Superceded per Rev. 1 To Response To NUREG 0737

6/18/81 50-280

VIRGINIA ELECTRIC AND POWER COMPANY

RESPONSE TO NUREG 0737

POST TMI - REQUIREMENTS

REVISION 0 DECEMBER 10, 1980

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 0696 and NUREG 0700:

To date the final requirements of NUREG 0696 and NUREG 0700 incorporated in NUREG 0737 have not been issued. In previous letters, Vepco has commented on NUREG CR/1580 and NUREG 0696. Vepco commitments to NUREG 0696 and 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 0696 and 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.D.1	Control-room design	1) None at this time - NUREG 0700 has not been issued. 2) Vecpo assumes the licensee submittal date should be "later" rather than 4/82.
I.D.2	Plant safety parameter display console	1) None at this time - NUREG 0696 has not been issued.
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. * 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) Vepco shielding review to the requirements of NUREG 0578 & 0660 did not consider HELB. This additional review is estimated to be complete by June, 1981 but may require additional equipment modifications that may extend beyond July, 1982. 2) Some systems included in the NRC clarifications were excluded from the review. Justifications are provided. 3) 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. 4) Radiation zone maps for personnel access will be revised after all designs are installed.

CLARIFICATION ITEM	SHORTENED TITLE	EXCEPTIONS
II.B.3	Post-accident sampling	NONE
II.B.4	Training for mitigating core damage	NONE
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	1) Installation beyond the requested date of July 1, 1981 could extend to Fall, 1981 if material is not received as scheduled.
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.

CLARIFICATION ITEM	SHORTENED TITLE		EXCEPTIONS
II.F.1	Accident-monitoring		
	1. Noble gas effluents	1)	Low range sensitivity of Main Steam Monitors is restricted by very high background radiation.
		2)	Some design information and procedures will not be available by January 1, 1981 but will be available by the required installation date.
	2. Sampling & analysis of effluents	1)	Final design details and analytical procedures will be available prior to the required implementation date of January 1, 1982 rather than the required documentation date of January 1, 1981 based on the availability of vendor information.
	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	* 1) Report to be submitted March 1, 1981, rather than January 1, 1981.
	.3 SV & RV Failures	NONE
	.5 Auto trip RCP's	NONE
	.9 PID Controller	NONE
	.10 Proposed Anticipatory Trip	NONE
	.11 Use of certain PORV's	NONE
	.12 Anticipatory Trip on Turbine Trip	NONE

CLARIFICATION ITEM	SHORTENED TITLE	EXCEPTIONS
II.K.3	Final recommendations, B&O task force (continued)	
.17	ECC system outages	1) None for North Anna * 2) Report to be submitted by March 1, 1981 for Surry.
.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	1) None at this time - additional clarification has not been issued.
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 * 2) Vepco requires a relaxation of the January 1, 1981 documentation date for Surry. The on-site analysis and off-site analysis will be submitted by January 19 and June 30, 1980, respectively.
Appendix B	Design Criteria	* 1) Equipment does not meet the seismic test results of R.G. 1.100. 2) No additional vendor documentation was required for "extended range" qualification.

CLARIFICATION ITEM	SHORTENED TITLE	EXCEPTIONS
Appendix B	Design Criteria (continued)	* 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:

2nd R - Second refueling outage currently scheduled for January 1, 1981.

SM - Spring, 1981 maintenance outage.

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

Unit 2 Outages:

FP - Fire protection modification outage scheduled for November 1, 1980.

SM - Spring, 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:

SM - Spring, 1981 maintenance outage.

