May 29, 1996

Virginia Electric and Power Company
ATTN: Mr. J. P. O'Hanlon
Senior Vice President - Nuclear
Innsbrook Technical Center
5000 Dominion Boulevard
Glen Allen, VA 23060

SUBJECT: MEETING SUMMARY - SURRY SELF-ASSESSMENT - SURRY NUCLEAR POWER STATION - DOCKETS 50-280 AND 50-281

Dear Mr. O'Hanlon:

This refers to the meeting conducted at your request at the NRC Region II Office in Atlanta, Georgia on May 21, 1996. The meeting's purpose was to discuss Virginia Electric and Power Company's Self-Assessment for the Surry nuclear power plant.

It is our opinion that this meeting was beneficial in that it provided us with a better understanding of your accomplishments and improvement initiatives at the Surry station. Specific topics discussed included management perspectives, self-assessment and continuing challenges.

A list of attendees and a copy of your handout are enclosed. The ten minute video presentation used during the meeting will be retained in the Region II office for approximately six months before being discarded.

In accordance with Section 2.790 of the NRC's "Rules of Practice," Part 2, Title 10 Code of Federal Regulations, a copy of this letter and its enclosures will be placed in the NRC Public Document Room.

Should you have any questions concerning this letter, please contact us.

Sincerely,

Original signed by George A. Belisle

George A. Belisle, Chief
Reactor Projects Branch 5
Division of Reactor Projects

Docket Nos. 50-280, 50-281
License Nos. DPR-32, DPR-37

Enclosures: 1. List of Attendees
2. Surry Power Station Self-Assessment May 21, 1996

cc w/encls: See page 2

IE45

cc w/encls:
M. L. Bowling, Manager
Nuclear Licensing & Operations Support
Virginia Electric & Power Company
Innsbrook Technical Center
5000 Dominion Boulevard
Glen Allen, VA 23060

David A. Christian, Manager
Surry Power Station
Virginia Electric & Power Company
5570 Hog Island Road
Surry, VA 23883

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Surry County Board of Supervisors
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Dendron, VA 23839

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Virginia State Corporation Commission
Division of Energy Regulation
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State Health Commissioner
Office of the Commissioner
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Attorney General
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Richmond, VA 23219

Distribution w/encls: See page 3

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R. Gibbs, RII
M. Thomas, RII
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W. Stansberry, RII
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NAME	LGarner alt					
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COPY?	YES NO	YES NO	YES NO	YES NO	YES NO	YES NO

OFFICIAL RECORD COPY DOCUMENT NAME: G:\SURMTG.SUM

LIST OF ATTENDEES

NRC Attendees:

S. D. Ebneter, Regional Administrator, Region II (RII)
L. A. Reyes, Deputy Regional Administrator, RII
E. W. Merschoff, Director, Division of Reactor Projects (DRP), RII
J. R. Johnson, Deputy Director, DRP, RII
J. P. Jaudon, Deputy Director, Division of Reactor Safety (DRS), RII
G. A. Belisle, Chief, Reactor Projects Branch 5, DRP, RII
E. V. Imbro, Director, Project Directorate II-1, Office of Nuclear Reactor Regulation (NRR)
G. E. Edison, Project Manager, NRR
L. W. Garner, Project Engineer, DRP, RII
R. K. Caldwell, Project Engineer, DRP, RII
H. O. Christensen, Chief, Maintenance Branch, DRS, RII
W. W. Stansberry, Physical Security Specialist, DRS, RII
M. W. Branch, Senior Resident Inspector, Surry Nuclear Power Station
R. A. Musser, Resident Inspector, Browns Ferry Nuclear Plant

Licensee Attendees:

R. G. Saunders, Vice President, Nuclear Operations
M. L. Bowling Jr., Manager, Nuclear Licensing and Operations Support
D. A. Christian, Station Manager
B. L. Shriver, Assistant Station Manager, Nuclear Safety and Licensing

Surry Power Station Self Assessment

May 21, 1996

Agenda

Introduction	*R. F. Saunders*
Management Perspective	*D. A. Christian*
Self-Assessment	*B. L. Shriver*
Continuing Challenges	*D. A. Christian*
Closing Remarks	*R. F. Saunders*

Virginia Power Nuclear Vision

We are a Safe, Competitive,

World-Class Nuclear Operator

in the Base Load Energy

Generation Market

VIRGINIA POWER

Management Perspective

D. A. Christian
Station Manager

Vigilance

The key to sustained success is our ability to recognize our weaknesses before they become our problems.

To that end, we have established the expectations of personal integrity, accountability, and self-motivation. These attributes and the Nuclear Safety Policy are the basis for our work ethic.

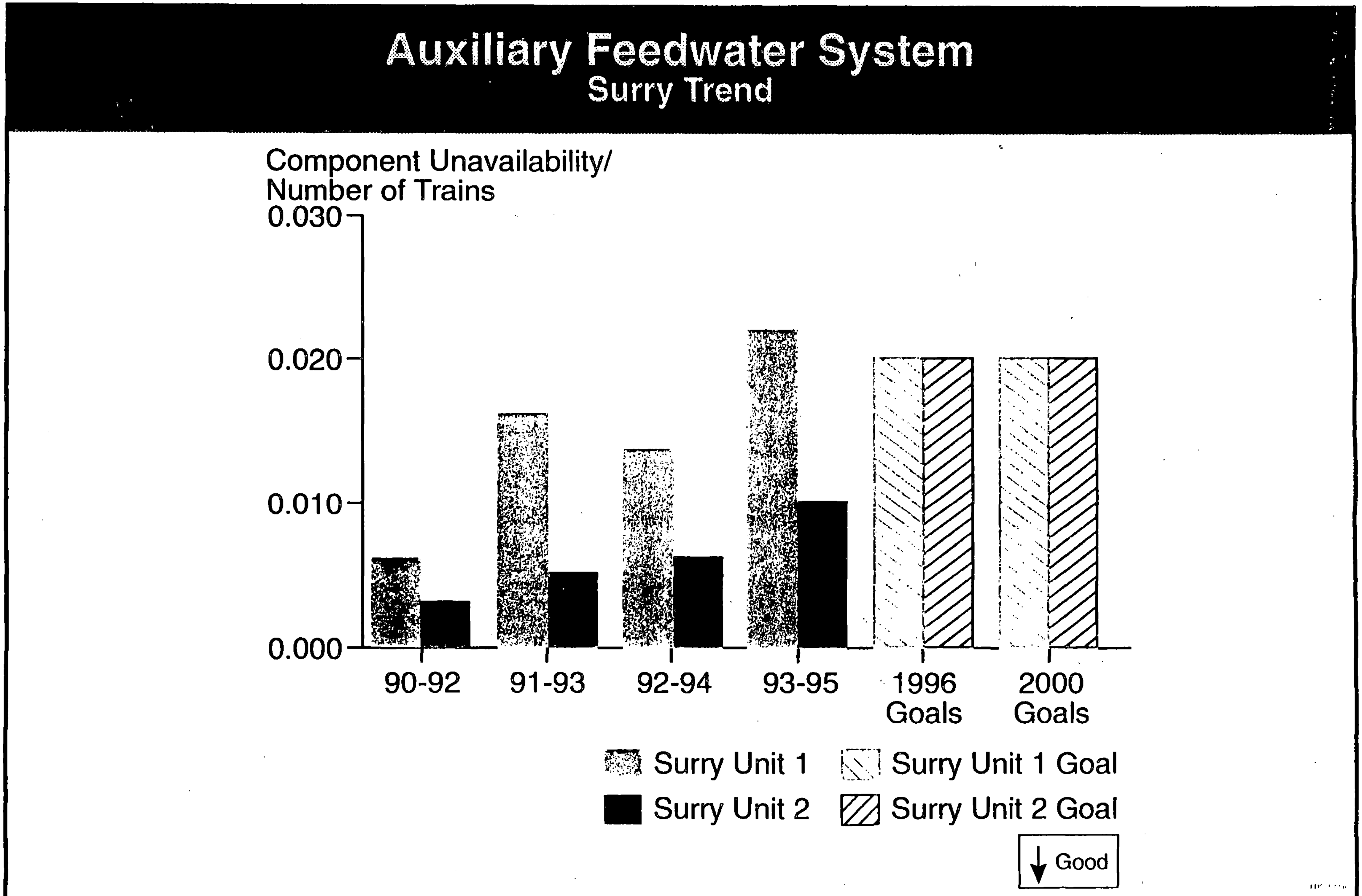
This work ethic and complementary programs enable our employees to make a difference -- to be effective at identifying and resolving problems.

Operating Performance

Plant improvement initiatives have resulted in excellent operating performance. The units have operated safely, reliably, and with minimal challenges to safety systems.

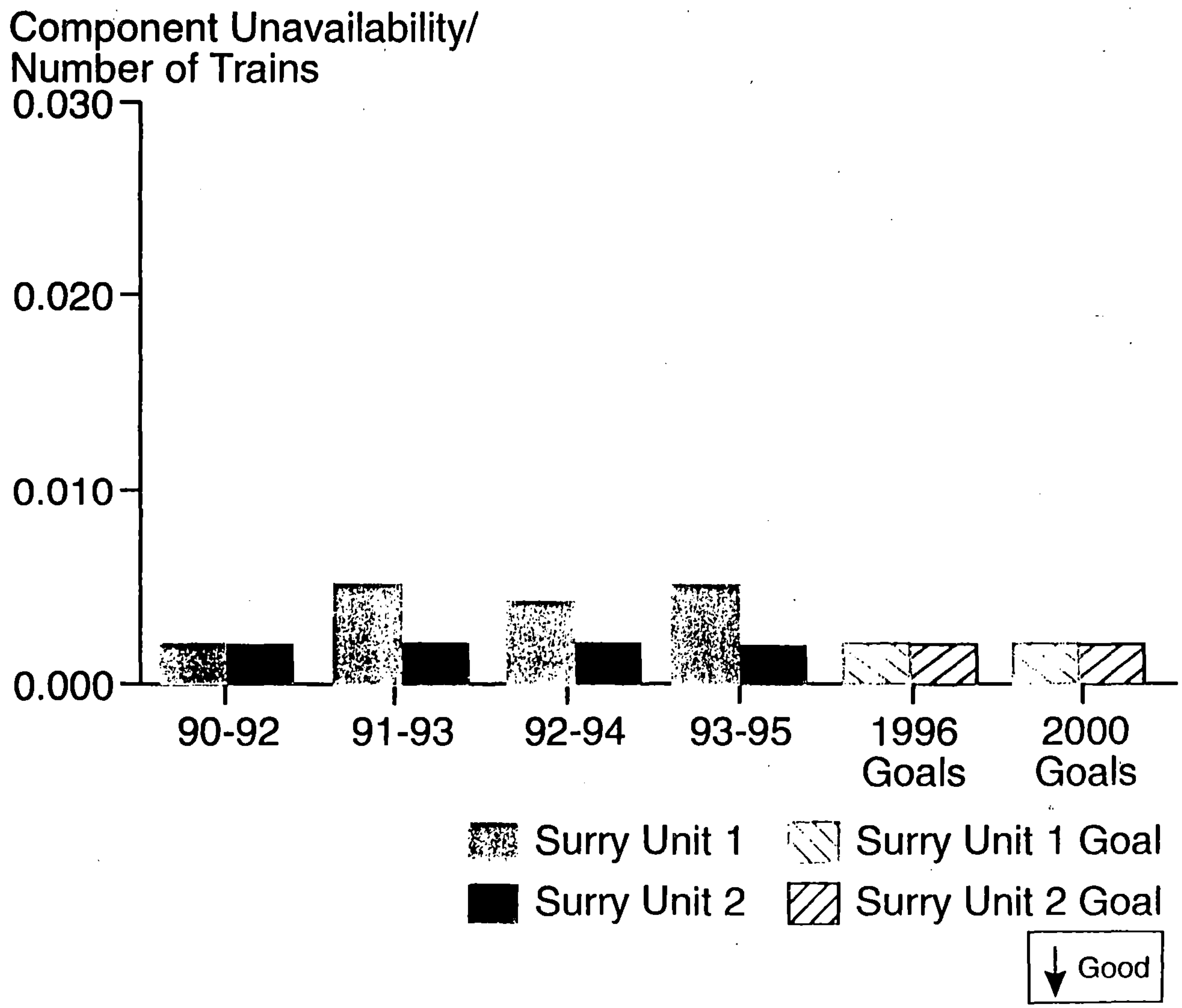
	Previous SALP (7/4/93 - 1/21/95)	Current * SALP (1/22/95 - 9/28/96)
Safety System Failures	1	0
Automatic Trips	5	2
Forced Outage Rate %	3.4	3.7
Capacity Factor % (MDC Net)	81.4	86.5

* Through 4/30/96

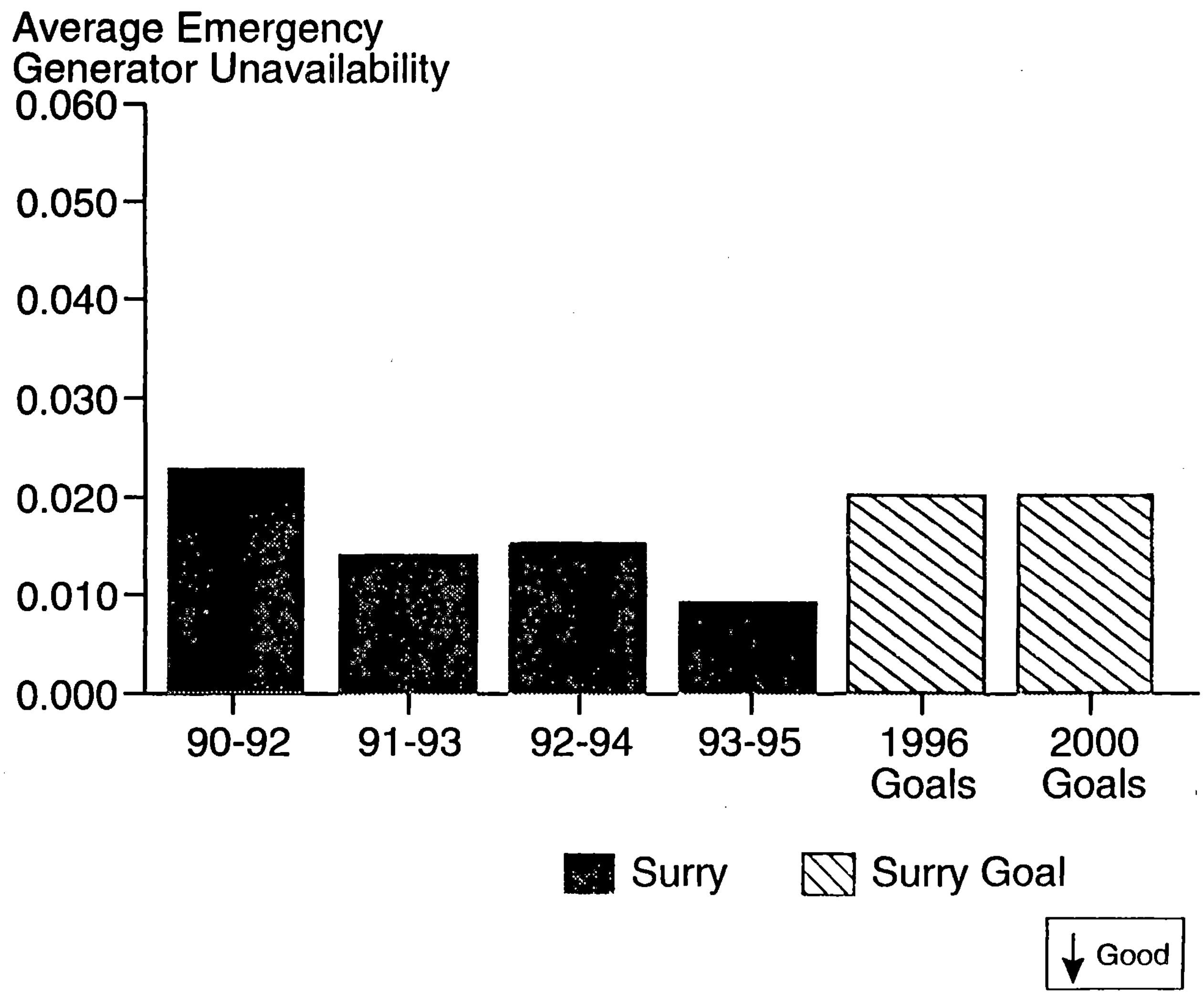


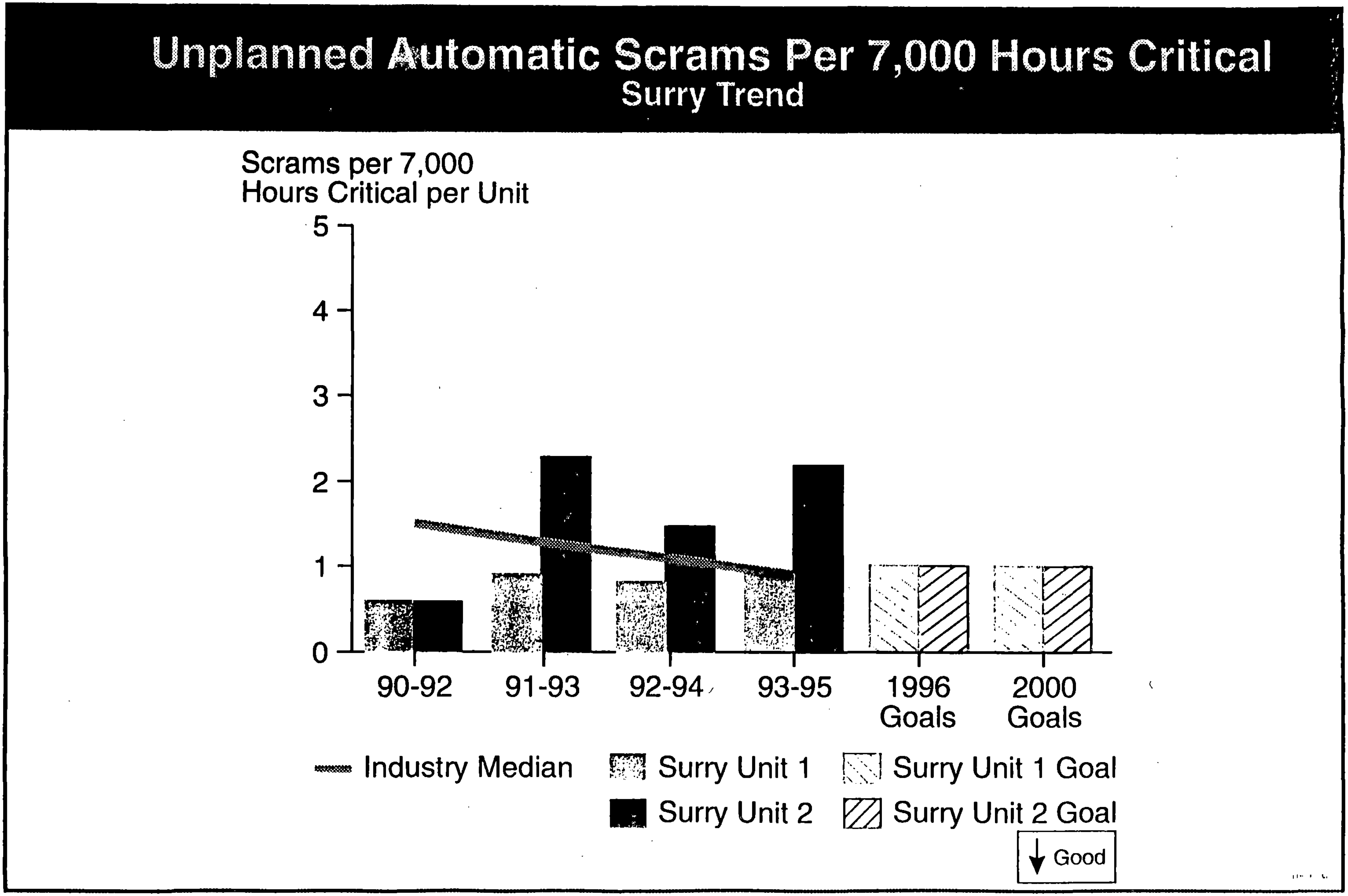
Auxiliary Feedwater System
Surry Trend
Component Unavailability/
Number of Trains
0.030
0.020
0.010
0.000
90-92
91-93
92-94
93-95
1996
Goals
2000
Goals
Surry Unit 1
Surry Unit 2
Surry Unit 1 Goal
Surry Unit 2 Goal
Good

High Pressure Safety Injection System
Surry Trend

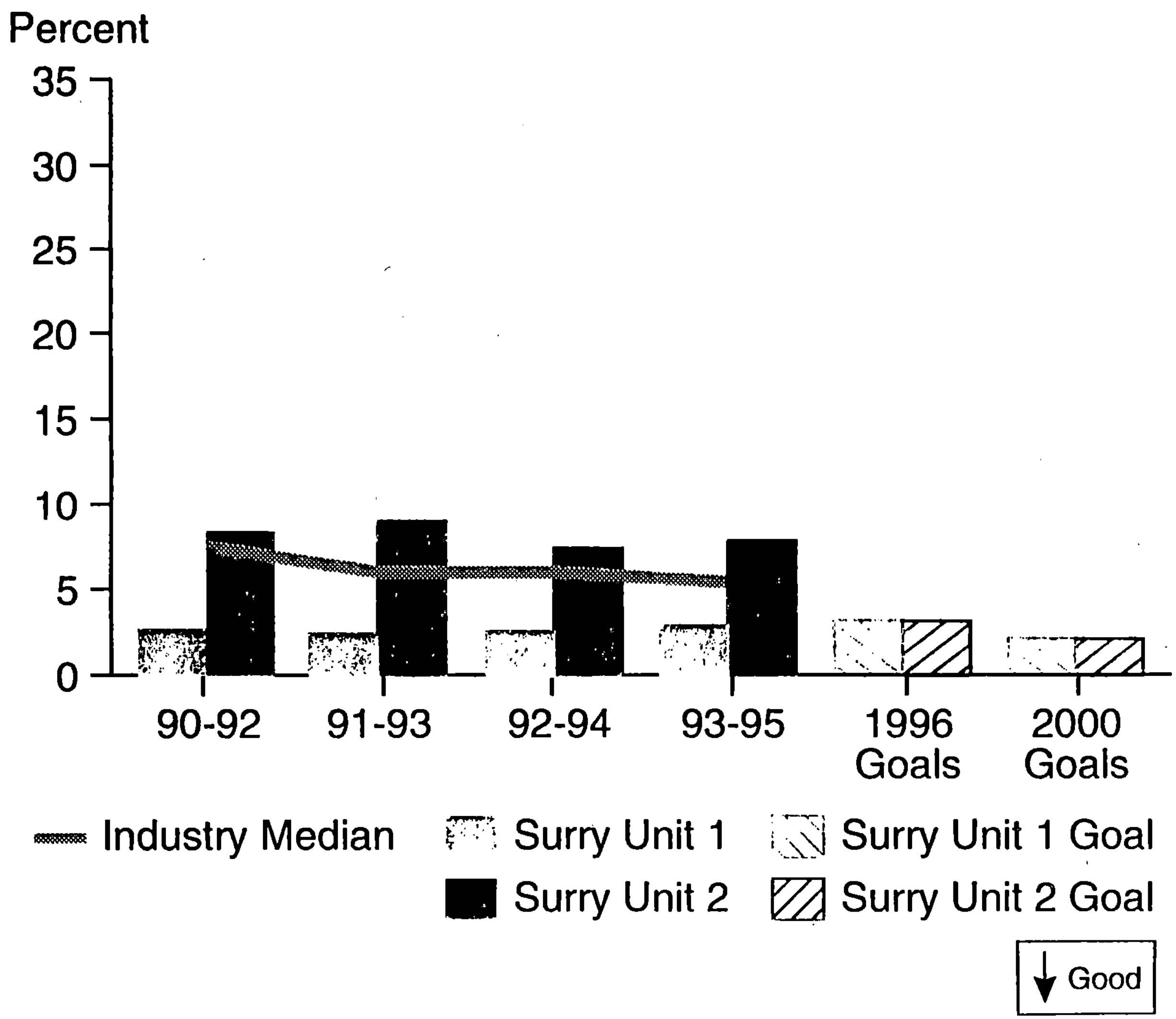


Emergency AC Power System
Surry Trend

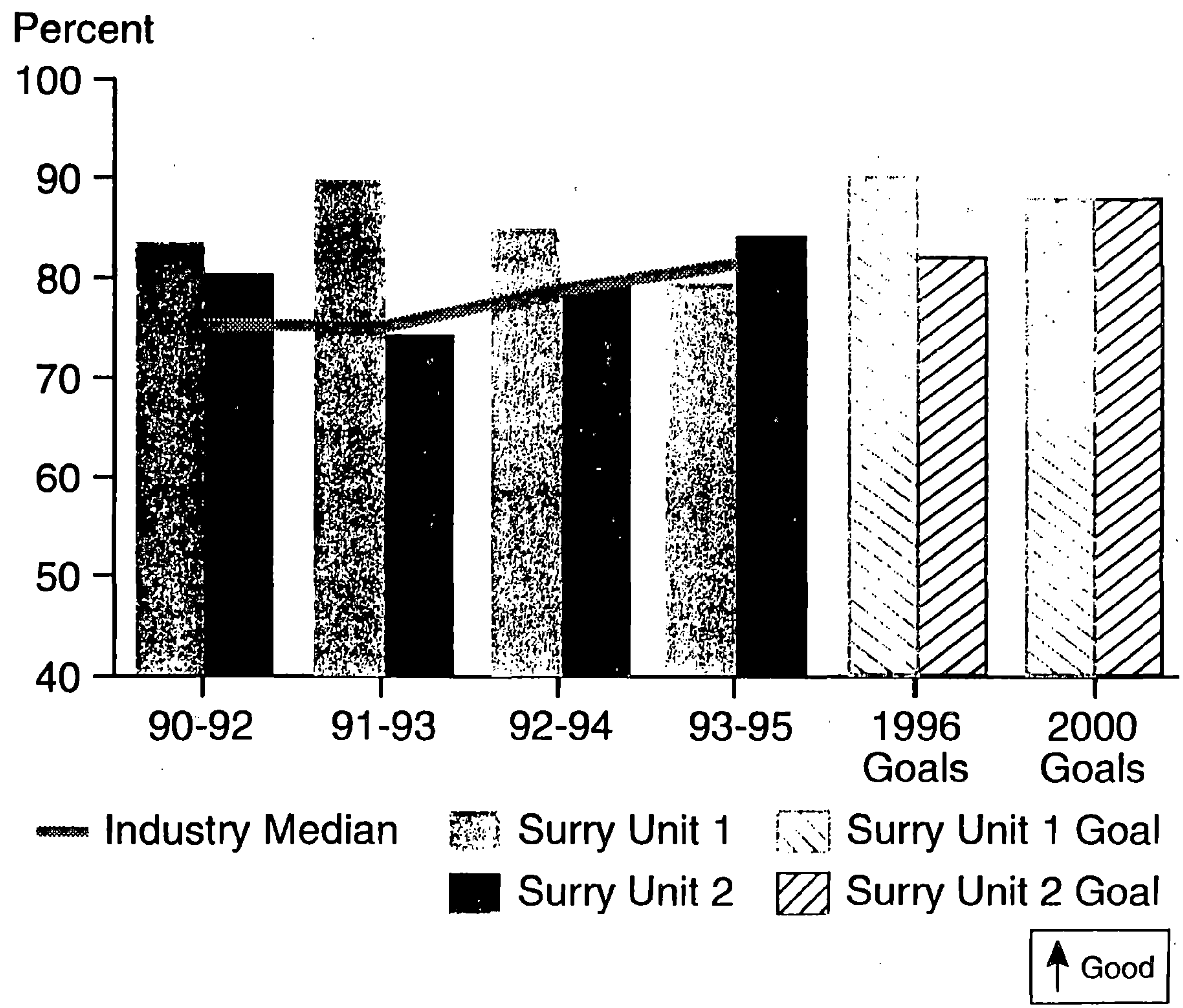



Unplanned Automatic Scrams Per 7,000 Hours Critical
Surry Trend
Scrams per 7,000
Hours Critical per Unit
5
4
3
2
1
0
90-92
91-93
92-94
93-95
1996 Goals
2000 Goals
Industry Median
Surry Unit 1
Surry Unit 2
Surry Unit 1 Goal
Surry Unit 2 Goal
Good

