VIRGINIA ELECTRIC AND POWER COMPANY
RICHMOND, VIRGINIA 23261

March 14, 1996

United States Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, D 20555

Serial No. 93-136
NAPS/JHL
Docket Nos. 50-338
50-339
License Nos. NPF-4
NPF-7

Gentlemen:

VIRGINIA ELECTRIC AND POWER COMPANY
NORTH ANNA POWER STATION UNIT NOS. 1 AND 2
SUMMARY OF FACILITY CHANGES, TESTS AND EXPERIMENTS

Pursuant to 10 CFR 50.59 (b)(2), enclosed is a summary description of facility changes, tests and experiments, including a summary of the safety evaluations, that were conducted at North Anna Power Station during 1995.

If you have any questions, please contact us.

Very truly yours,

M L Bowling

M. L. Bowling, Manager
Nuclear Licensing and Operations Support

Enclosure

cc: U. S. Nuclear Regulatory Commission
Region II
101 Marietta Street, N. W.
Suite 2900
Atlanta, Georgia 30323

Mr. R. D. McWhorter
NRC Senior Resident Inspector
North Anna Power Station

180139

IE47

1995 MODIFICATION SUMMARIES
NORTH ANNA POWER STATION

NUMBER	TITLE	INSTALLED DATE	SE NUMBER
D91-010	REPAIR/REPLACEMENT OF 24" SW HEADERS TO/FROM UNIT 2 RSHXs	10/18/1995	93-SE-MOD-008
D91-011	AUX SW SUPPLY LINE BYPASSES	02/23/1995	93-SE-MOD-059
D91-160	SUPPORT FOR LIMIT STRIKE PLATE	05/10/1995	93-SE-MOD-046
D92-010	SBO DIESEL GENERATOR BLDG INSTALLATION	09/06/1995	93-SE-MOD-080
D92-012	SBO DIESEL GENERATOR TIE IN TO STATION	11/06/1995	94-SE-MOD-009 94-SE-MOD-038 95-SE-MOD-019 95-SE-MOD-031 95-SE-MOD-037
D92-016	RS TRANSMITTER REPLACEMENT	05/15/1995	94-SE-MOD-015
D92-215	REPL PUMP TO ENABLE RESIN XFER SYST TO FUNCTION	02/22/1995	92-SE-MOD-035
D92-267	BLOCKWALL SB-254-4 REPAIRS	01/10/1995	93-SE-MOD-048
D92-345	MODS TO CASING COOLING RECIRC PUMPS, CHILLERS, & TEMP INDICATORS	01/23/1995	93-SE-MOD-083
D92-353	INSTALL STRAINER UPSTREAM OF HV SYSTEM PCV	08/09/1995	95-SE-MOD-047
D92-360	FUEL TRANS TUBE FLANGE	05/28/1995	94-SE-MOD-011
D93-009	EQUIPMENT HATCH PLATFORM EXTENSION	03/31/1995	94-SE-MOD-064
D93-010	PRE-SGRP / BIO SHIELD WALL CUT, OPERATING FLOOR & CRANE WALL MODS	05/28/1995	94-SE-MOD-083
D93-011	STEAM GENERATOR REPLACEMENT	12/20/1995	94-SE-MOD-084
D93-017	MAIN GENERATOR PROTECTION MODS	06/01/1995	95-SE-MOD-010
D93-167	REPLACE AFW PRESSURE TRANSMITTERS	05/08/1995	95-SE-MOD-015
D93-201	MODIFY CONTAINMENT RING DUCT VOLUME DAMPERS	05/12/1995	94-SE-MOD-020
D93-241	REMOVAL OF EDG MUFFLER BY-PASS VALVES	04/24/1995	94-SE-MOD-005

1995 MODIFICATION SUMMARIES
NORTH ANNA POWER STATION

NUMBER	TITLE	INSTALLED DATE	SE NUMBER*
D93-246	REPLACE TEEs WITH ELBOWS ON PRZR LEVEL TRANSMITTERS	04/12/1995	93-SE-MOD-075
D93-260	REWORK CONTROL CIRCUIT TO PROVIDE LIMIT-LIMIT OPERATION	07/19/1995	94-SE-MOD-051
D93-262	INSTALL GAI-TRONICS IN CLEAN WASTE SEGREGATION FACILITY	02/21/1995	94-SE-MOD-025
D93-267	PROVIDE TORQUE SWITCH BYPASS	05/12/1995	94-SE-MOD-081
D93-272	PERMANENT SHIELDING FOR U2 SEAL WATER INJECTION FILTER CUBICAL	08/07/1995	94-SE-MOD-032
D93-273	PERMANENT SHIELDING FOR U1 SEAL WATER INJECTION FILTER CUBICAL	01/23/1995	94-SE-MOD-032
D94-002	STEAM GENERATOR BLOWDOWN SYSTEM TIE- INS	05/31/1995	94-SE-MOD-068
D94-008	HHSI FLOW INSTRUMENTATION UPGRADES/U2	05/30/1995	95-SE-MOD-003
D94-012	MOD TO PIPE SUPPORT 2-RC-HSS-144 & SOVS W/2-RC-SOV-2456-1&2	05/26/1995	94-SE-MOD-073
D94-014	LETDOWN RADIATION MONITOR REPLACEMENT	05/30/1995	95-SE-MOD-018
D94-106	INSTALL HARD PIPING FROM VENTS TO FLOOR DRAIN	01/04/1995	94-SE-MOD-052
D94-110	PROVIDE INSTRUCTIONS FOR INSTALLATION OF ANTI-ROTATION DEVICE	05/03/1995	95-SE-MOD-029
D94-127	INSTL OF HARD PIPING FROM LHSI VENTS TO FLOOR DRAIN	05/25/1995	94-SE-MOD-076
D94-147	MOD FOR EDG FAST START PTs	05/03/1995	94-SE-MOD-072
D94-156	INSTALL REDUNDANT CHECK VALVE IN SERIES WITH 2-CH-153	05/30/1995	94-SE-MOD-041
D94-167	VALVE REPLACEMENT 2-CH-155	05/09/1995	94-SE-MOD-040
D94-170	INSTALL HIGH POINT VENT VALVES IN LHSI LINES, U1	01/04/1995	94-SE-MOD-053
D94-171	INSTALL HIGH POINT VENT VALVES IN LHSI LINES, U2	05/25/1995	94-SE-MOD-075

1995 MODIFICATION SUMMARIES
NORTH ANNA POWER STATION

NUMBER	TITLE	INSTALLED DATE	SE NUMBER
D94-172	LIMIT-LIMIT OPERATION OF 2-SI-MOV-2867A&B	05/30/1995	94-SE-MOD-021
D94-180	REPLACE FLOW XMTR MODULE & PERFORM WIRING MOD	05/18/1995	94-SE-MOD-066
D94-181	PERFORM WIRING MOD TO INSTRUMENT LOOP	05/18/1995	94-SE-MOD-067
D94-188	UPGRADE CONOSEAL ASSEMBLIES U-1	05/31/1995	94-SE-MOD-060
D94-193	UF & UV RELAY RESETS	05/23/1995	94-SE-MOD-034
D94-199	INSTALL QUICK DISCONNECTS AND REPLACE SWITCH CLAMP ASSEMBLY	05/30/1995	95-SE-MOD-016
D94-209	REMOVAL OF MECHANICAL AGITATOR	11/16/1995	95-SE-MOD-005
D94-211	ELECTRICAL AND MISSILE MANHOLE MODS	09/27/1995	95-SE-MOD-002
D94-212	MODIFICATION OF EDG ROOM DOORS	08/29/1995	95-SE-MOD-011
D94-216	INSTALLATION OF CONTAINMENT INSTRUMENT AIR HEADER MANUAL ISOLATION VALVES	05/13/1995	94-SE-MOD-078
D94-239	INSTALL LED LAMPS IN E30JF SWITCHES	05/27/1995	94-SE-MOD-059
D94-247	ABANDON HYDROGEN ANALYZERS	06/15/1995	94-SE-MOD-077
D94-249	REPLACE CASING COOLING FLOW SWITCHES	06/23/1995	94-SE-MOD-085
D94-255	RE-FURBISHMENT OF 2ND FLOOR, STATION ADM BUILDING	01/11/1995	94-SE-MOD-065
D94-260	GENERATOR B.U. IMPEDANCE RELAY RESET	04/05/1995	94-SE-MOD-069
D94-274	SEAL SFGD VALVE PIT ACCESS SHAFT TO PREV. GROUND WATER INTRUSION	03/23/1995	94-SE-MOD-070
D94-293	INSTRUMENT AIR LINE REMOVAL FROM 2ND FLOOR OF ADMIN BUILDING	01/12/1995	94-SE-MOD-082
D94-296	PERMANENT WALL ATTACHMENT FOR EDG CURTAINS	01/24/1995	94-SE-MOD-080
D95-004	ROD CONTROL SYSTEM TIMING CHANGES U2	05/30/1995	94-SE-MOD-017
D95-009	RESOLVE DC SEPARATION ISSUE FOR SOV PANELS U2	05/30/1995	95-SE-MOD-027
D95-109	VALVE REPLACEMENT (2-CH-MOV-2370)	05/28/1995	95-SE-MOD-021

1995 MODIFICATION SUMMARIES
NORTH ANNA POWER STATION

NUMBER	TITLE	INSTALLED DATE	SE NUMBER
D95-112	INSTALL SEPARATE DRAIN LINES FOR DEMISTER TANK & LO TANK EXHAUST FAN	04/05/1995	95-SE-MOD-009
D95-115	REACH ROD PENETRATION RELOCATION	11/06/1995	95-SE-MOD-055
D95-116	ROOF UPGRADE, NEW FUEL RECEIVING, U1 CASING COOLING - U1	11/17/1995	95-SE-MOD-050
D95-117	REPLACE SECURITY INVERTER & BATTERY CHARGER	07/11/1995	95-SE-MOD-023
D95-120	REPLACE 480V LOAD CENTER BREAKER U2	05/30/1995	95-SE-MOD-043
D95-128	CHARGING PUMP CASING REPLACEMENT U2	03/21/1995	95-SE-MOD-014
D95-129	VALVE REPLACEMENT 2-SI-MOV-2867B	05/30/1995	95-SE-MOD-024
D95-133	REMOVE VOLUME BOOSTERS	05/23/1995	95-SE-MOD-026
D95-135	RECONFIG. CHEMICAL ADDITION CONNECTION TO BEARING COOLING HEADER	06/01/1995	95-SE-MOD-022
D95-146	REPLACE RC FLOW TRANSMITTER U2	05/26/1995	95-SE-MOD-025
D95-158	VALVE REPLACEMENT 2-SI-100	05/31/1995	95-SE-MOD-028
D95-159	INSTALL STRAINER (N2 TO PRT) UPSTEAM OF 2-SI-PCV-200	05/27/1995	95-SE-MOD-030
D95-163	RELOCATION OF APPENDIX R FLEXIBLE DUCTING IN THE AUX BUILDING	06/01/1995	95-SE-MOD-041
D95-164	INCORE FLUX THIMBLE MODIFICATION U2	05/30/1995	95-SE-MOD-034
D95-165	INSTALL THRUST BEARING	05/15/1995	95-SE-MOD-033
D95-167	SSPS INTERNAL WIRING MODIFICATION U-2	05/30/1995	95-SE-MOD-036
D95-168	ELIMINATION OF SECONDARY SIDE BORIC ACID TREATMENT	05/26/1995	95-SE-MOD-042
D95-169	CHANGE MOTOR SET GEARING FOR 2-RS-MOV-201A/B	05/23/1995	95-SE-MOD-037
D95-170	CHANGE MOTOR SET GEARING FOR 2-RH-MOV-2720A/B	05/30/1995	95-SE-MOD-035
D95-175	ACTUATOR MODIFICATION TO 2-SI-MOV-2890A	05/24/1995	95-SE-MOD-039
D95-178	ACTUATOR GEAR CHANGE FOR 2-CH-MOV-2373	05/20/1995	95-SE-MOD-040

1995 MODIFICATION SUMMARIES
NORTH ANNA POWER STATION

NUMBER	TITLE	INSTALLED DATE	SE NUMBER
D95-179	ACTUATOR GEAR REPLACEMENT U2	05/20/1995	95-SE-MOD-038
D95-193	INSTALL LADDER TO ACCESS U2 SAFEGUARDS ROOF	10/30/1995	95-SE-MOD-063
	SPECIAL TESTS		
2-ST-097	STEAM GENERATOR MOISTURE CARRYOVER MEASUREMENT USING CHEMTRAC CHEMICAL TRACER METHOD	10/17/1995	95-SE-ST-004

1995 SAFETY EVALUATIONS - NAPS

Number	Unit	Document	Description	SNSOC Date
94-SE-OT-027 REV. 1	1,2	UFSAR FN 94-015	Storage of irradiated components (but not fuel) between the storage racks & spent fuel pool wall.	10-31-95
95-SE-JCO-01	1,2	JCO 95-01	Interim use of a station battery swing charger during unit power operation w/o entering LCO 3.8.2.3.	7-11-95
95-SE-JCO-02	2	JCO 95-02	Continued operation of U-2 without inspection / repair of 2-CH-MOV-2286B & 2287B (valves experienced excessive over thrust during closing)	9-08-95
95-SE-JCO-03	1,2	JCO 95-03	Evaluation of SW pump net positive suction head (NPSH).	11-22-95
95-SE-OT-01	1,2	UFSAR FN 94-039	Revises Table 3.8-12, Lake Anna main dam instrumentation monitoring frequencies.	1-09-95
95-SE-OT-02	1,2	TRM, Rev. 7	Corrects typographical errors & labeling errors.	1-24-95
95-SE-OT-03	1,2	TRM revision	Corrects errors in Section 12.1, Snubbers.	1-31-95
95-SE-OT-04	1,2	TS CHG #317	Allows 1 of 2 SW loops to be isolated from CCHXs during power operation to refurbish the isolated SW headers (to be worked under DCP 94-010).	2-16-95
95-SE-OT-04 REV. 1	1,2	TS CHG #317A	Adds exemption of T.S. 3.0.4.	3-22-95
95-SE-OT-04 REV. 2	1,2	DCP 94-010	Repair / replacement of deteriorated exposed SW piping around CCHXs.	11-09-95
95-SE-OT-05	1,2	UFSAR, TRM, SER APP. R Report	Replaces 8 Class 2A water extinguishers with 8 Class 4A60BC dry chemical extinguishers in U1&2 cable vault & tunnel, emergency. Switchgear rooms, & Instrument rack room.	2-21-95

1995 SAFETY EVALUATIONS - NAPS

Number	Unit	Document	Description	SNSOC Date
95-SE-OT-06	1,2	TS CHG #325	Type "A" test exemption.	2-23-95
95-SE-OT-07	2	95-TSR-007, 008, 009, 010 & 032	Temporary shielding requests to place lead blanket shielding over selected piping during U-2 SGR/RO.	3-01-95
95-SE-OT-08	1,2	UFSAR Sect. 8.1	Adds Gordonsville 230 kV line; better describes connections between switchyard & plant; corrects number of normal 480V buses per unit; makes administrative / editorial changes.	3-16-95
95-SE-OT-09	1,2	TRM, Rev. 9	Changes General Requirement 1.0.10 to have original written Special Report submitted to NRC Document Control Desk.	3-21-95
95-SE-OT-10	1,2	UFSAR FN 95-005	Revises description of Inside Recirc spray pump flow test in UFSAR Section 6.2.2.4.2.	3-22-95
95-SE-OT-11	1,2	ISFSI license application	North Anna independent spent fuel storage installation (ISFSI) safety analysis report/TS.	3-29-95
95-SE-OT-12	2	95-TSR-035, Rev. 1, 95-TSR-049, Rev. 1	Temporary shielding requests to place lead blanket shielding around specified RCS piping while U-2 is in either Mode 5 or Mode 6.	4-02-95
95-SE-OT-13	1,2	UFSAR Sec. 12.2	Upgrade to enhance its accuracy to reflect plant operation.	4-04-95
95-SE-OT-13 REV. 1	1,2	UFSAR FN-95-10	Upgrade Section 12.2 to enhance its accuracy to reflect plant operations.	6-19-95
95-SE-OT-14	1,2	UFSAR 15.2.13, UFSAR 15.4.2.1 NE Tech Report 1016	Reanalysis of North Anna main steam line break.	4-27-95
95-SE-OT-15	2	NE Tech Report 1019	Reload Safety Evaluation for N2 C11, Pattern UM .	4-27-95

1995 SAFETY EVALUATIONS - NAPS

Number	Unit	Document	Description	SNSOC Date
95-SE-OT-16	1	Proposed maintenance. on 1-FW-MOV-154A, B, C	De-energize fast acting main FW MOVs one at a time with the unit at power for modification to MOV torque switches.	4-27-95
95-SE-OT-17	1,2	TRM, Rev. 10	Enhances TRM Section 7 in response to QA audits, operator input; reflects equipment configuration. GR 1.0.3 & GR 1.0.4 notes will prevent unnecessary plant shutdowns.	5-09-95
95-SE-OT-18	1,2	UFSAR FN 95-013	Allows flexibility for use of quality assured computer codes & formalizes use of computer codes for piping analysis by any qualified personnel.	5-17-95
95-SE-OT-19	1,2	T.S. CHG #310 T.S. CHG #310A	Increases surveillance test interval for the turbine reheat stop & intercept valves.	6-27-95
95-SE-OT-20	1,2	T.S.. CHG #328	Operation at reduced power levels with inoperable MSSVs.	6-27-95
95-SE-OT-20 REV. 1	1,2	T.S.. CHG #328	Operation at reduced power levels with inoperable MSSVs.	8-17-95
95-SE-OT-21	1,2	TS CHG #330 FN 95-017	PSRV setpoint changes.	6-29-95
95-SE-OT-22	1,2	TS CHG #320	Charging pump operability during low temperature operation.	6-29-95
95-SE-OT-23	1	W.O. 321552-01	Air jumper installed around the SOV for 1-IA-TV-102B to maintain the valve open.	7-11-95
95-SE-OT-24	1,2	- QA Topical Report, Chgs 2,3,4,6,7,8,9 - UFSAR Ch. 17	Revises QA Topical Report VEP 1-5A to reflect current practices & current organization.	7-13-95
95-SE-OT-25	1,2	App. R Report, Rev. 13	Annual update of NAPS Appendix R report.	7-27-95

1995 SAFETY EVALUATIONS - NAPS

Number	Unit	Document	Description	SNSOC Date
95-SE-OT-26	1,2	UFSAR FN 95-027	Deletes most of the 4th paragraph on page 10.4-15 concerning operation of motor-driven AFW pumps.	7-27-95
95-SE-OT-27	1,2	UFSAR, Sec. 11.3	Deletes description of use of catalytic H2 recombiner.	7-31-95
95-SE-OT-28	1,2	UFSAR, Sec. 9.2.1.3.2	Deletes discussion of SW check valves in RSHX supply header preventing an operable header from back-feeding an inoperable header - replaces with SW-MOVs in cross-ties being powered from different emergency busses.	8-15-95
95-SE-OT-29	1,2	T.S. CHG #318	Extend EDG allowed outage time to 14 days.	8-18-95
95-SE-OT-30	1,2	UFSAR, Table 6.3-2	Revise MOV stroke time requirements.	9-19-95
95-SE-OT-31	1,2	(see DR N-95-1382)	Provides the basis for documentation revisions to support CC tank low level setpoint & safety analysis volume requirement changes.	9-19-95
95-SE-OT-32	1,2	TS CHG #323	PORV operability per GL 90-06. Re-approved on 10-18-95 for hand-correction of errors - only Engineering Transmittal 95140 was signed with new date.	9-27-95
95-SE-OT-33	2	TS CHG #329	Unit 2 post-SGRP inspections.	9-27-95
95-SE-OT-34	1,2	TS CHG #316	Allows both containment personnel airlock doors to remain open during core alterations or fuel movement.	9-28-95
95-SE-OT-35	1,2	UFSAR, Sec. 15.3.1 & 15.4.1	Updates large break LOCA & small break LOCA analyses to support up-flow conversion project (DCP 95-001).	10-31-95
95-SE-OT-36	1,2	TS CHG #333	Allows the use of 10 CFR 50, App. J, Option B, for Type A, B, C testing.	11-01-95

1995 SAFETY EVALUATIONS - NAPS

Number	Unit	Document	Description	SNSOC Date
95-SE-OT-37	1,2	QA Topical Report VEP-1-5A	Reflects reorganization of nuclear organization, deletion of QA Department, & creation of Nuclear Oversight.	11-02-95
95-SE-OT-38	1,2	DR N-95-1586	Operation without continuous radiation monitoring of service water effluent from the four component cooling heat exchangers as described in UFSAR 11.4.2.8.	11-13-95
95-SE-OT-39	1,2	TRM Chg. Request !%	EQ Doors - clarifies necessary compensatory actions to be taken when an EQ door will not function as an environmental barrier automatically or with assistance of an EQ watch.	11-20-95
95-SE-OT-40	1,2	B.C. memo from W. A. Thornton to A. H. Stafford	Bearing Cooling Water Tower Sludge Classification of Material memo from W. A. Thornton to A. H. Stafford, dated 11-09-95. - Allows operation of the bearing cooling system with radioactive contaminants.	11-22-95
95-SE-OT-41	1,2	DR N-95-1820 UFSAR CHG FN 92-130	Supports the change to continuity test the MSTV SOVs on a refueling cycle interval versus the original UFSAR stated monthly testing.	11-22-95
95-SE-OT-42	1,2	QA Topical Report VEP 1-5A	Clarifies that technical requirements during operations must be equivalent to original requirements (also reduces QC hold points).	12-11-95
95-SE-OT-43	1,2	UFSAR, Sec. 6.2 TS Sec. 6.2	Containment analysis - removal of containment floor plugs; reduces casing cooling flow rate; reduces casing cooling volume; extends RMT time for LHSI switch over; & provides a sensitivity of the ORS pump NPSH to casing cooling temperature.	12-12-95
95-SE-OT-44	1.2	UFSAR, Sec. 9.5.8.2	Removes the requirement for operators to write work requests for cleaning the EDG rooms.	12-12-95
95-SE-OT-45	1	DR N-95-1933 UFSAR Sec. 3.8.1.1	Evaluation of concrete in rattle space between auxiliary building basement floor slab & Unit 1 QSPH between elevation 242'-3: & 246'-6".	12-20-95
95-SE-OT-46	1	96-TSR 005, 010, 012, 013	Temporary shielding requests for pipe stress-related TSRs.	12-20-95
95-SE-OT-47	1,2	UFSAR Section 7.6.7	Reflects replacement of a B&W acoustic valve monitoring system with a qualified TEC acoustic valve monitoring system per DCPs 84-17 & 84-18.	12-28-95

1995 SAFETY EVALUATIONS - NAPS

Number	Unit	Document	Description	SNSOC Date
95-SE-PROC-01	1	1-ECM-2303-01 (OTO-1 chg)	Adds an additional temporary modification to Attachment 8 to allow running both 1-FW-P-1A1 & 1A2 simultaneously.	1-22-95
95-SE-PROC-01 REV. 1	1,2	1-ECM-2303-01, 2-ECM-2303-01	Revision to procedures to allow running A, B, & C main FW inboard/outboard motors coupled to each other but not coupled to their respective pump.	2-07-95
95-SE-PROC-02	1,2	1/2-AP-33.1 (Rev. 1)	Incorporates guidance for RCP #1 seal failure per IEIN 93-84: requires the RCP be tripped within 5 minutes & #1 seal leakoff isolation valve be closed immediately. after RCP stops rotating.	1-31-95
95-SE-PROC-03	1,2	1/2-OP-7.13	Allows depressurized venting of the headers.	2-06-95
95-SE-PROC-04	1	1-MOP-55.82 (OTO-1)	Verification that Ch. II of auto stop oil was in "trip" per W.O. 00310389-01.	2-10-95
95-SE-PROC-05	2	0-MCM-1303-01, 0-MCM-1106-01, 0-MCM-1106-04	Allows use of auxiliary crane for miscellaneous heavy load handling per 0-MCM-1303-01 & for handling Rx. vessel head studs per 0-MCM-1106-01 & 04.	2-23-95
95-SE-PROC-05 REV. 1	2	0-MCM-1303-01, 0-MCM-1106-01, 0-MCM-1106-04	PARs to procedures remove the word "horizontal" for the auxiliary crane position.	3-30-95
95-SE-PROC-06	2	2-OP-2.2 (P-4 to Rev. 29)	Closes #1 governor valve by isolating EHC to 2-MS-GOV-1A.	2-27-95
95-SE-PROC-07	2	2-PT-83.5J	Installs a temporary modification that will energize the BAST heater, 1-CH-EHR-8B, to support UV/DV testing.	3-16-95
95-SE-PROC-08	1	W.O. 00303634-01 W.O. 00303635-01	MDAP-19 for isolating one line at a time of the LP turbine crossover steam supply piping by closing the intercept valve.	3-23-95
95-SE-PROC-09	2	IMP-C-2-RVLIS-01 (P-1 to Rev. 1)	Install temporary spool piece in RVLIS system to allow RVLIS to operate longer & monitor RCS level.	3-28-95

1995 SAFETY EVALUATIONS - NAPS

Number	Unit	Document	Description	SNSOC Date
95-SE-PROC-10	1,2	70 Inst. proc.	ICPs & channel functional PTs for (a) S/G level protection, (b) steam flow / feed flow protection, (c) turbine 1st stage pressure. Changes procedures to always place AMSAC in manual bypass whenever procedure is performed for consistency.	3-29-95
95-SE-PROC-11	1	W.O. 00315173-01	Troubleshoot the 43 selector switch for 1A main transformer cooling circuit. Also provides contingency step to energize all cooling circuits, if needed, which requires a temporary modification.	3-30-95
95-SE-PROC-12	2	IMP-C-FP-01 (P-3 chg)	Adds steps to remove & return to service smoke & heat detectors in Unit 2 containment.	4-03-95
95-SE-PROC-13	1,2	D-NAT-91-010-2-1 (Rev. 2)	Incorporates a TM to provide a high pressure source for performing SW system hydrostatic test.	4-06-95
95-SE-PROC-14	2	2-OP-8.2 (P-1, Rev. 14) TM 95-N2-1089	Places 2-CH-I-3B in service for RCS purification during startup with a previously utilized lithium form mixed bed resin.	5-09-95
95-SE-PROC-14 REV. 1	2	2-OP-8.2 Rev. 14, P-1	Corrects errors in original 50.59 - reflects that 2-CH-I-3B will be used as a mixed bed IX.	6-26-95
95-SE-PROC-15	2	D-NAT-92-12-3-10 (P-1 CHG)	Testing of SBO diesel - P-1 defeats the trip of 05L1 breaker when 2B station service bus is being fed off of 'B' RSST.	5-13-95
95-SE-PROC-16	2	2-OP-5.1	Allows installation of a temporary modification to bypass the protective circuitry to open the cold leg loop stop valve.	5-16-95
95-SE-PROC-17	2	D-NAT-94-008-2-1	Performs system leakage testing on hot leg & cold leg safety injection piping inside containment.	5-24-95
95-SE-PROC-18	1,2	W.O. 318863-01, 0-MCM-1904-01	On-line leak repair of 1-SW-679.	5-25-95
95-SE-PROC-19	1,2	0-MCM-1410-03	Connecting, cleaning, & flushing U-1 or U-2 EHC fluid systems using the portable filtration skid.	7-13-95

1995 SAFETY EVALUATIONS - NAPS

Number	Unit	Document	Description	SNSOC Date
95-SE-PROC-19 REV. 1	1,2	0-MCM-1410-03	Revises Block 31 to state that temporary oil absorbent barriers will be installed around the filtration skid to prevent the possible spread of EHC fluid.	7-18-95
95-SE-PROC-20	1,2	0-FH-FEB-001	Temporary new fuel elevator basket replacement - Procedure. will be used to control a temporary modification.	7-13-95
95-SE-PROC-21	1,2	FP-VRA-FI1	Fuel inspection & repair for Unit 1.	7-13-95
95-SE-PROC-22	2	2-MOP-26.113 (P-1 to Rev. 0)	Provides more specific instructions on preparing the plant for removal of MCC-2C1-3 from service.	8-08-95
95-SE-PROC-23	1,2	FP-VRA-FI1 (P-3 to Rev. 0)	Adds steps to allow rod E-7 to be moved from AM2 F/A & placed into the rod canister for storage.	8-23-95
95-SE-PROC-24	2	2-OP-7.10 (P-1 to Rev. 17)	Raise Recirc spray sump level above the 6-inch requirement - source to sump is from RWST via suction side of B LHSI pump	8-19-95
95-SE-PROC-25	2	2-MOP-7.01 (R14-P1), 2-MOP-7.02	Provides for vacuum-assisted depressurized venting of "A" LHSI discharge header.	8-31-95
95-SE-PROC-26	1,2	GMP-M-162, Rev. 1	Rev. 1 provides for compensatory missile protection measures & temporary seismic restraints for excavation of the fire main.	9-26-95
95-SE-PROC-27	1,2	1-OP-7.10 (R19-P1), 2-OP-7.10 (R17-P2)	Provides steps to add PG water from temporary hose connections in safeguards area to Recirc spray sump.	10-18-95
95-SE-PROC-28	2	MDAP-19 W.O. 329415-02	Replacement of 27XB degraded voltage relay (2-EP-CB-28J).	11-03-95
95-SE-PROC-29	1,2	0-TOP-50.6	New procedure to determine if a cyclonic separator unit will separate bearing cooling tower sludge.	11-14-95

1995 SAFETY EVALUATIONS - NAPS

Number	Unit	Document	Description	SNSOC Date
95-SE-PROC-30	1,2	0-MOP-50.1 (Rev. 0)	De-sludging the bearing cooling tower basin to a sludge separator.	12-14-95
95-SE-PROC-31	1,2	1-PT-210.19 2-PT-210.19	Allows accumulator check valve full flow test with fuel in the vessel.	12-19-95
95-SE-PROC-32	2	MDAP-19 W.O. 331754-01	2-CC-TV-204C will be closed for air tubing replacement. Contingency plans allow for installation of an air jumper to reopen the valve if the RCP motor temperature approaches the required RCP trip Setpoints.	12-21-95
95-SE-ST-01	1	1-ST-104 (Rev. 0)	Safeguards exhaust flow verification of 1-HV-F-40B with inspection port of 1-HV-F-40A open.	7-20-95
95-SE-ST-02	1	1-ST-104 (Rev. 1)	Removes inspection port on discharge at 1-HV-F-40A; provides for clearing action statement during restoration.	7-24-95
95-SE-ST-03	2	2-ST-98	Discharge valve flow & differential pressure verification for 2-CH-MOV-2286B and 2287B.	8-15-95
95-SE-ST-04	2	2-ST-97	S/G moisture carryover measure using CHEMTRAC chemical tracer method.	10-17-95
95-SE-TM-01	1	N1-95-1611	Install temporary air eductor on top of manway of lube oil reservoir to replace 1-GM-F-6.	1-04-95
95-SE-TM-02	2	N2-95-1082	Connects a regulated service air system. jumper to condenser air ejector exhaust flow meter manifold test connections at 2-VP-331 & 2-VP-337.	2-07-95
95-SE-TM-03	1	N1-95-1612	Jumpers out the 4 min. timer in CW pump start/stop circuitry to allow starting/stopping pump as necessary.	2-07-95
95-SE-TM-04	2	N1-95-1083	Installs a controlled supply of N2 to condenser hot well to improve removal of non-condensable gases modified & (supplied from 2-GN-170 or N2 bottle). **Modified & re-signed 2/16/95**	2-15-95

1995 SAFETY EVALUATIONS - NAPS

Number	Unit	Document	Description	SNSOC Date
95-SE-TM-05	1,2	N1-95-1613	Temporarily install a Merlin-Gerin spectral analysis rad monitor on vent stack 'B' for 2-3 months for evaluation.	2-23-95
95-SE-TM-06	1	N1-95-1614	install electrical jumper from bypass transformer to distribution panel 1IHS-MO in security system.	3-16-95
95-SE-TM-07	2	N2-95-1084	Installs a controlled supply of N2 to condenser hot well to improve removal of non-condensable gases (supplied from 2-CN-170 or N2 bottle). **(Supersedes N1-95-1083)**	3-17-95
95-SE-TM-08	2	N2-95-1085	Install temporary hose between an SI accumulator vent & a drain off the RHR relief valve discharge line.	3-27-95
95-SE-TM-09	2	N2-95-1086	Disables 2 of 4 parking brakes on polar crane bridge due to failure of control card on electrical portion of brakes.	4-01-95
95-SE-TM-10	1,2	N2-95-1615	Provides an alternate means of adding H-900 biocide at the bearing cooling tower.	4-13-95
95-SE-TM-11	2	N2-95-1087	Block exhaust & supply ductwork in 2-I battery room to maintain control room pressure boundary during 2-PT-87H.	4-20-95
95-SE-TM-12	2	N2-95-1090	Uses the field cable for 2-RM-RMS-263 to return 2-RM-RMS-262 to service.	5-25-95
95-SE-TM-13	2	N2-95-1092	Low pressure N2 supplied from 2-GN-170 or N2 bottle to condenser hot well at 2-CN-167 or 2-CN-168 & 2-CN-171.	5-31-95
95-SE-TM-14	1	N1-95-1617	Low pressure N2 supplied from 1-GN-413 or N2 bottle to condenser hot well at 1-CN-165 & 1-CN-169.	6-15-95
95-SE-TM-15	1	N1-95-1859	Install portable AC unit outside U-1 rod drive room to provide additional cooling until 1-HV-AC-163 is repaired.	7-18-95

1995 SAFETY EVALUATIONS - NAPS

Number	Unit	Document	Description	SNSOC Date
95-SE-TM-16	1	N1-95-1620	Crimps the EHC line upstream of 1-EH-12 to stop EHC flow to the EHC filters.	10-25-95
95-SE-TM-16 REV. 1	1	N1-95-1620	Jumper changed to reflect MDAP-19 (WO. 328635-02) - after crimp, the crimping bracket will be left in place of the upstream crimp.	10-26-95
95-SE-TM-17	1,2	N2-95-1093	Allows operation of the bearing cooling fans with fire protection out of service to BC tower cells during repair of the fire protection pressure regulator for one of the BC cells.	12-14-95

DC 91-010
REPAIR/REPLACEMENT OF 24" SW HEADERS TO/FROM UNIT 2 RSHXs
NORTH ANNA / UNIT 2

DESCRIPTION

DCP 91-010, "Repair/Replacement of 24" SW Headers to/from Unit 2 RSHXs," is the third in a series of design change packages developed to clean, inspect, repair and coat the 24" diameter buried and concrete-encased service water (SW) piping which is currently uncoated and experiencing corrosion.

