superseded pgs per Rev. 2 to Response to NUREG 0737 dtd 12/9/81 50-280

# VIRGINIA ELECTRIC AND POWER COMPANY

## RESPONSE TO NUREG 0737

## POST TMI - REQUIREMENTS

REVISION 0 — DECEMBER 10, 1980

REVISION 1 — MAY 31, 1981

PURPOSE:

The purpose of this report to the Nuclear Regulatory Commission is to provide the total response to and current status of implementing the commitments made by Virginia Electric and Power Company to NUREG 0737 for the North Anna and Surry Power Stations.

BACKGROUND:

NUREG 0578 and NUREG 0660:

By letters dated August 14, October 24, 25, November 26, December 17, 1979, January 10, 31, February 1, 8, 27, April 1, 10, 28, June 6, 9, 30, July 7, 14, and 25, 1980, Vepco has previously submitted commitments and documentation of actions taken at North Anna and Surry Power Stations to implement the Lessons Learned Requirements of NUREG 0578. This information has been consolidated and revised as necessary for those requirements that were incorporated in NUREG 0660 and subsequently reissued with clarifications and changes as NUREG 0737. Responses to the remaining items of NUREG 0660 that have been reissued in NUREG 0737 provide documentation of previous actions and commitments to additional actions necessary to meet the new and/or clarified requirements.

NUREG 0700:

To date the final requirements of NUREG 0700 incorporated in NUREG 0737 have not been issued. By previous letter, Vepco has commented on NUREG CR/1580. Vepco commitments to NUREG 0700 will be provided after issuance by the NRC of the final requirements.

SCOPE:

This report is to stand alone as Vepco's response to NUREG 0737. The report is a compilation of previously submitted information referenced above for the items of NUREG 0578 and NUREG 0660. Future correspondence submitted in response to NUREG 0700 items will be included in this report. In addition, this report presents the current status of Vepco's efforts in implementing the commitments.

Section E of this report, Response, follows the NRC format in NUREG 0737 of three chapters of requirements: I. Operational Safety; II. Siting and Design; and III. Emergency Preparedness and Radiation Effects. To facilitate review, the individual response sections include the NRC position and clarifications along with the detailed Vepco response.

Each Response section attempts to provide a status of the engineering and construction and a design description which addresses each clarification item and identifies anticipated deviations from the technical or schedular requirements of NUREG 0737.

This report will be updated periodically by additional or replacement pages to maintain the required documentation and status of implementation. Section D of this report presents a list of the requirements and the status of Vepco's responses. This Status List will be updated with each addition to the report.

| CLARIFICATION ITEM | SHORTENED TITLE | EXCEPTIONS |
|---|---|---|
| I.A.1.1 | Shift technical advisor | NONE |
| I.A.1.2 | Shift supervisor responsibilities | NONE |
| I.A.1.3 | Shift manning | NONE |
| I.A.2.1 | Immediate upgrading of RO and SRO training and qualifications | NONE |
| I.A.2.3 | Administration of training programs | NONE |
| I.A.3.1 | Revise scope and criteria for licensing exams | NONE |
| I.C.1 | Short-term accident and procedures review | NONE |
| I.C.2 | Shift & relief turnover procedures | NONE |
| I.C.3 | Shift-supervisor responsibility | NONE |
| I.C.4 | Control-room access | NONE |
| I.C.5 | Feedback of operating experience | NONE |
| I.C.6 | Verify correct performance of operating activities | 1) None at Surry.<br>2) At North Anna administrative controls will be in effect until procedures are reviewed and revised. This will be completed by June 1, 1981. |

| CLARIFICATION ITEM | SHORTENED TITLE | EXCEPTIONS |
|---|---|---|
| I.D.1 | Control-room design | 1) None at this time - NUREG 0700 has not been issued.<br>2) Vecpo assumes the licensee submittal date should be "later" rather than 4/82. |
| I.D.2 | Plant safety parameter display console | NONE |
| II.B.1 | Reactor coolant system vents | 1) Vepco requires a relaxation of the final documentation submittal date to January 1, 1982 rather than July 1, 1981 for complete listing of qualification, final procedures and final electrical drawings for all units.<br>* 2) Vepco requires a relaxation of the required installation date to July 1, 1982 or first refueling after January 1, 1981, which ever is later, rather than installation by July 1, 1982. |
| II.B.2 | Plant shielding | 1) Some systems included in the NRC clarifications were excluded from the review. Justifications are provided.<br>2) Liners and seals in the recirculation spray valves at Surry Unit 2 will be replaced at next refueling currently scheduled for the first quarter, 1982. This date could potentially delay beyond the July 1, 1982 requested date.<br>3) Radiation zone maps for personnel access will be revised after all designs are installed. |
| II.B.3 | Post-accident sampling | NONE |
| II.B.4 | Training for mitigating core damage | NONE |

| CLARIFICATION ITEM | SHORTENED TITLE | EXCEPTIONS |
|---|---|---|
| II.D.1 | Relief & safety-valve test requirements | 1) The EPRI program has not formally included the testing of block valves. |
| II.D.3 | Valve position indication | 1) Equipment is undergoing seismic and environmental testing and is scheduled to be completed Summer, 1981. |
| II.E.1.1 | Auxiliary Feedwater system evaluation | NONE |
| II.E.1.2 | Auxiliary feedwater system initiation | |
| | Part 1 | 1) The original design criteria of the plants satisfy the requirements and meet the intent of the clarifications. |
| | Part 2 | 1) The original design criteria of the plants satisfy the requirements and meet the intent of the clarifications. |
| II.E.3.1 | Emergency power for pressurizer heaters | NONE |
| II.E.4.1 | Dedicated hydrogen penetrations | NONE |
| II.E.4.2 | Containment isolation dependability | 1) Justification is provided for the existing containment pressure isolation setpoint. |
| | | * 2) Category I containment radiation isolation signal is assumed not to be required for normally closed containment purge and vent valves. |
| | | 3) Original plant design criteria as described in the FSAR took exceptions to General Design Criteria. |
| II.F.1 | Accident-monitoring | |
| | 1. Noble gas effluents | 1) Low range sensitivity of Main Steam Monitors is restricted by very high background radiation. |

| CLARIFICATION ITEM | SHORTENED TITLE | | EXCEPTIONS |
|---|---|---|---|
| II.F.1 Cont'd | | 2) | Some design information will not be available by January 1, 1981 but will be available by November 1, 1981. Final calibration and operating procedures will be available by the required installation date. |
| | 2. Sampling & analysis of effluents | 1) | Final design details will be available prior to November 1, 1981 rather than the required documentation date of January 1, 1981 based on the availability of vendor information. |
| | | 2) | Analytical procedures will be available by the required implementation date. |
| | 3. Containment radiation monitors | 1) | Transmitters qualified to IEEE-344 & 323-1971 will be installed and upgraded per the IE Bulletin 79-01B program. The upgrade may not be complete by January 1, 1982. |
| | 4. Containment pressure | 1) | Best equipment available will be used and upgraded under the I.E. Bulletin 79-01B program. Upgrade may not be complete by January 1, 1982. |
| | 5. Containment water level | 1) | Best equipment available will be used and upgraded under the I.E. Bulletin 79-01B program. Upgrade may not be complete by January 1, 1982. |
| | 6. Containment hydrogen | 1) | Hydrogen analyzers qualification test has not been approved. |
| II.F.2 | Instrumentation for detection of inadequate core cooling | * 1) | Vepco requires a relaxation of the required installation date to the first refueling after system availability or January 1, 1982, which ever is later, rather than installation by January 1, 1982. |
| | | 2) | Clarification item (7) is a new requirement. Comparison of the proposed Westinghouse design with the Appendix B requirements is not complete. |

| CLARIFICATION ITEM | SHORTENED TITLE | EXCEPTIONS |
|---|---|---|
| II.F.2 | Instrumentation for detection of inadequate core cooling (continued) | * 3) The ability of any system to provide an unambiguous indication has not been demonstrated. Procurement has proceeded on the information available. The required installation date should be delayed until additional research is performed. If the implementation date is not delayed, the installed level system should not be required to be changed if future research provides a better system. |
| II.G.1 | Power supplies for pressurizer relief valves, block valves and level indicators | NONE |
| II.K.1 | IE Bulletins | NONE |
| II.K.2 | Orders on B&W plants | |
| | .13 Thermal Mechanical Supports | NONE |
| | .17 Voiding in RCS | NONE |
| | .19 Bench Mark Analysis AFW | NONE |
| II.K.3 | Final recommendations, B&O task force | |
| | .1 PORV isolation system | NONE |
| | .2 PORV Failures | NONE |
| | .3 SV & RV Failures | NONE |
| | .5 Auto trip RCP's | Vepco requires an extension of the July 1, 1981 date for submittal of proposed design information to three (3) months from NRC approval of the analysis model used. This request is consistent with the timetable provided in Item II.K.3.5. |
| | .9 PID Controller | NONE |
| | .10 Proposed Anticipatory Trip | NONE |

| CLARIFICATION ITEM | SHORTENED TITLE | EXCEPTIONS |
|---|---|---|
| II.K.3 | Final recommendations, B&O task force (continued) | |
| | .11 Use of certain PORV's | NONE |
| | .12 Anticipatory Trip on Turbine Trip | NONE |
| | .17 ECC system outages | NONE |
| | .25 Power on Pump Seals | NONE |
| | .30 SB LOCA Methods | NONE |
| | .31 Compliance with 10 CFR 50.46 | NONE |
| III.A.1.1 | Emergency preparedness, short-term | NONE |
| III.A.1.2 | Upgrade emergency support facilities | NONE |
| III.A.2 | Emergency preparedness | * 1) Implementation of improved meteorological data capabilities must be conducted on the same schedule as portions of III.A.1.2. |
| III.D.1.1 | Primary coolant outside containment | NONE |
| III.D.3.3 | Inplant radiation monitoring | NONE |
| III.D.3.4 | Control-room habitability | 1) None for North Anna<br>* 2) Vepco requires a relaxation of the January 1, 1981 documentation date for Surry. The analysis of off-site chemical shipments will be submitted by June 30, 1981. |
| Appendix B Design Criteria | | * 1) Equipment does not meet the seismic test results of R.G. 1.100. |

| CLARIFICATION ITEM | SHORTENED TITLE | EXCEPTIONS |
|---|---|---|
| Appendix B | Design Criteria (continued) | 2) No additional vendor documentation was required for "extended range" qualification. |
| | | * 3) Equipment does not meet the IEEE-323-1974 requirements of R.G. 1.89. |
| | | * 4) Existing plant systems meet original plant criteria for electrical separation but not R.G. 1.75. |
| | | 5) The individual requirements of the referenced Reg. Guides in Item (5) were not addressed. The design was done in accordance with the requirements of 10CFR50, Appendix B and ANSI N45.2. |
| | | 6) Existing plant instrumentation used for normal and post-accident conditions will be reviewed as part of the Control Room Design Review, Section I.D.1 for proper identification. |

# PROPOSED MODIFICATIONS AND TENTATIVE OUTAGE SCHEDULE NORTH ANNA AND SURRY

## North Anna

Unit 1 Outages:

FM - Fall, 1981 maintenance outage.

3rd R - Third refueling outage currently scheduled for Summer of 1982.

Unit 2 Outages:

FM - Fall, 1981 maintenance outage.

1st R - First refueling outage currently scheduled for the first quarter of 1982.

## Surry

Unit 1 Outage:

SGRP - Steam generator replacement outage (in progress).

Unit 2 Outages:

FM - Fall, 1981 maintenance outage.

5th R - Fifth refueling outage currently scheduled for the first quarter of 1982.

## PROPOSED TMI MODIFICATION SCHEDULE
## NORTH ANNA AND SURRY

| NUREG-0737 MODS | REQ DATE | UNIT | EST MAT'L DEL | OUT. REQ | NORTH ANNA #1 FM | NORTH ANNA #1 3rd R | NORTH ANNA #2 FM | NORTH ANNA #2 1st R | SURRY #1 SGR | SURRY #2 FM | SURRY #2 5th R | REMARKS |
|---|---|---|---|---|---|---|---|---|---|---|---|---|
| RCS & PZR Vent (II.B.1) | 7/82 | NA1 | Onsite | Yes | | X | | | | | | DC-79-S69A - Unit 1 |
| | 7/82 | NA2 | Onsite | Yes | | | | X | | | | DC-79-69B - Unit 2 |
| | 7/82 | S1 | Onsite | Yes | | | | | X | | | DC-80-S29 - Rx Vessel head removal required. |
| | 7/82 | S2 | Onsite | Yes | | | | | | | X | |
| Plant Shielding (II.B.2) | | | | | | | | | | | | |
| Hydrogen Recombiner Vault Shielding | 1/82 | NA1 | Onsite | No | | | | | | | | DC-80-S36 North Anna only. |
| | 1/82 | NA2 | Onsite | No | | | | | | | | |
| S.W. Rad Monitor Pump Mods. | 1/82 | NA1 | Onsite | Yes | | | | | | | | DC-80-S25A Unit 1 complete |
| | 1/82 | NA2 | Onsite | Yes | | | X | | | | | |
| | 1/82 | S1 | Onsite | No | | | | | | | | DC-80-57 (motors) |
| | 1/82 | S2 | Onsite | No | | | | | | | | |
| S.W. Rad Monitor Shielding | 1/82 | NA1 | Onsite | No | | | | | | | | DC-80-S33 |
| | 1/82 | NA2 | Onsite | No | | | | | | | | |
| | 1/82 | S1 | Onsite | No | | | | | | | | DC-80-S56 Piping & electrical to be installed |
| | 1/82 | S2 | Onsite | No | | | | | | | | |

| NUREG-0737 MODS | REQ DATE | UNIT | EST MAT'L DEL | OUT. REQ | NORTH ANNA #1 FM | NORTH ANNA #1 3rd R | NORTH ANNA #2 FM | NORTH ANNA #2 1st R | SURRY #1 SGR | SURRY #2 FM | SURRY #2 5th R | REMARKS |
|---|---|---|---|---|---|---|---|---|---|---|---|---|
| Replacement of Mission Check Valves | 1/82 | NA1 | Onsite | Yes | | X | | | | | | North Anna Unit 1 only. DC-80-S23 Valve Bodies |
| Replacement of Mission Check Valve Bearings | 1/82 | NA1 | Onsite | Yes | | X | | | | | | DC-80-S24A - Unit 1 Replace 13<br>DC-80-S24B - Unit 2 replace 13<br>DC-80-S24C - Requires 2 unit outage for 2 valves or entry into Tech Specs. action statement. |
| | 1/82 | NA2 | Onsite | Yes | | | | X | | | | |
| | 1/82 | S1 | Onsite | Yes | | | | | X | | | DC-80-53 |
| | 1/82 | S2 | Onsite | Yes | | | | | | X | | Complete |
| Replace Charging Pump Bearing Gaskets | 1/82 | S1 | Onsite | No | | | | | | | | Surry only. DC-80-S58 |
| | 1/82 | S2 | Onsite | No | | | | | | X | | Unit 1 complete |
| Replace Mech Seal - Charging Pump Cooling Water Pump | 1/82 | S1 | Onsite | No | | | | | | | | Surry only. DC-80-S55 |
| | 1/82 | S2 | Onsite | No | | | | | | | | |
| Replace LMC Ball Valves (Primary Drain Tank) | 1/82 | S1 | Onsite | Yes | | | | | X | | | Surry only. DC-80-S67 |
| | 1/82 | S2 | Onsite | Yes | | | | | | X | | |
| Replace Charging Pump Seal Water Cooler-O-Rings | 1/82 | S1 | Onsite | No | | | | | | | | Surry only. DC-80-S66 |
| | 1/82 | S2 | Onsite | No | | | | | | | | |
| Charging Pump Lube Oil Cooler TCV Mods | 1/82 | S1 | 8/81 | No | | | | | | | | Surry only. DC-80-54 |
| | 1/82 | S2 | 8/81 | No | | | | | | | | |

| NUREG-0737 MODS | REQ DATE | UNIT | EST MAT'L DEL | OUT. REQ | NORTH ANNA #1 FM | NORTH ANNA #1 3rd R | NORTH ANNA #2 FM | NORTH ANNA #2 1st R | SURRY #1 SGR | SURRY #2 FM | SURRY #2 5th R | REMARKS |
|---|---|---|---|---|---|---|---|---|---|---|---|---|
| Replace Liners & Seals in Recirc. Spray Plug Valves | 1/82 | S1 | Onsite | Yes | | | | | | | | Surry only. DC-80-S69 Unit 1 complete (valve vendor Rep. required) |
| | 1/82 | S2 | Onsite | Yes | | | | | | | X | INSTALLATION AFTER 1/82 FOR S2. |
| Safeguards Area Vent Fan Motors Replacement | 1/82 | NA1 | Onsite | No | | | | | | | | North Anna only. DC-80-S03 |
| | 1/82 | NA2 | Onsite | No | | | | | | | | |
| $H_2$ Recombiner Gas Cooler Outlet Mod | 1/82 | NA1 | 4/81 | Yes | | X | | | | | | DC-80-21A Reach Rods<br>DC-80-21B Shield Wall |
| | 1/82 | NA2 | 4/81 | Yes | | | X | | | | | DC-80-21C Control Panel/Trickle Htr. (Trickle heater not required by 1/82) |
| Post-Accident Sampling (II.B.3) | | | | | | | | | | | | |
| RCS & Containment Sampling Facility | 1/82 | NA1 | 8/81 | No | | | | | | | | DC-80-S48 - Outage required for some containment tie-ins only. |
| | 1/82 | NA2 | 8/81 | No | | | | | | | | |
| | 1/82 | S1 | 8/81 | No | | | | | | | | DC-80-31 - Outage required for some containment tie-ins only. |
| | 1/82 | S2 | 8/81 | Yes | | | | | | X | | |
| Sample System Isolation Valves | 1/82 | NA1 | Onsite | Yes | | X | | | | | | DC-80-S32A - Unit 1 |
| | 1/82 | NA2 | Onsite | Yes | | | X | | | | | DC-80-S32B - Unit 2 |
| | 1/82 | S1 | Onsite | Yes | | | | | | | | DC-80-60A - Unit 1 - complete |
| | 1/82 | S2 | Onsite | Yes | | | | | | X | | DC-80-60B - Unit 2 |

| NUREG-0737 MODS | REQ DATE | UNIT | EST MAT'L DEL | OUT. REQ | NORTH ANNA #1 FM | NORTH ANNA #1 3rd R | NORTH ANNA #2 FM | NORTH ANNA #2 1st R | SURRY #1 SGR | SURRY #2 FM | SURRY #2 5th R | REMARKS |
|---|---|---|---|---|---|---|---|---|---|---|---|---|
| Sample Containment Return Line | 1/82 | NA1 | Onsite | Yes | | X | | | | | | DC-80-60 |
| | 1/82 | NA2 | Onsite | Yes | | | | X | | | | |
| | 1/82 | S1 | Onsite | Yes | | | | | X | | | DC-80-78 |
| | 1/82 | S2 | Onsite | Yes | | | | | | | X | |
| Aux Feedwater Mods (II.E.1.1) | | | | | | | | | | | | |
| Pump Auto Control | 1/82 | S1 | 10/81 | Yes | | | | | X | | | Work includes replacing PCV & MOV in steam line to turbine driven pump. |
| | 1/82 | S2 | 10/81 | Yes | | | | | | | X | |
| Misc. Auxiliary Auxiliary Feedwater Mods (II.E.1.1) | | | | | | | | | | | | |
| AFW Flow Restricting Orifices | R | S1 | Onsite | Yes | | | | | | | | Unit 1 complete |
| | R | S2 | 8/81 | Yes | | | | | | | X | |
| ECST Level Alarm | 7/81 | NA1 | Onsite | C R | | | | | | | | DC-80-S02 Unit 1<br>DC-80-S42 Unit 2 |
| | 7/81 | NA2 | Onsite | C R | | | | | | | | NA complete |
| | 7/81 | S1 | Onsite | C R | | | | | | | | DC-80-S37 |
| | 7/81 | S2 | Onsite | C R | | | | | | | | Surry complete |

| NUREG-0737 MODS | REQ DATE | UNIT | EST MAT'L DEL | OUT. REQ | NORTH ANNA #1 FM | NORTH ANNA #1 3rd R | NORTH ANNA #2 FM | NORTH ANNA #2 1st R | SURRY #1 SGR | SURRY #2 FM | SURRY #2 5th R | REMARKS |
|---|---|---|---|---|---|---|---|---|---|---|---|---|
| Dedicate $H_2$ Penetration (II.E.4.1) | | | | | | | | | | | | |
| Replacement of $H_2$ Analyzer & Hydrogen Recombiner Valves | 7/82 | NA1 | 6/81 | Yes | | X | | | | | | DC-80-S31A&B Entry into Tech Spec action statement required. PAM&C panel must be installed. |
| | 7/82 | NA2 | 6/81 | Yes | | | | X | | | | |
| | 7/82 | S1 | 6/81 | Yes | | | | | X | | | DC-79-62B Entry into Tech Spec action statement required. PAM&C panel must be installed. |
| | 7/82 | S2 | 6/81 | Yes | | | | | | | X | |
| Containment Isolation (II.E.4.2) | | | | | | | | | | | | |
| Diverse Signal to Condenser Air Ejector | 1/81 | S1 | Onsite | Yes | | | | | | | | Surry only. DC-80-S90 Complete |
| | 1/81 | S2 | Onsite | Yes | | | | | | | | |
| T.D. AFW SI Reset | 1/81 | NA1 | Onsite | No | | | | | | | | DC-79-S75 Complete |
| | 1/81 | NA2 | Onsite | No | | | | | | | | |
| CRDM Fans SI Reset | 1/81 | NA1 | Onsite | No | | | | | | | | |
| | 1/81 | NA2 | Onsite | No | | | | | | | | DC-79-S76 Complete |
| Cond Air Ejector SI Reset | 1/81 | NA1 | Onsite | No | | | | | | | | DC-80-S11 Complete |
| | 1/81 | NA2 | Onsite | No | | | | | | | | |

| NUREG-0737 MODS | REQ DATE | UNIT | EST MAT'L DEL | OUT. REQ | NORTH ANNA #1 FM | NORTH ANNA #1 3rd R | NORTH ANNA #2 FM | NORTH ANNA #2 1st R | SURRY #1 SGR | SURRY #2 FM | SURRY #2 5th R | REMARKS |
|---|---|---|---|---|---|---|---|---|---|---|---|---|
| Increased Range of Rad Monitors (II.F.1) | | | | | | | | | | | | |
| Containment High Range | 1/82 | NA1 | 7/81 | Yes | | X | | | | | | DC-80-S35A Unit #1 Containment. DC-80-S35B Unit #2 Containment. DC-80-S35C Outside Containment. Non-outage - C.R. Equipment ~7/81. Electrical termination problems. |
| | 1/82 | NA2 | 7/81 | Yes | | | | X | | | | |
| | 1/82 | S1 | 7/81 | Yes | | | | | X | | | |
| | 1/82 | S2 | 7/81 | Yes | | | | | | | X | DC-80-S52 - Penetration problem. Non-outage - C.R. Equipment ~7/81. Electrical termination problems. |
| Main Steam Effluent | 1/82 | NA1 | ~9/81 | No | | | | | | | | DC-80-S37B - Electrical upgrade & recorders. |
| | 1/82 | NA2 | ~9/81 | No | | | | | | | | |
| | 1/82 | S1 | ~9/81 | No | | | | | | | | DC-80-64B - Electrical upgrade & recorders. |
| | 1/82 | S2 | ~9/81 | No | | | | | | | | |
| Steam Driven Aux Feed | 1/82 | NA1 | 9/81 | No | | | | | | | | DC-80-S37A |
| | 1/82 | NA2 | 9/81 | No | | | | | | | | |
| | 1/82 | S1 | 9/81 | No | | | | | | | | DC-80-64C |
| | 1/82 | S2 | 9/81 | No | | | | | | | | |
| Process & Vent Rad | 1/82 | NA1 | ~9/81 | No | | | | | | | | DC-80-S37C |
| | 1/82 | NA2 | ~9/81 | No | | | | | | | | |
| | 1/82 | S1 | ~9/71 | No | | | | | | | | DC-80-64A |
| | 1/82 | S2 | ~9/81 | No | | | | | | | | |

| NUREG-0737 MODS | REQ DATE | UNIT | EST MAT'L DEL | OUT. REQ | NORTH ANNA #1 FM | NORTH ANNA #1 3rd R | NORTH ANNA #2 FM | NORTH ANNA #2 1st R | SURRY #1 SGR | SURRY #2 FM | SURRY #2 5th R | REMARKS |
|---|---|---|---|---|---|---|---|---|---|---|---|---|
| Containment Accident Monitoring (II.F.1) | | | | | | | | | | | | |
| Containment Pressure | 1/82 | NA1 | Onsite | Yes | | X | | | | | | DC-79-S67 - Complete |
| | 1/82 | NA2 | Onsite | Yes | | | | X | | | | Non-qualified indicators will be used until qualified indicators are available. |
| | 1/82 | S1 | Onsite | Yes | | | | | X | | | DC-79-60 - Non-qualified indicators will be used until qualified indicators |
| | 1/82 | S2 | Onsite | Yes | | | | | | | X | are available. |
| Containment Water Level | 1/82 | S1 | 7/81 | Yes | | | | | X | | | Surry only. Gems transmitters undergoing qualification. May install |
| | 1/82 | S2 | 7/81 | Yes | | | | | | | X | without qualification. First shipment of equip defective. |
| Containment Hydrogen Monitor | 1/82 | NA1 | 7/81 | No | | | | | | | | DC-79-S68 |
| | 1/82 | NA2 | 7/81 | No | | | | | | | | |
| | 1/82 | S1 | 7/81 | No | | | | | | | | DC-79-62A |
| | 1/82 | S2 | 7/81 | No | | | | | | | | |
| PAM&C Panel | 7/81 | NA1 | Onsite | No | | | | | | | | DC-80-S29 - Installation required for completion of containment hydrogen, |
| | 7/81 | NA2 | Onsite | No | | | | | | | | pressure & RCS Vent. |
| | 7/81 | S1 | Onsite | No | | | | | | | | DC-80-SS - Installation required for completion of containment hydrogen, |
| | 7/81 | S2 | Onsite | No | | | | | | | | water level, pressure & RCS Vent. |

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|---|---|---|---|---|---|---|---|---|---|---|---|---|
| Inadequate Core Cooling (II.F.2) | | | | | | | | | | | | |
| Redundant RCS Wide | 1/81 | S1 | Onsite | Yes | | | | | | | | Surry only. Complete |
| | 1/81 | S2 | Onsite | Yes | | | | | | | | |
| Reactor Vessel Level | 1/82 | NA1 | ~9/81 | Yes | | X | | | | | | DC-XX-XXX (Westinghouse level system) |
| | 1/82 | NA2 | ~9/81 | Yes | | | | X | | | | |
| | 1/82 | S1 | ~9/81 | Yes | | | | | X | | | DC-XX-XX Sixth refueling outage if not available for work during |
| | 1/82 | S1 | ~9/81 | Yes | | | | | | | X | SGR. (Westinghouse level system) |

*Denotes a request for revised implementation date. Reference Vepco letters dated November 20, 1980 (Serial No. 924), and November 18, 1980 (Serial No. 922).