Unplanned Capability Loss Factor
Surry Trend

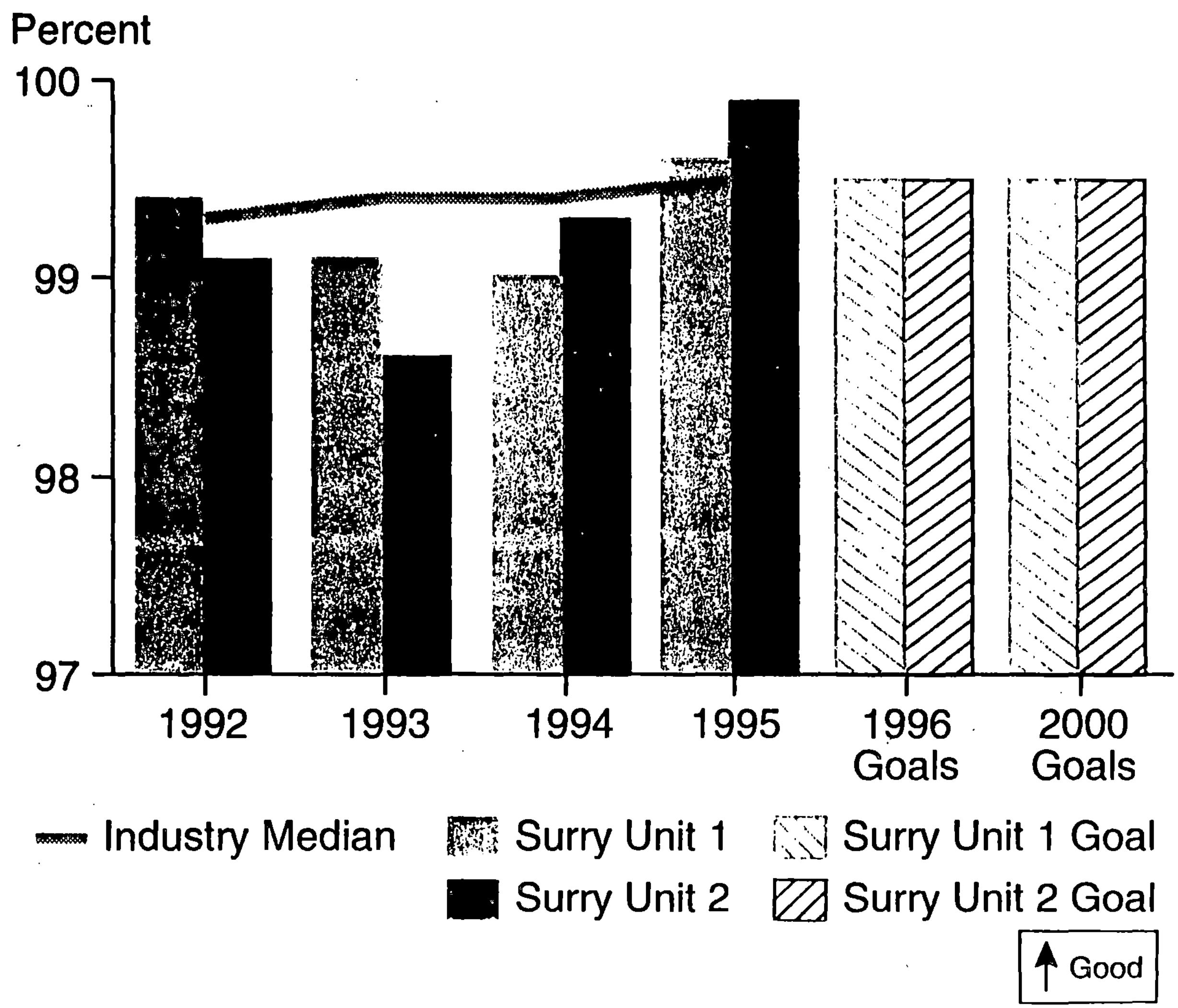


Unit Capability Factor
Surry Trend



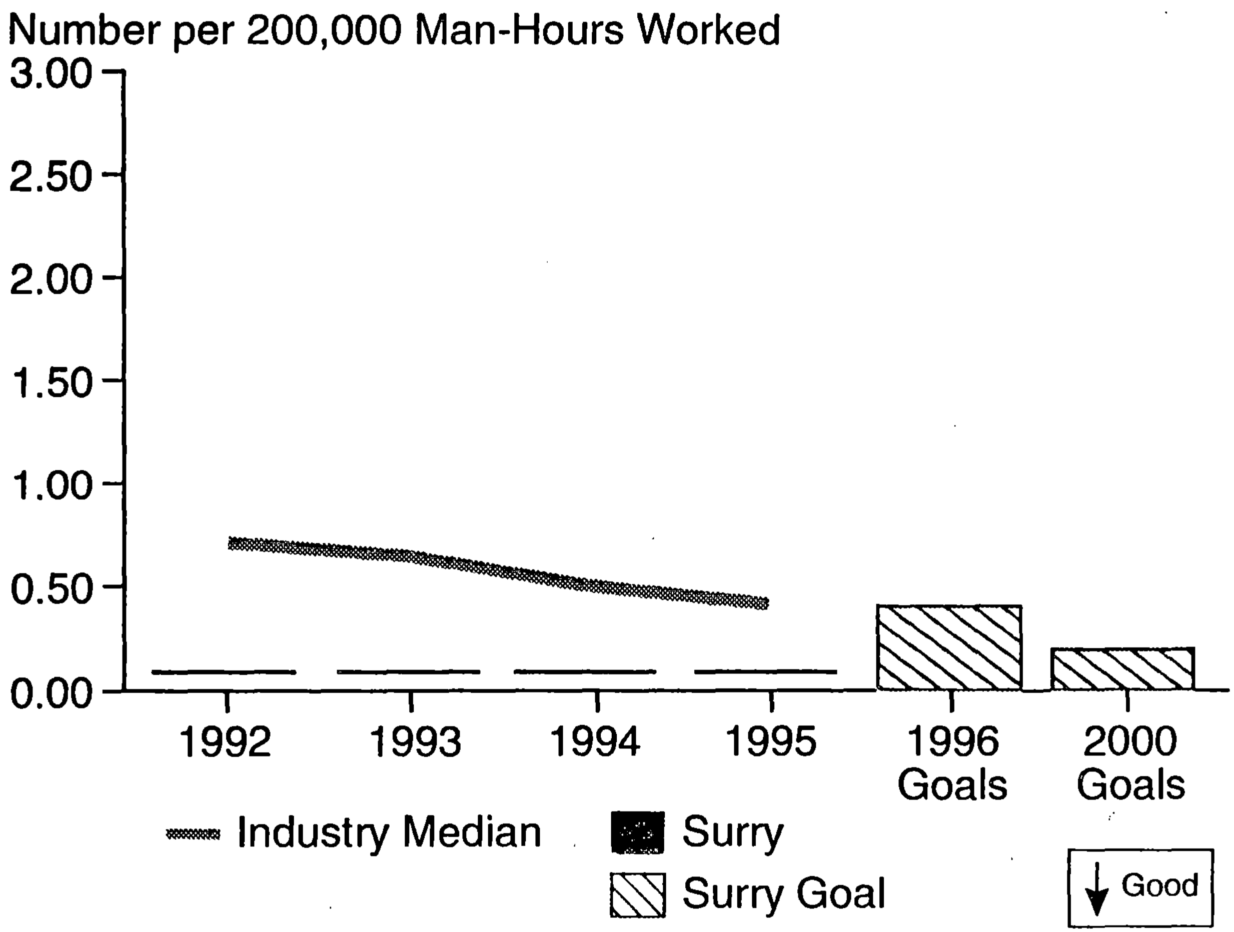
Thermal Performance

Surry Trend

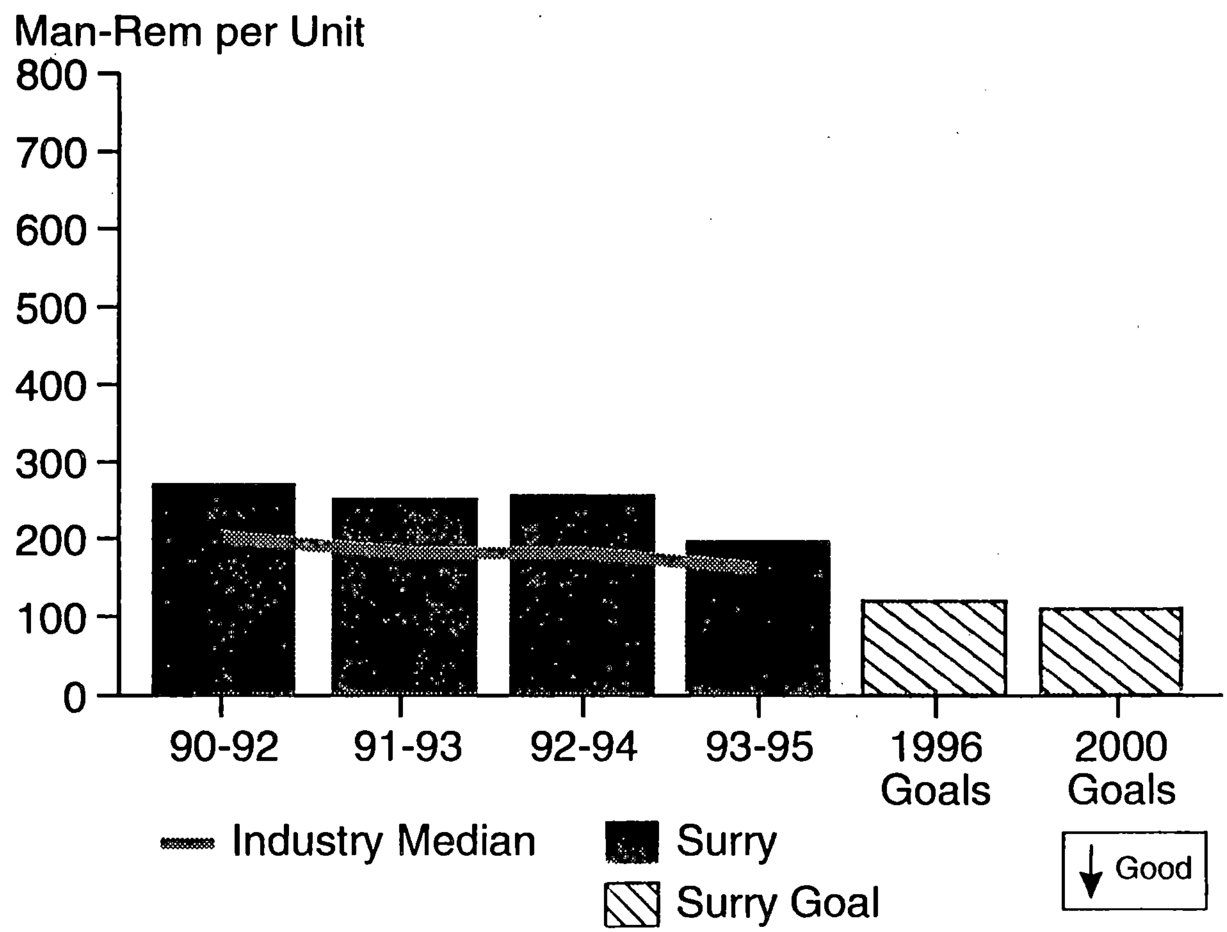


Industrial Safety Accident Rate

Surry Trend

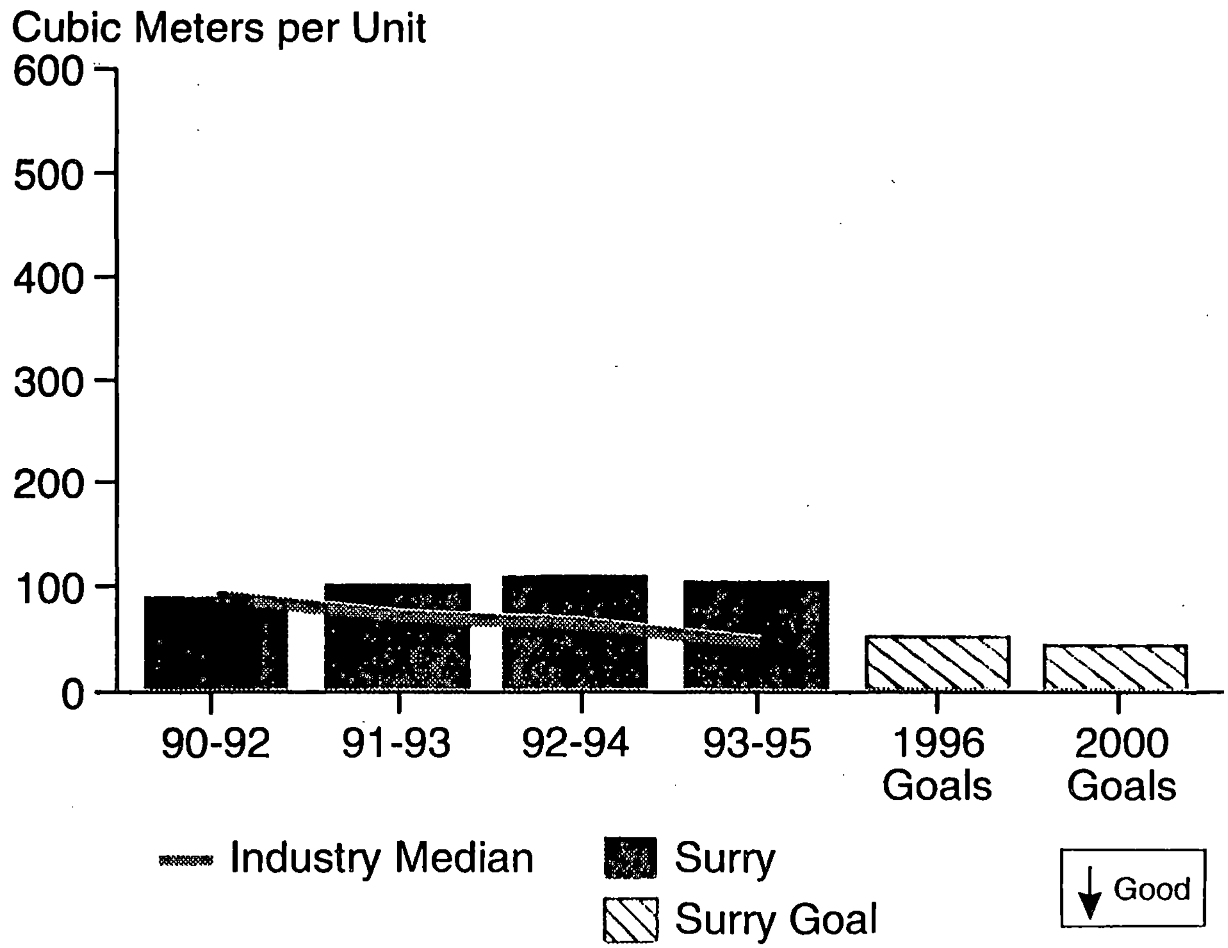


Collective Radiation Exposure (PWR)
Surry Trend

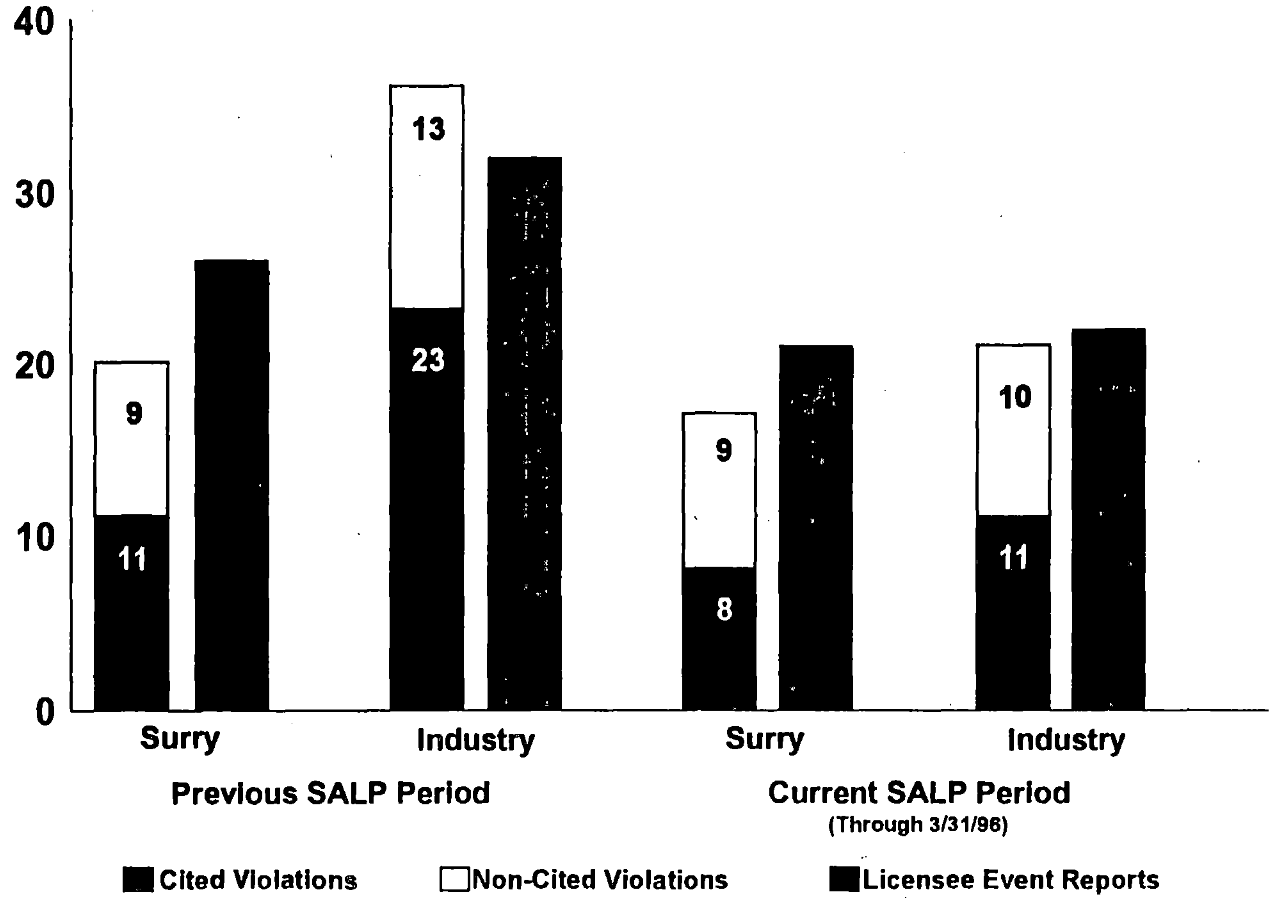


Volume of Low-Level Solid Radioactive Waste (PWR)

Surry Trend



Regulatory Performance



SALP Period Accomplishments

In addition to good operating performance, several significant milestones and successes have been achieved during this SALP period.

- ***Units 1 and 2 surpassed 100 million megawatt-hours gross generation. Unit 1 has operated continuously for 214 days.***

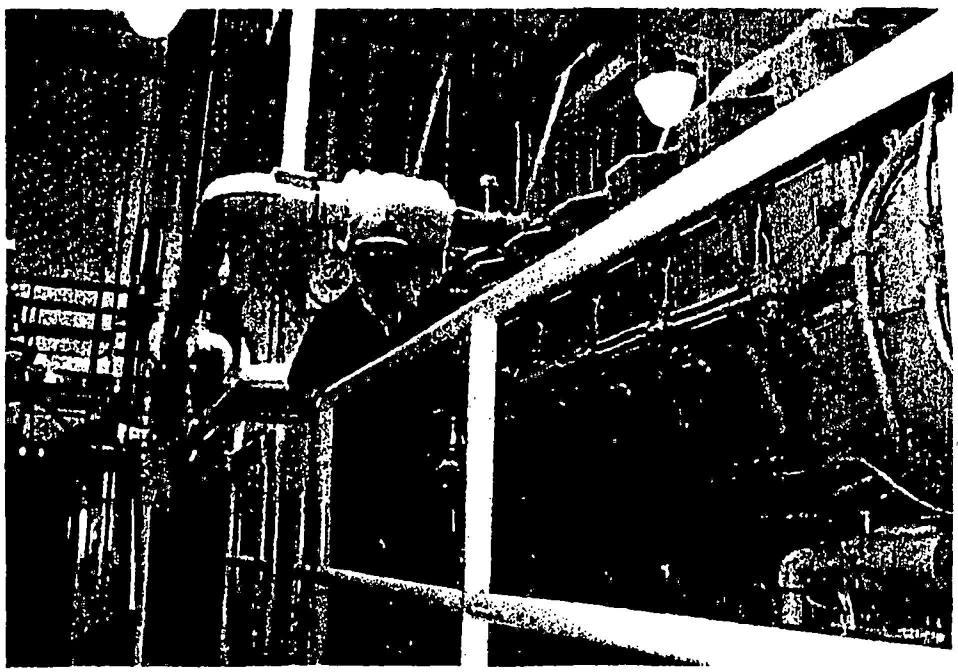
- ***Licensed power was increased from 2441 to 2546 MWt. Maximum dependable capacity was increased from 820 to 840 MWe.***

- ***Six million man-hours without a lost-time accident.***

- ***Generic Letter 89-10, Motor-Operated Valves close-out inspection was completed.***

SALP Period Accomplishments

- *Independent spent fuel storage installation capacity was expanded.*
- *Formal station performance improvement plans were initiated.*
- *An INPO 1 rating was maintained.*



Unit 1 station blackout emergency diesel generator was installed and tested. Unit 2 is being completed during the current refueling outage.

Focus Areas

We have evaluated the 1995 Unit 2 outage events and the underlying causes. As a result, we have focused our improvement efforts in the following areas:

- ***Human Performance***
- ***Self-Assessment***
- ***Accountability***

Human Performance

Improving human performance is one of the most difficult challenges we face. We are concentrating our efforts on four major elements that affect human performance:

- ***Communications***
- ***Teamwork***
- ***Processes***
- ***Supervisory Development***

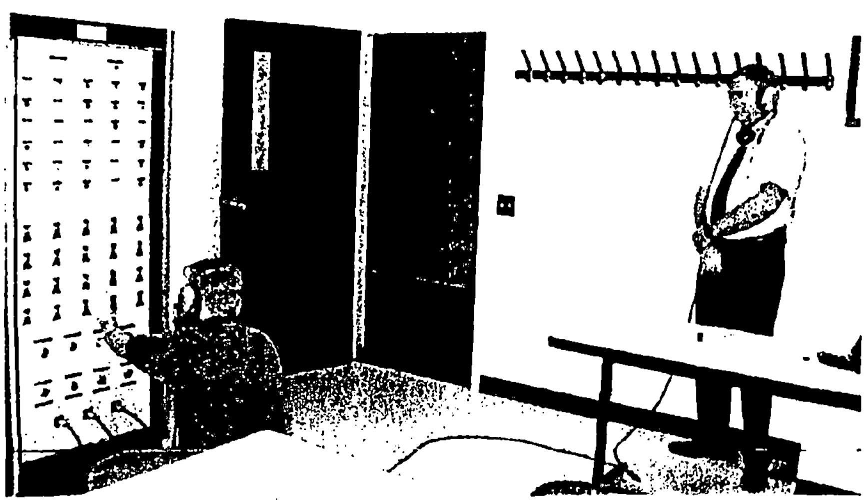
Communications Improvements

Several initiatives are underway to improve station communications and the dissemination of information. These efforts are being monitored to determine their effectiveness.

- ***Established specific station objectives***
- ***Conducted human performance focused "Standdown Days"***
- ***Increased management communications***
- ***Improved coordination through the Plan of the Day (POD)***
- ***Implemented an employee recognition process***

Communications Improvements

- ***New Operations Standards have been issued to more clearly define station expectations in the following areas:***
  - ***Communications***
  - ***Reactivity Management***
  - ***Control Room***
  - ***Outage Super Crew***



A computer based Self-Check Simulator was developed to reinforce self-checking practices, proper verbal communications, and procedural adherence.

Teamwork Improvements

- ***The use of interdepartmental teams to address significant station issues is being increased.***
  - ***Category I Root Cause Evaluations (e.g., Lifting/Rigging)***
  - ***Station Self-Assessments (e.g., Work Control Process)***

- ***The implementation of a performance based self-assessment program is being facilitated through the cooperative effort of Station Nuclear Safety and each station department.***

- ***The Plan of the Day (POD) is being used as a tool to improve interdepartmental coordination.***

- ***The outage management organization has been strengthened, including management personnel on backshift.***

Process Improvements

- ***A Radiological Review Board was established following on the success of the Operations and Maintenance Review Boards.***

- ***Root Cause Program has been improved.***

- ***Benchmarking trips have been made to other good performing nuclear power stations and other non-nuclear organizations. These observations are being used to improve station performance.***

Process Improvements

- ***Examples of benchmarking trips:***
  - ***Human performance (Callaway)***
  - ***Tagging, emergency lighting design (Diablo Canyon)***
  - ***Industrial safety (Brunswick)***
  - ***Contractor management (Tsuruga, Japan)***
  - ***Complacency, standards, accountability (St. Lucie)***
  - ***RCA tool control (Calvert Cliffs)***
  - ***Emergency preparedness (Fitzpatrick)***
  - ***Offsite reviews (V. C. Summer)***
  - ***Loss prevention (Farley)***
  - ***Resource sharing (Region II Health Physics Conference)***
  - ***Non-nuclear benchmarking (Ford, Canon, Union Camp)***

Supervisory Development

- *Key personnel are being developed through cross-training assignments.*

- *Selected engineering supervisors are attending the INPO Engineering Supervisor Professional Development Seminar.*

- *Monitoring of work activities and training has been increased (e.g., Daily Plant Observations).*

- *Hands-on Change Management training has been conducted for supervisory personnel.*

VIRGINIA POWER

The People Behind the Performance

VIRGINIA POWER

Self-Assessment

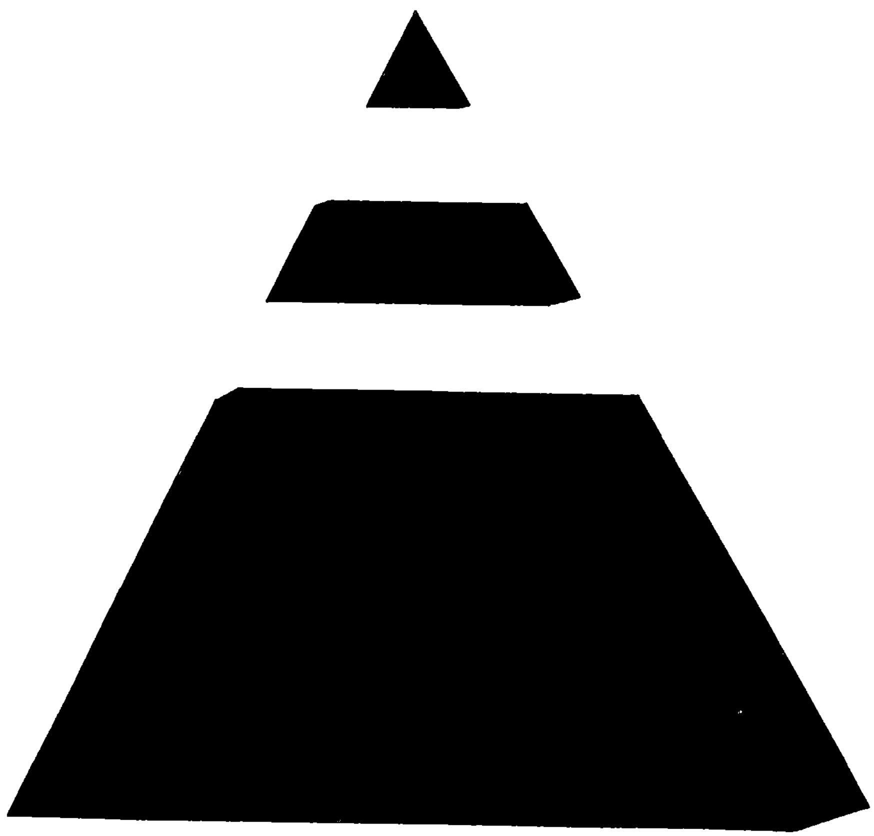
B. L. Shriver
Assistant Station Manager
Nuclear Safety and Licensing

Program Improvements

A Self-Assessment Program improvement plan has been developed:

- *Formalize self-assessment elements*
- *Enhance Station Performance Annunciator Panels*
- *Focus on corrective actions and accountability*
- *Conduct periodic effectiveness reviews*

Self-Assessment Hierarchy



- ***External Oversight***
- ***Internal Oversight***
- ***Management Self-Assessment***
  - → ***Integrated Station***
  - → ***Department***
  - → ***Continuous Monitoring***

Self-Assessment Matrix

- ***Continuous Monitoring***

Preventive

- *Performance Indicators*
    - *Morning Report*
    - *Critical Parameters*
- *Plant Observations*
    - *Operator Logs*
    - *Daily Plant Observations*
    - *Housekeeping Inspections*

Corrective

- *Deviation Report,*
  Work Request,
  Design Change Package
- *Category III Root Cause*
  Evaluations

Self-Assessment Matrix

- ***Department Self-Assessments***

Preventive

- *Department INPO Performance Objectives and Criteria*
- *Program Self-Assessments*
- *System Engineering Quarterly Report*
- *Training Observations*
- *Departmental Performance Indicators (e.g., Open Work Orders)*

Corrective

- *Category II Root Cause Evaluations*
- *Performance Based Department Assessments*

Self-Assessment Matrix

- *Integrated Station*

Preventive

- *Station Performance Annunciator Panel*
- *Nuclear Business Plan Goal Performance Report*
- *Deviation Trend Report*
- *Refueling Outage Assessment and Safety Evaluation*
- *Programmatic Team Assessments (e.g., 10 CFR 50.59, Industry Issues)*

Corrective

- *Integrated Trend Report*
- *Category I Root Cause Evaluations*
- *Significant and Precursor Events*
- *Issue-Driven Team Assessments*

Surry Performance Annunciator Panel

First Quarter 1996

Personnel Performance

Operation	Maintenance	Radiological Protection	NSS	Engineering	Station Support

Equipment Performance

Reactor Trips	ESF Actuations	RCS and Secondary System Integrity	Fuel Reliability	Key Safety System Performance/ Availability	System Materiel Condition
Net Generation/ Thermal Performance	Plant Capacity Factor	Emergency Preparedness Equipment Availability	Chemistry Performance		

Program Performance

Operating Program	Maintenance Program	Radiation Protection Program	Site Services Program	Site Engineering Program	Security Program
Regulatory Performance Program	Emergency Preparedness Program	Industrial Safety Program	Safety Assessment and Quality Program	Materials Program	Training Program
Planning & Scheduling Program	Procedure Program	Business Plan Performance	Adherence to Nuclear Safety Policy	Significant and Precursor Events	

LEGEND

2Q/95	3Q/95	4Q/95
First Quarter 1996		

Red – Significant Weakness

White – Satisfactory

Yellow – Improvement Needed

Green – Significant Strength

Human Performance

Strengths

- *Employee experience and knowledge*
- *Employee commitment to good performance*
- *High quality programs and procedures*
- *Industrial safety*
- *Re-engineering focus on safety*

Weaknesses

- *Human performance errors are the cause of most precursor events*
- *Long-term corrective action effectiveness*
- *Compliance with Radiation Work Permits*
- *Re-engineering*
  - *Initial implementation plan*
  - *Employee anxiety*

Equipment Performance

Strengths

- ***Safety-related equipment reliability***
- ***Decreased forced power reductions***
- ***Improved performance of support equipment***
- ***Containment isolation valve testing***

Weaknesses

- ***Equipment problems are responsible for most reactor trips and unplanned outages***
- ***Maintenance Rule SSC failures***
  - ***Rod control system***
  - ***Turbine-driven AFW pumps***
  - ***Emergency service water pumps***
  - ***CR/ESGR air handling units***
- ***Unit 1 fuel reliability***
- ***Unit 2 reactor coolant system leakage***
- ***Collective radiation exposure***

Plant Aging

Strengths

- *Foxboro digital instrumentation upgrade*
- *Radiation monitor digital upgrade*
- *Emergency diesel generator fuel oil line replacement*
- *Flow assisted corrosion program*
- *Feedwater pump motor replacement*
- *Turbine electro-hydraulic control system upgrades*
- *Station service transformer cable replacement*
- *Circulating water pumps overhaul*

Weaknesses

- *Rod control system*
- *Control room annunciators*
- *Plant computer systems*
- *Nuclear instrumentation cables*

Corrective Action Improvements

A corrective action improvement plan has been developed to:

- *Clearly identify as high priority those items important to nuclear safety, regulatory compliance, and plant reliability*
- *Establish clear ownership for the development, monitoring, and effectiveness of corrective actions*
- *Hold organizations and individuals accountable for completing and verifying the effectiveness of corrective actions*

Equipment Problems

We have succeeded in resolving several complex equipment problems during this SALP period. These experiences have demonstrated the effectiveness of using interdepartmental teams and a methodical approach.