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 2nd R	NORTH ANNA #1 SM	NORTH ANNA #1 3rd R	NORTH ANNA #2 FP	NORTH ANNA #2 SM	NORTH ANNA #2 1st R	SURRY #1 SGR	SURRY #2 SM	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
	* 1/82	NA2	Onsite	Yes						X					DC-79-69B - Unit 2
	7/82	S1	1/81	Yes							X				DC-80-S29 - Rx Vessel head removal required.
	7/82	S2	1/81	Yes										X	
Plant Shielding (II.B.2)															
Hydrogen Recombiner Vault Shielding	1/82	NA1	Onsite	No											DC-80-S36 North Anna only.
	* 1/81	NA2	Onsite	No											
S.W. Rad Monitor Pump Mods.	1/82	NA1	12/80	Yes	X										DC-80-S25A (motors & seals)
	* 1/81	NA2	12/80	Yes					X						
	1/82	S1	~3/81	No											DC-80-57 (motors)
	1/82	S2	~3/81	No											
S.W. Rad Monitor Shielding	1/82	NA1	1/81	No											DC-80-S33
	* 1/81	NA2	1/81	No											
	1/82	S1	~3/81	No											DC-80-S56 Piping & electrical to be installed per DC-80-S57
	1/82	S2	~3/81	No											

NUREG-0737 MODS	REQ DATE	UNIT	EST MAT'L DEL	OUT. REQ	NORTH ANNA #1			NORTH ANNA #2			SURRY #1	SURRY #2			REMARKS
					2nd R	SM	3rd R	FP	SM	1st R	SGR	SM	FM	5th R	
Replacement of Mission Check Valves	1/82	NA1	10/80	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 DC-80-S24B - Unit 2 replace 13
*	1/81	NA2	Onsite	Yes				X	X						DC-80-S24C - Requires 2 unit outage for 2 valves or entry into Tech Specs. action statement.
	1/82	S1	11/80	Yes							X				DC-80-53
	1/82	S2	11/80	Yes								X			
Replace Charging Pump Bearing Gaskets	1/82	S1	Onsite	No											Surry only. DC-80-S58
	1/82	S2	Onsite	No											
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	~2/81	Yes							X				Surry only. DC-80-S67
	1/82	S2	~2/81	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	2/81	Yes							X				Surry only. DC-80-54
	1/82	S2	2/81	Yes								X			

NUREG-0737 MODS	REQ DATE	UNIT	EST MAT'L DEL	OUT. REQ	NORTH ANNA #1 2nd R	NORTH ANNA #1 SM	NORTH ANNA #1 3rd R	NORTH ANNA #2 FP	NORTH ANNA #2 SM	NORTH ANNA #2 1st R	SURRY #1 SGR	SURRY #2 SM	SURRY #2 FM	SURRY #2 5th R	REMARKS
Replace Liners & Seals in Recirc. Spray Plug Valves	1/82	S1	12/80	Yes							X				Surry only. DC-80-S69 (valve vendor Rep. required)
	1/82	S2	12/80	Yes										X	INSTALLATION AFTER 1/82 FOR S2.
Safeguards Area Vent Fan Motors Replacement	1/82	NA1	3/81	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 DC-80-21B Shield Wall DC-80-21C Control Panel/Trickle Htr. (Trickle heater not required by 1/82)
	1/82	NA2	4/81	-											
Post-Accident Sampling (II.B.3)															
RCS & Containment Sampling Facility	1/82	NA1	3/81	Yes	X										DC-80-S48 - Outage required for some containment tie-ins only.
	* 4/81	NA2	3/81	Yes				X							
	1/82	S1	3/81	Yes							X				DC-80-31 - Outage required for some containment tie-ins only.
	1/82	S2	3/81	Yes								X			
Sample System Isolation Valves	1/82	NA1	1/81	Yes	X										DC-80-S32A - Unit 1
	1/82	NA2	1/81	Yes					X						DC-80-S32B - Unit 2
	* 4/81	S1	1/81	Yes							X				DC-80-60A - Unit 1
	1/82	S2	1/81	Yes								X			DC-80-60B - Unit 2