STATUS LIST OF RESPONSES
TO NUREG 0737

POST-TMI REQUIREMENTS FOR OPERATION REACTORS

| Clarification Item | Shortened Title | Description | Implementation Schedule | NRC Pre-Implementation Approval | NRC Post-Implementation Review Required | Tech Spec. Revision Required | Submittal Req. By | Vepco Remarks |
|---|---|---|---|---|---|---|---|---|
| I.A.1.1 | Shift technical advisor | 1. On duty | 1-1-80 | No | Yes | Yes | 1-1-80 | Complete |
| | | 2. Tech specs | 12-15-80 | Yes | No | Yes | 9-1-80 | Complete |
| | | 3. Trained per LL Cat B | 1-1-81 | No | Yes | Yes | 1-1-81 | Complete |
| | | 4. Describe long-term program | 1-1-81 | No | No | No | 1-1-81 | Submitted |
| I.A.1.2 | Shift supervisor responsibilities | Delegate non-safety duties | 1-1-80 | No | Yes | No | 1-1-80 | Complete |
| I.A.1.3 | Shift manning | 1. Limit overtime | 11-1-80 | No | Yes | No | 11-1-80 | Complete |
| | | 2. Min shift crew | 7-1-82 | No | Yes | Yes | 11-1-80 | Complete |
| I.A.2.1 | Immediate upgrading of RO and SRO training and qualifications | 1. SRO exper | 5-1-80 | No | Yes | No | None | Complete |
| | | 2. SRO's be RO's | 12-1-80 | No | Yes | No | None | Complete |
| | | 3. Three mo trng on shift | 8-1-80 | No | Yes | No | None | Complete |
| | | 4. Modify training | 8-1-80 | No | Yes | No | 8-1-80 | Complete |
| | | 5. Facility certification | 5-1-80 | No | Yes | No | None | Complete |
| I.A.2.3 | Administration of training programs | Instructors complete SRO exam | 8-1-80 | No | Yes | No | None | Complete |

| Clarification Item | Shortened Title | Description | Implementation Schedule | NRC Pre-Implementation Approval | NRC Post-Implementation Review Required | Tech Spec. Revision Required | Submittal Req. By | Vepco Remarks |
|---|---|---|---|---|---|---|---|---|
| I.A.3.1 | Revise scope and criteria for licensing exams | 1. Increase scope | 5-1-80 | No | No | No | None | Complete |
| | | 2. Increase passing grade | 5-1-80 | No | No | No | None | Complete |
| | | 3. Simulator exams | 6-1-80 | No | No | No | None | Complete |
| | | | 10-1-81 | No | No | No | None | Complete |
| I.C.1 | Short-term accident and procedures review | 1. SB LOCA | 6-1-80 | No | Yes | No | None | Complete |
| | | 2. Inadequate core cooling | | | | | | |
| | | a. Reanalyze and propose guidelines | 1-1-81 | Yes | No | No | 1-1-81 | Submitted |
| | | b. Revise procedures | 1st refueling outage after 1-1-82 | Yes | No | No | TBD | |
| | | 3. Transients & accidents | | | | | | |
| | | a. Reanalyze and propose guidelines | 1-1-81 | Yes | No | No | 1-1-81 | Submitted |
| | | b. Revise procedures | 1st refueling outage after 1-1-82 | Yes | No | No | TBD | |
| I.C.2 | Shift & relief turnover procedures | Implement shift turnover checklist | 1-1-81 | No | Yes | No | 1-1-80 | Complete |
| I.C.3 | Shift-supervisor responsibility | Clearly define superv & oper responsibilities | 1-1-80 | No | Yes | No | 1-1-80 | Complete |
| I.C.4 | Control room access | Establish authority limit access | 1-1-80 | No | Yes | No | 1-1-80 | Complete |

| Clarification Item | Shortened Title | Description | Implementation Schedule | NRC Pre-Implementation Approval | NRC Post-Implementation Review Required | Tech Spec. Revision Required | Submittal Req. By | Vepco Remarks |
|---|---|---|---|---|---|---|---|---|
| I.C.5 | Feedback of operating experience | License to implement procedures | 1-1-81 | No | Yes | No | None | Complete |
| I.C.6 | Verify correct performance of operating activities | Revise performance procedures | 1-1-81 E | No | Yes | No | None | |
| I.D.1 | Control room design | Preliminary assessment | TBD | | | | TBD E | Response after issue of NUREG 0700 |
| I.D.2 | Plant safety parameter display console | 1. Description | 6-1-81 | No | Yes | No | 6-1-81 | |
| | | 2. Installed | 10-1-82 | No | Yes | No | 10-1-82 | |
| | | 3. Fully implemented | 10-1-82 | Yes | No | Yes | 10-1-82 | |
| II.B.1 | Reactor coolant system vents | 1. Design vents | 7-1-81 | No | Yes | No | 7-1-81 | Submitted |
| | | 2. Install vents (LL Cat B) | 7-1-82 E | Yes | No | Yes | 7-1-81 E | |
| | | 3. Procedures | 7-1-82 | Yes | No | Yes | 1-1-81 E | |
| II.B.2 | Plant shielding | 1. Review designs | 1-1-80 | No | Yes | No | 1-1-80 | Submitted |
| | | 2. Plant modifications (LL Cat B) | 1-1-82 E | No | Yes | No | 1-1-81 submittal if deviation from position | Submitted |
| | | 3. Equipment qualification | 6-30-82 | No | Yes | No | 1-1-82 | |

| Clarification Item | Shortened Title | Description | Implementation Schedule | NRC Pre-Implementation Approval | NRC Post-Implementation Review Required | Tech Spec. Revision Required | Submittal Req. By | Vepco Remarks |
|---|---|---|---|---|---|---|---|---|
| II.B.3 | Post-accident sampling | 1. Interim system | 1-1-80 | No | Yes | No | 1-1-80 | Complete |
| | | 2. Plant modifications | 1-1-82 | No | Yes | Yes | 1-1-81 submittal if deviation from position | Submitted |
| II.B.4 | Training for mitigating core damage | 1. Develop training program | 1-1-81 | No | Yes | No | 1-1-81 | Complete |
| | | 2. Implement program | | | | | | |
| | | a. Initial | 4-1-81 | No | Yes | No | None | Complete |
| | | b. Complete | 10-1-81 | No | Yes | No | None | |
| II.D.1 | Relief & safety-valve test requirements | 1. Submit program | 1-1-80 | No | Yes | No | 1-1-80 | Complete |
| | | 2. RV & SV testing (LL Cat B) | | | | | | |
| | | a. Complete testing | 7-1-81 | No | No | No | 7-1-81 | |
| | | b. Plant-specific report | 10-1-81 | Yes | Yes | TBD | 1-1-82 | |
| | | 3. Block-valve testing | 7-1-82 | Yes | Yes | TBD | 7-1-82 | |
| II.D.3 | Valve position indication | 1. Install direct indications of valve position | 1-1-80 | No | Yes | Yes | 1-1-80 | Complete (qualifying equipment) |
| | | 2. Tech Specs | 12-15-80 | Yes | No | Yes | 9-1-80 | Submitted |
| II.E.1.1 | Auxiliary Feedwater system evaluation | 1. Short-term | 7-1-81 | Yes | Yes | Item Specific | Plant Specific | Submitted |
| | | 2. Long-term | 1-1-82 | Yes | Yes | Item Specific | Plant Specific | Submitted |

| Clarification Item | Shortened Title | Description | Implementation Schedule | NRC Pre-Implementation Approval | NRC Post-Implementation Review Required | Tech Spec. Revision Required | Submittal Req. By | Vepco Remarks |
|---|---|---|---|---|---|---|---|---|
| II.E.1.2 | Auxiliary feedwater system initiation | 1. Initiation | | | | | | |
| | | a. Control grade | 6-1-80 | Yes | No | Yes | 1-1-80 | Complete |
| | | b. Safety grade | 7-1-81 | No | Yes | Yes | 1-1-81 | Complete |
| | | 2. Flow indication | | | | | | |
| | | a. Control grade | 1-1-80 | No | Yes | Yes | 1-1-80 | Complete |
| | | b. LL A tech specs | 12-15-80 | Yes | No | Yes | 9-1-80 | Submitted |
| | | c. Safety grade | 7-1-80 | No | Yes | Yes | 1-1-81 | Complete |
| II.E.3.1 | Emergency power for pressurizer heaters | 1. Upgrade power supply | 1-1-80 | No | Yes | Yes | 1-1-81 | Complete |
| | | 2. Tech specs | 12-15-80 | Yes | No | Yes | 9-1-80 | Submitted |
| II.E.4.1 | Dedicated hydrogen penetrations | 1. Design | 1-1-80 | Yes | No | No | 1-1-80 | Complete |
| | | 2. Install | 7-1-81 | No | Yes | No | 7-1-81 | Submitted |
| II.E.4.2 | Containment isolation dependability | 1-4 Imp. diverse isolation | 1-1-80 | No | Yes | Yes | 1-1-80 | Complete |
| | | 5. Cntmt pressure setpoint | | | | | | |
| | | a. Specify pressure | 1-1-81 | No | Yes | No | 1-1-81 | Complete |
| | | b. Modifications | 7-1-81 | Yes | No | Yes | 1-1-81 | None identified |
| | | 6. Cntmt purge valves | 1-1-81 | No | Yes | Yes | 1-1-81 | Submitted |
| | | 7. Radiation signal on purge valves | 7-1-81 | No | Yes | Yes | 7-1-81 | Assume not applicable |
| | | 8. Tech specs | 12-15-80 | Yes | No | Yes | 9-1-80 | Submitted |
| II.F.1 | Accident-monitoring | 1. Noble gas monitor | 1-1-82 | No | Yes | Yes | 1-1-81 Submittal if deviation from position | Submitted |

| Clarification Item | Shortened Title | Description | Implementation Schedule | NRC Pre-Implementation Approval | NRC Post-Implementation Review Required | Tech Spec. Revision Required | Submittal Req. By | Vepco Remarks |
|---|---|---|---|---|---|---|---|---|
| II.F.1 | Accident-monitoring (continued) | 2. Iodine/particulate sampling | 1-1-82 | No | Yes | Yes | 1-1-81 submittal if deviation from position | Submitted |
| | | 3. Containment high-range monitor | 1-1-82 | No | Yes | Yes if deviation from position | 7-1-81 submittal | Submitted |
| | | 4. Containment pressure | 1-1-82 | No | Yes | Yes | 1-1-82 | Submitted |
| | | 5. Containment water level | 1-1-82 | No | Yes | Yes | 1-1-82 | Submitted |
| | | 6. Containment hydrogen | 1-1-82 | No | Yes | Yes | 1-1-82 | Submitted |
| II.F.2 | Instrumentation for detection of inadequate core cooling | 1. Subcool meter | 1-1-80 | No | Yes | Yes | 1-1-80 | Complete |
| | | 2. Tech spec (LL Cat A) | 12-15-80 | Yes | No | Yes | 9-1-80 | Submitted |
| | | 3. Install level instruments (LL Cat B) | 1-1-82 E | No | Yes | Yes | 1-1-81 Submittal if deviation from position | Submitted |
| II.G.1 | Power supplies for pressurizer relief valves, block valves and level indicators | 1. Upgrade to emerg sources | 1-1-80 | No | Yes | Yes | 1-1-80 | Complete |
| | | 2. Tech specs | 12-15-80 | Yes | No | Yes | 9-1-80 | Submitted |
| II.K.1 | IE Bulletins | 79-05, 06, 08 | Bulletin specific | No | Yes | No | Bulletin specific | NRR has evaluated Vepco responses |

| Clarification Item | Shortened Title | Description | Implementation Schedule | NRC Pre-Implementation Approval | NRC Post-Implementation Review Required | Tech Spec. Revision Required | Submittal Req. By | Vepco Remarks |
|---|---|---|---|---|---|---|---|---|
| II.K.2 | Orders on B&W plants | 13. Thermal mechanical report | 1-1-82 | No | Yes | As required | 1-1-82 | |
| | | 17. Voiding in RCS | b. 1-1-82 | No | Yes | No | 1-1-82 | |
| | | 19. Benchmark analysis of seq. AFW flow | b. 1-1-82 | No | Yes | No | 1-1-82 | |
| II.K.3 | Final recommendations, | 1. Auto PORV isolation | | | | | | Not required by II.K.3.2 |
| | B&O task force | 2. Report on PORV failures | 1-1-81 E | No | Yes | No | 1-1-81 E | Submitted |
| | | 3. Reporting SV & RV failures & challenges | 1-1-81 | No | Yes | Yes | 1-1-81 | Initiate data beginning 4-1-80 |
| | | 5. Auto trip of RCPs | | | | | | |
| | | a. Propose modifications | 7-1-81 | No | Yes | No | 2-15-81 | Submitted |
| | | b. Modify | 3-1-82 | Yes | No | Yes | 7-1-81 E | If required |
| | | 9. PID controller | 1-1-81 | No | Yes | No | 12-1-80 | Complete |
| | | 10. Proposed anticipatory trip modifications | Plant specifc | Yes | No | Yes | Plant specific | N.A. |
| | | 11. Justify use of certain PORV | Plant specific | No | Yes | No | Plant specific | N.A. |
| | | 12. Anticipatory trip on turbine trip | | | | | | |
| | | a. Confirmation or propose modifications | 1-1-81 | No | Yes | No | 1-1-81 | Complete |
| | | b. Modify | 1st refuel 6 mo after staff approval | Yes | No tech spec | Yes amend request | 1st refuel | N.A |
| | | 17. ECC system outages | 1-1-81 | No | Yes | As required | 1-1-81 | Submitted |
| | | 25. Power on pump seals | | | | | | |
| | | a. Propose mods | 1-1-82 | No | Yes | No | 1-1-82 | |
| | | b. Modifications | 7-1-82 | Yes | No | No | 7-1-82 | |

| Clarification Item | Shortened Title | Description | Implementation Schedule | NRC Pre-Implementation Approval | NRC Post-Implementation Review Required | Tech Spec. Revision Required | Submittal Req. By | Vepco Remarks |
|---|---|---|---|---|---|---|---|---|
| II.K.3 | Final recommendations, B&O task force (continued) | 30. SB LOCA methods | | | | | | |
| | | a. Schedule outline | 11-15-80 | No | Yes | No | 11-15-80 | Complete |
| | | b. Model | 1-1-82 | Yes | No | No | 1-1-82 | |
| | | c. New analyses | 1-1-83 or 1 yr after staff approval | Yes | No | No | 1-1-83 or 1 yr after staff approval | |
| | | 31. Compliance with CFR 50.46 | 1-1-83 or 1 yr after staff approval | Yes | No | TBD | 1-1-83 | |
| III.A.1.1 | Emergency preparedness, short-term | Short-term improvements | Complete | No | Yes | No | Complete | Complete |
| III.A.1.2 | Upgrade emergency support facilities | 1. Interim TSC OSC & EOF | Complete | No | Yes | No | Complete | Complete |
| | | 2. Design | 6-1-81 | Yes | No | No | 6-1-81 | |
| | | 3. Modifications | 10-1-82 | No | Yes | Yes | 10-1-82 | |
| III.A.2 | Emergency preparedness | 1. Upgrade emergency plans to App. E, 10 CFR 50 | 4-1-81 | No | Yes | Yes | 1-2-81 | Submitted |
| | | 2. Meteorological data | 6-1-83 | No | Yes | Yes | 1-2-81 | Staged implementation (E) |
| III.D.1.1 | Primary coolant outside containment | 1. Leak reduction | Complete | No | Yes | Yes | Complete | Complete |
| | | 2. Tech specs | 12-15-80 | Yes | No | Yes | 9-1-80 | Submitted |
| III.D.3.3 | Inplant radiation monitoring | 1. Provide means to determine presence of radioiodine | Complete | No | Yes | No | Complete | Complete |

| Clarification Item | Shortened Title | | Description | Implementation Schedule | NRC Pre-Implementation Approval | NRC Post-Implementation Review Required | Tech Spec. Revision Required | Submittal Req. By | Vepco Remarks |
|---|---|---|---|---|---|---|---|---|---|
| III.D.3.3 | Inplant radiation monitoring (continued) | 2. | Modifications to accurately measure $I_2$ | 1-1-81 | No | Yes | Yes | 1-1-81 | Complete |
| III.D.3.4 | Control-room habitability | 1. | Review | 1-1-81 E | No | Yes | No | 1-1-81 | |
| | | 2. | Modification | 1-1-83 | No | Yes | Yes | 1-1-81 | |

Note

E - Indicates those implementation and/or submittal dates to which Vepco has taken an exception.

## I.A.1.1 SHIFT TECHNICAL ADVISOR

1. Shift Technical Advisors have been on duty at North Anna Units 1 and 2 and Surry Units 1 and 2 since January 1, 1980. Details of the qualifications of currently assigned STA's are given on pages I.A.1.1-5 and I.A.1.1-6.

   Current Technical Specifications for both North Anna and Surry Power Stations require the presence of an STA on shift whenever either unit is in operation.

2. STA training per the requirements of Lessons Learned Category B was complete in December 1980 for both stations. This met the January 1, 1981 requirements. Details of the training conducted at each station are given on pages I.A.1.1-5 and I.A.1.1-6.

3. The long-term STA training program has been developed and was submitted for NRC Staff review on December 31, 1980. A copy of the submitted program is included in this section starting on page I.A.1.1-7.

# SHIFT TECHNICAL ADVISOR

# TRAINING PROGRAM

## SHIFT TECHNICAL ADVISOR TRAINING PROGRAM

The attached program compares favorably with the program proposed by INPO with regard to subject matter content of the education section, transient analysis and simulation; however, INPO's program is longer. STA's will be included in the VEPCO Licensed Operator Requalification Program to ensure that the STA's are provided with an understanding of operational concepts and philosophies in order that acceptance of the STA by operations personnel is enhanced. A comparison of time (contact hours) for various sections of the program follows:

| COLLEGE LEVEL FUNDAMENTAL EDUCATION | VEPCO | INPO |
|---|---|---|
| Math | 72 | 90 |
| Reactor Theory | 48 | 100 |
| Reactor Chemistry | 16 | 30 |
| Materials | 16 | 40 |
| Thermal Sciences | 40 | 120 |
| Health Physics | 16 | 40 |
| APPLIED FUNDAMENTALS | | |
| Reactor Technology | 32 | 120 |
| Instrumentation & Control | 32 | |
| Management Skills | 24 | 40 |
| Systems | 240 | 200 |
| Transient Analysis | 40 | 30 |
| Simulator Training | 80 | 100 |
| Retraining | 96 | 80 |

## SHIFT TECHNICAL ADVISOR
## ADMINISTRATIVE REQUIREMENTS

A. Education

Personnel selected to become Shift Technical Advisors should have as a minimum a bachelor degree in an engineering or scientific discipline. Other degree disciplines will be evaluated on a case-by-case basis.

B. Experience

Prospective Shift Technical Advisors shall have a minimum of six month's experience at the station at which they will assume the duties of Shift Technical Advisor.

C. Qualification

Shift Technical Advisors will be considered qualified to perform the duties of an STA after successful completion of the Shift Technical Advisor Training Program. Successful completion shall be based upon a cumulative average of 70%, or greater, on the written examinations plus "satisfactory" on the final oral/simulator evaluation.

D. Waiver

Participation of prospective Shift Technical Advisors in selected areas of the Shift Technical Advisor Training Program may be waived. The waiver will be on a case-by-case basis with the requirement that the individual attain a score of 70%, or greater, on an equivalency examination in the subject area for which the exemption is being sought. The equivalency examination shall be prepared and administered by the station Training Center Staff. In addition, the Manager-Nuclear Operations and Maintenance shall authorize the waiver based on the results of the equivalency examination and the recommendation of the Station Manager.

I. Academic

A. Mathematics (9 days)

1. Algebra

a. Basic concepts
b. Algebraic operations
c. Linear equations
d. Exponents and radicals
e. Roots

2. Logarithms

a. Definition
b. Rules of logarithms
c. Base 10 and natural logarithms
d. Exponential and logarithmic equations

3. Geometry and Trigonometry

a. Plane geometry
b. Solid geometry
c. Trigonometric functions
d. Vectors and vector operations

4. Specialized Mathematics

a. Ratios and proportions
b. Graphing techniques
c. Systems of equations

5. Differential Calculus

a. Limits and continuity
b. Derivatives
c. Applications of derivatives

6. Integral Calculus

a. Antiderivatives
b. Definite integrals
c. Applications of integrals

7. Differential Equations

a. First order
b. Applications of first order
c. Linear equations
d. Applications of linear equations

8. Partial Differential Equations

EXAMINATION - 1 DAY

B. Reactor Theory (6 days)

1. Basic Nuclear Physics

   a. Atomic structure
   b. Nuclear terminology
   c. Properties and structure of the nucleus
   d. Mass defect and binding energy

2. Nuclear Fission

   a. Microscopic and macroscopic cross sections
   b. Fission process
   c. Reaction rates and power production
   d. Fission products

3. Neutron Kinetics

   a. Neutron sources
   b. Reactions and cross sections
   c. Neutron flux
   d. Neutron moderation
   e. Neutron multiplication (including subcritical)
   f. Diffusion theory
   g. Criticality theory

4. Reactivity Coefficients and Poisons

   a. Fuel temperature coefficient
   b. Moderator temperature coefficient
   c. Void coefficient
   d. Pressure coefficient
   e. Redistribution coefficient
   f. Xenon
   g. Samarium
   h. Boron
   i. Other poisons (control rods, structural materials, etc.)

EXAMINATION - 1 DAY

C. Chemistry (2 days)

1. Purpose

2. Terminology

3. Sources of Makeup Water

   a. Flash Evaporator
   b. Polishing Demineralizer
   c. Condensate Storage Tanks

4. Secondary Chemistry

   a. Type of chemistry control
   b. Condensor leakage effects
   c. Steam Generator chemistry problems
   d. Steam Generator corrosion
   e. Secondary chemistry specifications

5. Primary Chemistry

   a. Chemical addition
   b. Demineralization
   c. Primary chemistry specifications
   d. Nuclear reactions

D. Materials (2 days)

1. Introduction

   a. Characteristics of an "Ideal metal"
   b. Failure modes of metals
   c. Metalic structure
   d. Radiation effects on metals

2. Fuel Element Design

3. Reactor Vessel Design

4. Steam Generator Design

EXAMINATION - 1 DAY

E. Thermal Sciences (5 days)

1. Thermodynamics

   a. Laws of Thermodynamics
   b. Properties of steam and water

2. Fluid Dynamics

   a. Fluid statics
   b. Bernoulli's Equation
   c. Static and dynamic pressures
   d. Laminar and turbulent flow
   e. Two phase flow

3. Heat Transfer

   a. Fundamentals (conduction, convection, radiation)
   b. Pool boiling
   c. Forced convection boiling
   d. Departure from Nucleate Boiling (DNB)
   e. Heat exchangers

4. Reactor Thermal Cycle

EXAMINATION - 1 DAY

F. Health Physics (2 days)

1. Radiation

a. Sources of radiation
b. Types of radiation
c. Effects of radiation on matter

2. Biological Effects

a. Micro (atomic) effects
b. Macro (cell) effects
c. Acute radiation effects
d. Chronic radiation effects

3. Radiation Protection

a. Time, distance, and shielding
b. Anti-contamination clothing
c. Respiratory protection

II. Applied Fundamentals

A. Reactor Technology (4 days)

1. Flux Distribution and Control

a. Axial flux distribution
b. Radial flux distribution
c. $\Delta\phi$ and axial offset
d. Hot channel factors
e. Control rods
f. Boron

2. Reactor Core

a. Core materials
b. Thermal performance
c. Core description
d. Core capability

3. Core Physics Data

a. Startup physics data
b. Core flux mapping
c. Core thermocouple mapping

4. Reactivity Procedures

a. Estimated Critical Position (ECP)
b. Shutdown Margin Calculation (OP-1F)

EXAMINATION - 1 DAY

B. Instrumentation and Control (4 days)

1. Basic Instrumentation

a. Flow measuring devices
b. Pressure measuring devices
c. Temperature measuring devices
d. Level/volume measuring devices
e. Miscellaneous measuring devices

2. Control Theory

a. Flow control circuits
b. Pressure control circuits
c. Temperature control circuits
d. Level/volume control circuits
e. Miscellaneous control circuits

EXAMINATION - 1 DAY

III. Management Skills

A. Communication (1 day)

B. Problem Solving (1 day)

C. Decisional Analysis (1 day)

IV. Systems Training (30 days)

A. Reactor Coolant System

B. Residual Heat Removal System

C. Chemical and Volume Control System

D. Nuclear Instrumentation System

1. Excore

2. Incore

E. $T_{ave}$/Rod Control

1. $\Delta T/T_{ave}$
2. $T_{ave}$ Control
3. Rod Control
4. Rod Position Indication

F. Reactor Protection and Control

1. Reactor Protection - General
2. Process Protection Instrumentation
3. Overpower/Overtemperature $\Delta T$ Protection
4. Pressurizer Pressure Control and Protection
5. Pressurizer Level Control and Protection
6. Steam Dump Control
7. Steam Generator Level Control and Protection

G. Secondary Systems

1. Main Steam System
2. Steam Generators
3. Main Feedwater System
4. Main Condensate System
5. Main Turbine
6. Turbine Control and Protection
7. Steam Generator Blowdown System
8. Steam Dumps
9. Circulating Water System
10. Service Water System

EXAMINATION - 1 DAY

H. Engineered Safeguards

1. Safety Injection System
2. Containment
3. Containment Vacuum System
4. Containment Spray System
5. Recirculation Spray System
6. Consequence Limiting Safeguards
7. Auxiliary Feedwater System

J. Electrical Distribution

1. Normal Distribution

   a. 4160 vac
   b. 480 vac
   c. Screenwell distribution
   d. Lighting

2. Emergency Distribution

   a. 4160 vac
   b. 480 vac
   c. Vital Busses
   d. DC distribution

K. Primary Support Systems

1. Component Cooling Water System
2. Chilled Component Cooling Water System
3. Charging Pump Cooling Water System
4. Primary Grade Water System
5. Liquid Waste System
6. Gaseous Waste System
7. Ventilation System
8. Core Cooling Monitor
9. PORV and Safety Valve System
10. Auxiliary Shutdown Panel
11. Status Lights
12. Loose Parts Monitoring System (North Anna only)

L. Radiation Monitoring

1. General Detector Curve

2. Radiation Monitoring System

EXAMINATION - 1 DAY

V. Accident/Transient Analysis (5 days)

A. Transient Analysis Methodology

B. Core Reload Design

C. Physics Related Safety Analysis Input

D. Core Thermal Hydraulic Analysis

E. System Transient Analysis

F. Non-LOCA Transient Analysis

1. Rod withdrawal from subcritical

2. Rod withdrawal at power

3. Dropped rod

4. Feedwater System malfunction

5. Excessive load increase

6. CVCS malfunction

7. Startup of inactive reactor coolant loop

8. Loss of flow

9. Single rod withdrawal

10. Loss of normal feedwater

11. Loss of offsite power

13. Feedline break

14. Steam Generator tube rupture

15. Inadvertant Safety Injection

16. Steamline break

17. Rod ejection

18. Locked rotor (fuel performance)

G. Loss of Coolant Accident

1. Large

2. Small

H. Best Estimate Transient Analysis

1. North Anna cooldown event

2. Surry pump coastdown

3. Simulator loss of load comparison

VI. Simulator Training (10 days)

A. Malfunctions

1. Single

a. Steam Generator tube rupture
b. Loss of rod control
c. Loss of feedwater
d. Reactor/turbine/generator trip
e. Small break LOCA
f. Large break LOCA
g. Loss of RCS pressure
h. Steam Generator level malfunction
i. Uncontrolled cooldown of RCS
j. Loss of flow
k. Loss of electrical bus (4160 vac)
l. Loss of RHR

2. Multiple Malfunctions
--to be determined based on the class's performance

3. Normal Operations

a. Reactor startup
b. Reactor shutdown
c. Power maneuvers

VII. COMPREHENSIVE EXAMINATION

A. Written - 1 day

B. Oral/Simulator - 1/2 day

## I.C.1 GUIDANCE FOR THE EVALUATION AND DEVELOPMENT OF PROCEDURES FOR TRANSIENTS AND ACCIDENTS

On December 15, 1980, the Westinghouse Owners Group submitted a detailed description of a program to comply with the requirements of I.C.1 for both Inadequate Core Cooling and Transients and Accidents (Ref: Letter No. OG-47). The submittal identified previous Owners Group submittals to the NRC, which are believed to comprise the bulk of the required information. Additional effort required to obtain full compliance with Item I.C.1 (with a proposed schedule for completion) was also identified. Additionally, on March 18, 1981, an update of the status of the Owners Group I.C.1 activities was submitted to the NRC (Ref: Letter No. OG-54). This approach was discussed during a November 12, 1980 meeting between Westinghouse Owners Group representatives at the NRC Staff, and is consistent with the alternate requirements on page I.C.1-4.

## I.C.5 PROCEDURES FOR FEEDBACK OF OPERATING EXPERIENCE TO PLANT STAFF

The operating experience assessment function has been implemented through both a system-level Safety Engineering and Control group and Safety Engineering staffs at each station. Procedures for the operation of the system organization, the North Anna SES and the Surry SES were in effect prior to January 1, 1981.

## I.C.6 GUIDANCE ON PROCEDURES FOR VERIFYING CORRECT PERFORMANCE OF OPERATING ACTIVITIES

Surry Units 1 and 2 and North Anna Units 1 and 2 have implemented a program to assure that an effective system of verifying the correct performance of operating activities is provided. Specific details of this program are as follows:

1. Shift Supervisor permission is required for performance of surveillance testing (periodic tests) and maintenance procedures.

2. The Shift Supervisor signs off on a system released for work or returned to service on maintenance orders.

3. Procedures are in place at Surry 1 and 2 to ensure that a second qualified person verifies the status of critical components. Two signatures are required for periodic tests. Efforts are underway at North Anna Units 1 and 2 to revise applicable procedures to require this second person verification. These revisions are scheduled to be completed by June 1, 1981. In the interim, the second person verification will be administratively required. Critical valve positions are controlled by the administrative key lock program. Verification is also made of proper light/control board switch assignment. The shift turnover checklists, implemented in accordance with 2.2.1.c of NUREG 0578, provide further verification, at each shift change, of the status of important plant parameters and equipment. The checklists require the signature of the oncoming and offgoing operator in each area, as well as the oncoming shift supervisor. There are no plans to implement a two man requirement for tagging of equipment.

4. The Shift Supervisor is informed of any change in plant status, and he relays this to the operators. The balance of plant operators report equipment status to the respective licensed control room operator on duty who relays this information to the Shift Supervisor. North Anna has implemented an Action Statement Status Log, which lists all equipment, covered by Technical Specifications, the status of which requires entry into an Action Statement. This log is reviewed by the Shift Supervisor periodically, and prior to any mode changes.

5. Equipment important to safety is returned to service by procedure (MOP). A periodic test or the MOP is used to verify operability when equipment is returned to service. Both procedures utilize a second person to ensure critical items are verified to be in proper alignment at Surry 1 and 2. North Anna procedures will be revised by June 1, 1981. In the interim, the second person verification will be administratively required.

## I.D.1 CONTROL ROOM DESIGN REVIEWS

Vepco has reviewed and provided comments on the draft document NUREG/CR-1580, "Human Engineering Guide to Control Room Evaluation".