As we continue to improve our Root Cause Program and apply the lessons learned, we anticipate more success in resolving equipment problems.

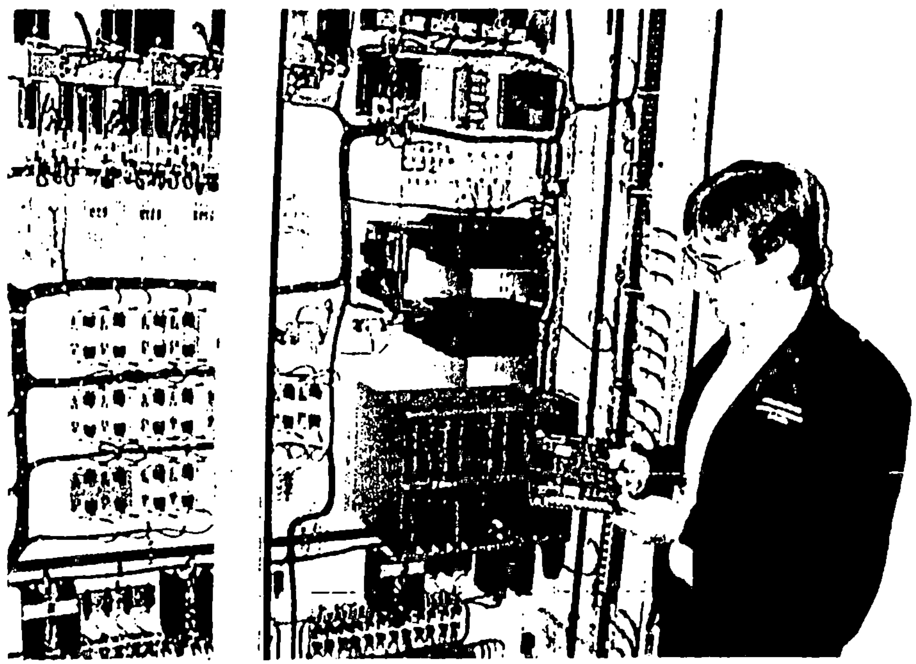
Rod Control System

Actions

- *Improved the material condition of the rod control and support systems.*
- *Enhanced preventive maintenance and testing.*
- *Improved the human interface.*
- *Improved rod control power cabinet environmental conditions.*
- *Addressed rod control circuit card service life concerns.*

Results

- *No reactor trips since May 1995.*
- *No rod urgent failures since September 1995.*



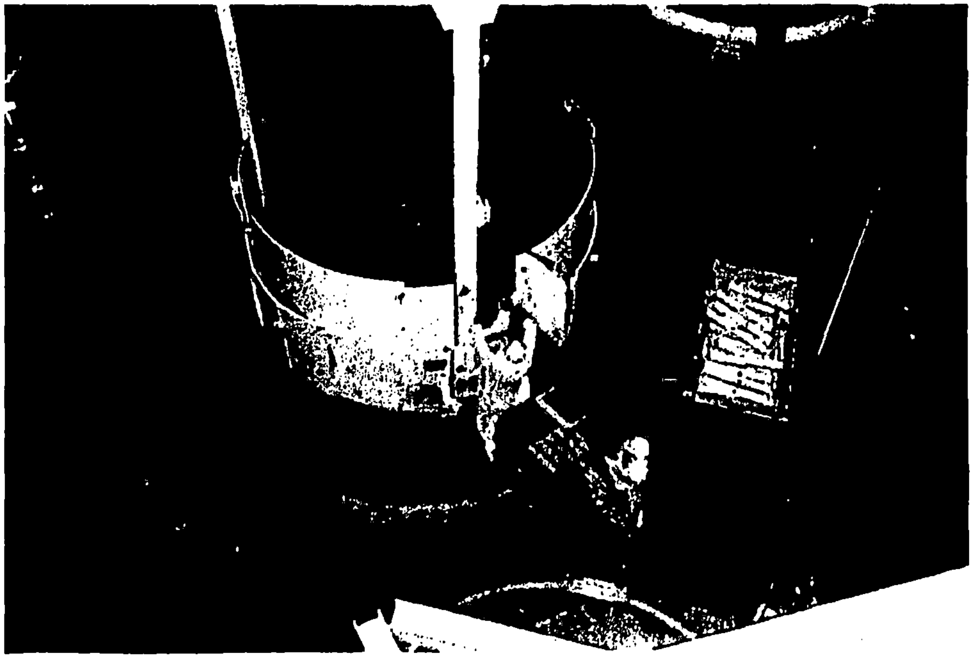
Unit 2 IRPI M-10

Actions

- *Performed significant troubleshooting, including time domain reflectometry testing, coil stack replacement, inspection/cleaning of IRPI connections, and control rod drive shaft replacement.*
- *Placed the search coil back in service.*
- *Replacing the pressure housing during the 1996 refueling outage.*

Results

- *Alternate indication of M-10 position provided.*
- *Eliminated the need to consider M-10 as not being fully inserted when performing Emergency Operating Procedure E-0.*



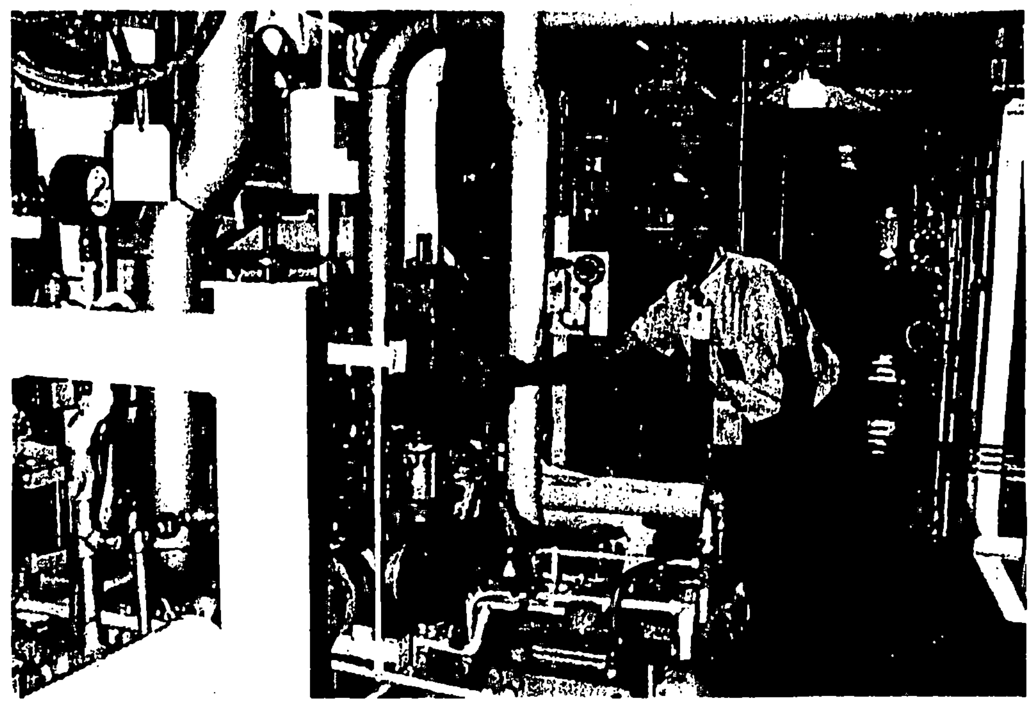
Turbine-Driven AFW Pump

Actions

- ***Revised maintenance and testing procedures to address deficiencies identified by root cause team.***
- ***Improved the perpendicular alignment of governor No. 227 servo with servo linkage and installed a solid control rod.***
- ***Improved the monitoring of pump performance.***

Results

- ***No overspeed trips since January 1995.***
- ***Documented effectiveness of torque checks and chrome stems.***



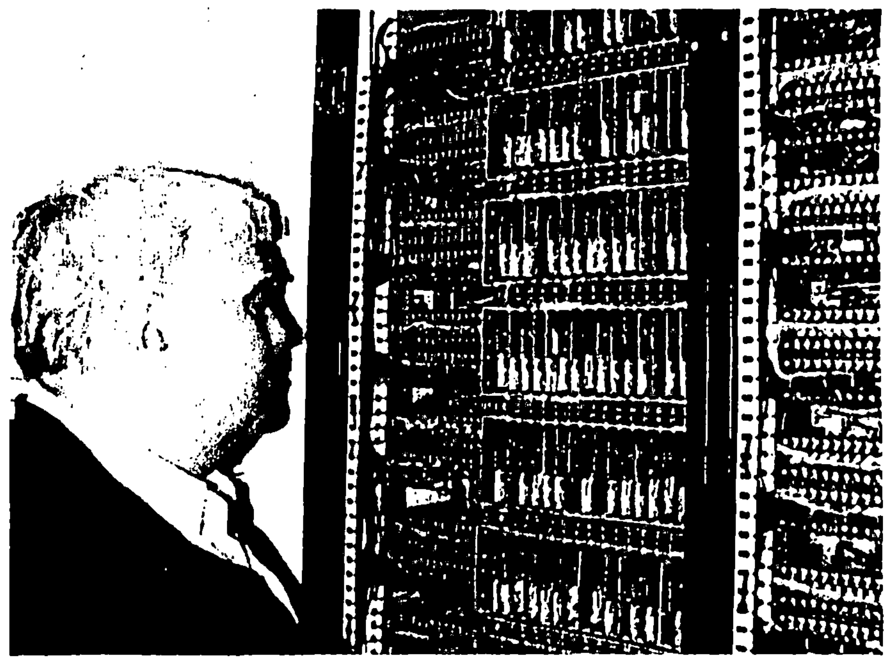
Control Room Annunciators

Actions

- ***Refurbished Unit 1 power supplies.***
- ***Refurbishing Unit 2 power supplies during the 1996 refueling outage.***

Results

- ***No deviations identified since Unit 1 power supplies were refurbished.***



Service Water Improvements

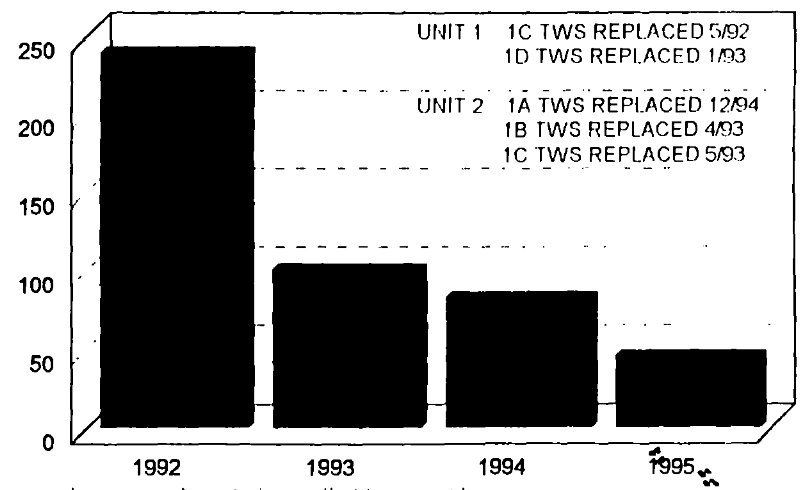
Actions

- ***Replaced 6 of 8 traveling water screens (TWS)***
- ***Applied protective coating to piping and antifoulant coating to canal level probes***
- ***Chemical treatment of SW to component cooling heat exchangers (CCHX)***
- ***Replaced intermediate seal cooler copper tubes with copper-nickel tubes***
- ***Improved materials and set-up/operation of MER No. 5 PCVs and charging SW TCVs***
- ***Eliminated circulating water pump external oil coolers***
- ***Replaced carbon steel fish spray (FS) system piping with fiberglass piping***
- ***Initiated bi-weekly flushing of charging lube oil coolers***
- ***Replaced recirculating spray heat exchanger carbon steel drain valves with bronze valves***

Results

- ***TWS replacement reduced the transport of biofouling into intake canal and station***
- ***CCHX tube cleaning frequency reduced since start of chemical treatment***
- ***Canal level probe response and reliability has been improved***
- ***No intermediate seal cooler failures since tube replacement***
- ***Increased FS system pressure and flow***
- ***Stable charging pump oil temperature and MER No. 5 operation***
- ***Significant reduction in internal pipe corrosion***

WATERBOX DATA



Resolution Plans

Although there have been successes, equipment performance issues remain. These issues will be prioritized and addressed with the utmost consideration for maintaining nuclear safety.

Resolution Plans are being developed to address equipment performance issues. These plans, which are approved and monitored by management, define a logical course of action for correcting and preventing the recurrence of equipment problems such as:

- ***Source Range Nuclear Instrumentation spiking***
- ***Emergency Service Water Pump performance***
- ***Waste Gas Decay Tank Oxygen Monitors***
- ***Emergency Response Facility Computer system reliability***

Addressing Complacency

A major challenge for Surry Power Station, as well as any other company seeking sustained high performance, is to avoid complacency.

We recognize that the onset of complacency can be insidious and that we must be alert to its symptoms and aggressive in our preventive actions. Therefore, we established a Complacency Avoidance Team:

- ***M. L. Bowling*** — ***Manager, Nuclear Licensing and Operations Support***
- ***A. H. Friedman*** — ***Manager, Nuclear Training***
- ***D. A. Heacock*** — ***NAPS Assistant Station Manager, Nuclear Safety and Licensing***
- ***B. L. Shriver*** — ***SPS Assistant Station Manager, Nuclear Safety and Licensing***

Recommendations

- *Incorporate complacency indicators into station windows and implement with 2nd quarter 1996 assessment*
- *Issue a Training Information Bulletin to address complacency*
- *"Organizational issues" to the Integrated Trending Program*
- *Increase the focus of the Human Resources program to address the reasons for complacency*
- *Evaluate the need and benefit of developing morale indicators*

Surry Performance Annunciator Panel

Panel

Complacency

DEADLINES BEING MET	INAPPROPRIATE ACTIONS DURING ROUTINE EVOLUTIONS	CORRECTIVE ACTION EFFECTIVENESS	NON-CITED / TOTAL VIOLATION TREND
NUCLEAR BUSINESS PLAN GOAL PERFORMANCE	SELF-ASSESSMENT EFFECTIVENESS	STATION BENCHMARKING	MANAGEMENT JOB ROTATION
SIGNIFICANT AND PRECURSOR EVENT TREND	HOUSEKEEPING STANDARDS		

VIRGINIA POWER

Continuing Challenges

D. A. Christian
Station Manager

Continuing Challenges

We remain committed to the challenge of continuous improvement in all aspects of station operation.

- ***We will continue to strive for excellent human performance since our people are the key to the company's viability.***

- ***Aging plant equipment and systems will be closely monitored and adequately maintained to ensure nuclear safety, reliable plant operation, and to preserve the option for license renewal.***

- ***Our Technical Specifications will be improved to enhance plant operations and foster good regulatory performance. The UFSAR will be carefully reviewed and revised, as necessary.***

VIRGINIA POWER

Closing Remarks

R. F. Saunders
Vice-President Nuclear Operations

May 14, 1999

Virginia Electric and Power Company
ATTN: Mr. J. P. O'Hanlon
Senior Vice President - Nuclear
Innsbrook Technical Center
5000 Dominion Boulevard
Glen Allen, VA 23060

SUBJECT: NRC EXAMINATION REPORT NOS. 50-280/99-301 AND 50-281/99-301

Dear Mr. O'Hanlon:

On March 29 - April 1, 1999 and April 12-15, 1999, the NRC administered examinations to employees of your company who had applied for licenses to operate the Surry Power Station. At the conclusion of the examination, the examiners discussed the examination questions and preliminary findings with those members of your staff identified in the enclosed report.

The Simulation Facility Report is included in this report as Enclosure 2. Enclosure 3 is the Facility Post-Examination Comments. Enclosure 4 is the NRC Resolution of post-examination comments. A copy of the written examination questions and answer key, as noted in Enclosure 5, was provided to the members of your training staff at the conclusion of the examination.

Of the ten senior reactor (SRO) and reactor operator (RO) applicants who received written examinations and operating tests nine candidates passed (and one failed) the examination, representing a 90 percent pass rate. Four of the ten candidates were identified as having some performance weaknesses in the Administrative Topic or systems areas. The individual examination reports should be reviewed to determine if adjustments to the training program, as well as individual remediation, are necessary.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter and its enclosures will be placed in the NRC Public Document Room.

Should you have any questions concerning this letter, please contact me at (404) 562-4638.

Sincerely,

(Original signed by H. O. Christensen)

Harold O. Christensen, Chief
Operator Licensing and
Human Performance Branch
Division of Reactor Safety

Docket Nos. 50-280, 50-281
License Nos. DPR-32, DPR-37

Enclosures:
1. Report Details
2. Simulation Facility Report
3. Facility Post-Examination Comments
4. NRC Resolution of Facility Post-Examination Comments
5. Written Examination and Answer Key (SRO)
   (Document Control Desk Only)

cc w/encl:
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cc w/encls cont'd: (See page 3)

cc w/encls cont'd:
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Distribution w/encls: (See page 4)

<u>Distribution w/encls:</u>
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OFFICE	RII:DRS	RII:DRS	RII:DRS	RII:DRS	RII:DRP		
SIGNATURE	RBB	[illegible] for	[illegible]	[illegible]			
NAME	RBaldwin	LMellen	DCPayne	HChristensen	BHagg		
DATE	5/12/99	5/12/99	5/13/99	5/14/99	5/14/99	5/ /99	5/ /99
COPY?	YES NO	YES NO	YES NO	YES NO	YES NO	YES NO	YES NO

OFFICIAL RECORD COPY DOCUMENT NAME: G:\LA\SUR\99301RPT.WPD

U. S. NUCLEAR REGULATORY COMMISSION

REGION II

Docket Nos.:	50-280, 50-281
License Nos.:	DPR-32, DPR-37
Report No.:	50-280/99-301, 50-281/99-301
Licensee:	Virginia Electric and Power Company (VEPCO)
Facility:	Surry Power Station, Units 1 & 2
Location:	5850 Hog Island Road Surry, VA 23883
Dates:	March 29 - April 15, 1999
Examiners:	Richard S. Baldwin, Chief License Examiner Larry S. Mellen, License Examiner D. Charles Payne, License Examiner
Approved by:	Harold O. Christensen, Chief Operator Licensing and Human Performance Branch Division of Reactor Safety

Enclosure 1

EXECUTIVE SUMMARY

Surry Nuclear Power Station Units 1 & 2
NRC Examination Report No. 50-280/99-301 and 50-281/99-301

During the periods of March 29 - April 1, 1999 and April 12-15, 1999, NRC examiners conducted an announced operator licensing initial examination in accordance with the guidance of Examiner Standards, NUREG-1021, Interim Revision 8. This examination implemented the operator licensing requirements of 10 CFR §55.41, §55.43, and §55.45.

Seven Senior Reactor Operator (SRO) candidates and three Reactor Operator (RO) candidates received written examinations and operating tests. The written examination was administered by the licensee on April 8, 1999, and the operating tests were administered by the NRC the weeks of March 29 - April 1, 1999 and April 12 - 15, 1999.

Operations

- The submitted written examination and operating tests met the requirements of NUREG-1021. The licensee's first time at development of the NRC administered examination was considered good. (Section O5.1)

- Nine of ten candidates passed the examination. One SRO candidate failed the operating examination. (Section O5.1)

- Overall performance on the operating test was satisfactory with a strength noted in the area of annunciator response procedure usage. Weaknesses were noted in the areas of 3 way communication, and crew briefings. (Section O5.1)

- Candidate Pass/Fail

	SRO	RO	Total	Percent
Pass	6	3	9	90
Fail	1	0	1	10

Report Details

Summary of Plant Status

During the period of the examinations Unit 1 was at 100 percent power and Unit 2 was in coast down at approximately 67 percent power.

I. Operations

O5 Operator Training and Qualifications

O5.1 Initial Licensing Examinations

a. Scope

NRC examiners conducted regular, announced operator licensing initial examinations during the periods of March 29 - April 1, 1999 and April 12-15, 1999. NRC examiners administered examinations developed by the licensee's training department, under the requirements of an NRC security agreement, in accordance with the guidelines of the Examiner Standards (ES), NUREG-1021, Interim Revision 8. Five Senior Reactor Operators (SRO) upgrade, two SRO instant and three Reactor Operators (RO) applicants received written examinations and operating tests.

b. Observations and Findings

The licensee developed the SRO and RO written examinations, two Job Performance Measure (JPM) sets, and two dynamic simulator scenarios, with one spare scenario, for use during this examination. All materials were submitted to the NRC on schedule. NRC examiners reviewed, modified as necessary, and approved the examination prior to administration. The NRC conducted an on-site preparation visit during the week of March 15, 1999, to validate examination materials and familiarize themselves with the details required for examination administration.

(1) Written Examination

Organization of the submitted examination materials expedited the examination review process. Relevant portions of the reference materials were attached to each test item allowing for faster retrieval of the associated reference.

This was the licensee's first time at developing the NRC administered examinations in accordance with the pilot program guidance. The NRC noted that the quality of the licensee's submittal was satisfactory. During the initial review, the NRC examiners did not agree with all the questions designated level of difficulty assigned by the licensee. Through discussion, consensus was made concerning the level of difficulty of all individual test items. During further discussions with the licensee, test item modifications were made to question stems or distractors. The number and type of corrections to the examination were consistent for a licensee's initial effort at examination development. Aside from minor editorial changes to clarify or improve the language of the questions, the number of technical errors noted were minimal. Most requested changes were to assure clarity in the question stem and to enhance the plausibility of incorrect distractors. The NRC recommended replacing only two questions due to quality of the questions. The final examination was considered an adequate product, in that, it could identify a less than competent candidate.

(2) Operating Test Development

The NRC reviewed two walkthrough examination sets submitted by the facility, one SRO(U) and a combined SRO(I)/RO set. Portions of the two JPM sets were shared. These were comprised of job performance measures (JPMs) and administrative JPMs and administrative questions. The examiners found the JPMs were developed at the appropriate level as described in NUREG-1021. Some minor technical errors were noted such as the incorrect designation of critical steps. The NRC also noted that the quality of some of the JPM follow-up questions were weak. There were two JPM questions that were considered direct look-up which lacked operational validity. These questions were either changed or references were not allowed to answer these questions.

The NRC reviewed two simulator scenarios (plus one spare) developed for the examination. Some changes and additions were made to the scenarios to enhance the examiners opportunity to observe candidates perform all required competencies. These changes were made during the examination preparation week. Overall, the scenarios were found to be challenging and at the appropriate level of difficulty. The final scenarios were considered an excellent examination tool providing the proper level of discrimination.

During the examination weeks the examiners found that the crews did not enter E-0, "Reactor Trip or Safety Injection" when the reactor was tripped from a subcritical condition. This situation occurred during the performance of one JPM and one simulator scenario. This was a concern because the Westinghouse Owners Group (WOG) delineates the entry conditions using reactor temperature rather than degree of criticality. The WOG requires entry into E-0 when the reactor is in mode 3, or greater than 350 degrees F. The facility acted promptly to resolve this procedural implementation weakness. This item was placed in the licensee's Training Impact Report (TIR) for resolution.

During the preparation week of March 15 - 17, 1999, a potential examination compromise occurred. During the copying of simulator scenario, SE-3, the individual performing the copying inadvertently left the original copy of SE-3 in the copier. The licensee immediately notified the NRC of the potential examination compromise. The simulator scenario was not used during the examination. The licensee developed an additional spare simulator scenario for potential use. Additionally, the licensee initiated a TIR to address this problem. The TIR identified a check list that will be used to secure the copy machine when copying NRC examination materials.

The facility administered the written on April 8, 1999 in accordance with NUREG-1021. The licensee administered the examination in five hours. No time limit extensions were requested.

(3) Examination Results

The facility licensee submitted post-examination comments for two written examination questions, of which the NRC accepted (see Enclosures 3 and 4). The acceptance of these comments did not change the outcome of the grading for any candidates.

The NRC noted that the quality of the licensee's proposed examination was satisfactory and above average when compared to examinations submitted to the NRC during the pilot period.

The examiners reviewed the results of the written examination and found that ten of ten candidates passed this examination. The review of the operating examination revealed nine of ten candidates passed the examination. Overall SRO and RO candidate performance on the written examination was satisfactory. The licensee conducted a post-examination item analysis of the SRO and RO written examinations. This analysis identified one question where both SRO and RO candidates exhibited knowledge deficiencies. The analysis also identified one other SRO specific knowledge weakness and one other RO specific knowledge weakness. The examiners concluded that no generic knowledge weaknesses existed where multiple questions on the same system or topic were missed by a large number of candidates.

Examiners also identified several weaknesses in candidate performance during the operations portion of the examination. Details of the weaknesses are described in each individual's examination report, Form ES-303-1, "Operator Licensing Examination Report." Copies of the evaluations have been forwarded under separate cover to the Training Manager in order to enable the licensee to evaluate the weaknesses and provide appropriate remedial training for those operators, as necessary.

In general, these weaknesses included the following: knowledge of radiological posting requirements, and the operation of Red T-handle on the guillotine valve for the AAC diesel generator.

During scenario performance examiners noted strengths in the use of annunciator response procedures. Weaknesses were noted in the areas of 3 way communication, and crew briefings. Three-way communication, was not always implemented properly and in general, crew briefings were done sporadically between the SRO and the crew.

c. Conclusion

The submitted examination met the requirements of NUREG-1021. Nine of ten candidates passed the examination. Overall performance on the operating test was satisfactory with a strength noted in the area of annunciator response procedure usage. Weaknesses were noted in the areas of 3 way communication and crew briefing.

V. Management Meetings

X1. Exit Meeting Summary

At the conclusion of the site visit, the examiners met with representatives of the plant staff listed on the following page to discuss the results of the examinations and other issues. No proprietary material provided was provided.