NUREG-0737 MODS	REQ DATE	UNIT	EST MAT'L DEL	OUT. REQ	NORTH ANNA #1 2nd R	NORTH ANNA #1 SM	NORTH ANNA #1 3rd R	NORTH ANNA #2 FP	NORTH ANNA #2 SM	NORTH ANNA #2 1st R	SURRY #1 SGR	SURRY #2 SM	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	1/81	Yes							X				DC-80-78
	1/82	S2	1/81	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	5/81	Yes							X				
	R	S2	5/81	Yes										X	
ECST Level Alarm	7/81	NA1	11/80	C R	X										DC-80-S02 Unit 1 DC-80-S42 Unit 2
	7/81	NA2	11/80	C R				X							
	7/81	S1	11/80	C R							X				DC-80-S37
	7/81	S2	11/80	C R								X			

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

NUREG-0737 MODS		REQ DATE	UNIT	EST MAT'L DEL	OUT. REQ	NORTH ANNA #1 2nd R	NORTH ANNA #1 SM	NORTH ANNA #1 3rd R	NORTH ANNA #2 FP	NORTH ANNA #2 SM	NORTH ANNA #2 1st R	SURRY #1 SGR	SURRY #2 SM	SURRY #2 FM	SURRY #2 5th R	REMARKS
Increased Range of Rad Monitors (II.F.1)																
Containment High Range		1/82	NA1	11/80	Yes	X	X									DC-80-S35A Unit #1 Containment. DC-80-S35B Unit #2 Containment.
	*	1/81	NA2	11/80	Yes				X	X						DC-80-S35C Outside Containment. Non-outage - C.R. Equipment ∿5/81.
		1/82	S1	12/80	Yes							X				Electrical termination problems.
		1/82	S2	12/80	Yes								X			DC-80-S52 - Penetration problem. Non-outage - C.R. Equipment ∿5/81. Electrical termination problems.
Main Steam Effluent		1/82	NA1	∿7/81	No											DC-80-S37B - Electrical upgrade & recorders.
	*	7/81	NA2	∿7/81	No											
		1/82	S1	∿7/81	No											DC-80-64B - Electrical upgrade & recorders.
		1/82	S2	∿7/81	No											
Steam Driven Aux Feed		1/82	NA1	3/81	No											DC-80-S37A
	*	7/81	NA2	3/81	No											
		1/82	S1	3/81	No											DC-80-64C
		1/82	S2	3/81	No											
Process & Vent Rad		1/82	NA1	∿7/81	No											DC-80-S37C
	*	7/81	NA2	∿7/81	No											
		1/82	S1	∿7/71	No											DC-80-64A
		1/82	S2	∿7/81	No											

NUREG-0737 MODS	REQ DATE	UNIT	EST MAT'L DEL	OUT. REQ	NORTH ANNA #1			NORTH ANNA #2			SURRY #1	SURRY #2			REMARKS
					2nd R	SM	3rd R	FP	SM	1st R	SGR	SM	FM	5th R	
Containment Accident Monitoring (II.F.1)															
Containment Pressure	1/82	NA1	1/81	Yes	X										DC-79-S67 - Recorders on PAM&C panel. Outage required for installation of panel. Non-qualified indicators will be used until qualified indicators are available.
	* 1/81	NA2	1/81	Yes				X							
	1/82	S1	12/80	Yes							X				DC-79-60 - Recorders on PAM&C panel. Outage required for installation of panel. Non-qualified indicators will be used until qualified indicators are available.
	1/82	S2	12/80	Yes								X			
Containment Water Level	1/82	S1	11/80	Yes							X				Surry only. Gems transmitters undergoing qualification. May install without qualification.
	1/82	S2	11/80	Yes								X			
Containment Hydrogen Monitor	1/82	NA1	1/81	Yes		X									DC-79-S68
	* 1/81	NA2	1/81	Yes					X						
	1/82	S1	1/81	Yes							X				DC-79-62A
	1/82	S2	1/81	Yes								X			
PAM&C Panel	7/81	NA1	12/80	Yes	X										DC-80-S29 - Installation required for completion of containment hydrogen, pressure & RCS Vent.
	* 1/81	NA2	12/80	Yes					X						
	7/81	S1	1/81	Yes							X				DC-80-SS - Installation required for completion of containment hydrogen, water level, pressure & RCS Vent.
	7/81	S2	1/81	Yes								X			