North Anna Units 1 & 2 control room has been reviewed and modified to reflect the human engineering recommendations of the NRC presented in the Essex Corporation Report. Surry Units 1 & 2 control room has been reviewed to establish short-term improvements. Vepco will respond to the requirements of NUREG 0700 once they are issued.

## I.D.1 CONTROL ROOM DESIGN REVIEWS - EXCEPTIONS

The Licensee submittal date for I.D.1 in Enclosure I appears to incorrectly indicate 4/82 rather than TBD (to be determined upon issuance of NUREG 0700) in the requirements section.

1. The majority of electrical equipment in these systems are not qualified to meet the integrated radiation dose to which they would be exposed in processing and concentrating the highly radioactive water or gas.

2. There is extensive piping for the recovery systems throughout the auxiliary building. The resulting dose rate from all these systems operating simultaneously would severely limit access for the required operation. Shielding for the recovery system piping and components would be very difficult and in some cases may be impossible to install due to the arrangement of the piping and equipment.

B. Modifications - Shielding or Equipment Changes for Reduction of Personnel Exposure

As a result of the plant radiation and shielding review, we have identified additional shielding and plant modifications to reduce the personnel exposure and equipment irradiation qualification required by NUREG 0578 which is more conservative than subsequent clarifications (NUREG 0737 allows the shielding to be based on a 30-day average base).

North Anna Only

1. The post-accident hydrogen recombiner vault requires shielding modifications to limit radiation exposure to the operators at the vault while realigning and operating the recombiner, and to reduce the levels in the continuous occupancy areas.

2. Manual valves, located in high radiation zones, which must be operated to line up and operate the post-accident hydrogen recombiner and hydrogen analyzer will be replaced with environmentally qualified remotely operated valves, such as direct-acting solenoid valve or air-operated valves.

Surry Only

1. In order to automatically adjust cooling water temperature to charging pumps, automatic temperature control valves are being added to the service water lines in the charging pump cooling water subsystem.

2. Manual valves, located in high radiation zones, which must be operated to line up and operate the post-accident hydrogen analyzer will be replaced with environmentally qualified remotely operated valves, such as direct-acting solenoid valve or air-operated valves.

North Anna and Surry

1. Shielding of portions of the lines added as part of the new post-accident sampling system may be required.

2. Shielding for the Post-Accident Sampling Facility is required.

3. Although no access is required in the lower level auxiliary building or safeguards building to mitigate an accident, the drain system for the auxiliary building sump and the safeguards building sump will be modified such that these sumps can be pumped to the affected unit's containment instead of to the high or low level waste tanks. This would eliminate a significant potential source of activity in the basement of the auxiliary building.

4. Sampling procedures have been modified and temporary shielding employed to limit dose rates at the present sample facility.

5. Additional shielding, area relocation, or procedural modifications are being evaluated to limit radiation dose rates in the technical support center, the operational support center, the counting lab, and the security control center.

6. System modifications to permit interfacing with external process systems designed and shielded after the accident are being made. The external process system design would be based on the extent of the accident and would utilize the most current technology available at the time of the accident.

C. Modifications - Equipment Qualification

The evaluation of radiation environmental qualification of equipment is proceeding slowly because of the difficulty in obtaining vendor data on older plants. The mechanical equipment review for LOCA is complete for North Anna and Surry. Vepco has reported the results of the electrical equipment review in conjunction with the responses to I.E. Bulletin 79-01B. Any necessary modifications will be made as material becomes available.

In addition, Vepco has concluded its review of the radiation effects on mechanical equipment required to mitigate a high energy line break (HELB) for North Anna and Surry Power Stations. This review indicates that no equipment modifications are required to mitigate the HELB transient.

Replacement of, or shielding for, material with insufficient radiation resistance in the following equipment has been identified to date and is in progress as noted. See Section C for proposed installation scheduled. These materials meet the requirements of the FSAR but not the extended requirements of NUREG 0578, NUREG 0660, NUREG 0737 and IE Bulletin 79-01B.

1. Replacement Safeguards area ventilation fan motors have been received. (North Anna)

2. Stainless steel bearings for component cooling water and service water insert check valves to replace teflon lug and plate bearings have been ordered. (North Anna and Surry)

3. Replacement service water radiation monitor pump motors have been delivered to North Anna. New pumps and motors are being purchased for Surry.

4. Replacement mechanical seal bellows for the service water radiation monitor pumps have been ordered. (North Anna)

5. Additional shielding is being designed for the service water radiation monitors at North Anna. Relocation of the service water radiation monitors to a lower background radiation area is required at Surry.

6. Replacement O-rings in the high head safety injection pump seal cooler are being ordered. (Surry)

7. Valve seat replacements are on order for component cooling water valves to the reactor coolant pumps. (North Anna Unit 1)

8. Charging pump gaskets and mechanical seals on cooling water pump on charging pump are on order. (Surry)

9. Containment Isolation Valve Buna-N diaphragms will be replaced with qualified material during normal maintenance. (Surry)

10. Outside recirculation sprary pump plug valve seats (teflon) will be replaced at Surry.

11. Electrical equipment as identified in response to I.E. Bulletin 79-01B.

12. The low head safety injection and outside recirculation spray pumps will have Torlon replace Teflon for support pads. (North Anna and Surry)

D. Additional Information

The basis for systems excluded:

1. The design of the Reactor Coolant System (RCS) and supporting systems is such that the letdown portion of the Chemical and Volume Control System (CVCS) is not required to take the plant to a safe shutdown (hot standby) condition or mitigate the effects of a LOCA. The use of the CVCS letdown could create significant radiological problems. The letdown portion of the CSVS was not considered in this review for the following detailed reasons.

   At TMI-2, high airborne activity levels and high radiation levels outside the containment resulted from using systems which carry highly radioactive fluids from inside the containment to other buildings. One of the lessons learned from TMI-2 is to isolate from the containment all nonessential systems. This is also a requirement of NUREG 0737, Section II.E.4.2. The letdown and normal charging portions of the CVCS is automatically isolated by the phase A containment isolation signal. Use of the letdown portion of the CVCS presents the potential for increasing activity levels outside the containment. This portion of the CVCS is not required to mitigate an accident and this is kept isolated from the containment.

The CVCS letdown provides various functions during normal operation. First, it provides a method for the reduction of RCS water inventory. If the accident resulted from a ruptured pipe in the RCS, this function is not desired immediately. If, as at TMI-2, the integrity of the primary system is reestablished, other methods of reducing RCS inventory are available. The pressurizer PORV's and safety valves are designed to reduce pressure resulting from increased system water level.

The letdown system provides for control of chemistry and radioactivity (i.e. RCS hydrogen inventory, boric acid dilution and filtration, and ion exchange). Boric acid addition to ensure proper concentration in the RCS is provided by the Safety Injection System. Hydrogen addition is not required to maintain plant safety after an accident, as it is added only for long term corrosion control.

Degassing of the RCS is normally provided through the letdown portion of the CVCS. The installation of the RCS head and pressurizer venting system, required by NUREG 0578 and NUREG 0737, provides this function during accident conditions.

The CVCS letdown also provides some ability to clean up the RCS by means of filtration and demineralizaion. This function is not required nor desired during the mitigation phase of an accident. A large increase in the radiation levels in the CVCS would lead to high radioactivity levels outside the containment on resin beds and filters for which the system is not designed.

There are additional problems associated with using the CVCS letdown during an accident. Overpressure in the volume control tank, which could result from RCS degassing, is relieved automatically, first to the Boron Recovery System via the Vent and Drain System, then at a higher pressure to the liquid waste system. In addition, there is an automatic deverting valve, which sends RCS water to the Boron Recovery System if the level in the volume control tank gets too high. These parts of relief and automatic diversion would result in spreading radioactive fluids and gases in a large amount of piping and equipment throughout the Auxiliary building with the resulting increase in area dose rates, while deriving no substantial benefit.

Taking all these facts into consideration, we believe that the letdown portion of the CVCS should not be used during or after an accident and thus we have not considered that source in our shielding review.

2. The Residual Heat Removal (RHR) system was not considered in our shielding review. This system was not considered because all piping in this system is located inside the containment. Therefore, the source activity for the containment also includes any source derived from the RHR system inventory.

3. The Boron Recovery, Liquid Waste, Solid Waste, and Gaseous Waste systems were not considered in the shielding review for the following reasons:

   a. These systems were not designed or arranged to accommodate the activity levels that could be present after an accident, but rather to operate at the design conditions of one percent failed fuel as discussed in FSAR Section 11. The calculated activity concentration based on Regulatory Guide 1.4 and TID-14844 of the influent to the Liquid Waste or Boron Recovery system is approximately 2000 uci/cc, even after six months of radioactive decay.

      Thus, concentrated effluent in the evaporators of these systems would be so highly radioactive that shielding, processing and handling of the waste by conventional methods would not be possible. The area radiation dose rate from the concentrated waste and storage tanks would severely limit access to parts of the Auxiliary Building. It is proposed to keep this waste inside containment until the recovery phase when cleanup operations begin.

   b. Modifications are being designed to add connections to these and other systems to permit interfacing with an external waste processing system specifically selected for post-accident clean-up, during the recovery phase, of radioactive fluids and gases resulting from the accident.

Inclusion of all essential sources in the review:

All essential system piping and equipment required to mitigate the effects of a LOCA which contain or could contain highly radioactive fluids were considered as sources in our shielding review. These systems include; the High Head Safety Injection (HHSI) portions of the CVCS and SI Systems, the Low Head Safety Injection (LHSI) System, Recirculation Spray System, Sample System, and Containment Atmosphere Cleanup (hydrogen recombiner) System. In addition, other systems which are not required to mitigate a LOCA and are not required by NUREG 0737, but which could contain significant radioactivity, were considered such as drain lines and standing water in sumps and waste tanks. All branch connections to and from these systems were considered as sources to the first isolation valve. Other sources such as the shine from the containment dome, shine through containment penetrations, and shine through the personnel hatch were considered. The location of field run pipe, which is part of the systems listed above, was considered in our analysis. As noted in the response to the North Anna FSAR Comment 12.3, the routing and location of radioactive piping is such that the piping is in shielded areas. The exact routing of our field run pipe is not critical in the production of the radiation zone maps. The highest activity level in each zone is calculated and that level is considered for the entire zone. For instance, the highest activity may be 12 inches from a pipe, regardless of its exact location within the zone.

Indirect radiation was not considered as a source. Buildup factors in shield walls are considered but scatter over walls or through labyrinth doorways was not considered. Airborne activity was not considered as a source in our shielding review.

Consideration of all vital areas:

Vital areas for personnel exposure are defined as those areas which require continuous or frequent occupancy in order to control, monitor, and evaluate the accident. These areas include the Control Room, Technical Support Center, the Counting Lab/Health Physics area, the Operational Support Center, and Security Control Center. In addition, any area which requires access to perform manual operation of equipment in systems which are used to mitigate the accident were considered. Vital areas for equipment qualification inlcude all areas in which mitigating equipment is located. Zone maps have been developed for many non-vital areas as well. These areas include the entire Auxiliary Building, Main Steam Valve House, Quench Spray Pump House, Safeguards Building, Service Building, and selected areas in the yard.

a. Post-Accident Sampling Modification

The interim modifications made to the sampling system and the proposed long term modifications are designed to minimize the exposure to personnel during sampling using time, distance and shielding.

Shielding for portions of the interim sampling system lines has been installed. Shielding for portions of the long term post-accident sampling system lines and the sampling facility is required.

b. Technical Support Center

Sufficient shielding will be designed into the technical support center to limit personnel exposure to acceptable levels.

c. Operations Support Center (North Anna)

The Operations Support Center at North Anna will be located in an area with acceptable radiation dose rates.

d. Counting Laboratory

Shielding for some instruments in the counting laboratory or relocation of the counting equipment is necessary due to background activities.

e. Hydrogen Recombiner Modifications (North Anna Only)

The hydrogen recombiner system at North Anna is external to the containment. Because of the potential for a large dose contribution after post-accident operation modifications are being made.

The vault will be modified to provide a monitored release path for leakage from the hydrogen recombiner system to the Auxiliary ventilation system.

The post-accident hydrogen recombiner vault requires shielding modifications to limit radiation exposure to the operators at the vault while realigning and operating the recombiner and to reduce the levels in some of the vital areas. Relocation of the recombiner control panel is also required.

f. Containment Atmosphere Cleanup System

Manual valves, located in high radiation zones, which must be operated to line up and operate the post-accident recombiner at North Anna, and hydrogen analyzer, will be replaced with environmentally qualified, remote operated valves.

Section II.E.4.1 shows the proposed modifications in the valve arrangement for the containment atmosphere cleanup system. These modifications provide a double valve barrier between the accident unit's containment atmosphere which is being processed by either hydrogen recombiner, all other systems, and the unaffected containment. Remote operators will be provided for those valves where personnel access is restricted by post accident radiation levels.

g. Security Control Center

The emergency procedures will be revised to incorporate instruction on relocation of the security boundary if radiation doses in the yard are not acceptable.

In addition, Vepco has committed to implementation of the following modifications at North Anna Units 1 & 2 and Surry Units 1 & 2. These modifications are not required to validate the results of the shielding review or satisfy the requirements of NUREG 0578 but would reduce personnel exposure during the Recovery Phase.

- Waste cleanup system tie-ins (Additional design information in Attachment E to July 7, 1980 letter)

- Auxiliary building and safeguards building sump drain modifications (Design information provided in April 1, 1980 letter)

## II.B.2 DESIGN REVIEW OF PLANT SHIELDING AND EQUIPMENT QUALIFICATION - EXCEPTIONS

The shielding review is in compliance with or exceeds all the requirements of this Section except as noted below:

1. Not all systems listed in the clarification were assumed to contain the Accident Level Radiation Source Term as described in the "Basis for Systems Excluded" and "Inclusion of all Essential Sources in the Review".

2. In accordance with the section titled Changes to Previous Requirements and Guidance, Vepco will install remotely operated valves for the Containment Atmosphere Cleanup System for North Anna Power Station during the first sufficient outage but no later than July 1, 1982.

3. The replacement of recirculation spray valve liners and seals at Surry Unit 2 is scheduled for the next refueling outage (currently scheduled for the first quarter of 1982). This replacement requires extended access to the containment and vendor representatives to make the changes. This modification should be complete by the required date of July 1, 1982 unless the refueling is extended by unscheduled outages during which the vendor representatives would not be available.

4. The radiation zone maps for personnel access will be revised after all the post-TMI modifications are complete on January 1, 1982 to ensure the impact of all changes are incorporated. This revision will take a few months and will replace the radiation zone maps prepared by January 1, 1980 that identified the required modifications for personnel access.

An in-line chemical analysis panel is included to facilitate remote measurement of important chemical parameters with a minimum of manual action or exposure to the operator. This chemical analysis panel has the capability to measure primary coolant pH, boron, oxygen and hydrogen concentrations. The capability to measure chloride in-line at Surry is also provided. In-line chloride analysis is not planned for North Anna since it is a fresh water site. A diluted grab sample would be sent to Surry Power Station for chloride analysis using an Ion Chromatograph. Each parameter is either indicated or recorded on a remote control panel located in a lower background area of the station.

The HRSS sample panels will be located within existing space in the auxiliary building. The reactor coolant is drawn from existing sample system lines, outside of containment, upstream of the existing sample system coolers. Controls are provided to prevent post-accident samples from being inadvertently introduced to the existing sample room.

Specific design criteria from NUREG 0578 are addressed in this document as follows:

| NUREG 0578 Position | Design Input Requirements Attachment A-1 |
|---|---|
| Radiation exposure limit of 3 rem whole body and 18.75 rem extremity at t = 1 hour after the accident | Section 17.0 |
| Accident source term basis (Reg. Guide 1.4) | Section 4.0 |
| Consideration of operator exposure from sources external to sampling equipment | Section 17.0 |
| Analyses of boron and chloride | Section 13.0; Chloride analysis at Surry only, North Anna uses Surry equipment |

Sample liquid resulting from recirculation, purging, and drainage can be routed to a new HRSS Waste Tank from where the fluid can be pumped or displaced with nitrogen back to the containment sump. Connections are provided to recirculate, purge, and drain non-accident liquid samples via normal sample system flow paths for purposes of operator training and periodic equipment testing.

The containment atmosphere sample panel will have the capability to take suction from within the containment hydrogen analyzer system. Motive force for the HRSS containment atmosphere sample panel is provided by an integral nitrogen eductor. The discharge of the containment atmosphere panel will be routed back to the containment via the existing hydrogen monitor system piping.

f. Provides a ventilated cabinet held below atmospheric pressure to contain potential CASP system leakage. Cabinet ventilation is connected to the auxiliary building HVAC system.

3. Chemical Analysis Panel

The Chemical Analysis Panel (CAP) performs the following functions:

a. Accept a preconditioned, cooled, depressurized and degassed, liquid sample from the LSP for post-accident chemical analysis for boron, pH, dissolved hydrogen and dissolved oxygen, and hydrogen concentration in post-accident containment atmophere samples. (In addition, it will provide in-line chloride analysis of post-accident liquid samples for Surry.)

b. Provide remote readout of chemical analysis panel parameters on the remote HRSS Process Control Panel (PCP).

c. Provide an integrally shielded panel front to minimize post-accident operator dose rates.

d. Provide a ventilated cabinet held below atmospheric pressure to contain potential CAP subsystem leakage. Cabinet ventilation is connected to the auxiliary building HVAC system.

Table III-2 lists the type of instrumentation to be used for determination of post-accident chemical parameters. Instrumentation has been selected based upon the following criteria:

a. the ability to measure accurately the full anticipated range of parameters,

b. the ability to withstand high radiation fields,

c. the ability to reproduce results after calibration,

d. the ability to measure chemical parameters with small sample volumes.

The CAP is designed with built-in instrument calibration equipment. Instrument calibration will be performed by station personnel on a periodic basis to maintain the CAP in a ready condition and to minimize instrument drift.

4. Waste Tank, Pumps, and Evacuating Compressor

The Waste Tank and Pumps collect and return system purge and flush liquids to containment. The Waste Tank and Pumps will be bypassed during those periods of line purging when primary system pressure has sufficient motive force to return

2. Direct radiation from samples lines which are routed behind the shielded HRSS Sample and Analysis Panels.

3. Backscatter from the walls and roof behind and above the shielded HRSS Sample and Analysis panels.

The operator exposure from source (1) is limited by the stay time associated with sample panel manual operations and by selecting entrance and exit routes to the sample room via the lowest dose rate paths.

The operator exposure from source (2) is limited by the integral shielding located in the front of each of the system sample analysis panels (LPS, CAP, CASP). This shielding will consist of up to six inches of lead shot poured into panel front sections.

The operator exposure from source (3) is limited by positioning the panel in an orientation such that the distance from the back of the panel to the nearest wall is maximized to the greatest extent practicable. The shielding analysis indicates for the locations selected at North Anna and Surry, additional shield walls above the panels will not be required for this source.

For the worst case assumption of obtaining and analyzing a one-hour reactor coolant sample, the maximum operator exposure will be less than 2.5 rem whole body and 15 rem to the extremities.

## V. TEST AND INSPECTIONS

The HRSS is designed to be used under post-accident conditions and will not be used regularly during power operation, cooldown, and/or shutdown. Therefore, the system will be tested and maintained on a regular schedule to ensure all system components are in the ready condition. Station personnel will undergo regular training sessions to ensure good familiarity with the function and operation of the system. The Chemical Analysis Panel Instrumentation will be recalibrated and tested on a regular basis to ensure the accuracy and readiness of the instruments.

## VI. INSTRUMENT APPLICATION

The Chemical Analysis Panel (CAP) measured parameters will be indicated and recorded on the Remote Process Control Panel. Parameters to be measured are boron concentration, pH, dissolved oxygen, chloride (for Surry), dissolved hydrogen and containment air hydrogen concentration. Local flow and pressure indication will be on the face of the LSP, CASP, and CAP to enable the operator to manually align and adjust system flows.

The Process Control Panel will permit remote operation of all HRSS automatic valves including the existing inside containment sample system valves which are normally operated from a panel in the existing sample room.

Isotopic analysis of reactor coolant and containment atmosphere samples will be available within one hour of sample acquisition. The postulated activity concentration of post-accident samples is far in excess of the

capabilities of normal counting equipment and geometries. Thus, sample dilution will be required prior to analysis. The LSP provides a 1,000 to 1 dilution of reactor coolant samples. However, depending upon the accident condition, additional final dilution can be accomplished by the LSP, if necessary. The diluted sample can then be analyzed by existing laboratory counting equipment.

The LSP provides a shielded syringe sample of diluted reactor coolant gases which can also be further diluted, if necessary, in the adjacent shielded fume hood. These samples can then be analyzed in existing laboratory counting equipment.

The containment atmosphere samples are collected in 5 ml shielded sample bombs in the CASP. These samples will be isotopically analyzed by a Ge detector which measures through a one-fourth inch aperture in the sample vessel head shield. The 5 ml sample shield apertures will be designed to allow measurement in several orientations. Halides and noble gases can be analyzed together. Successive analyses of containment air samples collected on a known time sequence will enable the operator to determine the extent of the accident and the effectiveness of the containment spray system.

Table III-2
Chemical Analysis Panel Instrumentation

| Parameter | Instrument or Method | Range of Measurement |
|---|---|---|
| **Reactor Coolant and Containment Sump** | | |
| Boron | Selective ion electrode | 200-2,000 ppm |
| pH | Probe, Cole-Parmer or equal | 1-13 |
| Dissolved oxygen | Probe, Yellow Springs Instrument | 1-20 ppm |
| Dissolved hydrogen | Gas chromatograph, Baseline or equal | 10-2,000 cc/kg |
| Chloride (Surry only) | Ion chromatograph, Dionex or equal | 0- 5 ppm |
| **Containment Atmosphere** | | |
| Hydrogen | Gas chromatograph, Baseline or equal | 0-10 percent |

## II.B.4 TRAINING FOR MITIGATING CORE DAMAGE

1. A training program to teach the use of installed equipment and systems to control or mitigate accidents in which the core is severely damaged has been developed for use at both North Anna and Surry. The program is currently being revised to incorporate newly acquired data.

2. The program has already been implemented and taught at both stations. Attendance was required for all licensed operator, licensed senior operators and non-licensed operators. Personnel identified by the Emergency Plan qualified to become Emergency Directors are required to attend, as are all Shift Technical Advisors and Nuclear Training Coordinators. The training program has been incorporated into the operator training and retraining programs.

3. The program for managers and technicians in the Instrumentation and Control (I&C), health physics and chemistry departments has been implemented in conjunction with the Station Emergency Plan training on EPIP's specific to each technical area and will be complete prior to October 1, 1981 at both the Surry and North Anna Power Stations.

## II.D.1 PERFORMANCE TESTING OF BOILING-WATER REACTOR AND PRESSURIZED-WATER REACTOR RELIEF AND SAFETY VALVES (NUREG-0578, SECTION 2.1.2)

### Position

Pressurized-water reactor and boiling-water reactor licensees and applicants shall conduct testing to qualify the reactor coolant system relief and safety valves under expected operating conditions for design-basis transients and accidents.

### Changes to Previous Requirements and Guidance

A. Safety and Relief Valves and Piping--The types of documentation required for safety and relief valves and piping and the specific submittal dates are considered to be a clarification of item II.D.1 as described in NUREG-0660. The submittal of information was implied but not explicitly discussed in that report.

B. Block Valves--Qualification of PWR block valves is a new requirement. Since block valves must be qualified to ensure that a stuck-open relief valve can be isolated, thereby terminating a small loss-of-coolant accident due to a stuck-open relief valve. Isolation of a stuck-open power-operated relief valve (PORV) is not required to ensure safe plant shutdown. However isolation capability under all fluid conditions that could be experienced under operating and accident conditions will result in a reduction in the number of challenges to the emergency core-cooling system. Repeated unnecessary challenges to these system are undesirable.

C. ATWS Testing--Testing of anticipated transients without scram (ATWS) for later phases of the valve qualification program was noted in item II.D.1 of NUREG-0660. The clarification below provides updated information on PWR ATWS temperature and pressure conditions and clarifies that ATWS testing need not be accomplished by July 1981.

### Clarification

Licensees and applicants shall determine the expected valve operating conditions through the use of analyses of accidents and anticipated operational occurrences referenced in Regulatory Guide 1.70, Revision 2. The single failures applied to these analyses shall be chosen so that the dynamic forces on the safety and relief valves are maximized. Test pressures shall be the highest predicted by conventional safety analysis procedures. Reactor coolant system relief and safety valve qualification shall include qualification of associated control circuitry, piping, and supports, as well as the valves themselves.

A. Performance Testing of Relief and Safety Valves--The following information must be provided in report form by October 1, 1981:

(1) Evidence supported by test of safety and relief valve functionability for expected operating and accident (non-ATWS) conditions must be provided to NRC. The testing should demonstrate that the valves will open and reclose under the expected flow conditions.

(2) Since it is not planned to test all valves on all plants, each licensee must submit to NRC a correlation or other evidence to substantiate that the valves tested in the EPRI (Electric Power Research Institute) or other generic test program demonstrate the functionability of as-installed primary relief and safety valves. This correlation must show that the test conditions used are equivalent to expected operating and accident conditions as prescribed in the final safety analysis report (FSAR). The effect of as-built relief and safety valve discharge piping on valve operability must also be accounted for, if it is different from the generic test loop piping.

(3) Test data including criteria for success and failure of valves tested must be provided for NRC staff review and evaluation. These test data should include data that would permit plant-specific evaluation of discharge piping and supports that are not directly tested.

B. Qualification of PWR Block Valves--Although not specifically listed as a short-term lessons-learned requirement in NUREG-0578, qualification of PWR block valves is required by the NRC Task Action Plan NUREG-0660 under task item II.D.1. It is the understanding of the NRC that testing of several commonly used block valve designs is already included in the generic EPRI PWR safety and relief valve testing program to be completed by July 1, 1981. By means of this letter, NRC is establishing July 1, 1982 as the date for verification of block valve functionability. By July 1, 1982, each PWR licensee, for plants so equipped, should provide evidence supported by test that the block or isolation valves between the pressurizer and each power-operated relief valve can be operated, closed, and opened for all fluid conditions expected under operating and accident conditions.

C. ATWS Testing--Although ATWS testing need not be completed by July 1, 1981, the test facility should be designed to accommodate ATWS conditions of approximately 3200 to 3500 (Service Level C pressure limit) psi and 700°F with sufficient capacity to enable testing of relief and safety valves of the size and type used on operating pressurized-water reactors.

Applicability

This requirement applies to all operating reactors and operating license applicants.

Implementation

See implementation schedules in the "Documentation Required" section.

Type of Review

Preimplementation review will be performed for EPRI and BWR test programs with respect to qualification of relief and safety valves. Also, the applicants' proposal for functional testing or qualification of PWR valves will be reviewed.

Postimplementation review will also be performed of the test data and test results as applied to plant-specific situations.

Documentation Required

Preimplementation review will be based on EPRI, BWR, and applicant submittals with regard to the various test programs. These submittals should be made on a timely basis as noted below, to allow for adequate review and to ensure that the following valve qualification dates can be met:

Final PWR (EPRI) Test Program--July 1, 1980
Final BWR Test Program--October 1, 1980
Block Valve Qualification Program--January 1, 1981

Postimplementation review will be based on the applicants' plant-specific submittals for qualification of safety relief valves and block valves. To properly evaluate these plant-specific applications, the test data and results of the various programs will also be required by the following dates:

PWR (EPRI)/BWR Generic Test Program Results--July 1, 1981
Plant-specific submittals confirming adequacy of safety and relief valves based on licensee/applicant preliminary review of generic test program results--July 1, 1981
Plant-specific reports for safety and relief valve qualification--October 1, 1981
Plant-specific submittals for piping and support evaluations--January 1, 1982
Plant-specific submittals for block valve qualification--July 1, 1982

Technical Specification Changes Required

No technical specification changes are required.

References

NUREG-0578

NUREG-0660, Item II.D.1

## II.D.1 PERFORMANCE TESTING OF POWER RELIEF VALVE AND SAFETY VALVES

As a sponsor of the EPRI PWR Safety and Relief Valve Test Program, Vepco intends to comply with the requirements of NUREG 0578, Item 2.1.2. By letter dated December 15, 1980, R. C. Youngdahl of Consumers Power Company has provided the current PWR Utilities' positions of NUREG 0737, Item II.D.1 clarifications. Briefly, those positions are:

A. Safety and Relief Valves and Piping - the EPRI "Program Plan for Performance Testing of PWR Safety and Relief Valves", Revision 1, dated July 1, 1980, does provide a program that satisifes the NRC requirements. Discussion with the NRC staff and their consultants are resolving specific detailed issues.

B. Block Valves - The EPRI Program has not formally included the testing of block valves. However, a small number of block valves have been tested at the Marshall Steam Station Test Facility. The PWR Utilities and EPRI cannot provide a detailed block valve test program until results of the Wyle and CE relief valve tests are available. Therefore, a block valve test program will not be provided before July, 1981. The PWR Utilities and EPRI believe that the proper operation of the TMI-2, and Crystal River block valves and other operational experience, plus knowledge of the Marhsall tests, support a less hurried and more rational approach to block valve testing.