PARTIAL LIST OF PERSONS CONTACTED

Licensee:

A. Brown, Supervisor-Training Support
M. Crist, Superintendent-Operation
E. Grecheck, Site Vice President
K. Grover, Operations Instructor
D. Llewellyn, Superintendent-Nuclear Training
H. McCallum, Supervisor-Operations Training
P. Orrison, Operations Instructor
E. Shore, Operations Instructor
C. Silcox, Nuclear Specialist
R. Simmons, Operations Instructor
M. Small, Supervisor-SNS
J. Spence, Operations Instructor
K. Spencer, Operations Instructor

NRC:

R. Musser, Senior Resident Inspector, Surry
K. Poertner, Resident Inspector, Surry
L. Mellen, Examiner, RII
D. Payne, Examiner, RII

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened:

None

Closed:

None

Discussed:

None

SIMULATION FACILITY REPORT

Facility Licensee: Virginia Power - Surry Power Station Units 1 & 2

Facility Docket Nos.: 50-280 and 50-281

Operating Tests Administered on: March 29 - April 1, 1999, April 12- April 15, 1999

This form is to be used only to report observations. These observations do not constitute audit or inspection findings and are not, without further verification and review, indicative of noncompliance with 10 CFR 55.45(b). These observations do not affect NRC certification or approval of the simulation facility other than to provide information that may be used in future evaluations. No licensee action is required in response to these observations.

While conducting the simulator portion of the operating tests, the following items were observed:

ITEM	DESCRIPTION
There were no simulator fidelity problems.	

Enclosure 2

VIRGINIA ELECTRIC AND POWER COMPANY
RICHMOND, VIRGINIA 23261

April 14, 1999

Regional Administrator	Serial No.	99-229
United States Nuclear Regulatory Commission	BAG	R0
Region II	Docket No.	50-280
Atlanta Federal Center		50-281
61 Forsyth Street S.W., Suite 23T85	License No.	DPR-32
Atlanta, Georgia 30303		DPR-37

Dear Mr. Reyes:

VIRGINIA ELECTRIC AND POWER COMPANY
SURRY POWER STATION UNITS 1 AND 2
WRITTEN LICENSE EXAMINATION COMMENTS

In accordance with NUREG-1021, Section ES-402, the following comments are submitted concerning the Reactor Operator and Senior Reactor Operator written initial examinations administered at Surry on April 8, 1999.

QUESTION: #24

An intermediate break LOCA has occurred on Unit 2, all four RMT channel trip annunciators are illuminated and automatic Recirculation Mode Transfer (RMT) is in progress. The amber RMT light has just illuminated. Which ONE of the following identifies valves which are expected to be cycling during this period of RMT?

a) 1-SI-MOV-1885 A/B/C/D (LHSI recirculation isolation valves)

b) 1-SI-MOV-1863 A/B (LHSI discharge to HHSI suction)

c) 1-SI-MOV-1862 A/B (LHSI suction from the RWST)

d) 1-SI-MOV-1864 A/B (LHSI discharge to cold legs)

ANSWER: c)

Reference: ND-91-LP-3, Objective E, "Recirculation Mode Transfer (RMT) System"

ENCLOSURE 3

COMMENTS:

The stem of the question referenced Unit 2, while the answers referenced Unit 1 valves. During administration of the exam, the stem was modified to reference Unit 1 as the accident Unit.

RECOMMENDATIONS:

Accept modification and leave question "as is" on the examination.

QUESTION: RO #95

Which ONE of the following identifies how a LOSS of "A" DC bus affects the operation of the associated reactor trip breaker?

a) The shunt coil will deenergize

b) The shunt coil will energize

c) The UV coil will deenergize

d) The UV coil will energize

ANSWER: c)

Reference: 1-ES-1.3, ND-93.3-LP-10, Objective B, "Reactor Trip Breaker Operation"

COMMENTS:

Both (a) and (c) are correct answers. The outcome of the examination was not affected as none of the candidates being examined picked distracter (a) and therefore had no impact on the candidate's scores. The question will be corrected for future use.

RECOMMENDATIONS:

Accept both answers (a) and (c) as correct answers.

Please find attached a copy of reference material associated with the above comments that was not included in the original reference material submittal.

If you have any questions or require additional information, please contact us.

Very truly yours,

E. S. Grecheck
Site Vice President

Attachment

Commitments contained in this letter: None

copy:

Mr. Harold O. Christensen, Chief
Operator Licensing and Human Performance Branch
United States Nuclear Regulatory Commission
Region II
Atlanta Federal Center
61 Forsyth Street, S. W., Suite 23T85
Atlanta, Georgia 30303

Mr. Richard S. Baldwin
United States Nuclear Regulatory Commission
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Atlanta, Georgia 30303

Document Control Desk
United States Nuclear Regulatory Commission
Washington, D.C. 20555

Mr. R. A. Musser
Senior Resident Inspector
Surry Power Station

ATTACHMENT

WRITTEN LICENSE EXAMINATION COMMENTS

REFERENCE MATERIAL TO SUPPORT COMMENTS

Surry Power Station - Units 1 & 2

VIRGINIA ELECTRIC AND POWER COMPANY

g. Class I

The RPS will continue to operate under seismic and accident environment conditions.

If desired, use AIA-10.1, Logic Matrices, to review how logic matrices work for the RPS. **WARNING!** In the logic diagrams, the contacts are open when the relay is de-energized. However, their normal state is energized, with the contacts closed. A protective signal causes the contacts to OPEN.

B. Reactor Trip Breaker Operation

Refer to/display H/T-10.3, Rx Trip Breaker Setup.

1. The Reactor Trip Breakers supply power from the Rod Drive M/G sets. This power is normally directed through closed Reactor Trip Breakers RTA and RTB.

2. Both reactor trip bypass breakers are normally racked in but remain open. For testing or maintenance, one bypass breaker can be closed and its corresponding trip breaker opened. For example, if RTA was to be tested, then RTB A would be closed. Then RTA could be opened without resulting in a trip.

Use H/T-10.3 to illustrate the above concept.

3. The reactor trip bypass breakers are interlocked. They are interlocked so that if both bypass breakers are closed at the same time, each bypass breaker's trip coil will be energized, and both bypass breakers will open. The effect of this interlock is that both bypass breakers cannot be closed at the same time.

4. Reactor trip and bypass breaker status lights provide breaker position indication in the

MCR.

5. **The Train A reactor trip signal controls the A Reactor Trip Breaker (RTA) and the B bypass breaker (RTB B).**

6. **The Train B reactor trip signal controls the B Reactor Trip Breaker (RTB) and the A bypass breaker (RTB A).**

7. When Train A is to be tested, then RTB A is closed. This allows the protection testing to cycle RTA open and closed to verify that the trip signals will actually cause the breaker to open. Should an actual reactor trip be necessary, then the Train B trip signal will open RTB and RTB A, causing the rods to drop.

Refer to/display H/T-10.4, Protection Circuitry.

8. The reactor trip breakers are kept closed by maintaining their undervoltage coils (UV) **energized**. When a reactor trip coincidence logic is made up, then the reactor trip bistable de-energizes, breaking the DC circuit. The undervoltage coil is de-energized, and the reactor trip breaker opens (the other train's bypass breaker also opens).

9. The Manual reactor trip pushbuttons are wired directly into the DC circuit. Pushing the trip button causes a break in the circuit.

Refer to/display H/T-10.5, Rx Trip Breaker and Bypass Breaker Trp Logics. This is a more detailed look at the DC circuitry associated with the trip breakers.

10. The DC power to the UV coil and the automatic shunt trip relay (STA) is supplied through the closed reactor trip relays and the closed contacts of the trip pushbuttons in the MCR. Any power interruption of this DC will cause the UV coil to de-energize and trip the breaker.

11. De-energizing the STA relay causes the STA relay contact to close. Closing the STA contact allows DC power to be supplied to the Trip Coil (TC). Energizing the TC to trip the breaker is a backup to tripping the breaker by de-energizing the UV coil.

Question #9 Reference

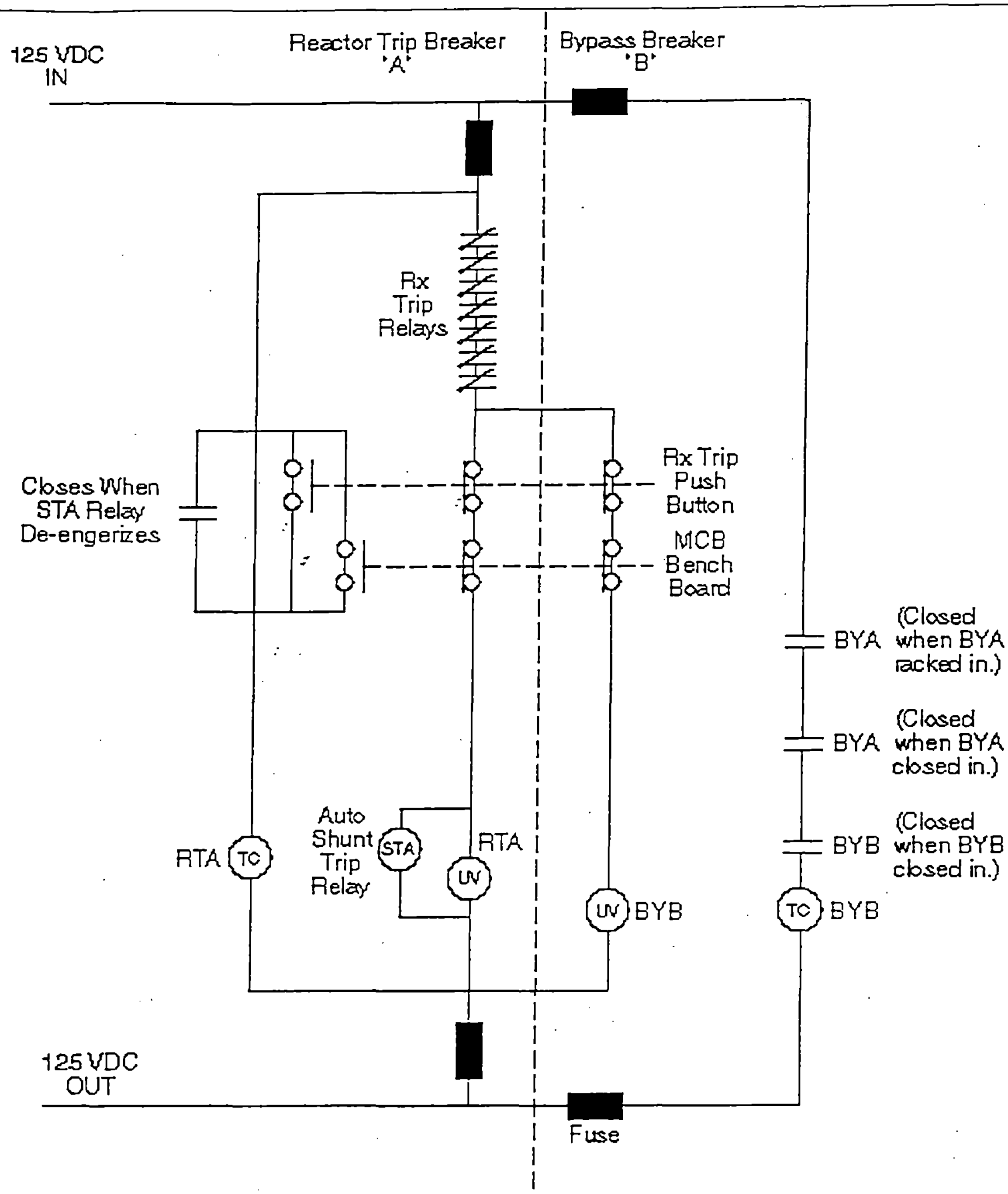
Direct trainees to look at the bypass breaker contacts on the right hand side of H/T-10.5.

12. If both bypass breakers are closed, then this logic circuit energizes the trip coil for RTB B. A similar logic circuit exists for RTB A. Note that this interlock is only in effect if both bypass breakers are racked in.

C. Reactor Trip Signals

AMSAC is not included as a direct reactor trip. AMSAC trips the reactor by opening the MG Set supply breakers. Refer to/display H/T-10.6, Reactor Trip Signals, and use the following material to cover the reactor trips, one at a time.

1. There are 17 reactor trip signals. A <u>Manual</u> reactor trip can be initiated at any time by depressing 1/2 pushbuttons on the MCR benchboard. It also functions as a backup to any automatic reactor trip.



NOTE: Trip Coils (TC) energize to trip breaker.
Under Voltage Coils (UV) de-energize to trip breaker.
Auto Shunt Trip Realy (STA) de-energizes to actuate.

Rx TRIP BREAKER A AND BYPASS BREAKER TRIP LOGICS

NRC RESOLUTION OF FACILITY RECOMMENDATIONS

Question: #24 *Recommendation accepted.* The typographical error will be corrected on the master examination.

Question: RO # 95 *Recommendation accepted.* Both answers (a) and (c) will be accepted as correct answers. The answer key will be corrected on the master examination.

ENCLOSURE 5

WRITTEN EXAMINATION AND ANSWER KEY

U. S. Nuclear Regulatory Commission
Site-Specific
Written Examination

Applicant Information

Name:	Region : II
Date: 4/8/99	Facility/Unit: Surry
License Level: RO	Reactor Type: W
Start Time: 0900	Finish Time:

Instructions

Use the answer sheets provided to document your answers. Staple this cover Sheet on top of the answer sheets. The passing grade requires a final Grade of at least 80.00 percent. Examination papers will be collected four Hours after the examination starts.

Applicant Certification

All work done on this examination is my own. I have neither given nor Received aid.

____
Applicant's Signature

Results

Examination Value	100	Points
Applicant's Score	______	Points
Applicant's Grade	______	Percent

RO ANSWER KEY

001	B	026	B	051	D	076	D
002	C	027	D	052	D	077	D
003	C	028	B	053	A	078	C
004	A	029	A	054	C	079	C
005	D	030	C	055	A	080	C
006	A	031	D	056	D	081	C
007	A	032	B	057	C	082	D
008	B	033	A	058	D	083	B
009	B	034	B	059	C	084	B
010	D	035	B	060	B	085	D
011	A	036	B	061	A	086	B
012	B	037	A	062	D	087	B
013	B	038	B	063	A	088	A
014	D	039	C	064	C	089	C
015	C	040	C	065	A	090	A
016	C	041	C	066	C	091	B
017	D	042	A	067	C	092	B
018	D	043	B	068	C	093	B
019	D	044	D	069	D	094	B
020	B	045	C	070	A	095	C or A
021	D	046	D	071	B	096	A
022	A	047	D	072	C	097	D
023	D	048	A	073	A	098	C
024	C	049	B	074	B	099	C
025	D	050	D	075	D	100	A

1.
During troubleshooting on the Rod Control System at 100% power, a Power Cabinet 2BD Non-Urgent alarm was received. The Test Director (I&C Supervisor) directs the "Internal Alarm Reset" push-button to be depressed, in accordance with a SNSOC approved test procedure. The Unit Reactor Operator mistakenly depresses the "Startup Reset" push-button. Which ONE of the following automatic responses is expected?

a) The Reactor Trip Breakers will open.

b) All control rod bank low and low-low annunciators will illuminate.

c) Group B and D group step counters reset to 0 steps, A and C groups remain at the all rods out position.

d) All IRPI indicators reset to 0 and all rod bottom lights illuminate (actual rod position does not change.

2.
Which ONE of the following would initiate an automatic Unit 2 Reactor trip?

a) An underfrequency condition of 49 hertz on the Unit 2 "B" and "C" Station Service busses with Unit 2 stable at 2% power.

b) An overcurrent trip of the Unit 2 "B" RCP with reactor power at 25% power.

c) With reactor power at 45%, the Reactor Operator manually secures the Unit 2 "A" RCP due to high shaft vibrations.

d) The Unit 1 Reactor Operator accidentally opens breaker 15D1, RES STA SERVICE XFER SUP BKR, with Unit 2 at 100% power.

3.
Which ONE of the following events would require AP-39.00, NATURAL CIRCULATION OF RCS, to be initiated to establish/verify Natural Circulation Flow?

a) A major steam line break where at least one Steam Generator has pressure which is 350 psig less than RCS pressure.

b) A Small Break LOCA in which RCS subcooling is 28 °F with containment pressure at 24 psia.

c) Following a reactor trip, all Station Service busses fail to automatically swapover to Reserve Station Service.

d) While at 200°F/300 psig, the running RHR pump trips and the standby pump cannot be started.

4.
With a saturated mixed bed ion exchanger in service, 1-CC-TCV-103, CC return from NRHX, drifts 30% in the closed direction. Letdown temperature is now 139°F. Which ONE of the following plant responses is expected?

a) Rods step out slowly.

b) Rods step in slowly.

c) 1-CH-TCV-1143 (Letdown IX Temperature Divert Valve) automatically diverts letdown flow around the Ion Exchangers.

d) 1-CH-PCV-1145 (Letdown Pressure Control Valve) throttles closed to maintain letdown pressure.

5.
The following conditions exist:
- Unit 1 is at 100% power.
- 1-CC-TV-120B ("B" RCP Thermal barrier return) and its manual isolation valve, 1-CC-57, are closed to isolate a small thermal barrier leak on the "B" RCP.
- All other systems are operating normally.

Which ONE of the following events would allow continued power operation of Unit 1?

a) 1-CC-TV-105A (RCP "A" CLR CC RTN TV) closes and will not reopen.

b) 1-CH-HCV-1186 (RCP Seal injection flow) closes and will not reopen.

c) Actual seal leakoff flow on "C" RCP increases off scale high.

d) 1-CH-MOV-1381 (RCP Seal Return) closes and will not reopen.

6.
Ten Minutes ago a Steam Dump valve failed partially open at 2% power.
The Operating team has corrected the problem.
Reactor power reached a maximum of 5.7% Power Range indication, currently stable at 2%.
RCS pressure reduced to a minimum of 2175 psig and is recovering.
RCS temperature reduced to a minimum of 529°F and is recovering.

Which ONE of the following identifies the **Tech Spec LCO** that has been exceeded?

a) Section 3.12 DNB Low Pressure limit.

b) Section 2.1 Low Pressure Safety limit.

c) Section 3.1 Minimum Temperature for Criticality limit.

d) Section 2.3.2 Overpower Delta T limit.

7.
During 20% steady state reactor power operation, "A" Steam Generator PORV fails full open. Which ONE of the following describes the operation of the control rods to this event?

a) The control rods would move out due to a Tave/Tref mismatch.

b) The control rods would not move.

c) Control rods would not be affected, only power would increase.

d) The rods would trip into the core due to high steam line flow SI being generated.

8.
Following a major steam line break on the Unit 1 "B" Main Steam line, the Reactor Operator is directed to control RCS temperature following "B" Steam Generator dryout.
Which ONE of the following identifies the basis for performing this action?

a) Prevent Pressurizer Relief Tank (PRT) rupture leading to possible Containment integrity concerns.

b) Prevent RCS repressurization, leading to possible Pressurized Thermal Shock concerns.

c) Minimize the temperature perturbation on the RCS which could lead to possible "B" SG tube failure.

d) Minimize RCS heatup which could cause a loss of the subcooling margin necessary to secure SI.

9.
Which ONE of the following indications, if sustained for 5 minutes, on 1-CN-PR-101A and 101B, CONDENSER VACUUM, would require an immediate reactor trip?

a) 25" Hg at 100% power.

b) 25.5" Hg with turbine power at 20%.

c) 26" Hg with turbine power at 35%.

d) 28" Hg and decreasing at a rate of 0.5" every 2 minutes.

10.
The following conditions exist:

- Unit 2 is at Hot Shutdown preparing for a Unit startup.
- SG narrow range levels are 35% in all Steam Generators.
- All decay heat is being removed via SG blowdown.
- SGs are being fed from the "A" Main Feed Pump through the Main Feed Bypass HCVs.
- Main Steam Trip valves are open.

A Station Blackout occurs (Emergency Busses are reenergized from their associated EDGs) simultaneous with a loss of all Unit 2 Instrument air.

Which ONE of the following identifies an expected response during the initial phase of the transient (Assume no Operator actions are taken)?

a) Steam Generator Blowdown is diverted to the river.

b) Pressurizer level decreases.

c) Main Feed Regulating Valve demand increases.

d) Auxiliary Building Central Ventilation realigns to filtered exhaust.

11.
With Unit 1 at 100% power and all systems functioning normally, 1-CH-LT-1112 (VCT level) fails low due to Vital Bus 1-IIIA breaker 26 tripping open. Which ONE of the following is the expected plant response for this transient?

a) No apparent response other than 1-CH-LI-1112 failed low.

b) Charging pump suction MOVs swap to the RWST from the VCT (1-CH-MOV-1115B/D open, 1-CH-MOV-1115C/E close).

c) Letdown flow diverts to the Primary Drain Tank (1-CH-LCV-1115A diverts).

d) Automatic VCT makeup actuates.

12.
A fire has been reported in the Unit 1 Relay Room.

Which ONE of the following actions should be taken to extinguish the fire?

a) Rotate the Unit 1 selector switch on the halon control panel. (located in the Unit 1 Turbine basement near the Unit 1 blowdown coolers).

b) Depress the Unit 1 Halon Push-button on the Unit 2 side of the Control Room (Behind the Vertical Panel) .

c) Actuate the LP CO_2 pull station located in the Unit 1 ESGR (At the entrance to the Unit 1 cable vault).

d) Actuate the Halon Pull station located in the Unit 1 Turbine Building basement (at the entrance to the Unit 2 ESGR).

13.
During a Limiting Fire in the Main Control Room, swapover for the Unit 2 Station Service busses failed. All Unit 2 Station Service busses are deenergized. All emergency busses are energized by off-site power. Which ONE of the following identifies the heater capacity available at the Unit 2 Auxiliary Shutdown Panel?

a) 400 KW

b) 450 KW

c) 500 KW

d) 600 KW

14.
The failure of a "B" train Hi CLS (phase II) relay has caused a partial initiation of "B" train of Hi CLS. Annunciators AF3, SI INITIATED TRAIN A, and AF4, SI INITIATED TRAIN B, are ***NOT*** lit. Also, ***NO*** automatic actions associated with SI have occurred. The Unit remains at 100% power.

Which ONE of the following manual actions will need to be performed to recover from this event

a) Secure the 1-VS-F-58B and realign the auxiliary ventilation system to normal status.

b) Align the Containment Instrument Air Compressors to an outside alignment.

c) Secure the Hydrogen Analyzer Heat Tracing.

d) Realign the Containment Particulate and Gaseous Radiation Monitor (1-RMS-159/160) to the Containment.

15.
During 75% reactor power operation, which ONE of the following events would place the unit closer to a Departure from Nucleate Boiling (DNB) condition? (Assume all systems and components are operable and in auto.)

a) Group "C" pressurizer heater output fails to high output.

b) Selected 1st stage impulse pressure fails low.

c) Median Tave fails low.

d) 1-RC-PT-1445, PRESSURIZER PRESSURE CONTROL CHANNEL II, fails low.

16.
The Operating team entered FR-C.1, RESPONSE TO INADEQUATE CORE COOLING. The team failed in all attempts to establish High Head Flow. Core Exit Thermocouples are 820°F and increasing slowly.

Which ONE of the following methods is required to respond to the core cooling challenge?

a) Open available Pressurizer PORVs to lower pressure to the SI accumulator and LHSI injection pressures.

b) Open the Pressurizer and Head vent SOVs to allow venting of any hard bubble and allow natural circulation to progress.

c) Depressurize all intact Steam Generators to 150 psig to allow RCS depressurization to the SI accumulator and LHSI injection pressures.

d) Enter the Severe Accident Mitigation Guidelines.

17.
Following a reactor trip and safety injection, indications of extensive failed fuel exist. Which ONE of the following criteria would require the Main Control Room Staff to begin using conservative setpoints due to the potential unreliability of installed instrumentation?

a) Specific activity >10 microcuries/cc dose equivalent I-131, as indicated on any RCS sample.

b) SI accumulator level indicators deviate from pre-trip levels.

c) RCS subcooling indicates 26°F on ICCM.

d) 1.3E5 R/hr as indicated on the Containment High Range Monitor.

18.
With Unit 1 operating at 100% power (normal operating pressure and temperature), the median Tave output fails to 570°F. If "D" bank started at 210 steps, which ONE of the following approximates how many ***seconds*** would elapse to reach the all out rod position on "D" bank indications?
(Assume rod motion does not affect temperature).

a) 22

b) 29

c) 112

d) 143

19.
During recovery of a dropped rod, an urgent failure alarm is received immediately after initiating rod motion on the dropped rod. Which ONE of the following identifies the cause of this alarm?

a) Non-urgent failure condition coincident with rod motion demand.

b) Deviating condition is generated in the Bank Overlap Unit.

c) Disagreement between Individual Rod Position Indicators and Group Step Counters.

d) The lift coils of the remaining rods in the affected bank are deenergized.

20.
During normal 100% operation the RO acknowledges annunciator C-C-8, PRZR HI LEVEL HTRS ON.
RCS pressure is 2203 psig and decreasing slowly.
VCT level trend is decreasing slowly.
Which ONE of the following diagnoses this off normal trend?

a) Pressurizer heaters have failed to minimum output.

b) Pressurizer level detector reference leg has separated from the Pressurizer.

c) 1-CH-FCV-1122, CHG FLOW CONT, failed open.

d) 1-CH-HCV-1200A, LETDOWN ORIFICE ISOL, failed closed.

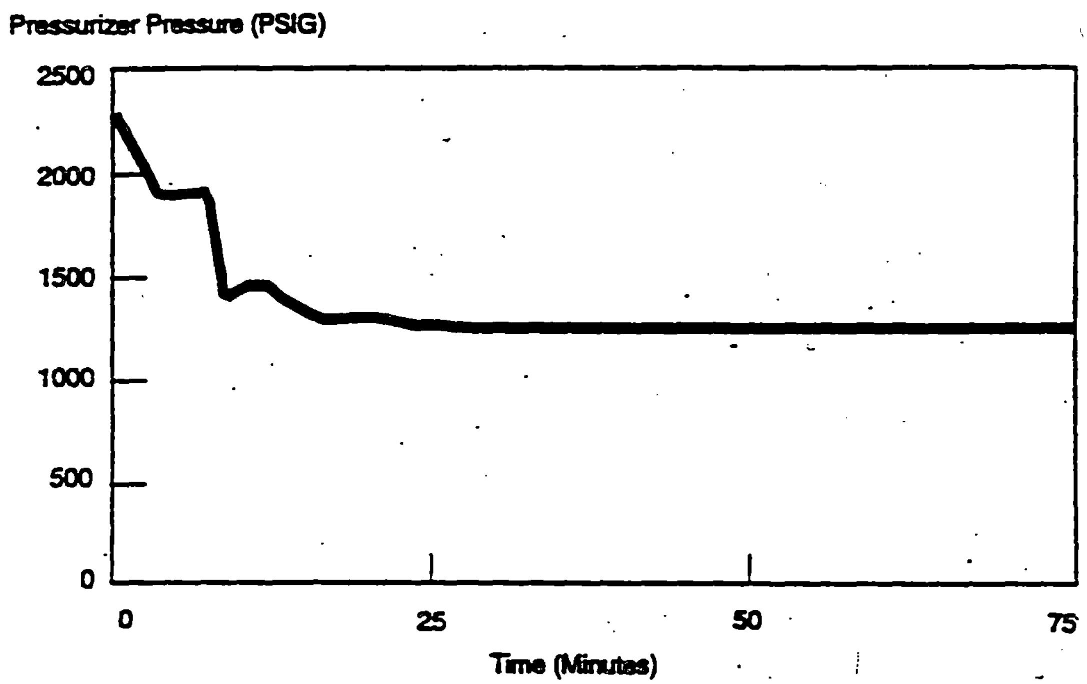
21.
The graph included below gives typical RCS pressure response following a Small Break LOCA. Which ONE of the following identifies the cause of stable RCS pressure from time 25 minutes to 75 minutes?

a) All charging pumps have reached their low pressure auto start setpoint.

b) RCS level has decreased out of the Pressurizer and surge leg. This pressure is indicative of Reactor Vessel head pressure.

c) RCP trip criteria was met and the change in slope is indicative of static RCS pressure.

d) SI flow has matched break flow.



22.
During a Large Break LOCA, a low pressurizer pressure SI signal initiates an automatic trip of the unit's Main Feed Pumps. Which ONE of the following identifies the purpose of this trip?

a) Feedwater isolation is designed for a main steam line break accident and does not provide any substantial benefits during a LBLOCA.

b) Minimizes the thermal stresses associated with rapid RCS depressurization by minimizing the feed water injection.

c) Required to allow AFW pumps to deliver cooler water to the SG. This minimizes the temperature differential across the SG tubes.

d) Allows the RSSTs to maintain a constant voltage profile during the accident with additional required loads.

23.
The following Unit 2 conditions exist:

- The unit has sustained a Small Break Loss Of Coolant Accident (SBLOCA).
- The team is in 2-E-1, LOSS OF REACTOR OR SECONDARY COOLANT.
- Pressurizer pressure is 1405 psig.
- Pressurizer level is off-scale low.
- RCS subcooling is 29°F.
- Containment Pressure is 13 psia.