NUREG-0737 MODS		REQ DATE	UNIT	EST MAT'L DEL	OUT. REQ	NORTH ANNA #1			NORTH ANNA #2			SURRY #1	SURRY #2			REMARKS
						2nd R	SM	3rd R	FP	SM	1st R	SGR	SM	FM	5th R	
Inadequate Core Cooling (II.F.2)																
Redundant RCS Wide	*	1/81	S1	~4/81	Yes							X				Surry only. DC-80-44 Input to sub-cooling meter. WORK ON ELECTRONICS WILL EXTEND BEYOND 1-1-81 DUE TO MATERIAL DELIVERY.
	*	1/81	S2	~4/81	Yes								X			
Reactor Vessel Level		1/82	NA1	~9/81	Yes			X								DC-XX-XXX (Westinghouse level system)
	*	1/81	NA2	~9/81	Yes						X					
		1/82	S1	~9/81	Yes							X				DC-XX-XX Sixth refueling outage if not available for work during SGR. (Westinghouse level system)
		1/82	S1	~9/81	Yes										X	

*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	Submitted
		3. Trained per LL Cat B	1-1-81	No	Yes	Yes	1-1-81	Submitted
		4. Describe long-term program	1-1-81	No	No	No	1-1-81	
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	
		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	TBD	
		b. Revise procedures	1st refueling outage after 1-1-82					
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	
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				4/82 E	Response after issue of NUREG 0700
I.D.2	Plant safety parameter display console	1. Description 2. Installed 3. Fully implemented	TBD TBD TBD				Later	Response after issue of NUREG 0696
II.B.1	Reactor coolant system vents	1. Design vents 2. Install vents (LL Cat B) 3. Procedures	7-1-81 7-1-82 E 1-1-82	No Yes Yes	Yes No No	No Yes Yes	7-1-81 7-1-81 1-1-81 E	Submitted
II.B.2	Plant shielding	1. Review designs 2. Plant modifications (LL Cat B) 3. Equipment qualification	1-1-80 1-1-81 6-30-82 E	No No No	Yes Yes Yes	No No No	1-1-81 1-1-81 submittal if deviation from position 1-1-82	Submitted 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.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	
		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
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

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 (continued)	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
		2. Iodine/particulate sampling	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)	3. Containment high-range monitor	1-1-82	No	Yes	Yes	7-1-81 submittal if deviation from position	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
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	

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	1. Auto PORV isolation						
		a. Design	7-1-81	Yes	No	Yes	7-1-81	If required by II.K.3.2
		b. Test/install	1st refuel 6 mo after staff approval	Yes	No	Yes	7-1-81	
		2. Report on PORV failures	1-1-81 E	No	Yes	No	1-1-81 E	Submittal by 3-1-81
		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	
		b. Modify	3-1-81	Yes	No	Yes	7-1-81	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 E	No	Yes	As required	1-1-81 E	Submittal by 3-1-81
		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	TBD	TBD	TBD	TBD	TBD	
		3. Modifications	TBD	TBD	TBD	TBD	TBD	
III.A.2	Emergency preparedness	1. Upgrade emergency plans to App. E, 10 CFR 50	4-1-81	No	Yes	Yes	1-2-81	Procedures to be submitted 3-1-81
		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 E	
		2.	Modification	1-1-83	No	Yes	Yes	1-1-81 E	

Note

E - Indicates those implementation 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 Units 1 and 2 require the presence of an STA on shift whenever either unit is in operation. Similar provisions have been proposed for the Surry 1 and 2 Technical Specifications.

2. STA training per the requirements of Lessons Learned Category B will be complete by December 17, 1980 for both stations. This will successfully meet 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 is currently under internal review. The program will be submitted for NRC Staff review prior to January 1, 1981.

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

By January 1, 1981, the Westinghouse Owners Group will submit a detailed description of a program to comply with the requirements of I.C.1 for both Inadequate Core Cooling and Transients and Accidents. The submittal will identify 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) will also be identified. 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 is being 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 and the North Anna SES have already been put into effect. Procedures for the Surry SES are under final development and will be in effect prior to January 1, 1981.