C. ATWS Testing - PWR Utilities will not support additional efforts for ATWS valve testing until regulatory issues are resolved. The major safety and relief valve test facility (CE) is nearing completion and some measures were taken to provide additional test capability beyond the current program requirements. The NRC should recognize that results from the current program are likely to provide most of the information necessary to address ATWS events (i.e., relief capability at high pressure).

## II.D.3 DIRECT INDICATION OF RELIEF AND SAFETY VALVE POSITION - EXCEPTIONS

The acoustical monitoring system is still undergoing seismic and environmental qualification by the vendor per an approved owner's group test plan. The test is scheduled for completion in the Summer of 1981.

- The automatic initiation signals and circuits shall be designed so that their failure will not result in the loss of manual capability to initiate the AFW system from the control room.

Response

Vepco has verified that the automatic start AFW signals and associated circuitry are safety grade. The AFW system is initiated automatically by a safety injection signal, a loss of offsite power, or a low-low steam generator level. These actuation signals are testable and these signals are the system actuations on which the FSAR Chapter 15 accident analysis is based. The AFW system is also automatically initiated on loss of the main feedwater pumps in anticipation of low steam generator level. This anticipatory actuation is not testable during normal operation. All initiation signals and circuits are designed to prevent a single failure from causing a loss of the AFW system.

ADDITIONAL SHORT-TERM RECOMMENDATIONS

The following additional short-term recommendations resulted from the staff's Lessons Learned Task Force review and the Bulletins and Orders Task Force review of AFW systems.

1. Recommendation

The licensee should provide redundant level indications and low level alarms in the control room for the AFW system primary water supply to allow the operator to anticipate the need to make up water or transfer to an alternate water supply and prevent a low pump suction pressure condition from occurring. The low level alarm setpoint should allow at least 20 minutes for operator action, assuming that the largest capacity AFW pump is operating.

Response

An additional level transmitter has been installed on the Emergency Condensate Storage Tank (ECST) for North Anna Units 1 and 2. This instrument provides the operator with both ECST level indication and a low level alarm in the main control room. The alarm is set to alert the operator of ECST low level at least twenty (20) minutes before the ECST could be emptied by the largest auxiliary feedwater (AFW) pump.

2. Recommendation

The licensee should perform a 72-hour test on all AFW system pumps, if such a test or continuous period of operation has not been accomplished to date. Following the 72-hour pump run, the pumps should be shut down and cooled down and then restarted and run for one hour. Test acceptance criteria should include demonstrating that the pumps remain within design limits with respect to bearing/bearing oil temperatures and vibration and that pump room ambient conditions (temperature, humidity) do not exceed environmental qualification limits for safety-related equipment in the room.

Vepco indicated to the staff that a potential conflict existed between the requirement for a 72-hour endurance test on the turbine driven AFW pump and the LCO in Technical Specification 3.7.1.2. The Additional Short-Term Recommendations No. 2 was then revised. The revision was issued as follows:

Revision to Recommendation No. 2 of "Additional Short-Term Recommendations" Regarding Auxiliary Feedwater Pump Endurance Test

The licensee should perform an endurance test on all AFW system pumps. The test should continue for at least 48 hours after achieving the following test conditions:

- Pump/driver operating at rated speed, and

- Pump developing rate discharge pressure and flow or some higher pressure at a reduced flow but not exceeding the pump vendor's maximum permitted discharge pressure value for a 48-hour test.

- For turbine drivers, steam temperature should be as close to normal operating steam temperature as practicable but in no case should the temperature be less than 400°F.

Following the 48-hour pump run, the pumps should be shut down and allowed to cool down until pump temperatures reduce to within 20°F of their values at the start of the 48-hour test and at least 8 hours have elapsed. Following the cool down, the pumps should be restarted and run for one hour. Test acceptance criteria should include demonstrating that the pumps remain within design limits with respect to bearing/bearing oil temperatures and vibration and that ambient pump room conditions (temperature, humidity) do not exceed environmental qualification limits for safety-related equipment in the room.

The licensee should provide a summary of the conditions and results of the tests. The summary should include the following: 1) A brief description of the test method (including flow schematic diagram) and how the test was instrumented (i.e., where and how bearing temperatures were measured), 2) A discussion of how the test conditions (pump flow, head, speed and steam temperature) compare to design operating conditions, 3) Plots of bearing/bearing oil temperature vs. time for each bearing of each AFW pump/driver demonstrating that temperature design limits were not exceeded, 4) A plot of pump room ambient temperature and humidity vs. time demonstrating that the pump room ambient conditions do not exceed environmental qualification limits for safety-related equipment in the room, 5) A statement confirming that the pump vibration did not exceed allowable limits during tests.

Response

The motor driven and turbine driven pumps have been endurance tested for North Anna Unit 2 and the motor driven pumps have been tested for Unit 1. These results have previously been transmitted

to the NRC. The turbine driven pump for Unit 1 has been tested and results will be submitted as soon as they are available.

3. Recommendation

The licensee should implement the following requirements as specified by Item 2.1.7.b on page A-32 of NUREG-0578:

> "Safety grade indication of auxiliary feedwater flow to each steam generator shall be provided in the control room.
>
> The auxiliary feedwater flow instrument channels shall be powered from the emergency buses consistent with satisfying the emergency power diversity requirements for the auxiliary feedwater system set forth in Auxiliary Systems Branch Technical Position 10-1 of the Standard Review Plan, Section 10.49."

Response

Modifications are complete for upgrading the safety-grade indication of AFW flow from semi-vital bus power to vital bus power.

4. Recommendation

Licensees with plants which require local manual realignment of valves to conduct periodic tests on one AFW system train and which have only one remaining AFW train available for operation, should propose Technical Specifications to provide that a dedicated individual who is in communication with the control room be stationed at the manual valves. Upon instruction from the control room, this operator would re-align the valves in the AFW system train from the test mode to its operational alignment.

Response

Periodic testing does not require local manual realignment of valves. Also, there are three AFW trains available. Therefore, no further action is required.

LONG-TERM

Long-term recommendations for improving the system are as follows:

1. Recommendation

GL-5 - The licensee should upgrade the AFW system automatic initiation signals and circuits to meet safety grade requirements.

Response

The AFW system automatic initiation signals and circuits are presently designed to meet safety grade requirements.

Item 3 of the Position corresponds to Enclosure 2 of the NRC Letter to Vepco dated September 28, 1979. These requirements can be found along with the corresponding response in Attachment A.

## II.E.1.1 AUXILIARY FEEDWATER SYSTEM EVALUATION SURRY UNITS 1 & 2

As stated in the clarification, the Staff reviewed Items 1 and 2, and issued letters to those plants that required the implementation of certain short and long-term AFW system upgrade requirements. This was the September 25, 1979 letter from the NRC to Vepco for Surry Units 1 and 2. Enclosure 1 of this letter listed those short and long-term requirements, while Enclosure 2 requested additional information concerning reevaluation of the AFW system flow rate design bases and criteria (Item 3 of Position). The short and long-term requirements are listed below with a description of the modifications implemented, as required.

### SHORT-TERM

1. Recommendation GS-1

   The licensee should propose modifications to the Technical Specifications to limit the time that one AFW system pump and its associated flow train and essential instrumentation can be inoperable. The outage time limit and subsequent action time should be as required in current Standard Technical Specifications; i.e., 72 hours and 12 hours, respectively.

   Response

   A modification to the Technical Specifications to provide limited conditions of operation of the Auxiliary Feedwater System have been submitted. This modification will limit the time that one AFW system pump and its associated flow train and essential instrumentation can be inoperable. The limits of the Standard Technical Specifications were utilized.

2. Recommendation GS-4

   Emergency procedures for transferring to alternate sources of AFW supply should be available to the plant operators. These procedures should include criteria to inform the operator when, and in what order, the transfer to alternate water sources should take place. The following cases should be covered by the procedures:

   - The case in which the primary water supply is not initially available. The procedures for this case should include any operator actions required to protect the AFW system pumps against self-damage before water flow is initiated; and,

   - The case in which the primary water supply is being depleted. The procedure for this case should provide for transfer to the alternate water sources prior to draining of the primary water supply.

   Response

   Procedure modifications have been made to provide operators with guidance to diagnose availability of the primary water supply and protect the AFW system pumps against self-damage before water flow

is initiated. The procedure also provides a prioritized list of alternate water sources and defines when and how to shift to the alternate sources as the primary source is depleted.

3. Recommendation GS-5

The as-built plant should be capable of providing the required AFW flow for at least two hours from one AFW pump train independent of any alternating current power source. If manual AFW system initiation or flow control is required following a complete loss of alternating current power, emergency procedures should be established for manually initiating and controlling the system under these conditions. Since the water for cooling of the lube oil for the turbine-driven pump bearings may be dependent on alternating current power, design or procedural changes shall be made to eliminate this dependency as soon as practicable. Until this is done, the emergency procedures should provide for an individual to be stationed at the turbine-driven pump in the event of the loss of all alternating current power to monitor pump bearing and/or lube oil temperatures. If necessary, this operator would operate the turbine-driven pump in an on-off mode until alternating current power is restored. Adequate lighting powered by direct current power sources and communications at local stations should also be provided if manual initiation and control of the AFW system is needed. (See Recommendation GL-3 for the longer-term resolution of this concern.)

Response

- Our primary water source provides sufficient water volume to insure we can supply 700 gpm for 2 hours; independent of any alternating current power source. No change is therefore required.

- Procedure modifications have been made to give the operator guidance for controlling feed to the steam generators manually and how the steam driven auxiliary feedwater pump can be manually started and controlled if necessary.

- Each of our Auxiliary Feedwater pumps is cooled by a flow path from its own discharge back to its suction, independent of any alternating current power. Therefore, no change is necessary.

- The requirement to station an operator or modify procedures in the event of the loss of all AC power to insure bearing cooling is also unnecessary, as the cooling water is provided by the individual pump.

- The concern for the operator to operate the turbine-driven pump in an on-off mode until alternating power is restored has been incorporated in the manual procedure as described in Item b above.

- Emergency (D.C.) lighting has been provided to give sufficient lighting to manually control the turbine-driven AFW pump and manually control pump discharge valves.

- Sound power phone communication is already available in the area of the AFW pumps and discharge valves. This circuit has been checked to be operable and head/hand sets have been provided to insure we have ready communication with the Control Room should it be necessary.

We feel one general comment is necessary. The probability of a total loss of all normal and emergency alternating current sources would seem very small. Only a series of failures without operator action would allow a complete loss of AC power. With the present capability to supply the affected unit's S/G's with auxiliary feedwater from the unaffected unit, it is extremely unlikely that we could deteriorate to having only steam driven auxiliary feedwater pumps available to both or either unit.

4. Recommendation GS-6

The licensee should confirm flow path availability of an AFW system flow train that has been out of service to perform periodic testing or maintenance as follows:

- Procedures should be implemented to require an operator to determine that the AFW system valves are properly aligned and a second operator to independently verify that the valves are properly aligned.

- The licensee should propose Technical Specifications to assure that prior to plant startup following an extended cold shutdown, a flow test would be performed to verify the normal flow path from the primary AFW system water source to the steam generators. The flow test should be conducted with AFW system valves in their normal alignment.

Response

- The periodic test (PT-15) for testing the operability of the Auxiliary Feedwater pumps have been modified to provide for a second operator to verify that the valves manipulated as part of the test are in proper alignment following the completion of the test.

- Our present start-up procedure, OP-1.4, provides for an actual flow verification of the Auxiliary Feedwater systems prior to taking the reactor critical. A change to our Technical Specifications has been proposed to require the flow test to verify normal flow path for the primary AFW system water source to the steam generators prior to plant start-up following an extended cold shutdown.

5. Recommendation GS-7

The licensee should verify that the automatic start AFW signals and associated circuitry are safety grade. If this cannot be verified, the AFW system automatic initiation system should be modified in the short-term to meet the functional requirements listed below. For the longer term, the automatic initiation signals and circuits should be upgraded to meet safety grade requirements as indicated in Recommendation GL-5.

- The design should provide for the automatic initiation of the auxiliary feedwater system flow.
- The automatic initiation signals and circuits should be designed so that a single failure will not result in the loss of auxiliary feedwater system function.
- Testability of the initiation signals and circuits shall be a feature of the design.
- The initiation signals and circuits should be powered from the emergency buses.
- Manual capability to initiate the auxiliary feedwater system from the control room should be retained and should be implemented so that a single failure in the manual circuits will not result in the loss of system function.
- The alternating current motor-driven pumps and valves in the auxiliary feedwater system should be included in the automatic actuation (simultaneous and/or sequential) of the loads to the emergency buses.
- The automatic initiation signals and circuits shall be designed so that their failure will not result in the loss of manual capability to initiate the AFW system from the control room.

Response

- The current design of the Auxiliary Feedwater System provides for automatic initiation.
- All initiation signals and circuits are designed to prevent a single failure from causing a loss of the Auxiliary Feedwater System.
- The Auxiliary Feedwater System is initiated automatically by a safety injection signal, loss of offsite power, and on low-low steam generator level of any one steam generator. These actuation signals are testable and these signals are the system actuations on which the FSAR Chapter 14 safety analysis is based. The Auxiliary Feedwater System is also automatically initiated on loss of the main feedwater pumps in anticipation of low steam generator level. This anticipatory actuation is not testable during normal operation.

- All initiating circuits which automatically start the Auxiliary Feedwater System, are powered from vital buses and are backed-up by the emergency power system.

- The capability presently exists to manually initiate the Auxiliary Feedwater System from the control room. A single failure in the manual circutis will not result in the loss of system function.

- The AC motor feed pumps in the Auxiliary Feedwater System are automatically initiated. The motor operated valves required to establish generators are left in the open position and also receive automatic signals. These valves are under strict administrative control and can be operated from the control room. The motor operated valves are powered from the emergency bus.

  The capability of cross-connecting auxiliary feedwater and supplying Auxiliary Feed from the unaffected unit has been installed at Surry. The valves receive automatic signals to open during certain steam rupture conditions. They are powered from the vital bus and are controlled manually from the control room. The same flow indications, individual steam generator isolation valve, and flow paths are utilized.

- The automatic signals are designed in such a manner that their failure will not result in the loss of manual capability to start the Auxiliary Feedwater System. The automatic initiation circuits are presently safety-grade equipment and meet the long-term requirements.

6. Recommendation

Procedures should be established to lock open and periodically verify open position of all manual AFWS valves inside containment.

Response

Our startup procedure (OP-1B) presently verifies the open position of all manual AFWS valves inside containment. In addition, our checklist (CL-53) which is done periodically, checks the Auxiliary Feedwater Pump Manual Discharge valves in the open position. Procedures have been modified to lock open the manual valves inside the containment.

7. Recomendation

The licensee should require staggering of the periodic pump train tests (e.g., one train at North Anna is tested every 10 days rather than all three trains tested at once on a monthly basis). This reduces the potential for inadvertently leaving closed the discharge valves of all trains after test.

Response

Our periodic testing has been staggered to test the motor driven and steam driven auxiliary feedwater pumps at different times to reduce the potential for inadvertently leaving closed the discharge valves of all trains after a test.

8. Recommendation

Emergency procedures should be available to the operators for operating the AFWS of one unit such that it is supplying water to the steam generator(s) of the opposite unit in the event that such an operating mode should be necessary.

Response

Emergency procedures have been modified to give operational guidance to utilize the other unit's AFWS should this condition become necessary.

ADDITIONAL SHORT-TERM RECOMMENDATIONS

The following additional short-term recommendations resulted from the staff's Lessons Learned Task Force review and the Bulletins and Orders Task Force review of AFW systems.

1. Recommendation

The licensee should provide redundant level indications and low level alarms in the control room for the AFW system primary water supply to allow the operator to anticipate the need to make up water or transfer to an alternate water supply and prevent a low pump suction pressure condition from occurring. The low level alarm setpoint should allow at least 20 minutes for operator action, assuming that the largest capacity AFW pump is operating.

Response

An additional safety-grade level transmitter has been installed on the Emergency Condensate Storage Tank (ECST). This instrument provides the operator with both ECST level indication and a low level alarm in the main control room. The alarm is set to alert the operator of ECST low level at least twenty (20) minutes before the ECST could be emptied by the largest auxiliary feedwater (AFW) pump. This modification should be fully implemented on Unit 1 during the steam generator replacement outage and on Unit 2 by the required implementation date. The existing level indication loop will be upgraded at the same time.

2. Recommendation

The licensee should perform a 72-hour test on all AFW system pumps, if such a test or continuous period of operation has not been accomplished to date. Following the 72-hour pump run, the

pumps should be shut down and cooled down and then restarted and run for one hour. Test acceptance criteria should include demonstrating that the pumps remain within design limits with respect to bearing/bearing oil temperatures and vibration and that pump room ambient conditions (temperature, humidity) do not exceed environmental qualification limits for safety-related equipment in the room.

The NRC, in a February 8, 1980 letter to Vepco, revised the pump endurance test requirements reducing the test from 72 hours to 48 hours. The revision was issued as follows:

Revision to Recommendation No. 2 of "Additional Short-Term Recommendations" Regarding Auxiliary Feedwater Pump Endurance Test

The licensee should perform an endurance test on all AFW system pumps. The test should continue for at least 48 hours after achieving the following test conditions:

- Pump/driver operating at rated speed, and
- Pump developing rate discharge pressure and flow or some higher pressure at a reduced flow but not exceeding the pump vendor's maximum permitted discharge pressure value for a 48-hour test.
- For turbine drivers, steam temperature should be as close to normal operating steam temperature as practicable but in no case should the temperature be less than 400°F.

Following the 48-hour pump run, the pumps should be shut down and allowed to cool down until pump temperatures reduce to within 20°F of their values at the start of the 48-hour test and at least 8 hours have elapsed. Following the cool down, the pumps should be restarted and run for one hour. Test acceptance criteria should include demonstrating that the pumps remain within design limits with respect to bearing/bearing oil temperatures and vibration and that ambient pump room conditions (temperature, humidity) do not exceed environmental qualification limits for safety-related equipment in the room.

The licensee should provide a summary of the conditions and results of the tests. The summary should include the following: 1) A brief description of the test method (including flow schematic diagram) and how the test was instrumted (i.e., where and how bearing temperatures were measured), 2) A discussion of how the test conditions (pump flow, head, speed and steam temperature) compare to design operating conditions, 3) Plots of bearing/bearing oil temperature vs. time for each bearing of each AFW pump/driver demonstrating that temperature design limits were not exceeded,

4) A plot of pump room ambient conditions do not exceed environmental qualification limits for safety-related equipment in the room, 5) A statement confirming that the pump vibration did not exceed allowable limits during tests.

Response

The motor driven and turbine driven pumps have been endurance tested for both Surry Units 1 and 2. These results have previously been previously transmitted to the NRC.

3. Recommendation

The licensee should implement the following requirements as specified by Item 2.1.7.b on page A-32 of NUREG-0578:

> "Safety grade indication of auxiliary feedwater flow to each steam generator shall be provided in the control room.
>
> The auxiliary feedwater flow instrument channels shall be powered from the emergency buses consistent with satisfying the emergency power diversity requirements for the auxiliary feedwater system set forth in Auxiliary Systems Branch Technical Position 10-1 of the Standard Review Plan, Section 10.49."

Response

To meet the diversity requirements of ASTB-10-1, the Auxiliary Feedwater Flow Indication power supplies have been moved to an existing cabinet which meets the diversity requirements.

4. Recommendation

Licensees with plants which require local manual realignment of valves to conduct periodic tests on one AFW system train and which have only one remaining AFW train available for operation, should propose Technical Specifications to provide that a dedicated individual who is in communication with the control room be stationed at the manual valves. Upon instruction from the control room, this operator would re-align the valves in the AFW system train from the test mode to its operational alignment.

Response

Our plant does not require local manual realignment of valves to conduct periodic tests on the AFW system. In the event that a periodic test is performed with only one AFW system train available, a dedicated individual who is in communication with the Control Room will be stationed at the manual valves and upon instruction from the Control Room, would re-align the AFW valves to their operational alignment. To insure this, a precaution has been added to the Periodic Test Procedure.

LONG-TERM

Long-term recommendations for improving the system are as follows:

1. Recommendation

GL-3 - At least one AFW system pump and its associated flow path and essential instrumentation should automatically initiate AFW system flow and be capable of being operated independently of any alternating current power source for at least two hours. Conversion of direct power to alternating current is acceptable.

Response

Our present design has at least one AFW system pump and flow path which is automatically initiated upon a loss of all AC power. However, in order to control the amount of flow delivered, the pump must be manually operated in an on-off mode. In order to provide automatic, or in this case, remote on-off operation from the control room, a design modification must be made. This modification should be completed by January 1, 1982.

2. Recommendation

GL-5 - The license should upgrade the AFW system automatic initiation signals and circuits to meet safety-grade requirements.

Response

The AFW system automatic initiation signals and circuits are presently designed to meet safety grade requirements.

3. Recommendation

The AFWS flow control valves for both the motor and turbine pump trains are AC powered, normally open, fail as-is motor operated valves which are located inside containment. Also, manual, normally open valves are located inside containment. The AFW design should be reevaluated, including the possibility of relocating the valves outside containment, assuming an accident inside containment which necessitates AFWS operation and which creates a containment environment (humidity, radiation) that precludes access to the valves. The reevaluation should consider the following:

a. A possible common mode failure (environmentally induced) casuing spurious closure or failure of the MOV's in a throttled position.

b. An AFWS line break downstream of the MOV's and failure of the MOV's to operate.

Response

As a result of Vepco's I.E. Bulletin 79-01B Review for Surry Units 1 and 2, it has been determined that the AFW flow control valves have environmental qualification data requiring further corroborating information from the manufacturers before final judgement can be made.

These valves are of the same general type and were purchased during the same time period as those which have been determined to be qualified. We do not foresee any problem in having these components qualified for nuclear use. Futhermore, these components will have performed their accident safety function prior to receiving a radiation dose above the threshold of 2500 rads.

Since we did not receive vendor qualification documentation in time for the December 1, 1980 update to I.E. Bulletin 79-01B, we have placed a purchase order for replacement operators. If in the interim, prior to installation, we receive substantiation of qualification for the presently installed operators, the existing operators will not be replaced. Therefore, only those installed operators that are proven not to be qualified will be replaced during the first outage of sufficient duration upon receipt of material.

Item 3 of the Position corresponds to Enclosure 2 of the NRC letter to Vepco dated September 25, 1979. These requirements along with the corresponding response can be found in Attachment B.

8. Requirement

IEEE 279-1971 Paragraph 4.10 Capability for Test and Calibration

Response

FSAR Section 7.2.1.7 article titled "Capability for Test and Calibration" and FSAR Section 7.5.3 titled "Engineered Safeguards Calibration and Test" both contain discussions on the Capability for Test and Calibration. The letter article states: "The engineered safeguards actuation channel are designed with sufficient redundancy to provide the capability for channel calibration and test during power operation."

Also, automatic initiation of auxiliary feed or loss of main feed pumps and loss of reserve station power is tested each refueling. The tests are defined by Procedure EMP-P-RT-20 and EMP-P-RT-30.

9. Requirement

IEEE 279-1971 Paragraph 4.11 Channel Bypass

Response

FSAR Section 7.2.1.7 article titled "Channel Bypass on Removal from Operation" contains supporting statements to Paragraph 4.11 of IEEE 279-1971.

10. Requirement

IEEE 279-1971 Paragraph 4.12 Operating Bypasses

Response

FSAR Section 7.2.1.7 article "Completion of Protective Action (Interlock)" states: "Where operating requirements necessitate automatic or manual bypass of a protective function, the design is such that the bypass is removed automatically whenever permissive conditions are not met. Devices used to achieve automatic removal of the bypass of a protective function are a part of the protective system and are designed in accordance with the criteria of this section". These statements directly comply with Paragraph 4.12 of IEEE 279-1971.

11. Requirement

IEEE 279-1971 Paragraph 4.13 Indication of Bypasses

FSAR Section 7.2.1.7 article "Information Readout and Indication of Bypass" states: "Indication is provided in the control room if some part of the system has been administratively bypassed or taken out of service." This statement directly complies with IEEE 279-1971 Paragraph 4.13.

Part 2: Auxiliary Feedwater System Flowrate Indiciation
Surry Units 1 & 2

The Auxiliary Feedwater System Flowrate Indication requirements are consistent with those set forth in General Design Criterion 13 to provide the capability in the control room to ascertain the actual performance of the AFWS when it is called to perform its intended function. Those requirements for Westinghouse and Combustion Enginering Plants, as found in the clarification, are listed below with the corresponding response.

1. Requirement

To satisfy these requirements, Westinghouse and CE plants must provide, as a minimum, one auxiliary feedwater flowrate indicator and one wide-range steam generator level indicator for each steam generator or two flowrate indicators.

Response

One auxiliary feedwater flowrate indicator and one wide-range steam generator lever indicator is provided for each steam generator on both Surry Units 1 and 2's main control boards.

2. Requirement

The flow indication system should be environmentally qualified.

Response

Vepco is presently participating in a transmitter Qualification Program in response to IE Bulletin 79-01B, which is scheduled to be completed by the Spring of 1982. Upon successful completion of the test program and receipt of materials, these transmitters will be replaced at the first outage of sufficient duration. Transmitters which fall into this category for auxiliary feedwater flow indication are FT-FW-100A, B, C for Surry Unit 1 and FT-FW-200A, B, C for Surry Unit 2.

Flow indicators on the main control board (FI-FW-100A, B, & C and FI-FW-200A, B, C) are type Westinghouse VX-252. In a letter to Vepco from Westinghouse dated September 2, 1980, Westinghouse stated that their VX-252 instruments are Class IE and meet the intent of the NRC programs including their Directive 79-01B released January 14, 1980.

Since the control room will not be subject to a change in environment during accident conditions (including accidents requiring the auxiliary feedwater system), the flow indicators are environmentally qualified.

3. Requirement

The flow indication system should be powered from highly reliable, battery backed non-Class IE power source.

Response

Auxiliary Feedwater Flow Indication is powered from a highly reliable, battery backed non-class IE power supply when the steam generator wide range level is considered a back-up indication of auxiliary feed flow. Feed flow and wide range steam generator level indication are powered from diverse class IE buses, with at least one of the buses backed by batteries for each steam generator.

4. Requirement

The flow indication system should be periodically testable.

Response

The capability of periodic testing of the auxiliary feedwater flow indication system exists presently at Surry Units 1 and 2.

Periodic tests to cover the auxiliary feedwater indication system are currently being written. When complete, they will be located in the station's two series of periodic testing.

5. Requirement

The flow indication system should be part of the plants Quality Assurance program.

Response

Auxiliary feedwater flow indication is covered and discussed in Section 2 of Vepco Nuclear Power Station Quality Assurance Manual.

6. Requirement

The flow indication system should be capable of display on demand.

Response

The auxiliary feedwater flow indication system continuously displays auxiliary feedwater flowrate on the main control board for each steam generator. Since the capability to instantly display any auxiliary feedwater flow is always present, the flow indication system is capable of display on demand.

7. Requirement

It is important that the displays and controls added to the control room as a result of this requirement not increase the potential for operator error. A human-factor analysis should be performed taking into consideration:

a. the use of this information by an operator during both normal and abnormal plant conditions,

## II.E.4.1 DEDICATED HYDROGEN PENETRATIONS

In a letter dated March 5, 1981 (Serial No. 142), Vepco provided clarification of our response to the requirements of Item II.E.4.1. In brief, this letter states that the requirements of II.E.4.1 are met for North Anna and are not applicable to Surry. Additional information is provided in the following response on the current plant configurations including information on proposed modifications to enhance to system capability. These modifications are not required to meet the requirements of II.E.4.1 and are provided as supplemental information.

### North Anna Design

The original North Anna design uses redundant external Hydrogen Recombiners shared between Units 1 and 2. The Hydrogen Recombiner line takes suction from the same penetration used for the suction of the Containment Vacuum pumps, the Hydrogen Purge lines and the Hydrogen Analyzers. Since radioactive gases could be flowing through these penetrations during the post-accident mode, these systems were considered to become extensions of containment.