Which ONE of the following identifies the procedure(s) which will provide long term guidance to stabilize the plant given the above RCS conditions?

a) 2-GOP-2.4, Unit Cooldown, HSD to 351°F.

b) 2-ES-0.2, NATURAL CIRCULATION COOLDOWN.

c) 2-ES-0.3, NATURAL CIRCULATION COOLDOWN WITH STEAM VOID IN RX VESSEL.

d) 2-ES-1.2, POST LOCA COOLDOWN AND DEPRESSURIZATION.

24.
An intermediate break LOCA has occurred on Unit 1, all four RMT channel trip annunciators are illuminated and automatic Recirc Mode Transfer (RMT) is in progress. The amber RMT light has just illuminated. Which ONE of the following identifies valves which are expected to be cycling during this period of RMT?

a) 1-SI-MOV-1885 A/B/C/D (LHSI recirculation isolation valves).

b) 1-SI-MOV-1863 A/B (LHSI discharge to HHSI suction).

c) 1-SI-MOV-1862 A/B (LHSI suction from the RWST).

d) 1-SI-MOV-1864 A/B (LHSI discharge to cold legs).

25.
The operating team is responding to a Unit 1 "A" main steam line break inside containment. The following conditions presently exist:

- The operating team has transitioned from 1-E-2, FAULTED STEAM GENERATOR ISOLATION, to 1-E-1, LOSS OF REACTOR OR SECONDARY COOLANT.
- "A" SG is dry with pressure <100 psig.
- "B" SG narrow range level is 20% and increasing, with pressure stable at 950 psig.
- "C" SG narrow range level is 5% and increasing, with pressure stable at 950 psig.
- Pressurizer level is 15% and increasing.
- Containment pressure is 18 psia and decreasing.
- RCS subcooling indicates stable at 120°F.
- RCS pressure is 2150 psig and increasing slowly.

Which ONE of the following conditions will complete the transition criteria from 1-E-1 to 1-ES-1.1, SI TERMINATION?

a) "C" SG level increases to 14%.

b) "B" SG level increases to greater than 25%.

c) RCS pressure increases to 2215 psig.

d) Pressurizer level increases to 27%.

26.
A Process Vent "HIGH" alarm was initiated from the 1-GW-RI-101, PRCS VENT PART, Radiation Monitor.
Which ONE of the following **AUTOMATIC** action(s) must be verified?

a) HCV-GW-106, AERATED VENT ISOLATION, closed.

b) FCV-GW-101, DECAY TK BLEED ISOL, closed.

c) 1-VS-MOV-100A-D, CTMT PURGE ISOLATION, closed.

d) 1-CV-P-1A/1B, CTMT VACUUM PUMPS, off.

27.
The following conditions exist while RHR is in service on Unit 1:

- Hot leg temperature is 232°F.
- RCS pressure is 310 psig and decreasing rapidly.
- Containment sump level is increasing.
- Containment pressure is 9.8 psia and increasing slowly.
- Pressurizer level is 19% and decreasing.
- Charging flow is at maximum.

Which ONE of the following describes the reason that the Safety Injection pushbuttons are **NOT** depressed in response to these conditions?

a) Results in a no-load condition for the Emergency Diesel Generators since ESF components are in Pull to Lock.

b) SI Accumulator isolation MOVs will automatically open, leading to a possible OPMS actuation.

c) PTS concern when HHSI flow is established to the cold legs.

d) Phase I Containment isolation, leading to CC loss in the Containment.

28.
Which ONE of the following indication(s) in conjunction with Power Range Nuclear Instruments provide **ALLOWABLE** assurance that a Reactor Trip has occurred?

a) Annunciator E-A-8 "Reactor Tripped by Turbine Trip" backlit red.

b) Reactor Trip Breaker indicating lights illuminated green.

c) All Individual Rod Position Indicators (IRPI) at 0 steps.

d) Annunciator F-B-3 "AMSAC Initiated" illuminated white.

29.
During 70% power operation Intermediate Range Nuclear Instrument N-36 fails low. Which ONE of the following actions is required to respond to this event?

a) Place the "Level Trip" switch in "Bypass".

b) Reduce power to less than 10%.

c) Perform E-0, REACTOR TRIP OR SAFETY INJECTION, due to an automatic reactor trip.

d) Remove the Instrument power fuses.

30.
The following Unit 2 conditions exist:

- 100% operation.
- KAMAN annunciator A-6 "UNIT 2 MN STEAM ABC RAD MON ALERT/HI" illuminated.
- Air ejector radiation monitor reading 00.0E0
- Charging line flow is 142 gpm being controlled manually.
- Letdown flow is gpm 100 gpm.
- Combined seal return flow is 8 gpm.
- Total seal injection flow is 27 gpm.
- RCS Tave is stable
- Pressurizer level is stable at 55%
- The team is initiating 2-AP-16.00.

Which ONE of the following describes the expected procedure transition(s) for the given conditions?

a) Go to 2-AP-24.00, MINOR SG TUBE LEAK.

b) Go to 2-AP-24.01, LARGE STEAM GENERATOR TUBE LEAK.

c) Go to 2-E-0, REACTOR TRIP OR SAFETY INJECTION, and initiate 2-AP-24.01, LARGE STEAM GENERATOR TUBE LEAK.

d) Go to 2-E-0, REACTOR TRIP OR SAFETY INJECTION, with eventual transition to 2-E-3, STEAM GENERATOR TUBE RUPTURE.

31.
Which ONE of the following identifies the power level above which a manual reactor trip is required for a loss of one out of two running Main Feed pumps?

a) No trip required unless the unit approaches an automatic trip setpoint.

b) Power greater than 65% power.

c) Power greater than 75% power.

d) Power greater than 85% power.

32.
Unit 1 is at 400°F with a stable 30°F/hr heatup in progress for the last 4 hours. The following conditions also exist:

- All Main Steam Trip Valves and bypass valves are closed.
- All feed to all steam generators is secured.
- For the last 30 minutes "A" SG narrow range levels have increased from 35% to 55%, prior to this levels were stable at 35%.
- "B" and "C" have been steady at 35% throughout the heatup.
- No other abnormal indications exist.

Which ONE of the following events would produce the indicated parameters.

a) Small steam break on "A" Steam Generator.

b) "A" Steam Generator Tube Rupture.

c) Small feed line break on "B" and "C" Steam Generators.

d) Variable leg leak on "A" Steam Generator.

33.
Which ONE of the following lists the proper order of cooling restoration to Unit 1 (by priority) directed by 1-FR-H.1, RESPONSE TO LOSS OF SECONDARY HEAT SINK? Assume all Steam Generator Wide Range levels are 40%.

a) Unit 1 AFW, Unit 2 AFW, MFW, Condensate, Bleed and Feed.

b) MFW, Unit 1 AFW, Unit 2 AFW, Condensate, Bleed and Feed.

c) Feed and bleed, Unit 1 AFW, Unit 2 AFW.

d) Unit 2 AFW, Unit 1 AFW, MFW, Condensate, Bleed and Feed.

34.
During a Steam Generator Tube Rupture event, Health Physics determines Unit 1 Turbine Building sump activity levels are in excess of permissible release values and cannot be discharged to the river. Which ONE of the following provides the required course of action per E-3, STEAM GENERATOR TUBE RUPTURE?

a) Leave the water in the Turbine Building sump to allow radioactive decay.

b) Pump the Turbine Building Sump to the CCHX trough.

c) Use condensate to dilute the sump to acceptable release values.

d) Pump the Turbine Building sump to the Unit 1 Waste Neutralization sump.

35.
Which ONE of the following immediate actions is required upon notification from the Containment that a fuel handling accident has occurred, coinciding with this report are 1-RM-RI-162 "Manipulator Crane" ALERT and HIGH alarms?

a) Secure running 1-VS-F-58 fan(s) to isolate possible release paths.

b) Close MCR Motor Operated Dampers (1-VS-MOD-103C/D) and then manually initiate an air bottle dump.

c) Place the fuel in the nearest safe location and then evacuate the Containment.

d) Mobilize the Containment Closure Team, to set Refueling Containment Integrity.

36.
At 100% power operation, the lower selected channel of pressurizer level fails low. Which ONE of the following identifies the unit response with no operator action?

a) Charging pump suction (1-CH-MOV-1115B/D open, 1-CH-MOV-1115C/E close) swaps to the RWST.

b) Charging flow decreases to the minimum setpoint (25 gpm).

c) Pressurizer heater output goes to maximum.

d) No effect on system operation.

37.
Unit 2 has sustained a Hi Hi CLS due to a Large Break LOCA. One minute following the Hi-Hi initiation signal the "B" Reserve Station Service Transformer locked out. The corresponding EDG auto started and loaded on the emergency bus as required.
Assume it is now 70 seconds after the EDG loaded on the emergency bus.
All the listed loads are not running. Which ONE of the following loads has **not sequenced properly** onto the **affected** emergency bus?

a) "B" 58 fan (while on the normal supply).

b) "E" group of pressurizer heaters.

c) "A" Motor Driven AFW pump.

d) "B" ISRS pump.

38.
Assuming that station instrument air system is in a normal system alignment with the Unit 1 Service Air Compressor (1-SA-C-1) tagged out, which one of the following identifies the automatic system response to a trip of the Unit 2 Service Air Compressor (2-SA-C-1).

a) 1-CP-FCV-101, CP AIR SUPPLY TO TURBINE BUILDING, automatically opens to supply air to both units instrument air systems.

b) Both Units Instrument Air compressors auto start (1/2-IA-C-1) to supply air to the respective Units Instrument Air system.

c) The first Unit's Instrument Air compressor (1-IA-C-1/2-IA-C-1) to start will load and supply both Units Instrument Air systems.

d) The Sullair Diesel automatically starts to supply both Units Instrument and Service air systems.

39.
Unit 1 is operating at 100% power with rod control in manual due to hunting. The RO verifies that rods start to move inward as seen on the rod direction light and a decrease in Group Step Counters.

Which ONE of the following identifies the required Reactor Operator response to this transient?

a) Verify a proper temperature mismatch exists for the given rod speed.

b) Place rods in automatic and verify rod motion stopped.

c) Trip Unit 1 reactor.

d) Depress the "Startup Reset" pushbutton.

40.
The following conditions exist on Unit 2

- A large Steam line break has occurred inside the Unit 2 Containment.
- The team is performing 2-E-0, REACTOR TRIP OR SAFETY INJECTION, actions.
- "B" SG pressure is 100 psig and decreasing slowly.
- RCS pressure is 1725 psig and slowly recovering.
- RCS temperature is 501°F and decreasing slowly.
- Containment pressure is 26 psia and increasing slowly.
- Total Safety Injection flow is 420 gpm.

Which ONE of the following identifies the desired status of the Unit 2 Reactor Coolant Pumps?

a) Leave running to ensure even mixing of the injected RWST water.

b) Secure due to the status of RCS subcooling.

c) Secure due to the status of RCP motor cooling.

d) Leave running because RCS conditions are not conducive for Natural Circulation flow.

41.
Assume a typical Reactor Startup is in progress. 1/M plot data is as follows:

- At 98 steps "C" Control Bank (CB) Source Range (SR) counts were 800 cps.
- At 143 steps "C" CB, SR counts were 1300 cps.
- At 188 steps "C" CB, SR counts were 2500 cps.

Using the attached ICCR plot, which ONE of the following gives the current projected "D" bank Critical Rod Height?

a) 85 steps.

b) 98 steps.

c) 110 steps.

d) 135 steps.

ATTACHMENT 2

(Page 1 of 1)

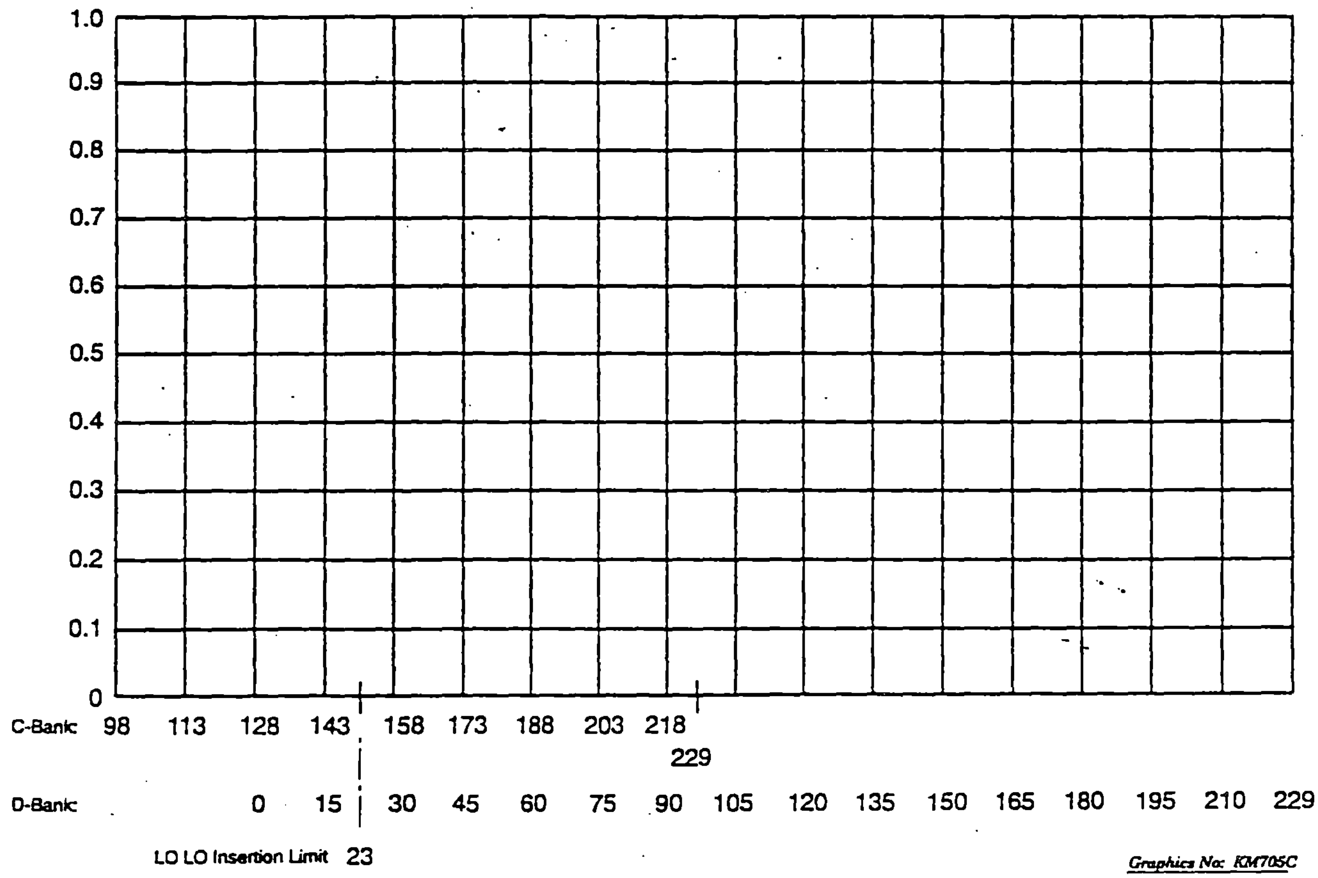
INVERSE COUNT RATE RATIO PLOT (PULL TO CRITICALITY)

Start Up No. ________
Ch: SR ________
IR ________

ECP: Bank _____ at _____ Steps
-500 PCM _____ Steps
+500 PCM _____ Steps

ACP: Bank _____ at _____ Steps
CB ________

Time	Steps		Power Level		1/M	Time	Steps		Power Level		1/M
	C	D	SR	IR			C	D	SR	IR	



INVERSE COUNT RATE RATIO PLOT

Performed by: ____________________ ________ ____________________ ________
Signature Initial Print Date

42.
Which ONE of the following is **NOT** a function provided by the CVCS system?

a) Purify Spent Fuel Pit water.

b) Purify Reactor Cavity water.

c) RCS solid plant pressure control.

d) Inventory control during mid-loop operations.

43.
Which ONE of the following conditions results in automatic initiation of Hi-Hi CLS?

a) Loss of Vital bus IV and IVA.

b) Containment Pressure protection channels 1 and 4 reach 24 psia, with channel 3 in trip.

c) Actual containment pressure increases to 18 psia.

d) Loss of Power to the "A" and "B" train Hi-Hi CLS cabinets.

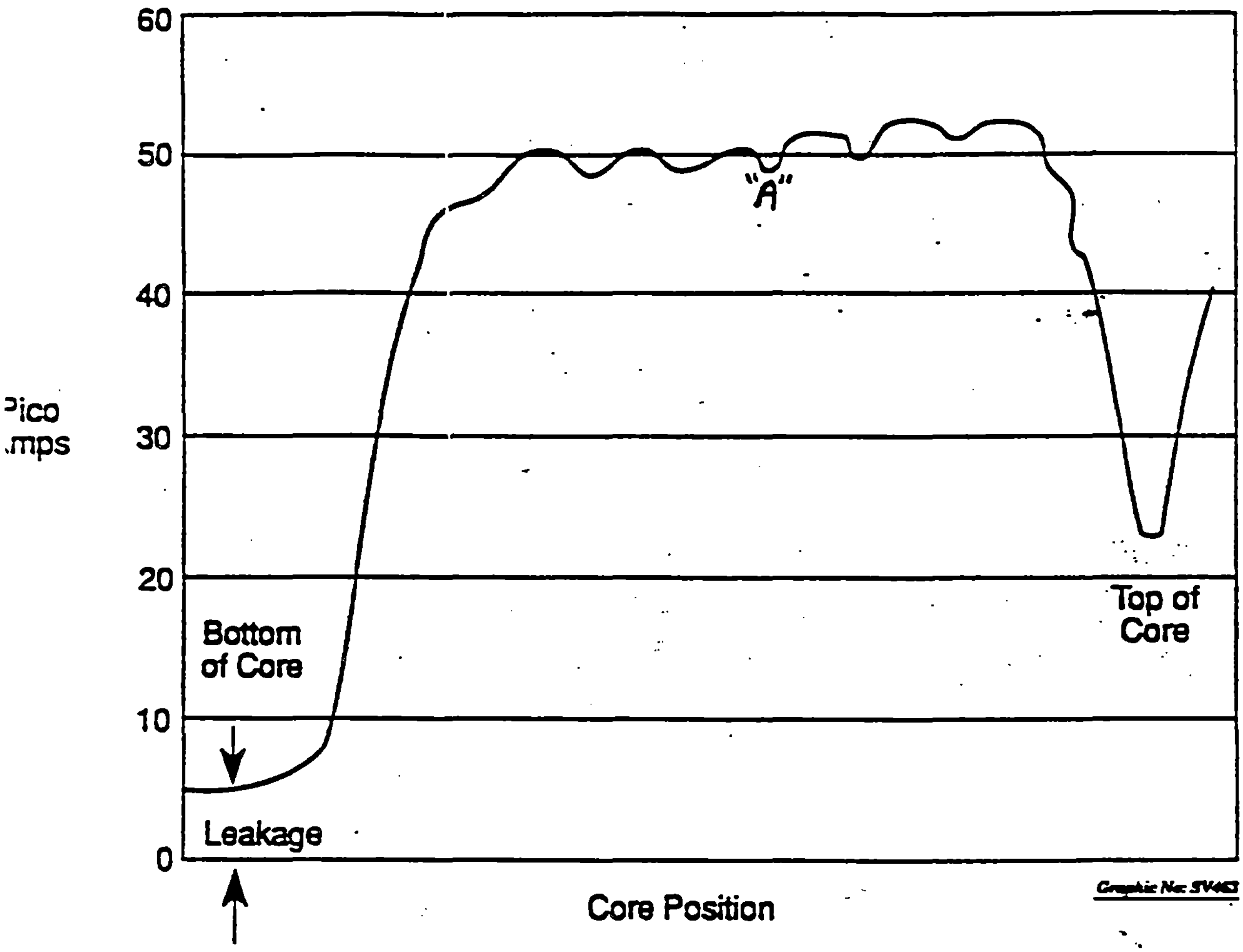
44.
Using the attached Flux Map for reference, which ONE of the following identifies the cause of the large perturbation at point "A"?

a) Region of high fuel enrichment.

b) Region of low fuel enrichment.

c) Region of partial core boiling.

d) Grid strap location.



GAMMA RESPONSE TRACE

45.
With the Unit at 100% power, a control rod is suspected of partially dropping into the core. Currently the control rod indicates approximately 40 steps by Individual Rod Position Indication (IRPI). This control rod is located adjacent to Power Range instrument N-44. Which ONE of the following selections support diagnosis of an actual mispositioned rod?

a) The N-44 benchboard meter reads higher than the other three, due to lower localized temperatures in the RCS.

b) All Power Range Nuclear instruments will read slightly lower than pre-event values due to negative reactivity insertion.

c) Annunciator G-H-1, NIS DROPPED ROD FLUX DECREASE > 5% PER 2 SEC, illuminated due to high localized power reduction.

d) N-44 delta flux indicates more positive due to flux shift to a higher enriched region of the core.

46.
Which ONE of the following describes the power supply arrangement to the Containment Air Recirc Fans?

a) All emergency bus powered.

b) All Station Service bus powered.

c) One emergency bus powered, two station service bus powered.

d) One station service bus powered, two emergency bus powered.

47.
During normal 100% power operations on Unit 1, all three Steam Generator Feed Flow/Steam Flow mismatch alarms illuminate. The SRO immediately recognizes both Main Feed Pumps are running and that 1-CN-FCV-107, Condensate Recirculation Valve, indicates full open.
Which ONE of the following identifies the required procedural actions of the Reactor Operator?

a) Immediately trip the reactor and perform the immediate operator actions of E-0.

b) Immediately start the 3rd Condensate Pump. If feed flow does not increase above steam flow, trip the reactor and perform the immediate operator actions of E-0.

c) Immediately brief and dispatch an Operator to locally close 1-CN-FCV-107.

d) Immediately start the 3rd Condensate pump and reduce turbine load to decrease steam flow below feed flow.

48.
During 35% power operation, selected first stage impulse pressure transmitter 1-MS-PT-446 (1st Stage Impulse Pressure) fails low. Which ONE of the following identifies the effect on the Steam Generator Level Control system? (Assume all Main Feed Regulating Valves are in automatic)

a) Steam Generator levels decrease to 33%.

b) Steam Generator levels remain at 44%.

c) Main Feed Regulating Valve controllers all shift to auto-hold.

d) Manual Control of Main Feed Regulating valves is required to prevent a reactor trip.

49.
Which ONE of the following identifies the purpose of isolating AFW to a faulted SG?

a) Minimize thermal stresses on the SG tubes.

b) Minimize RCS cooldown.

c) Prevent underfeeding the intact Steam Generators.

d) Minimize AFW pump runout.

50.
Which ONE of the following identifies a basis for the ES-0.1, REACTOR TRIP RESPONSE, direction to realign Main Feed Water flow to the Steam Generators?

a) Minimize long-term SG U-tube thermal stresses generated by prolonged AFW system operation.

b) Minimize potential for AFW pump overheating.

c) Allow rapid recovery of Steam Generator Inventory.

d) Conserve Emergency Condensate Storage Tank Level.

51.
Which ONE of the following tanks **cannot** be pumped DIRECTLY to the Surry Radwaste Facility (SRF)?

a) Contaminated Drain Tank.

b) High Level Liquid Waste Tank.

c) Low Level Liquid Waste Tank.

d) Primary Drains Tank.

52.
During performance of fuel reconstitution in the Fuel Building the following monitors indicate "HIGH" alarms:
1-RM-RI-153, FUEL PIT BRDG
1-VG-RI-109, VENT VENT PART
1-VG-RI-110, VENT VENT GAS

Which ONE of the following response(s) is expected?

a) 1-VS-F-58A/B automatically start and align to ventilate the Fuel and Auxiliary Buildings.

b) Fuel Building exhaust dampers swap to filtered exhaust through 1-VS-F-59.

c) Fuel Building supply fans and unfiltered exhaust fans trip.

d) No automatic actions are expected.

53.
Which ONE of the following failures would initiate an "APPROACH TO SATURATION TEMPERATURE " alarm?

a) RCS wide range pressure transmitter 1-RC-PT-1402 fails low.

b) Train "A" RVLIS fails low.

c) Median Tave 1-RC-TI-1408A, fails high.

d) Any CETC fails high.

54.
The following conditions exist during 100% Unit 2 operation:
AP-16.00 has been performed up to step 6, CHECK SI - NOT REQUIRED.
RCS pressure is 2014 psig decreasing slowly.
RCS temperature is stable at 573°F.
Pressurizer level is 49% and decreasing slowly.
Letdown flow is 0 gpm.
Charging flow is 145 gpm.
Annunciator VSP-F4, AUX BLDG SUMP HI LEVEL, is illuminated.
Containment Sump Narrow Range level is 40% and stable.

Upon transition from E-0, REACTOR TRIP OR SI, which ONE of the following procedural flowpaths is the team expected to use to mitigate this event?

a) ES-0.1, REACTOR TRIP RESPONSE to GOPs for cooldown.

b) E-1, LOSS OF REACTOR OR SECONDARY COOLANT to ECA-1.1, LOSS OF EMERGENCY COOLANT RECIRCULATION.

c) ECA-1.2, LOCA OUTSIDE CONTAINMENT, to E-1, LOSS OF REACTOR OR SECONDARY COOLANT.

d) E-1, LOSS OF REACTOR OR SECONDARY COOLANT, to ES-1.2, POST-LOCA COOLDOWN AND DEPRESSURIZATION.

55.
Which ONE of the following identifies the time requirement and basis for securing one of two running Low Head Safety Injection pumps if RCS pressure is greater than 185 psig?

a) 30 minutes, due to undersized recirculation piping.

b) 60 minutes, due to undersized recirculation piping.

c) 30 minutes, prevent heating RWST water above Tech Spec limits.

d) 60 minutes, prevent heating RWST water above Tech Spec limits.

56.
With Unit 1 at 100% power, describe unit response to a failure of 1-RC-PT-1456, PZR PRESSURE PROTECTION CHANNEL II, in the high direction. Assume 1-RC-PT-1455, PZR PRESSURE PROTECTION CHANNEL I had previously failed low and is now in trip. Assume all other plant systems are operable.

a) Both Pressurizer spray valves modulate open.

b) Pressurizer Power Operated Relief Valve, 1-RC-PCV-1456 opens.

c) Channel II OPDT activates.

d) High pressure reactor trip.

57.
Which ONE of the following identifies the cause of a loss of maximum Pressurizer heater capability?

a) Loss of Motor Control Center 1J1-2.

b) Any pressurizer heater group breaker loses its associated DC control power.

c) 1-RC-LT-1459, PRESSURIZER LEVEL PROTECTION CHANNEL I, fails low while selected to the upper channel.

d) Load Shed.

58.
Which ONE of the following will generate a "COMPUTER PRINTOUT ROD CONT SYS" alarm at 60% power.

a) "ROD CONT MODE SELECTOR" switch in any position other than "AUTO" or "MAN".

b) Any rod control power or logic cabinet experiences a "NON-URGENT" alarm.

c) Any high power rod stop permissive is met.

d) Any IRPI deviates from it's Group step counter by 10 steps.

59.
A 2-OPT-RC-10.1, RCS LEAKAGE MANUAL CALCULATION, has just been performed with Unit 2 at HSD. The STA feels the 0.5 gpm identified leakage increase is from the Reactor Vessel Flange Leakoff Line. Which ONE of the following indications would you use to verify the STA's diagnosis?

a) Trend for increased PRT level.

b) Monitor for increased Containment sump level.

c) Monitor for increased PDTT level.

d) Trend for increasing VCT level.

60.
Aside from providing flow to the Containment Spray Rings, which ONE of the following identifies a function of the Containment Spray Pumps?

a) Cool the ISRS pump recirculation flow to aid NPSH.

b) Provide water to the OSRS pump suction.

c) Provide flow to one half of each Recirculation Spray ring.

d) During outages, provide rapid RWST temperature reduction.

61.
Which ONE of the following components MUST be in service during refueling operations to allow Containment purge to remain in operation.

a) At least one Containment Air Recirculation Fan (1-VS-F-1A/B/C).

b) Manipulator Crane radiation monitor (1-RM-RI-162) <u>AND</u> RX CTMT radiation monitor (1-RM-163)

c) Manipulator Crane radiation monitor (1-RM-RI-162) <u>OR</u> RX CTMT radiation monitor (1-RM-163).

d) At least one Containment purge supply fan (1-VS-F-4A/B).

62.
Which ONE of the following design features prevent a loss of Spent Fuel Pool level if a SFP cooling system leak develops?

a) Suction weir and Cooling pump discharge check valves both located 20 feet above the fuel.

b) Cooling pump low level lockout and automatic high volume firemain makeup.

c) Return line siphon breaker, and SFP bridge radiation monitor automatic actions.

d) Suction weir and return line siphon breaker located 20 feet above the fuel.

63.
Which ONE of the following is indicative of an impending loss of natural circulation flow?

a) RCS delta T at 57°F and increasing.

b) RCS subcooling at 42°F and increasing.

c) Source range detectors counts decreasing.

d) RCS cold leg temperature slowly decreasing.

64.
During 100% power operation, channel III first stage pressure transmitter (1-MS-PT-446) fails low. Which ONE of the following describes how the steam dump system will operate?

a) The steam dumps are armed.

b) The steam dump system will be unaffected during a load reject signal.

c) The steam dumps will modulate closed properly during a Unit trip.

d) All steam dumps open fully.