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, 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.

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.

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.

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 cleanup, 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. Equipment qualification review for mechanical equipment required to mitigate a High Energy Line Break (HELB) for effects of increased radiation is considered to be a new requirement. This review is estimated to be completed by June 1, 1981 and may require additional equipment modification that could extend beyond the equipment qualification date of June 30, 1982. The LOCA review is expected to be more conservative; therefore, few additional modifications are expected.

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.

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.

. 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 Unit 2 and is expected to be installed for Unit 1 during the upcoming refueling outage. This instrument will provide the operator with both ECST level indication and a low level alarm in the main control room. The alarm will be 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.

Until the additional level transmitter is installed, operating personnel have been instructed on how to correlate pump suction pressure with ECST level. The operation of the AFW pump distorts suction pressure such that accurate level indication is difficult; however, the suction pressure indicator is still useful 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.

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 is expected to be tested prior to January 1, 1981 and results will then be submitted.

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 will be installed on the Emergency Condensate Storage Tank (ECST). This instrument will provide the operator with both ECST level indication and a low level alarm in the main control room. The alarm will be 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. Hence, an analysis is in progress to determine the best method for providing this capability. The analysis and subsequent modifications should be completed and/or implemented 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.

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 which is scheduled to be completed by the end of 1981. 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.

II.E.4.1 DEDICATED HYDROGEN PENETRATIONS

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. Two containment isolation valves located outside of containment will be installed on each of these lines. Since radioactive gases could be flowing through these penetrations during the post-accident mode, these systems were considered to become extensions of containment and the modifications listed below were proposed.

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. This modification 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. 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. Throttle valves will be added in the suction lines from Unit 2 containment to the hydrogen recombiners to balance the system ensuring the required flow rate can be established. The pipe run from Unit 2 containment to the recombiners is considerably shorter than from Unit 1 containment; therefore, throttling for Unit 2 may be required to provide the proper flow balance from both units.

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. Vepco has committed to relocation of remote manual valves to areas accessible within five days per the requirements of the North Anna FSAR.

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 SV & DV and auxiliary FWPT exhaust monitors

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

2. The Main Steam SV & DV and Auxiliary FWPT exhaust monitors have a total range of concentration extending from a minimum design condition of 18 uci/cc to a maximum of 10^{9} uci/cc (Kr^{85}). Three detectors are required to cover this range and have a range overlap.

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 Detector - C^{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
30 values of one-hour averages for radiation, uci/cc
30 values of one-day averages for radiation, uci/cc
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 RV & DV and Auxiliary FWPT exhaust monitors will provide continuously recorded radiation readings by analog strip chart recorders.

h. Refer to the response to Item 4.a.i for qualification details.

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 monitors are seismically installed but a seismic system design is not required. 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 5.2×10^{-7} to 3.7×10^{-1} uci/cc

Mid Range 9.6×10^{-4} to 1×10^{2} uci/cc

High Range 7.1×10^{-1} to 1×10^{5} uci/cc

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

Low Range 1.2×10^{-6} to 3.7×10^{-1} uci/cc

Mid Range 1.7×10^{-2} to 1×10^{2} uci/cc

High Range 1.8 to 1×10^{5} uci/cc

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

The energy response of the detectors are:

Low Range
Mid Range (To be supplied later)
High Range

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.h 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) or release rates (uci/sec) 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 line upstream of the safety valves/dump valves and downstream of the auxiliary feedwater pump turbine. 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.		1.8×10^1 to 1×10^9 uci/cc (Kr^{85} equiv. conc.*)	.75 MR/hr - Co^{60}
	or	(later) (Xe^{133} equiv. conc.)	
AFWPT		** to 1×10^9 uci/cc (Kr^{85} equiv. Conc.*)	
	or	(later) (Xe^{133} equiv. conc.)	