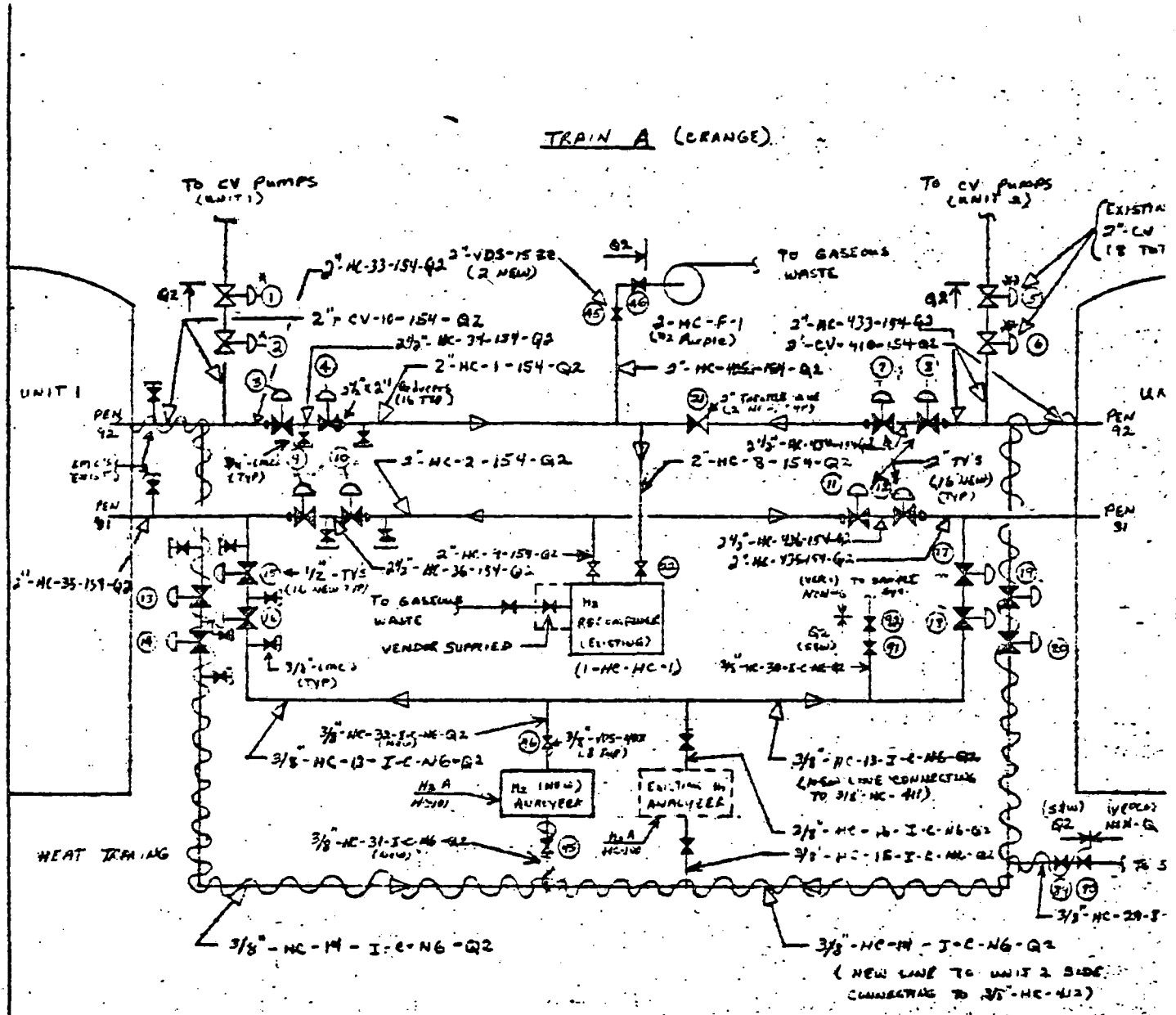
The discharge line from the hydrogen recombiner shares the same penetration with the discharge line from the hydrogen analyzer. Containment isolation is provided by a check valve inside containment and two remote manual valves outside containment. The combined hydrogen recombiner suction and discharge line is sized such that the flow requirements for the use of the combustible gas control system are satisfied.

The attached basic flow diagram shows the modified containment atmosphere clean-up system layout. The modification shown will enable the control room operator to line up the system using remotely operated valves and establish flow from the containment to the hydrogen analyzer and the hydrogen recombiner without exposure to high radiation. This modification resulted from the shielding review conducted as part of Item II.B.2 and is included here for completeness. The modified system will retain the design basis for single failure criteria. Valves will be added to the existing valves relocated in the suction of the containment atmosphere purge blower to provide double barrier isolation. Also, the Post-Accident Sampling System, Containment Vacuum & Gaseous Waste System connections will be provided with double isolation valves.

The System under normal plant operations will be lined up with all containment isolation valves closed except the containment isolation valves for the Containment Vacuum system which are periodically opened during vacuum pump operation.

Under accident conditions, the Containment Vacuum isolation valves will receive a containment isolation Phase "A" signal to close. The operator will open from the control room either the "A" or "B" train isolation valves to place the hydrogen recombiner and/or hydrogen analyzer in operation. The non-accident unit as well as all tie-in systems (Containment Vacuum, Post-Accident Sampling and Gaseous Waste) will be isolated from the containment atmosphere by a double valve barrier.

The backup Hydrogen Purge system is presently isolated from the hydrogen analyzers and recombiners by an administratively locked closed valve. This system is not operated during normal plant operations. Its use would only be comtemplated if both hydrogen recombiners fail.




NOTES

* VALVES CLOSE OR ASSURE CLOSE ON ϕA ISOLATION FROM UNIT 1

** VALVES CLOSE OR ASSURE CLOSE ON ϕA ISOLATION FROM UNIT 2

EXISTING MANUAL VALVES FOR POSSIBLE REUSE

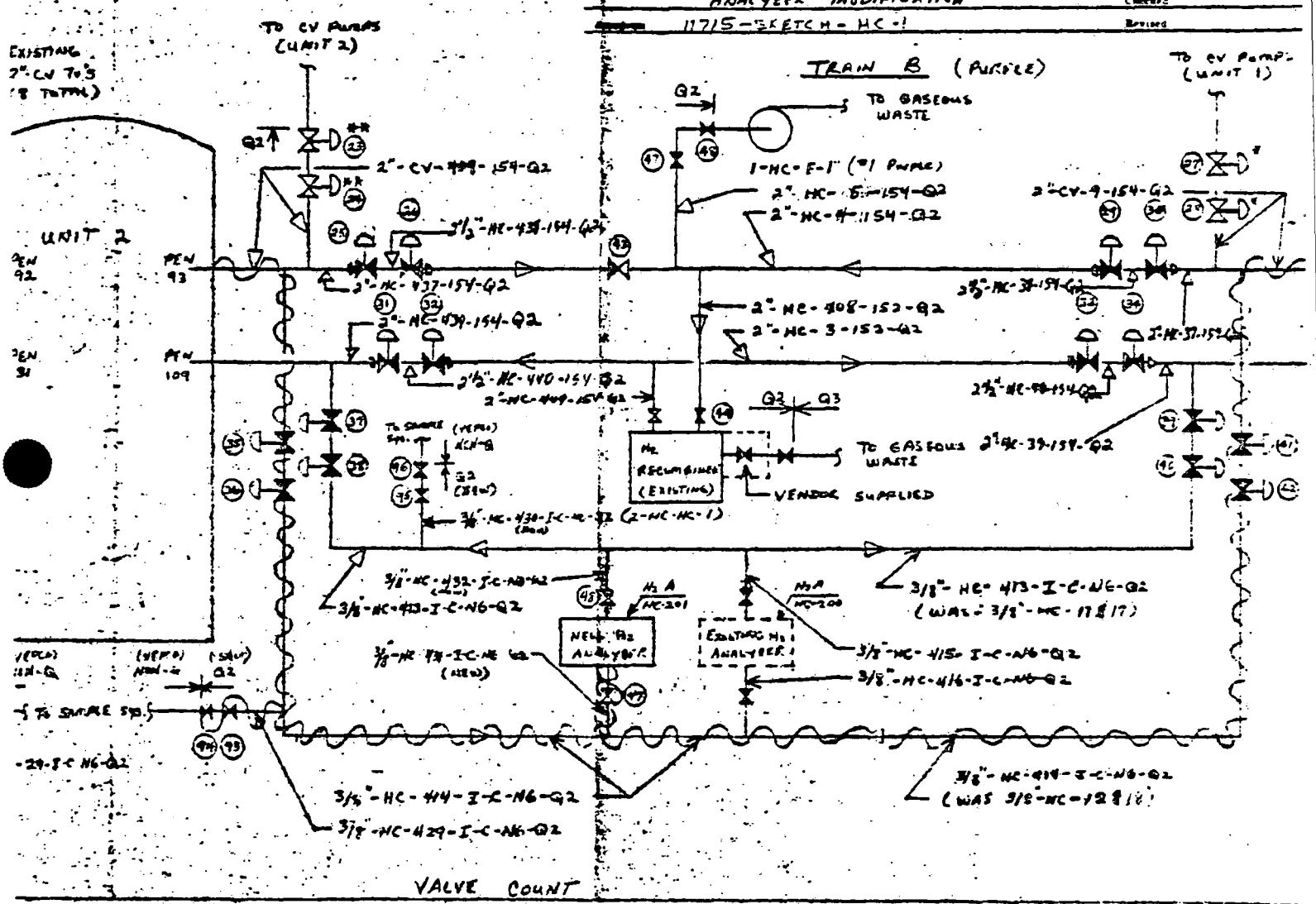
- 25 - 3/8" - VDS-402
- 8 - 3/4" - VDS-1532
- 8 - 2" - VDS-1522

EXISTING VI
FOR POSSIBLE
28 3/8"-VDS-4
8 3/4"-VDS-15
8 2"-VDS-15

North Anna

STONE & WEBSTER ENGINEERING CORPORATION
CALCULATION SHEET

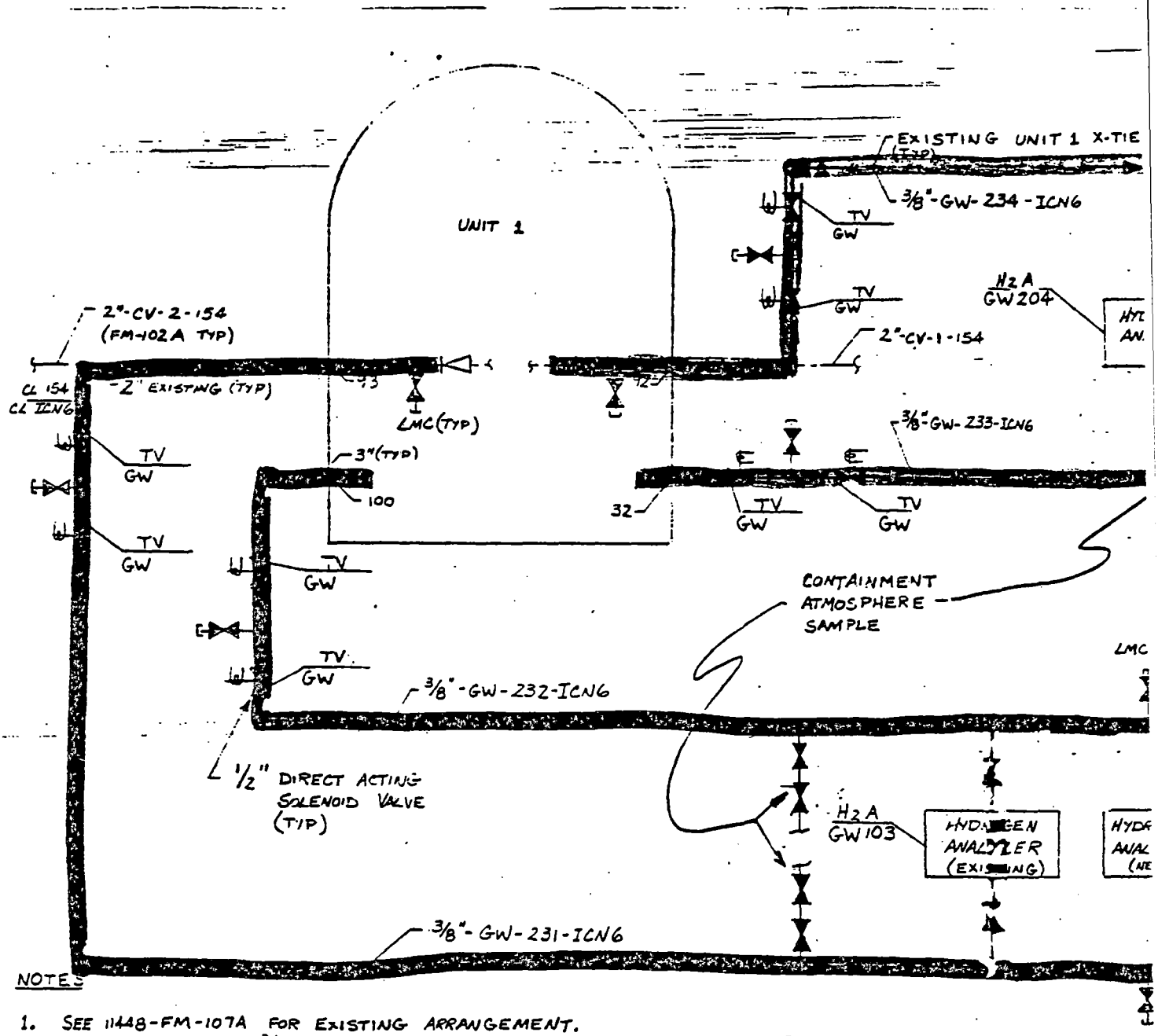
Client VEPCO Location NA
Subject or Application $H_2$ RECOMBINER & ANALYZER MODIFICATION
Calculation 11715-SKETCH-HC-1




## VALVE COUNT

| ~6 VALVES ARE REUSE | NEW VALVES REQD. | | | ANA. VALVES ON HAND | | |
|---|---|---|---|---|---|---|
| 26-40X | 16 | 3/4" | VDS-152; LMC'S | 12 | 3/8" - VDS-402 | 10/23 |
| 26-15Z | 16 | 1/2" | CONTAINMENT ISO. VALVES, DIRECT ACTING SELENOID TYPE | 24 | 3/8" - VOM-500T | 12/23 |
| 26-152Z | 16 | 2 1/2" | CONTAINMENT ISO. VALVES, AIR OPERATED TRIP VALVES | 18 | 3/4" - VDS-152 | 8/16 |
| | 12 | 3/8" | VDS-402; 4 $H_2$ ANALYZER ISO.; 8 SAMPLE ISO.; 4 LMC'S | 2 | 2" - VDS-152Z | 7/... |
| | 2 | 2" | VDS-152P; PURGE BLOWER ISO. | 2 | 2" - VDS-60... | A/... |
| | 2 | 2" | VDS-60E; THROTTLE VALVES $H_2$ RECOMBINER UNIT 2 SUPPLY LINE. | | | |
| | 24 | 3/8" | VOM-500T (OR 3/8"-VDS-402 IF AVAIL.); LMC'S | | | |

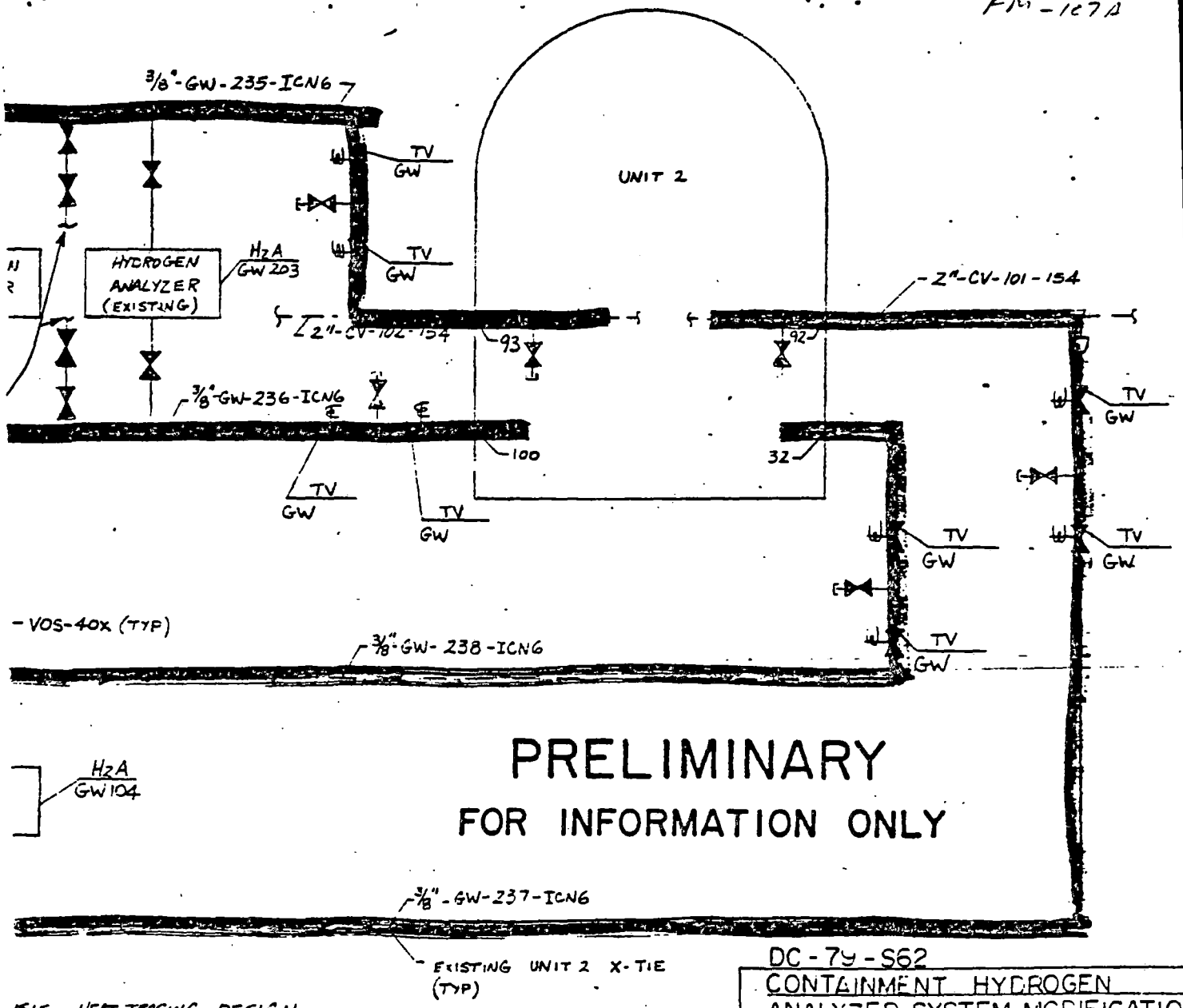
Sorry,




1. SEE 11448-FM-107A FOR EXISTING ARRANGEMENT.
2. EXISTING 2 IN. AND 3/4 IN. CLASS 154 PIPING WILL BE REPLACED WITH 3/8 IN. ICN6 TUBING TO EASE SEISMIC AND INSTALLATION OF NEW SOLENOID VALVES.
3. VALVES: 16 - 1/2" DIRECT ACTING SOLENOID VALVES (NEW)
   12 - 3/8" VOS-40X FOR LMC (NEW)
   8 - 1/2" OR 3/8" $H_2$ ANALYZER STOP VALVES (EXISTING)

Sorry

Fig - 107A




SIS, HEAT TRACING DESIGN

DC-79-S62

CONTAINMENT HYDROGEN ANALYZER SYSTEM MODIFICATION

VEPCO SURRY UNITS 1&2

STONE & WEBSTER ENGINEERING CORPORATION

12846.46-SK-1-

## II.E.4.1 DEDICATED HYDROGEN PENETRATIONS - EXCEPTIONS

The remotely operated valves will be operated from new Post-Accident Monitoring and Control Panels located in the control room. These panels are undergoing final testing prior to shipment to North Anna and Surry. The valves are scheduled to be shipped in January, 1981. Any delay in these shipments would impact Vepco's ability to install the system since "back-to-back" outages for Units 1 and 2, such as those scheduled for the Spring, are required for installation.

## II.F.1 ATTACHMENT 1, NOBLE GAS EFFLUENT MONITOR

A. Process vent and ventilation vent monitors

1. Refer to item 4 below for the upper range capacity.

2. The vent effluent monitors have a total range of concentration extending from a minimum design condition of $10^{-7}$ uci/cc to a maximum of $10^{5}$ uci/cc - $Xe^{133}$ (Table II.F.1-1 requires a maximum of only $10^{3}$ uci/cc). Three detectors are required to cover this range and have an expected minimum range overlap of a factor of ten.

B. Main steam Safety Valve (SV), Atmospheric Dump Valves (ADV) and auxiliary Feedwater Pump Turbine (FWPT) exhaust monitors

1. Refer to item 4 below for upper range capacity.

2. The Main Steam SV & ADV monitors have a total range of concentration extending from a minimum design condition of $10^{-1}$ uci/cc to a maximum of $10^{7}$ uci/cc ($Xe^{133}$ equivalent dose). Three detectors are required to cover this range and have a range overlap.

   The Auxiliary FWPT exhaust monitors will detect effluent concentrations of $10^{-1}$ uci/cc to $10^{+3}$ uci/cc ($Xe^{133}$ equivalent dose) as a minimum. The actual range is dependent on the final design.

1. The gaseous effluent monitors meet the following requirements specified in Table II.F.1-1:

   a. The following potential accident release paths are monitored:

      - Process vent stack
      - Ventilation vent stack(s)
      - Main steam safety valve and atmospheric dump valve discharge
      - Auxiliary feedwater pump turbine (FWPT) exhaust

   b. Design basis maximum range (requirements - $10^{+3}$ uci/cc) - Refer to item 4 for each effluent monitor range and radionuclide spectrum distribution.

   c. Redundancy - None

   d. The offline monitors (process vent and ventilation vents) sampling design criteria will be per ANSI N13.1-1969.

   e. The effluent monitors in item (a) will be powered by the emergency power supply system.

f. The following calibration sources will be used:

i) Process vent and ventilation vent monitors

Beta Scintillation Detector - $Ce^{136}$

GM Detector - $Co^{60}$

ii) Main steam and auxiliary FWPT exhaust monitors

GM Detector - $Cs^{137}$

Ion Chamber Detector - $Cs^{137}$

g. A continuous display with recording capability is provided.

i) Process vent and ventilation vent monitors are digital based systems which contain the following information on demand:

30 values of ten-minute averages for radiation, uci/cc ($Xe^{133}$)
30 values of one-hour averages for radiation, uci/cc ($Xe^{133}$)
30 values of one-day averages for radiation, uci/cc ($Xe^{133}$)
30 values of ten-minute averages for integrated release
30 values of one-hour averages for integrated release
30 values of one-day averages for integrated release

Also, radiation readings will be recorded by analog strip chart recorders.

ii) Main Steam SV & ADV and Auxiliary FWPT exhaust monitors will provide continuously recorded radiation readings by analog strip chart recorders.

h. The effluent monitors are designed to operate in the following maximum predicted environmental conditions:

(i) Main Steam/Aux. FWPT Exhaust Monitors

| | | |
|---|---|---|
| Temperature: | Detector | 40° to 167° F |
| | Control Unit | 70° to 80°F |
| Humidity: | Detector | 45° to 95% RH (non-condensing) |
| | Control Unit | 40 to 60% RH (non-condensing) |
| Radiation: | Detector* | $2 \times 10^4$ Rads integrated (background .75mR/hr to 200 R/hr |
| | Control Unit | $1 \times 10^3$ Rads integrated |

(ii) Process Vent and Ventilation Vent Effluent Monitors

| | | |
|---|---|---|
| Temperature: | Sampler/Detector | 55° to 120°F |
| | Micro Computer | 55° to 120°F |
| | Control Unit | 70° to 80°F |

| | | |
|---|---|---|
| Humidity: | Sampler/Detector | 45 to 95% RH (N.C.) |
| | Micro Computer | 45 to 95% RH (N.C.) |
| | Control Unit | 40 to 60% RH (N.C.) |
| Radiation: | Sampler/Detector* | $10^3$Rads integrated (background .75 to 600 mR/hr |
| | Control Unit | $10^3$ Rads integrated |

*Note: Refer to the "System Description", Item (4) for the affect of Background Radiation on monitor sensitivity.

i. Refer to the response to Item 4.a for design considerations.

2. The post-accident gaseous effluent monitors will be capable of functioning both during and following an accident. The monitoring system is not Seismic, however, isokinetic nozzles that attach to seismic duct or pipe will be seismically designed. The effluent monitors are designed to accomodate a design-basis release ($10^{+3}$ uci/cc $Xe^{133}$ equivalent) and respond to decreasing concentrations of noble gases down to the sensitivity (i.e., minimum detectable level, MDL) of each monitor. Refer to Item 4.a.i for the sensitivity of each monitor type.

3. The Main Steam RV & DV and Auxiliary FWPT exhaust monitors will be externally mounted monitors viewing the Main Steam line and Auxiliary Feed Water Pump Turbine exhaust line, respectively. Procedures will be developed to correct for the low energy gammas that the external monitors will not detect. Refer to Item 4.b for a description of these procedures.

4. The design decription of the post-accident gaseous monitors is as follows:

Process Vent and Ventilation Vent Effluent Monitors

a. System Description:

(i) Kaman Sciences Corporation's model KMG-HR effluent monitoring system will be used to monitor noble gas effluents. The system consists of noble gas monitors, particulate and Iodine collectors, isokinetic nozzle, and control room display. The noble gas monitors consist of three detectors, one Beta scintillation and two GM tubes, and cover the following ranges (sensitivity to $Xe^{133}$):

Normal Background - 1 mr/hr $Co^{60}$

| | |
|---|---|
| Low Range | $5.2 \times 10^{-7}$ to $3.7 \times 10^{-1}$ uci/cc |
| Mid Range | $9.6 \times 10^{-4}$ to $1 \times 10^{2}$ uci/cc |
| High Range | $7.1 \times 10^{-1}$ to $1 \times 10^{5}$ uci/cc |

Maximum Accident Background - 600 mr/hr $Co^{60}$

Low Range $1.2 \times 10^{-6}$ to $3.7 \times 10^{-1}$ uci/cc

Mid Range $1.7 \times 10^{-2}$ to $1 \times 10^{2}$ uci/cc

High Range $1.8$ to $1 \times 10^{5}$ uci/cc

NOTE: The above sensitivity values are based on ANSI N13.10 - 1974 Procedures.

The energy response of the detectors are listed in Table II.F.1A.

Calibration - Each prototype detector/sampler design is processed through a primary radionuclide calibration. Two or more isotopes are prepared to simulate the effluent for which the system will be used. These isotopes are assayed by direct comparison with NBS or commerically certified isotope standards and thus become primary calibration sources. A secondary point source is directly referenced to the primary calibrated source and is used to calibrate each detector supplied by Kaman.

Field calibration sources and fixtures will be used to periodically recalibrate each detector and in accordance with the Technical Specification requirements.

(ii) Sampler Probe Design - The sampler probes will be designed in accordance with ANSI N13.1 and will be located down-stream of the last effluent entry point prior to discharge.

Sampler/Detector Location - The sampler will be located in a low background area (less than 600 mr/hr under accident conditions) such that the detector sensitivities are not adversly affected. Refer to the list of monitor ranges for maximum background conditions provided above.

(iii) & (iv) Location of Instrument Readouts - The readouts will be located in the control room providing the information discussed by item 1.g above. The readings will be continuously recorded and can be obtained at least every 15 minutes during and following an accident. The procedure for transmitting or disseminating this information will be developed and provided prior to operation.

(v) Source of Power - The monitors will receive power from the station emergency power supply system.

b. Methods used to convert radiation readings (uci/cc) to release rate per unit time (uci/sec):

The process vent and ventilation vent monitors are digital based systems that will receive signals from the detectors and vent flow meters, process these signals, and display results as radiation concentrations uci/cc, $Xe^{133}$ equivalent, or release rates (uci/sec), $Xe^{133}$ equivalent, on demand. The effluent concentrations and release rates (or duct flow rates) will be continuously recorded.

Periodic grab samples can be obtained for a laboratory spectral analysis to determine the radionuclide distribution as a function of time after shutdown.

Main Steam RV & DV and Auxiliary FWPT Exhaust Monitors

a. System Description

(i) Externally mounted monitors will view the main steam lines upstream of the safety valves/dump valves and the steam driven auxiliary feedwater pump turbine exhaust line. Nuclear Research Corporation's model TA900-TA600 area monitoring system detectors are located within a collimated two inch thick lead shielded enclosure. The system consists of a three detector area monitor (two GM tubes and an Ion Chamber) and a remote readout/control unit. The detector range overlap and cover a dynamic range of .01 MR/hr to 10,000 R/hr, which represents the following approximate effluent concentrations based on a time zero TID-14844 nuclide source term and various background conditions:

Normal Background Conditions

| Monitor Location | | Approximate Range | Background |
|---|---|---|---|
| M.S. | | $7.5 \times 10^{-3}$ to $1 \times 10^{6}$ uci/cc (TID mixture)* | .75 MR/hr - $Co^{60}$ |
| | or | | |
| | | $1.1 \times 10^{-1}$ to $1.4 \times 10^{7}$ uci/cc ($Xe^{133}$ equiv. dose)** | |
| AFWPT | | (later)*** (TID mixture)* | |
| | or | | |
| | | (later)*** ($Xe^{133}$ equiv. dose)** | |

LOCA W/O SI Recirc. (TMI)

| Monitor Location | | Approximate Range | Background |
|---|---|---|---|
| M. S. | | $2 \times 10^{1}$ to $1 \times 10^{6}$ uci/cc (TID mixture)* | 2 R/hr |
| | or | | |
| | | $2.9 \times 10^{2}$ to $1.4 \times 10^{7}$ uci/cc ($Xe^{133}$ equiv. dose)** | |
| AFWPT | | (later)*** (TID mixture)* | |
| | or | | |
| | | (later)*** ($Xe^{133}$ equiv. dose)** | |

LOCA With SI Recirc.

| Monitor Location | Approximate Range | Background |
|---|---|---|
| M.S. | $2 \times 10^3$ to $1 \times 10^6$ uci/cc (TID mixture)* | 200 R/hr |
| or | $2.9 \times 10^4$ to $1.4 \times 10^7$ uci/cc ($Xe^{133}$ equiv dose)** | |
| AFWPT | (later)*** (TID mixture)* | |
| or | (later)*** ($Xe^{133}$ equiv. dose)** | |

NOTES: *The effluent concentrations are based on a time zero TID nuclide mixture.