65.
Which ONE of the following types of radiation monitor detectors is used on the Air Ejector system to allow sensitive response?

a) Geiger-Mueller detector.

b) Gamma Scintillation detector.

c) Uncompensated ion chamber.

d) Beta Scintillation detector.

66.
Following a Unit 1 Hi-Hi CLS, a loss of offsite power occurs. Both Unit 1 emergency busses are reenergized from their associated EDGs. Unit 2 "H" bus reenergizes from its associated EDG. Which ONE of the following actions identifies which Component Cooling water pump **SHOULD** be restored. (Assume CC is crosstied)?

a) 1-CC-P-1A

b) 1-CC-P-1B

c) 1-CC-P-1C

d) 1-CC-P-1D

67.
During performance of 2-ES-0.1, REACTOR TRIP RESPONSE, the Reactor Operator notes the only running AFW pump (2-FW-P-3A) has an extinguished white light with the following parameters indicated:

- 0 amps on the benchboard meter.
- Red light on, green and amber lights out on the 2-FW-P-3A control switch.
- 2-FW-MOV-251A/B/C/D/E/F green lights on red lights off.
- "A", "B", "C" AFW flow indication all indicate "0"
- The light bulb is verified not burned out.

Which ONE of the following identifies the cause of the white light being extinguished?

a) Normal condition.

b) Breaker trip power is lost.

c) Breaker is racked to a position other than "CONNECT".

d) An undervoltage condition exists on the "2H" emergency bus.

68.
A Unit 2 periodic test of #3 EDG is in progress with the diesel in parallel with offsite power. During the test a tornado initiates a loss of the switchyard and lockout of all Reserve Station Transformers. Assuming all systems operate as designed, which ONE of the following identifies the minimum action(s) to restore all four emergency busses.

a) Automatic actions will restore all four emergency busses.

b) #3 EDG must be manually aligned to Unit 1 and the AAC diesel manually aligned to the 2H bus.

c) The AAC diesel must be manually aligned to the 1J bus.

d) Manually realign #3 EDG to 2J bus and manually align the AAC diesel to the 1J bus.

69.
A 20 foot length of piping filled with radioactive fluid has a dose rate of 8 REM/hr at 6 feet. Which ONE of the following APPROXIMATES the dose rate at 4 feet?

a) 18 REM/hr

b) 16 REM/hr

c) 14 REM/hr

d) 12 REM/hr

70.
Which ONE of the following **DOES NOT** result from intake canal level dropping to 23 feet?

a) An "INTAKE CANAL Lo LVL" reactor trip signal is generated.

b) Component Cooling Service Water supply valves receive closed signals.

c) Bearing Cooling Service Water supply valves receive closed signals.

d) Waterbox outlet Motor Operated Valves receive closed signals.

71.
Which ONE of the following determines which Unit's Service air compressor (1-SA-C-1/2-SA-C-1) will run to supply air to the station?

a) Each Compressor's control panel has a "HAND/STBY/OFF" control switch.

b) A local LEAD/LAG control switch designates which compressor runs and which one is in standby.

c) Control switches on each MCR vertical panel allows each Reactor Operator to control his compressor. This allows both compressors to run at once.

d) A stanchion located near each Unit's air dryer contains an "AUTO/STBY/OFF" control switch.

72.
Which ONE of the following conditions would initiate a start of the Diesel Driven Fire Pump?

a) Fire main pressure drops to 99 psig with the local control switch in "AUTO".

b) Breaker 15H8 "NORMAL SUPPLY TO 4160V BUS" opens with the local control switch in any position other than "OFF".

c) Local control switch taken to "TEST".

d) Any ROBERTSHAW system "FIRE" alarm received with the local control switch in "AUTO".

73.
During solid plant conditions with RHR in service, Containment Instrument Air is lost due to inadvertent closing of 1-IA 446. Which ONE of the following identifies the RCS response?

a) Pressure increases, temperature decreases.

b) Pressure increases, temperature increases.

c) Pressure decreases, temperature decreases.

d) Pressure decreases, temperature increases.

74.
Following a reactor trip from 100% power , RCS median Tave is 562°F.
Which ONE of the following describes the position of the steam dump valves?

a) All dumps tripped open.

b) 1-MS-TCV-105A/B, 1-MS-TCV-106A/B tripped open, 1-MS-TCV-107A/B 100% open.

c) All dumps 75% open.

d) 1-MS-TCV-105A/B, 1-MS-TCV-106A/B, 1-MS-TCV-107A/B all 75% open.

75.
During 100% operation, all cooling water to the Containment Air Recirculation Fans , 1-VS-F-1A/B/C is lost. Which ONE of the following is the expected Containment response?

a) Actual partial pressure decreases, indicated partial pressure decreases.

b) Actual partial pressure decreases, indicated partial pressure increases.

c) Actual partial pressure increases, indicated partial pressure increases.

d) Actual partial pressure increases, indicated partial pressure decreases.

76.
During normal 100% operation on Unit 2, 2-VS-F-1B "Containment Air Recirculation Fan" amber light is illuminated.
Local investigations reveal a bell alarm lockout on the breaker.
Which ONE of the following actions is required to remedy the situation?

a) Rotate the control switch to the pull to lock position to reset the 86 device, one restart attempt from the MCR is allowed.

b) Have the operator locally reset the breaker (with SRO concurrence), restart the fan locally at the breaker.

c) Have the operator locally reset the breaker (with SRO concurrence), restart the fan from the MCR, one restart attempt from the MCR is allowed.

d) The Electricians MUST investigate prior to restart.

77.
The following Unit 1 timeline exists:

- At 1400 selected first stage impulse pressure failed low.
- At 1410:30 a common mode failure occurred on all three Main Feed Regulating Valves, causing all valves to go closed.
- At 1411 the Reactor Operator manually tripped the reactor due to all Steam Generator levels decreasing rapidly.
- Current time is 1414, Steam Generator levels are at 34% wide range level.

Which ONE of the following AMSAC system indicators is consistent with given unit conditions?

a) Annunciator F-B-3, AMSAC INITIATED, lit.

b) Annunciator H-D-1, AMSAC ARMED, lit.

c) Annunciator H-E-1, AMSAC TRBL, lit.

d) Bypass Status Light G-1, AMSAC OPERATIONAL BYPASS, lit.

78.
The following plant conditions exist:

- CSD during heatup after a 3 week plant shutdown.
- Train "A" RHR operating.
- Train "B" RHR in standby.
- PRZR level 100% (solid plant).
- RCS pressure 300 psig.
- RCS temperature 180 °F.

Which ONE of the following procedures provides guidance in the event of a excessive RCS leakage while operating in this condition?

a) Excessive RCS Leakage, AP-16.00.

b) Shutdown LOCA, AP-16.01.

c) Loss of Decay Heat Removal, AP-27.00.

d) Reactor Trip or Safety Injection, E-0.

79.
Which ONE of the following identifies an **<u>OFF-NORMAL</u>** parameter concerning Reactor Coolant Pump operation at 100% power?

a) "B" RCP seal injection flow indicates 7.7 gpm.

b) Combined Thermal Barrier CC return header flow indicates 123 gpm.

c) #1 seal D/P indicates 180 psid.

d) #1 seal leakoff flow indicates 3.4 gpm.

80.
Which ONE of the following personnel monitoring devices detects both beta and gamma to determine whole body exposure.

a) Digital Alarming Dosimeter

b) Self Reading Pocket Dosimeter

c) Thermo Luminescent Dosimeter

d) RO-2 monitor with the detector window closed

81.
During a Hi-Hi CLS, a station blackout occurs.
Which ONE of the following identifies the status of the Component Cooling Service Water supply valves (1-SW-MOV-102A/B)?

a) Receives an automatic open signal.

b) Throttled to the 25% open position.

c) Closed with the ability to be manually reopened after 5 minutes.

d) Closed with the ability to automatically reopen after Hi-Hi CLS is reset.

82.
The following plant conditions exist:

- Plant S/U and ramp in power in progress
- Power level is currently 5%
- At this time, compensating voltage fails high on Nuclear Instrument N-35, Intermediate Range detector.

Which ONE of the following will occur?

a) Automatic Reactor trip.

b) Slight decrease on N-35 IR amps.

c) Slight increase on N-35 IR amps.

d) No observable effect on N-35 IR amps.

83.
During 100% power operation, PRT in-leakage is identified as increased. Which ONE of the following identifies a possible source?

a) Reactor vessel head vent valve leakage.

b) RCP seal return relief valve leakage.

c) RCS loop stop valve stem leakoff.

d) RCP #2 seal leakoff.

84.
Which ONE of the following conditions would initiate a turbine runback on Unit 2?

a) 2 out of 4 Power range channels at 103%.

b) 2 out of 3 Overtemperature Delta T channels are within 2% of the reactor trip setpoint.

c) Power Range Channel N-43 fails from 100% to 0% in 10 seconds.

d) 1of 48 Individual Rod Position Indicators reads less than 20 steps.

85.
Following a transition to E-1 the STA reports the following:

- An Orange path on Core cooling
- A Red path on Containment
- A Yellow path on Subcriticality
- A Red path on Integrity

Which ONE of the following identifies the required procedure to be implemented?

a) FR-C.2

b) FR-Z.1

c) FR-S.2

d) FR-P.1

86.
Which ONE of the following items is NOT required to be performed by the relieving Control Room Operator?

a) Review Chemistry Status.

b) Verify Blended Flow.

c) Review Delta Flux Log.

d) Review the Action Statement Log.

87.
Which ONE of the following identifies the purpose of ensuring RCS pressure is less than 2335 psig during performance of FR-S.1, RESPONSE TO NUCLEAR POWER GENERATION/ATWS?

a) Prevents immediate RCS bleed and feed to restore an adequate heat sink.

b) Ensures adequate boration delivery to the core.

c) Prevents unacceptable reactivity void coefficients.

d) Ensures that the RCS remains below all Tech Spec safety limit thresholds.

88.
Health Physics has requested you to start the Iodine Removal fans (1-VS-F-3A/B). Which ONE of the following identifies the location in which these fans can be controlled?

a) Unit 1 MCR Ventilation Panel.

b) Unit 1 Upper Cable vault.

c) Unit 1 Normal Switchgear Room.

d) MCR Common Ventilation Panel (VSP).

89.
During withdrawal of control bank "C" to critical conditions, the STA reports 1/M indicates criticality is projected at 12 steps on "D" bank. Which ONE of the following actions is required of the team?

a) Pull rods another 45 steps to get another set of independent data.

b) Trip the reactor and emergency borate.

c) Open the reactor trip breakers and re-evaluate 1-OP-RX-004, THE CALCULATION OF ESTIMATED CRITICAL CONDITIONS.

d) Open the reactor trip breakers, rackout the MG set input breakers, and close 1-CH-FCV-1114A (PG WATER TO BLENDER).

90.
Which ONE of the following describes why both 1st point feedwater heaters must be taken out of service at the same time?

a) Uneven feedwater temperatures will produce radial flux tilt conditions.

b) Unequal feed temperatures will create an Overtemperature delta T trip condition.

c) The unequal steaming rates on the opposite ends of the steam header could initiate RCS loop differential temperature alarms.

d) Unequal feed temperatures will create an Overpower T trip condition.

91.
Which ONE of the following Auxiliary Feedwater system design features limits the probability of AFW pump runout?

a) An orifice installed in each pump's discharge.

b) Cavitating flow venturis in each SG supply line.

c) Tech Spec minimum levels for ECST levels.

d) Tapered pump volute.

92.
During a Site Area Emergency, the on-shift fire team members will do which ONE of the following?

a) Report to the OSC and respond as directed by the OSC director.

b) Report to the Annex and perform normal duties unless called out for a fire.

c) Report to the TSC and respond as directed by the Station Emergency Manager.

d) Report to the MCR and respond as directed by the Shift Supervisor.

93.
During a normal shutdown on Unit 1, the Unit RO is unable to close the "A" MSTV using the benchboard switch. She attempts the "APP R EMERG CLOSURE" control switch on the MCR vertical panel, but this also fails.
Which ONE of the following identifies another location at which closure can be attempted?

a) Unit 1 AUX SHUTDOWN PANEL in U-1 ESGR.

b) Unit 1 "APP R" panel in U-1 ESGR.

c) Unit 2 "APP R" panel in U-1 ESGR.

d) Hot-short panel in U-1 lower cable vault.

94.
During a Large Break LOCA, Hi-Hi CLS fails to automatically and manually actuate.

Which ONE of the following actions should be performed first to meet the <u>design function</u> of the Hi-Hi CLS system?

a) Secure all three Reactor Coolant Pumps.

b) Start and align the Containment Spray pumps.

c) Secure all three Containment Air Recirculation Fans.

d) Align Service Water to the Recirc Spray Heat Exchangers (RSHX).

95.
Which ONE of the following identifies how a LOSS of "A" DC bus affects the operation of the associated reactor trip breaker?

a) The shunt coil will deenergize.

b) The shunt coil will energize.

c) The UV coil will deenergize.

d) The UV coil will energize.

96.
Which ONE of the following identifies a difference between Unit 1 and Unit 2?

a) Unit One RCP seal return flow transmitters utilize a magnetic flow transmitter, Unit 2 RCP seal return utilizes a rotometer.

b) Common Radiation Monitors can only be powered from Unit 1.

c) MER #5 chillers can only supply Unit 2 Air Handling Units.

d) 1-VS-F-58A backup power supply is Unit 2 "H", 1-VS-F-58B backup power supply is Unit 1 "J".

97.
During a declared "GENERAL EMERGENCY" you volunteer to perform an action to minimize equipment damage. While briefing with the Radiological Assessment Director, you are informed you will exceed your normal exposure limits.
Which ONE of the following individuals can approve use of Emergency Exposure limits?

a) Radiological Assessment Director.

b) Accident Unit SRO.

c) Emergency Operations Director.

d) Station Emergency Manager.

98.
Which ONE of the following events would initiate an AAC diesel automatic start?

a) A lightening strike causes a loss of switchyard bus #5.

b) A system voltage spike initiates a simultaneous lockout of the "A" and "B" Reserve Station Service Transformers.

c) With "A" Reserve Station Service Transformer out of service for relay replacement, a switching error initiates a loss of switchyard bus #6.

d) With Unit 2 at Hot Shut Down, a sequencing error during a Unit 1 reactor trip from 100% power occurs. Breakers 15C1(Normal supply from RSST) and 15C2 (Normal supply from SST) are closed at the same time, leading to an overcurrent trip of breaker 15F1("F" transfer bus supply).

99.
Which set of procedure's immediate actions are required to be performed in order?

a) FR-S.1, ECA-0.0

b) ECA-0.0, E-0

c) E-0, FR-S.1

d) ECA-0.0, E-0, FR-S.1

100.
FCA-16.00, LOCAL OPERATION OF AIR OPERATED VALVES, provides guidance to operate air operated valves locally. Using the provided (on the next page) "ATTACHMENT 2" , which ONE of the following identifies the purpose of Quick disconnects 4 and 5.

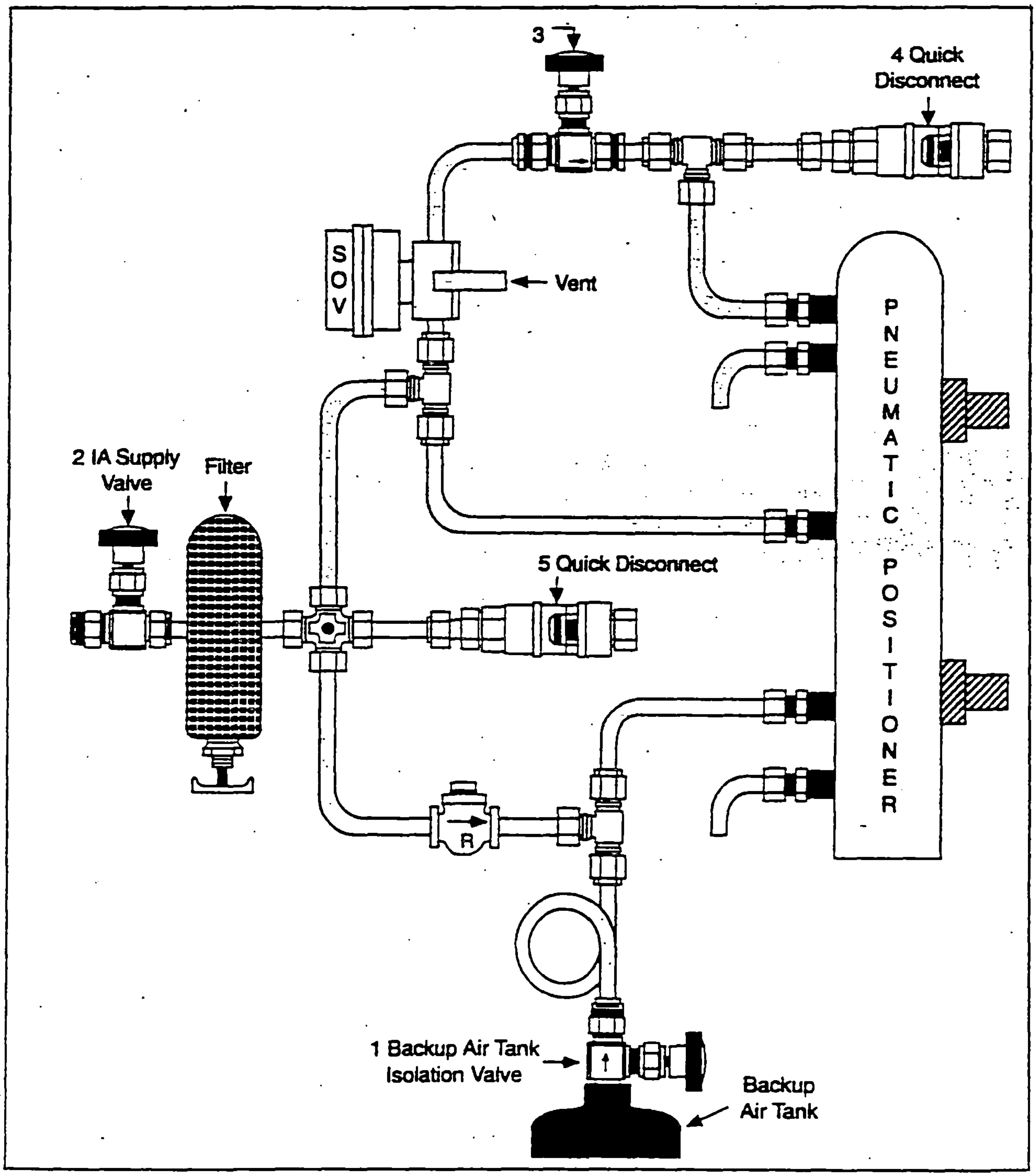
a) 4 positions the pneumatic positioner to the open position, 5 strokes the valve.

b) 4 closes the valve, 5 opens the valve.

c) 4 strokes the valve, 5 positions the pneumatic positioner to the open position.

d) 4 opens the valve, 5 closes the valve.

NUMBER 0-FCA-16.00	ATTACHMENT TITLE	REVISION 2
ATTACHMENT 2	LOCAL OPERATION OF CC TVs	PAGE 1 of 1



Graphics No. CB31B

U. S. Nuclear Regulatory Commission
Site-Specific
Written Examination

Applicant Information

Name:	Region : II
Date: 4/8/99	Facility/Unit: Surry
License Level: SRO	Reactor Type: W
Start Time: 0900	Finish Time:

Instructions

Use the answer sheets provided to document your answers. Staple this cover Sheet on top of the answer sheets. The passing grade requires a final Grade of at least 80.00 percent. Examination papers will be collected four Hours after the examination starts.

Applicant Certification

All work done on this examination is my own. I have neither given nor Received aid.

____
Applicant's Signature

Results

Examination Value	100	Points
Applicant's Score	______	Points
Applicant's Grade	______	Percent

SRO ANSWER KEY

001 B	026 B	051 D	076 D
002 C	027 D	052 D	077 B
003 C	028 B	053 A	078 B
004 A	029 A	054 C	079 D
005 D	030 C	055 A	080 C
006 A	031 D	056 D	081 D
007 A	032 B	057 C	082 C
008 B	033 A	058 D	083 B
009 B	034 B	059 C	084 B
010 D	035 B	060 B	085 C
011 A	036 B	061 A	086 B
012 B	037 A	062 D	087 B
013 B	038 B	063 A	088 B
014 D	039 C	064 C	089 B
015 C	040 C	065 A	090 D
016 C	041 C	066 C	091 C
017 D	042 A	067 C	092 C
018 D	043 B	068 C	093 C
019 D	044 D	069 D	094 A
020 B	045 C	070 A	095 C
021 D	046 D	071 B	096 A
022 A	047 D	072 C	097 D
023 D	048 A	073 A	098 D
024 C	049 B	074 B	099 C
025 D	050 D	075 D	100 B

RO/SRO License Class 1999 Initial License Exam Key

1.
During troubleshooting on the Rod Control System at 100% power, a Power Cabinet 2BD Non-Urgent alarm was received. The Test Director (I&C Supervisor) directs the "Internal Alarm Reset" push-button to be depressed, in accordance with a SNSOC approved test procedure. The Unit Reactor Operator mistakenly depresses the "Startup Reset" push-button. Which ONE of the following automatic responses is expected?

a) The Reactor Trip Breakers will open.

b) All control rod bank low and low-low annunciators will illuminate.

c) Group B and D group step counters reset to 0 steps, A and C groups remain at the all rods out position.

d) All IRPI indicators reset to 0 and all rod bottom lights illuminate (actual rod position does not change.

2.
Which ONE of the following would initiate an automatic Unit 2 Reactor trip?

a) An underfrequency condition of 49 hertz on the Unit 2 "B" and "C" Station Service busses with Unit 2 stable at 2% power.

b) An overcurrent trip of the Unit 2 "B" RCP with reactor power at 25% power.

c) With reactor power at 45%, the Reactor Operator manually secures the Unit 2 "A" RCP due to high shaft vibrations.

d) The Unit 1 Reactor Operator accidentally opens breaker 15D1, RES STA SERVICE XFER SUP BKR, with Unit 2 at 100% power.

3.
Which ONE of the following events would require AP-39.00, NATURAL CIRCULATION OF RCS, to be initiated to establish/verify Natural Circulation Flow?

a) A major steam line break where at least one Steam Generator has pressure which is 350 psig less than RCS pressure.

b) A Small Break LOCA in which RCS subcooling is 28 °F with containment pressure at 24 psia.

c) Following a reactor trip, all Station Service busses fail to automatically swapover to Reserve Station Service.

d) While at 200°F/300 psig, the running RHR pump trips and the standby pump cannot be started.

4.
With a saturated mixed bed ion exchanger in service, 1-CC-TCV-103, CC return from NRHX, drifts 30% in the closed direction. Letdown temperature is now 139°F. Which ONE of the following plant responses is expected?

a) Rods step out slowly.

b) Rods step in slowly.

c) 1-CH-TCV-1143 (Letdown IX Temperature Divert Valve) automatically diverts letdown flow around the Ion Exchangers.

d) 1-CH-PCV-1145 (Letdown Pressure Control Valve) throttles closed to maintain letdown pressure.

5.
The following conditions exist:

- Unit 1 is at 100% power.
- 1-CC-TV-120B ("B" RCP Thermal barrier return) and its manual isolation valve, 1-CC-57, are closed to isolate a small thermal barrier leak on the "B" RCP.
- All other systems are operating normally.

Which ONE of the following events would allow continued power operation of Unit 1?

a) 1-CC-TV-105A (RCP "A" CLR CC RTN TV) closes and will not reopen.

b) 1-CH-HCV-1186 (RCP Seal injection flow) closes and will not reopen.

c) Actual seal leakoff flow on "C" RCP increases off scale high.

d) 1-CH-MOV-1381 (RCP Seal Return) closes and will not reopen.

6.
Ten Minutes ago a Steam Dump valve failed partially open at 2% power.
The Operating team has corrected the problem.
Reactor power reached a maximum of 5.7% Power Range indication, currently stable at 2%.
RCS pressure reduced to a minimum of 2175 psig and is recovering.
RCS temperature reduced to a minimum of 529°F and is recovering.

Which ONE of the following identifies the **Tech Spec LCO** that has been exceeded?

a) Section 3.12 DNB Low Pressure limit.

b) Section 2.1 Low Pressure Safety limit.

c) Section 3.1 Minimum Temperature for Criticality limit.

d) Section 2.3.2 Overpower Delta T limit.

7.
During 20% steady state reactor power operation, "A" Steam Generator PORV fails full open. Which ONE of the following describes the operation of the control rods to this event?

a) The control rods would move out due to a Tave/Tref mismatch.

b) The control rods would not move.

c) Control rods would not be affected, only power would increase.

d) The rods would trip into the core due to high steam line flow SI being generated.

8.
Following a major steam line break on the Unit 1 "B" Main Steam line, the Reactor Operator is directed to control RCS temperature following "B" Steam Generator dryout.
Which ONE of the following identifies the basis for performing this action?

a) Prevent Pressurizer Relief Tank (PRT) rupture leading to possible Containment integrity concerns.

b) Prevent RCS repressurization, leading to possible Pressurized Thermal Shock concerns.

c) Minimize the temperature perturbation on the RCS which could lead to possible "B" SG tube failure.

d) Minimize RCS heatup which could cause a loss of the subcooling margin necessary to secure SI.

9.
Which ONE of the following indications, if sustained for 5 minutes, on 1-CN-PR-101A and 101B, CONDENSER VACUUM, would require an immediate reactor trip?

a) 25" Hg at 100% power.

b) 25.5" Hg with turbine power at 20%.

c) 26" Hg with turbine power at 35%.

d) 28" Hg and decreasing at a rate of 0.5" every 2 minutes.

10.
The following conditions exist:

- Unit 2 is at Hot Shutdown preparing for a Unit startup.
- SG narrow range levels are 35% in all Steam Generators.
- All decay heat is being removed via SG blowdown.
- SGs are being fed from the "A" Main Feed Pump through the Main Feed Bypass HCVs.
- Main Steam Trip valves are open.

A Station Blackout occurs (Emergency Busses are reenergized from their associated EDGs) simultaneous with a loss of all Unit 2 Instrument air.

Which ONE of the following identifies an expected response during the initial phase of the transient (Assume no Operator actions are taken)?

a) Steam Generator Blowdown is diverted to the river.

b) Pressurizer level decreases.

c) Main Feed Regulating Valve demand increases.

d) Auxiliary Building Central Ventilation realigns to filtered exhaust.

11.
With Unit 1 at 100% power and all systems functioning normally, 1-CH-LT-1112 (VCT level) fails low due to Vital Bus 1-IIIA breaker 26 tripping open. Which ONE of the following is the expected plant response for this transient?

a) No apparent response other than 1-CH-LI-1112 failed low.

b) Charging pump suction MOVs swap to the RWST from the VCT (1-CH-MOV-1115B/D open, 1-CH-MOV-1115C/E close).

c) Letdown flow diverts to the Primary Drain Tank (1-CH-LCV-1115A diverts).

d) Automatic VCT makeup actuates.

12.
A fire has been reported in the Unit 1 Relay Room.

Which ONE of the following actions should be taken to extinguish the fire?

a) Rotate the Unit 1 selector switch on the halon control panel. (located in the Unit 1 Turbine basement near the Unit 1 blowdown coolers).

b) Depress the Unit 1 Halon Push-button on the Unit 2 side of the Control Room (Behind the Vertical Panel) .

c) Actuate the LP CO_2 pull station located in the Unit 1 ESGR (At the entrance to the Unit 1 cable vault).

d) Actuate the Halon Pull station located in the Unit 1 Turbine Building basement (at the entrance to the Unit 2 ESGR).

13.
During a Limiting Fire in the Main Control Room, swapover for the Unit 2 Station Service busses failed. All Unit 2 Station Service busses are deenergized. All emergency busses are energized by off-site power. Which ONE of the following identifies the heater capacity available at the Unit 2 Auxiliary Shutdown Panel?

a) 400 KW

b) 450 KW

c) 500 KW

d) 600 KW

14.
The failure of a "B" train Hi CLS (phase II) relay has caused a partial initiation of "B" train of Hi CLS. Annunciators AF3, SI INITIATED TRAIN A, and AF4, SI INITIATED TRAIN B, are ***NOT*** lit. Also, ***NO*** automatic actions associated with SI have occurred. The Unit remains at 100% power.

Which ONE of the following manual actions will need to be performed to recover from this event

a) Secure the 1-VS-F-58B and realign the auxiliary ventilation system to normal status.

b) Align the Containment Instrument Air Compressors to an outside alignment.

c) Secure the Hydrogen Analyzer Heat Tracing.

d) Realign the Containment Particulate and Gaseous Radiation Monitor (1-RMS-159/160) to the Containment.

15.
During 75% reactor power operation, which ONE of the following events would place the unit closer to a Departure from Nucleate Boiling (DNB) condition? (Assume all systems and components are operable and in auto.)

a) Group "C" pressurizer heater output fails to high output.

b) Selected 1st stage impulse pressure fails low.

c) Median Tave fails low.

d) 1-RC-PT-1445, PRESSURIZER PRESSURE CONTROL CHANNEL II, fails low.