Service water piping has experienced both general area and localized pitting corrosion due to the aggressive nature of the water and the presence of several forms of bacteria which accelerate attack of base metal and welds.

While significant progress has been made recently in corrosion protection of system materials and in control of the bacteria population, certain portions of the system have reached a point where weld repair is necessary to maintain long term structural integrity and pressure boundary capability. This repair, combined with application of a corrosion inhibiting epoxy-based coating, is the scope of this multi-year project.

The specific scope of DCP 91-010 was the refurbishment of approximately 300 feet of 24" piping which forms the supply and return headers to/from the Unit 2 recirculation spray heat exchangers (RSHXs). The work was performed during the Unit 2 outage during the time period when these lines are not required to be operable.

Generally, the work scope involved the following activities for piping which was being repaired:

1) isolation and draining the affected piping,
2) initial blast cleaning to remove corrosion product to facilitate inspection,
3) inspection by engineering personnel to identify areas requiring weld repair,
4) weld repairs,
5) blasting and application of a two-coat epoxy based coating system, and,
6) return to service.

Steps 2 through 5 above were accomplished by personnel entry into the piping. Piping which was replaced was cut out and removed and replaced with shop-coated piping. Field welds were coated from inside the piping after installation.

SUMMARY OF SAFETY ANALYSIS (93-SE-MOD-008)

An unreviewed safety question did not exist because:

A. The implementation of this DCP did not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety and previously evaluated in the UFSAR since the DCP was implemented while Unit 2 was shutdown. The SW lines to/from the RSHXs are not required to be operable during operating modes 5 and 6. These lines were temporarily plugged during the 168-hour Technical Specification (TS) Action Statement (service water TS Section 3.7.4.1.d) with code-qualified welded pipe plugs at the point of origin with the Unit 1 and common sections of the SW system. This allowed the Unit 2 portions to be isolated from the operating portion of the SW system for extended durations to perform the pipe refurbishment activities. Following completion of the pipe work, the temporary plugs were removed.

A portion of the work involved an excavation to expose and subsequently replace a portion of the buried SW lines. The backfill over these lines provides the tornado missile protection to meet GDC-2 for protection from natural phenomena. The minimum required cover over these lines was not removed until the lines were isolated and out of service. Following pipe replacement, the minimum cover was restored prior to returning the lines to operable status.

The engineering inspection identified some areas which were below the minimum wall thickness criteria established for various portions of the piping system. For these areas, weld repairs were imposed. Due to the random nature of the corrosion found, it was concluded that the structural integrity of the SW system and its ability to perform all of its required safety functions, were preserved, both for the as-found conditions (prior to weld repair) and as-left conditions (following repair.)

B. The implementation of this DCP did not create a possibility for an accident or a malfunction of a different type than any evaluated previously in the UFSAR since this DCP did not alter the function or capability of the SW system. The pipe cleaning, inspection, repair, and coating enhanced the materiel condition of the system by identifying and repairing isolated minimum wall thickness locations and by providing a barrier to future

corrosion. The work was performed during the Unit 2 outage as stated above, therefore, there was no impact on the ability of the system to perform its intended functions during the implementation of the DCP.

C. The implementation of this DCP did not reduce the margin of safety as defined in the basis of any Technical Specification since the function of the SW system was not changed nor were any changes made in the operability requirements of the system. During the course of the refurbishment process, the existing 168-hour TS Action Statement was entered four times to permit installation of the temporary pipe plugs and their subsequent removal. During these Action Statements, one of the two redundant SW loops was taken out of service. Components which are supplied with service water for cooling were temporarily aligned to only one SW header while the other header was isolated. Use of the Action Statement in this manner was evaluated for any impact on SW system reliability using probabilistic risk assessment techniques. It was found that the time period (potentially up to 28 days) that the system would be operating with one header isolated resulted in an insignificant impact on the base core damage frequency/probability plant model.

DC 91-011
REPAIR/REPLACEMENT OF 24" AUXILIARY SW SUPPLY HEADERS
NORTH ANNA / UNITS 1 & 2

DESCRIPTION

DCP 91-011, "Repair/Replacement of 24" Auxiliary SW Supply Headers," was the second in a series of design change packages developed to clean, inspect, repair and coat the 24" diameter buried and concrete-encased service water (SW) piping which is currently uncoated and experiencing corrosion.

Service water piping has experienced both general area and localized pitting corrosion due to the aggressive nature of the water and the presence of several forms of bacteria which accelerate attack of base metal and welds.

While significant progress has been made recently in corrosion protection of system materials and in control of the bacteria population, certain portions of the system have reached a point where weld repair is necessary to maintain long term structural integrity and pressure boundary capability. This repair, combined with application of a corrosion inhibiting epoxy-based coating, is the scope of this multi-year project.

The specific scope of DCP 91-011 was the refurbishment of approximately 750 feet of 24" piping which forms the two auxiliary service water supply headers. These lines provide flow from Lake Anna to the main SW system using the auxiliary SW pumps. The work was performed during non-outage conditions.

Generally, the work scope involved the following activities for piping which was being repaired:

1) isolation and draining the affected piping,
2) initial blast cleaning to remove corrosion product to facilitate inspection,
3) inspection by engineering personnel to identify areas requiring weld repair,
4) weld repairs,
5) blasting and application of a two-coat epoxy based coating system, and,
6) return to service.

Steps 2 through 5 above were accomplished by personnel entry into the piping. Piping which was replaced was cut out and removed and replaced with shop-coated piping. Field welds were coated from inside the piping after installation.

SUMMARY OF SAFETY ANALYSIS (93-SE-MOD-059)

An unreviewed safety question did not exist because:

A. The implementation of this DCP did not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety and previously evaluated in the UFSAR since the DCP did not affect operation of the main portion of the SW system. The auxiliary SW system is isolated from the main SW system by redundant normally-closed motor operated valves in series. Use of the auxiliary SW system during normal unit(s) operation (other than for service water reservoir make-up) is prohibited except when both units are shutdown. However, the auxiliary SW system does provide an Appendix "R" safe shutdown function. The auxiliary SW system is required to place the affected unit(s) in cold shutdown in the event of a fire in the Service Water Pump House (SWPH) which disables the main SW pumps. During the implementation of this DCP, the requirements in the Technical Requirements Manual were complied with to ensure that adequate equipment remained available in the event of a SWPH fire or that appropriate contingency measures were implemented.

A portion of the work involved an excavation to expose and subsequently replace a portion of the buried auxiliary SW lines. The backfill over these lines provides the tornado missile protection to meet GDC-2 for protection from natural phenomena. It would have been impractical to maintain adequate cover over one of the two lines while the other line was exposed and replaced because the lines are buried very close together. As a result, a request for temporary relief from GDC-2 was submitted to the NRC and was approved to allow exposure of both auxiliary SW lines during replacement of one line. This request was based on probabilistic risk assessment techniques and implementation of numerous contingency measures. It was found that the time period (potentially up to nine months) that the system would be operating with the auxiliary service water headers exposed resulted in an insignificant impact on the base core damage frequency/probability plant model.

The engineering inspection identified some areas which were below the minimum wall thickness criteria established for various portions of the piping system. For these areas, weld repairs were imposed. Due to the random nature of the corrosion found, it was concluded that the structural integrity of the SW system and its

ability to perform all of its required safety functions, were preserved, both for the as-found conditions (prior to weld repair) and as-left conditions (following repair.)

B. The implementation of this DCP did not create a possibility for an accident or a malfunction of a different type than any evaluated previously in the UFSAR since this DCP did not alter the function or capability of the SW system. As stated above, this DCP did not affect normal operation of the main portion of the SW system. The pipe cleaning, inspection, repair, and coating enhanced the materiel condition of the auxiliary SW system by identifying and repairing isolated minimum wall thickness locations and by providing a barrier to future corrosion.

C. The implementation of this DCP did not reduce the margin of safety as defined in the basis of any Technical Specification since the function of the SW system was not changed nor were any changes made in the operability requirements of the system. The temporary exposure of the buried portions of the two auxiliary SW lines was evaluated for any impact on SW system reliability using probabilistic risk assessment techniques and was found to have an insignificant impact on the base core damage frequency/probability plant model.

DCP 91-160

SUPPORT FOR LIMIT STRIKE PLATE
02-SI-MOV-2836, 2869A, 2869B, 2890A AND 2890B
NORTH ANNA UNIT 2

DESCRIPTION

Limit strike plates associated with 2-SI-MOV-2836, 2-SI-MOV-2869B, and 2-SI-MOV-2890B attached to the all thread rod tend to move sideways on coming in contact with the switch arm of the stem mounted limit switch when the valve is being closed. This causes the limit not to make up resulting in erroneous indication at the Control Room. It is required that the valve monitor light No. 1 on the Main Control Vertical section 2-3B be lit when all three valves MOV-2836, MOV-2869B, and MOV-2890B are fully closed, which is the NPO (Normal Plant Operating) position and the valve monitor light Nc. 2 on the Main Control Vertical Board section 2-3A be lit when the valves MOV-2869A and MOV-2890A are fully closed. The movement of the limit strike plate is to be prevented to provide reliability of valve position indication at the control room.

SUMMARY OF SAFETY ANALYSIS

This design change in accordance with DCP 93-132 does not create an "unreviewed safety question" as defined in 10 CFR 50.59.

A. The implementation of this modification does not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety and previously evaluated in the Final Safety Analysis Report.

The limit strike plates and the stem mounted limit switches associated with the SI valves are installed to provide indication of the valve position in the Main Control Vertical Board sections 2-3A & 2-3B. The valve monitor lights no. 2 & 1 respectively illuminate whenever the respective valves are fully closed. This modification of the strike plate ensures the reliability of the indication. The limit strike plates and the stem mounted limit switches are passive components that have no control on the operation of the valves and therefore,are unrelated to the initiation of any of the accidents considered. Thus, the modification of the limit strike plate does not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety and previously evaluated in the Final Safety Analysis Report.

B. The implementation of this modification does not create a possibility for an accident or malfunction of a different type than any previously evaluated in the Final Safety Analysis report.

The limit strike plates and the stem mounted limit switches associated with the SI valves are installed to provide indication of the valve position in the Main Control Vertical Board sections 2-3A & 2-3B. The valve monitor lights no. 2 & 1 respectively illuminate whenever the respective valves are fully closed. This modification of the strike plate ensures the reliability of the indication. The limit strike plates and the stem mounted limit switches are passive components that have no control on the operation of the valves, the Safety Injection System or any other system and therefore, are unrelated to the initiation of any accident or malfunction of equipment of a different type than was previously considered. Therefore, this modification will not create a possibility for any other accident or malfunction and will not jeopardize any equipment, system or procedure required to operate the plant safely and achieve and maintain safe shut down or to prevent the release of radiation for any condition.

C. The implementation of this modification does not reduce the margin of safety as defined in any Technical Specification.

Technical Specifications, Section 3/4.5.2 requires that the physical SI valve positions be verified. The limit strike plate and the Stem Mounted Limit Switch have no capability to control the position of the valves. They are only provided for Control Room indications of the valve positions. Therefore, the modification of the strike plate will have no impact on the Margin of Safety.

DC 91-010
REPAIR/REPLACEMENT OF 24" SW HEADERS TO/FROM UNIT 2 RSHXs
NORTH ANNA / UNIT 2

DESCRIPTION

DCP 91-010, "Repair/Replacement of 24" SW Headers to/from Unit 2 RSHXs," is the third in a series of design change packages developed to clean, inspect, repair and coat the 24" diameter buried and concrete-encased service water (SW) piping which is currently uncoated and experiencing corrosion.

Service water piping has experienced both general area and localized pitting corrosion due to the aggressive nature of the water and the presence of several forms of bacteria which accelerate attack of base metal and welds.

While significant progress has been made recently in corrosion protection of system materials and in control of the bacteria population, certain portions of the system have reached a point where weld repair is necessary to maintain long term structural integrity and pressure boundary capability. This repair, combined with application of a corrosion inhibiting epoxy-based coating, is the scope of this multi-year project.

The specific scope of DCP 91-010 was the refurbishment of approximately 300 feet of 24" piping which forms the supply and return headers to/from the Unit 2 recirculation spray heat exchangers (RSHXs). The work was performed during the Unit 2 outage during the time period when these lines are not required to be operable.

Generally, the work scope involved the following activities for piping which was being repaired:

1) isolation and draining the affected piping,
2) initial blast cleaning to remove corrosion product to facilitate inspection,
3) inspection by engineering personnel to identify areas requiring weld repair,
4) weld repairs,
5) blasting and application of a two-coat epoxy based coating system, and,
6) return to service.

Steps 2 through 5 above were accomplished by personnel entry into the piping. Piping which was replaced was cut out and removed and replaced with shop-coated piping. Field welds were coated from inside the piping after installation.

SUMMARY OF SAFETY ANALYSIS (93-SE-MOD-008)

An unreviewed safety question did not exist because:

A. The implementation of this DCP did not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety and previously evaluated in the UFSAR since the DCP was implemented while Unit 2 was shutdown. The SW lines to/from the RSHXs are not required to be operable during operating modes 5 and 6. These lines were temporarily plugged during the 168-hour Technical Specification (TS) Action Statement (service water TS Section 3.7.4.1.d) with code-qualified welded pipe plugs at the point of origin with the Unit 1 and common sections of the SW system. This allowed the Unit 2 portions to be isolated from the operating portion of the SW system for extended durations to perform the pipe refurbishment activities. Following completion of the pipe work, the temporary plugs were removed.

   A portion of the work involved an excavation to expose and subsequently replace a portion of the buried SW lines. The backfill over these lines provides the tornado missile protection to meet GDC-2 for protection from natural phenomena. The minimum required cover over these lines was not removed until the lines were isolated and out of service. Following pipe replacement, the minimum cover was restored prior to returning the lines to operable status.

   The engineering inspection identified some areas which were below the minimum wall thickness criteria established for various portions of the piping system. For these areas, weld repairs were imposed. Due to the random nature of the corrosion found, it was concluded that the structural integrity of the SW system and its ability to perform all of its required safety functions, were preserved, both for the as-found conditions (prior to weld repair) and as-left conditions (following repair.)

B. The implementation of this DCP did not create a possibility for an accident or a malfunction of a different type than any evaluated previously in the UFSAR since this DCP did not alter the function or capability of the SW system. The pipe cleaning, inspection, repair, and coating enhanced the materiel condition of the system by identifying and repairing isolated minimum wall thickness locations and by providing a barrier to future

corrosion. The work was performed during the Unit 2 outage as stated above, therefore, there was no impact on the ability of the system to perform its intended functions during the implementation of the DCP.

C. The implementation of this DCP did not reduce the margin of safety as defined in the basis of any Technical Specification since the function of the SW system was not changed nor were any changes made in the operability requirements of the system. During the course of the refurbishment process, the existing 168-hour TS Action Statement was entered four times to permit installation of the temporary pipe plugs and their subsequent removal. During these Action Statements, one of the two redundant SW loops was taken out of service. Components which are supplied with service water for cooling were temporarily aligned to only one SW header while the other header was isolated. Use of the Action Statement in this manner was evaluated for any impact on SW system reliability using probabilistic risk assessment techniques. It was found that the time period (potentially up to 28 days) that the system would be operating with one header isolated resulted in an insignificant impact on the base core damage frequency/probability plant model.

DC 91-010
REPAIR/REPLACEMENT OF 24" SW HEADERS TO/FROM UNIT 2 RSHXs
NORTH ANNA / UNIT 2

DESCRIPTION

DCP 91-010, "Repair/Replacement of 24" SW Headers to/from Unit 2 RSHXs," is the third in a series of design change packages developed to clean, inspect, repair and coat the 24" diameter buried and concrete-encased service water (SW) piping which is currently uncoated and experiencing corrosion.

Service water piping has experienced both general area and localized pitting corrosion due to the aggressive nature of the water and the presence of several forms of bacteria which accelerate attack of base metal and welds.

While significant progress has been made recently in corrosion protection of system materials and in control of the bacteria population, certain portions of the system have reached a point where weld repair is necessary to maintain long term structural integrity and pressure boundary capability. This repair, combined with application of a corrosion inhibiting epoxy-based coating, is the scope of this multi-year project.

The specific scope of DCP 91-010 was the refurbishment of approximately 300 feet of 24" piping which forms the supply and return headers to/from the Unit 2 recirculation spray heat exchangers (RSHXs). The work was performed during the Unit 2 outage during the time period when these lines are not required to be operable.

Generally, the work scope involved the following activities for piping which was being repaired:

1) isolation and draining the affected piping,
2) initial blast cleaning to remove corrosion product to facilitate inspection,
3) inspection by engineering personnel to identify areas requiring weld repair,
4) weld repairs,
5) blasting and application of a two-coat epoxy based coating system, and,
6) return to service.

Steps 2 through 5 above were accomplished by personnel entry into the piping. Piping which was replaced was cut out and removed and replaced with shop-coated piping. Field welds were coated from inside the piping after installation.

SUMMARY OF SAFETY ANALYSIS (93-SE-MOD-008)

An unreviewed safety question did not exist because:

A. The implementation of this DCP did not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety and previously evaluated in the UFSAR since the DCP was implemented while Unit 2 was shutdown. The SW lines to/from the RSHXs are not required to be operable during operating modes 5 and 6. These lines were temporarily plugged during the 168-hour Technical Specification (TS) Action Statement (service water TS Section 3.7.4.1.d) with code-qualified welded pipe plugs at the point of origin with the Unit 1 and common sections of the SW system. This allowed the Unit 2 portions to be isolated from the operating portion of the SW system for extended durations to perform the pipe refurbishment activities. Following completion of the pipe work, the temporary plugs were removed.

   A portion of the work involved an excavation to expose and subsequently replace a portion of the buried SW lines. The backfill over these lines provides the tornado missile protection to meet GDC-2 for protection from natural phenomena. The minimum required cover over these lines was not removed until the lines were isolated and out of service. Following pipe replacement, the minimum cover was restored prior to returning the lines to operable status.

   The engineering inspection identified some areas which were below the minimum wall thickness criteria established for various portions of the piping system. For these areas, weld repairs were imposed. Due to the random nature of the corrosion found, it was concluded that the structural integrity of the SW system and its ability to perform all of its required safety functions, were preserved, both for the as-found conditions (prior to weld repair) and as-left conditions (following repair.)

B. The implementation of this DCP did not create a possibility for an accident or a malfunction of a different type than any evaluated previously in the UFSAR since this DCP did not alter the function or capability of the SW system. The pipe cleaning, inspection, repair, and coating enhanced the materiel condition of the system by identifying and repairing isolated minimum wall thickness locations and by providing a barrier to future

corrosion. The work was performed during the Unit 2 outage as stated above, therefore, there was no impact on the ability of the system to perform its intended functions during the implementation of the DCP.

C. The implementation of this DCP did not reduce the margin of safety as defined in the basis of any Technical Specification since the function of the SW system was not changed nor were any changes made in the operability requirements of the system. During the course of the refurbishment process, the existing 168-hour TS Action Statement was entered four times to permit installation of the temporary pipe plugs and their subsequent removal. During these Action Statements, one of the two redundant SW loops was taken out of service. Components which are supplied with service water for cooling were temporarily aligned to only one SW header while the other header was isolated. Use of the Action Statement in this manner was evaluated for any impact on SW system reliability using probabilistic risk assessment techniques. It was found that the time period (potentially up to 28 days) that the system would be operating with one header isolated resulted in an insignificant impact on the base core damage frequency/probability plant model.

DCP 92-012
STATION BLACKOUT DIESEL GENERATOR TIE IN TO STATION
NORTH ANNA POWER STATION UNITS 1 & 2

DESCRIPTION

On July 21, 1988, the Nuclear Regulatory Commission (NRC) amended its regulations in Volume 10 of the Code of Federal Regulation (CFR) to include a new section 10 CFR 50.63, which requires North Anna to be able to withstand and recover from a Station Black Out (SBO) of specific duration without sustaining reactor damage.

The initial Virginia Power approach to meet this regulation was submitted to the NRC in letters on April 17 and 20, 1989. The NRC rejected this approach in their response with Safety Evaluation dated October 18, 1990. Virginia Power provided information on a revised approach to dealing with an SBO by letters dated November 29, 1990 and April 30 and July 31, 1991. The revised approach involved installation of an Alternate AC (AAC) diesel generator to provide power in the event of an SBO. The NRC in a letter dated December 6, 1991, found this revised approach acceptable providing several open items were resolved. Submittals dealing with open items were made on February 10 and March 17, 1992. The NRC on June 8, 1992 issued a Supplemental Safety Evaluation for North Anna which found the plan for dealing with SBO acceptable with only one open item, "Emergency Diesel Generator Reliability".

The SBO AAC building was erected in accordance with DCP 92-010 and the diesel generator and related equipment were installed in accordance with DCP 92-011. Fuel oil lines, service air system, and fire mains were installed per these DCPs. The sprinkler system was made operational under DCP 92-012.

This DCP installed and made functional the tie in of 480 volt power from the station to the building and the 4160 volt feeders from the building to buses in the existing station normal switchgear room. These 4160 volt feeders allow power from the SBO AAC diesel generator to be provided to any of the three existing Transfer Buses and subsequently from them to any of the four (two per unit) emergency busses.

D92-012

SUMMARY OF SAFETY ANALYSES (94-SE-MOD-009, 94-SE-MOD-038, 95-SE-MOD-019, 95-SE-MOD-031, & 95-SE-MOD-037

Safety Analysis 94-SE-MOD-009 was prepared to support initial construction and installation, but did not allow for tie in to any buses or existing equipment.
Safety Analysis 94-SE-MOD-038 supported tie in to existing equipment and operation when the tie in was completed. This included: (1) non-outage modification of breaker cell switches, (2) non-outage tie in to Transfer Bus D and testing of the SBO AAC diesel (emergency bus normally fed from this Transfer Bus fed from alternate source), (3) performance of modifications during appropriate windows in the Unit 1 1994 refueling outage, (4) performance of remaining modifications during appropriate windows in the Unit 2 1995 outage, and (5) operation after completion.

Safety Analyses 95-SE-MOD-019, 95-SE-MOD-031, AND 95-SE-MOD-037 were prepared in support of changes in scope and logic for tie ins during the Unit 2 1995 outage caused by further evaluation of when work should be performed and caused by changes in plant equipment status from that expected when initial planning was performed.

The majority of the information presented in each of these Safety Analyses did not change from document to document and the conclusion of no adverse impact on safety did not change, but details did change to reflect expected conditions and evolutions.

All accidents were reviewed and the following were found to be possibly impacted by this design: Partial Loss of Reactor Coolant Flow and a partial or complete Loss Of Off-Site Power. However, the Partial Loss of Reactor Coolant Flow discussed here is different from and not as severe as that discussed in UFSAR 15.2.5, since the condition discussed here involves a partial loss of flow after a reactor trip and generally after the RCP has continued to run fed from the generator leads and grid for thirty seconds. The UFSAR discusses a loss of flow due to RCP related failure at power.

1. Neither Accident probability, accident consequences, nor possible accident type were impacted by this DCP.

   During the modification of the cell switches of the non-safety related reserve station service feeder breakers for the station service buses, these breakers were out of service. When the units are running these breakers are normally open and therefore do not impact operation.

Had Unit 2 trip while work was in progress on one of its breakers, the load from that Station Service Bus would not transfer to its respective RSST and some loads (including the RCP) would not be powered. However, the emergency buses would still be powered by their RSST with their EDG as back up.

Similarly if Unit 1 had tripped and its 500 kV switchyard breakers opened, then its station service loads would transfer to their respective RSST. If work had been in progress, then the load associated with that breaker would not be powered. A more severe version of this event has previously been evaluated in UFSAR section 15.2.5.

Some of these breakers were worked during outages which precluded any impact. Additionally, only one breaker was worked at a time and each breaker was generally out of service for less than a shift.

There was no impact on probability of the UFSAR accident, since the breaker out of service would have only impacted RCP operation after a unit trip and attempt to load to the RSST.

SBO modifications did not increase the probability of occurrence of malfunction of safety related equipment since interfaces with such equipment were avoided.

Once installation of the SBO AAC diesel was completed in 1995 the probability of a LOOP could be considered to be reduced since the SBO AAC diesel constitutes another source of power to the emergency buses other than their specific on-site sources (their EDGs).

In the event of the loss of off-site power concurrent with the loss of power to the emergency bus, the emergency diesel generator will power the emergency bus for the safe shutdown of the plant. The AAC system will provide backup to the emergency diesel generator in case of failure. The installation of the AAC system did not and increase the probability of occurrence of malfunction of any equipment. Since installation is complete it will be an additional backup to minimize the consequences of the malfunctions identified.

During implementation, two off site power supplies were maintained to the emergency busses of each unit while it was on line and no modifications were performed which could impact the operability of an EDG or other equipment important to safety. Modifications related to the tie

In the event of the loss of off-site power concurrent with the loss of power to the emergency bus, the emergency diesel generator will power the emergency bus for the safe shutdown of the plant. The AAC system will provide backup to the emergency diesel generator in case of failure. The installation of the AAC system will not increase the consequences of occurrence of malfunction of any equipment but will be an additional backup to reduce the consequences of the malfunctions identified since the SBO diesel can power an unpowered emergency bus for either unit.

This design change will not create the possibility for a malfunction of equipment different than was previously evaluated in the SAR. The AAC system provides a backup power supply which can power the emergency bus when required through operator action. Power from the AAC system may be routed to the emergency bus through the D, E or F transfer bus. The operators must take manual action in order to tie the AAC system in to the normal station electrical system. This would only be done if the operator needed the AAC as a power source and could be reversed as needed. The AAC diesel will supply power to an emergency bus in a manner similar to and at the same location as the normal off-site power system, therefore it will not create the possibility of a different malfunction.

3. The margin of safety has not been reduced.

During implementation care was taken to assure that Technical Specification Limiting Conditions of Operation were not entered. Detailed planning was performed to assure that work was done in appropriate windows and with the plant in appropriate configurations.

There was no reduction in the margin of safety since safety related system and important to safety system performance was unaffected. Additionally, it can be argued that the margin of safety has been improved because there is now an additional highly reliable power supply available in case of a loss of off-site power. Based on subsequent probabilistic analysis review of the risk of core damage, a significant reduction in this frequency was determined to have occurred due to this installation.

DCP 92-016
INSIDE RS TRANSMITTER REPLACEMENT
NORTH ANNA UNITS 1 & 2

DESCRIPTION

The existing discharge pressure transmitters (RS-PT-152A, -152B, -252A, and -252B) for the Inside Recirculation Spray (Inside RS) Pumps are located inside Containment (at floor elevation 241'-0"), but were not qualified for a LOCA or MSLB harsh environment, and were not on the Equipment Qualification Master List (EQML). In order to comply with RG 1.97 for a Category 2, type D variable, these transmitters must be environmentally qualified for the environment in which they must operate. Similarly, the (existing) cable installed inside Containment between the transmitter and the penetrations was not qualified for the environment in which the circuits must function. The cable inside Containment was also replaced.

SUMMARY OF SAFETY EVALUATION

This Design Change was performed to replace Foxboro transmitters 1/2-RS-PT-X52A and B with qualified Rosemount transmitters. These transmitters provide Control Room indication of Regulatory Guide 1.97 variable D-23 (Containment Spray flow).

This Design Change does not constitute an Unreviewed Safety Question since the transmitters were replaced with environmentally qualified transmitters located above the submergence level inside Containment; the new transmitters will be operated within their qualified voltage range. These transmitters provide indication only; no protection or control loops are affected. No new failure modes exist, since this is a replacement of existing equipment with equal to or better than existing equipment.