**The $Xe^{133}$ concentration is derived from a normalized nuclide mixture based on time zero TID-14844.

***Range dependent upon final design.

Energy Dependence - ± 20% at 80 Kev to 1.5 Mev energies.

Calibration - Each detector is calibrated with a point source ($Cs^{137}$) traceable to NBS. The frequency of the recalibrations will be in accordance with Technical Specification requirements.

Analytical methods are utilized to convert radiation readings (mR/hr) to effluent concentrations (uci/cc - TID mixture) in lieu of a primary radionuclide calibration.

(ii) Monitor Location - The Main Steam monitors are located in the mainsteam valve house oriented such that one monitor "views" each relief header for a total of three monitors per unit.

The radiation background in the main steam valve house varies with the type of accident postulated. The effect of background on monitor sensitivity for normal conditions, a LOCA with out Safety Injection recirculation, and a LOCA with Safety Injection recirculation are listed in the Table above.

The Auxiliary Feedwater Pump Turbine exhaust monitors will be located in the auxiliary feedwater pump house for North Anna and on the wall outside the main steam valve house for Surry. A monitor will "view" one of two six (6) inch auxiliary FWPT exhaust lines for a total of one (1) monitor per unit.

The radiation background for the Auxiliary FWPT exhaust monitors will be less than the main steam monitors and its effect on monitor sensitivity will be assessed.

(iii) Location of Instrument Readouts - The readouts are located in the control room at North Anna and in the electrical switch gear room at Surry. Both areas are accessible following an accident.

(iv) The readings will be continuously recorded in the control room and can be obtained at least every 15 minutes during and following an accident.

The procedure for transmitting or disseminating this information have been developed and issued (EPIP's) for the Main Steam effluent monitors.

(v) Source of Power - The monitors will receive power from the station emergency power supply system.

b. Methods used to convert radiations readings to release rates per unit time:

The detector radiation readings (mR/hr) will be converted to effluent concentrations (uci/cc - TID nuclide mixture) by using analytically derived conversion factors based on a time zerio TID 14844 mixture. Correction factors as a function of time will be used to correct for changes in the radionuclide spectrum after shutdown.

Steam flow released from the safety valves or dump valves is determined by the design flow through each valve times the number of valves opened.

The steam flow from the Auxiliary Feedwater pump turbine exhaust will be determined by direct steam measurment.

TABLE II.F.1A

LOW RANGE
BETA DETECTOR PERFORMANCE TABLE - GAS
MODEL KDG IN KSG-HR

| Isotope | *Count Rate ($x10^8$) | Isotope | *Count Rate ($x10^8$) |
|---|---|---|---|
| BR-84 | 1.17 | Cs-134 | 0.432 |
| Rb-88 | 1.32 | Cs-136 | 0.284 |
| Rb-89 | 1.00 | Cs-137 | 0.490 |
| Sr-89 | 0.943 | Cs-138 | 1.22 |
| Sr-90 | 0.494 | Ba-140 | 0.650 |
| Sr-91 | 0.920 | La-140 | 0.935 |
| Sr-92 | 0.544 | Ce-144 | 0.176 |
| Y-90 | 1.10 | Pr-144 | 1.19 |
| Y-91 | 0.966 | Kr-85 | 0.607 |
| Y-92 | 1.24 | Kr-85M | 0.584 |
| Zr-95 | 0.340 | Kr-87 | 1.25 |
| Nb-95 | 0.0253 | Kr-88 | 0.644 |
| Mo-99 | 0.774 | Xe-133 | 0.268 |
| TC-99m | 0.203 | Xe-133M | 0.472 |
| I-129 | 0.0165 | Xe-135 | 0.748 |
| I-131 | 0.504 | Xe-135M | 0.163 |
| I-132 | 0.874 | Xe-138 | 1.13 |
| I-133 | 0.843 | Mn-54 | 0.0002 |
| I-134 | 0.750 | Mn-56 | 0.958 |
| I-135 | 0.704 | Co-58 | 0.422 |
| Te-129 | 0.169 | Co-60 | 0.222 |
| Te-132 | 0.107 | Fe-59 | 0.296 |
| Te-134 | 0 | | |
| N-16 | 1.20 | AR-41 | 0.841 |

*Counting rates are in counts per minute per microcurie per cc.

MID RANGE**
GM DETECTOR

$Xe^{133}$ $6.16 \times 10^3$

HIGH RANGE**
GM DETECTOR

$Xe^{133}$ $1.36 \times 10^1$

**Energy dependence for the mid and high range detectors is ±20% 80 keV to 3.0 meV.

## II.F.1. ATTACHMENT 1, NOBLE GAS EFFLUENT MONITOR - EXPECTIONS

### I. Process Vent and Ventilation Vent Effluent Monitors

Some design information was available by January 1, 1981 for NRC review but will be available by November 1, 1981. In addition, procedures and operating instructions will be developed as additional information is received from the vendor and will be available by the required implementation date, but not by January 1, 1981.

### II. Main Steam SV&DV and Auxiliary FWPT Exhaust Effluent Monitors

Some design information will not be available for the Auxiliary FWPT exhaust effluent monitors by January 1, 1981, the conceptual design is similar to the Main Steam SV & DV effluent monitors but the details of shielding and mounting are still in design. All design information will be available by November 1, 1981. Preliminary procedures are in effect for the installed main steam SV & DV effluent monitors. However, final procedures for both effluent monitors are still under development to adequately consider the requirements of this section. These final procedures will be in effect by the required implementation date.

II.F.1, Attachment 2 - Sampling and Analysis of Plant Effluents

1. The continuous collection of post accident releases of radioactive iodines and particulates meets the following requirements as specified in Table II.F.1-2:

Applicability - Process vent and ventilation vent effluent release paths.

Equipment - Kaman Sciences Model KMG-HR effluent monitoring system will continuously sample and collect particulates and iodines on shielded filters with provisions for removal and laboratory analysis. The samples will be analyzed with site Germanium detectors for low activity samples. The highly radioactive samples will be analyzed as discussed in the "Analysis" section below.

Design Basis Shielding Enveloped - The high range particulate and iodine collectors are mounted inside of a three inch 4 lead shielded assemblies. A portable transfer housing containing three inches of 4 lead shielding will be used for personnel protection during the transfer of the filters. Occupational exposure for filter removal will be negligible due to the lead shields. Assuming a collecting time of thirty minutes at the maximum posulated accident of $10^2$ uci/cc (.5 mev gamma) with a design sample flow rate of 1000 cc/min through the filters, the radiation dose rate one foot from the filter assembly will be 0.1 mR/hr.

Sampling Media - The iodine collector is a charcoal cartridge with an effective absorption (for methyl iodines) of not less than 95%.

The particulate collector is a paper filter disk with an effective retention of 99% of 0.3 micron particles.

Sampling Considerations -

A. Representative Sampling - The sampler probes will be designed in accordance with ANSI-N13.1-1969 and located downstream of the last effluent entry point prior to discharge.

The sample piping will be routed using ANSI-N13.1, Appendix B as a guide (i.e., minimize sample line bends and lengths). The sample flow isokinetics is addressed in item (3).

B. Entrained Moisture - Refer to Item (4)

C. Continuous Collection - The effluent monitors will continuously operate, in accordance with Technical Specifications, as long as the exhaust flow occurs.

D. Limiting Occupational Dose - The particulate and iodine collector design is sufficiently shielded to minimize personnel exposure during filter transfer to the count laboratory. Refer to "Design Basis Shielding Envelope" response above.

A detector dedicated to each collector will limit the activity on the filters to preset levels by switching the sample flow to an unused filter when the set point radiation level is obtained.

Three foot tongs will be used to transfer the unshielded filter from the collector to the portable transfer housing thus avoiding direct contact with the filters.

Shielding will be designed to protect personnel during sample analysis.

E. Analysis - (To be supplied later)

2. Shielding Design Basis - Refer to Item (1)

3. Isokinetic Sampling - The effluent monitoring sampling design is currently being formulated and will be submitted later.

4. Entrained Moisture - Effluent streams will be analyzed to determine the potential of containing entrained moisture. If a potential exists, then the sample lines will be heat traced to prevent condensation.

## II.F.1, ATTACHMENT 2, SAMPLING AND ANALYSIS OF PLANT EFFLUENTS - EXCEPTIONS

Some design information will not be available by January 1, 1981 for NRC review. Information will be made available as it is received from the vendor. It is anticipated that all design information will be available by November 1, 1981.

## II.F.1, ATTACHMENT 3, CONTAINMENT HIGH RANGE RADIATION MONITORS - EXCEPTIONS

Detectors and cable are being mounted in the containment during scheduled outages as shown in Section C. However, the vendor has not qualified an in-containment electrical termination procedure. Outside containment work continues as material is received. The latest scheduled material delivery is April, 1981. However, testing and subsequent operation of the system requires containment entry after vendor qualification of electrical terminations.

Clarification of items 5 and 7 of Appendix B to NUREG-0737 are presented under items 1a and 1b.

## II.F.1, ATTACHMENT 5, CONTAINMENT WATER LEVEL - EXCEPTIONS

Three vendors are currently attempting to qualify equipment to meet the requirements. Extension of the implementation date to January 1, 1982 should be sufficient for completion of qualification testing. However, if the vendor supplying the equipment to Surry does not meet the requirements, it is not sufficient time to procure and install qualified equipment.

## II.F.1, ATTACHMENT 6, CONTAINMENT HYDROGEN MONITOR - EXCEPTIONS

The Comsip, Inc. hydrogen analyzers for North Anna and Surry have just completed their qualification test program. Pending review and approval of the test results, installation of fully qualified equipment by the required date cannot be assured.

## II.F.2 INSTRUMENTATION FOR DETECTION OF INADEQUATE CORE COOLING - EXCEPTIONS

NUREG 0737 requires the implementation of the reactor vessel level systems by January 1, 1982 and the documentation associated with these systems by January 1, 1981. Present schedules show that the Westinghouse reactor vessel level monitoring system along with the supporting analyses will be available June of 1981 at the earliest. The balance of plant engineering and stress analysis will require approximately three months following receipt of the Westinghouse reactor vessel level monitoring system design. This places the earliest availability of the reactor vessel level system for installation in the Fall of 1981. Installation would require an approximate two-month outage and the removal of the reactor vessel head. Therefore, Vepco intends to install the reactor vessel level monitoring systems in each of our nuclear units during the first refueling outage following availability of these systems. As shown in Section C, installation of reactor vessel level monitoring systems should be completed on all units by August, 1982. Therefore, Vepco requests a change to the required installation date for the Reactor Vessel Level System to be January 1, 1982 or the first refueling outage after system availability, which ever is later.

The ability of a system to provide an unambiguous indication has not been demonstrated. The required installation date should be delayed until additional research is performed. If the implementation date is not delayed, the installed system should not be required to be changed if future research provides a better system.

## II.K.3.1 INSTALLATION AND TESTING OF AUTOMATIC POWER OPERATED RELIEF VALVE ISOLATION SYSTEM

A PORV failure report, WCAP-9804, was submitted for NRC review in March, 1980 in response to Item II.K.3.2. In WCAP-9804, an assessment of an automatic PORV isolation system was provided. The conclusions of the report are that such an automatic isolation system would not provide a significant benefit in terms of plant safety and consequently should not be required.

Vepco supports this conclusion. Therefore, installation of an automatic PORV isolation system is not planned.

## II.K.3.2 REPORT ON OVERALL SAFETY EFFECT OF POWER-OPERATED RELIEF VALVE ISOLATION SYSTEM

The Westinghouse Owners Group has prepared and submitted a report which addresses historical valve failure rate data and documentation of actions taken since the TMI-2 accident to decrease the probability of a stuck-open PORV. This report provides the basis for a decision on the necessity of incorporating an automatic PORV isolation system, as specified in Item II.K.3.1.

This report, WCAP-9804, was transmitted to the NRC for review on March 13, 1981 (Letter No. OG-52).

## II.K.3.3 REPORTING SAFETY AND RELIEF VALVE FAILURES AND CHALLENGES

These requirements are being implemented at North Anna Units 1 and 2 and Surry Units 1 and 2.

North Anna Unit 2 Technical Specification 6.9.1.6 requires documentation of all challenges to the PORV's and Safety Valves with the routine monthly operating report. Failures of these valves will be reported in accordance with Specification 6.9.1.8.

Similar provisions have been proposed for North Anna Unit 1 and Surry Technical Specifications.

## II.K.3.5 AUTOMATIC TRIP OF REACTOR COOLANT PUMPS DURING LOSS-OF-COOLANT ACCIDENT

The Westinghouse Owners Group has submitted LOFT test predictions in accordance with the schedule provided by II.K.3.5. Currently, additional information, in response to NRC questions on the LOFT predictions, is being prepared and will be submitted in June, 1981. NRC approval of Westinghouse models for the RCP trip evaluations has not yet been received. Therefore, Vepco's response regarding the need for automatic RCP trips during a LOCA will be provided within three (3) months of approval of the Westinghouse models. This is consistent with the timetable of Item II.K.3.5 but not the July 1, 1981 submittal date for design information of an automatic RCP trip system.

## II.K.3.17 REPORT ON OUTAGES OF EMERGENCY CORE-COOLING SYSTEMS LICENSEE REPORT AND PROPOSED TECHNICAL SPECIFICATION CHANGES

Operating records for the last five years at Surry Units 1 and 2, and since the commencement of operation at North Anna Units 1 and 2, have been reviewed to determine the dates, lengths and causes of outages of emergency core cooling systems.

The results of this review for North Anna 1 and 2 and Surry 1 and 2 were submitted for NRC review on December 31, 1980 (North Anna) and February 27, 1981 (Surry). This information is provided in the following pages. Pages II.K.3.17-4 through II.K.3.17-26 supply the information for North Anna. Pages II.K.3.17-27 through II.K.3.17-44 supply the information for Surry.

ECCS OUTAGE REPORT

NORTH ANNA POWER STATION

UNITS 1 AND 2

## ECCS OUTAGE REPORT

This report is concerned with the outages of the Emergency Core Cooling System (ECCS) of North Anna Units 1 and 2, commencing with the issuing of the fuel loading license of each unit. The data collected is presented as a response to the requirements of NUREG 0737, Item II.k.3.17-1 and will address the following four criteria as required by NUREG 0737. These four criteria are: (1) outage dates and duration of outages; (2) cause of the outage; (3) ECCS systems or components involved in the outage; (4) corrective action taken.

The primary methods utilized in collecting the data consisted of identifying the ECCS systems/components to be examined, and of reviewing all action statement logs for the appropriate time period in order to answer question no. 1, no. 3 and to an extent question no. 2. In order to answer question no. 4, and to an extent question no. 2, completed maintenance reports and periodic surveillance tests were retrieved from Station Records and reviewed. Other documents used in the review included control room operator logs, shift supervisor logs, equipment tagging reports and completed outstanding maintenance lists.

The equipment identified as ECCS equipment and the total outage duration times are as follows for each unit.

| | | HOURS | |
|---|---|---|---|
| Emergency Diesel Generator and Associated Subsystems | Unit 1 | 468 | (1H Diesel) |
| | | 166.5 | (1J Diesel) |
| | Unit 2 | 184.25 | (2H Diesel) |
| | | 92.5 | (2J Diesel) |
| High Head Safety Injection System | Unit 1 | 172 | |
| | Unit 2 | 4.5 | |
| Low Head Safety Injection System | Unit 1 | 44.5 | |
| | Unit 2 | 88 | |
| Refueling Water Storage Tank | Unit 1 | 24.5 | |
| | Unit 2 | 1 | |
| Inside and Outside Recirculation Spray | Unit 1 | 21.5 | |
| | Unit 2 | 59.5 | |
| Casing Cooling Systems | Unit 1 | 445.5 | |
| | Unit 2 | 7 | |

| | | HOURS |
|---|---|---|
| Safety Injection Accumulators | Unit 1 | 24.5 |
| | Unit 2 | 32 |
| Boron Injection System | Unit 1 | 68.5 |
| | Unit 2 | 272.5 |
| Service Water System | Unit 1 | 100.75 |
| | Unit 2 | 29.5 |

NORTH ANNA POWER STATION
UNIT #1
ECCS OUTAGE REPORT

ELECTRICAL POWER SYSTEM

| COMPONENT | CAUSE OF OUTAGE | CORRECTIVE ACTION | OUTAGE DATES | OUTAGE DURATION |
|---|---|---|---|---|
| 1-H Emergency Diesel Generator | Preventative Maintenance | Completed Maintenance | 09-01-78 | 6 hours |
| | Repair and re-install diesel battery racks | Fastened racks to concrete wall | 12-02-78 | 10 hours |
| | Preventative Maintenance | Completed Maintenance | 03-02-79 to 03-05-79 | 77.5 hrs |
| | Remove angle iron on lip of exhaust ducting outlet | Ground out pieces of angle iron | 05-03-79 | 11 hours |
| | Semi-Annual Preventative Maintenance | Completed Maintenance | 08-02-79 | 7.5 hours |
| | Diesel tripped on high crank-case Pressure | Pressure switch recali-brated - SAT. | 02-02-80 to 02-04-80 | 47 hours |
| | Diesel tripped on Overspeed | Reset speed control governor | 02-15-80 to 02-17-80 | 46 hours |
| | Diesel tripped on Overspeed | Replaced hi speed relay | 03-02-80 to 03-04-80 | 38 hours |
| | Diesel tripped on Lube Oil Crank-case Low Pressure | Inspected diesel, performed periodic test | 03-06-80 | 14 hours |
| | Fitting leaks at pressure switch (EG-PS-603H) inside skid mounted Control Panel | Tightened fitting and recali-brated pressure switch. Ran diesel - no leaks | 03-16-80 | 4.5 hours |
| | Gasket on oil filter inside radiator house - leaking | tightened packing and plug on drain valve | 03-21-80 | 4 hours |
| | Tripped on overspeed on emer. start (SI) | Replaced Governor | 04-03-80 to 04-04-80 | 1 hour |

NORTH ANNA POWER STATION
UNIT #1
ECCS OUTAGE REPORT

ELECTRICAL POWER SYSTEM
(continued)

| COMPONENT | CAUSE OF OUTAGE | CORRECTIVE ACTION | OUTAGE DATES | OUTAGE DURATION |
|---|---|---|---|---|
| 1-H Emergency Diesel Generator | Preventative Maintenance | Complete Maintenance | 04-30-80 to 05-02-80 | 49.5 hrs |
| | Tripped on overspeed - emer. start (SI) | Replace with a new modified woodward governor | 05-23-80 to 05-26-80 | 83.5 hrs |
| | Preventative Maintenance | Complete Maintenance | 09-22-80 | 6.5 hours |
| | Preventative Maintenance | Complete Maintenance | 10-02-80 to 10-04-80 | 51 hours |
| | Remove 1H from service to install agastat in diesel output breaker | Install agastat and tested | 11-16-80 | 5.5 hours |
| 1-J Emergency Diesel Generator | Semi-Annual Preventative Maintenance | Complete Maintenance | 11-16-78 | 7 hours |
| | Repair and re-install diesel battery racks | Fastened battery racks to concrete wall | 12-02-78 | 10 hours |
| | Semi-Annual Preventative Maintenance | Complete Maintenance | 02-16-79 | 12 hours |
| | Preventative Maintenance | Complete Maintenance | 03-16-79 | 5.5 hours |
| | Remove angle iron on lip of exhaust ducting outlet | Ground out pieces of angle iron | 05-04-79 | 1.5 hours |
| | Tripped on overspeed | Governor reset-cooldown for refuel | 09-26-79 to 09-27-79 | 21.5 hrs |

NORTH ANNA POWER STATION
UNIT #1
ECCS OUTAGE REPORT

ELECTRICAL POWER SYSTEM
(continued)

| COMPONENT | CAUSE OF OUTAGE | CORRECTIVE ACTION | OUTAGE DATES | OUTAGE DURATION |
|---|---|---|---|---|
| 1-J Emergency Diesel Generator | Accuracy check on crankcase pressure switch | Calibrated and installed new switch | 03-12-80 | 49.5 hrs |
| | Auxiliary oil pump out of service | Replaced overload assembly auxiliary contact | 03-24-80 | 11 hours |
| | Preventative Maintenance | Complete Maintenance | 05-22-80 to 05-23-80 | 27 hours |
| | Preventative Maintenance | Complete Maintenance | 09-22-80 | 6 hours |
| | Preventative Maintenance | Complete Maintenance | 10-16-80 to 10-18-80 | 47.5 hrs |
| | Remove J diesel from service to install agastats in diesel output breaker | Install agastat, perform periodic test | 11-16-80 | 9 hours |

NORTH ANNA POWER STATION
UNIT #1
ECCS OUTAGE REPORT

HIGH HEAD SAFETY INJECTION SYSTEM

| COMPONENT | CAUSE OF OUTAGE | CORRECTIVE ACTION | OUTAGE DATES | OUTAGE DURATION |
|---|---|---|---|---|
| A-Charging Pump | Periodic Surveillance Test | Complete test<br>Two pumps still available | 01-27-79 | 10 hours |
| | Vent valve on suction line cracked | Crack ground out and re-welded. Two pumps remain available | 05-13-79 | 10 hours |
| B-Charging Pump | Outboard Seal Leakage | *Seal replaced and pump tested. Two pumps remain available | 07-14-78 | 72 hours |
| | Periodic Testing | Complete Testing<br>Two pumps remain available | 03-20-80 | 10 hours |
| | Preventative Maintenance | Complete Maintenance<br>Two pumps still available | 09-30-80 to 10-01-80 | 32 hours |
| C-Charging Pump | Blown oil seal on inboard bearing | Replace oil seal<br>Two pumps remain available | 04-03-78 | 13 hours |
| | Inboard pump seal leakage | Repair inboard mechanical seal | 07-17-78 | 13 hours |
| | Oil leak on oil strainer | Oil filter retainer ring replaced. Two pumps remain available | 03-19-80 | 12 hours |

* Placement of charging pump 1B in operable status was delayed in order to shim the outboard seal to reduce charging pump 1B outboard seal leakage. During this time, 3 hours 11 minutes, only one pump was operable.

NORTH ANNA POWER STATION
UNIT #1
ECCS OUTAGE REPORT

LOW HEAD SAFETY INJECTION SYSTEM

| COMPONENT | CAUSE OF OUTAGE | CORRECTIVE ACTION | OUTAGE DATES | OUTAGE DURATION |
|---|---|---|---|---|
| B-Low Head Pump | Preventative Maintenance | Complete Maintenance | 10-16-78 | 2 hours |
| | Preventative Maintenance | Complete Maintenance | 11-28-78 | 1.5 hrs |
| A-Low Head Pump | Preventative Maintenance | Complete Maintenance | 11-03-78 | 7 hours |
| | Preventative Maintenance | Complete Maintenance | 11-28-78 | 2 hours |
| MOV 1863A Discharge to Charging Pumps | Perform Periodic Surveillance Test | Complete Testing | 05-11-79 | 6 hours |
| Check Valves in SI discharge lines | Inservice Inspection - Periodic Testing | Complete testing | 05-03-80 | 1.5 hrs |
| Discharge Valve 1890D | No light indication | Fuse holder tightened | 05-19-80 | 22.5 hrs |
| MOV 1885D B-Pump | Valve would not electrically operate. | Replaced locking ring on light module | 06-23-80 | 2 hours |

NORTH ANNA POWER STATION
UNIT #1
ECCS OUTAGE REPORT

QUENCH SPRAY SYSTEM

| COMPONENT | CAUSE OF OUTAGE | CORRECTIVE ACTION | OUTAGE DATES | OUTAGE DURATION |
|---|---|---|---|---|
| Refueling Water Storage Tank | Level below specifications due to a Safety Injection | Made up to tank | 04-29-78 | 3 hours |
| | Temperature >50°F due to a Safety Injection | Chillers placed in service - Temp. lowered | 04-29-78 | 13 hours |
| | Boron concentration low due to a Safety Injection | Borate and make up to tank | 04-29-78 | 2.5 hrs |
| | Engineering determined that an excessive column of water is being maintained in the tank | Lowered level to new specification | 09-12-79 | 1.5 hrs |
| | Level low due to a safety injection | Refill Tank | *09-25-79 | *Cool down for refueling |
| | Level low due to a safety injection | Refill Tank | 04-03-80 to 04-04-80 | 4.5 hrs |

NORTH ANNA POWER STATION
UNIT #1
ECCS OUTAGE REPORT

RECIRCULATION SPRAY SYSTEM

| COMPONENT | CAUSE OF OUTAGE | CORRECTIVE ACTION | OUTAGE DATES | OUTAGE DURATION |
|---|---|---|---|---|
| Casing Cooling Tank | Temperature > 50°F | Placed on chiller | 09-22-78 to 09-23-78 | 18 hours |
| | Freon leak from the tank chiller unit caused the temperature to go out high. | System recharged with freon. Vendor to locate freon leak. | 08-19-79 | 5.5 hrs |
| Casing Cooling Pump 3A | ΔP high as found on Periodic surviellance test | Vented pressure gage - test redone | 08-12-78 to 08-17-78 | 120 hrs |
| | Preventative Maintenance | Changed oil, completed inspection | 05-25-79 | .5 hour |
| | Leakage from outboard pump seal | Bearing housing disassembled cleaned and inspected. Casing surface reground to set O-ring | 6-6-79 | 8 hours |
| | Preventative Maintenance | Complete Maintenance | 06-12-80 | 6 hours |
| | Preventative Maintenance | Complete Maintenance | 07-03-80 | 3 hours |
| | ΔP high as found on Periodic surviellance test | Vented pressure gage - test redone | 08-11-80 to 08-13-80 | 57 hours |
| Casing Cooling Pump 3B | Preventative Maintenance | Complete Maintenance | 05-25-79 | .5 hours |
| | Testing indicated pump ΔP too high | Vented pressure gage | 08-27-79 to 08-29-79 | 54 hours |

NORTH ANNA POWER STATION
UNIT #1
ECCS OUTAGE REPORT

RECIRCULATION SPRAY SYSTEM
(continued)

| COMPONENT | CAUSE OF OUTAGE | CORRECTIVE ACTION | OUTAGE DATES | OUTAGE DURATION |
|---|---|---|---|---|
| Casing Cooling Pump 3B | Unknown | Unknown | 09-14-79 to 09-20-79 | 139 hrs |
| | Preventative Maintenance | Complete Maintenance | 07-26-80 | 3.5 hrs |
| Casing Cooling Tank level xmitter | Level controller frozen | Built temporary enclosure ground gage | 12-29-79 | 5 hours |
| | Design change | Place heat tracing around transmitter | 01-30-80 | 2.5 hrs |
| | Level xmitter not approved | Replace with a seismic approved xmitter | 02-29-80 to 03-01-80 | 23 hours |
| Outside Recirc. Spray Pump A | Special testing | Completed testing | 01-10-80 | .5 hour |
| | Preventative Maintenance | Completed Maintenance | 11-28-80 | 1.5 hrs |
| Outside Recirc. Spray Pump B | Periodic Testing | Completion of testing | 11-07-80 to 11-08-80 | 15 hours |
| | Preventative Maintenance | Complete Maintenance | 11-28-80 | 1.5 hrs |

NORTH ANNA POWER STATION
UNIT #1
ECCS OUTAGE REPORT

RECIRCULATION SPRAY SYSTEM
(continued)

| COMPONENT | CAUSE OF OUTAGE | CORRECTIVE ACTION | OUTAGE DATES | OUTAGE DURATION |
|---|---|---|---|---|
| Inside Recirc. Spray Pump A | Preventative Maintenance | Complete Maintenance | 11-28-78 | 2 hours |
| Inside Recirc. Spray Pump B | Design change 79-S64 | Fuse and fuse blocks changed out on design change | 10-06-80 | 1 hour |

NORTH ANNA POWER STATION
UNIT #1
ECCS OUTAGE REPORT

SAFETY INJECTION SYSTEM

| COMPONENT | CAUSE OF OUTAGE | CORRECTIVE ACTION | OUTAGE DATES | OUTAGE DURATION |
|---|---|---|---|---|
| Accumulator C | Level > 7756 gal (T.S. limit) | Lowered level | 10-02-78 | 1 hour |
| | Pressure low when accum. discharged to relief tank thru RHR relief after RHR outlet valves opened. | Returned level and pressure to limit. | 03-06-79 | 1 hour |
| | Boron concentration too high due to stratification within the accumulators | Feed and bleed accumulator until within specifications | 01-26-80 to 01-27-80 | 4 hours |
| | Boron concentration too high due to stratification within the accumulators | Feed and bleed accumulator until within specifications | 11-24-80 | 2 hours |
| Accumulators B, C | Boron concentration too high due to stratification within the accumulators | Feed and bleed accumulator until within specifications | 01-26-80 | 12 hours |
| Accumulator B | Boron concentration too high due to stratification within the accumulators | Feed and bleed accumulator until within specifications | 08-01-80 | 2 hours |
| | Boron concentration too high due to stratification within the accumulators | Feed and bleed accumulator until within specifications | 09-04-80 | 2.5 hrs |
| Boron Injection Tank (BIT) | Level low due to a Safety Injection | Make up to BIT | 04-30-78 | 9 hours |
| | Low level due to a Safety Injection | Make up to BIT | *09-25-79 | *Cooled down for refueling |