16.
The Operating team entered FR-C.1, RESPONSE TO INADEQUATE CORE COOLING. The team failed in all attempts to establish High Head Flow. Core Exit Thermocouples are 820°F and increasing slowly.

Which ONE of the following methods is required to respond to the core cooling challenge?

a) Open available Pressurizer PORVs to lower pressure to the SI accumulator and LHSI injection pressures.

b) Open the Pressurizer and Head vent SOVs to allow venting of any hard bubble and allow natural circulation to progress.

c) Depressurize all intact Steam Generators to 150 psig to allow RCS depressurization to the SI accumulator and LHSI injection pressures.

d) Enter the Severe Accident Mitigation Guidelines.

17.
Following a reactor trip and safety injection, indications of extensive failed fuel exist. Which ONE of the following criteria would require the Main Control Room Staff to begin using conservative setpoints due to the potential unreliability of installed instrumentation?

a) Specific activity >10 microcuries/cc dose equivalent I-131, as indicated on any RCS sample.

b) SI accumulator level indicators deviate from pre-trip levels.

c) RCS subcooling indicates 26°F on ICCM.

d) 1.3E5 R/hr as indicated on the Containment High Range Monitor.

18.
With Unit 1 operating at 100% power (normal operating pressure and temperature), the median Tave output fails to 570°F. If "D" bank started at 210 steps, which ONE of the following approximates how many ***seconds*** would elapse to reach the all out rod position on "D" bank indications?
(Assume rod motion does not affect temperature).

a) 22

b) 29

c) 112

d) 143

19.
During recovery of a dropped rod, an urgent failure alarm is received immediately after initiating rod motion on the dropped rod. Which ONE of the following identifies the cause of this alarm?

a) Non-urgent failure condition coincident with rod motion demand.

b) Deviating condition is generated in the Bank Overlap Unit.

c) Disagreement between Individual Rod Position Indicators and Group Step Counters.

d) The lift coils of the remaining rods in the affected bank are deenergized.

20.
During normal 100% operation the RO acknowledges annunciator C-C-8, PRZR HI LEVEL HTRS ON.
RCS pressure is 2203 psig and decreasing slowly.
VCT level trend is decreasing slowly.
Which ONE of the following diagnoses this off normal trend?

a) Pressurizer heaters have failed to minimum output.

b) Pressurizer level detector reference leg has separated from the Pressurizer.

c) 1-CH-FCV-1122, CHG FLOW CONT, failed open.

d) 1-CH-HCV-1200A, LETDOWN ORIFICE ISOL, failed closed.

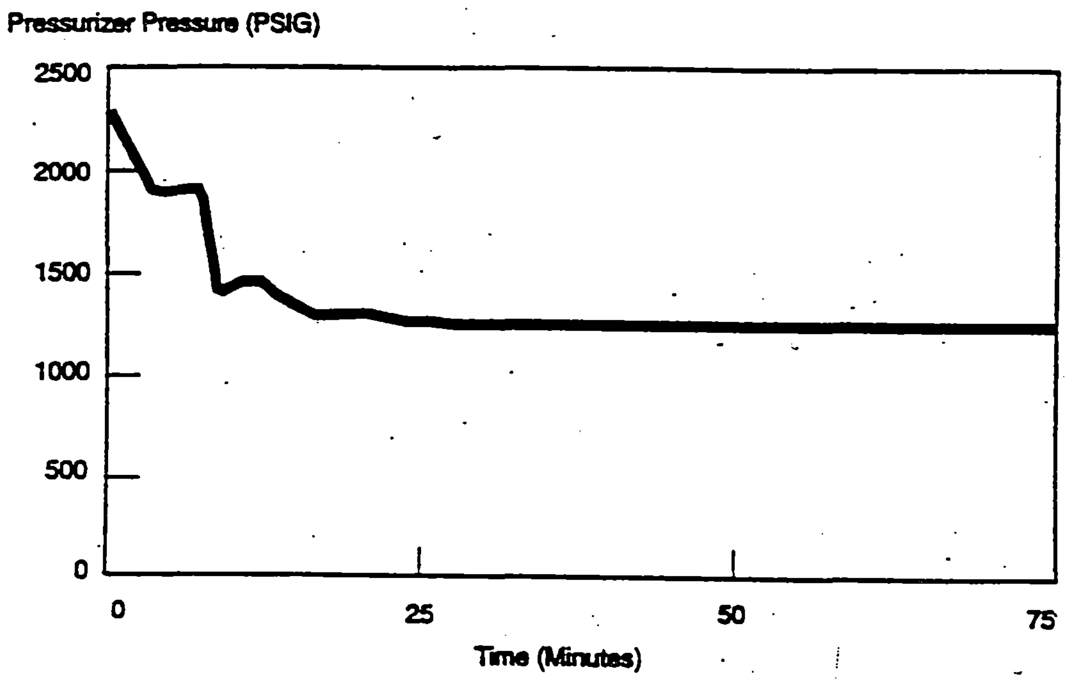
21.
The graph included below gives typical RCS pressure response following a Small Break LOCA. Which ONE of the following identifies the cause of stable RCS pressure from time 25 minutes to 75 minutes?

a) All charging pumps have reached their low pressure auto start setpoint.

b) RCS level has decreased out of the Pressurizer and surge leg. This pressure is indicative of Reactor Vessel head pressure.

c) RCP trip criteria was met and the change in slope is indicative of static RCS pressure.

d) SI flow has matched break flow.



22.
During a Large Break LOCA, a low pressurizer pressure SI signal initiates an automatic trip of the unit's Main Feed Pumps. Which ONE of the following identifies the purpose of this trip?

a) Feedwater isolation is designed for a main steam line break accident and does not provide any substantial benefits during a LBLOCA.

b) Minimizes the thermal stresses associated with rapid RCS depressurization by minimizing the feed water injection.

c) Required to allow AFW pumps to deliver cooler water to the SG. This minimizes the temperature differential across the SG tubes.

d) Allows the RSSTs to maintain a constant voltage profile during the accident with additional required loads.

23.
The following Unit 2 conditions exist:

- The unit has sustained a Small Break Loss Of Coolant Accident (SBLOCA).
- The team is in 2-E-1, LOSS OF REACTOR OR SECONDARY COOLANT.
- Pressurizer pressure is 1405 psig.
- Pressurizer level is off-scale low.
- RCS subcooling is 29°F.
- Containment Pressure is 13 psia.

Which ONE of the following identifies the procedure(s) which will provide long term guidance to stabilize the plant given the above RCS conditions?

a) 2-GOP-2.4, Unit Cooldown, HSD to 351°F.

b) 2-ES-0.2, NATURAL CIRCULATION COOLDOWN.

c) 2-ES-0.3, NATURAL CIRCULATION COOLDOWN WITH STEAM VOID IN RX VESSEL.

d) 2-ES-1.2, POST LOCA COOLDOWN AND DEPRESSURIZATION.

24.
An intermediate break LOCA has occurred on Unit 1, all four RMT channel trip annunciators are illuminated and automatic Recirc Mode Transfer (RMT) is in progress. The amber RMT light has just illuminated. Which ONE of the following identifies valves which are expected to be cycling during this period of RMT?

a) 1-SI-MOV-1885 A/B/C/D (LHSI recirculation isolation valves).

b) 1-SI-MOV-1863 A/B (LHSI discharge to HHSI suction).

c) 1-SI-MOV-1862 A/B (LHSI suction from the RWST).

d) 1-SI-MOV-1864 A/B (LHSI discharge to cold legs).

25.
The operating team is responding to a Unit 1 "A" main steam line break inside containment. The following conditions presently exist:

- The operating team has transitioned from 1-E-2, FAULTED STEAM GENERATOR ISOLATION, to 1-E-1, LOSS OF REACTOR OR SECONDARY COOLANT.
- "A" SG is dry with pressure <100 psig.
- "B" SG narrow range level is 20% and increasing , with pressure stable at 950 psig.
- "C" SG narrow range level is 5% and increasing , with pressure stable at 950 psig.
- Pressurizer level is 15% and increasing.
- Containment pressure is 18 psia and decreasing.
- RCS subcooling indicates stable at 120°F.
- RCS pressure is 2150 psig and increasing slowly.

Which ONE of the following conditions will complete the transition criteria from 1-E-1 to 1-ES-1.1, SI TERMINATION?

a) "C" SG level increases to 14%.

b) "B" SG level increases to greater than 25%.

c) RCS pressure increases to 2215 psig.

d) Pressurizer level increases to 27%.

26.
A Process Vent "HIGH' alarm was initiated from the 1-GW-RI-101, PRCS VENT PART, Radiation Monitor.
Which ONE of the following **AUTOMATIC** action(s) must be verified?

a) HCV-GW-106, AERATED VENT ISOLATION, closed.

b) FCV-GW-101, DECAY TK BLEED ISOL, closed.

c) 1-VS-MOV-100A-D, CTMT PURGE ISOLATION, closed.

d) 1-CV-P-1A/1B, CTMT VACUUM PUMPS, off.

27.
The following conditions exist while RHR is in service on Unit 1:
- Hot leg temperature is 232°F.
- RCS pressure is 310 psig and decreasing rapidly.
- Containment sump level is increasing.
- Containment pressure is 9.8 psia and increasing slowly.
- Pressurizer level is 19% and decreasing.
- Charging flow is at maximum.

Which ONE of the following describes the reason that the Safety Injection pushbuttons are **NOT** depressed in response to these conditions?

a) Results in a no-load condition for the Emergency Diesel Generators since ESF components are in Pull to Lock.

b) SI Accumulator isolation MOVs will automatically open, leading to a possible OPMS actuation.

c) PTS concern when HHSI flow is established to the cold legs.

d) Phase I Containment isolation, leading to CC loss in the Containment.

28.
Which ONE of the following indication(s) in conjunction with Power Range Nuclear Instruments provide **ALLOWABLE** assurance that a Reactor Trip has occurred?

a) Annunciator E-A-8 "Reactor Tripped by Turbine Trip" backlit red.

b) Reactor Trip Breaker indicating lights illuminated green.

c) All Individual Rod Position Indicators (IRPI) at 0 steps.

d) Annunciator F-B-3 "AMSAC Initiated" illuminated white.

29.
During 70% power operation Intermediate Range Nuclear Instrument N-36 fails low. Which ONE of the following actions is required to respond to this event?

a) Place the "Level Trip" switch in "Bypass".

b) Reduce power to less than 10%.

c) Perform E-0, REACTOR TRIP OR SAFETY INJECTION, due to an automatic reactor trip.

d) Remove the Instrument power fuses.

30.
The following Unit 2 conditions exist:

- 100% operation.
- KAMAN annunciator A-6 "UNIT 2 MN STEAM ABC RAD MON ALERT/HI" illuminated.
- Air ejector radiation monitor reading 00.0E0
- Charging line flow is 142 gpm being controlled manually.
- Letdown flow is gpm 100 gpm.
- Combined seal return flow is 8 gpm.
- Total seal injection flow is 27 gpm.
- RCS Tave is stable
- Pressurizer level is stable at 55%
- The team is initiating 2-AP-16.00.

Which ONE of the following describes the expected procedure transition(s) for the given conditions?

a) Go to 2-AP-24.00, MINOR SG TUBE LEAK.

b) Go to 2-AP-24.01, LARGE STEAM GENERATOR TUBE LEAK.

c) Go to 2-E-0, REACTOR TRIP OR SAFETY INJECTION, and initiate 2-AP-24.01, LARGE STEAM GENERATOR TUBE LEAK.

d) Go to 2-E-0, REACTOR TRIP OR SAFETY INJECTION, with eventual transition to 2-E-3, STEAM GENERATOR TUBE RUPTURE.

31.
Which ONE of the following identifies the power level above which a manual reactor trip is required for a loss of one out of two running Main Feed pumps?

a) No trip required unless the unit approaches an automatic trip setpoint.

b) Power greater than 65% power.

c) Power greater than 75% power.

d) Power greater than 85% power.

32.
Unit 1 is at 400°F with a stable 30°F/hr heatup in progress for the last 4 hours. The following conditions also exist:

- All Main Steam Trip Valves and bypass valves are closed.
- All feed to all steam generators is secured.
- For the last 30 minutes "A" SG narrow range levels have increased from 35% to 55%, prior to this levels were stable at 35%.
- "B" and "C" have been steady at 35% throughout the heatup.
- No other abnormal indications exist.

Which ONE of the following events would produce the indicated parameters.

a) Small steam break on "A" Steam Generator.

b) "A" Steam Generator Tube Rupture.

c) Small feed line break on "B" and "C" Steam Generators.

d) Variable leg leak on "A" Steam Generator.

33.
Which ONE of the following lists the proper order of cooling restoration to Unit 1 (by priority) directed by 1-FR-H.1, RESPONSE TO LOSS OF SECONDARY HEAT SINK? Assume all Steam Generator Wide Range levels are 40%.

a) Unit 1 AFW, Unit 2 AFW, MFW, Condensate, Bleed and Feed.

b) MFW, Unit 1 AFW, Unit 2 AFW, Condensate, Bleed and Feed.

c) Feed and bleed, Unit 1 AFW, Unit 2 AFW.

d) Unit 2 AFW, Unit 1 AFW, MFW, Condensate, Bleed and Feed.

34.
During a Steam Generator Tube Rupture event, Health Physics determines Unit 1 Turbine Building sump activity levels are in excess of permissible release values and cannot be discharged to the river. Which ONE of the following provides the required course of action per E-3, STEAM GENERATOR TUBE RUPTURE?

a) Leave the water in the Turbine Building sump to allow radioactive decay.

b) Pump the Turbine Building Sump to the CCHX trough.

c) Use condensate to dilute the sump to acceptable release values.

d) Pump the Turbine Building sump to the Unit 1 Waste Neutralization sump.

35.
Which ONE of the following immediate actions is required upon notification from the Containment that a fuel handling accident has occurred, coinciding with this report are 1-RM-RI-162 "Manipulator Crane" ALERT and HIGH alarms?

a) Secure running 1-VS-F-58 fan(s) to isolate possible release paths.

b) Close MCR Motor Operated Dampers (1-VS-MOD-103C/D) and then manually initiate an air bottle dump.

c) Place the fuel in the nearest safe location and then evacuate the Containment.

d) Mobilize the Containment Closure Team, to set Refueling Containment Integrity.

36.
At 100% power operation, the lower selected channel of pressurizer level fails low. Which ONE of the following identifies the unit response with no operator action?

a) Charging pump suction (1-CH-MOV-1115B/D open, 1-CH-MOV-1115C/E close) swaps to the RWST.

b) Charging flow decreases to the minimum setpoint (25 gpm).

c) Pressurizer heater output goes to maximum.

d) No effect on system operation.

37.
Unit 2 has sustained a Hi Hi CLS due to a Large Break LOCA. One minute following the Hi-Hi initiation signal the "B" Reserve Station Service Transformer locked out. The corresponding EDG auto started and loaded on the emergency bus as required.
Assume it is now 70 seconds after the EDG loaded on the emergency bus.
All the listed loads are not running. Which ONE of the following loads has **not sequenced properly** onto the **affected** emergency bus?

a) "B" 58 fan (while on the normal supply).

b) "E" group of pressurizer heaters.

c) "A" Motor Driven AFW pump.

d) "B" ISRS pump.

38.
Assuming that station instrument air system is in a normal system alignment with the Unit 1 Service Air Compressor (1-SA-C-1) tagged out, which one of the following identifies the automatic system response to a trip of the Unit 2 Service Air Compressor (2-SA-C-1).

a) 1-CP-FCV-101, CP AIR SUPPLY TO TURBINE BUILDING, automatically opens to supply air to both units instrument air systems.

b) Both Units Instrument Air compressors auto start (1/2-IA-C-1) to supply air to the respective Units Instrument Air system.

c) The first Unit's Instrument Air compressor (1-IA-C-1/2-IA-C-1) to start will load and supply both Units Instrument Air systems.

d) The Sullair Diesel automatically starts to supply both Units Instrument and Service air systems.

39.
Unit 1 is operating at 100% power with rod control in manual due to hunting. The RO verifies that rods start to move inward as seen on the rod direction light and a decrease in Group Step Counters.

Which ONE of the following identifies the required Reactor Operator response to this transient?

a) Verify a proper temperature mismatch exists for the given rod speed.

b) Place rods in automatic and verify rod motion stopped.

c) Trip Unit 1 reactor.

d) Depress the "Startup Reset" pushbutton.

40.
The following conditions exist on Unit 2

- A large Steam line break has occurred inside the Unit 2 Containment.
- The team is performing 2-E-0, REACTOR TRIP OR SAFETY INJECTION, actions.
- "B" SG pressure is 100 psig and decreasing slowly.
- RCS pressure is 1725 psig and slowly recovering.
- RCS temperature is 501°F and decreasing slowly.
- Containment pressure is 26 psia and increasing slowly.
- Total Safety Injection flow is 420 gpm.

Which ONE of the following identifies the desired status of the Unit 2 Reactor Coolant Pumps?

a) Leave running to ensure even mixing of the injected RWST water.

b) Secure due to the status of RCS subcooling.

c) Secure due to the status of RCP motor cooling.

d) Leave running because RCS conditions are not conducive for Natural Circulation flow.

41.
Assume a typical Reactor Startup is in progress. 1/M plot data is as follows:

- At 98 steps "C" Control Bank (CB) Source Range (SR) counts were 800 cps.
- At 143 steps "C" CB, SR counts were 1300 cps.
- At 188 steps "C" CB, SR counts were 2500 cps.

Using the attached ICCR plot, which ONE of the following gives the current projected "D" bank Critical Rod Height?

a) 85 steps.

b) 98 steps.

c) 110 steps.

d) 135 steps.

ATTACHMENT 2

(Page 1 of 1)

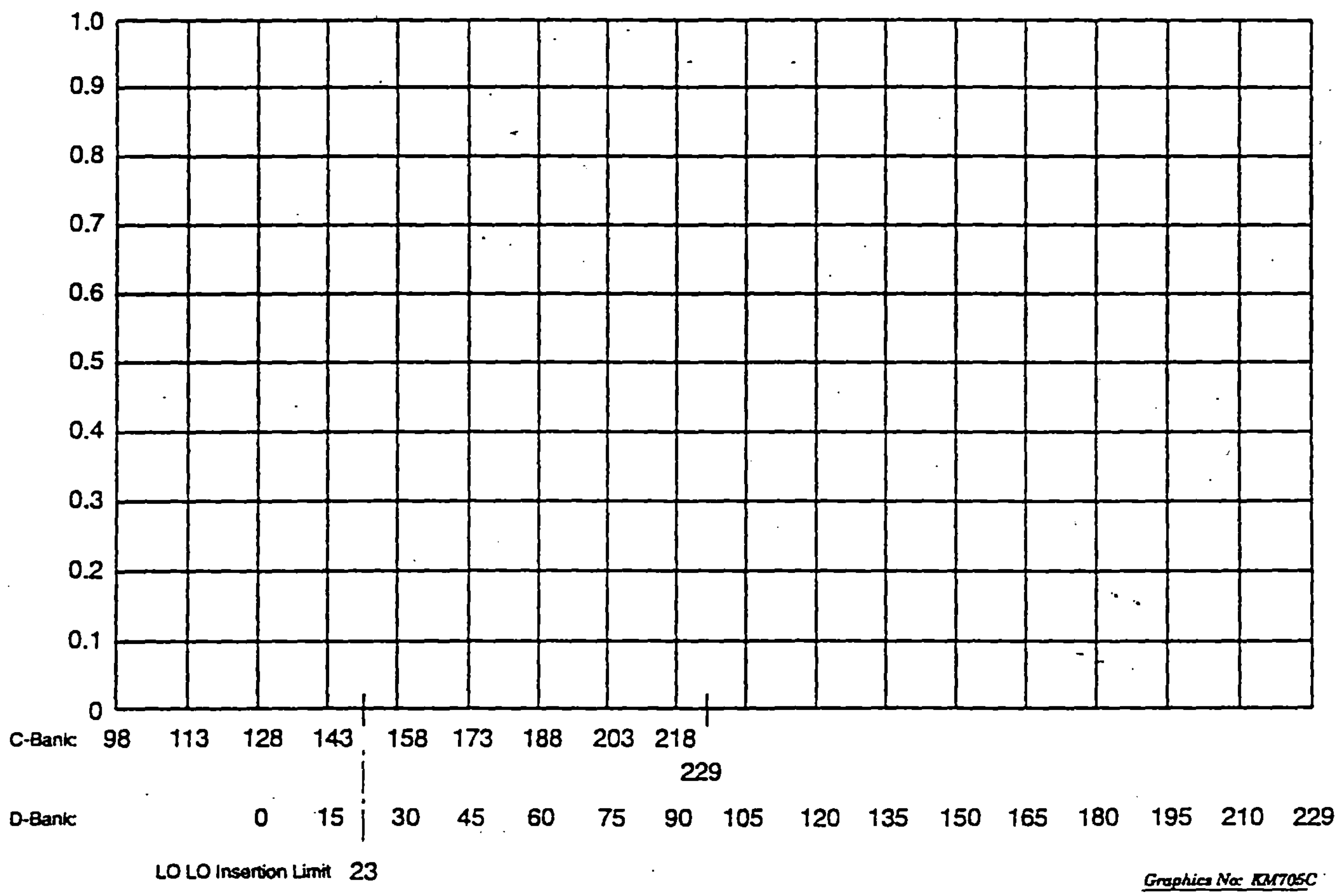
INVERSE COUNT RATE RATIO PLOT (PULL TO CRITICALITY)

Start Up No. ________
Ch: SR ________
IR ________

ECP: Bank _____ at _____ Steps
-500 PCM _____ Steps
+500 PCM _____ Steps

ACP: Bank _____ at _____ Steps
CB ________

Time	Steps		Power Level		1/M	Time	Steps		Power Level		1/M
	C	D	SR	IR			C	D	SR	IR	



INVERSE COUNT RATE RATIO PLOT

rformed by: ____________________ ________ ____________________ ________
Signature Initial Print Date

42.
Which ONE of the following is **NOT** a function provided by the CVCS system?

a) Purify Spent Fuel Pit water.

b) Purify Reactor Cavity water.

c) RCS solid plant pressure control.

d) Inventory control during mid-loop operations.

43.
Which ONE of the following conditions results in automatic initiation of Hi-Hi CLS?

a) Loss of Vital bus IV and IVA.

b) Containment Pressure protection channels 1 and 4 reach 24 psia, with channel 3 in trip.

c) Actual containment pressure increases to 18 psia.

d) Loss of Power to the "A" and "B" train Hi-Hi CLS cabinets.

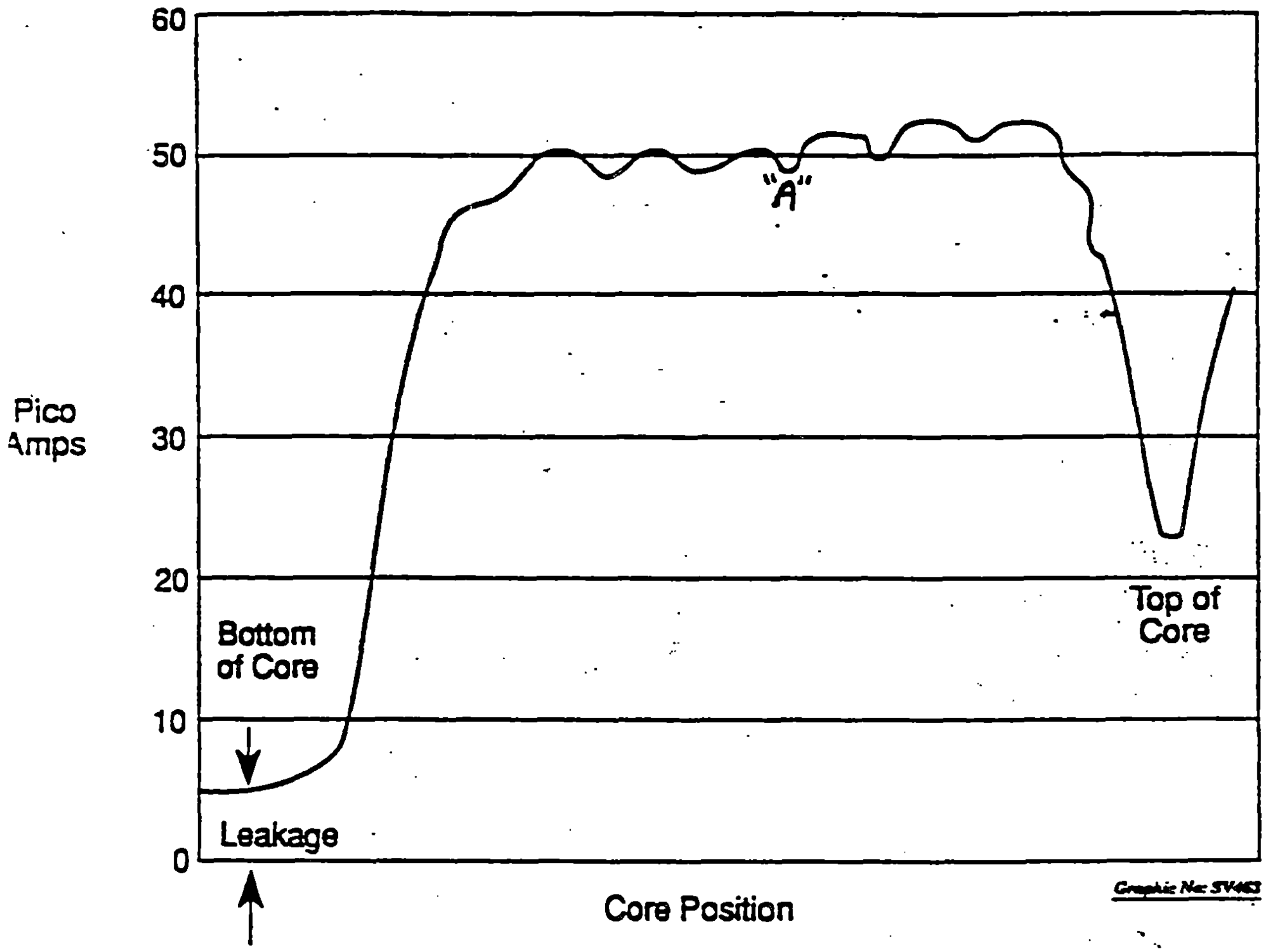
44.
Using the attached Flux Map for reference, which ONE of the following identifies the cause of the large perturbation at point "A"?

a) Region of high fuel enrichment.

b) Region of low fuel enrichment.

c) Region of partial core boiling.

d) Grid strap location.



GAMMA RESPONSE TRACE

45.
With the Unit at 100% power, a control rod is suspected of partially dropping into the core. Currently the control rod indicates approximately 40 steps by Individual Rod Position Indication (IRPI). This control rod is located adjacent to Power Range instrument N-44. Which ONE of the following selections support diagnosis of an actual mispositioned rod?

a) The N-44 benchboard meter reads higher than the other three, due to lower localized temperatures in the RCS.

b) All Power Range Nuclear instruments will read slightly lower than pre-event values due to negative reactivity insertion.

c) Annunciator G-H-1, NIS DROPPED ROD FLUX DECREASE > 5% PER 2 SEC, illuminated due to high localized power reduction.

d) N-44 delta flux indicates more positive due to flux shift to a higher enriched region of the core.

46.
Which ONE of the following describes the power supply arrangement to the Containment Air Recirc Fans?

a) All emergency bus powered.

b) All Station Service bus powered.

c) One emergency bus powered, two station service bus powered.

d) One station service bus powered, two emergency bus powered.

47.
During normal 100% power operations on Unit 1, all three Steam Generator Feed Flow/Steam Flow mismatch alarms illuminate. The SRO immediately recognizes both Main Feed Pumps are running and that 1-CN-FCV-107, Condensate Recirculation Valve, indicates full open.
Which ONE of the following identifies the required procedural actions of the Reactor Operator?

a) Immediately trip the reactor and perform the immediate operator actions of E-0.

b) Immediately start the 3rd Condensate Pump. If feed flow does not increase above steam flow, trip the reactor and perform the immediate operator actions of E-0.

c) Immediately brief and dispatch an Operator to locally close 1-CN-FCV-107.

d) Immediately start the 3rd Condensate pump and reduce turbine load to decrease steam flow below feed flow.

48.
During 35% power operation, selected first stage impulse pressure transmitter 1-MS-PT-446 (1st Stage Impulse Pressure) fails low. Which ONE of the following identifies the effect on the Steam Generator Level Control system? (Assume all Main Feed Regulating Valves are in automatic)

a) Steam Generator levels decrease to 33%.

b) Steam Generator levels remain at 44%.

c) Main Feed Regulating Valve controllers all shift to auto-hold.

d) Manual Control of Main Feed Regulating valves is required to prevent a reactor trip.

49.
Which ONE of the following identifies the purpose of isolating AFW to a faulted SG?

a) Minimize thermal stresses on the SG tubes.

b) Minimize RCS cooldown.

c) Prevent underfeeding the intact Steam Generators.

d) Minimize AFW pump runout.