The following accidents were considered in this evaluation:

1. LOCA
2. MSLB inside Containment

DCP 92-215
REPLACE PUMP TO ENABLE RESIN TRANSFER SYSTEM
NORTH ANNA UNIT 1

DESCRIPTION

The waste solids pump skid was originally installed for transferring spent demineralizer slurry, and dewatering and solidifying radioactive waste in preparation for shipping. Since the equipment was installed, the methods of conducting radwaste operations had changed. Most of the equipment on the skid had been abandoned in place. The only remaining operating equipment on the skid was the radioactive waste metering pump and its related piping and valves. The pump was undersized for the application and was unreliable.

The abandoned waste solids equipment was a source of personnel radiation exposure as was the metering pump due to its high maintenance. The abandoned waste solids equipment was removed and the positive displacement radioactive waste metering pump was replaced with a centrifugal pump with higher capacity and reliability. None of the equipment affected was safety related, however, UFSAR Section 11.5, Solid Waste System, required revision as a result of this change.

SUMMARY OF SAFETY ANALYSIS (92-SE-MOD-035)

All accidents were reviewed and none were considered to be applicable.

1) Accident probability was not increased as the waste solidification system had no affect on the operation of the unit. The equipment had no role in the occurrence of any of the analyzed accidents.

2) The consequences of an accident was not affected. The location of the equipment affected and their interfaces with safety related equipment was sufficiently remote that there was change in the consequences of an accident.

3) No unique accident possibilities were created. The equipment removed had not been used for more than ten years. The change in the type of resin transfer pump did not affect system operation.

4) Margin of Safety was maintained because the liquid waste system is not described in the Technical Specification basis.

DCP 92-267 SUMMARY
BLOCKWALL SB-254-4 REPAIRS

Major Issues:

The major issue associated with this Safety Evaluation is as follows:

As a long-term corrective action to NAPS Deviation Report No. N-92-1585, structural reinforcement was required on block wall SB-254-4 to restore the originally intended end fixity condition (i.e. pinned-end). This block wall is a Safety-Related. Class 1, Seismic block wall that runs along column row "8", between rows "C" and "E", separating NAPS Units 1 & 2 ESGR's. The block wall was previously analyzed and reinforced as part of the US NRC IE Bulletin 80-11 review. The reinforcement associated with this Design Change Package was required to restore the originally intended pinned-end condition of the block wall that was invalidated by a cracked mortar joint at the point where this block wall abuts column row "C". Structural reinforcement was anchored to the concrete wall at "C" line, with the outstanding legs of the reinforcing angles laterally supporting the block wall end. No equipment tag-outs or special operating conditions were required to implement these repairs.

As a Safety-Related, Class 1, Seismic block wall, this wall supports Safety-Related electrical equipment and must be able to withstand the effects of an OBE and DBE seismic event. It has been previously concluded (see memo attached to DCP from C. E. Sorrell to G. E. Modzelewski, dated 08-06-92) that the cracked mortar joint did not jeopardize any Safety-Related equipment within the collapse envelope of this block wall. These structural repairs have restored the originally assumed end fixity conditions, maintaining seismic stresses within allowable design limits, as specified in calculation no. 13075.63-SB-254-4R, Rev. 4, Addendum 4A.

UNREVIEWED SAFETY QUESTION ASSESSMENT:

These modifications did not increase the probability of experiencing an OBE or DBE seismic event, as seismic activity is unrelated to the implementation of these repairs. The consequences of an OBE or DBE seismic event are unchanged, in fact the consequences are restored to the originally assumed levels, since seismic stress levels in the block wall will be maintained as analyzed by the completion of these repairs. In as much as these structural reinforcements are anchored into "C" line wall, via standard Work Procedures, without special equipment tag-outs, no concerns for an accident of a different kind, other than previously analyzed, were known to exist.

DCP 92-345
"MODIFICATIONS TO THE CASING COOLING RECIRCULATION PUMPS, CHILLERS AND TEMPERATURE INDICATORS"

SUMMARY:

Temperature control of the unit 2 casing cooling tank was superior to the configuration of the unit 1 tank since the unit 2 recirculation pump runs continuously. The temperature controls for the unit 1 casing cooling tank was changed so that it operates similar to unit 2. Running the recirculation pump continuously is superior since it prevents temperature stratification and promotes chemical homogeneity.

As a human factors improvement, the range of the casing cooling tank temperature indicators were changed from 0°F-150° to 0°F-75°F since the tank must be maintained between 35°F and 50°F.

UNREVIEWED SAFETY QUESTION ASSESSMENT:

1) Accident probability has not increased because the casing cooling subsystem does not perform a safety function.

2) Accident consequences have not increased. The casing cooling tank is still maintained at the proper temperature. The tank's ability to deliver water to the outside recirculation spray pumps has not been altered.

3) No unique accident probabilities have been created. The implementation of this DCP has not created a possibility for an accident or a malfunction of a different type than any evaluated previously in the UFSAR because the casing cooling recirculation pumps and chillers do not perform a safety function.

4) Margin of Safety has been maintained because the casing cooling tank is still able to deliver cool borated water to the outside recirculation spray pumps.

DCP 92-353
NORTH ANNA POWER STATION

REMOVAL OF MAKEUP PCV TO THE CONTROL & RELAY ROOM CHILLED WATER SYSTEM

DESCRIPTION:

Rust and debris becoming lodged between the seating surfaces of 02-HV-PCV-2303 did not allow the pressure control valve to completely seat. This resulted in over filling of the chilled water system. Maintenance tried cleaning the PCVs 01-HV-PCV-1303 and 02-HV-PCV-2303 several times and never was able to get the PCV to operate correctly for more than a couple of days (Ref. Attachment #1 of EWR 90-040C). The control room and relay room air conditioning chilled water system is closed system and makeup is only required periodically. The problem on unit 1 was resolved via DCP 93-352-1 by removing the PCV and rerouting the piping to allow a manual valve to be accessible to the operator.

The pressure control valve 2-HV-PCV-2303 and it's associated valves were eliminated. The 2-CD-181 gate valve was replaced with a globe valve; and it is being used to maintain water inventory.

SUMMARY:

This activity required the UFSAR figure 10.4-2B to be updated; therefore the facility as described in the safety analysis report required change.

The safety analysis report was reviewed. No accidents previously evaluated were identified as being applicable. The work to be performed is not safety related, is non seismic, and non EQ. This activity will not increase the probability or the consequences of any malfunction. Nor will it create the possibility for an accident or malfunction of a different type than was previously evaluated in the UFSAR. The operating license and technical specification will not require a change as a result of this activity. The margin of safety as defined in the Technical Specifications will not be reduced. The fire protection system will be unaffected by this activity. The activity is benign and will not have an adverse environmental impact. There will be no change in effluents or power level as a result of the proposed change. For these reasons a unreviewed safety question did not exist and this modification was allowed.

DCP 92-360
Fuel Transfer Tube Flange
North Anna Unit 2

DESCRIPTION

The existing blind flange on the containment side of the fuel transfer tube was attached utilizing 20 bolts which must be removed and reinstalled for each refueling. In order to reduce to time to take off and put back the flange, a new flange was installed needing only 4 bolts to secure. The new flange was fabricated with a double concentric ring of grooves to accept "Quad ring" seals. The advantages also include decreased radiational exposure.

SUMMARY OF SAFETY ANALYSIS

Design Change DCP 92-360 does not create an "unreviewed safety question" as defined in 10 CFR 50.59.

A. The implementation of this modification does not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety and previously evaluated in the Safety Analysis Report.

   The fuel transfer tube blind flange is not the initiator of or a contributor to a previously analyzed accident. The bolt stresses have been shown to remain within the allowable values. The leak-tight sealing capability has been demonstrated. Therefore, the design function of the reactor containment is maintained.

B. The implementation of this modification does not create a possibility for an accident or malfunction of a different type than any previously evaluated in the Safety Analysis Report.

   It has been demonstrated by calculation that the design requirements for the fuel transfer tube cover and bolting continue to be met, and the containment isolation function is maintained.

C. The implementation of this modification does not reduce the margin of safety as defined in any Technical Specification.

   The margin of safety of the transfer tube and cover is defined, in part, by the requirements imposed by the ASME Code. These design requirements have been shown to be met by Westinghouse Engineering calculation. The containment isolation is maintained.

Equipment Hatch Platform Extension
North Anna / Unit 2

DESCRIPTION

This Design Change modified and extended the existing equipment hatch platform outside Unit 2 containment as a prerequisite for the steam generator replacement (SGR). The Steam Generator Lower Assemblies (SGLA) had to be rigged through the containment equipment hatch. The original equipment hatch platform was too small to fit the SGLA, the necessary runway beams, rigging hardware, and to provide space for the rigging personnel. In addition, the original platform was stiffened for the SGLA loads.

New concrete footings and structural steel were used to erect a permanent non-safety platform extension approximately 26′ by 40′ which is structurally independent of the original safety related platform. The original access ladder was removed and a new 4-foot wide stair was installed to provide access to the grating level at elevation 291′-10". Platform extension framing included support for a future jib crane.

Prior to implementation, consideration was given to protecting existing underground utilities (casing cooling piping, storm drain, electrical conduit) in the vicinity of excavations for platform and stair footings. Special implementation instructions were followed (including soil removal by hand or a vacuum method) which avoided potential damage and maintained the availability of these utilities during and after construction.

SUMMARY OF SAFETY ANALYSIS

An unreviewed safety question did not exist because:

A. The accident previously considered by the SAR involves the protection of structures, systems and components from the effects of natural phenomena such as earthquakes and tornadoes. The original platform and missile shield were specifically designed to maintain protection from the effects of natural phenomena. The only reason that the original platform is classified as safety related is that it provides support for the tornado missile shield (labyrinth) at the containment equipment hatch. Implementation of DC 93-009 had no effect on the probability of the occurrence of these events. These modifications were specifically designed to maintain existing protection from the effects of natural phenomena; therefore the consequences of accidents identified will not change.

B. Operability of the casing cooling system was not affected by construction; therefore, consequences of a LOCA or MSLB inside containment were not increased. The ability of the original platform to perform its safety related function was not diminished. The potential for damage to buried casing cooling lines caused by excavating, tornado missile or a seismic event was evaluated prior to implementation and special instructions were followed which maintained operability of these lines. Therefore, no new accident was created.

C. No malfunctions of equipment associated with the original platform or missile shield were considered in the SAR because these structures were provided specifically to provide missile protection for the equipment hatch and equipment inside containment. No malfunction of the casing cooling system was considered since provisions of DC 93-009 maintained the seismic qualification of the buried lines and missile protection for casing cooling is not required per the UFSAR. There are no different equipment malfunctions which could have occurred as a result of implementing DC 93-009. The ability of the original platform to support the labyrinth missile shield in its design position at the equipment hatch cannot affected by the postulated failure of any new work installed by this Design Change.

D. Containment integrity is addressed in TS 3/4.6.1.1. Since the DCP did not degrade the safety related performance of the original platform or missile shield, there was no reduction in any TS Margin of Safety. T.S. 3/4.6.2 bases were not affected since excavation did not result in loss of seismic integrity of the buried casing cooling lines.

PRE-SGR INSIDE CONTAINMENT MODIFICATIONS
NORTH ANNA - UNIT NO. 2

DESCRIPTION

The removal and replacement of the steam generators requires that interferences to steam generator movement, within the containment between the permanent steam generator (SG) locations and the equipment hatch, be removed. These interferences required concrete cutting and included portions of the SG biological shield walls, a section of the crane wall in front of the equipment hatch at the existing opening, and a portion of the operating floor in front of the equipment hatch. Miscellaneous electrical components attached to the biological shield wall and crane wall were temporarily detached in order to allow concrete cutting operations and replacement of the steam generators to be performed.

In addition, the shims at the hot and crossover leg elbow whip restraints were removed. The support tower for the auxiliary crane, which was used during the SG replacement, was installed at the beginning of the replacement outage.

1. Concrete Cutting

   a. Sections of the biological shield walls for the steam generators were cut above the operating deck and re-secured with through bolted splice plates. This modification was part of the rigging path for the steam generators.

   b. An overhead section 1′ X 8′ of the polar crane wall at the equipment hatch was removed to allow passage of the steam generator transition cone.

   c. A 5′-9" X 8′ X 10" section of the operating deck located in front of the equipment hatch was removed to allow alignment of the transition cone with the hatch barrel. A removable steel floor was installed in place of the removed concrete section.

2. Relocation/Removal of Interferences

   Electrical conduits, receptacles, instrument air tubing, secondary sample system piping and their respective supports were temporarily removed and subsequently reinstalled as required to support cutting and removal of the steam generator biological shield wall. Additionally the containment dome air recirculation fan supports on biological shield wall C were modified.

3. Pipe Whip Restraint Shims

In order to provide adequate clearance for the reinstallation of the reactor coolant loop piping, the hot leg and crossover leg pipe whip restraint shim assemblies were permanently removed. The restraint structures were left in place.

4. Auxiliary Crane Support Tower

An auxiliary crane support tower was installed temporarily on the operating floor slab. During the steam generator replacement outage, the auxiliary crane was mounted on top of the support tower to support material handling. Both the crane and the support tower were removed from the containment prior to the end of the steam generator replacement outage.

SUMMARY OF SAFETY ANALYSIS

Modifications implemented by this design change did not create an unreviewed safety question as defined by 10CFR50.59. A discussion of the four (4) activities follows.

1. Concrete Cutting

The containment structure is designed to sustain, without loss of required integrity, all effects of gross equipment failures up to and including the rupture of the largest pipe in the reactor coolant system and any condition resulting from a LOCA. The in-containment structural modifications do not affect the performance or integrity of the containment. The modifications to the steam generator biological shield wall and the section of the operating floor have been analyzed and found structurally acceptable under normal and accident loadings. The substitution of the structural steel platform for the removed section of the operating floor did not affect seismic loads. The permanent enlargement of the opening in the polar crane wall was analyzed and found structurally acceptable under seismic and crane loading.

2. Relocation/Removal of Interferences

Relocation, removal and subsequent reinstallation of electrical conduits, receptacle, instrument air tubing, sample system piping and respective supports meets plant specifications and does not affect any safety function.

3. Pipe Whip Restraint Shims

The current NRC philosophy concerning the Leak Before Break analysis for the North Anna reactor coolant loops allows the removal of the pipe whip restraints, including shims. The restraints were not considered in the containment integrity analysis and therefore the containment integrity analysis is unaffected by this modification. Allowing the restraint, minus the shims, to remain in place does not alter the ability of the reactor coolant system to perform its intended function. The pipe whip restraints are no longer part of the North Anna design basis.

4. The installation of the auxiliary crane support tower inside the containment was evaluated for design basis seismic loading and found to be acceptable. The tower was removed from the containment prior to completion of the steam generator replacement outage under design change 93-011.

STEAM GENERATOR REPLACEMENT
NORTH ANNA UNIT 2

DESCRIPTION

Due to the degradation of the previous steam generator tubing, the lower steam generator tube bundle assemblies have been replaced at North Anna Power Station Unit 1. The new steam generator lower assemblies were fabricated in accordance with ASME Code Section III, 1986 Edition and have physical, mechanical, and thermal characteristics that are consistent with the original design and safety analysis presented in the Updated Final Safety Analysis Report (UFSAR). The new steam generator lower assemblies are designed and fabricated to be physical duplicates of the original lower assemblies since all major external dimensions and orientation angles for both the original and new components are essentially the same.

Certain design changes and enhancements have been made in the new steam generator lower tube bundle assemblies which address the operating experience of the original steam generators and which enhance the overall reliability and maintainability of the steam generators. These changes and enhancements do not adversely affect the mechanical or thermal-hydraulic performance of the new steam generators.

Specifically, some of these enhancements are the utilization of thermally-treated alloy 690 tubing to reduce the susceptibility of stress and intergranular corrosion experienced by the previous mill-annealed alloy 600 tubing. In addition, the incorporation of an additional row of anti-vibration bars uniformly inserted into the tube bundle provides increased support in the tube bundle region, reducing the susceptibility of the tubes to vibration. The number of tubes has also increased for additional plugging margin.

The steam generator replacement was performed in accordance with the requirements of the ASME Code, Section XI, 1983 Edition and Summer 1983 Addenda. Welding, postweld heat treatment, nondestructive examination, and baseline inservice inspection were performed in accordance with the ASME Code, Section XI, 1983 edition; ASME Code, Section III, 1986 edition; and ANSI B31.7, 1969 edition through 1970 addenda, as applicable.

The steam generator lower assemblies were be removed and replaced through the existing containment equipment hatch. This replacement process is commonly referred to as the two-piece replacement method. The two-piece replacement through the equipment hatch was determined to be the best overall method for North Anna Unit 2 due to limitations on the diameter of equipment that can be moved through the equipment hatch. The containment equipment hatch is large enough to allow passage of the steam

generator lower assemblies only, not the steam domes. The previous SG lower assemblies were removed by severing the reactor coolant and all other attached piping at the steam generator nozzles, severing the steam generators within the transition cone, and removing the lower assemblies from the containment. Following removal of the old lower assemblies, the new shop-fabricated steam generator lower assemblies were transported into the containment through the equipment hatch and connected to the original steam domes and reactor coolant piping. Use of the equipment hatch eliminated the need to modify the containment wall concrete or pressure boundary.

Additional enhancements resulting from the replacement of the steam generators are:

- The steam generator blowdown nozzles coupling size has increased to 2-1/2". The previous 2" blowdown piping from these nozzles to the 3" headers has been replaced with 2-1/2" chrome-moly steel pipe for increased erosion/corrosion resistance. Additionally, the previous 1" carbon steel drain line has been replaced with 1" chrome-moly material. These modifications will provide for additional blowdown capability in the future.

- Portions of the feedwater piping loop seal at the steam generator has been be replaced with chrome-moly material. This piping has been replaced to alleviate future erosion/corrosion concerns.

- The steam generator and adjacent piping has been covered with new blanket insulation which has stainless steel jacketing. The replacement blanket insulation meets or exceeds the design requirements of the previous insulation.

- The steam generator upper restraints have been replaced with an equivalent restraint. Demolition of the previous upper restraints was required due to difficulties associated with their removal from the steam generators and the effort involved to reuse them.

SUMMARY OF SAFETY EVALUATION

The Safety Evaluation addresses eight major activities associated with steam generator replacement as follows:

I. Steam Generator Vessel Repair
II. Piping Removal and Replacement
III. Instrumentation Removal and Replacement
IV. Blowdown System Modification
V. Insulation Removal and Replacement
VI. Rigging Activities Inside the Containment
VII. Rigging and Transport of Heavy Loads Outside the Containment

VIII. Temporary Services (Impact on permanent facilities only)

I. Steam generator vessel repair constitutes replacement of all three steam generator lower assemblies, including replacement of tubes, tubesheet, lower vessel shell, channelhead, and a portion of the wrapper plate and transition cone. The repair also includes the installation of a flow restrictor in the main steam nozzle and removal of the downcomer flow resistance plates, replacement of the steam generator upper lateral restraints, and new materials for the steam generator lower supports.

II. Piping removal and replacement includes all piping systems attached to the steam generators. These systems include the reactor coolant, main steam, feedwater, chemical feed, wet lay-up, and sample piping which were severed at the steam generator nozzles to allow for the removal of the original steam generator lower assemblies and the installation of the new lower assemblies. The severed piping was reinstalled in essentially the same configuration. Material upgrades from carbon steel to chrome-moly were utilized on a portion of the feedwater loop seals for improved erosion/corrosion characteristics. In addition, decontamination of the reactor coolant system piping following the severance cuts is addressed.

III. The steam generator level instrument piping and tubing was severed to allow removal of the original steam generator lower assemblies and installation of the new lower assemblies. The condensate pots and instrument root valves were removed and replaced. The severed piping and tubing as well as the condensate pots were reinstalled to satisfy the original design requirements with material upgrades. In addition, the optical templating bracket installed under DC 93-007-2 was removed.

IV. The previous 1 and 2 inch carbon steel blowdown lines connected to each steam generator lower assembly were replaced with new 1 and 2 1/2 inch chrome-moly lines respectively. The supports associated with the piping were removed and modified/replaced.

V. The original steam generator insulation, of which part was reflective and part was encapsulated fiberglass, was replaced with a blanket-type of insulation that exhibits equivalent thermal properties.

VI. Lifting and handling activities required to support removal and installation of steam generator lower assemblies were evaluated.

VII. Erection, operation, load test, and disassembly of the outside lifting system adjacent to the containment equipment hatch. Establishing and testing the proposed haul route to be used to transport the new steam generators to the equipment

hatch, transport the old steam generators to the Old Steam Generator Storage Facility, and transport other heavy loads to and from the containment.

VIII. Temporary modifications to support steam generator replacement were required. These temporary modifications included attachment of a flexible duct and volume control damper to the purge system, modification to RCP-1B power supply for temporary steam generator replacement power, modification to security door A-95-1, an auxiliary crane, a jib crane, temporary main steam work platforms, and a reactor cavity cover.

The probability of occurrence for the accidents previously identified have not been increased as discussed below.

I. The probability of accident occurrence associated with the replacement of the SGs has not increased because the design, materials, and code standards for installation are equal to or more conservative than those used in the original licensing basis.

II. All reactor coolant system and secondary side piping and supports have been restored to their original design configuration in accordance with ASME Section XI and ANSI B31.7 code requirements. Replacement materials, including all weld metal utilized, satisfy the original code requirements and meet the existing installation specification. All modified piping systems were subjected to nondestructive examination and hydrostatic testing in accordance with the Section XI, the Special Processes Manual (DC 93-11, Reference 6.5) and DC 93-11, Appendix 4-21, as applicable. In addition, periodic inspection will continue throughout the remaining life of the plant.

III. The piping, instrument tubing, and condensate pots were reinstalled to satisfy the original design requirements with material upgrades. The piping, instrument tubing, condensate pots, root isolation valves, and vent valves were replaced with a material which meets or exceeds existing material characteristics. Therefore, the probability of an accident has not increased from the original licensing basis.

IV. The modification to the steam generator blowdown system enhances the reliability of the blowdown system with respect to erosion/corrosion concerns. All supports associated with the blowdown system modifications have been reviewed to ensure that the blowdown system meets the original seismic design requirements. The improvements made to the blowdown system meet or exceed the current licensing basis requirements and do not increase the probability of occurrence of an accident.

V. The replacement blanket insulation has been procured and installed to meet or exceed the original design requirements for

heat transfer and has been seismically qualified. Failure of the blanket insulation cannot initiate any of the relevant accidents addressed by this design change. Therefore, the issue of probability of accident occurrence is not applicable.

VI. Rigging activities covered by this safety evaluation that may be performed during defueling/refueling were performed in accordance with the existing station heavy loads procedures and in a manner that will not interfere in any way with defueling or refueling operations that may be in progress.

VII. Activities associated with the rigging and transport of heavy loads outside the containment including the steam generator lower assemblies, haul route test load, etc. did not increase the probability of occurrence of an accident.

VIII. The temporary modifications do not have the potential to increase the probability of occurrence of relevant accidents while the unit is defueled.

The modifications addressed in this design change package do not increase the consequences of an accident for the following reasons:

I. As a result of the steam generator vessel repair, Technical Report NE-983, Revision 0 was prepared to consolidate and summarize the safety analyses and evaluations supporting North Anna 2 operation following the steam generator repair. In this report, each UFSAR Chapter 15 accident analysis applicable to North Anna Unit 2 operation with the repaired steam generator has been evaluated and it has been determined that none of the accident consequences were found to be more limiting than those currently documented in the UFSAR.

II. The reactor coolant, main steam, feedwater, wet layup, sample, and chemical feed systems will be restored to their original configuration after SG replacement. As described in approved calculations, all applicable design basis seismic stress and support analyses have been evaluated/performed, as applicable, to verify the capability of the repaired systems to perform their intended functions.

III. The piping, instrument tubing, and condensate pots were reinstalled to satisfy the original design requirements using upgraded material. Therefore, all design basis evaluations of the consequences of accidents for which steam generator level instrumentation is assumed to be operable remain valid.

IV. The function and operation of the steam generator blowdown system did not change. The increase in pipe size does not increase the consequences of an accident since these pipe sizes are bounded by the larger break sizes used in the accident

analysis. Therefore, all design basis evaluation of the consequences of accidents involving the blowdown system remain valid.

V. The replacement thermal insulation has been qualified for use within containment and has been procured to meet the post accident environmental conditions within containment. The replacement insulation is seismically installed to ensure the insulation remains attached to the generator in the event of a seismic occurrence. The insulation performs no safety function in the event of a design basis accident. Appropriate evaluations have been performed in accordance with the recommendations of Regulatory Guide 1.82 to ensure that the replacement insulation does not adversely affect emergency core cooling and engineered safeguards systems. A debris and NPSH analysis has been performed by calculation (ER&D 93-011, References 6.12.1.108 and 6.12.1.109) with the results incorporated into Technical Report NE-983 (DC 93-011, Appendix 4-23, Safety Analyses and Evaluations Supporting North Anna 2 Operation Following Steam Generator Replacement). The impact of the replacement fiberglass blanket insulation with respect to the NPSH requirements for the low head safety injection pumps and the inside and outside containment recirculation spray pumps as well as the potential long term blockage of the sump suction screens has been evaluated and determined to be acceptable. Therefore, all design basis evaluations of the consequences of accidents are unaffected by the replacement insulation.

VI. All movements of heavy loads within containment while fuel remain within the reactor containment were conducted in accordance with the existing station heavy loads procedures to ensure that the loads remain within the established safe load paths and to ensure that, in the inadvertent event of a load drop, the consequences remain within the established acceptance criteria. No safety-related equipment would be adversely impacted by a drop outside the containment. However To assess the radiological consequences associated with this drop, an analysis was performed. The acceptability of the offsite dose consequences associated with a postulated drop have been evaluated and compared to the consequences of other events in the same class of postulated accidents for waste gas or waste liquid releases. The evaluated consequences of a steam generator lower assembly drop are within the applicable regulatory guidelines and are less than the limiting, and more permanent, licensing basis accidents currently evaluated in the UFSAR. Thus, the consequences associated with this class of accidents will not be increased.

VII. Activities associated with the rigging and transport of heavy loads outside the containment including the steam generator lower assemblies, haul route test load, etc. will not increase the consequences of any accidents.

VIII. In the defueled condition, all UFSAR accidents associated with reactor operation, reactor criticality and reactor decay heat removal are not credible occurrences. The only accident which required consideration regarding temporary modifications was the fuel handling accident inside containment. During all fuel handling, containment integrity was maintained. Modification to the containment purge system was not made until the vessel was defueled. The temporary main steam work platforms were installed in accordance with the Station Heavy Loads Procedure (0-MCM-1303-01) and/or STD-CEN-0042, Control of Heavy Loads.

The possibility for an accident of a different type than was previously evaluated in the Safety Analysis Report has not been increased as justified below:

I. The possibility of an accident that is different from that already evaluated in the UFSAR is not created because, as evaluated in Westinghouse Safety Evaluation SECL-93-274, the replacement steam generators have been designed and fabricated to criteria that are equivalent to or better than the existing steam generators. The replacement steam generators have also been determined to have no adverse impact on the function or performance of connected systems, components and structures. The upper restraints were not removed prior to fuel offload and were reinstalled prior to refueling.

II. All reactor coolant system and secondary side piping and supports were restored to their original configuration in accordance with the original code requirements using materials which meet or exceed the original design requirements. Various configurations of primary and secondary pipe cuts with fuel in the pool were evaluated and seismic supports specified where required to maintain a seismicly acceptable system.

III. The piping, instrument tubing, were reinstalled to satisfy the original design requirements using upgraded material. The function and operation of the system following the modification did not change. Therefore, the possibility of an accident of a different type from that evaluated previously would not be created.

IV. The blowdown piping changes implemented by this modification did not change the function and operation of the steam generator blowdown system. Therefore, the possibility of an accident of a different type from that evaluated previously would not be created.

V. The replacement insulation performs the same function as the original insulation and is seismically installed. Failure of the SG insulation does not, in itself, initiate any existing type of accident. The change from encapsulated insulation to blanket

insulation does not alter this conclusion. Therefore, no new accident is possible as a result of the replacement insulation.

VI. Rigging activities performed during defueling operations were conducted in accordance with the existing heavy load handling procedures to ensure that load handling occurs only in currently analyzed and approved safe load paths. The radiological consequences of a postulated drop of an old steam generator lower assembly inside the containment or within the protected area have been evaluated and determined to be within applicable regulatory limits and less than the limiting case events within the same classification of accidents currently evaluated in the UFSAR. Thus, no new accidents are created as a result of the rigging activities.

VII. Activities associated with the rigging and transport of heavy loads outside the containment including the steam generator lower assemblies, haul route test load, etc. did not create the possibility of an accident of a different type than previously evaluated.

VIII. Following the completion of the SGR and prior to a return to power, all temporary modifications were removed. Thus, this activity did not create the possibility of an accident of a different type as the operating performance of the plant following SGR is identical to the operating performance before SGR.

The margin of safety as defined in the basis for any Technical Specification is not reduced by this design change. The replacement steam generators have been demonstrated to insignificantly affect the transient system response during postulated UFSAR Chapter 15 accidents. Accident analyses for all UFSAR Chapter 15 transients have been performed which bound allowable operation in accordance with the North Anna 2 Technical Specifications that will be applicable following steam generator replacement. All accident analyses meet their respective acceptance criteria. It may, therefore, be concluded that steam generator replacement does not decrease the margin of safety as defined in the basis for any Technical Specification.

DC 93 017
MAIN GENERATOR PROTECTION MODIFICATIONS

DESCRIPTION

This modification implemented the following upgrades to circuits providing Main Generator protection:

1) The existing electromechanical negative sequence relay was replaced with a static relay which provides protection for the entire negative sequence range of the generator.

2) Two reverse power relays were installed to improve the reliability of the anti-motoring trip function.

3) A new synchronizing system was installed to improve the operator's ability to place the Unit on line with very small voltage, phase and frequency differences.

SUMMARY OF SAFETY EVALUATION(95-SE-MOD-10)

All UFSAR chapter 15 accidents and malfunctions were reviewed and the following were found applicable:

- Loss of electrical load and/or turbine trip
- Loss of normal feedwater
- Loss of offsite power to station auxiliaries

This design change did not create an unreviewed safety question as defined by 10CFR50.59 because:

1) Accident or malfunction probability was not increased. The relay changes provide improved protection of the main Generator in accordance with industry standards and Virginia Power System Protection Requirements.

2) Accident or malfunction consequences were not increased. The applicable accidents and malfunctions are mitigated by the reactor protection system,which is not affected by the relay additions implemented.

3) No new accident or malfunction possibilities were created. The relay changes improve the reliability of existing generator protection functions.

4) The relay changes affect any margin of safety as defined in the Bases of any Technical Specification.

DCP 93-167

REPLACE AFW PRESSURE TRANSMITTER 2-FW-PT-203B
NORTH ANNA POWER STATION
UNIT #2

Executive Summary

The existing Foxboro E11GM suction pressure transmitter (2-FW-PT-203B) on the motor driven auxiliary feed water pump (2-FW-P-3A) has been replaced with an equivalent Rosemount.