II.K.3.17-16

NORTH ANNA POWER STATION
UNIT #1
ECCS OUTAGE REPORT

SAFETY INJECTION SYSTEM
(continued)

| COMPONENT | CAUSE OF OUTAGE | CORRECTIVE ACTION | OUTAGE DATES | OUTAGE DURATION |
|---|---|---|---|---|
| Boron Injection Tank (BIT) | Boron concentration and temp low | Return to specification | 02-23-80 to 02-24-80 | 8 hours |
| | Boron concentration level, temp low - Safety Inj. | Return to specifications | 04-03-80 to 04-04-80 | 7 hours |
| | Boron concentration low | Return to specifications | 05-30-80 | 10 hours |
| BIT - Check Valve 1-SI-66 | Check valve failed to hold pressure for testing | Unknown | 08-22-78 | 25 hours |
| BIT - 1-SI-304 | Valve plugged with Boric Acid | Cleaned out line | 02-13-80 | 9.5 hrs |

II.K.3.17-17

NORTH ANNA POWER STATION
UNIT #1
ECCS OUTAGE REPORT

SERVICE WATER SYSTEM

| COMPONENT | CAUSE OF OUTAGE | CORRECTIVE ACTION | OUTAGE DATES | OUTAGE DURATION |
|---|---|---|---|---|
| Service Water Pump 1A | Preventative Maintenance - Change oil | Complete Maintenance | 09-18-78 to 09-19-78 | 5 hours |
| | Preventative Maintenance - Polarization Index Ratio | Complete Maintenance | 11-29-78 | 1.5 hrs |
| | Periodic Test III - Pump Tilt Measurements | Complete Measurements | 05-16-79 | 3 hours |
| | Periodic Test III - Pump Tilt Measurements | Complete Measurements | 01-10-80 | 1 hour |
| | Preventative Maintenance | Complete Maintenance | 03-07-80 | 4 hours |
| Service Water Pump 1B | Unknown | Unknown | 07-06-78 | 5 hours |
| | Hi discharge pressure as per Periodic test 75.2B | Retested using more accurate flow device | 07-26-78 to 07-28-78 | 33.5 hrs |
| | Periodic Test III - Pump Tilt Measurements | Complete Measurements | 03-15-79 | 2.5 hrs |
| | Periodic Test III - Pump Tilt Measurements | Complete Measurements | 01-10-80 | 2.5 hrs |
| | Periodic Test III - Pump Tilt Measurements | Complete Measurements | 11-24-80 | .25 hrs |

NORTH ANNA POWER STATION
UNIT #1
ECCS OUTAGE REPORT

SERVICE WATER SYSTEM
(continued)

| COMPONENT | CAUSE OF OUTAGE | CORRECTIVE ACTION | OUTAGE DATES | OUTAGE DURATION |
|---|---|---|---|---|
| MOV 108A, B | Water in Limitorque | No water found, no maintenance done | 08-18-78 | 5 hours |
| Flow xmitter Service Water | Transmitter out of tolerance | Transmitter replaced and calibrated | 10-29-80 to 10-30-80 | 37.5 hrs |

NORTH ANNA POWER STATION
UNIT #2
ECCS OUTAGE REPORT

ELECTRICAL POWER SYSTEM

| COMPONENT | CAUSE OF OUTAGE | CORRECTIVE ACTION | OUTAGE DATES | OUTAGE DURATION |
|---|---|---|---|---|
| Emergency Diesel Gen. 2H | Preventative Maintenance | Complete Maintenance | 05-28-80 | 7 hours |
| | Preventative Maintenance | Complete Maintenance | 05-28-80 to 05-29-80 | 27 hours |
| | Preventative Maintenance | Complete Maintenance | 08-05-80 to 08-06-80 | 34 hours |
| | Preventative Maintenance | Complete Maintenance | 10-01-80 to 10-02-80 | 4.5 hrs |
| | XF-1 Jacket Heater Transformer burned out | Replaced transformer XF-1 | 10-15-80 | 4.5 hrs |
| 2H Diesel Fuel Oil Pump | Fuel oil pump did not start | Level switch, on fuel oil tank, operation tested | 10-15-80 | 16 hours |
| Emergency Diesel Gen. 2J | Diesel tripped after 20 seconds | Performed periodic test 82.1B | 05-17-80 | 8 hours |
| | Preventative Maintenance | Complete Maintenance | 05-28-80 | 15.5 hrs |
| | Auxiliary lube oil xformer burned out | Replaced xformer and tested - sat. | 06-06-80 | 6.5 hrs |
| | Preventative Maintenance | Complete Maintenance | 10-16-80 to 10-18-80 | 55 hours |

NORTH ANNA POWER STATION
UNIT #2
ECCS OUTAGE REPORT

ELECTRICAL POWER SYSTEM
(continued)

| COMPONENT | CAUSE OF OUTAGE | CORRECTIVE ACTION | OUTAGE DATES | OUTAGE DURATION |
|---|---|---|---|---|
| Emergency Diesel Gen. 2J | Diesel batteries failed Periodic test | Recalculated test by normalizing data | 12-01-80 | 6.25 hrs |

NORTH ANNA POWER STATION
UNIT #2
ECCS OUTAGE REPORT

HIGH HEAD SAFETY INJECTION SYSTEM

| COMPONENT | CAUSE OF OUTAGE | CORRECTIVE ACTION | OUTAGE DATES | OUTAGE DURATION |
|---|---|---|---|---|
| Charging Pump 1C | Outboard pump seal leaking | Shutdown pump - repair seal Two pumps remain available | 10-08-80 | 4.5 hrs |
| Safety Injection Pump 1A | Fitting on seal supply leaks | Repair fitting (tighten and test) | 06-10-80 | 10 hours |
| | Lamp test design change mod. | Complete modification | 07-01-80 to 07-03-80 | 41 hours |
| Safety Injection Pump 1B | Lamp test design change mod. | Complete modification | 07-01-80 to 07-03-80 | 41 hours |
| | Relief valve on discharge line lifting at too low a pressure | Pulled valve, adjusted lift-setting and placed back in system | 09-19-80 | 12.5 hrs |
| MOV 2890 A, C LHSI Discharge Valves | Packing Leaks | Tighten Packing | 07-28-80 | .5 hour |
| MOV 2862 A MOV 2885A MOV 2864A | Packing Leaks | Tighten Packing | 07-28-80 | .5 hour |
| MOV 2890B MOV 2862B MOV 2885B | Packing Leaks | Tighten Packing | 10-30-80 | .5 hour |

NORTH ANNA POWER STATION
UNIT #2
ECCS OUTAGE REPORT

RECIRCULATION SPRAY SYSTEM

| COMPONENT | CAUSE OF OUTAGE | CORRECTIVE ACTION | OUTAGE DATES | OUTAGE DURATION |
|---|---|---|---|---|
| Casing Cooling Pump 3A | Preventative Maintenance | Complete Maintenance | 06-13-80 | 3 hours |
| Casing Cooling Pump 3B | Preventative Maintenance | Complete Maintenance | 06-21-80 | 4 hours |
| Recirc. Spray Pump 2A | Lamp Test Modification | Modification Complete | 06-29-80 to 06-30-80 | 9 hours |
| Recirc. Spray Pump 2B | Lamp Test Modification | Modification Complete | 06-24-80 to 06-25-80 | 15.5 hrs |
| Inside Recirc. Spray Pump 1A | Lamp Test Modification | Modification Complete | 06-30-80 to 07-01-80 | 32.5 hrs |
| Inside Recirc. Spray Pump 1B | Mechanical Linkage on breaker 24J1-2 for pump - did not operate | Breaker removed, cleaned and replaced. Tested satisfactorily | 11-25-80 | 2.5 hrs |

NORTH ANNA POWER STATION
UNIT #2
ECCS OUTAGE REPORT

QUENCH SPRAY SYSTEM

| COMPONENT | CAUSE OF OUTAGE | CORRECTIVE ACTION | OUTAGE DATES | OUTAGE DURATION |
|---|---|---|---|---|
| Refueling Water Storage Tank | Boron concentration > 2105 PPM | Feed and bleed tank until within limits | 09-09-80 | 1 hour |

NORTH ANNA POWER STATION
UNIT #2
ECCS OUTAGE REPORT

SAFETY INJECTION SYSTEM

| COMPONENT | CAUSE OF OUTAGE | CORRECTIVE ACTION | OUTAGE DATES | OUTAGE DURATION |
|---|---|---|---|---|
| Accumulator A, B, C | Boron concentration >2100 PPM | Feed and bleed until within specs. | 08-07-80 to 08-13-80 | In Mode 5 - Acc. not needed |
| Accumulator A | Boron concentration >2100 PPM | Feed and bleed until within specs. | 09-11-80 to 09-12-80 | 27.5 hrs |
| Accumulator C | Boron concentration > 2100 PPM | Feed and bleed until within specs. | 10-09-80 | 1.5 hrs |
| | Boron concentration > 2100 PPM | Feed and bleed until within specs. | 10-09-80 | 3 hours |
| Boron Injection Tank (BIT) | Boron concentration low | Batch to Boric Acid Storage tank | 05-20-80 to 05-21-80 | 25 hours |
| | Boron concentration low | Batch to Boric Acid Storage tank | 05-21-80 to 05-26-80 | 120 hrs |
| | Boron concentration low | Batch to Boric Acid Storage tank | 05-27-80 to 06-02-80 | 124 hrs |
| | | Batch to Boric Acid Storage tank | 10-08-80 | .5 hour |
| | | Batch to Boric Acid Storage tank | 10-20-80 | 3 hour |

NORTH ANNA POWER STATION
UNIT #2
ECCS OUTAGE REPORT

SERVICE WATER SYSTEM

| COMPONENT | CAUSE OF OUTAGE | CORRECTIVE ACTION | OUTAGE DATES | OUTAGE DURATION |
|---|---|---|---|---|
| Service Water Pump 1A | Lamp Test Modification | Complete Modification | 06-29-80 | 8 hours |
| Service Water Pump 1B | Lamp Test Modification | Complete Modification | 06-23-80 to 06-24-80 | 21.5 hrs |

ECCS OUTAGE REPORT

SURRY POWER STATION

UNITS 1 AND 2

## ECCS OUTAGE REPORT

This report is concerned with the outages of the Emergency Core Cooling System (ECCS) of Surry Units 1 and 2 for the 5 year period ending December 1980. The data collected is presented as a response to the requirements of NUREG 0737, Item II.k.3.17-1 and will address the following four criteria as required by NUREG 0737. These four criteria are: (1) outage dates and duration of outages; (2) cause of the outage; (3) ECC Systems or components involved in the outage; (4) correction action taken.

The primary methods utilized in collecting the data consisted of identifying the ECC Systems/Components to be examined, and of reviewing the Licensee Event Reports and Periodic Surveillance Tests for the appropriate time period. Other documents used in this review included Maintenance Reports, Control Room Operators' Logs, and the Shift Supervisor's Logs.

Outage data for the #3 Emergency Diesel Generator will be found under both Unit 1 and Unit 2 headings as this is a shared generator. Placement of the data was under the unit that was most affected by the outage.

The equipment identified as ECCS equipment and the total outage duration times are as follows for each unit.

| UNIT 1 | MAINT. HOURS | TESTING HOURS |
|---|---|---|
| Emergency Power System | | |
| #1 EDG | 115.5 | 156.0 |
| #3 EDG | 1.0 | 99.0 |
| Safety Injection System | 74.5 | 1.5 |
| Recirculation Spray System | 0.25 | 157.5 |
| Charging Pump Component Cooling | 48.2 | |
| Service Water System | 14.6 | |

| UNIT 2 | MAINT. HOURS | TESTING HOURS |
|---|---|---|
| Emergency Power System | | |
| #2 EDG | 0.0 | 126.0 |
| #3 EDG | 1.0 | 81.0 |
| Safety Injection System | 51.7 | 6.5 |
| Recirculation Spray System | | 126.5 |
| Charging Pump Component Cooling | 7.0 | |
| Service Water System | 73.9 | |

The equipment identified as ECCS equipment and the average outage duration times per month are as follows for each unit.

| | | HOURS | |
|---|---|---|---|
| Emergency Power System | Unit 1 | 4.5 | (#1 EDG) |
| | | 1.7 | (#3 EDG) |
| | Unit 2 | 2.1 | (#1 EDG) |
| | | 1.4 | (#3 EDG) |
| Safety Injection System | Unit 1 | 1.3 | |
| | Unit 2 | 1.0 | |
| Recirculation Spray System | Unit 1 | 3.0 | |
| | Unit 2 | 2.1 | |
| Charging Pump Component Cooling | Unit 1 | 0.8 | |
| | Unit 2 | 0.1 | |
| Service Water System | Unit 1 | 0.2 | |
| | Unit 2 | 1.2 | |

SURRY POWER STATION
UNIT #1
ECCS OUTAGE REPORT

EMERGENCY POWER SYSTEM

| COMPONENT | CAUSE OF OUTAGE | CORRECTIVE ACTION | OUTAGE DATES | OUTAGE DURATION (HOURS) |
|---|---|---|---|---|
| #1 Emergency Diesel Generator | Crack in cylinder #17 allowed water to enter causing extensive damage when diesel started. | Replaced all damaged parts. | 04-16-76 | 77.00 |
| | Crack in cylinder head #1 allowed water to enter cylinder. | Replaced cylinder head #1, 2, and 16. | 05-08-76 | 10.50 |
| | Cylinder head #19 cracked. | Replaced cylinder head #1, 2 and 16. | 07-02-76 | 13.80 |
| | Cylinder head #7 cracked. | Replaced cylinder head. | 07-23-76 | 11.92 |
| | Base tank switch sticking preventing fuel flow from day tank. | Repaired switch. | 09-02-80 | 2.25 |
| | Periodic Test 22.3A | Completed Test. | 01-05-76 | 3.00 |
| | Periodic Test 22.3A | Completed Test. | 01-30-76 | 3.00 |
| | Periodic Test 22.3A | Completed Test. | 03-02-76 | 3.00 |
| | Periodic Test 22.3A | Completed Test. | 04-04-76 | 3.00 |
| | Periodic Test 22.3A | Completed Test. | 05-06-76 | 3.00 |
| | Periodic Test 22.3A | Completed Test. | 05-14-76 | 3.00 |
| | Periodic Test 22.3A | Completed Test. | 06-02-76 | 3.00 |
| | Periodic Test 22.3A | Completed Test. | 07-03-76 | 3.00 |
| | Periodic Test 22.3A | Completed Test. | 07-13-76 | 3.00 |
| | Periodic Test 22.3A | Completed Test. | 07-21-76 | 3.00 |
| | Periodic Test 22.3A | Completed Test. | 08-02-76 | 3.00 |
| | Periodic Test 22.3A | Completed Test. | 08-10-76 | 3.00 |
| | Periodic Test 22.3A | Completed Test. | 08-24-76 | 3.00 |
| | Periodic Test 22.3A | Completed Test. | 09-01-76 | 3.00 |
| | Periodic Test 22.3A | Completed Test. | 10-01-76 | 3.00 |
| | Periodic Test 22.3A | Completed Test. | 10-11-76 | 3.00 |
| | Periodic Test 22.3A | Completed Test. | 02-02-77 | 3.00 |
| | Periodic Test 22.3A | Completed Test. | 02-02-77 | 3.00 |
| | Periodic Test 22.3A | Completed Test. | 03-02-77 | 3.00 |
| | Periodic Test 22.3A | Completed Test. | 03-02-77 | 3.00 |
| | Periodic Test 22.3A | Completed Test. | 06-02-77 | 3.00 |
| | Periodic Test 22.3A | Completed Test. | 07-02-77 | 3.00 |
| | Periodic Test 22.3A | Completed Test. | 08-02-77 | 3.00 |
| | Periodic Test 22.3A | Completed Test. | 09-02-77 | 3.00 |

SURRY POWER STATION
UNIT #1
ECCS OUTAGE REPORT

EMERGENCY POWER SYSTEM

| COMPONENT | CAUSE OF OUTAGE | CORRECTIVE ACTION | OUTAGE DATES | OUTAGE DURATION (HOURS) |
|---|---|---|---|---|
| #3 Emergency | Periodic Test 22.3C | Completed Test. | 06-08-76 | 3.00 |
| Diesel Generator | Periodic Test 22.3C | Completed Test. | 06-09-76 | 3.00 |
| | Periodic Test 22.3C | Completed Test. | 06-10-76 | 3.00 |
| | Periodic Test 22.3C | Completed Test. | 07-02-76 | 3.00 |
| | Periodic Test 22.3C | Completed Test. | 08-10-76 | 3.00 |
| | Periodic Test 22.3C | Completed Test. | 09-01-76 | 3.00 |
| | Periodic Test 22.3C | Completed Test. | 10-11-76 | 3.00 |
| | Periodic Test 22.3C | Completed Test. | 03-02-77 | 3.00 |
| | Periodic Test 22.3C | Completed Test. | 07-03-77 | 3.00 |
| | Periodic Test 22.3C | Completed Test. | 07-06-77 | 3.00 |
| | Periodic Test 22.3C | Completed Test. | 09-03-77 | 3.00 |
| | Periodic Test 22.3C | Completed Test. | 11-03-77 | 3.00 |
| | Periodic Test 22.3C | Completed Test. | 01-03-77 | 3.00 |
| | Periodic Test 22.3C | Completed Test. | 03-06-78 | 3.00 |
| | Periodic Test 22.3C | Completed Test. | 04-18-78 | 3.00 |
| | Periodic Test 22.3C | Completed Test. | 04-19-78 | 3.00 |
| | Periodic Test 22.3C | Completed Test. | 01-05-79 | 3.00 |
| | Periodic Test 22.3C | Completed Test. | 11-06-79 | 3.00 |
| | Periodic Test 22.3C | Completed Test. | 02-06-80 | 3.00 |
| | Periodic Test 22.3C | Completed Test. | 06-11-80 | 3.00 |
| | Periodic Test 22.3C | Completed Test. | 07-06-80 | 3.00 |
| | Periodic Test 22.3C | Completed Test. | 07-25-80 | 3.00 |
| | Periodic Test 22.3C | Completed Test. | 09-07-80 | 3.00 |
| NOTE: The following Periodic Tests were accomplished using Unit 2 Periodic Test. However, only Unit 1 was operating | | | | |
| | Periodic Test 22.3C | Completed Test. | 10-05-76 | 3.00 |
| | Periodic Test 22.3C | Completed Test. | 04-06-78 | 3.00 |
| | Periodic Test 22.3C | Completed Test. | 10-06-78 | 3.00 |
| | Periodic Test 22.3C | Completed Test. | 02-06-79 | 3.00 |
| | Periodic Test 22.3C | Completed Test. | 12-06-79 | 3.00 |

SURRY POWER STATION
UNIT #1
ECCS OUTAGE REPORT

EMERGENCY POWER SYSTEM

| COMPONENT | CAUSE OF OUTAGE | CORRECTIVE ACTION | OUTAGE DATES | OUTAGE DURATION (HOURS) |
|---|---|---|---|---|
| #1 Emergency Diesel Generator | Periodic Test 22.3A | Completed Test. | 10-02-77 | 3.00 |
| | Periodic Test 22.3A | Completed Test. | 11-02-77 | 3.00 |
| | Periodic Test 22.3A | Completed Test. | 01-02-78 | 3.00 |
| | Periodic Test 22.3A | Completed Test. | 01-03-78 | 3.00 |
| | Periodic Test 22.3A | Completed Test. | 02-02-78 | 3.00 |
| | Periodic Test 22.3A | Completed Test. | 03-02-78 | 3.00 |
| | Periodic Test 22.3A | Completed Test. | 04-03-78 | 3.00 |
| | Periodic Test 22.3A | Completed Test. | 04-19-78 | 3.00 |
| | Periodic Test 22.3A | Completed Test. | 08-02-78 | 3.00 |
| | Periodic Test 22.3A | Completed Test. | 09-02-78 | 3.00 |
| | Periodic Test 22.3A | Completed Test. | 10-02-78 | 3.00 |
| | Periodic Test 22.3A | Completed Test. | 11-02-78 | 3.00 |
| | Periodic Test 22.3A | Completed Test. | 11-02-78 | 3.00 |
| | Periodic Test 22.3A | Completed Test. | 12-02-78 | 3.00 |
| | Periodic Test 22.3A | Completed Test. | 01-04-79 | 3.00 |
| | Periodic Test 22.3A | Completed Test. | 01-15-79 | 3.00 |
| | Periodic Test 22.3A | Completed Test. | 02-02-79 | 3.00 |
| | Periodic Test 22.3A | Completed Test. | 03-02-79 | 3.00 |
| | Periodic Test 22.3A | Completed Test. | 11-02-79 | 3.00 |
| | Periodic Test 22.3A | Completed Test. | 12-02-79 | 3.00 |
| | Periodic Test 22.3A | Completed Test. | 12-14-79 | 3.00 |
| | Periodic Test 22.3A | Completed Test. | 02-03-80 | 3.00 |
| | Periodic Test 22.3A | Completed Test. | 06-10-80 | 3.00 |
| | Periodic Test 22.3A | Completed Test. | 07-02-80 | 3.00 |
| | Periodic Test 22.3A | Completed Test. | 07-25-80 | 3.00 |
| | Periodic Test 22.3A | Completed Test. | 09-02-80 | 3.00 |
| | Periodic Test 22.3A | Completed Test. | 09-03-80 | 3.00 |
| | Periodic Test 22.3A | Completed Test. | 07-27-76 | 3.00 |
| #3 Emergency | Gauge in cooling water system blew out. | Replaced gauge. | 08-02-78 | 1.00 |
| | Periodic Test 22.3C | Completed Test. | 01-02-76 | 3.00 |
| | Periodic Test 22.3C | Completed Test. | 03-03-76 | 3.00 |
| | Periodic Test 22.3C | Completed Test. | 05-06-76 | 3.00 |
| | Periodic Test 22.3C | Completed Test. | 05-21-76 | 3.00 |
| | Periodic Test 22.3C | Completed Test. | 06-07-76 | 3.00 |

SURRY POWER STATION
UNIT #1
ECCS OUTAGE REPORT

SAFETY INJECTION SYSTEM

| COMPONENT | CAUSE OF OUTAGE | CORRECTIVE ACTION | OUTAGE DATES | OUTAGE DURATION (HOURS) |
|---|---|---|---|---|
| Boron Injection Tank | Recirculation of BIT stopped to allow repair of 1-CH-85. | Replaced valve 1-CH-85. | 03-26-76 | 0.20 |
| | Recirculation of BIT stopped to allow repair of 1-CH-125. | Repaired valve 1-CH-125. | 08-23-76 | 0.58 |
| | Recirculation of BIT stopped to allow installation and subsequent removal of a blind flange on line 1"-CH-56-152. | The flange was installed to clear a suspected Boric Acid Plug - no obstruction was found. | 09-17-76 | 0.13 |
| | BIT boron concentration less than T.S. due to erroneous valve line-up. | Corrected valve line-up, Recirc. BIT with "B" Boric Acid Storage Tank. | 11-29-78 | 2.00 |
| | Recirculation of BIT stopped to allow repair of valves 1-CH-83 and 1-CH-95. | Replaced diaphragms. | 01-26-77 | 0.23 |
| | Recirculation of BIT stopped to allow repair of valve 1-CH-98. | Replaced diaphragm. | 02-20-77 | 0.33 |
| | Recirculation of BIT stopped to allow repairs to 1-CH-98. | Replaced diaphragm. | 04-07-78 | 0.05 |
| 1-CH-P-2A | Pump shaft broken. | Replaced pump shaft. | 01-23-80 | 13.00 |
| 1-SI-TK-1A | Moisture in air line to HCV-1852A caused a delay in closing of the valve allowing the "A" accumulator level to decrease below T.S. levels. | Cleared air line and refilled accumulator | 02-06-80 | 0.42 |
| 1-SI-TK-1B | Check valves 1-SI-128 and 1-SI-130 leaking by causing dilution of "B" accumulator. | Initiated recirculation of "B" accumulator with RWST to increase boron concentration. | 07-02-76 | 1.33 |
| 1-SI-TK-1C | When attempting to pressurize the accumulator, a partially open valve (HCV-1936) and a body to bonnet leak on valve HCV-1549, caused acc. pressure to decrease to 540 psig. | Repaired HCV-1549 and HCV-1936. | 09-09-76 | 4.55 |
| CLS | Jumper used to defeat CLS not removed after refueling. | Removed jumper. | 07-21-76 | 3.50 |

SURRY POWER STATION
UNIT #1
ECCS OUTAGE REPORT

SAFETY INJECTION SYSTEM

| COMPONENT | CAUSE OF OUTAGE | CORRECTIVE ACTION | OUTAGE DATES | OUTAGE DURATION (HOURS) |
|---|---|---|---|---|
| RWST | Volume of tank specified by A.E. different than that assumed in the order for modification of license dated 06-29-78. | Calculated correct level and filled RWST to greater than this level. | 08-19-80 | |
| | Periodic Test 18.7 | Completed Test. | 02-05-80 | 0.50 |
| | Periodic Test 18.7 | Completed Test. | 08-11-80 | 0.50 |
| | Periodic Test 18.7 | Completed Test. | 09-05-80 | 0.50 |

SURRY POWER STATION
UNIT #1
ECCS OUTAGE REPORT

RECIRCULATION SPRAY SYSTEM

| COMPONENT | CAUSE OF OUTAGE | CORRECTIVE ACTION | OUTAGE DATES | OUTAGE DURATION (HOURS) |
|---|---|---|---|---|
| 1-RS-6 | Recirculation valve left open. | Closed valve. | 09-09-80 | 0.25 |
| | Periodic Test 17.3 | Completed Test. | 01-03-76 | 4.50 |
| | Periodic Test 17.3 | Completed Test. | 02-03-76 | 4.50 |
| | Periodic Test 17.3 | Completed Test. | 03-03-76 | 4.50 |
| | Periodic Test 17.3 | Completed Test. | 04-08-76 | 4.50 |
| | Periodic Test 17.3 | Completed Test. | 06-02-76 | 4.50 |
| | Periodic Test 17.3 | Completed Test. | 07-03-76 | 4.50 |
| | Periodic Test 17.3 | Completed Test. | 08-04-76 | 4.50 |
| | Periodic Test 17.3 | Completed Test. | 09-04-76 | 4.50 |
| | Periodic Test 17.3 | Completed Test. | 10-05-76 | 4.50 |
| | Periodic Test 17.3 | Completed Test. | 01-19-77 | 4.50 |
| | Periodic Test 17.3 | Completed Test. | 02-03-77 | 4.50 |
| | Periodic Test 17.3 | Completed Test. | 03-03-77 | 4.50 |
| | Periodic Test 17.3 | Completed Test. | 06-04-77 | 4.50 |
| | Periodic Test 17.3 | Completed Test. | 07-03-77 | 4.50 |
| | Periodic Test 17.3 | Completed Test. | 08-03-77 | 4.50 |
| | Periodic Test 17.3 | Completed Test. | 09-04-77 | 4.50 |
| | Periodic Test 17.3 | Completed Test. | 10-09-77 | 4.50 |
| | Periodic Test 17.3 | Completed Test. | 11-06-77 | 4.50 |
| | Periodic Test 17.3 | Completed Test. | 01-05-78 | 4.50 |
| | Periodic Test 17.3 | Completed Test. | 02-03-78 | 4.50 |
| | Periodic Test 17.3 | Completed Test. | 03-10-78 | 4.50 |
| | Periodic Test 17.3 | Completed Test. | 04-16-78 | 4.50 |
| | Periodic Test 17.3 | Completed Test. | 07-15-78 | 4.50 |
| | Periodic Test 17.3 | Completed Test. | 10-16-78 | 4.50 |
| | Periodic Test 17.3 | Completed Test. | 11-15-78 | 4.50 |
| | Periodic Test 17.3 | Completed Test. | 01-16-79 | 4.50 |
| | Periodic Test 17.3 | Completed Test. | 02-16-79 | 4.50 |
| | Periodic Test 17.3 | Completed Test. | 11-15-79 | 4.50 |
| | Periodic Test 17.3 | Completed Test. | 12-15-79 | 4.50 |
| | Periodic Test 17.3 | Completed Test. | 01-06-80 | 4.50 |
| | Periodic Test 17.3 | Completed Test. | 02-15-80 | 4.50 |
| | Periodic Test 17.3 | Completed Test. | 05-15-80 | 4.50 |
| | Periodic Test 17.3 | Completed Test. | 06-15-80 | 4.50 |
| | Periodic Test 17.3 | Completed Test. | 08-15-80 | 4.50 |
| | Periodic Test 17.3 | Completed Test. | 09-04-80 | 4.50 |

SURRY POWER STATION
UNIT #1
ECCS OUTAGE REPORT

COMPONENT COOLING SYSTEM

| COMPONENT | CAUSE OF OUTAGE | CORRECTIVE ACTION | OUTAGE DATES | OUTAGE DURATION (HOURS) |
|---|---|---|---|---|
| 1-CC-P-2A | Motor bearing failed. | Replaced bearings. | 02-08-77 | 28.00 |
| 1-CC-P-2B | Pump bearing failed. | Replaced bearings. | 10-09-78 | 20.17 |