LOCA W/O SI Recirc. (TMI)

Monitor Location		Approximate Range	Background
M. S.		3.5×10^4 to 1×10^9 uci/cc (Kr^{85} equiv. conc.*)	2 R/hr
	or	(later) (Xe^{133} equiv. conc.)	
AFWPT		** to 1×10^9 uci/cc (Kr^{85} equiv. conc.*)	
	or	(later) (Xe^{133} equiv. conc.)	

LOCA With SI Recirc.

Monitor Location		Approximate Range	Background
M.S.		3.5×10^6 to 1×10^9 uci/cc (Kr^{85} equiv. conc.*)	200 R/hr
	or (later)	(Xe^{133} equiv. conc.)	
AFWPT		** to 1×10^9 uci/cc (Kr^{85} equiv. conc.*)	
	or (later)	(Xe^{133} equiv. conc.)	

NOTES: * The effluent concentrations are based on a time zero TID nuclide mixture converted to an equivalent Kr^{85} (or Xe^{133}) mixture.

**Low 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 - Kr^{85} equiv.) 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. One monitor will "view" the turbine exhaust line for each unit.

The radiation background will be less for the Auxiliary FWPT exhaust and the 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 - Kr^{85} equivalent) by using analytically derived conversion graphs. The graphs display dose rate (mr/hr) versus Kr^{85} equivalent concentration (uci/cc) based on time zero TID 14844 mixture. Also 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 measuring the AFW pump flow rate (GPM) and correlating the pump BHP requirement at that flow to the turbine steam requirements.

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

I. Process Vent and Ventilation Vent Effluent Monitors

Some design information will not be available by January 1, 1981 for NRC review. 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. 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 4Π lead shielded assemblies. A portable transfer housing containing three inches of 4Π 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 or silver zeolite 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 and located downstream fo the last effluent 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 and provides indication on the rate of radioactivity buildup.

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.

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

Vepco will respond to the requirements of this section after the studies specified in II.K.3.2 have been completed.

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

The Westinghouse Owners Group is in the process of developing a report which will address 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 will be used to support a decision on the necessity of incorporating an automatic PORV isolation system, as specified in Item II.K.3.1.

Due to the time-consuming process of data gathering, breakdown, and evaluation, it will not be possible to submit the report by January 1, 1981. The report is scheduled for submittal on or about March 1, 1981.

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 will be proposed for North Anna Unit 1 and Surry Technical Specifications on or about January 1, 1981.

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

The Westinghouse Owners Group will submit the results of a blind post-test prediction of the LOFT L3-6 test by February 15, 1981 or two (2) months following the completion of the test, whichever is later. This is consistent with the schedule provided on page II.K.3.5-1.

Should the NRC review of the submitted information indicate a need for automatic trip of the RCS pumps during a LOCA, proposed design modifications and an implementation schedule will be submitted by July 1, 1981.

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, are being 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 will be compiled into a report and submitted prior to January 1, 1981. Since the ECCS outage records for Surry cover a longer period of time and are in a form less amenable to the compilation of the required information, additional time will be required to complete the report. Submittal of the report will be made prior to March 1, 1981.

III.A.1.2 UPGRADE EMERGENCY SUPPORT FACILITIES

Additional clarification will be provided in the near future.

III.A.1.2 UPGRADE EMERGENCY SUPPORT FACILITIES

Vepco will respond once the addition clarification is provided.

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. Based on recent comments from the Staff, the majority of the first amendment is scheduled to be submitted prior to 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 are currently under revision. They will be submitted for review by the Staff on or prior to March 1, 1981.

3. Implementation of the revised radiological response plans is scheduled to be 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, including the provision of backup power supplies to both primary and backup meteorological instrument towers, and display of this data in the control room, will be operational by April 1, 1981. An operable dose calculational methodolgy (DCM) shall be 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 July 1, 1982, 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 is being compiled and will be submitted by January 1, 1981.

The control room habitiability review for Surry Units 1 and 2 is underway. Based on current progress in this program, it will not be possible to complete the review by the January 1, 1981 date. We propose to submit an analysis of habitability with respect to all on-site hazards by January 19, 1981. 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.