SUMMARY OF SAFETY ANALYSIS (95-SE-MOD-015)

This design change did not create an unreviewed safety question as defined in 10 CFR 50.59.

MAJOR ISSUES:

DCP 93-167 replaced an existing Foxboro pressure transmitter with an equivalent Rosemount pressure transmitter. Suction pressure for auxiliary feedwater pump 2-FW-P-3A is monitored by this pressure transmitter 2-FW-PT-203B, which provides indication in the control room. This indication is utilized to alleviate the AFW pump from cavitating when started due to low suction pressure. The replacement is a like-for-like replacement and the indication in the MCR will not be affected. The pressure transmitter is classified as safety related for system pressure boundary. The replacement pressure transmitter is qualified as safety related and will maintain the system pressure boundary. The MCR indication is classified as non-safety.

JUSTIFICATION:

This modification eliminated the problem of the output signal of pressure transmitter (2-FW-PT-203B) from experiencing an exaggerated bow. Although this output signal is not out of calibration, the pressure transmitter could become out of calibration at any time and no repair parts are available. The Rosemount transmitter is a much more reliable piece of equipment.

UNREVIEWED SAFETY QUESTION ASSESSMENT:

1) Accident probability has not been increased because this design change conformed to standards and admins. The operation of the AFW system or MCR indication has not be affected. This pressure transmitter and MCR indication is not required for safe shutdown. This design change has provided a more reliable transmitter output signal to the MCR.

2) Accident consequences have not been increased. The operation of the AFW system has not been affected due to this design change. 2-FW-P-3A suction pressure in the MCR has not been affected. Consequences have not been increased because a failure of the DCP will not corrupt mitigating systems.

3) No unique accident probabilities are created. The implementation of this DCP did not create a possibility for an accident or a malfunction of a different type than any evaluated previously in the UFSAR because the operation of the AFW system has not been affected.

4) Margin of Safety is maintained because the integrity and reliability of the AFW is unchanged.

DCP 93-201
MODIFY CONTAINMENT RING DUCT DAMPERS
NORTH ANNA UNIT 2

DESCRIPTION

The manually operated containment ring duct dampers were difficult to inspect as it required personnel to enter the duct which is a contaminated, high radiation area. Test taps were added to the duct at the dampers to allow inspection of the dampers from outside the duct.

The containment recirculation system is used to maintain containment temperatures within limits as defined by Technical Specifications. This is to ensure that equipment is not exposed to temperatures higher than it is qualified for.

SUMMARY OF SAFETY ANALYSIS (94-SE-MOD-020)

The accidents considered were a LOCA and MSLB.

This design change did not create an unreviewed safety question as defined by 10CFR50.59.

1. Accident probability was not increased because the system has no role in the occurrence of an accident.

2. Accident consequences were not increased. Containment temperatures will still be maintained within limits as defined by Technical Specifications.

3. No unique accident probabilities were created. The test taps are passive components which will not affect the operation or function of the system or the ring duct.

4. Margin of safety was maintained because containment temperatures will still be maintained within limits as required by Technical Specifications.

DCP 93-241-3

EDG EXHAUST BYPASS VALVE REMOVAL
NORTH ANNA UNITS 1 AND 2

DESCRIPTION

Each exhaust system on the emergency diesel generators has two flow paths. The normal flow path exhausts the gases directly into a seismic, missile protected enclosure which is located on the Service Building Roof. The other flow path is through a muffler which is also mounted on the Service Building roof. The muffler is neither missile protected nor seismically mounted.

A chain operated muffler bypass valve, which is normally locked open allows the exhaust to pass directly into the seismic enclosure. The bypass valve is only closed to force exhaust gases through the muffler during test runs of the diesel generator to minimize noise.

This design change replaces the emergency diesel generator exhaust bypass valves with open piping spool pieces. The diesel exhaust system flow alignments will be through the seismic, tornado missile protected structures at all times, effectively bypassing the exhaust mufflers. The ability to use the exhaust mufflers during emergency diesel generator testing will be eliminated. This design change was initiated to prevent operator error or valve malfunction, which cause blockage or restriction of the exhaust flow and possibly affect the electrical output of the emergency diesel generator.

SUMMARY OF SAFETY ANALYSIS (94-SE-MOD-005)

This design change did not create an unreviewed safety question as defined in 10 CFR 50.59.

A. The implementation of this modification did not increase the probability of occurrence or consequences of an accident or malfunction of equipment important to safety and previously evaluated in the Final Safety Analysis Report.

   The emergency diesel generators are the backup power supplies during a loss of offsite power to the station auxiliaries (Station Blackout). A review of the effects of replacement of normally locked open emergency diesel generator exhaust bypass valves with open pipe spool pieces was performed. No specific accidents or malfunctions were previously postulated concerning these

valves since they are passive components (normally locked open) and after proper operator positioning do not affect the operability, reliability, or performance of the emergency diesel generators. The open pipe spool pieces installed by this design change are totally passive components. The installation of open pipe spool pieces did not increase the probability of occurrence or consequences of an accident or malfunction of equipment important to safety and previously evaluated in the Final Safety Analysis Report.

B. The implementation of this modification did not create a possibility for an accident or a malfunction of a different type than any previously evaluated in the Final Safety Analysis Report.

The replacement of normally locked open valves with open pipe spool pieces does not affect the operability, reliability, or performance of the emergency diesel generators. The removed locked open valves and pipe spool pieces are passive components. Potential problems due to operator error (exhaust bypass valve improperly positioned) are eliminated. Therefore, the modifications did not create the possibility of an accident or malfunction of a different type than any previously evaluated in the Final Safety Analysis Report.

C. The implementation of this modification did not reduce the margin of safety as defined in the basis of any Technical Specification.

The piping spool pieces installed by this design change are passive components. The piping changes have no effect on the operability, performance, or reliability of the emergency diesel generators. Therefore, this design change did not reduce the margin of safety, as described in the bases of any Technical Specification.

DCP 93-246
REPLACEMENT OF TEE'S WITH ELBOWS
NORTH ANNA UNIT 2

DESCRIPTION

The pressurizer level transmitters were originally Barton transmitters. These were replaced with Environmentally Qualified Rosemount transmitters per DCP 81S-08 A&B. The Barton transmitters were equipped with calibration taps which were no longer required. The tubing tee's to the calibration ports were capped using Swagelock compression fittings. Swagelock compression fittings are no longer acceptable in containment in order to eliminate possible locations for RCS leakage. The tee's were removed and replaced with welded elbows.

The level transmitters provide inputs to the pressurizer high level trip, level control logics, control room annunciator, control room recorder, backup heater logics, CVCS flow control, CVCS letdown isolation and pressurizer heater cutoff controls. The reactor trip is a backup to the high pressure trip. No credit was taken for the trip in the accident analysis and they were installed only to enhance overall RCS reliability.

SUMMARY OF SAFETY ANALYSIS (93-SE-MOD-075)

All accidents were reviewed and a small break LOCA, FW Pipe Rupture, Steam Generator Tube Rupture and Large Break LOCA were considered to be applicable.

1) Accident probability was not increased as the tubing elbows eliminated the compression caps and reduced the number of sites for possible RCS leakage and decreased the probability for a small break LOCA. The transmitters are to detect accidents and do not contribute to their probability.

2) The consequences of any of these accidents was not affected. Failure of the elbows is still bounded by the small break LOCA analysis in the UFSAR. The trip function of the transmitters was not affected as the tubing operation was not affected. A reactor trip signal shall still be received with the 2/3 criteria if a high level is detected in the pressurizer.

3) No unique accident probabilities were created. The only function of the tee's was as system pressure boundaries. The elbows will serve the same function with less possible leakage sites.

4) Margin of Safety was maintained because the integrity and reliability of the RCS and the transmitters was not affected. All components used for this modification were in accordance with all applicable codes, standards and specifications.

DC 93-260
MOV LIMIT-LIMIT CONTROL CIRCUIT WIRING MODIFICATION
NORTH ANNA / UNIT 2

DESCRIPTION

During resolution of the Liberty Technologies Part 21 notification for Unit 2, Motor Operated Valve (MOV) as-left control switch trip (CST) settings were reviewed against the maximum thrust values. The CST settings for several of MOVs were left above the manufacturer's (Velan) specified continuous valve allowable because the targeted thrust band was too narrow to envelope inertial forces. Interim justification for this condition utilized the fatigue limit of the valve. Subsequent conversations with the manufacturer (Velan) resulted in the determination that increasing the continuous allowable may not be possible when considering stresses induced by seismic events. The corporate recommendation for resolving this issue was to modify the control circuitry to make the MOV limit-limit controlled. This method of control was in concert with STD-GN-0002. The MOV Engineer agreed to pursue this option. Additionally, 2-CH-MOV-2380 and 2-CH-MOV-2381 were modified to limit-limit operation, since the existing thrust bands were too limiting, to prevent continual overthrusting of the MOVs.

This package addressed the Unit 1 Valves identified as requiring the limit-limit modification. All valves were 3" Velan valves in Groups 36 and 37 of the NAPS grouping methodology, with the exception of 2-CH-MOV-2380 and 2-CH-MOV-2381. 2-CH-MOV-2380 and 2-CH-MOV-2381 were required to be limit-limit due to thrust band limitations.

SUMMARY OF SAFETY ANALYSIS

An unreviewed safety question did not exist because:

A. The implementation of this DCP did not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety and previously evaluated in the UFSAR because the torque switch was still wired into the circuit to provide a back-up means of de-energizing the motor if catastrophic failure of the limit switch gearing or rotors occurs or if the MOV encounters a line obstruction prior to reaching the preset position.

B. The implementation of this DCP did not create a possibility for an accident or a malfunction of a different type than any evaluated previously in the UFSAR because this modification did not replace the limit switch but modified the wiring. This did not have any impact on increasing the probability of a switch failure. In addition, following this modification, diagnostic testing was performed to ensure the wiring change was correct and the limit switch operated correctly. A leak test was performed to ensure work performed on the valves was adequate to seat the Type C MOVs at the lower thrust values. The non Type C MOVs utilized diagnostic testing to verify seating.

C. The implementation of this DCP did not reduce the margin of safety as defined in the basis of any Technical Specification because these valves still perform the same function to mitigate a LOCA, MSLB, or SGTR and did not change the system operation, flows, or safety related requirements. Therefore, an unreviewed safety question did not exist. In fact, the rewiring change improved the long-term reliability of the valve component portion of the MOVs since the change does better at controlling the seating thrust.

DCP 93-262
"INSTALLATION OF GAI-TRONICS IN THE CLEAN WASTE SEGREGATION FACILITY"

SUMMARY:

Announcements and emergencies broadcast over the Gai-tronics could not be heard from within the Clean Waste Segregation Facility(CWSF). A Gai-tronics wall unit, speaker and two desk-top subsets were installed inside the CWSF. This equipment is non-safety related however it is powered from a vital bus.

UNREVIEWED SAFETY QUESTION ASSESSMENT

1) Accident probability was not increased because the additional load was evaluated and it was determined that the EDG has the capacity to carry the additional load. The new equipment is isolated from safety related loads via an existing breaker which currently supplies other Gai-tronics equipment.

2) Accident consequences were not increased. Implementation of this DCP will not affect the operation of the Gai-tronics system or any other system.

3) No unique accident probabilities were created. The implementation of this DCP does not create a possibility for an accident or a malfunction of a different type than any evaluated previously in the UFSAR because the design change will not effect the operation of the Gai-tronics or any safety-related system.

4) Margin of Safety is maintained because the integrity and reliability of the vital bus is unchanged.

DC 93-267
PROVIDE TORQUE SWITCH BYPASS
NORTH ANNA / UNIT 2

DESCRIPTION

Virtually all rising stem MOVs at North Anna were provided with open direction torque switch (TS) bypass as part of the response to IEB 85-03. A small portion (notably 2-HV-MOV-213C and the SR quarter turn MOVs) were excluded at that time. When GL 89-10 was being dealt with, it was decided to go ahead and provide open torque switch bypass for all remaining valves so that this application would be uniform for all SR MOVs. Thus, 2-HV-MOV-213C was modified to provide open TS bypass.

SUMMARY OF SAFETY ANALYSIS

An unreviewed safety question did not exist because:

A. The implementation of this DCP did not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety and previously evaluated in the UFSAR because the components ability to function as designed was not modified.

B. The implementation of this DCP did not create a possibility for an accident or a malfunction of a different type than any evaluated previously in the UFSAR because this modification did not all credible failure modes are bounded by the single failure criterion. The specific component involved was an installed spare in conjunction with two 100% capacity trains of control room chillers.

C. The implementation of this DCP did not reduce the margin of safety as defined in the basis of any Technical Specification because a properly functioning, reliable chiller MOV did not alter the design attributes assumed in the Bases. The overall HV system was not be changed.

DCP SUMMARY FOR DCP #93-272

MAJOR ISSUES:

In an effort to reduce the number of locked, high radiation areas, Health Physics has installed lead sheet shielding in front of the reactor coolant filter vessels in the NAPS Unit 1 and 2 Seal Water Injection Filter Cubicles. These cubicles were previously posted as locked, high radiation areas. Local shielding has significantly reduced dose rates, in the normally accessible areas of the cubicles, below 1R, at a distance of one foot from the source. At this level, the cubicles may be posted as normal radiation areas with unrestricted access. By reducing the number of locked, high radiation areas, the potential for unlocked gate violations is decreased, while area access and dose rates are favorably improved for workers in these cubicles. Design Engineering was tasked to design seismic supports to hold the lead sheet shielding in place.

The major design issues associated with eliminating the locked, high radiation requirements for these areas (i.e. NAPS Units 1 and 2 Seal Water Injection Filter Cubicles) are personnel safety, ALARA, and seismic concerns for nearby Safety-Related equipment. Personnel safety and ALARA are linked design issues in this Design Change. Reducing radiation exposures, while improving access to work areas, are integrally connected elements of personnel safety and ALARA goals. The supporting frames, together with the lead sheet shielding, form a radiation and a personnel barrier. Dose rates in the unrestricted areas of the cubicles are sufficiently, reduced by the thickness of the lead sheet shielding, and access to areas of the cubicle where dose rates may exceed 1R, at a distance of one foot from the source, are blocked by the balance of the supporting frame structure. Personnel would have to leave the floor or reach into the overhead to get near the vessels, in violation of the Radiation Work Permit for a normal radiation area with unrestricted access. By definition, this modification permits the cubicles to be posted as a normal radiation area with unrestricted access, eliminating the need for a locked gate. Personnel safety and ALARA concerns were also considered during the installation phase, by restricting installation work, within the cubicles, to refueling outage periods when the filters have been changed out and dose rates are at an acceptable level.

Seismic concerns for the supporting frames ensured that nearby Safety-Related equipment, within the collapse envelope of these frames, were not impacted during a design basis seismic event. To ensure that seismic concerns were adequately addressed, the frames were seismically designed, not only to survive the stresses induced during an OBE or DBE loading, but to maintain a safe clear distance, from neighboring equipment. This will prevent "pounding" in case of a seismic event.

UNREVIEWED SAFETY QUESTION ASSESSMENT:

1. The probability of seismic activity, as well as the probability for unusually high filter vessel dose rates, are unrelated to the installation of these lead sheet shielding structures. Consequently, there is no reason to expect a design basis seismic event or an unusually high filter vessel dose rates to occur as a result of this modification.
2. Since the supporting frames for the lead sheet shielding are seismically designed, and have been checked for "pounding" against nearby Safety-Related equipment during a seismic event, there is no identified increase in the consequences related to a seismic event or higher than normal filter vessel dose rates, to be associated with this modification.
3. Plant safety/ALARA and seismic concerns were the main issues which were addressed independently, via this Design Change. There is no adverse affect, singular or cumulative, associated with this modification upon any existing system, structure or component. As such, no accidents of a different kind are expected.

4. Margins of safety are not reduced since no system, structure or component is affected by this modification or is required to interface with this modification.

DCP SUMMARY FOR DCP #93-273

MAJOR ISSUES:

In an effort to reduce the number of locked, high radiation areas, Health Physics has installed lead sheet shielding in front of the reactor coolant filter vessels in the NAPS Unit 1 and 2 Seal Water Injection Filter Cubicles. These cubicles were previously posted as locked, high radiation areas. Local shielding has significantly reduced dose rates, in the normally accessible areas of the cubicles, below 1R, at a distance of one foot from the source. At this level, the cubicles may be posted as normal radiation areas with unrestricted access. By reducing the number of locked, high radiation areas, the potential for unlocked gate violations is decreased, while area access and dose rates are favorably improved for workers in these cubicles. Design Engineering was tasked to design seismic supports to hold the lead sheet shielding in place.

The major design issues associated with eliminating the locked, high radiation requirements for these areas (i.e. NAPS Units 1 and 2 Seal Water Injection Filter Cubicles) are personnel safety, ALARA, and seismic concerns for nearby Safety-Related equipment. Personnel safety and ALARA are linked design issues in this Design Change. Reducing radiation exposures, while improving access to work areas, are integrally connected elements of personnel safety and ALARA goals. The supporting frames, together with the lead sheet shielding, form a radiation and a personnel barrier. Dose rates in the unrestricted areas of the cubicles are sufficiently, reduced by the thickness of the lead sheet shielding, and access to areas of the cubicle where dose rates may exceed 1R, at a distance of one foot from the source, are blocked by the balance of the supporting frame structure. Personnel would have to leave the floor or reach into the overhead to get near the vessels, in violation of the Radiation Work Permit for a normal radiation area with unrestricted access. By definition, this modification permits the cubicles to be posted as a normal radiation area with unrestricted access, eliminating the need for a locked gate. Personnel safety and ALARA concerns were also considered during the installation phase, by restricting installation work, within the cubicles, to refueling outage periods when the filters have been changed out and dose rates are at an acceptable level.

Seismic concerns for the supporting frames ensured that nearby Safety-Related equipment, within the collapse envelope of these frames, were not impacted during a design basis seismic event. To ensure that seismic concerns were adequately addressed, the frames were seismically designed, not only to survive the stresses induced during an OBE or DBE loading, but to maintain a safe clear distance, from neighboring equipment. This will prevent "pounding" in case of a seismic event.

UNREVIEWED SAFETY QUESTION ASSESSMENT:

1. The probability of seismic activity, as well as the probability for unusually high filter vessel dose rates, are unrelated to the installation of these lead sheet shielding structures. Consequently, there is no reason to expect a design basis seismic event or an unusually high filter vessel dose rates to occur as a result of this modification.
2. Since the supporting frames for the lead sheet shielding are seismically designed, and have been checked for "pounding" against nearby Safety-Related equipment during a seismic event, there is no identified increase in the consequences related to a seismic event or higher than normal filter vessel dose rates, to be associated with this modification.
3. Plant safety/ALARA and seismic concerns were the main issues which were addressed independently, via this Design Change. There is no adverse affect, singular or cumulative, associated with this modification upon any existing system, structure or component. As such, no accidents of a different kind are expected.

4. Margins of safety are not reduced since no system, structure or component is affected by this modification or is required to interface with this modification.

DCP 94-002
Steam Generator Blowdown System Tie-Ins
North Anna Unit 2

Description

Nuclear Steam Generator corrosion problems throughout the industry have caused significant loss of plant availability, increased total personnel radiation exposure and eventually resulted in replacement of the damaged generators at great capital expense. (As was recently done for North Anna Unit 1 and is being planned for Unit 2.) One of the key methods of protecting steam generators (S/G's) from the various forms of corrosive attach is the use of continuous blowdown (BD) systems. This approach is highly recommended by both EPRI and Westinghouse.

The current NAPS S/G BD system is both expensive from an operating standpoint and marginal from a design standpoint. It does not satisfy current requirements for capacity to minimize corrosion product buildup in the S/G's, or for efficiency (no heat recovery). The existing S/G BD system provides a maximum continuous operating blowdown rate of 20-25 gpm per S/G with the system at capacities exceeding design has resulted in continual maintenance problems with substantial corresponding costs.

An operating S/G BD rate of 0.2% to 1.0% of main steam (MS) flow is recommended by EPRI. This corresponds to a BD rate (at a fluid temperature of 60°F) of approximately 17 gpm to 85 gpm per S/G, based on the current NAPS MS flow of approximately 12.75 million lb/hr. The new NAPS Unit 1 S/G's have a 1% of MS flow BD design recommendation. Clearly, the existing S/G BD system provides a marginal BD rate at best when operating at flow rates or more than twice design capacity. However, a back fit system allowing a continuous BD of 85 gpm per S/G is cost prohibitive.

Therefore, the design objectives for S/G BD system improvements are:

1) To provide an increased blowdown rate of as close as possible to 1% of the main steam flow during startup and upset conditions, which will assist in minimizing corrosion of the S/G's.

2) To provide a cost and energy efficient continuous BD system design with heat recovery function that has a continuous operating flow rate of at least 45 gpm per S/G.

Because of the complexity and time required for installation of S/G BD system improvements with the design attributes described above, it is desirable to complete as much of the installation work during non-outage periods as possible.

The abandoned high capacity S/G BD system can be made operational with the secondary side of the Unit in its current operational configuration by reworking or replacing, as required, the existing control instrumentation and by rerouting the drains from the BD flash tank to the circulating water discharge as waste, instead of the original design of routing the drains to the condenser hotwell. The original design concept of returning the untreated drains from the high capacity system BD flash tank to the condenser would result in zero net blowdown from the secondary side. The drains from the BD flash tank will be cooled via a shell and tube heat exchanger (the BD flash tank drains cooler) with condensate as the cooling medium. Thus, energy from the flash tank drains will be recovered to the steam cycle, while cooling the BD discharge to approximately the same temperature as the circulating water to which it is discharging. These upgrades to the high capacity BD system will permit the system to operate at its design capacity of 100,000 lb/hr (approximately 200 gpm at 60°F), or 67 gpm per S/G, if sufficient makeup water capacity is available. Present makeup water capacity allows a continuous S/G BD rate of approximately 45 gpm per S/G for both Units.

In order to be able to complete the majority of the proposed high capacity S/G BD system improvements during non-outage periods, installation of tie-ins (with isolation valves and welded caps as needed) to existing piping were completed during the refueling outage of the Unit. These tie-ins were installed in locations for future connections of the upgraded high capacity S/G BD system piping to the existing interfacing piping systems, which will permit installation of all new S/G BD equipment (BD flash tank drains cooler and radiation monitor) and piping, including the removal of the tie-in stub weld caps, and the process tie-in, to be performed during non-outage times. This Design Change dealt only with the installation of the tie-ins to existing piping systems for future connections of the upgraded high capacity S/G BD system. The installation of the upgrades to the high capacity S/G BD system and the corresponding system process tie-ins will be dealt with by other Design Change Packages.

Required system tie-ins designed and installed by this DCP include: tie-in for condensate supply to the BD flash tank drains coolers (from the 24 in. condensate pump discharge line, 24"-WCPD-4-301); tie-in for condensate return from the BD flash tank drains coolers (to the 24 in. condensate line, 24"-WCPD-14-301, between the gland steam condenser outlet and the condensate polishing inlet connection); tie-in for the divert line from the BD flash tank drains cooler condensate return line to condenser 2-CN-SC-1B spare penetration number 57; and tie-in for cooled blowdown discharge line (via abandoned 6"-WED-411-151 to 20"-WMD-402-121 to the circulating water discharge tunnel).

Summary of Safety Analysis

This Design Change did not constitute an unreviewed safety question as defined in 10CFR50.59 since:

1. This modification did not affect or impact any safety related equipment or systems. Therefore, this Design Change did not increase the probability of occurrence or the

consequences of an accident or malfunction of equipment important to safety previously evaluated in the UFSAR.

2. This Design Change is consistent with the affected systems' design bases and existing design basis criteria. The tie-ins installed by this modification did not change the operation or performance of the affected systems, nor did they alter or create any process flow paths. Therefore, this modification did not create a possibility for an accident or malfunction of a different type than any previously evaluated in the UFSAR.

3. This Design Change did not impact or change the basis of any of the Technical Specifications, and, therefore, the margin of safety as defined in the bases of the Technical Specifications remained unchanged.

Which accidents previously evaluated in the Safety Analysis Report were considered?

Loss of Normal Feedwater

A. Could the activity increase the probability of occurrence for the accidents identified above? State the basis for this conclusion.

No, the addition of tie-ins to the condensate system installed by this Design Change are in accordance with the original design code and did not result in an increased probability of occurrence for the Loss of Normal Feedwater Accident. None of the other accidents previously evaluated by the SAR were impacted by this modification.

B. Could the activity increase the consequences of the accidents identified above? State the basis for this conclusion.

No, the addition of tie-ins to the condensate system installed by this Design Change are in accordance with the original design code. Blowdown tie-ins did not affect equipment required for accident mitigation.

C. Could the activity create the possibility for an accident of a different type than was previously evaluated in the Safety Analysis Report? State the basis for this conclusion.

No, the addition of tie-ins to the condensate system installed by this Design Change are in accordance with the original design code and no operational or process flow changes were made as a result of this Design Change.

What malfunctions of equipment related to safety, previously evaluated in the Safety Analysis Report, were considered?

None, this Design Change did not involve or impact any safety related equipment or systems.

A. Could the activity increase the probability of occurrence of malfunctions identified above? Sate the basis for this conclusion.

No, this Design Change did not involve or impact any safety related equipment or systems.

B. Could the activity increase the consequences of the malfunctions identified above? State the basis for this conclusion.

No, this Design Change did not involve or impact any safety related equipment or systems.

C. Could the activity create the possibility for a malfunction of equipment of a different type than was previously evaluated in the Safety Analysis Report? State the basis for this conclusion.

No, this Design Change did not involve or impact any safety related equipment or systems, and no operational or process flow changes were made as a result of this Design Change.

Has the margin of safety of any part of the Technical Specifications as described in the bases section been reduced? Explain.

No, this Design Change had no impact on the Technical Specifications.

Does the proposed change, test, or experiment require a change to the Technical Specifications? Explain.

No, this Design Change had no impact on the Technical Specifications.

DCP 94-008
HHSI Flow Instrumentation Upgrades
North Anna Unit 2

Description

North Anna Technical Specification 4.5.2.h gives requirements for High Head Safety Injection (HHSI) branch line flow balance testing. The Technical Specification requires flow balance testing during plant shutdown to be accomplished following modifications to Emergency Core Cooling (ECCS) subsystems which altered subsystem flow characteristics.

The Technical Specifications require that the total HHSI flow rate remain less than or equal to 660 gpm when flowing one pump to an atmospheric RCS. The basis of this total flow limitation is to prevent pump run out (approximately 675 gpm). The flow balance testing is performed with HHSI pumps taking suction from the Refueling Water Storage Tank (RWST). This total flow is the sum of three HHSI branch line flows plus simulated Reactor Coolant Pump (RCP) seal injection flow.

Technical Specification 4.5.2.h also requires that the sum of the injection line flow rates, excluding the highest flow rate, is greater than or equal to 359 gpm. This 359 gpm minimum value is based on ensuring that the ECCS criteria for peak clad temperature (2200°F) is met.

Pre-operational HHSI flow balance testing was performed for Unit 1 and Unit 2 on 5/5/77 and 3/14/80, respectively using the installed flow instrumentation. This instrumentation consist of two pressure taps off of the HHSI branch lines which are separated by only two feet. This configuration relies only on the frictional losses between the two taps which results in a head loss of approximately 4 feet at a flow rate of 200 gpm. Accuracy for this type of instrumentation is generally assumed to be on the order of $\pm$ 10%.

Special test 1-ST-41 was performed on 3/27/81 in response to Westinghouse technical bulletins NSD-TB-80-11 and NSD-TB-81-10. Both of these bulletins discussed potential problems associated with the delivery of flow and the HHSI branch line flow balancing related to pump degradation/replacement and instrument inaccuracies. The HHSI throttle valves were set by the number of turns open and the use of mechanical spacers to establish accurate and repeatable throttle valve stem heights.

A Westinghouse Letter, Serial #VRA-89-757, discussed several topics related to ECCS flow inconsistencies. These inconsistencies were 1) actual plant system resistances and pump curve assumptions may be inconsistent with Westinghouse assumptions, 2) RCP seal injection flow may be greater than the value used in the Westinghouse analysis, 3) pump degradation has resulted in some plants opening up throttle valves which could result in actual delivered flow to

the RCS during small and intermediate break LOCAs being less than that assumed in the safety analysis, and 4) the Westinghouse analysis assumed a maximum flow imbalance between branch lines of 10 gpm, but this may not be the case at some plants.

As a result of the Westinghouse letter, Station Deviation DR 89-2250 was written which resulted in an action plan to perform testing per NA&F report titled "Final Report for Evaluation of Flow Inconsistencies for Surry Units 1&2 and North Anna Units 1&2" dated 2/14/89. In an effort to obtain better flow measurement accuracy, it was decided to pursue the use of clamp on ultrasonic Controlotron flowmeters. HHSI flow balance testing was subsequently performed each refueling outage for both units using the Controlotron flowmeters. These flowmeters were calibrated by Controlotron to have an error of less than 1%.

In October 1990, during the performance of Unit 2 HHSI flow balance testing it was discovered that the as-found cold leg branch line flows were not sufficient to meet Technical Specification requirements. At this time, Technical Specification 4.5.2.h required the sum of the branch flows, excluding the highest branch flow, to be greater than or equal to 384 gpm. Using the single most limiting pump, the sum of the two lowest branch line flows was only 347 gpm. The cause of the event was attributed to the previous positioning of the HHSI branch line throttle valves using the permanently installed in line flow instruments.

On April 13, 1992, while reviewing the Unit 2 HHSI flow balance test results, it was determined that the as-found cold leg branch line flows were insufficient to meet Technical Specification requirement 4.5.2.h. Using the most limiting HHSI pump, the sum of the two lowest flows was only 347 gpm. The cause of the event was attributed to repositioning of the throttle valves using previously taken stem height measurements after the installation of stem locking devices.

On March 20, 1993, it was discovered during the performance of Unit 1 HHSI flow balance testing that the as-found cold leg line flows were insufficient to meet Technical Specification requirements. The cause of this event was attributed to the Technical Specification change (#259) which was issued for both units on August 4, 1993. The change reduced the minimum flow requirement for the two lowest flow branch lines from 384 to 359 gpm with no measurement uncertainty. Therefore, flow balance acceptance criteria values the TS change also increased the maximum total pump flow from 650 to 660 gpm. Also, a surveillance requirement was added to define a value greater than or equal to 43.8 gpm to be used for simulated RCP seal injection flow during the flow balance testing.

On October 14, 1993, during the performance of Unit 2 flow balance testing, the sum of the two lowest branch line flows, excluding the highest flow, was equal to 356 gpm. The cold leg throttle valves were adjusted so that the sum of the two lowest flow rates was equal to 347 gpm. Review of previous flow balance and valve stem height measurements revealed that very small changes in stem height results in relatively large changes in flow rate. The Unit 2 HHSI branch line throttle valves were fixed in position using Locktite brand "Threadlocker 290". As a result of the Technical Specification violation, an emergency Technical Specification change (#259A) was submitted to the NRC to eliminate the simulated seal injection flow rate.