SURRY POWER STATION
UNIT #1
ECCS OUTAGE REPORT

SERVICE WATER SYSTEM

| COMPONENT | CAUSE OF OUTAGE | CORRECTIVE ACTION | OUTAGE DATES | OUTAGE DURATION (HOURS) |
|---|---|---|---|---|
| 1-SW-132 | Stem corroded causing valve disc to drop and stop flow from Charging Pump intermediate Seal Cooler 1-SW-E-1B. | Replaced valve stem. | 03-05-78 | 0.75 |
| 1-SW-110 | Stem corroded causing valve disc to drop and stop flow to Charging Pump Intermediate Seal Cooler 1-SW-E-1B. | Replaced valve stem. | 12-04-78 | 0.83 |
| 1-SW-107 | Stem corroded allowing valve disc to drop and stop flow to 1-SW-P-10A Charging Pump Service Water Pump. | Replaced valve. | 11-27-79 | 12.50 |

SURRY POWER STATION
UNIT #2
ECCS OUTAGE REPORT

EMERGENCY POWER SYSTEM

| COMPONENT | CAUSE OF OUTAGE | CORRECTIVE ACTION | OUTAGE DATES | OUTAGE DURATION (HOURS) |
|---|---|---|---|---|
| #3 Emergency Diesel Generator | Air motor worn out. | Rebuilt motor. | 11-15-80 | 4.32 |
| #2 Emergency Diesel Generator | Periodic Test 22.3B | Completed Test. | 01-01-76 | 3.00 |
| | Periodic Test 22.3B | Completed Test. | 02-01-76 | 3.00 |
| | Periodic Test 22.3B | Completed Test. | 03-01-76 | 3.00 |
| | Periodic Test 22.3B | Completed Test. | 04-01-76 | 3.00 |
| | Periodic Test 22.3B | Completed Test. | 06-03-76 | 3.00 |
| | Periodic Test 22.3B | Completed Test. | 07-21-76 | 3.00 |
| | Periodic Test 22.3B | Completed Test. | 07-01-76 | 3.00 |
| | Periodic Test 22.3B | Completed Test. | 01-01-77 | 3.00 |
| | Periodic Test 22.3B | Completed Test. | 02-01-77 | 3.00 |
| | Periodic Test 22.3B | Completed Test. | 05-01-77 | 3.00 |
| | Periodic Test 22.3B | Completed Test. | 06-01-77 | 3.00 |
| | Periodic Test 22.3B | Completed Test. | 07-01-77 | 3.00 |
| | Periodic Test 22.3B | Completed Test. | 08-01-77 | 3.00 |
| | Periodic Test 22.3B | Completed Test. | 09-02-77 | 3.00 |
| | Periodic Test 22.3B | Completed Test. | 11-01-77 | 3.00 |
| | Periodic Test 22.3B | Completed Test. | 12-02-77 | 3.00 |
| | Periodic Test 22.3B | Completed Test. | 12-04-77 | 3.00 |
| | Periodic Test 22.3B | Completed Test. | 01-01-78 | 3.00 |
| | Periodic Test 22.3B | Completed Test. | 01-03-78 | 3.00 |
| | Periodic Test 22.3B | Completed Test. | 02-01-78 | 3.00 |
| | Periodic Test 22.3B | Completed Test. | 03-04-78 | 3.00 |
| | Periodic Test 22.3B | Completed Test. | 05-04-78 | 3.00 |
| | Periodic Test 22.3B | Completed Test. | 05-11-78 | 3.00 |
| | Periodic Test 22.3B | Completed Test. | 05-24-78 | 3.00 |
| | Periodic Test 22.3B | Completed Test. | 06-05-78 | 3.00 |
| | Periodic Test 22.3B | Completed Test. | 07-04-78 | 3.00 |
| | Periodic Test 22.3B | Completed Test. | 08-04-78 | 3.00 |
| | Periodic Test 22.3B | Completed Test. | 09-04-78 | 3.00 |
| | Periodic Test 22.3B | Completed Test. | 10-04-78 | 3.00 |
| | Periodic Test 22.3B | Completed Test. | 12-04-78 | 3.00 |
| | Periodic Test 22.3B | Completed Test. | 01-04-79 | 3.00 |
| | Periodic Test 22.3B | Completed Test. | 01-15-79 | 3.00 |
| | Periodic Test 22.3B | Completed Test. | 01-18-79 | 3.00 |
| | Periodic Test 22.3B | Completed Test. | 02-04-79 | 3.00 |

SURRY POWER STATION
UNIT #2
ECCS OUTAGE REPORT

EMERGENCY POWER SYSTEM

| COMPONENT | CAUSE OF OUTAGE | CORRECTIVE ACTION | OUTAGE DATES | OUTAGE DURATION (HOURS) |
|---|---|---|---|---|
| #2 Emergency Diesel Generator | Periodic Test 22.3B | Completed Test. | 08-07-80 | 3.00 |
| | Periodic Test 22.3B | Completed Test. | 09-04-80 | 3.00 |
| | Periodic Test 22.3B | Completed Test. | 11-05-80 | 3.00 |
| | Periodic Test 22.3B | Completed Test. | 11-15-80 | 3.00 |
| | Periodic Test 22.3B | Completed Test. | 11-16-80 | 3.00 |
| | Periodic Test 22.3B | Completed Test. | 11-25-80 | 3.00 |
| | Periodic Test 22.3B | Completed Test. | 12-01-80 | 3.00 |
| | Periodic Test 22.3B | Completed Test. | 12-09-80 | 3.00 |
| #3 Emergency Diesel Generator | Periodic Test 22.3C | Completed Test. | 02-03-76 | 3.00 |
| | Periodic Test 22.3C | Completed Test. | 04-05-76 | 3.00 |
| | Periodic Test 22.3C | Completed Test. | 06-04-76 | 3.00 |
| | Periodic Test 22.3C | Completed Test. | 08-03-76 | 3.00 |
| | Periodic Test 22.3C | Completed Test. | 02-03-77 | 3.00 |
| | Periodic Test 22.3C | Completed Test. | 06-03-77 | 3.00 |
| | Periodic Test 22.3C | Completed Test. | 08-03-77 | 3.00 |
| | Periodic Test 22.3C | Completed Test. | 10-02-77 | 3.00 |
| | Periodic Test 22.3C | Completed Test. | 12-03-77 | 3.00 |
| | Periodic Test 22.3C | Completed Test. | 02-01-78 | 3.00 |
| | Periodic Test 22.3C | Completed Test. | 06-05-78 | 3.00 |
| | Periodic Test 22.3C | Completed Test. | 08-06-78 | 3.00 |
| | Periodic Test 22.3C | Completed Test. | 11-03-78 | 3.00 |
| | Periodic Test 22.3C | Completed Test. | 12-06-78 | 3.00 |
| | Periodic Test 22.3C | Completed Test. | 01-16-79 | 3.00 |
| | Periodic Test 22.3C | Completed Test. | 01-17-79 | 3.00 |
| | Periodic Test 22.3C | Completed Test. | 10-06-80 | 3.00 |
| | Periodic Test 22.3C | Completed Test. | 11-24-80 | 3.00 |
| | Periodic Test 22.3C | Completed Test. | 12-06-80 | 3.00 |
| NOTE: The following Periodic Tests were accomplished using Unit 1 Periodic Test. However, only Unit 2 was operating | | | | |
| #3 Emergency Diesel Generator | Periodic Test 22.3C | Completed Test. | 01-03-77 | 3.00 |
| | Periodic Test 22.3C | Completed Test. | 05-03-77 | 3.00 |
| | Periodic Test 22.3C | Completed Test. | 05-06-78 | 3.00 |
| | Periodic Test 22.3C | Completed Test. | 06-06-78 | 3.00 |
| | Periodic Test 22.3C | Completed Test. | 09-07-78 | 3.00 |
| | Periodic Test 22.3C | Completed Test. | 08-07-80 | 3.00 |
| | Periodic Test 22.3C | Completed Test. | 09-07-80 | 3.00 |
| | Periodic Test 22.3C | Completed Test. | 11-05-80 | 3.00 |

SURRY POWER STATION
UNIT #2
ECCS OUTAGE REPORT

SAFETY INJECTION SYSTEM

| COMPONENT | CAUSE OF OUTAGE | CORRECTIVE ACTION | OUTAGE DATES | OUTAGE DURATION (HOURS) |
|---|---|---|---|---|
| Boron Injection Tank | Leakage through MOV-2867C and D caused a dilution of the BIT. | Initiated recirculation of BIT with "B" Boric Acid Storage Tank to increase boron concentration. | 11-09-77 | 0.67 |
| | The Contents of the Batch Tank were transferred to the "C" Boric Acid Storage Tank and the BIT causing the boric acid concentration to rise above T.S. limits. | Dilute to the required concentration. | 08-22-80 | 1.12 |
| | Improper valve lineup caused a dilution of BIT and "C" BAST. | Increased boric acid concentration. | 11-05-80 | 0.96 |
| | Leakage through MOV-2876A and/or MOV-2867B diluting BIT to less T.S. low limit. | Closed valves manually and recirculated BIT with "C" Boric Acid Storage Tank to increase concentration. | 12-30-76 | 3.50 |
| | An obstruction stopped recirculation of the BIT. | Established recirculation with "B" Boric Acid Storage Tank. | 01-26-77 | 1.75 |
| | Dilution of BIT because of a cracked diaphragm on 1-CH-144. | Recirculate BIT with "B" Boric Acid Storage Tank and replaced diaphragm on valve 1-CH-144. | 08-15-77 | 1.20 |
| | Leakage through MOV-2867C and D diluting the BIT. | Closed valves manually and increased boric acid concentration of BIT. | 10-31-77 | 0.57 |
| 2-SI-TK-1B | When attempting to pressurize the accumulator, the low $N_2$ header pressure caused a further reduction of accumulator pressure. | Repaired Leaking $N_2$ fittings and repressurized. | 08-26-80 | 1.58 |
| | Contents of the boric acid bottoms tank were transferred to the "C" Boric Acid Storage Tank & thence the BIT causing the boric acid concentration to rise above T.S. limits. | Dilute to required concentrations. | 08-15-80 | 0.81 |

SURRY POWER STATION
UNIT #2
ECCS OUTAGE REPORT

SAFETY INJECTION SYSTEM

| COMPONENT | CAUSE OF OUTAGE | CORRECTIVE ACTION | OUTAGE DATES | OUTAGE DURATION (HOURS) |
|---|---|---|---|---|
| 2-SI-TK-1B | The accumulator was filled to a level greater than allowed by T.S. | Repaired instrumentation and drained accumulator. | 08-16-80 | 4.03 |
| | Check valve 2-SI-127 leaking through allowing dilution of "B" accumulator. | Isolated valve and recirculated accumulator with RWST. | 11-08-77 | 3.92 |
| 2-SI-TK-1C | Check valves 2-SI-147 and 2-SI-145 leaking allowing dilution of "C" accumulator. | Initiated recirculation of accumulator with RWST to increase boron concentration. | 08-13-76 | 2.75 |
| 2-SI-P-1A | Rain water from a leaking roof hatch grounded out the motor connection box. | Repaired roof hatch and dried motor connections. | 11-05-80 | 4.83 |
| MOV-2862B | Valve binding caused motor overloads to function. | Adjusted valve and reset overloads. | 06-15-76 | 21.92 |
| 4"-CH-389 | Pipe support for alternate charging header not installed as designed. | Modified to conform to drawings. | 10-15-80 | 2.12 |
| | Periodic Test 18.7 | Completed Test. | 08-09-80 | 0.50 |
| | Periodic Test 18.7 | Completed Test. | 09-05-80 | 0.50 |
| | Periodic Test 18.7 | Completed Test. | 09-07-80 | 0.50 |
| | Periodic Test 18.7 | Completed Test. | 09-15-80 | 0.50 |
| | Periodic Test 18.7 | Completed Test. | 10-07-80 | 0.50 |
| | Periodic Test 18.7 | Completed Test. | 11-07-80 | 0.50 |
| | Periodic Test 18.7 | Completed Test. | 11-19-80 | 0.50 |
| | Periodic Test 18.7 | Completed Test. | 11-21-80 | 0.50 |
| | Periodic Test 18.7 | Completed Test. | 12-07-80 | 0.50 |
| | Periodic Test 18.7 | Completed Test. | 12-12-80 | 0.50 |
| | Periodic Test 18.7 | Completed Test. | 12-13-80 | 0.50 |
| | Periodic Test 18.7 | Completed Test. | 12-13-80 | 0.50 |
| | Periodic Test 18.7 | Completed Test. | 12-27-80 | 0.50 |

SURRY POWER STATION
UNIT #2
ECCS OUTAGE REPORT

RECIRCULATION SPRAY SYSTEM

| COMPONENT | CAUSE OF OUTAGE | CORRECTIVE ACTION | OUTAGE DATES | OUTAGE DURATION (HOURS) |
|---|---|---|---|---|
| Recirculation Spray | Periodic Test 17.3 | Completed Test. | 01-26-76 | 4.50 |
| | Periodic Test 17.3 | Completed Test. | 02-25-76 | 4.50 |
| | Periodic Test 17.3 | Completed Test. | 06-24-76 | 4.50 |
| | Periodic Test 17.3 | Completed Test. | 07-25-76 | 4.50 |
| | Periodic Test 17.3 | Completed Test. | 08-25-76 | 4.50 |
| | Periodic Test 17.3 | Completed Test. | 12-30-76 | 4.50 |
| | Periodic Test 17.3 | Completed Test. | 01-24-77 | 4.50 |
| | Periodic Test 17.3 | Completed Test. | 04-23-77 | 4.50 |
| | Periodic Test 17.3 | Completed Test. | 05-25-77 | 4.50 |
| | Periodic Test 17.3 | Completed Test. | 06-24-77 | 4.50 |
| | Periodic Test 17.3 | Completed Test. | 07-24-77 | 4.50 |
| | Periodic Test 17.3 | Completed Test. | 12-23-77 | 4.50 |
| | Periodic Test 17.3 | Completed Test. | 01-23-78 | 4.50 |
| | Periodic Test 17.3 | Completed Test. | 02-24-78 | 4.50 |
| | Periodic Test 17.3 | Completed Test. | 04-25-78 | 4.50 |
| | Periodic Test 17.3 | Completed Test. | 06-25-78 | 4.50 |
| | Periodic Test 17.3 | Completed Test. | 06-30-78 | 4.50 |
| | Periodic Test 17.3 | Completed Test. | 08-24-78 | 4.50 |
| | Periodic Test 17.3 | Completed Test. | 09-24-78 | 4.50 |
| | Periodic Test 17.3 | Completed Test. | 10-23-78 | 4.50 |
| | Periodic Test 17.3 | Completed Test. | 12-23-78 | 4.50 |
| | Periodic Test 17.3 | Completed Test. | 01-24-79 | 4.50 |
| | Periodic Test 17.3 | Completed Test. | 08-23-80 | 4.50 |
| | Periodic Test 17.3 | Completed Test. | 09-10-80 | 4.50 |
| | Periodic Test 17.3 | Completed Test. | 10-12-80 | 4.50 |
| | Periodic Test 17.3 | Completed Test. | 11-10-80 | 4.50 |
| | Periodic Test 17.3 | Completed Test. | 11-25-80 | 4.50 |
| | Periodic Test 17.3 | Completed Test. | 12-11-80 | 4.50 |

SURRY POWER STATION
UNIT #2
ECCS OUTAGE REPORT

CHARGING PUMP COMPONENT COOLING SYSTEM

| COMPONENT | CAUSE OF OUTAGE | CORRECTIVE ACTION | OUTAGE DATES | OUTAGE DURATION (HOURS) |
|---|---|---|---|---|
| 2-CC-P-2B | Pump worn beyond service limits. | Rebuilt pump. | 11-13-77 | 7.00 |

SURRY POWER STATION
UNIT #2
ECCS OUTAGE REPORT

SERVICE WATER SYSTEM

| COMPONENT | CAUSE OF OUTAGE | CORRECTIVE ACTION | OUTAGE DATES | OUTAGE DURATION (HOURS) |
|---|---|---|---|---|
| MOV-SW-203, A,B,C,D. | Drain valves inadvertently left open flooded to the valve pit. | Drained pit and dried motors. | 10-11-77 | 21.50 |
| | Valve stem corroded allowing disc to drop and obstruct service water flow to the intermediate seal cooler. | Replaced valve stem. | 03-05-78 | 1.33 |
| 2-SW-P-10A | Motor bearings failed. | Replaced motor. | 04-10-78 | 3.00 |
| | Eel found in impeller. | Removed eel. | 10-12-80 | 22.42 |
| | Material entrained in water was deposited on pump impeller causing loss of discharge pressure. | Cleaned pump | 11-14-80 | 6.63 |
| 2-SW-P-10B | Trash in pump caused a loss of flow. | Cleaned pump. | 09-24-80 | 12.00 |
| | Trash on strainer caused a reduction in discharge pressure. | Cleaned strainer. | 11-05-80 | 0.62 |
| 2-SW-113 | The valve disc was found to be missing from the check valve for 2-SW-P-10A. | Replaced check valve. | 09-26-80 | 6.40 |

## III.A.1.2 UPGRADE EMERGENCY SUPPORT FACILITIES

### Position

Each operating nuclear power plant shall maintain an onsite technical support center (TSC) separate from and in close proximity to the control room that has the capability to display and transmit plant status to those individuals who are knowledgeable of and responsible for engineering and management support of reactor operations in the event of an accident. The center shall be habitable to the same degree as the control room for postulated accident conditions. The licensee shall revise his emergency plans as necessary to incorporate the role and location of the TSC. Records that pertain to the as-built conditions and layout of structures, systems, and components shall be readily available to personnel in the TSC.

An operational support center (OSC) shall be established separate from the control room and other emergency response facilities as a place where operations support personnel can assemble and report in an emergency situation to receive instructions from the operating staff. Communications shall be provided between the OSC, TSC, EOF, and control room.

An emergency operations facility (EOF) will be operated by the licensee for continued evaluation and coordination of all licensee activities related to an emergency having or potentially having environmental consequences.

### Changes to Previous Requirements and Guidance

(1) NUREG-0696, "Functional Criteria for Emergency Response Facilities," to be issued shortly, will provide more detailed design and functional criteria.

(2) Table III.A.1.2-1 (Table B-1 to NUREG-0654, Revision 1), establishes staffing levels for emergency situations. The revision of NUREG-0654 establishes staging of staffing for 30 to 60 minutes rather than requiring capability for required augmentation at 30 minutes. The implementation schedule for licensed operators and STA on shift shall be as specified in Task Action Item I.A.1.3. Any deficiencies in the other staffing requirements of the table must be capable of augmentation within 30 minutes by September 1, 1981 and such deficiencies must be fully removed by July 1, 1982.

(3) Implementation schedule has been changed.

(4) Establishes a requirement to submit emergency response facilities (ERFs) conceptual design information by June 1, 1981.

(5) Establishes guidance on EOF location and habitability

Clarification

NUREG-0696 was issued in draft for comment. The staff has analyzed the comments, prepared the final version for Commission review, and will issue the final version of NUREG-0696 in the near future.

NUREG-0696, "Functional Criteria for Emergency Response Facilities," will provide more detail design and functional criteria than previously prescribed. The operational date for the final emergency response facilities has been changed to October 1, 1982. The interim TSC and EOF completed by January 1, 1980 shall continue to be operated until the upgraded facilities become fully operational.

An emergency operations facility (EOF) will be operated by the licensee for continued evaluation and coordination of all licensee activities related to an emergency having or potentially having environmental consequences. The criteria regarding the location and habitability of the EOF is given in Table III.A.1.2-2.

Applicability

This requirement applies to all operating reactors and applicants for operating license.

Implementation

For operating reactors, the upgraded emergency response facilities conceptual design shall be submitted by June 1, 1981. For operating license applications, such design information should be provided in connection with the OL review process. The upgraded facilities shall be operational by October 1, 1982, for all facilities licensed for operation prior to that date. For OL expected to be issued after October 1, 1982, the upgraded facilities shall be operational prior to receiving an operating license.

Type of Review

A pre-implementation review of the conceptual design submittals will be performed. A post-implementation review will be performed for the October 1, 1982 requirement.

Documentation Required

Facility conceptual design description shall be provided by June 1, 1981 including:

(1) Task functions of the individuals required to report to the TSC and EOF upon activation and for each emergency class; and

(2) Descriptions of TSC instrumentation, instrument quality, instrument accuracy and reliability.

(3) Descriptions of TSC power supply systems, power supply quality, reliability and availability, and consequences of power supply interruption.

(4) Descriptions of the design of the TSC data display systems, plant records and data available and record management systems.

(5) Descriptions of the data transmission system to be installed between the TSC and control room.

(6) Description of data to be provided to the EOF.

Technical Specification Changes Required

Changes to technical specifications will be required.

References

NUREG-0696, "Functional Criteria for Emergency Response Facilities," to be issued.

NUREG-0654 (FEMA-REP-1), Revision 1 "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants," November 1980.

NUREG-0578, "TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendations," July 1979.

NUREG-0585, "TMI-2 Lessons Learned Task Force Final Report," October 1979.

Letter from D. G. Eisenhut, NRC, to All Power Reactor Licensees, "Discussion of Upgraded Emergency Plans and Facilities," dated October 10, 1979.

Letter from H. R. Denton, NRC, to All Operating Plants, "Discussion of Lessons Learned Short-Term Requirements," dated October 30, 1979.

Letter from D. G. Eisenhut, NRC, to All Licensees, "Clarification of NRC Requirements for Emergency Response Facilities at Each Site," dated April 25, 1980.

TABLE III.A.1.2-1

MINIMUM STAFFING REQUIREMENTS FOR NRC LICENSEES
FOR NUCLEAR POWER PLANT EMERGENCIES

| Major Functional Area | Major Tasks | Position Title or Expertise | Capability for Additions | | |
|---|---|---|---|---|---|
| | | | On Shift* | 30 min. | 60 min. |
| Plant Operations and Assessment of Operational Aspects | | Shift supervisor (SRO) | 1 | -- | -- |
| | | Shift foreman (SRO) | 1 | -- | -- |
| | | Control-room operators | 2 | -- | -- |
| | | Auxiliary operators | 2 | | |
| Emergency Direction and Control (Emergency Coordinator)*** | | Shift technical advisor, shift supervisor, or designated facility manager | 1** | -- | -- |
| Notification/ Communication**** | Notify licensee, state, local, and federal personnel & maintain communication | | 1 | 1 | 2 |
| Radiological Accident Assessment and Support of Operational Accident Assessment | Emergency operations facility (EOF) director | Senior manager | -- | -- | 1 |
| | Offsite dose assessment | Senior health physics (HP) expertise | -- | 1 | -- |
| | Offiste surveys | | -- | 2 | 2 |
| | Onsite (out-of-plant) | | -- | 1 | 1 |
| | Inplant surveys | HP technicians | 1 | 1 | 1 |
| | Chemistry/radio-chemistry | Rad/chem technicians | 1 | -- | 1 |

NOTE: Source of this table is NUREG-0654, "Functional Criteria for Emergency Response Facilities."

TABLE III.A.1.2-1 (CONTINUED)

| Major Functional Area | Major Tasks | Position Title or Expertise | Capability for Additions On Shift* | 30 min. | 60 min. |
|---|---|---|---|---|---|
| Plant System Engineering, Repair and Corrective Actions | Technical support | Shift technical advisor | 1 | -- | -- |
| | | Core/thermal hydraulics | -- | 1 | -- |
| | | Electrical | -- | -- | 1 |
| | | Mechanical | -- | -- | 1 |
| | Repair and corrective actions | Mechanical maintenance/ | 1** | -- | 1 |
| | | Radwaste operator | | | 1 |
| | | Electrical maintenance/ | 1** | 1 | 1 |
| | | instrument and control | | 1 | |
| | | (I&C) technician | -- | 1 | -- |
| Protective Actions (In-Plant) | Radiation protection:<br>a. Access control<br>b. HP Coverage for repair, corrective actions, search and rescue, first-aid, & firefighting<br>c. Personnel monitoring<br>d. Dosimetry | HP technicians | 2** | 2 | 2 |
| Firefighting | -- | -- | Fire brigade per technical specifications | Local support | |
| Rescue Operations and First-Aid | -- | -- | 2** | Local support | |

TABLE III.A.1.2-1 (CONTINUED)

| Major Functional Area | Major Tasks | Position Title or Expertise | Capability for Additions On Shift* | 30 min. | 60 min. |
|---|---|---|---|---|---|
| Site Access Control and Personnel Accountability | Security, firefighting communications, personnel accountability | Security personnel | All per security plan | | |
| | | Total | 10 | 11 | 15 |





* For each unaffected nuclear unit in operation, maintain at least one shift foreman, one control-room operator, and one auxiliary operator except that units sharing a control room may share a shift foreman if all functions are covered.

** May be provided by shift personnel assigned other functions.

*** Overall direction of facility response to be assumed by EOF director when all centers are fully manned. Director of minute-to-minute facility operations remains with senior manager in technical support center or control room.

**** May be performed by engineering aide to shift supervisor.

## EMERGENCY OPERATIONS FACILITY

### Option 1
### Two Facilities

A. Close-in Primary: Reduced Habitability*
- within 10 miles
- protection factor = 5
- ventilation isolation with HEPA (no charcoal)

B. Backup EOF
- between 10-20 miles
- no separate, dedicated facility
- arrangements for portable backup equipment
- strongly recommend location be coordinated with offsite authorities
- continuity of dose projection and decision making capability

### Option 2
### One Facility

- At or Beyond 10 miles
- No special protection factor
- If beyond 20 miles, specific approval required by the Commission, and some provision for NRC site team closer to site
- Strongly recommend location be coordinated with offsite authorities

For both Options:
- located outside security boundary
- space for about 10 NRC employees
- none designed for severe phenomena, e.g., earthquakes

* Habitability requirements are only for that part of the EOF in which dose assessments communications and decision making take place.

If a utility has begun construction of a new building for an EOF that is located within 5 miles, that new facility is acceptable (with less than protection factor of 5 and ventilation isolation with HEPA) provided that a backup EOF similar to "B" in Option 1 is provided.

## III.A.1.2 UPGRADE EMERGENCY SUPPORT FACILITIES

The requirements of the Emergency Response Facilities (ERF's) have been detailed in the final version of NUREG 0696. The schedule for ERF implementation given in Item II.A.1.2 (and NUREG 0696) will be met. Preliminary design information (based on the draft of NUREG 0696) regarding the proposed Vepco ERF's was submitted to the NRC on December 18, 1980. An update of this information will be provided on June 1, 1981.

## III.A.2 IMPROVING LICENSEE EMERGENCY PREPAREDNESS - LONG-TERM

1. The Radiological Emergency Response Plan for North Anna Units 1 and 2 was filed with the Commission on May 1, 1980, and subsequently amended in response to Staff comments. The plan included a description of the program to provide the elements of NUREG 0654, Appendix 2. The NRC Staff's review and approval of the North Anna plan is documented in Supplements 11 and 12 to NUREG 0053, the Safety Evaluation Report for North Anna Unit 2. The Surry emergency plan was submitted for review on June 16, 1980. The first amendment was submitted on January 2, 1981.

   The requirement to submit radiological emergency response plans has been met for both Surry and North Anna.

2. The Emergency Plan Implementing Procedures for both Surry and North Anna were revised. These revisions were submitted for review by the Staff on February 27, 1981.

3. Implementation of the revised radiological response plans was accomplished prior to April 1, 1981.

   A meteorological measurements program which incorporates the features of both element 1 and element 2 of Appendix 2 of NUREG 0654, Rev. 1, including the display of data in the control room, will be operational by the required implementation date. An operable dose calculational methodolgy (DCM) is currently available for emergency use.

   Implementation of element 3, Real-Time Predictions of Atmospheric Effluent Transport and Diffusion, and element 4, Remote Interrogation of the Atmospheric Measurement at Prediction Systems, is related to the installation of equipment to meet the requirements of NUREG 0696, "Functional Criteria for Emergency Response Facilities". Installation of equipment required by NUREG 0696, including new computer capabilities, will not be complete prior to July 1, 1982. Consequently, milestone 4, which requires installation of Emergency Response Facility hardware and software by March 1, 1982, must be deferred to be consistent with NUREG 0696. At that time, the Class A model should be available, in accordance with milestone 5. Futher modifications to the Class A model, as well as implementation of the Class B model, will take place on a schedule to be determined based on ongoing discussions with the Commission.

## III.D.3.4 CONTROL ROOM HABITABILITY REQUIREMENTS

A control room habitability review was conducted for the North Anna control room as a part of the Unit 2 full power licensing review. Based on the reviews conducted in accordance with Standard Review Plan Sections 2.2.1, 2.2.2, 2.2.3 and 6.4, and Regulatory Guide 1.78 and 1.95, we concluded that the control room meets the specifications and guidance in these SRP sections and Regulatory Guides; and therefore, no modifications are required. Since the control room is common to both units, this conclusion applies to both Units 1 and 2.

Additional information required by item III.D.3.4, beyond that which was required for full power licensing was compiled and submitted on December 30, 1980 (Serial No. 1013) for North Anna.

The control room habitiability review for Surry Units 1 and 2 has been partially completed. We have submitted an analysis of habitability with respect to all on-site hazards on January 19, 1981 (Serial No. 036). The special case of chemical shipments along the James River will require additional study. We expect to complete this portion and submit the report by June 30, 1981.

## III.D.3.4 CONTROL ROOM HABITABILITY REQUIREMENTS - EXCEPTIONS

Vepco proposes to submit an analysis of off-site hazards by June 30, 1981 rather than the required date of January 1, 1981.