50.
Which ONE of the following identifies a basis for the ES-0.1, REACTOR TRIP RESPONSE, direction to realign Main Feed Water flow to the Steam Generators?

a) Minimize long-term SG U-tube thermal stresses generated by prolonged AFW system operation.

b) Minimize potential for AFW pump overheating.

c) Allow rapid recovery of Steam Generator Inventory.

d) Conserve Emergency Condensate Storage Tank Level.

51.
Which ONE of the following tanks **cannot** be pumped DIRECTLY to the Surry Radwaste Facility (SRF)?

a) Contaminated Drain Tank.

b) High Level Liquid Waste Tank.

c) Low Level Liquid Waste Tank.

d) Primary Drains Tank.

52.
During performance of fuel reconstitution in the Fuel Building the following monitors indicate "HIGH" alarms:
1-RM-RI-153, FUEL PIT BRDG
1-VG-RI-109, VENT VENT PART
1-VG-RI-110, VENT VENT GAS

Which ONE of the following response(s) is expected?

a) 1-VS-F-58A/B automatically start and align to ventilate the Fuel and Auxiliary Buildings.

b) Fuel Building exhaust dampers swap to filtered exhaust through 1-VS-F-59.

c) Fuel Building supply fans and unfiltered exhaust fans trip.

d) No automatic actions are expected.

53.
Which ONE of the following failures would initiate an "APPROACH TO SATURATION TEMPERATURE " alarm?

a) RCS wide range pressure transmitter 1-RC-PT-1402 fails low.

b) Train "A" RVLIS fails low.

c) Median Tave 1-RC-TI-1408A, fails high.

d) Any CETC fails high.

54.
The following conditions exist during 100% Unit 2 operation:
AP-16.00 has been performed up to step 6, CHECK SI - NOT REQUIRED.
RCS pressure is 2014 psig decreasing slowly.
RCS temperature is stable at 573°F.
Pressurizer level is 49% and decreasing slowly.
Letdown flow is 0 gpm.
Charging flow is 145 gpm.
Annunciator VSP-F4, AUX BLDG SUMP HI LEVEL, is illuminated.
Containment Sump Narrow Range level is 40% and stable.

Upon transition from E-0, REACTOR TRIP OR SI, which ONE of the following procedural flowpaths is the team expected to use to mitigate this event?

a) ES-0.1, REACTOR TRIP RESPONSE to GOPs for cooldown.

b) E-1, LOSS OF REACTOR OR SECONDARY COOLANT to ECA-1.1, LOSS OF EMERGENCY COOLANT RECIRCULATION.

c) ECA-1.2, LOCA OUTSIDE CONTAINMENT, to E-1, LOSS OF REACTOR OR SECONDARY COOLANT.

d) E-1, LOSS OF REACTOR OR SECONDARY COOLANT, to ES-1.2, POST-LOCA COOLDOWN AND DEPRESSURIZATION.

55.
Which ONE of the following identifies the time requirement and basis for securing one of two running Low Head Safety Injection pumps if RCS pressure is greater than 185 psig?

a) 30 minutes, due to undersized recirculation piping.

b) 60 minutes, due to undersized recirculation piping.

c) 30 minutes, prevent heating RWST water above Tech Spec limits.

d) 60 minutes, prevent heating RWST water above Tech Spec limits.

56.
With Unit 1 at 100% power, describe unit response to a failure of 1-RC-PT-1456, PZR PRESSURE PROTECTION CHANNEL II, in the high direction. Assume 1-RC-PT-1455, PZR PRESSURE PROTECTION CHANNEL I had previously failed low and is now in trip. Assume all other plant systems are operable.

a) Both Pressurizer spray valves modulate open.

b) Pressurizer Power Operated Relief Valve, 1-RC-PCV-1456 opens.

c) Channel II OPDT activates.

d) High pressure reactor trip.

57.
Which ONE of the following identifies the cause of a loss of maximum Pressurizer heater capability?

a) Loss of Motor Control Center 1J1-2.

b) Any pressurizer heater group breaker loses its associated DC control power.

c) 1-RC-LT-1459, PRESSURIZER LEVEL PROTECTION CHANNEL I, fails low while selected to the upper channel.

d) Load Shed.

58.
Which ONE of the following will generate a "COMPUTER PRINTOUT ROD CONT SYS" alarm at 60% power.

a) "ROD CONT MODE SELECTOR" switch in any position other than "AUTO" or "MAN".

b) Any rod control power or logic cabinet experiences a "NON-URGENT" alarm.

c) Any high power rod stop permissive is met.

d) Any IRPI deviates from it's Group step counter by 10 steps.

59.
A 2-OPT-RC-10.1, RCS LEAKAGE MANUAL CALCULATION, has just been performed with Unit 2 at HSD. The STA feels the 0.5 gpm identified leakage increase is from the Reactor Vessel Flange Leakoff Line. Which ONE of the following indications would you use to verify the STA's diagnosis?

a) Trend for increased PRT level.

b) Monitor for increased Containment sump level.

c) Monitor for increased PDTT level.

d) Trend for increasing VCT level.

60.
Aside from providing flow to the Containment Spray Rings, which ONE of the following identifies a function of the Containment Spray Pumps?

a) Cool the ISRS pump recirculation flow to aid NPSH.

b) Provide water to the OSRS pump suction.

c) Provide flow to one half of each Recirculation Spray ring.

d) During outages, provide rapid RWST temperature reduction.

61.
Which ONE of the following components MUST be in service during refueling operations to allow Containment purge to remain in operation.

a) At least one Containment Air Recirculation Fan (1-VS-F-1A/B/C).

b) Manipulator Crane radiation monitor (1-RM-RI-162) <u>AND</u> RX CTMT radiation monitor (1-RM-163)

c) Manipulator Crane radiation monitor (1-RM-RI-162) <u>OR</u> RX CTMT radiation monitor (1-RM-163).

d) At least one Containment purge supply fan (1-VS-F-4A/B).

62.
Which ONE of the following design features prevent a loss of Spent Fuel Pool level if a SFP cooling system leak develops?

a) Suction weir and Cooling pump discharge check valves both located 20 feet above the fuel.

b) Cooling pump low level lockout and automatic high volume firemain makeup.

c) Return line siphon breaker, and SFP bridge radiation monitor automatic actions.

d) Suction weir and return line siphon breaker located 20 feet above the fuel.

63.
Which ONE of the following is indicative of an impending loss of natural circulation flow?

a) RCS delta T at 57°F and increasing .

b) RCS subcooling at 42°F and increasing.

c) Source range detectors counts decreasing.

d) RCS cold leg temperature slowly decreasing.

64.
During 100% power operation, channel III first stage pressure transmitter (1-MS-PT-446) fails low. Which ONE of the following describes how the steam dump system will operate?

a) The steam dumps are armed.

b) The steam dump system will be unaffected during a load reject signal.

c) The steam dumps will modulate closed properly during a Unit trip.

d) All steam dumps open fully.

65.
Which ONE of the following types of radiation monitor detectors is used on the Air Ejector system to allow sensitive response?

a) Geiger-Mueller detector.

b) Gamma Scintillation detector.

c) Uncompensated ion chamber.

d) Beta Scintillation detector.

66.
Following a Unit 1 Hi-Hi CLS, a loss of offsite power occurs. Both Unit 1 emergency busses are reenergized from their associated EDGs. Unit 2 "H" bus reenergizes from its associated EDG. Which ONE of the following actions identifies which Component Cooling water pump **SHOULD** be restored. (Assume CC is crosstied)?

a) 1-CC-P-1A

b) 1-CC-P-1B

c) 1-CC-P-1C

d) 1-CC-P-1D

67.
During performance of 2-ES-0.1, REACTOR TRIP RESPONSE, the Reactor Operator notes the only running AFW pump (2-FW-P-3A) has an extinguished white light with the following parameters indicated:

- 0 amps on the benchboard meter.
- Red light on, green and amber lights out on the 2-FW-P-3A control switch.
- 2-FW-MOV-251A/B/C/D/E/F green lights on red lights off.
- "A", "B", "C" AFW flow indication all indicate "0"
- The light bulb is verified not burned out.

Which ONE of the following identifies the cause of the white light being extinguished?

a) Normal condition.

b) Breaker trip power is lost.

c) Breaker is racked to a position other than "CONNECT".

d) An undervoltage condition exists on the "2H" emergency bus.

68.
A Unit 2 periodic test of #3 EDG is in progress with the diesel in parallel with offsite power. During the test a tornado initiates a loss of the switchyard and lockout of all Reserve Station Transformers. Assuming all systems operate as designed, which ONE of the following identifies the minimum action(s) to restore all four emergency busses.

a) Automatic actions will restore all four emergency busses.

b) #3 EDG must be manually aligned to Unit 1 and the AAC diesel manually aligned to the 2H bus.

c) The AAC diesel must be manually aligned to the 1J bus.

d) Manually realign #3 EDG to 2J bus and manually align the AAC diesel to the 1J bus.

69.
A 20 foot length of piping filled with radioactive fluid has a dose rate of 8 REM/hr at 6 feet. Which ONE of the following APPROXIMATES the dose rate at 4 feet?

a) 18 REM/hr

b) 16 REM/hr

c) 14 REM/hr

d) 12 REM/hr

70.
Which ONE of the following **DOES NOT** result from intake canal level dropping to 23 feet?

a) An "INTAKE CANAL Lo LVL" reactor trip signal is generated.

b) Component Cooling Service Water supply valves receive closed signals.

c) Bearing Cooling Service Water supply valves receive closed signals.

d) Waterbox outlet Motor Operated Valves receive closed signals.

71.
Which ONE of the following determines which Unit's Service air compressor (1-SA-C-1/2-SA-C-1) will run to supply air to the station?

a) Each Compressor's control panel has a "HAND/STBY/OFF" control switch.

b) A local LEAD/LAG control switch designates which compressor runs and which one is in standby.

c) Control switches on each MCR vertical panel allows each Reactor Operator to control his compressor. This allows both compressors to run at once.

d) A stanchion located near each Unit's air dryer contains an "AUTO/STBY/OFF" control switch.

72.
Which ONE of the following conditions would initiate a start of the Diesel Driven Fire Pump?

a) Fire main pressure drops to 99 psig with the local control switch in "AUTO".

b) Breaker 15H8 "NORMAL SUPPLY TO 4160V BUS" opens with the local control switch in any position other than "OFF".

c) Local control switch taken to "TEST".

d) Any ROBERTSHAW system "FIRE" alarm received with the local control switch in "AUTO".

73.
During solid plant conditions with RHR in service, Containment Instrument Air is lost due to inadvertent closing of 1-IA 446. Which ONE of the following identifies the RCS response?

a) Pressure increases, temperature decreases.

b) Pressure increases, temperature increases.

c) Pressure decreases, temperature decreases.

d) Pressure decreases, temperature increases.

74.
Following a reactor trip from 100% power , RCS median Tave is 562°F.
Which ONE of the following describes the position of the steam dump valves?

a) All dumps tripped open.

b) 1-MS-TCV-105A/B, 1-MS-TCV-106A/B tripped open, 1-MS-TCV-107A/B 100% open.

c) All dumps 75% open.

d) 1-MS-TCV-105A/B, 1-MS-TCV-106A/B, 1-MS-TCV-107A/B all 75% open.

75.
During 100% operation, all cooling water to the Containment Air Recirculation Fans , 1-VS-F-1A/B/C is lost. Which ONE of the following is the expected Containment response?

a) Actual partial pressure decreases, indicated partial pressure decreases.

b) Actual partial pressure decreases, indicated partial pressure increases.

c) Actual partial pressure increases, indicated partial pressure increases.

d) Actual partial pressure increases, indicated partial pressure decreases.

76.
The following conditions exist:
Unit 1 is at 30% power.
Breaker 15H3 (#1 EDG output breaker) was placed in PTL 4 hours ago to allow #1 EDG auto start relay testing.
The Operating team has just received notification that control power to the 1-SI-P-1B breaker cubicle is completely lost due to a ground.
All other Unit 1 and Unit 2 components are operable.

Which ONE of the following Limiting Conditions for Operations exist?

a) 7 day to HSD

b) 72 hours to HSD

c) 30 hours to CSD

d) 6 hours to HSD

77.
Which ONE of the following indicates the set of conditions for which ES-0.0, REDIAGNOSIS, could be appropriately used?

a) ES-0.1 is being implemented. The team identifies RCS Tave decreasing.

b) ES-1.2 is being implemented. The team identifies "B" SG level increasing in an uncontrolled manner.

c) ECA-0.0 is being implemented on step 7. "B" SG pressure is identified decreasing in an uncontrolled manner.

d) The team is implementing E-0 following a reactor trip and spurious SI actuation from 100% power. On step 14 it is identified that CTMT sump level is increasing and RCS temperature and pressure are decreasing.

78.
If RCS activity limits are in excess of Tech Spec limits, the Unit is required to be cooled down to 500°F or less within 6 hours after detection.

Which ONE of the following explains the basis for this requirement?

a) Limits peak containment pressure during a LOCA, thereby ensuring site boundary doses remain within 10CFR100 limits with maximum permissible containment boundary leakage.

b) Ensures secondary pressures are maintained well below the SG PORV setpoints should a Steam Generator Tube Rupture occur.

c) Iodine spiking is prevented with RCS temperatures less than 500°F.

d) Ensures the site boundary dose is maintained within 10 CFR 100 limits following a steam line break.

79.
The Unit 1 Turbine Runback circuit is initiating EHC power perturbations during steady state operation. The System Engineer supports a decision to defeat the automatic runback function until the Unit 1 outage starts (approximately 40 days). Which ONE of the following actions support continued operation with the Unit 1 Turbine Runbacks defeated?

a) The Turbine must be placed in "Turbine Manual"; and load reduced to less than 50%.

b) A Tech Spec Amendment is required.

c) No actions required since all Turbine Runback signals are permanently defeated.

d) A 10CFR50.59 safety analysis shall be performed.

80.
Which one of the following is a required action performed in AP-10.13, LOSS OF MAIN CONTROL BOARD ANNUNCIATORS for Unit 1?

a) Secure the ERFCS computer.

b) Check the black battery voltage normal.

c) Check station battery voltage normal.

d) Place the unit in HSD within 6 hours.

81.
Assume that conditions exist which require implementation of the following procedures within each of the following groups. Which ONE of the following has the procedures arranged in the correct order of priority?

a) **E-1**, LOSS OF REACTOR OR SECONDARY COOLANT, **E-2**, FAULTED STEAM GENERATOR ISOLATION, **E-3**, STEAM GENERATOR TUBE RUPTURE.

b) **FR-C.2**, RESPONSE TO DEGRADED CORE COOLING, **FR-H.1**, RESPONSE TO LOSS OF SECONDARY HEAT SINK, **ECA-0.0**, LOSS OF ALL AC POWER.

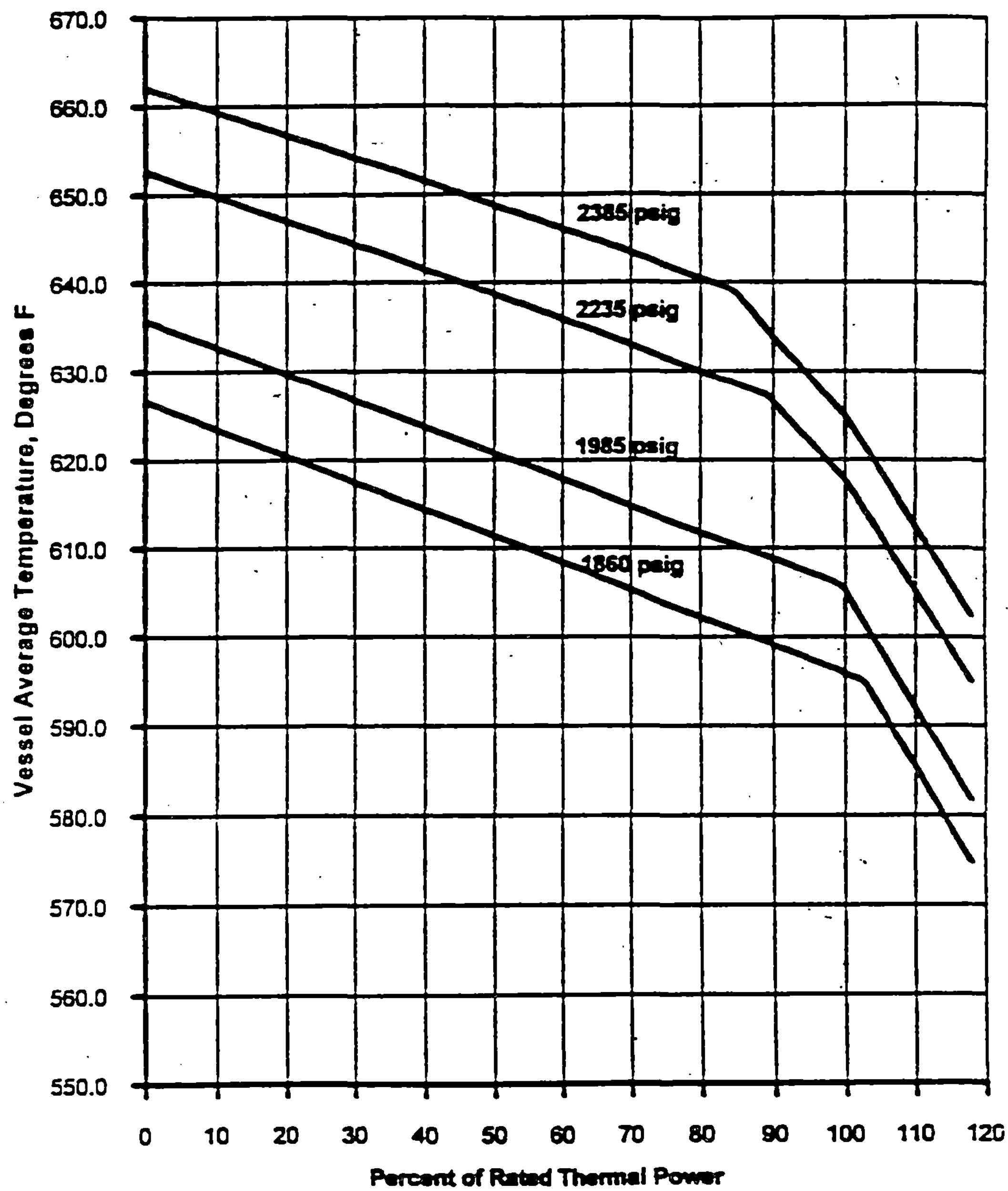
c) **ECA-0.0**, LOSS OF ALL AC POWER, **E-1**, LOSS OF REACTOR OR SECONDARY COOLANT, **FR-S.1**, RESPONSE TO NUCLEAR POWER GENERATION/ATWS.

d) **ES-1.3**, TRANSFER TO COLD LEG RECIRCULATION, **FR-H.1**, RESPONSE TO LOSS OF SECONDARY HEAT SINK, **E-1**, LOSS OF REACTOR OR SECONDARY COOLANT.

82.
Using the attached Tech Spec figure 2.1, Which ONE of the following conditions would be a safety limit violation?

	RCS Pressure (psig)	Rx Power (% rated thermal)	Cold leg temperature (°F)	Hot leg Temperature (°F)
a.	2385	90	580	680
b.	2385	60	570	670
c.	1985	100	560	660
d.	2235	90	560	660

TS FIGURE 2.1-1
REACTOR CORE THERMAL AND HYDRAULIC SAFETY LIMITS
THREE LOOP OPERATION, 100% FLOW



83.
Which ONE of the following conditions identifies the threshold per Annunciator Response Procedure B-E-6, IA LOW HEADER PRESSURE/IA COMPR 1 TRBL, for implementing AP-40.00, NON-RECOVERABLE LOSS OF INSTRUMENT AIR?

a) Instrument air pressure decreases to the low pressure alarm setpoint (80 psig).

b) Instrument air pressure decreases to 50 psig.

c) Instrument air dryer bypasses due to low pressure.

d) A leak is discovered <u>and</u> the affected unit's instrument air compressor is running <u>and</u> instrument air pressure not recovering.

84.
A limiting MCR fire has occurred. The fire has initiated a complete loss of Vital Bus I-1. Which ONE of the following explains how operability of remote indicating Excore channel I can be restored on Unit 1?

a) Rotate unit selector switch on the Remote Monitoring Panel (ASC/RMP) to the "UNIT 1" position.

b) Rotate the selector switch on the Unit 2 Emergency Diesel Generator Isolation Panel to the "ALT" position.

c) Place the "H" bus transfer switch on the Unit 1 Auxiliary Shutdown Panel (ASDP) to the "LOCAL" position.

d) Open the normal supply breaker inside the Unit 2 Appendix "R" panel (2-PP-ESR) then close the alternate supply breaker inside the Unit 1 Appendix "R" panel (1-PP-ESR).

85.
The following conditions exist:

At time 1328 the SEM declared a Alert based on RCS leakage.

At time 1338 the State and Local Communicator completed initial notifications.

At time 1342 the SEM declared a Site Area Emergency based on high CTMT radiation.

At time 1348 the SEM upgraded to General Emergency due to CTMT leakage.

Which ONE of the following identifies the time in which the Protective Action Recommendations must be made to the State?

a) 1353

b) 1357

c) 1403

d) 1428

86.
During 30% power operation on Unit 1 , annunciator 1K-A8, UPS SYSTEM 1A TROUBLE, alarms. The operator in the field reports a "BATTERY CHARGER 1A-1 INPUT FUSE" amber light illuminated with the 1A-1 Battery Charger breaker open. Which ONE of the following identifies the long term operability of the "1A" DC bus?

a) The "A" battery will be the sole source of supply to the bus. Actions need to be taken to minimize DC loads.

b) The UPS 1A-2 will be the sole source of supply to charge the "A" battery.

c) The UPS 1A-1 supply source will automatically shift to MCC 1H1-2 and continue to supply "1A" battery bus.

d) Manually align UPS 1B-1 to charge the "A" battery.

87.
Unit One has just tripped from 100% power due to a loss of both running Main Feed Water Pumps. The Reactor Operators have reported that the immediate actions of 1-E-0, REACTOR TRIP OR SAFETY INJECTION, have been performed from memory and that SI is not required. The Balance of Plant Operator also reports that all Steam Generator levels are off-scale low (narrow range) and all auxiliary feedwater pumps have failed to start (both automatically and manually). Which ONE of the following actions is required?

a) Transition directly to 1-FR-H.1, RESPONSE TO LOSS OF SECONDARY HEAT SINK.

b) Upon transition to 1-ES-0.1, REACTOR TRIP RESPONSE, immediately transition to 1-FR-H.1, RESPONSE TO LOSS OF SECONDARY HEAT SINK.

c) Perform 1-E-0 until transition to FR-H.1, RESPONSE TO LOSS OF SECONDARY HEAT SINK, is directed by the procedure.

d) Upon transition to 1-ES-0.1, REACTOR TRIP RESPONSE, transition to 1-FR-H.5, RESPONSE TO STEAM GENERATOR LOW LEVEL.

88.
The team is currently in E-0, at step 10, responding to excessive RCS leakage (approximately 200 gpm). The Unit RO states that she believes the leakage is from piping upstream of 1-SI-MOV-1890C (LHSI to the cold legs). She asks permission to energize the breaker and close the valve. Which ONE of the following identifies the proper response to this request to deviate from verbatim procedural adherence.

a) The request cannot be granted until ECA-1.2, LOCA OUTSIDE CONTAINMENT, procedural guidance directs actions.

b) The request can be granted, provided the team transitions to ECA-1.2, LOCA OUTSIDE CONTAINMENT, when directed by E-0.

c) The request can be granted provided immediate transition to ECA-1.2, LOCA OUTSIDE CONTAINMENT, is made.

d) The request cannot be granted since the action may affect parameters used within the E-0 diagnostic steps. The deviation from Emergency Operating Procedures is only allowed after transition to 1-ES-0.0, REDIAGNOSIS.

89.
Procedure 1-FR-H.2, STEAM GENERATOR OVERPRESSURE, requires performance if Steam Generator Pressure exceeds the highest safety valve setpoint. Which ONE of the following pressures signifies the highest Steam Generator safety valve setpoint?

a) 1185 psig.

b) 1135 psig.

c) 1110 psig.

d) 1035 psig.

90.
The operating team has been operating with primary to secondary leakage on the Unit 2 "B" Steam Generator. The leakage has been quantified at 0.8 gpd. Which ONE of the following could be used to identify a doubling of primary to secondary leakage?

a) 2-OPT-RC-10.00, RCS LEAKAGE.

b) VCT level trend.

c) 2-AP-16.00, EXCESSIVE RCS LEAKAGE, mass balance.

d) Air Ejector sample.

91.
Which one of the following ensures that an adequate shutdown margin is maintained during a rod ejection accident?

a) Core KW per linear foot.

b) Enthalpy rise hot channel factor limits.

c) Minimum insertion limits.

d) Heat flux hot channel factor limits.

92.
Which one of the following is **NOT** an allowable CTMT purge lineup?

a) 1-VS-F-59 (GENERAL AREA CAT II FILTER EXHAUST FAN) on the CTMT jumper.

b) 1-VS-F-58A (CAT I FILTER EXHAUST FAN) aligned to CTMT and the fuel building.

c) 1-VS-F-58A & B (CAT I FILTER EXHAUST FANS) aligned solely to CTMT.

d) 1-VS-F-58A & B (CAT I FILTER EXHAUST FANS) aligned to the unit 2 safeguards, aux building central, and CTMT.

93.
Which ONE of the following identifies the minimum level of authority that can approve a jumper within a tagging boundary?

a) SNSOC

b) Superintendent of Operations

c) Shift Supervisor

d) MSRC

94.
The "A" WGDT has been in holdup for 14 days and requires release. Which ONE of the following sequences is required to perform a gaseous release?

a) RP obtains and analyzes a gas sample, RP generates a release permit, Operations verifies release information and commences release.

b) RP obtains and analyzes a gas sample, Operations verifies the sample is within the existing batch release permit, and commences release.

c) RP obtains and analyzes a gas sample, Operations verifies the sample is within the existing continuous release permit, and commences release.

d) Based on initial tank contents and decay time since the tank was placed in holdup, RP generates a release permit, Operations verifies release information and commences release.

95.
During core off-load, the refueling team identifies that refueling cavity level is decreasing rapidly in an uncontrolled manner. RWST level is 30%. Which one of the following identifies the priority of aligning a makeup flowpath?

a) Gravity drain from the RWST to the cold legs.

b) HHSI to the hot legs.

c) LHSI to the cold legs.

d) Charging Crosstie.

96.
Which ONE the following identifies the possible source of a loss of Service Water cooling to the Unit 1 Component Cooling system?

a) Intake canal level decreases to 22.7 feet.

b) CW pump discharge vacuum breakers stays closed after a manual pump stop.

c) Unit 1 Hi-Hi CLS.

d) Unit 1 "A" and "C" Hi level structures clog.

97.
Which ONE of the following describes the SROs responsibility concerning reactivity management in accordance with OP-STD-006, REACTIVITY MANAGEMENT?

a) Perform a "Peer Check" by visually verifying each switch manipulation for every reactivity manipulation.

b) Ensure a team brief is held prior to every reactivity manipulation.

c) Inform OMOC prior to every reactivity manipulation.

d) Be informed of and maintain direct supervision over significant reactivity manipulations.

98.
During a Small Break LOCA with failed fuel (peak containment pressure was 16 psia, currently decreasing), the STA reports a CONTAINMENT yellow path due to Containment radiation levels reading 2.5R/hr. The STA recommends going to FR-Z.3, RESPONSE TO CTMT HIGH RADIATION LEVEL. Which ONE of the following actions is directed by FR-Z.3?

a) Verify closed or close ALL phase I and phase II containment isolation valves regardless of function being performed.

b) Manually start the Containment Spray system and ensure the CAT suction is aligned.

c) Align the CAT I Ventilation filters to service.

d) Verify closed or close ALL Containment isolation valves not required for recovery actions.

99.
A special test to determine SI accumulator check valve leakage is scheduled for Unit 1. The test has been approved by all required cognizant departments and individuals. An ICCE brief is being conducted. Which ONE of the following denotes an allowable configuration for the test.

a) The SS will be the Test Coordinator, with the Superintendent of Operations acting as the Senior Operations Manager.

b) A System Engineer will be the Test Coordinator with the Shift Supervisor acting as the Senior Operations Manager.

c) The Superintendent of Operations will be the Senior Operations Manager, the Supervisor Shift Operations will be the Operations Manager On Call.

d) The Unit SRO will be the Test Director, an off-shift Shift Supervisor will be the Senior Operations Manager.

100.
The following conditions exist on Unit 1:

The Unit was tripped 10 minutes ago.
Each RCS hot leg temperature is approximately 500°F
Each RCS cold leg temperature is approximately 496°F
"A" SG pressure is 400 psig and decreasing.
"B" SG pressure is 600 psig and decreasing.
"C" SG pressure is 660 psig and decreasing.
All RCPs are running.
All SG narrow range levels are off-scale low.
AFW flow is 120 gpm to each Steam Generator.

Which ONE of the following identifies the accident in progress?

a) "A" SG is faulted, "B" and "C" pressures are decreasing due to the RCS cooldown.

b) "A" and "B" SGs are faulted, "C" pressure is decreasing due to RCS cooldown.

c) All SGs are faulted.

d) A loss of heat sink has occurred.