Because of the 10/14/93, Unit 2 Technical Specification violation event, a Root Cause Evaluation was initiated, RCE 93-05 "HHSI Flow Balance". It was determine that the uncertainty in measurement using Controlotrons is on the order of 5.5%.

As a result of the uncertainty associated with using the existing installed flow instrumentation configuration (± 10%) or the use of Controlotron flow instrumentation (± 5.5%), the Root Cause Evaluation recommended that a local means of flow measurement should be installed which would provide an accuracy of ± 0.5% (error only for differential producer).

In order to provide for accurate HHSI branch line flow indication, this DCP installed a flow venturi assembly with an integral upstream flow conditioner in each of the three HHSI branch lines to the RCS cold legs. Each flow venturi assembly was laboratory calibrated to verify a flow accuracy of ± 0.5% or better. One flow venturi was flow tested to verify the ± 0.5% accuracy in a piping arrangement which simulates the worst case "as-built" HHSI branch line piping arrangement. The flow venturi assemblies were installed upstream of the existing throttle valves, 2-SI-89, 97 and 103. These valves were replaced at part of this DCP. The valves were previously locked in place with the use of "lock tight" and has lost the ability to be repositioned. A pressure reducing orifice assembly were installed downstream of each HHSI branch line throttle valve to provide a portion of the pressure drop which was required to be taken solely by the throttle valve. This permitted the throttle valves to be positioned more open with less pressure drop. Therefore, better throttle control was provided over the expected flow balancing range of 170 gpm to 210 gpm.

This DCP also replaced the throttle valves on the hot leg HHSI lines, 2-SI-111, 116, 123. These valves had also lost their ability to be repositioned due to previous maintenance efforts.

Summary of Safety Analysis

This design change did not create an unreviewed safety question as defined in 10CFR50.59.

Flow venturis were selected as the differential producer because of an expected flow measurement accuracy of ± 0.5%. The procurement specification for the venturi assemblies required a flow accuracy requirement of better than or equal to ± 0.5%. As part of the venturi assembly, an integral flow conditioner was utilized to provide a satisfactory flow profile for the pressure drop measurement. In order to ensure the flow accuracy requirements for the venturis was met, Alden Research Laboratory, Inc. calibrated all three of the Unit #1 flow venturis over the expected flow balance range of 170 gpm to 210 gpm (minimum). In addition, one flow venturi was flow tested over the expected flow range to verify the ± 0.5% accuracy requirement in a piping arrangement which simulates the worst case "as-built" HHSI branch line piping configuration at North Anna.

ASME standard MFC-3M-1989 was used in designing the venturi assemblies with integral flow conditioners. The venturis were designed to provide a differential pressure of approximately 84.5 psi at a flow of 200 gpm. Flow passage through the flow conditioner and venturi bore

were designed to pass post LOCA debris with a frontal dimension of 0.12" x 0.12" which could pass through the containment sump fine mesh screens.

The flow venturi assembly consisted of a venturi body section with an integral flow conditioner and two instrument root valves as shown on vendor drawing N-94007-1-V-800 and N-94007-1-V-801. The venturi body and valves were made out of stainless steel SA 479 316. The flow conditioner was made out of stainless steel tubing SA 213 TP 316. The design weight of the venturi assembly was kept to a minimum (20 lbs.) to eliminate the need of any additional piping supports.

The new trottle valves on both the hot and cold legs were designed to ASME Section III, Class 1 requirements. The Y pattern Edwards plug valves were selected on the basis of control of flow (Cv) in the desired flow range and the requirement for passage of post LOCA debris. All valves materials are consistent with RCS requirements.

Which accidents previously evaluated in the Safety Analysis Report were considered?

Loss of Coolant Accident

A. Could the activity increase the probability of occurrence for the accidents identified above? State the basis for this conclusion.

   No, the venturis, pressure reducing orifice assemblies, and valves are isolated from the Reactor Coolant system via two downstream check valves. The venturis and pressure reducing orifice assemblies meet or exceed the design requirements for the SI system.

B. Could the activity increase the consequences of the accidents identified above? State the basis for this conclusion.

   No, the venturis, pressure reducing orifice assemblies, and valves were designed to meet or exceed the requirements of the SI system. In addition, the flow passage through the venturis and pressure reducing orifices were sized so as not to create an obstruction due to post LOCA debris which passes through the containment sump screens. The SI system performance was not impacted as a result of this modification. Thus, ECCS flow delivered during a postulated LOCA remained unchanged by the modification.

C. Could the activity create the possibility for an accident of a different type than was previously evaluated in the Safety Analysis Report? State the basis for this conclusion.

   No, the venturis, pressure reducing orifice assemblies, and valves were assigned to meet or exceed the requirements of the SI system.

Malfunctions of equipment related to safety, previously evaluated in the Safety Analysis Report, were considered?

HHSI branch line rupture

A. Could the activity increase the probability of occurrence of malfunctions identified above? State the basis for this conclusion.

No, the venturis, pressure reducing orifice assemblies, and valves were designed to meet or exceed the requirements of the SI system.

B. Could the activity increase the consequences of the malfunctions identified above? State the basis for this conclusion.

No. the venturis, pressure reducing orifice assemblies, and valves were designed to meet or exceed the requirements of the SI system. In addition, the flow passage through the venturis and pressure reducing orifices were sized so as not to create an obstruction due to post LOCA debris which passes through the containment sump screens. The SI system performance was not impacted as a result of this modification.

C. Could the activity create the possibility for a malfunction of equipment of a different type than was previously evaluated in the Safety Analysis Report? State the basis for this conclusion.

No, the venturis, pressure reducing orifice assemblies, new valves were designed to meet or exceed the requirements of the SI system. In addition, the flow passage through the venturis and pressure reducing orifices was sized so as not to create an obstruction due to post LOCA debris which passes through the containment sump screens. The SI system performance was not impacted as a result of this modification.

Has the margin of safety of any part of the Technical Specifications as described in the bases section been reduced? Explain.

No, the operation, function and performance of the SI system was not impacted as a result of this modification. No technical specification margin of safety was reduced.

Does the proposed change test, or experiment require a change to the Technical Specifications? Explain.

No, the operation, function and performance of the SI system was not altered as a result of this modification. ECCS flows were not degraded. No technical specification margin of safety was reduced.

DC 94-012
Modification to Pipe Support 2-RC-HSS-144 and Valves 2-RC-SOV-2455C-1 and 2
North Anna Unit 2

Description

During the walkdowns of Safe Shutdown Equipment for Unresolved Safety Issue (USI) A-46 and IPEEE(Seismic) programs, the following seismic interaction concerns were noted for Power Operated Relief Valve (PORV) 2-RC-PCV-2455C and associated SOVs 2-RC-SOV-2455C-1 and 2:

A) The diaphragm housing of valve 2-RC-PCV-2455C was close to a pipe support structural steel member. Failure of valves due to impact with stiff adjacent structures during a seismic event has occurred in previous earthquakes at other facilities.

B) The diaphragm housing of valve 2-RC-PCV-2455C was also close to a condulet box connected to the solenoid casing for 2-RC-SOV-2455C-1. The concern was that the possible impact between the PORV and the condulet box during a seismic event may be sufficient to cause the SOV to malfunction.

To alleviate these potential concerns the adjacent pipe support was modified and the solenoid casing and attached condulet box for 2-RC-SOV-2455C-1 were rotated away from the PORV diaphragm housing to ensure there was sufficient gap between the PORV diaphragm housing and these items. It should be noted that in order to rotate the solenoid casing for 2-RC-SOV-2455C-1 it was required to rotate the solenoid casing and attached condulet box for 2-RC-SOV-2455C-2. Valves 2-RC-SOV-2455C-1 and 2 were tested after rotating to ensure their functionality.

Summary Of Safety Analysis

This design change did not create an unreviewed safety question as defined by 10CFR50.59 since the function of the modified pipe support and rotated SOVs were not altered. Therefore, this design change did not increase the probability of a previously evaluated accident, increase the consequences of a previously evaluated accident, or introduce the possibility of a new accident not previously evaluated in the Safety Analysis Report. Implementing these changes eliminated interaction concerns that potentially could have impacted the ability of the PORV to function properly during a seismic event.

DCP 94-014
LETDOWN RADIATION MONITOR REPLACEMENT
2-CH-RM-228 AND 2-CH-RM-229
NORTH ANNA UNIT 2

DESCRIPTION

The high and low range radiation monitors in the Reactor Coolant Letdown Sample Piping, 2-CH-RM-228 and 2-CH-RM-229, respectively, were designed to provide Local and Control Room indications of gross reactor coolant activity and fuel element failure. Due to the piping configuration and flow velocity, these monitors acted as crud traps, which resulted in high background radiation (at least 100 Mr/Hr overall, with several hot spots at 5-7 R/hr). This rendered the low range monitor ineffective and created higher dose areas in the Auxiliary Building. Temporary shielding consisted of 6' high lead blankets installed to reduce the general area field in the vicinity of the monitors to about 4 Mr/Hr (outside the shielding).

The existing high range and low range monitors were physically removed. This was accomplished by cutting the 1/2" sample piping to and from the monitors on the 291' elevation of the Auxiliary Building, and welding on pipe caps. In order to maintain the capability to detect fuel failure, one non-invasive monitor was installed on the letdown line, downstream of the Non-Regenerative HX in the Non-Regen HX Room. The new detector was set to alarm at coolant radioactivity levels that correspond to 0.1% fuel failure (Hi Alarm) and 1.0% fuel-failure (Hi-Hi Alarm).

SUMMARY OF SAFETY ANALYSIS

All accidents were reviewed and the following were found to be applicable: Loss of Pressure Boundary

This design change does not create an unreviewed safety question as defined by 10CFR50.59.

1) The accident identified above is caused by failure of the pipe or piping components. This Design Change will cut and cap 1/2" sample lines which are connected to the letdown piping downstream of the first RCS isolation valve. Approved station procedures will be used to perform this work, therefore, the activity will not increase the probability of a loss of pressure boundary.

2) Accident consequences will not be affected. The purpose of the new letdown radiation monitoring system is to provide plant personnel with early indication of fuel failure. The new monitors will not be connected to the letdown or RCS pressure boundary.

SUMMARY OF SAFETY ANALYSIS (Con't)

3) No unique accident possibilities will be created. The new detectors are external to the piping and seismically anchored in place. The new detectors will have no effect on the CVCS pressure boundary or any associated control/protection functions.

4) The Margin of Safety was maintained. Safe operation of the plant will not be adversely affected since the new monitors will be functionally equivalent to the existing high range monitors, and the existing low range monitors are not required for accident mitigation or compliance with Technical Specifications. All existing alarm, indication, and recording functions will continue to be provided by the new monitoring system. The new detectors and ratemeters will be seismically installed to prevent damage to nearby SR equipment.

DCF 94-106
INSTALL HARD PIPE FROM SI VENTS TO FLOOR DRAINS
NORTH ANNA UNIT #1

DESCRIPTION

The LHSI discharge vents are located in the overhead of the QSPH basement. The area is very congested and there was difficulty in reaching the valves. The system was being vented by using tygon tubing which was routed to a floor drain. To facilitate venting, tubing was permanently routed to the floor drains with double isolation valves located in an accessible area. The original vent valves are maintained open with the system pressure boundary relocated to the first isolation valve. The vents were provided with double isolation valves to ensure that there is no leakage from the system.

LHSI pump runs have shown that sudden transient movement of the pump discharge piping occurs upon pump start. Tubing, rather than piping, was routed from the original vents to the new isolation valves as the tubing allows greater flexibility. The tubing was installed to accommodate movement of the discharge lines upon pump start.

Leakage from the vent assemblies is not allowable. The QSPH ventilation system exhausts directly to atmosphere. No filters or radiation monitors are installed in the exhaust prior release to atmosphere. Any leakage from these vent assemblies creates the potential for unmonitored radiological release during an accident with the ECCS in the recirculation mode. Leakage from the vent valves is not expected as the tubing and valves meet or exceed the system design requirements and are seismically qualified. The lines downstream of the vent valves have also been capped.

SUMMARY OF SAFETY ANALYSIS (94-SE-MOD-52)

The accidents which are applicable to this change are those which result in an SI actuation. The applicable malfunctions are ECCS piping failures, particularly a LHSI pump discharge line.

This design change did not create an unreviewed safety question as defined by 10CFR50.59.

1) Accident and malfunction probabilities were not increased as function and operation of the safety injection system was not affected. All components were purchased and installed in accordance with Q2 and LHSI system design requirements.

2) Accident consequences were not affected. Double isolation valves and a pipe cap were installed on each vent line in the QSPH to ensure that any radioactive material that may leak by the valve does not exit the SI piping into the QSPH.

3) No unique accident possibilities were created. The tubing and valves installed meet or exceed the requirements of the SI system. The tubing is seismically supported. The double isolation valves and pipe caps ensure that there be no unfiltered, unmonitored radioactive release.

4) Margin of Safety was maintained because the function and operation of the SI system was not altered by this modification.

DCP 94-110
Anti-Rotation Device for Accumulator Test Line Trip Valve
NAPS Unit 1

Description

Trip valve 1-SI-TV-1859 and 1-CH-FCV-1114A are Masoneilan air operated valves that are used as a containment isolation valve for the accumulator test line and the primary grade water inlet valve to the boric acid blender, respectively. The valve position indication for these valves has not been operating properly. This is because the shaft in these air operated Masomeilan valves has a tendency to rotate as it strokes. When the shaft rotates, the actuating arm may not line up with the valve position limit switches.

An anti-rotation device has been designed for these valves. This device will prevent shaft rotation by acting as a guide for the limit switch actuating arm. It will be fabricated from stainless steel sheet metal and will be attached to the valves by the bolts that hold the lower limit switch mounting bracket.

Summary of Safety Evaluation (95-SE-MOD-29)

The anti-rotation device is a simple passive component. It will be mounted on the outside of the valve and will not affect the system pressure boundary. Valve position indication will operate in the same manner, but be more reliable. This modification will not diminish the ability of 1-SI-TV-1859 and 1-CH-FCV-1114A to perform their safety functions. Seismic qualification of the valve and piping is not affected.

DCP 94-127
Install Hard Pipe From SI Vents to Floor Drains
NAPS Unit 2

Description

The LHSI pump discharge piping vents 2-SI-302 and 2-SI-305 and the RWST to charging pump suction header vent valve 2-SI-307 are located in the overhead of the QSPH basement. The area is very congested and difficulty exists in reaching the valves. To facilitate venting of the LHSI system, tubing has been permanently installed downstream of the vents and routed to new isolation valves installed in a more accessible location. The original vent will be maintained open with the second and third valves kept closed. Each vent line has been routed from the vents to a floor drain. Those vents installed off of the 8" LHSI discharge piping have been provided with double isolation. The vent off of the 10" SI line from the RWST to charging pump suction header only required single isolation as it is not exposed to containment sump water.

Summary of Safety Evaluation (94-SE-MOD-076)

The LHSI pump discharge piping vents 2-SI-302 and 2-SI-305 are located in the overhead of the QSPH basement. The area is very congested and difficulty exists in reaching the valves. To facilitate venting of the LHSI system, tubing has been permanently installed downstream of the vents. Double isolation valves have been installed in a more accessible location. The original vent will be maintained open with the second and third valve acting as the system pressure boundary. Each vent line has been capped to provide further assurance that no leakage will exit the SI system. A detached, non safety related drain line has been routed from the vents to a floor drain. The drain line has been supported in accordance with NAS-1009 such that it will not damage any safety related equipment during a seismic event. A similar arrangement has also been installed downstream of 2-SI-307, the RWST to charging pump suction header vent valve, however only a single isolation valve was required as this line has no interface with containment sump water.

Unreviewed Safety Question Evaluation

1. The probability of occurrence of an accident or malfunction of equipment previously evaluated in the

UFSAR does not increase since the function and operation of the Safety Injection System did not change.

2. All components were purchased and installed in accordance with Q2 (Q3 on 10" SI-408-153A-Q3 line) and LHSI system design requirements. A pipe cap has been installed on each vent ensuring any radioactive material that may leak by the valve does not exit the SI piping into the QSPH. Therefore, there is no increase in consequences of an accident or malfunction of equipment.

3. The possibility of an accident of a different type than previously evaluated does not exist since the modification met system design criteria.

4. The margin of safety was not affected since the function and operation of the SI system was not altered by this modification.

DCP 94-147
WIRING MOD FOR DIESEL
FAST START FREQUENCY
NAPS UNIT 2

DESCRIPTION

Performing the Emergency Diesel Generator (fast start) periodic tests involved the installation of several electrical jumpers between the Control Room, switchgear room and the instrument rack room. These cables were carrying 150VDC which can be considered a personnel hazard. Also, the test recorder was connected using alligator clips. These clips did not provide a secure termination and were susceptible to disconnecting.

Sufficient spare cables existed between the emergency switchgear cubicles and the associated emergency diesel generator control cabinets in the Control Room to permanently install test points. These terminations, located in the diesel control panel in the Control Room, are used exclusively for performance of the Periodic Tests. The termination points in the emergency diesel cabinet are made using Pamona "banana jacks". Additionally, some internal wiring was installed between the 27W relays and the interconnection terminal blocks in the emergency switchgear to accomplish the monitoring.

SUMMARY OF SAFETY ANALYSIS (94-SE-MOD-072)

Installation of permanent test points with Pamona "banana jacks" in the EDG control panels does not constitute an unreviewed safety question as defined in 10CFR50.59 since it does not:

A) Increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety and previously evaluated in the UFSAR.

   Since the extension of the EDG start signal circuits with the addition of the Pamona "banana jacks" is passive and does not modify the circuit function and is only used during the performance of periodic tests, the probability of the occurrence or the consequences of an accident or malfunction of equipment important to safety and previously evaluated in the UFSAR are not increased.

DCP 94-147 (cont.)
WIRING MOD FOR DIESEL
FAST START FREQUENCY
NAPS UNIT 2

B) Create a possibility of for an accident or malfunction of a different type than any evaluated previously in the UFSAR.

The change made allows the performance of EDG periodic tests to be conducted more efficiently by permanently providing test points with "banana jacks" in the EDG control panels. This eliminates the possibility of personnel injury as temporary jumpers carrying 150VDC were previously installed. The possibility of creating an accident or malfunction of a different type than previously evaluated is not introduced.

C) Reduce the margin of safety as defined in the basis of any Technical Specification.

This design change does not change the basis of any Technical Specifications. Implementation of this modification simplifies the performance and eliminates a personnel hazard for the associated periodic tests.

DCP 94-156

INSTALL REDUNDANT CHECK VALVE IN RIES WITH 2-CH-153
NORTH ANNA UNI_ 2

DESCRIPTION

Westinghouse Nuclear Safety Advisory Letter 92-012 identified a potential post-LOCA ECCS leakage path to the environment during the SI recirculation mode of operation. The component of concern was the single check valve which isolated the combined VCT and seal water heat exchanger returns to the suction of the charging pumps. A single failure of the check valve could result in it not fully seating with back leakage occurring during LOCA with SI in recirculation mode. LHSI pump discharge pressure would then pressurize the seal water heat exchanger line and potentially cause its relief valve to lift. If the relief valve did lift, water recirculated from the containment sump would be discharged to the VCT. Eventually the VCT would fill sufficiently to compress the H_2 overpressure bubble and cause the VCT relief valve to lift. The discharge of potentially radioactive fluid from the VCT would be sent to the waste drain system and eventually provide a gaseous fission product release path to the environment.

The design change installed a check valve in the 3" seal water heat exchanger return line near its junction with the VCT o tlet line would provide a redundant check valve to prevent single failure of an active component (VCT outlet check valve) from initiating the event leading to the postulated ECCS leakage to the environment. A leak-off drain/vent valve was also installed between the two check valves to allow for future ISI leak testing of the check valves.

SUMMARY OF SAFETY ANALYSIS (94-SE-MOD-041)

The check valve installation did not constitute an unreviewed safety question as defined in 10CFR50.59 since it did not:

A) Increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety and previously evaluated in the UFSAR.

Activity installed a check valve in the seal water heat exchanger return line to provide redundancy and thus, eliminate single active failure of the VCT outlet check valve, as the cause of the potential ECCS leakage to the environment. Function and operation of the CVCS and ECCS did not change as a result of the modification. The check valve was sized to ensure that the charging pump minimum recirc flow rate was

maintained to prevent overheating of the pumps. The check valve and test valve assembly conformed to existing design codes and standards and qualified for use in this application. The modification did not corrupt mitigating systems or redundant features associated with the charging pumps, CVCS or ECCS.

B) Create a possibility for an accident or malfunction of a different type than any evaluated previously in the UFSAR.

Activity eliminated the potential for an unmonitored radiological release to the environment. Addition of a redundant check valve did not create any accident scenario that would not have been previously analyzed for the existing check valve arrangement. Modification met system design requirements. Addition of the check valve and test valve assembly had been seismically analyzed to ensure the seismic integrity of the line was maintained. UFSAR listed the most probable failure mechanism was back leakage through a normally closed valve. The redundant feature added by the check valve eliminated the possibility of creating an accident or malfunction not previously evaluated for the single check valve arrangement. Scenarios of failure due to a stuck closed check valve did not differ from that of existing conditions. New or unique malfunctions were not introduced.

C) Reduce the margin of safety as defined in the basis of any Technical Specification.

Operation and integrity of the CVCS remained unchanged. The operability of ECCS subsystems was not compromised by this activity. Verification that the charging pumps developed the required discharge pressure on recirculation flow was performed to ensure Tech Spec compliance. Charging pump recirc flow rate was verified to ensure minimum flow rate was maintained.

DCP 94-167
REPLACE CHECK VALVE
2-CH-155
NAPS UNIT 2

DESCRIPTION

The boric acid filter to blender check valve, 2-CH-155, had experienced repeated failures. The valve was a piston check. Piston check valves are not suitable for applications where there are solids in the fluid as the particulates may cause fouling of the piston guides causing the piston to hang up and allow back leakage through the valve. The valve was replaced with a ball check valve which is better suited for the application.

The check valve is in the normal makeup system to the VCT and is not part of the emergency boration flow path although it may be used as an alternate manual emergency boration flow path.

The valve is seismically qualified and the valve replacement resulted in a weight reduction of approximately 5 lbs. The weight change was evaluated and was found to be acceptable.

SUMMARY OF SAFETY ANALYSIS (94-SE-MOD-040)

All accidents were reviewed and the following were found to be applicable: LOCA, Control Rod Drive Mechanism Rupture, MSLB< Steam Generator Tube Rupture, Feedwater Line Break

This design change did not create an enreviewed safety question as defined by 10CFR50.59.

1) Accident probability was not affected. The emergency boration function is for accident mitigation only. The valve replacement, therefore, did not contribute to accident probability.

2) Accident consequences were not affected. The check valve, although not part of a normal emergency boration flow path, is part of an alternate manual emergency boration flow path. This flow path would only be used if a failure occurred in the normal emergency flow path. Under the single failure criteria, the check valve is assumed to operate correctly and the alternate emergency boration flow path would be used.

3) No unique accident possibilities were created. The replacement valve still acts as a backflow preventer and its safety function as a system pressure boundary was maintained.

4) Margin of safety was maintained because compliance with Technical Specifications for the boric acid storage tanks and

emergency boration flow paths was not affected.

DCP 94-170
LHSI VENT VALVES
NORTH ANNA UNIT #1

DESCRIPTION

Ultrasonic testing of the LHSI pump discharge lines had located air voids in the piping. Compression of these air voids during pump runs resulted in brief pressure transients. New SI vents were added to the LHSI discharge lines to facilitate venting of the piping system. Two new vents were added to the LHSI discharge lines. One vent per line was added in the Aux Bldg and one in the quench spray pump house. Each vent line was routed to an accessible location to facilitate venting of the system. The new vents installed in the QSPH were provided with double isolation valves to ensure that there is no leakage from the system. The vents installed in the Aux Bldg were provided with a single isolation valve. Hard piping was installed from the vents to a floor drain.
Leakage from the vent assemblies installed in the QSPH is not allowable. The QSPH ventilation system exhausts directly to atmosphere. No filters or radiation monitors are installed in the exhaust prior release to atmosphere. Any leakage from these vent assemblies creates the potential for unmonitored radiological release during an accident with the ECCS in the recirculation mode. Leakage from the vent valves is not expected as the piping, tubing and valves meet or exceed the system design requirements and are seismically qualified. Pipe caps have also been installed downstream of the isolation valves.

SUMMARY OF SAFETY ANALYSIS (94-SE-MOD-53)

The accidents which are applicable to this change are those which result in an SI actuation. The applicable malfunctions are ECCS piping failures, particularly a LHSI pump discharge line.

This design change did not create an unreviewed safety question as defined by 10CFR50.59.

1) Accident and malfunction probabilities were not increased as function and operation of the safety injection system was not affected. All components were purchased and installed in accordance with Q2 and LHSI system design requirements.

2) Accident consequences were not affected. Double isolation valves and a pipe cap were installed on each vent line in the QSPH to ensure that any radioactive material that may leak by the valve does not exit the SI piping into the QSPH.

3) No unique accident possibilities were created. The piping, tubing and valve installed meet or exceed the requirements of the SI system. The new lines are seismically supported. The double isolation valves and pipe caps ensure that there be no unfiltered, unmonitored radioactive release.

4) Margin of Safety was maintained because the function and operation of the SI system was not altered by this modification.

DCP 94-171
New LHSI Vent Valves
NAPS Unit 2

Description

In order to mitigate the pressure transients being experienced during LHSI pump start, additional vents have been added to the LHSI pump discharge piping. Two new vents have been added to the LHSI pump discharge lines in the Aux. Bldg. One new vent has been installed on each 8" LHSI pump discharge line at the high point of each piping run. One new vent has been added to 8-SI-440-153A-Q2, the "A" LHSI pump discharge line, at its high point in the QSPH. Tubing has been routed from the new vent valves to isolation valves installed in easily accessible areas to facilitate venting of the system. In addition, the existing vent off of 8"-SI-449-153A-Q2 found in the QSPH has been modified to include double isolation valves.

Summary of Safety Evaluation (94-SE-MOD-075)

Ultrasonic testing of the discharge lines had located air voids in the LHSI pump discharge piping. Compression of these air voids during pump runs resulted in brief pressure transients. New SI vents were added to the LHSI pump discharge lines to facilitate venting of the piping system. A new vent was added to each LHSI pump discharge line, 8"-SI-440-153A-Q2 and 8"-SI-449-153A-Q2 in the Aux. Bldg. Another new vent was also installed on 8"-SI-440-153A-Q2 in the QSPH. Each vent line has been routed to an accessible location to facilitate venting of the system. The new vents installed in the QSPH were provide with double isolation ensuring no leakage will exit the system. The vents installed in the Aux. Bldg. were provided with a single isolation valve. Hard piping has been installed from the vents to a floor drain. The new valves and vent lines are seismically installed. The supporting and routing are such that enough flexibility (i.e. expansion loops) has been built in to account for thermal, earthquake and transient movement/displacement of the main pipe run. The detached non safety related drain piping was supported in accordance with piping specification NAS-1009 such that it will not damage any safety related equipment during a seismic event.

Unreviewed Safety Question Analysis

1. The probability of occurrence of an accident or malfunction of equipment previously evaluated in the

UFSAR does not increase since the function and operation of the Safety Injection System did not change. All components were purchased and installed in accordance with Q2 and LHSI design requirements.

2. Double isolation valves and a pipe cap were provided on the vent lines installed in the QSPH ensuring any radioactive material that may leak by does not exit the SI piping into the QSPH. Therefore, there will be no increase in consequences of an accident or malfunction of equipment.

3. The possibility of an accident of a different type than previously evaluated does not exist since the modification meets system design criteria.

4. The margin of safety was not affected since the function and operation of the SI system was not altered by this modification and compliance with the ECCS Tech Specs was maintained during performance of the modification.

DC 94-172
Limit-Limit Operation of 2-SI-MOV-2867A&B
NORTH ANNA / UNIT 2

DESCRIPTION

During resolution of the Liberty Technologies Part 21 notification for Unit 2, Motor Operated Valve (MOV) as-left control switch trip (CST) settings were reviewed against the maximum thrust values. The CST settings for several of MOVs were left above the manufacturer's (Velan) specified continuous valve allowable because the targeted thrust band was too narrow to envelope inertial forces. Interim justification for this condition utilized the fatigue limit of the valve. Subsequent conversations with the manufacturer (Velan) resulted in the determination that increasing the continuous allowable may not be possible when considering stresses induced by seismic events. The corporate recommendation for resolving this issue was to modify the control circuitry to make the MOV limit-limit controlled. This method of control was in concert with STD-GN-0002.

Thirty-five valves (3" Velan) were identified as potential candidates for the limit-limit modification. All valves are 3" Velan valves in Groups 36 and 37 of the NAPS grouping methodology. The Unit 2 MOVs 2-SI-MOV-2867A&B were among those identified. Those valves were identified as suspects in a Boron Injection Tank dilution phenomenon observed by the station (indicating valve leak by). Station Management decided to inspect and repair (as appropriate) the valves in an attempt to remedy the leak-by. Station Planning identified that tagging window as adequate for the limit-limit modification.

SUMMARY OF SAFETY ANALYSIS

An unreviewed safety question did not exist because:

A. The implementation of this DCP did not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety and previously evaluated in the UFSAR thermal overload was still wired into the circuit to provide a back-up means of de-energizing the motor if catastrophic failure of the limit switch gearing or rotors occurs or if the MOV encounters a line obstruction prior to reaching the preset position.

B. The implementation of this DCP did not create a possibility for an accident or a malfunction of a different type than any evaluated previously in the UFSAR because this modification did not replace the limit switch but modified the wiring. This did not have any impact on increasing the probability of a switch failure. In addition, following this modification, diagnostic testing was performed to ensure the wiring change was correct and the limit switch operated correctly.

C. The implementation of this DCP did not reduce the margin of safety as defined in the basis of any Technical Specification because these valves still perform the same function to mitigate a LOCA, MSLB, or SGTR and did not change the system operation, flows, or safety related requirements. Therefore, an unreviewed safety question did not exist. In fact, the rewiring change improved the long-term reliability of the valve component portion of the MOVs since the change does better at controlling the seating thrust.

DCP 94-180

PERFORM WIRING MODIFICATION TO THE FLOW TRANSMITTER MODULE NORTH ANNA POWER STATION UNIT #2

Executive Summary

This design change interlocked a set of contacts on the emergency boration isolation valve (2-CH-MOV-2350) with the emergency boration flow transmitter module (2-CH-FM-2110). This set of contacts close when 2-CH-MOV-2350 is closed and open once the valve has been lifted from its seat. This set of contacts has been connected to the EXT CON of 2-CH-FM-2110. When the contacts are closed the EXT CON internal circuitry of the flow transmitter module (2-CH-FM-2110) drives the output to 4 mA, i.e. 0 gpm indication in the MCR. When the contacts are open, the EXT CON internal circuitry is bypassed and the output of the flow transmitter module (2-CH-FM-2110) will reflect the actual emergency boration flow conditions. The instrument accuracy of 2-CH-FM-2110 has not be affected by this design modification.

SUMMARY OF SAFETY ANALYSIS (94-SE-MOD-066)

This design change did not create an unreviewed safety question as defined in 10 CFR 50.59.

MAJOR ISSUES:

The major issue in this DCP was to ensure that physical independence and overfill in cable trays has been maintained in accordance with NAS-3012. The newly installed cable was color coded neutral and routed in neutral tray and conduit between 2-CH-FM-2110 and 2-CH-MOV-2350 in the Aux Building.

JUSTIFICATION:

This modification eliminated the problem of the emergency boration flow indicator (2-CH-FI-2110) from indicating full flow when flow does not exist. This erroneous indication in the MCR caused the operator to make additional checks to ensure that there was not flow to the suction header of the charging pumps. This modification has eliminated this problem by causing the output of the emergency boration flow transmitter/module

to be a constant 4 mA, i.e. 0 gpm, when emergency boration valve (2-CH-MOV-2350) is closed. The new cable has been installed in accordance with NAS-3012 to ensure physical independence and overfill in cable trays and conduit has been maintained.

UNREVIEWED SAFETY QUESTION ASSESSMENT:

1) Accident probability has not been increased because this design change conforms to standards and admins. The operation of 2-CH-MOV-2350 and the CVCS emergency boration system has not be affected. The reliability of the emergency borate flow indication has been improved. This has enhanced the reactor reactivity control. If the flow transmitter/module were to fail, the emergency boration function has not been affected, however the emergency boration flow indication in the MCR has been lost. The fact that emergency boration was taking place would be apparent to the CRO through other instrumentation, i.e. alarms, Tavg, valve position indication.

2) Accident consequences have not increased. The implementation of this DCP was performed during an outage. The operation of the CVCS emergency boration function has not been affected due to this design change. The reliability of the emergency borate flow indication in the MCR has been improved and has enhanced the reactor reactivity control. Consequences are not increased because a failure of the DCP has not corrupted any mitigating systems.

3) No unique accident probabilities have been created. The implementation of this DCP has not created a possibility for an accident or a malfunction of a different type than any evaluated previously in the UFSAR because the design change was performed during an outage and operation of the emergency boration function of the CVCS system has not been affected.

4) Margin of Safety has been maintained because the integrity and reliability of the CVCS has not been affected. The reliability of the emergency borate flow indication in the MCR has improved and this has enhanced the reactor reactivity control.

DCP 94-181

PERFORM WIRING MODIFICATION TO THE INSTRUMENT LOOP NORTH ANNA POWER STATION UNIT #2

DESCRIPTION

This design change will replace the flow transmitter module (2-CH-FM-2113) and interlock a set of contacts (MU/X5A) in the auxiliary relay cabinet (2-EI-CB-48A) with the new emergency boration flow transmitter module. These contacts will open anytime a MANUAL or BORATE makeup function is performed by the operator on the make up control switch (43/MU) on benchboard 1-1 in the MCR. In addition, these contacts will open anytime the make up control switch (43/MU) is in AUTO and a low VCT tank level signal has been received by the make up control circuitry. These contacts will be closed anytime there is not flow demanded, automatically or manually, from the Boric Acid Storage Tanks (BASTs) to the blender (line 1"-CH-456-153A-Q3). This set of contacts will be connected to the EXT CON of 2-CH-FM-2113. When the contacts are closed the EXT CON internal circuitry of the new flow transmitter module will drive the output to 4 mA, i.e. 0 gpm indication in the MCR. When the contacts are open, the EXT CON internal circuitry will be bypassed and the output of the flow transmitter module will reflect the actual makeup boration flow conditions.

SUMMARY OF SAFETY ANALYSIS (94-SE-MOD-067)

This design change did not create an unreviewed safety question as defined in 10 CFR 50.59.

A. The implementation of this modification did not increase the probability of occurrence or consequences of an accident or malfunctions of equipment important to safety and previously evaluated in the Final Safety Analysis Report.

   Accident probability has not been increased because this design change conforms to standards and admins. The operation of the CVCS makeup system will not be affected. The CVCS makeup system is not required for safe shutdown because the emergency borate flow path will be used to provide boric acid to the suction header of the charging pumps. This design change will provide a more accurate representation of the boric acid flow through the flow path of 02-CH-FCV-2113A.

B. The implementation of this modification did not create a possibility for an accident or a malfunction of a different type than any previously evaluated in the Final Safety Analysis Report.

The implementation of this DCP does not create a possibility for an accident or a malfunction of a different type than any evaluated previously in the UFSAR because the design change will be performed during an outage and operation of the CVCS makeup system will not be affected. Consequences have not been increased because a possible failure of the DCP would not have corrupted any mitigating systems.

C. The implementation of this modification did not reduce the margin of safety as defined in the basis of any Technical Specification.

The integrity and reliability of the CVCS have not been affected.

DCP 94-188-2

INSTRUMENT PORT COLUMN ASSEMBLY (IPCA) UPGRADE
NORTH ANNA UNIT 2

DESCRIPTION

In order to reduce the installation time and reduce radiation exposure to personnel working on disassembly/reassembly of the reactor, a new upgraded instrumentation port column assembly was installed on each of the four reactor head penetrations. The instrumentation port column assembly is the pressure retaining device that permits the incore instrumentation thermocouples to penetrate the pressure retaining boundary of the reactor head. The upgraded assemblies were designed, furnished and installed by Westinghouse, the original reactor supplier. The upgraded assemblies use the existing female flanges, thermocouple columns (conduit seal assemblies), and conoseal gaskets. The new upgraded assembly parts are new upper and lower articulated clamps, new male flanges, and new upper positioner clamps.

SUMMARY OF SAFETY ANALYSIS (94-SE-MOD-060)

This design change did not create an unreviewed safety question as defined in 10 CFR 50.59.

A. The implementation of this modification did not increase the probability of occurrence or consequences of an accident or malfunction of equipment important to safety and previously evaluated in the Final Safety Analysis Report.

   The evaluation of the instrumentation port column assembly modifications indicated that they have no affect on the operability or integrity of the reactor vessel. All service stress limits have been shown by analysis to be bounded by ASME Code limits. The instrumentation port column assembly modifications do not result in a condition where design, material, and construction standards that were applicable prior to the modification are altered. Therefore, the probability of an accident previously evaluated in the FSAR did not increase due to the instrumentation port column assembly modification.

   The instrumentation port column assembly modifications did not affect the integrity of the reactor vessel and penetrations such that their function in the control of radiological consequences was affected. In addition, the instrumentation port column assembly modifications did

not affect any fission barrier. The instrumentation port column assembly modifications did not change, degrade, or prevent the response of the reactor vessel to accident scenarios, as described in the FSAR Chapter 15. In addition, the instrumentation port column assembly modification did not alter any assumption previously made in the radiological consequence evaluations nor affect the mitigation of the radiological consequences of an accident described in the FSAR. Therefore, the consequences of an accident previously evaluated in the FSAR will not be increased.

The instrumentation port column assembly modification did not result in an increased probability of scenarios previously deemed improbable. The instrumentation port column assembly modification did not create any new failure modes for the reactor vessel and penetrations or other safety-related equipment. The instrumentation port column assembly modification did not result in original design specifications, such as seismic requirements, electrical separation requirements, and environmental qualification, being altered. In addition, the instrumentation port column assembly modification does not result in equipment used in accident mitigation to be exposed to an adverse environment. Therefore, the instrumentation port column assembly modification will not increase the probability of a malfunction of equipment important to safety previously evaluated in the FSAR.

The performance and integrity of the reactor vessel and penetrations are not affected such that their control of radiological consequences was altered. The instrumentation port column assembly modification does not result in a different response of safety-related systems and components to accident scenarios than that postulated in the FSAR. No new equipment malfunctions have been introduced that will affect fission barrier integrity. Therefore, the instrumentation port column assembly modification will not increase the consequences of a malfunction of equipment important to safety previously evaluated in the FSAR.

B. The implementation of this modification did not create a possibility for an accident or a malfunction of a different type than any previously evaluated in the Final Safety Analysis Report.

The instrumentation port column assembly modifications do not have any effect on the ability of the reactor vessel and penetrations to perform their intended safety functions. The instrumentation port column assembly modification did not create failure modes that could adversely impact safety-related equipment. Therefore, it did not create the possibility of a malfunction of equipment important to safety different than previously evaluated in the FSAR.

The instrumentation port column assembly modifications do not cause the initiation of any accident nor create any new credible limiting single failure. The instrumentation port column assembly modifications do not result in any event previously deemed incredible being made credible. Structural integrity will be maintained as indicated by compliance with ASME Code stress criteria. As such, the modifications did not create the possibility of an accident different than any evaluated in the FSAR.

C. The implementation of this modification did not reduce the margin of safety as defined in the basis of any Technical Specification.

The margin of safety with respect to integrity of the reactor coolant system is provided in part by the safety factors inherent to the ASME Boiler and Pressure Vessel Code. The instrumentation port column assembly modifications comply with the ASME Boiler and Pressure Vessel Code and have no effect on the availability, operability, or performance of the reactor vessel. Therefore, the instrumentation port column assembly modifications did not reduce the margin of safety, as described in the bases of any Technical Specification.

DCP 94-193

UF & UV RELAY RESETS
02-EP-27-VARIOUS
02-RPS-81-VARIOUS
UNIT 2

DESCRIPTION

The standard for Nuclear Plant Setpoints, STD-GN-0030, was revised November 20, 1992, and it included a more specific definition of where "Channel Statistical Allowance", CSA, was to be included in determining setpoints. This revision specifically states, in paragraph 5.3.4, that CSA is to be part of the calculation of setpoints for Technical Specification Underfrequency and Undervoltage setpoints. Calculation EE-0524 was revised. The relays were reset for Unit 2. The North Anna Setpoint Document, NASD, was also revised to show the new Underfrequency and Undervoltage setpoints, 56.55Hz and 86.95V (3043.5V primary), respectively.

SUMMARY OF SAFETY ANALYSIS (94-SE-MOD-034)

All accidents involving the loss of Primary Reactor Coolant flow were reviewed. The Loss of Reactor Coolant System flow due to undervoltage and/or underfrequency was found to be applicable. The new undervoltage and underfrequency relay setting changes enhanced protection of the Unit from the Loss of Reactor Coolant Flow by the additional conservatism based on the Channel Statistical Allowance methodology.

This design change did not create an unreviewed safety question as defined by 10CFR50.59.

1) Neither the probability of occurrence, nor the consequences of undervoltage or underfrequency were increased. The revised relay setpoints add conservatism to the existing relay setpoints by including the Channel Statistical Allowance.

2) This change did not create the possibility for an accident or malfunction of a different type. The setpoints are higher (i.e., more conservative) than the previous setpoints by the CSA plus the allowable relay setting range.

3) There is no reduction of the Margin of Safety shown in the SAR, and no changes were required to the Technical Specifications. The undervoltage and underfrequency relays will continue to provide protection against the loss of coolant flow in the Reactor Coolant System.

DCP 94-199

INSTALL QUICK DISCONNECTS AND REPLACE SWITCH CLAMP ASSEMBLY
NORTH ANNA POWER STATION
UNIT #2

Executive Summary

Station Management approved ALARA suggestion 90-ASR-039(s) and quick disconnects were installed on the four Reactor Vessel Head Vent System (RVHVS) isolation SOVs (2-RC-SOV-201A1/A2/B1/B2). In addition, new indicator switch clamp assemblies were installed on the four RVHVS isolation SOVs (2-RC-SOV-201A1/A2/B1/B2). This work was performed during the Unit #2 Steam Generator Replacement (SGR) Outage.

SUMMARY OF SAFETY ANALYSIS (95-SE-MOD-016)

This design change did not create an unreviewed safety question as defined in 10 CFR 50.59.

MAJOR ISSUES:

DCP 94-199 installed quick disconnects on the conduit/cabling and new switch clamp assemblies on the RVHVS isolation SOV's (2-RC-SOV-201A1/A2/B1/B2). The basic function of the RVHVS is to remove steam or non-condensable gases from the reactor vessel head to the refueling cavity during recovery from accident conditions. The system is designed to mitigate inadequate core cooling, inadequate natural circulation flow, and inability to depressurize to the RHRS initiation conditions resulting from the accumulation of non-condensable gases in the reactor coolant system. The most important thing considered by these design changes is that the operation of the RVHVS isolation SOV's has not been affected. The quick disconnects will not degrade the electrical connection between the control room and the RVHVS isolation SOV's in containment. The quick disconnects have been qualified for use in this application. The new switch clamp assemblies secure the position in place around the valve indicator tube and provide means for adjusting the position of the switch assemblies along the indicator tube. The new switch clamp assembly is designed to simplify the adjustment of the position indication switches. This simplified adjustment will eliminate incorrect position indications of these SOV's in the control room during a Unit startup.

These switch clamp assemblies only affect the indication of the SOV position in the control room and will have no affect on the operation of the SOV. In addition, these switch clamp assemblies have been qualified for this application.

JUSTIFICATION:

This modification will reduce the radiation exposure required to disconnect the RVHVS SOV's for reactor head removal during an outage and enhance the RVHVS isolation SOV indication in the control room.

UNREVIEWED SAFETY QUESTION ASSESSMENT:

1) Accident probability has not been increased because this design change conformed to standards and admins. The operation of the RVHVS has not been affected. The operation of the RVHVS isolation SOV's will remain the same. A more reliable RVHVS SOV indication will be provided in the control room.

2) Accident consequences are not increased. The operation of the RVHVS system has not be affected due to this design change. RVHVS SOV indication in the MCR has not only not been affected, but enhanced during Unit startup. Consequences are not increased because a failure of the DCP will not corrupt mitigating systems.

3) No unique accident probabilities are created. The implementation of this DCP does not create a possibility for an accident or a malfunction of a different type than any evaluated previously in the UFSAR because the operation of the RVHVS system has not been affected.

4) Margin of Safety is maintained because the integrity and reliability of the RVHVS is unchanged.

DCP 94-209

REMOVE MECHANICAL AGITATOR
1-DC-TK-6, 1-DC-WASH-1
UNIT 1

The mechanically agitated cleaning tank, dishwasher and PRV station which supplied steam to the tank for decontamination activities were removed. None of the equipment affected had any safety related function. However, UFSAR figures 9.5-14 and 9.5-15, Arrangement, Decon Building, were affected by the equipment removal. The mechanical agitator and dishwasher were used to decontaminate equipment and tools after they were removed from the nuclear plant. A new CO_2 decontamination system is being used and the mechanical agitator was being removed to free space in the decon building.

There are no accidents or malfunctions which were applicable to this change.

UNREVIEWED SAFETY QUESTION ASSESSMENT

1) Accident probability was not increased as the decon equipment had no affect on the cause for any accident.

2) Accident consequences were not affected. The decontamination equipment had no role in accident mitigation.

3) No unique accident possibilities were created. The mechanical agitator and PRV station were used to decontaminate equipment and tools already removed from the nuclear plant. The removal of the tank and PRV station had no affect on the operating plant.

4) Margin of Safety was maintained because the decontamination equipment had no affect on any Technical Specification equipment or any of the Technical Specification bases.

DCP 94-211

ELECTRICAL AND MISSILE MANHOLE MODIFICATIONS
North Anna/ Unit 1 & 2

DESCRIPTION

Continuous submergence of electrical cable at North Anna posed a serious threat to the reliable operation of the cable and plant. This DCP performed various modifications to stop water intrusion through manholes into cable vaults. Some manholes were raised. Some manhole covers were replaced with self-sealing covers and or gaskets to stop the infiltration of rainwater. In some instances a drain was installed in the vault to remove any water inundating the cable vault. Also, a maintenance procedure was created to periodically look at manholes and pump them if necessary.

While preparing this DCP it was found that some existing manhole covers over safety related cabling were not missile protecting covers. As a result of investigating Deviation Report Nos. N-94-1734 & N-94-1868, Virginia Power's Civil Engineering groups ascertained that the original manhole covers were provided as gray cast iron and were not adequate as missile protection. The existing covers were replaced with 2" thick solid steel covers. Protection of safety related equipment or components inside electrical vaults was ensured due to installation of the 2" thick steel covers.

SUMMARY OF SAFETY ANALYSIS

This design change in accordance with DCP 94-211 does not create an "unreviewed safety question" as defined in 10 CFR 50.59.

A. The implementation of this modification does not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety and previously evaluated in the Final Safety Analysis Report. Calculations were performed to confirm the adequacy of the 2" thick steel manhole covers. These covers are able to withstand the design tornado missiles described in NAPS UFSAR, Section 3.5.4.

B. The Implementation of this modification does not create a possibility for an accident or malfunction of a different type than any previously evaluated in the Final Safety Analysis Report because based on engineering calculation it has been demonstrated that the design requirements for tornado missile protection have been met.

C. The implementation of this modification does not reduce the margin of safety as defined in any Technical Specification because the margin of safety of the manhole covers to resist the postulated UFSAR tornado missiles has improved. Gray cast iron manhole covers have been replaced with 2" thick solid carbon ASTM A-36 steel covers.

DC 94-212
MODIFICATION OF EDG ROOM DOORS
NORTH ANNA / UNITS 1 & 2

DESCRIPTION

The Emergency Diesel Generator (EDG) room doors opening to the Turbine Building hallway were modified to make it possible to latch the doors open during periodic test operation of each EDG. A slide bolt was mounted on the inside lower corner of each EDG door. A mortise, to accept the slide bolt, was grouted into the Turbine Building floor which allows the door to be held open. This modification was performed to minimize the potential for injury to personnel trying to enter and exit the room while the EDG is running during test operation. EDG air intake during operation creates a strong negative pressure in the EDG room which makes the door difficult to open and subject to slamming closed. Damage due to repeated slamming has required the doors to be periodically replaced. The door is allowed to be latched and held open only when the EDG is considered inoperable during performance of EDG periodic testing.

SUMMARY OF SAFETY ANALYSIS

An unreviewed safety question did not exist because:

A. The implementation of this DCP did not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety and previously evaluated in the UFSAR because the EDG door will be latched and held open only when the EDG is declared inoperable for periodic testing. The doors are considered fire, security and EQ barriers important to the safe shutdown of the plant. A continuous fire watch is required as a compensatory action whenever the door is held open during periodic test operation.

B. The implementation of this DCP did not create a possibility for an accident or a malfunction of a different type than any evaluated previously in the UFSAR because the door is only held open when the EDG is considered inoperable during periodic testing. The ability for an operable EDG to respond to an emergency start signal was not affected. The use of screws to attach the slide bolt mechanism on the inner door leaf (only) satisfies 10CFR50 Appendix "R" design requirements. Integrity of the EDG room doors to provide an effective fire barrier is maintained.

C. The implementation of this DCP did not reduce the margin of safety as defined in the basis of any Technical Specification because the ability of an operable EDG to respond to an emergency start signal is not affected.

DCP 94-216
North Anna Power Station

Installation of Containment Instrument Air Header Manual Isolation Valves

DESCRIPTION:

The containment instrument air dryer skids 2-IA-D-2A and 2B could not be isolated from the containment instrument air header. Isolation valves 2-IA-2278 and 2-IA-2279 were installed in the lines to the containment instrument air dryer skids 2-IA-D-2A and 2B. These valves allow isolation of the dryer skids from the instrument air header when maintenance is required.

This activity required the UFSAR figure 9.3-4 to be updated; therefore the facility as described in the safety analysis report required change.

SUMMARY:

The safety analysis report was reviewed. No accidents previously evaluated were identified as being applicable. The work to be preformed is not safety related, is non seismic, and non EQ. All safety related air operated valves that may be operated during accident conditions have air available from air storage tanks of sufficient capacity. The rupture of a main instrument air header and complete loss of air was a malfunction considered in the UFSAR. This activity will not increase the probability or the consequences of this malfunction. Nor will it create the possibility for an accident or malfunction of a different type than was previously evaluated in the UFSAR. The operating license and technical specification will not require a change as a result of this activity. The margin of safety as defined in the Technical Specifications will not be reduced. The fire protection system will be unaffected by the addition of two isolation valves to the containment instrument air system. For these reasons a unreviewed safety question did not exist and the installation of the two instrument air isolation valves was allowed.

DCP 94-239
INSTALLATION OF LED LAMPS IN THE E30JF SWITCHES
NAPS UNIT 1 & 2

DESCRIPTION

The 120PSB incandescent light bulbs installed in the Cutler-Hammer E30JF switches powered from the 125VDC buses had been experiencing a high rate of failure. These switches are used for valve control and position indication. During bulb replacement Operations had been experiencing bulbs pulling apart from their base, blown fuses and switch damage due to the light bulb shorting in the socket.

This design change replaced the Sylvania 120PSB incandescent light bulbs with LED lamps. All of the affected 120PSB incandescent light bulbs were installed in the Cutler-Hammer E30JF switches mounted on the unit 1 and unit 2 vertical boards in the control room. All of the affected switches are powered from a 125VDC bus. The associated light bulbs in the simulator have also been replaced with LED lamps.

SUMMARY OF SAFETY ANALYSIS (94-SE-MOD-059):

An unreviewed safety question does not exist for the following reasons:

1 Neither the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety have been increased due to the failure of an LED lamp. The failure of an LED lamp will not adversely affect the operability of a vital bus or reduce the capability to control any equipment.

2 The possibility of an accident of a different type has not been created by this modification. All equipment and plant conditions remain unchanged. The operations ability to monitor the plant has not be diminished.

3 The margin of safety has not been reduced. No plant parameter will change due to the replacement of incandescent bulbs with LED lamps. The intent of using LED lamps is to increase the reliability of control room indications.

DCP 94-247
North Anna Power Station

Abandonment of Containment Hydrogen Analyzers

DESCRIPTION:

The Bendix Hydrogen Analyzers 1-HC-H2A-100 and 2-HC-H2A-200 are no longer used. All data pertaining to the alignment of this equipment has been deleted from operating procedure 1-OP-63.2. Technical Specification Section 3.6.4.1 requires that two independent hydrogen analyzers are to be operational during modes 1 and 2. This post accident monitoring requirement is already met by having the pair of Comsip Hydrogen Analyzers, 1-HC-H2A-101 and 2-HC-H2A-201, on standby. Therefore, the Bendix Hydrogen Analyzers are not required and can be officially abandoned. (Ref REA 94-222)

This DCP abandoned the Bendix Analyzers.

SUMMARY:

The probability of occurrence of an accident or malfunction of equipment does not increase since the function and operation of the Containment Atmosphere Cleanup System does not change. The Comsip Analyzers are still available to monitor hydrogen concentration inside containment. Therefore, there will be no increase in the consequences of an accident or malfunction of equipment. The possibility of an accident or malfunction of a different type than previously evaluated is not created as the abandoned analyzers will be physically isolated and will have no interface with the operating systems. The margin of safety is not affected as the function and operation of the Containment Atmosphere Cleanup System is not changed. The Comsip Analyzers are still available to perform their safety function.

DCP 94-249
REPLACEMENT OF CASING COOLING CHILLER FLOW SWITCHES

SUMMARY:

Due to a history of failure, the casing cooling chiller flow switches were replaced with switches that are more reliable. The old paddle type switches were replaced with switches incorporating a flow disk. This DCP required minor piping changes to the unit 1 casing cooling tank recirculation piping. The flow switches and related piping are not safety related however UFSAR figure 6.2-83 will be affected.

UNREVIEWED SAFETY QUESTION ASSESSMENT

1) Accident probability was not increased as the casing cooling system is for accident mitigation only.

2) Accident consequences were not affected. The casing cooling water is still subject to the same Tech. Spec. requirements. The piping change has no affect on the operation and function of the casing cooling system.

3) No unique accident possibilities were created. The operation and function of the casing cooling water system were not affected. The system is normally in standby mode and is needed in the event of an accident with CDA.

4) Margin of safety was maintained because the Tech. Spec. requirements for the casing cooling water were not affected.

DC 94-255
Second Floor Admin Building Mods
NORTH ANNA / UNIT 1&2

DESCRIPTION

The second floor of the Administration Building was remodelled. The bulk of the work was non-safety related. Part of the remodeling effort involved demolition of obsolete and abandoned fire protection piping, passive sprinklers and smoke detectors as well as relocation of two hose stations.

SUMMARY OF SAFETY ANALYSIS

An unreviewed safety question did not exist because:

A. The implementation of this DCP did not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety and previously evaluated in the UFSAR because the fire protection components were either relocated (hose stations) or demolished due to their obsolescence.

B. The implementation of this DCP did not create a possibility for an accident or a malfunction of a different type than any evaluated previously in the UFSAR because this modification did not alter any active aspects of the present fire protection or Appendix "R" scheme.

C. The implementation of this DCP did not reduce the margin of safety as defined in the basis of any Technical Specification because the relocated hose stations still provide their intended protection and the demolished components were no longer in service.

DCP 94-260

GENERATOR B.U. IMPEDANCE RELAY RESET
02-EP-21-2SPGN02
UNIT 2

DESCRIPTION

Power protective relay settings are determined by System Protection, Circuit Calculations, in accordance with Virginia Power Circuit Calculations Methods Manual. The philosophy of application for the Impedance Back Up relay changed such that it will provide greater coverage of the Main Transformer winding. The Unit 2 Main Generator Back Up Impedance protective relay was reset. The former setting was 50% of the transformer winding, and the new setting is 80% of that impedance. The Setpoint Document and Calculation EE-0310 were revised to reflect the new relay setting.

SUMMARY OF SAFETY ANALYSIS (94-SE-MOD-069)

All accidents were reviewed, and the following were found to be applicable: Loss of Electrical Load and/or Turbine Trip, Loss of Normal Feedwater and Loss of Offsite Power to Station Auxiliaries.

This design change did not create an unreviewed safety question as defined by 10CFR50.59.

1) Neither the probability of occurrence, nor the consequences of a Loss of Electrical Load and/or Turbine Trip, Loss of Offsite Power to Station Auxiliaries or Loss of Normal Feedwater were increased as a result of this setpoint change to increase the Impedance relay zone of protection.

2) This change did not create the possibility for an accident or malfunction of a different type because the Impedance relay provides the same function as it did before the new, more conservative, setting was applied.

3) There was no reduction in the margin of safety shown in the SAR, and no changes were required to the Technical Specifications. The Impedance Backup relay will continue to provide protection for the Generator for an electrical fault condition initiating a generator/turbine trip as it did before the reset.

DC 94-274
SEAL SAFEGUARDS VALVE PIT ACCESS SHAFT TO PREVENT GROUND WATER INTRUSION
NORTH ANNA / UNITS 1 & 2

DESCRIPTION

Ground water was leaking into the unit 1 and unit 2 Safeguards valve pit through rattlespace joints between the concrete access shaft and containment exterior wall. The ground water in-leakage is removed and processed by pumping from the Safeguards valve pit sump to the radioactive liquid waste system. The ground water has a high mineral content (calcium) that accelerated depletion of the radioactive liquid waste processing resins. Cost savings were acheved by implementation of this Design Change due to reduction of the radioactive disposal and replacement of filter resins. The leaking joints were sealed by placement of a flexible hydroactive urethane grout into the rattlespace.

SUMMARY OF SAFETY ANALYSIS

An unreviewed safety question did not exist because:

A. The implementation of this DCP did not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety and previously evaluated in the UFSAR because repair of the Safeguards valve pit access shaft ground water in-leakage has no effect on the probability of having a LOCA, MSLB or earthquake. Unlikely gross failure of the flexible urethane grout seal could possibly create flooding in the Safeguards valve pit. The Safeguards building was designed for operation if the valve pit is flooded due to a LHSI or outside RS pipe break. Operability was not affected since the LHSI and outside RS pump are a vertical shaft design with the discharge head and motor drives approximately 45 feet above the valve pit. The flexible urethane grout seal in the rattlespace does not restrict design movement of containment ensuring that containment integrity is maintained. The flexible joint seal does not alter the dynamic response of containment during a DBE seismic event.

B. The implementation of this DCP did not create a possibility for an accident or a malfunction of a different type than any evaluated previously in the UFSAR because flooding of the valve pit does not affect operability of the LHSI or outside RS pumps due to design of the Safeguards building.

C. The implementation of this DCP did not reduce the margin of safety as defined in the basis of any Technical Specification because operability of the LHSI and outside RS pumps is not affected by worst case ground water flooding of the Safeguards valve pit.

DCP 94-293
North Anna Power Station

Instrument Air Line Removal from
2nd Floor Admin. Bldg.

DESCRIPTION:

An instrument air test connection was located in the new office library, on the second floor of the administration building. IPR# 1319 to DCP 94-255

The instrument air test connection which was located on the second floor of administration building was longer needed. This test connection and line that feeds it was removed. The instrument tubing between 1-IA-1491 and 1-IA-1492 was replaced. The tee between these valves for the line being removed was eliminated.

This activity required the UFSAR figure 9.3-11 to be updated; therefore the facility as described in the safety analysis report required change.

SUMMARY:

The safety analysis report was reviewed. No accidents previously evaluated were identified as being applicable. The work to be preformed is not safety related, is non seismic, and non EQ. This activity will not increase the probability or the consequences of any malfunction. Nor will it create the possibility for an accident or malfunction of a different type than was previously evaluated in the UFSAR. The operating license and technical specification will not require a change as a result of this activity. The margin of safety as defined in the Technical Specifications will not be reduced. The fire protection system will be unaffected by this activity. The activity is benign and will not have an adverse environmental impact. There will be no change in effluents or power level as a result of the proposed change. For these reasons a unreviewed safety question did not exist and the removal of the test connection and the line feeding it was allowed.

DCP 94-296

PERMANENT WALL ATTACHMENTS FOR EDG CURTAINS
01-EE-EG-1J, 02-EE-EG-2H AND 02-EE-EG-2J
NORTH ANNA UNIT 1 & 2

DESCRIPTION

In order to maintain Emergency Diesel Generator operability in extreme cold weather, additional space heating and weather protection may be required to maintain the required temperature in the EDG cubicles. Installation of temporary space heaters in the area of the governor and temporary Herculite curtains in each of the EDG cubicles, 2H, 1J and 2J, between the louvered E-line wall and the missile shield block wall was approved in response to REA # 94-068 via the memo from G. T. Bischof to D. W. Roberts dated 01-19-94. EDG cubicle 1H has a missile door, at the opening, which can be closed during extreme cold weather. Safety evaluations 94-SE-JMP-003 and 94-SE-OT-007 were performed to ensure that no unreviewed safety questions existed due to these temporary modifications. General Operations Procedure 0-GOP-4.2, Extreme Cold Weather Operation, was developed to install and remove the above temporary modifications during extreme cold weather (<10°F) in addition to several other temporary modifications related to extreme cold weather.

The response to the REA 94-068 and the Procedure 0-GOP-4.2 required that the temporary Herculite curtains be suspended across the opening between the missile shield wall and the louver wall at the walkway around the missile barrier and that nails and fire retardant wood be used to attach the curtains to the walls.

This method of hanging the Herculite curtains results in damage to the concrete wall when the curtains are removed. Therefore, a permanent wall attachment is required to hang the curtains when required during extreme cold weather.

SUMMARY OF SAFETY ANALYSIS

This design change in accordance with DCP 93-132 does not create an "unreviewed safety question" as defined in 10 CFR 50.59.

A. The implementation of this modification does not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety and previously evaluated in the Final Safety Analysis Report.

The permanent angle attachment on the EDG walls are to facilitate hanging of temporary herculite curtains in the EDG rooms during extreme cold weather. This will increase ambient temperatures in the rooms and preclude equipment damage due to frigid conditions and consequent inoperability of the EDGs.

Installation of the Herculite curtains will reduce the air flow through the access opening around the missile shield wall. This will reduce the loads on the existing heating system in the rooms. Installation of the Herculite will not have a detrimental effect on the EDG since the design air flow is through the 16'X16' louvers and over the missile barrier not through this passage and around the missile barrier. It should be noted that the 1H EDG room has a missile door installed which performs the same function as the Herculite would in terms of blocking the air flow. If the Herculite should tear away from its reinforced eye-lets, it would only be to a point when the resistance to air flow is overcome. If the Herculite would fall during an accident it would fall to the ground and would have a tortuous path to follow to end up at the EDG air inlet. Based on the path, the fact that the air intake is approximately eight feet off of the ground, it is not possible for the Herculite to dislodge and block the EDG air intake. The Herculite is hung with the reinforced eye-lets and twist nuts permanently attached to the angle sections. The angle sections are attached to the wall with ⅝" dia. Hilti-Kwik anchor bolts, in accordance with NAS-1023, whose capacity will far exceed the forces due to the installation of the herculite curtains. The installation is seismically rugged and failure of the angle attachment is inconceivable during a Design Basis Seismic event.

Installation of the Herculite sheeting and the additional painting of the angle sections will not adversely impact the fire protection for the EDG rooms. The additional combustible loading is insignificant when compared to the total loading for each room. The effectiveness of the CO_2 system is not changed since CO_2 nozzles are not blocked or room gas boundary is not impacted. There is no significant change in the probability of a fire since Herculite used at the site meets the fire retardant requirements of NFPA 701. The man-size opening in the herculite curtains, normally kept closed by velcro could easily be opened during emergency which will provide adequate lighting, from the emergency lighting fixture on the west of the curtain, for the passageway to the exit

door opening into the alleyway. The temporary use of Herculite for the intended purpose does not require revision of the Appendix "R" Report. The installation and removal of the temporary Herculite curtains will be controlled by 0-GOP-4.2.

Therefore, the implementation of the modification will not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety and previously evaluated in the Final Safety Analysis Report.

B. The implementation of this modification does not create a possibility for an accident or malfunction of a different type than any previously evaluated in the Final Safety Analysis report.

The implementation of the actions will have no impact on the operation, actuation or function of any equipment and the actions are simple in nature and all failure modes have been evaluated. Therefore, this modification will not create a possibility for any other accident or malfunction and will not jeopardize any equipment, system or procedure required to operate the plant safely and achieve and maintain safe shut down or to prevent the release of radiation for any condition.

C. The implementation of this modification does not reduce the margin of safety as defined in any Technical Specification.

All requirements of the Technical Specifications, Section 3/4.8 will still be complied with subsequent to the installation of the permanent wall attachments to hang temporary herculite curtains in the EDG rooms in accordance with 0-GOP-4.2, Extreme Cold Weather Operation. Therefore, the installation of the permanent wall attachments to hang temporary herculite curtains in the EDG rooms will have no impact on the Margin of Safety of the operation of the EDGs.

DCP 95-004

ROD CONTROL SYSTEM TIMING CHANGES

DESCRIPTION

On May 27, 1993, an event occurred at the Salem Nuclear Generating Station involving failures in Rod Control System circuitry which resulted in rod withdrawal when rod insertion was demanded. As a result of this event, the NRC issued Generic Letter 93-04, **Potential Problem with Westinghouse Rod Control System and Inadvertent Withdrawal of a Single RCCA.** In response to the Generic Letter, a Westinghouse Owners Group (WOG) program was initiated to assess the safety impact of this type of failure and to recommend modifications to prevent recurrence of the problem experienced at Salem. The WOG program concluded that the licensing basis continued to be met but identified options which would enhance the basis for that determination. One of the options was to implement new current order timing, which would preclude any single rod uncontrolled withdrawal event due to a single failure in the Rod Control System.

DC 95 004 implemented the modified Rod Control System current order timing recommended by the WOG. Westinghouse provided Technical Bulletin NSD-TB-94-05-RO containing the information required to make the modifications to the Slave Cycler Decoder cards to obtain the new timing. The modifications consisted of repositioning diodes on the cards in accordance with the Westinghouse instructions.

SUMMARY OF SAFETY EVALUATION(95-SE-MOD-17)

All UFSAR chapter 15 accidents and malfunctions were reviewed and the following were found applicable:

- Dropped rod event
- Uncontrolled withdrawal of rod cluster control assembly at power and when subcritical
- Rod cluster control assembly misalignment
- Single rod cluster control assembly withdrawal at power
- Improper rod movement due to Rod Control System malfunction

This design change did not create an unreviewed safety question as defined by 10CFR50.59 because:

1) Accident or malfunction probability was not increased. Implementation of the timing changes will preclude any uncontrolled single rod withdrawal event from occurring due to a single failure in the rod control system.

ROD CONTROL SYSTEM TIMING CHANGES

2) Accident or malfunction consequences were not increased. The applicable accidents and malfunctions are mitigated by the reactor protection system,which is not affected by the control system timing changes

3) No new accident or malfunction possibilities were created. The timing changes did not alter the functional capabilities of the reactor protection or control systems, therefore no additional failure modes were introduced.

4) The implementation of the timing changes did not affect any margin of safety as defined in the Bases of any Technical Specification.

DCP 95-009-2
RESOLVE DC SEPARATION ISSUE FOR SOV PANELS
NORTH ANNA / UNIT 2

DESCRIPTION

NRC IN 95-10 identified a concern of potential impact to vital power supplies which serve protection systems due to failure of nonsafety-related circuits. Because of the concern raised by IN 95-10, a more general DC coordination review was performed for North Anna to ensure that failure of nonvital circuits could not disable safety-related circuits. During this review, it was noted that some vital circuits were identified as being powered from the Unit 2 DC SOV panels. However, these panels were powered via neutral (i.e., black) circuits from the main DC switchboards. Therefore, portions of the circuits which fed the safety-related Reactor Vessel Head Vent and Pressurizer Vent SOVs did not have proper electrical separation. This DCP implemented the required changes to provide adequate electrical separation for the two trains of the Unit 2 Reactor Vessel Head Vent and Pressurizer Vent SOVs.

SUMMARY OF SAFETY EVALUATION

Safety Evaluation 95-SE-MOD-27 was prepared in support of this design change. This design change did not create an unreviewed safety question as defined in 10CFR50.59.

A. The implementation of this DCP did not increase the probability of occurence or the consequences of an accident or malfunction of equipment important to safety and previously evaluated in the Safety Analysis Report.

   Powering the two trains of the Unit 2 Reactor Coolant System Vent SOVs from redundant and properly separated circuits (in accordance with this DCP) reduces the consequences of an accident, since the single failure of a common raceway no longer can cause the loss of both trains.

B. The implementation of this modification did not create a possibility for an accident or malfunction of a different type than any previously evaluated in the Safety Analysis Report.

   This DCP assured operation of the RCS Vent SOVs by obtaining separation in the 125VDC circuits feeding these loads. Previously, failure of a common raceway would result in the loss of both trains of the RCS Vent SOVs; after the DCP was implemented, a single failure would result in the loss of only one train.

C. The implementation of this modification did not reduce the margin of safety as defined in the basis of any Technical Specification.

The implementation of this DCP enhanced the margin of safety by ensuring that at least one train of RCS Vents would be available for a single failure.

[87] From: William E. Carter at innc01 2/6/96 7:58AM (4917 bytes: 1 ln, 1 fl)
To: Larry Runyon at NAIS01, Teresa G. Wilson at NAIS01, John C. Lusk at NAIS01
cc: William C. Stallings
ject: DCP 95-009-2 SUMMARY FOR ANNUAL NRC REPORT
---------- Message Contents ----------

Text item 1:

Attached is the summary for DCP 95-009-2, "Resolve DC Separation Issue for SOV Panels". I typed this summary in WordPerfect 5.1 and attached it to this cc:mail.

If you require any additional information, please advise.

DCP 95-109

2-CH-MOV-2370 REPLACEMENT
NORTH ANNA UNIT 2

DESCRIPTION

During a previous VOTES test, 2-CH-MOV-2370 was stroked closed and the MOV went to locked rotor. The resulting thrust exceeded the maximum allowable thrust and the valve was damaged beyond repair. Valve replacement was therefore required. Activity replaced 2-CH-MOV-2370 with a similar valve that had a different stem configuration than the existing valve. Due to the different stem thread geometry, modification to the motor operator was performed to reduce the overall gear ratio (OGR) to ensure that the stall thrust of the operator was not exceeded. The replacement valve exhibited an increased stroke time due to the difference in stem geometry and operator OGR. Control circuitry modification per DCP 93-260 was performed in conjunction with this design change since the resulting available thrust band was not adequate to allow the motor operator scheme for the replacement valve to be torque seated. The new valve incorporated a different packing/stuffing box design so the lantern ring leak-off line from the existing valve was eliminated.

The valve replacement and operator modification was acceptable because the new valve met the original design requirements. Slightly improved pressure drop and flow characteristics associated with the new valve increased valve performance. The existing operator with a modified OGR had been reviewed by the valve vendor to ensure that the valve motor operator was sized to equal or exceed the thrust required to open or close the valve under all conditions that required valve operation. Since the valve was not subject to any Tech Spec stroke time or ESF response time requirements, the increased stroke time was not critical to valve operation. The safety function of the valve was not compromised by this activity. The packing design of the replacement valve did not adversely affect valve operation.

SUMMARY OF SAFETY ANALYSIS (95-SE-MOD-021)

The valve replacement and operator modifications did not constitute an unreviewed safety question as defined in 10CFR50.59 since it did not:

A) Increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety and previously evaluated in the UFSAR.

The replacement valve met or exceeded the design requirements of the original equipment. Since the activity replaced the

valve with an equivalent, the operability and performance capability of the valve and CVCS was not compromised. Existing motor operator characteristics had been reviewed to ensure that sufficient thrust was developed by the new valve. The valve's pressure boundary and isolation safety function remained unaffected. The increased stroke time did not adversely affect the safety function of the valve. Since the valve did not function for accident mitigation, the valve replacement and increased stroke time did not generate or increase accident consequences.

B) Create a possibility for an accident or malfunction of a different type than any evaluated previously in the UFSAR.

The activity involved replacement of 2-CH-MOV-2370 with an equivalent valve which in no way adversely affected operation of the valve or CVCS. The valve replacement was essentially a one-for-one replacement. The different stem configuration increased the calculated valve stroke time but this change did not impact the safety function of the valve. No Tech Spec stroke time or ESF response time requirements existed for 2-CH-MOV-2370. Accidents or malfunction of equipment of a different type than was previously evaluated were not credible due to the one-for-one type of valve replacement.

C) Reduce the margin of safety as defined in the basis of any Technical Specification.

Replacement of the valve and modification to the operator did not adversely affect operation or valve function. Since the valve thermal overloads were not affected by the activity, the Tech Specs margin of safety associated with MOV thermal overloads was not impacted by this modification.

DCP 95-112
North Anna Power Station

Install Separate Drain Line for Turbine Lube Oil Demister

DESCRIPTION:

The turbine lube oil tank vapor extractor and vent line demister shared a common drain line and loop seal to drain. Since December of 1994 and the implementation of this DCP there was a problem with drainage of oil from the demister. A high oil level within the demister resulted in oil droplets being carried out with the exhaust air. The result was oil mess on the equipment and ground, north side of the turbine building.

The turbine lube oil tank exhaust fan uses the original drain line and loop seal. A new drain line and loop seal were installed in a similar routing to the existing drain line for dedicated use by the demister. This allows drainage of the demister without back pressure and flow from the exhauster fan's drain line. The old connection to the existing line was capped off.

SUMMARY:

This activity required the UFSAR figure 10.4-16 to be updated; therefore the facility as described in the safety analysis report required change.

The safety analysis report was reviewed. No accidents previously evaluated were identified as being applicable. The work to be preformed is not safety related, is non seismic, and non EQ. This activity will not increase the probability or the consequences of any malfunction. Nor will it create the possibility for an accident or malfunction of a different type than was previously evaluated in the UFSAR. The operating license and technical specification will not require a change as a result of this activity. The margin of safety as defined in the Technical Specifications will not be reduced. The fire protection system will be unaffected by this activity. The activity is benign and will not have an adverse environmental impact. There will be no change in effluents or power level as a result of the proposed change. For these reasons a unreviewed safety question did not exist and the installation of a separate drain line for the turbine lube oil demister was allowed.

DC 95-115
RELOCATION OF REACH ROD PENETRATION FOR 1-CH-4
NORTH ANNA / UNIT 1

DESCRIPTION

Cation Bed Demineralizer inlet isolation valve 1-CH-4 could not be remotely operated due to misalignment of the reach rod penetration and valve stem. The concrete penetration through the north wall of Demin Alley was enlarged to allow alignment and reinstallation of the reach rod for valve 1-CH-4. Enlarging the concrete wall penetration required the cutting of reinforcing bars in the wall. Cutting rebar was evaluated by Design Engineering and found acceptable.

SUMMARY OF SAFETY ANALYSIS

An unreviewed safety question did not exist because:

A. The implementation of this DCP did not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety and previously evaluated in the UFSAR because structural integrity of the north wall of Demin Alley to resist a design basis earthquake is maintained. The amount of concrete and reinforcing steel removed is very small compared to the overall size of the massive wall. The 24" thick wall is sized primarily for radiation shielding to allow remote operation of valves within Demin Alley. No heavy equipment is mounted on the wall in the area of the penetration being enlarged.

B. The implementation of this DCP did not create a possibility for an accident or a malfunction of a different type than any evaluated previously in the UFSAR because RCS inventory loss from 1-CH-4 due to a ruptured diaphragm is well within the make up capability of one Charging Pump. Inventory loss and radioactive consequences for a leak at 1-CH-4 due to a ruptured diaphragm is enveloped by the existing safety analysis for a VCT rupture.

C. The implementation of this DCP did not reduce the margin of safety as defined in the basis of any Technical Specification because uncontrolled boron dilution is unlikely due to existing procedural precautions to equalize the Cation Bed Demineralizer with RCS boron concentration prior to placing the Demineralizer in service. The original system design basis for remote operation of valve 1-CH-4 was restored.

DC 95-116
ROOF UPGRADE, FUEL RECEIVING, U1 CASING COOLING
NORTH ANNA / UNIT 1

DESCRIPTION

The Fuel Receiving and U1 Casing Cooling Buildings were retrofitted with a single membrane roofing system that is ballasted,(ie. held down with masonry pavers), thus adding additional dead load to the roof.

UFSAR section 3.2, Classification of Structures, indicates that the Casing Cooling Pump House (CCPH) is Seismic Class 1 and non-tornado missile protected. Calculation 11715-DC-12-001, originally designed the roof slab and indicates that a higher dead load was assumed than was actually constructed. Therefore, the actual dead load plus the addition dead load is bounded by this calculation. Note, the Fuel Building is classified non-safety and non-seismic.

SUMMARY OF SAFETY ANALYSIS

An unreviewed safety question did not exist because:

A. The implementation of this DCP did not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety and previously evaluated in the UFSAR because the additional dead load did not exceed the design basis of the CCPH roof. The structural integrity of the roof to resist a seismic event remains intact.

B. The implementation of this DCP did not create a possibility for an accident or a malfunction of a different type than any evaluated previously in the UFSAR because this modification did not over stress the CCPH roof structure. Calculation 11715-DC-12-001, Addendum A, bounds appropriate combinations of dead, live and seismic loads and proves that the roof is structurally sound for the required load combinations.

C. The implementation of this DCP did not reduce the margin of safety as defined in the basis of any Technical Specification because the dead load did not exceed the design capacity of the CCPH roof. The roof will still be structurally capable of carrying the additional dead load and withstanding a seismic event. Therefore, an unreviewed safety question did not exist.

DC 95-116
ROOF UPGRADE, FUEL RECEIVING, U1 CASING COOLING
NORTH ANNA / UNIT 1

DESCRIPTION

The Fuel Receiving and U1 Casing Cooling Buildings were retrofitted with a single membrane roofing system that is ballasted,(ie. held down with masonry pavers), thus adding additional dead load to the roof.

UFSAR section 3.2, Classification of Structures, indicates that the Casing Cooling Pump House (CCPH) is Seismic Class 1 and non-tornado missile protected. Calculation 11715-DC-12-001, originally designed the roof slab and indicates that a higher dead load was assumed than was actually constructed. Therefore, the actual dead load plus the addition dead load is bounded by this calculation. Note, the Fuel Building is classified non-safety and non-seismic.

SUMMARY OF SAFETY ANALYSIS

An unreviewed safety question did not exist because:

A. The implementation of this DCP did not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety and previously evaluated in the UFSAR because the additional dead load did not exceed the design basis of the CCPH roof. The structural integrity of the roof to resist a seismic event remains intact.

B. The implementation of this DCP did not create a possibility for an accident or a malfunction of a different type than any evaluated previously in the UFSAR because this modification did not over stress the CCPH roof structure. Calculation 11715-DC-12-001, Addendum A, bounds appropriate combinations of dead, live and seismic loads and proves that the roof is structurally sound for the required load combinations.

C. The implementation of this DCP did not reduce the margin of safety as defined in the basis of any Technical Specification because the dead load did not exceed the design capacity of the CCPH roof. The roof will still be structurally capable of carrying the additional dead load and withstanding a seismic event. Therefore, an unreviewed safety question did not exist.

DCP 95-117

SECURITY UPS SYSTEM UPGRADE
NORTH ANNA POWER STATION
UNITS #1 & #2

Executive Summary

The existing Exide 30 KVA UPS System, i.e. battery charger and inverter, for the Security System have been completely replaced with a new Solidstate Controls, Inc. (SCI) 10 KVA SE series UPS System. Disconnect switches and a manual bypass transfer switch were installed to allow preventive maintenance to be performed on the Security UPS System without a power interruption to the Security System. Equipment that is not required by Security has been removed from the Security UPS System and the Security UPS System load list was corrected.

SUMMARY OF SAFETY ANALYSIS (95-SE-MOD-023)

This design change did not create an unreviewed safety question as defined in 10 CFR 50.59.

MAJOR ISSUES:

The Security UPS System is classified as a non-safety related system. The new Security UPS System upgrade will be designed in accordance with IEEE-Std-692-1986, section 4. The alternate supply for the Security UPS inverter is provided from plant auxiliary bus 14A3 (1-EP-MCC-1B2-1), breaker E1L, which is located in Unit #1 normal switchgear. 1-EP-MCC-1B2-1 is also classified as non-safety related, but is shown in figure 8.3-6 of the UFSAR and has been revised as a result of this design change. A revision to the UFSAR requires a safety evaluation to be performed in accordance with VPAP-3001.

JUSTIFICATION:

An Electrical System Analysis has been performed for the added 37.5 KVA to 1-EP-MCC-1B2-1. This analysis is shown in Appendix 4-6 of design change and adding 37.5 KVA to 1-EP-MCC-1B2-1 was found to be acceptable. A UFSAR change request is shown in Appendix 4-9 of the design change.

UNREVIEWED SAFETY QUESTION ASSESSMENT:

1) Accident probability has not been increased because this design change conforms to all applicable standards and administrative procedures. The operation of 1-EP-MCC-1B2-1 or the electrical power system has not been affected by this modification.

2) Accident consequences are not increased. The implementation of this modification was performed and controlled using station approved procedures. The operation of the electrical power system was not be affected by this modification. Loss of the Security UPS system or 1-EP-MCC-1B2-1 will not affect the safe shutdown capability of the plant.

3) The implementation of this modification will not create a possibility for an accident or a malfunction of a different type than previously analyzed in the SAR. Loss of the Security UPS system or 1-EP-MCC-1B2-1 will not affect the safe shutdown capability of the plant.

4) The Margin of Safety will not be compromised. The integrity of the Security system and the electrical power system was maintained during the installation of this modification.

DCP 95-120

REPLACE 480V LOAD CENTER BREAKER
NORTH ANNA POWER STATION
UNIT #2

Executive Summary

The existing ITE, model K-600, trip device OD-6 480V load center breaker was replaced with a new Asea Brown Boveri (ABB), K-600S, trip device SS-4 480V load center breaker. The existing ITE K-600 breaker contained a OD-6 electro-mechanical 350 amp overcurrent trip device and the new ABB K-600S breaker contains a Solid State SS-4 600 amp overcurrent trip device.

SUMMARY OF SAFETY ANALYSIS (95-SE-MOD-043)

This design change did not create an unreviewed safety question as defined in 10 CFR 50.59.

MAJOR ISSUES:

During the development of a Setpoint Change calculation for the 480V Safety Related (SR) buses, it was identified that the characteristic of the trip device associated with the pressurizer heater panel feeder (breaker 24H1-6) could lead to the loss of an entire panel from a single heater/circuit failure. The instantaneous unit of the existing ITE type OD-6 trip device cannot be guaranteed to coordinate with the magnetic unit of the molded case circuit breakers in the pressurizer heater panel. For high magnitude currents, a "race" will occur between the tripping units possibly resulting in tripping of both circuit breakers. Since the heater circuits are fused prior to entering containment, the most probable initiating event would be a cable failure/fault outside of containment.

JUSTIFICATION:

Replacement of the ITE breaker and type OD-6 trip device with a solid state SS-4, containing a short-time delay unit, has ensured proper selective tripping.

UNREVIEWED SAFETY QUESTION ASSESSMENT:

1) Accident probability has not been increased because this design change conforms to all applicable standards and administrative procedures. The breaker coordination between 24H1-6 and the pressurizer heater panel feeder

molded circuit breakers has been enhanced.

2) Accident consequences are not increased. The implementation of this modification will be performed and controlled using station approved procedures. The operation of the pressurizer heaters and emergency power supply 24H1 will not be affected by this modification. This modification does not affect the safe shutdown capability of the plant.

3) The implementation of this modification will not create a possibility for an accident or a malfunction of a different type than previously analyzed in the SAR. The breaker coordination between 24H1-6 and the pressurizer heater panel feeder molded circuit breakers has been enhanced.

4) The Margin of Safety will not be compromised. The integrity of the pressurizer heaters and emergency electrical power distribution system will be maintained during the installation of this modification.

DCP 95-128

CHARGING PUMP CASING REPLACEMENT
2-CH-P-1C
NORTH ANNA UNIT 2

DESCRIPTION

The charging pump manufacturer had previously issued a bulletin advising owners of the pumps that had casings constructed of carbon steel, cladded internally with stainless steel to inspect them for cladding cracks, erosion or damage when disassembled. Past inspections of the carbon steel charging pump casings at NAPS discovered indications which were severe enough to warrant casing replacement rather than repair the existing casing. As a result of these inspections, the existing pump casings for 1-CH-P-1A, 2-CH-P-1A & 2-CH-P-1E were replaced with solid stainless steel casings.

Due to the failure rate exhibited by previous inspections, the carbon steel pump casing associated with 2-CH-P-1C was replaced in lieu of performing additional inspections. The replacement stainless steel casing was supplied by the original pump manufacturer, Ingersoll-Dresser Pump Company (Pacific Pumps). The replacement pump casing was superior to the original due to improved corrosion resistance. The new casing met or exceeded all design requirements for the original equipment. All nozzles and connections on the new casing were of the same size and location, so no piping changes were required. The pump internals, which determine the pump's performance characteristics, were reinstalled in the new casing to avoid generating changes to the pumps pressure and flow features. Minor changes to the pump mounting were reviewed and approved by the pump vendor.

SUMMARY OF SAFETY ANALYSIS (95-SE-MOD-014)

The replacement of the charging pump casing did not constitute an unreviewed safety question as defined in 10CFR50.59 since it did not:

A) Increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety and previously evaluated in the UFSAR.

The new casing met or exceeded the design requirements of the original equipment. The operational characteristics of the pump remained the same since the original pump internals were retained for use in the new casing. The HHSI pump continued to perform its intended function for mitigation of applicable accidents. All modifications involved in the casing replacement were external to the pump and in no way affected pump performance or operation.

B) Create a possibility for an accident or malfunction of a different type than any evaluated previously in the UFSAR.

Pump casing replacement was essentially a one-for-one replacement and since the original pump internals were retained for use in the new casing, the pump would continue to operate in the same manner as before the modification was performed. The possibility of creating a different accident or malfunction that wasn't previously evaluated was not credible.

C) Reduce the margin of safety as defined in the basis of any Technical Specification.

Replacement of the pump casing was essentially a one-for-one replacement with minor modifications to external mounting configurations. Pump operation remained unchanged as a result of the design change. The modification had no impact on the Tech Specs nor was the margin of safety affected by this work.

DCP 95-129

2-SI-MOV-2867B REPLACEMENT
NORTH ANNA UNIT 2

DESCRIPTION

The Boron Injection Tank (BIT) had experienced water in-leakage and the inlet isolation valve, 2-SI-MOV-2867B had been identified as the source of this leakage. Activity replaced 2-SI-MOV-2867B with an equivalent valve that had a different stem configuration. The different stem thread geometry required modification to the valve operator to ensure valve stroke time and developed thrust were within acceptable limits. In order to maintain an adequate operational thrust margin, the operator required a motor change in addition to modifying the actuator gear ratio. Since the control circuitry for 2-SI-MOV-2867B was set for limit-limit control, the operator torque switch was set to the maximum setting in order to effectively bypass the torque switch during normal operation and prevent inadvertent switch trip. A different valve packing configuration utilizing a live loading feature was an acceptable design to replace the existing packing arrangement.

The valve was evaluated as an acceptable replacement which met the design requirements of the original equipment. The valve replacement, motor replacement and motor operator modification were reviewed to ensure reliable operation and compliance with GL 89-10. The valve replacement and operator modification was acceptable because the new valve met the original design requirements. Slightly improved pressure drop and flow characteristics associated with the new valve increased valve performance. The existing operator equipped with a larger 25 ft-lb motor and a modified overall gear ratio (OGR) was evaluated to ensure that the valve motor operator was sized to equal or exceed the thrust required to open or close the valve under all conditions that required valve operation. Operation modifications ensured that ESF valve response time requirements were still satisfied. The safety function of the valve was not compromised by the activity. The packing design of the replacement valve did not adversely affect valve operation.

SUMMARY OF SAFETY ANALYSIS (95-SM-MOD-024)

The valve replacement did not constitute an unreviewed safety question as defined in 10CFR50.59 since it did not:

A) Increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety and previously evaluated in the UFSAR.

The replacement valve met or exceeded the design requirements of the original equipment. Since the activity replaced the

valve with an equivalent, the operability and performance capability of the valve and SI system was not compromised. Actuator motor replacement and modification to the OGR ensured reliable operation such that the MOV would develop the required thrust to overcome design basis differential pressure conditions. The valve's pressure boundary, flow and isolation safety function remained unaffected. The ESF response time requirement was satisfied with the modification. The valve's function for accident mitigation was not affected and the valve replacement did not generate or increase accident consequences.

B) Create a possibility for an accident or malfunction of a different type than any evaluated previously in the UFSAR.

The activity involved replacement of 2-SI-MOV-2867B with an equivalent valve which in no way adversely affected operation of the valve or SI system. The valve replacement, motor replacement and operator modification ensured reliable operation and the MOV would develop the required thrust to overcome design basis differential pressure conditions. The different stem configuration changed the valve stroke time but the operator had been modified to ensure that the ESF response time requirement was maintained. The change did not impact the safety function of the valve. Accidents or malfunction of equipment of a different type than was previously evaluated were not credible for this valve replacement.

C) Reduce the margin of safety as defined in the basis of any Technical Specification.

The activity replaced one of the BIT's inlet isolation valves, eliminating water ingress into the BIT and thus reducing the possibility of boron concentration dilution. Tech Specs margin of safety associated with boron injection flowpaths, ECCS operability, BIT operability and MOV thermal overloads were not affected by this activity.

DCP 95-133
REMOVE VOLUME BOOSTERS
2-RC-PCV-2455A/B

DESCRIPTION

Problems have been experienced with the pressurizer spray valves opening spuriously and with the valves failing to fully close after they have partially opened. To help correct this, the obsolete Bailey positioners were replaced with Bailey's recommended replacement and the Fisher volume boosters were removed. The volume boosters affected stroke time of the valves if there was a large, rapid input signal change (sudden demand for full sprays). Removal of the boosters increased this stroke time slightly. If the valves are operating normally, with small demands for spray increases or decreases, the removal of the volume boosters had no impact on valve operation.

SUMMARY OF SAFETY ANALYSIS (95-SE-MOD-026)

Transients and accidents considered were Loss of Electrical Load or Turbine Trip, Loss of Normal FW, Major Rupture of Main FW Pipe, Steam Generator Tube Rupture, and Single RCP Locked Rotor. No previously evaluated malfunctions were applicable to this change.

This design change did not create an unreviewed safety question as defined by 10CFR50.59.

1) Accident probability was not increased as the sprays are used after initiation of a transient or accident and have no role in their initiation.

2) Accident consequences were not affected. No credit was taken for the sprays during an accident. The loss of load transient was analyzed both with and without sprays. In all cases protection of the RCS will be provided by the pressurizer and steam generator safety valves.

3) No unique accident possibilities were created. The spray valves will still operate to maintain pressurizer pressure within prescribed limits.

4) Margin of Safety was maintained because the setpoints for pressurizer pressure were not affected. Nuclear controls will still operate to maintain DNB parameters as prescribed in the Technical Specifications.

DCP 95-135
North Anna Power Station

Reconfigure the Chemical Addition Connection to the Bearing Cooling Header

DESCRIPTION:

A pin hole leak developed in the phosphate chemical addition line connection to the 24" bearing cooling water header, just upstream of 2-BC-346. This short section of carbon steel piping has a history of developing leaks.

This DCP reconfigured the connection of the chemical addition line to the bearing cooling water header. A two inch flange connection with a piece of 1/2" stainless steel pipe passing through it was utilized. The carbon steel gate valve 2-BC-346 was replaced with a stainless steel valve. The new connection design: Provides a continuous stainless steel pathway for the phosphate until it is injected into the bearing cooling water flow, and isolates the carbon steel and stainless steel from each other via a flange gasket kit. This modification was completed on Unit 1 via DCP 94-196.

This activity required the UFSAR figure 10.4-19 to be updated; therefore the facility as described in the safety analysis report required change.

SUMMARY:

The safety analysis report was reviewed. No accidents previously evaluated were identified as being applicable. The work to be preformed is not safety related, is non seismic, and non EQ. This activity will not increase the probability or the consequences of any malfunction. Nor will it create the possibility for an accident or malfunction of a different type than was previously evaluated in the UFSAR. The operating license and technical specification will not require a change as a result of this activity. The margin of safety as defined in the Technical Specifications will not be reduced. The fire protection system will be unaffected by this activity. The activity is benign and will not have an adverse environmental impact. There will be no change in effluents or power level as a result of the proposed change. For these reasons a unreviewed safety question did not exist and this modification was allowed.

DCP 95-146
REPLACE RC FLOW TRANSMITTERS
2-RC-FT-2414/2424

UNIT 2

DE CRIPTION

The reactor coolant flow transmitters, 2-RC-FT-2414/2424, were changed from Foxboro transmitters to Rosemount due to the failure of the original transmitters. The original Kerotest manifolds fit up directly to the Foxboro transmitters and could not be used with the Rosemount so manifold arrangements, fabricated with small Whitey instrument valves were used. This safety evaluation was performed to ensure the identical, safe and reliable operation of the RC system and ensure that this modification would not affect the pressure boundary of the RCS or the reactor trip function of the transmitters. The 5-valve manifolds and the transmitters are qualified to be used in the environment that they are exposed to. The tubing was be connected to the five valve manifold using approved fittings and welds. Thus, the RCS boundary was not affected. The new transmitters send the same flow indication in the MCR, alarms, computers, and low flow trip signal to the RPS as the original transmitters. The replacement transmitters and manifolds met all of the specifications and requirements of the original equipment.

The flow transmitters are part of the reactor protection system. They provide a trip signal when low RCS flow is detected in the related loop. There are three flow transmitters for each loop with trip criteria being a trip signal from two out of the three transmitters. The transmitters are required to meet the requirements of General Design Criteria #21 and 23 which require redundant channels, transmitters to be electrically isolated and physically separated and the loss of power to a transmitter results in a trip signal.

The operability of the reactor protection system is addressed in Technical Specification 3.3.1.1 which requires an inoperable trip circuit to be placed in the trip condition within 1 hour of discovery. The trips are to be operable in mode 1 with different permissives in effect depending on reactor power levels.

SUMMARY OF SAFETY ANALYSIS (95-SE-MOD-025)

The accidents considered to be applicable for the transmitter replacements were partial loss of reactor coolant, station black out, loss of forced reactor coolant flow, and single RCP locked rotor. These are the accidents which may result in a trip from low RCS flow. A small break LOCA was also considered as the safety

function of the valve manifold is as a system pressure boundary.

This design change did not create an unreviewed safety question as defined by 10CFR50.59.

1) Accident probability was not increased because the new transmitters and manifolds did not increase the chances of occurrence of any of the accidents considered. The transmitters detect low flow conditions in order to mitigate an accident by tripping the reactor and do not affect the probability for an accident. The manifold was installed in accordance with all applicable specifications and the probability for leakage was not changed.

2) Accident consequences were not affected. The new transmitters met all of the specifications, requirements, codes and general design criteria which were required for the original. They operate in the same manner to ensure that the reactor trips in the event of an accident in order to mitigate an analyzed Design Basis Event.

3) No unique accident possibilities were created. The transmitter replacements and 5-valve manifold arrangements did not affect the operation of the RCS or RPS. System design bases were unchanged.

4) Margin of Safety was maintained because the integrity and reliability of the systems, RCS and RPS, that the transmitters and 5-valve manifold arrangements serve was unchanged.

DCP 95-158
2-SI-100 Valve Replacement
NAPS Unit 2

Description

Swing check valve 2-SI-100 was defective and replaced. The Velan valve was replaced with a Westinghouse valve that met the requirements of the original equipment. The new valve was relocated approximately 16" to permit automatic welding equipment to be used to expedite the modification in a high radiation area.

Summary of Safety Evaluation (95-SE-MOD-028)

LHSI swing check valve 2-SI-100 was defective and could not be repaired. The Velan valve was replaced with a Westinghouse valve that met or exceeded the requirements of the original equipment. The new valve was relocated approximately 16" to permit automatic welding equipment to be used to expedite the modification in a high radiation area. The replacement valve was evaluated as an acceptable replacement which met the design requirements of the original equipment. Improved pressure drop and flow characteristics associated with the new valve will increase valve performance. The safety function of the valve has not been compromised by this modification.

The valve replacement did not constitute an unreviewed safety question as defined in 10CFR50.59 because it did not:

A) Increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety and previously evaluated in the UFSAR.

The replacement valve met or exceeded the design requirements of the original equipment. Since the activity replaced the valve with an equivalent, the operability and performance capability of the valve and SI system was not compromised. The valve's pressure boundary, flow and backflow isolation function remained unaffected. The valve's function for accident mitigation was not affected. The valve replacement did not generate or increase accident consequences.

B) Create a possibility for an accident or malfunction of a different type than any evaluated previously in the UFSAR.

The activity involved replacement of 2-SI-100 with an equivalent valve which has in no way adversely affected operation of the valve or SI system. The valve replacement has ensured reliable operation of the SI system. This change has not impacted the safety function of the valve. Accidents or malfunctions of equipment of a different type than was previously evaluated were not credible for this valve replacement.

C) Reduce the margin of safety as defined in the basis of any Technical Specification.

The operation , function and performance of the SI system has not been impacted as a result of this modification. The lower L/D for the new valve will result in improved flow with a reduced pressure drop through the valve

DCP 95-159
North Anna Power Station

Install Strainer (N2 to PRT) Upstream of 2-SI-PCV-200

DESCRIPTION:

Problems were experienced with getting adequate nitrogen flow to the PRT when pressurizing the tank for RCS drain down. It was taking hours to raise the pressure in the tank just a couple of pounds. An inspection of 2-SI-PCV-200 revealed rust, scale, and debris clogging the valve's 5/64" orifice. The nitrogen supply to the Pressure Relief Tank (PRT) is at the very end of the line and is essentially a continuation of the main nitrogen header. This arrangement is conducive to the small orifice being a debris trap (Ref: DR N-95-0461). In addition the old configuration did not allow for in line pressure setting of 2-SI-PCV-200. A strainer 2-SI-YS-200 was installed prior to 2-SI-PCV-200 to prevent rust and debris from clogging the pressure control valve. An isolation valve 2-SI-457 was installed prior to the strainer to facilitate cleaning the strainer screen. In addition, a pressure gage 2-SI-PI-2945 with an isolation valve 2-SI-456 was installed between 2-SI-PCV-200 and 2-SI-REG-2002 to allow adjustment of the pressure control valve.

The valve 2-RC-HCV-2550 could not be removed easily, as the support bracket was tight against the tapered configuration of the valve's actuator. A bolted configuration was provided in the pipe support to 2-RC-HCV-2550 to facilitate removal of the valve in future outages.

During normal plant operation there is no flow in the nitrogen line to the PRT as it is at a steady state condition with a constant pressure being maintained in the PRT. There is a check valve on the inlet to the PRT so that pressure in the PRT would be maintained even if the supply line prior to the check valve were to fail. The function of the nitrogen to the PRT is to inert any H_2 which may come out of solution. During outages, flow occurs when the nitrogen is used to provide pressure during RCS drain down and during repressurization of the PRT. This is when rust could be carried in the line and the strainer could become clogged. This is easily detectable by the reduced flow and the strainer would be cleaned prior to return of the unit to operation.

All previously evaluated accidents were reviewed and none were found to address the nitrogen supply to the PRT. Consideration was given to the function of the nitrogen in the event of an overpressurization event.

SUMMARY:

UNREVIEWED SAFETY QUESTION ASSESSMENT

1) Accident probability will not be increased as the nitrogen is used to mitigate the consequences of an overpressurization by inerting any H_2 which may come out of solution and does not affect accident probability.

2) Accident consequences are not affected. Nitrogen supply to the PRT will not be affected. Nitrogen will still be available to inert any H_2 which may be in the PRT.

3) No unique accident possibilities are created. The design function and operation of the nitrogen line to the PRT is not affected.

4) Margin of Safety is maintained because there are none which apply to the nitrogen supply to the PRT or for the PRT itself.

DCP 95-163
RELOCATION OF APPENDIX R DUCTING IN AUXILIARY BUILDING
NAPS UNITS 1&2

DESCRIPTION

Thermo-Lag concerns were identified by the NRC in NRC Bulletin 92-01, NRC Bulletin 92-01 Supplement 1, Generic Letter 92-08, and a December 28, 1994 letter. Thermo-Lag is used for a 1-hour fire rated enclosure around the Appendix R flexible ventilation ducting on elevation 259'-6" of the Auxiliary Building. This enclosure is located around the steel duct outlets and it is used as the storage area for flexible ducting that is routed to areas requiring ventilation in the event ventilation is impacted by a Auxiliary Building fire. The existing enclosure will remain in place.

Appendix R flexible ducting is being relocated from a Thermo-Lag enclosure in the Auxiliary Building to a separate fire area in the Auxiliary Building Stairwell. The flexible duct is being relocated to eliminate the reliance on Thermo-Lag for a rated fire enclosure because of concerns for Thermo-Lag and the associated regulatory issues.

Flexible ducting will be stored in metal enclosures in the bottom of the Auxiliary Building stairwell. The enclosures have wheels so that the enclosures could be moved to elev. 259'-6" to the metal duct outlet using the elevator, or the flexible duct could be hand carried up the steps. Connection to the fixed metal duct outlets will use connecting bands with a hand operated lever that does not require tools for attachment of the flexible duct to the metal duct outlets. No new flexible ducting is being provided. A cap will be provided in the locker to be placed over the 18" duct outlet to maximize flow to the charging pumps prior to connection of the penetration area flexible duct. A spare cap will be left on the 18" outlet.

DCP 95-164
95-SE-MOD-34

INCORE FLUX THIMBLE MODIFICATION
NORTH ANNA POWER STATION
UNIT 2

EXECUTIVE SUMMARY

Eddy Current Examination of the Incore Flux Detection Thimbles identified a thimble that required retraction due to tube wall thinning. Thimble B07 was retracted approximately 2" and the retracted piece was cut off to provide a new wear surface for the thimble. Removal of a small portion of the thimble has no adverse impact on the operation of the Incore Flux Detection System since the original thimbles can accommodate retraction of up to 16 inches. Unit 2 thimble B07 was retracted and cut off using maintenance procedure MP2.3.1 VGB-2, "Seal Table Repair at North Anna."

SUMMARY OF SAFETY ANALYSIS (95-SE-MOD-34)

The Incore Flux Detector Thimbles provide a path for inserting the miniature fission chambers into the reactor for flux mapping. The thimble serves as a pressure boundary between reactor coolant and containment atmosphere. Because of flow induced vibration causing wear of the thimble at the bottom of the reactor, the thimble was retracted to provide a new wear surface on the thimble. By varying the inserted position of the thimble tube over several cycles, this wear is distributed over a larger area and is less likely to reach through-wall and result in pressure boundary leakage. If a leak were to occur, an operator would receive considerable dose while isolating the tube locally at the seal table. If the thimble tube was not isolated or the leak is large enough, the leakage could result in flooding of a ten-path transfer device, rendering a large portion of the Incore Monitoring System inoperable.

The activity was performed during a unit outage when incore flux monitoring was neither required nor possible. This activity was performed under appropriate RWP's. The work should save occupational exposure in the long run by reducing the potential of through-wall wear, which would necessitate manual isolation to stop a LOCA.

Small changes in the fully inserted position of an incore flux monitoring thimble tube will have a negligible effect on the axial flux profile since the upper end of the fuel assemblies are in a low flux region.

This DCP preserved the monitoring and control capabilities of the Incore Monitoring System by preventing through-wall fretting wear. Failure to have implemented this DCP may result in substantial degradation of the monitoring capabilities since an

entire ten-path device may flood.

In summary, reducing localized fretting wear of the incore thimble tubes reduces the probability of a small break LOCA since the tubes will not experience through-wall wear and result in RCS leakage. The consequences of an accident are unaffected since the operability of the Incore Flux Monitoring System has no effect on the outcome of a LOCA. No unique accident possibilities are created since the Incore Flux Monitoring System is typically inactive and does not physically interact with other systems to create any other type of accident. Operability and reliability of the system were enhanced which maintained the Margin of Safety. Therefore, it is concluded that the above Incore Flux Monitoring System modification did not result in an unreviewed safety question.

DCP 95-165
INSTALL THRUST BEARINGS
2-SI-MOV-2860A/B
NAPS UNIT 2

DESCRIPTION

During Coefficient of Friction (COF) testing on MOV's, the low head safety injection (LHSI) pump suction valves from the containment sump were found to have a COF of .35. This was the maximum allowable COF for these valves. These valves are equipped with 49' reach rods which connect to a motor operator. At the valve, there is a splined connection to the reach rod and the valve is operated through the use of a drive sleeve arrangement. In the original arrangement, the drive sleeve rode on thrust plates to move the valve stem. It was determined that the COF could be reduced by installing needle thrust bearings at the drive sleeve, thrust plate sliding surface. The valves are required to open after a LOCA to provide long term cooling from the containment sump. They may be operated either manually from the control room or automatically. The stroke time of the valves was not be affected by this change.

SUMMARY OF SAFETY ANALYSIS

All accidents which require SI actuation were considered, although long term cooling is only required after a LOCA. Other accidents requiring SI actuation are Steam Generator Tube Rupture, Main Seam Line Break and CRDM Rupture. Malfunctions considered were valve malfunction, pipe failure in suction line and LHSI pump failure to start.

This design change did not create an unreviewed safety question as defined by 10CFR50.59.

1) Accident probability was not be increased as the SI system is for accident mitigation and has no role in their initiation.

2) Accident consequences were not affected. The bearing installation did not affect the ability of the valves to operate. Failure of one valve is also addressed by the redundancy of the system as only one train is adequately sized for the maximum postulated LOCA.

3) No unique accident possibilities were created. The valves are only used in an accident situation and can not create the possibility for one of a different type.

4) Margin of Safety was maintained because the modification did not affect the design requirements of the SI system for accident mitigation.

DCP 95-167

SSPS INTERNAL WIRING MODIFICATION
NORTH ANNA POWER STATION
UNIT #2

Executive Summary

The internal wiring in the SSPS was modified to isolate the affected MCR benchboard reset switch returns from fuse 6FV2 and input power to K645 from fuses 6FV1 and connect them to fuses 6FU1 and 6FU2. This eliminated the possibility of having 120 VAC on the output relay cabinet input power fuse block after fuses 6FU1 and 6FU2 have been removed.

SUMMARY OF SAFETY ANALYSIS (95-SE-MOD-036)

This design change did not create an unreviewed safety question as defined in 10 CFR 50.59.

MAJOR ISSUES:

The SSPS Engineered Safety Feature Actuation System Instrumentation is not required to be operable in modes 5 or 6 for any functional unit in accordance with Technical Specification 3/4.3.2.1 and table 3.3-3. This design change will be implemented in mode 6. The SSPS internal wiring problem is associated with the reset function, i.e. unlatch function, of several output slave relays. The unlatching function of the output slave relay allows the operator in the control room to reset an output slave relay after the SSPS has automatically or an operator has manually actuated it. Failure of the reset function will not affect the automatic or manual actuation of an output slave relay. If the 6FV2 fuse was to inadvertently open, the reset function of the associated output slave relay would not operate. The reset function of the associated output relays are only required after an accident event or during PT's. Thus, the internal wiring problem is not an operational issue, but is a personnel safety concern.

JUSTIFICATION:

The SSPS internal wiring modification will not affect the operation of the SSPS. The power wiring, i.e. hot and neutral leads, between 6FU1 and 6FU2 will be isolated from 6FV1 and 6FV2. Thus, the operation of the RHR RCS pressure permissive for the opening of 1/2-RH-MOV-1700/1701/2700/2701 will not be

affected by removing power input fuses 6FU1 and 6FU2. In addition, the reset function, i.e. unlatching function, for the output slave relays will not be affected by this modification. Fuses 6FV1 and 6FU1 and 6FV2 and 6FU2 have the same common tie to the neutral and 120 Vac bus in the SSPS.

The input power to the K645 unlatching coil will be supplied from the same power source after the design change has been implemented. However, it will be supplied input power through 6FU1 rather than 6FV1. Both 6FU1 and 6FV1 are supplied power from the same source. The reason it was originally supplied through 6FV1 could not be determined by Design Engineering. SSPS ESF containment spray actuation (K643, K644 & K645) was added by Westinghouse for NAPS because contaiment operates below atomspheric pressure. The unlatch coils for K643 and K644 are supplied power from 6FU1. This design change will wire the unlatch coils of K643, K644 & K645 as shown on station drawing NA-DW-1082H41, sh 27. To ensure the unlatching function of K645 has not been affected by this design change, the K645 section of the SSPS output slave relay test (1/2-PT-36.5.3A/B) will be performed by the I&C department. This test will energize the slave coil of K645 and unlatch (reset) it by energizing the unlatching coil.

UNREVIEWED SAFETY QUESTION ASSESSMENT:

1) Accident probability has not been increased because this design change conforms to all applicable standards and administrative procedures. The operation of SSPS will remain the same. The reset switches in the MCR will receive a path to neutral through 6FU1 instead of 6FV1. The neutral ties for 6FU1 and 6FV1 are identical.

2) Accident consequences are not increased. The implementation of this modification will be performed and controlled using station approved procedures. The operation of the SSPS will not be affected by this modification. Loss of the unlatching function of the output slave relays will not affect the safe shutdown capability of the plant.

3) The implementation of this modification will not create a possibility for an accident or a malfunction of a different type than previously analyzed in the SAR. Loss of the unlatching function of the output slave relays will not affect the safe shutdown capability of the plant.

4) The Margin of Safety will not be compromised. The integrity of the SSPS system will be maintained during the installation of this modification.

DCP 95-168
Elimination of Secondary Boric Acid Treatment
NAPS Unit 2

Description

The purpose of the Design Change Package was to document the changes required by the termination of the secondary system boric acid feed to the steam generators and the increased upper pH limit change on the secondary feedwater. The change eliminated secondary side boric acid treatment on North Anna Unit 1 and Unit 2 and increased the secondary side pH upper limit on North Anna Units 1 and 2 from 9.6 to 10.5. No physical modification to plant structures, systems or components were required to implement the change. Existing boric acid feed equipment will continue to be used to add other secondary system treatments such as hydrazine and ethanolamine.

Summary of Safety Evaluation (95-SE-MOD-42)

A) Secondary System Boric Acid Elimination

Boric acid treatment was recommended by Westinghouse for the old steam generators to minimize tube denting by corrosion of the carbon steel tube support plate annuli. It has also been used in some steam generators to minimize outside diameter stress corrosion cracking on Inconel 600 tubes that are not thermally treated. The major issue associated with eliminating secondary boric acid is steam generator material integrity. With replacement of the North Anna Unit 2 steam generators, the addition of boric acid to the secondary system is no longer required. The replacement steam generators contain stainless steel tube support plates and Inconel 690 tubing which are highly resistant to corrosion. Inconel 690 has also demonstrated excellent resistance for alkaline induced stress corrosion cracking in laboratory testing. The change is also recommended by Westinghouse.

Eliminating boric acid does not degrade other secondary system materials. The materials were originally selected based on a chemical treatment scheme which did not include boric acid. The slight secondary system pH increase expected from the elimination of boric acid is bounded by the evaluation of the increase in the pH upper limit discussed below

B) Secondary System pH Increase

The Nuclear Plant Chemistry Manual limited secondary system pH to a range of 8.0 to 9.6. The boric acid treatment mentioned above buffers the secondary system pH level. Therefore, a slight increase in pH will occur when the boric acid treatment is eliminated. A further pH increase to a normal operating range of 9.8 to 10.2, with occasional excursions allowed up to 10.5, will be used to reduce corrosion product transport to the new steam generators and allow the station to achieve the EPRI recommended hydrazine concentration of greater than 100 ppb. Consequently an increase in the pH upper limit will enable the station to accommodate ammonia increases and thereby optimize secondary system chemistry.

Westinghouse evaluated the potential for elevated pH to degrade the materials in the steam generators and the turbine and endorses the change. Operational feedwater pH values in the vicinity of 10 will significantly aid in the reduction of corrosion product transport and sludge accumulation in the steam generators. Those factors, in combination with the materials used in the new steam generators, will improve steam generator integrity.

There are two alloys in the turbine assembly which are potentially susceptible to high pH because they contain small amounts of copper. As a precaution, Westinghouse (Turbine Division) recommends a photographic record of a representative sample of L-0 and L-1 blades be used to monitor the condition of the stellite strip brazing alloy. Westinghouse was requested to add this to their turbine action item list for tracking during future maintenance periods.

The balance of secondary system materials were evaluated by Corporate Materials Engineering and Corporate Mechanical Engineering. The potential for elevated pH to affect general corrosion, environmental cracking, fatigue and wear/flow assisted corrosion of metals was considered. High pH will increase the general corrosion rate of amphoteric (copper, copper based alloys and brazing) materials, however their use in the system is limited by design. No new degradation mechanisms are created by the change. Periodic excursions above a pH of 10.2 are not expected to have any significant impact on the performance of any system material. Continued operation near the upper pH limit of 10.5 will be restricted by the Nuclear Plant Chemistry Manual.

Flow assisted corrosion will be reduced by elevated pH. The relationship between piping wear rates and previous

increases in secondary pH is trended under the Secondary Piping and Component Inspection Program. Based on models of the most susceptible secondary system piping using the EPRI computer code CHECKWORKS, the trend indicates decreasing degradation due to flow assisted corrosion.

The pH increase will not affect the safety related functions or performance of the Main Steam, Steam Dump, Feedwater Control or Auxiliary Feedwater systems. The expected reduction in system corrosion and corrosion product transport will reduce the potential for valve malfunctions, fouling of instrumentation or pressure boundary failures.

DCP 95-169

ACTUATOR GEAR REPLACEMENT
2-RS-MOV-201A/B
UNIT 2

DESCRIPTION

During Coefficient of Friction (COF) testing on MOV's, the normally open casing cooling pump discharge MOVs (RS-MOV-201A/B) were found to be overthrusting due to inertial effects. The motor pinion gear and actuator wormshaft gear were replaced to increase the overall gear ration (OGR) of the valves. This reduced valve stem speed and, thus, reduced inertial effects. The stroke time for the valves was increased from 30 seconds to 45 seconds.

The valves are in series with the normally closed casing cooling pump discharge MOVs (200). When a CDA signal is received, the RS-MOV-200 valves open. 2-RS-MOV-200A/B will close on low casing cooling tank level. 2-RS-MOV-201A/B are closed if the other valves fail to close, either manually or automatically on low casing cooling pump flow. Containment isolation is maintained in each train by one of the MOV's and the pump discharge check valve. The increased valve stroke time did not affect containment isolation even if one of the 2-RS-MOV-200 valves failed open because of the check valves.

SUMMARY OF SAFETY ANALYSIS (95-SE-MOD-037)

Accidents which require CDA actuation were considered. These were Loss of Coolant Accident and Main Steam Line Break. These two accidents envelop the consequences of other accident (rod ejection, FW line break). Malfunctions considered were loss of electrical power to the MOVs and failure of the RS pumps to operate.

This design change did not create an unreviewed safety question as defined by 10CFR50.59.

1) Accident and malfunction probability were not increased as the RS system is for accident mitigation and has no role in their initiation.

2) Accident and malfunction consequences were not affected. The MOVs are normally open and will receive an open signal when a CDA actuation occurs to ensure this. Each pump has two discharge MOVs in series, with each MOV being connected to different emergency buses. This ensures that one train will operate and that at least one MOV will close for containment isolation. While the MOVs are closing, containment isolation is maintained by the pump discharge check valves.

3) No unique accident and malfunction possibilities were created. The valves are only used in an accident situation and can not create the possibility for one of a different type.

4) Margin of Safety was maintained because the modification did not affect the design requirements of the RS system for accident mitigation.

DCP 95-170

ACTUATOR GEAR REPLACEMENT FOR 2-RH-MOV-2720A/B
NORTH ANNA UNIT 2

DESCRIPTION

Operation of 2-RH-MOV-2720A/B was unacceptable due the failure of Coefficient of Friction testing. The inertial effects exhibited by 2-RH-MOV-2720A/B were reduced by modifying the valve actuator's overall gear ratio (OGR). The MOVs OGR was increased by replacing the motor pinion gear and actuator wormshaft gear. An increase in the OGR reduced the valve stem speed and thus reduced the inertial effects. This reduced the final thrust experienced by the valves. The gear change also increased the motor operator torque capability and increased the overall valve stroke time.

The operator modification was acceptable because operation and function of the valves was not compromised. Increasing the OGR reduced valve stem speed and increased valve stroke time. However, there were no maximum stroke time requirements for these valves other than those for the Inservice Test Program (IST). Since the valves were not subject to any Tech Spec stroke time or ESF response time requirements, the increased stroke time was not critical to valve operation. The safety function of the valves was not be compromised by this activity.

SUMMARY OF SAFETY ANALYSIS (95-SE-MOD-035)

The OGR change did not constitute an unreviewed safety question as defined in 10CFR50.59 since it did not:

A) Increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety and previously evaluated in the UFSAR.

The activity changed the operator OGR for 2-RH-MOV-2720A/B by replacing the motor pinion and actuator worm shaft gear with ones that met the design requirements of the original equipment. Increased stroke time did not affect the safety function of the valves. Operability and performance capability of the valves and RHR system would not be adversely affected. The existing operator and modified OGR were reviewed to ensure that it was sized to ensure adequate operation to close the valve under all conditions that require valve operation. The status of the RHR system for accident recovery and cooldown would not be impacted by this activity. Since the valve did not function for accident mitigation, the increased stroke time would not generate or increase the consequences of the above accidents. The consequences of currently analyzed FSAR accidents remained fully bounding.

B) Create a possibility for an accident or malfunction of a different type than any evaluated previously in the UFSAR.

Loss of RHR due to MOV failure as a result of modification to the valve operator would not create a different type of accident that had not been previously analyzed. The activity involved operator gear replacements with manufactured qualified gears which would in no way adversely affect operation of the valve or RHR system. Different types or modes of valve failure that lead to loss of RHR would not be generated by this activity. The activity did not create any accident scenario that had not been previously analyzed. Therefore there was no possibility of generating a different type of accident than previously evaluated. RHR outlet MOV isolation valve redundancy ensured reliable system operation. No Tech Spec stroke time or ESF response time requirements existed for 2-RH-MOV-2720A/B so increasing the valves' stroke time did not adversely affect system operation.

C) Reduce the margin of safety as defined in the basis of any Technical Specification.

Tech Specs associated with RHR operability and normal water level were not affected by this modification. Modification was performed during fuel off-load. RCS leakage limit for RHR isolation valves did not change since the valves continued to seat with a developed thrust necessary to ensure isolation. No change to the Tech Spec limit was therefore required. Evaluation of the existing thermal overload (TOL) settings determined that the increased stroke times would not nullify the margin of safety currently established for TOL protection. The Tech Specs margin of safety associated with the activity would not be impacted.

DCP 95-175

ACTUATOR MODIFICATION FOR 2-SI-MOV-2890A
NORTH ANNA UNIT 2

DESCRIPTION

2-SI-MOV-2890A was found to have excessive inertia during VOTES testing after stem cleaning and lubrication. The MOV could not be torque-closed without exceeding the actuator or valve thrust ratings. The modification was necessary to reduce the inertial forces in the geartrain that caused the valve to exceed the allowable thrust bands after the control switch was tripped. The inertial effects which were exhibited by 2-SI-MOV-2890A were reduced by modifying the existing actuator. A Limitorque SB, single compensating spring conversion kit was installed on the existing Limitorque SMB-2 actuator, effectively converting the actuator into a SB-2 unit. Addition of the compensating spring conversion unit was intended to minimize the effect of both motor starter "lag" time and inertia thrust which resulted in overthrust conditions for the MOV.

The operator modification was acceptable because operation and function of the valves was not compromised. Valve stroke time and thermal overload (TOL) protection was not affected by the modification. The ability to develop the required seating thrust to ensure adequate valve isolation functions was not compromised. Changes to thrust band setpoints were not required. The safety function of the valve was not affected by this activity.

SUMMARY OF SAFETY ANALYSIS (95-SE-MOD-039)

The modification did not constitute an unreviewed safety question as defined in 10CFR50.59 since it did not:

A) Increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety and previously evaluated in the UFSAR.

The activity modified the operator for 2-SI-MOV-2890A by adding the spring compensator to the existing valve operator. Valve operational characteristics remained the same. Operability and performance capability of the valve and SI system were not adversely affected. Actuator modification ensured reliable operation for the MOV to develop the necessary thrust for all operating conditions without creating an overthrust condition that could jeopardize valve/operator integrity. The probability of occurrence of analyzed accidents were not increased by this modification. The valve's flow, pressure boundary and isolation safety functions were unaffected by the modification and the status of the SI

system for accident mitigation was not impacted by this activity. The normally closed valve was opened as necessary by the control room operator after an accident had occurred. The consequences of currently analyzed FSAR accidents remained fully bounded.

B) Create a possibility for an accident or malfunction of a different type than any evaluated previously in the UFSAR.

The activity involved a modification to the valve operator to ensure proper operation of the valve and eliminated the possibility of valve/operator failure due to an overthrust condition. The valve continued to perform as before the modification. The activity did not create any accident scenario that had not been previously analyzed. Therefore there was no possibility of generating a different type of accident than previously evaluated. LHSI pump discharge to hot-leg isolation valve redundancy ensured reliable system operation.

C) Reduce the margin of safety as defined in the basis of any Technical Specification.

Tech Specs associated with ECCS operability, containment isolation valve status and RCS pressure isolation valve status were not affected by this activity. No change to the operator motor or valve stroke times were involved with this design change and thermal overload (TOL) protection was not be affected. Tech Specs TOL applicability was not impacted by this design change. The Tech Specs margin of safety associated with activity were not impacted.

DCP 95-178
Actuator Gear Change for 2-CH-MOV-2373
NAPS Unit 2

Description

As part of the 1995 Unit 2 SGR/RFO MOV coefficient of friction (COF) testing/maintenance, valve 2-CH-MOV-2373 required a gear change modification. The purpose for making the gear change is to increase the available actuator output torque. There was insufficient margin of torque being provided by the motor operator on the valve to ensure complete closure of the valve when degraded coefficient of friction and dynamic effects were included. The MOVs overall gear ratio (OGR) was increased by replacing the motor pinion gear and actuator wormshaft gear. The gear change also increased the motor operator torque capability and increased the overall valve stroke time. Modifying the valve actuator's overall gear ratio increased the actuator output torque.

The operator modification is acceptable because operation and function of the valve is not compromised. Increasing the OGR reduced valve stem speed and increased valve stroke time. However, there are no maximum stroke time requirements for this valve. Since the valve is not subject to any Tech Spec stroke time or ESF response time requirements, the increased stroke time is not critical to valve operation. The safety function of the valve is not compromised by this activity.

Summary of Safety Evaluation (95-SE-MOD-040)

The OGR change does not constitute an unreviewed safety question as defined in 10CFR50.59 since it does not:

A) Increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety and previously evaluated in the UFSAR.

Operability and performance capability of the valve and CVCS system is not adversely affected. The existing operator and modified OGR have been reviewed to ensure that it is sized to ensure adequate operation to close the valve under all conditions that require valve operation. The status of the CVCS system for accident recovery and cooldown is not impacted by this activity. Since the valve does not function for accident mitigation, the increased stroke time will not generate or increase the consequences of the above accidents. The

cosequences of currently analyzed FSAR accidents remain fully bounding.

B) Create a possibility for an accident or malfunction of a different type than any evaluated in the UFSAR.

Loss of CVCS due to MOV failure as a result of modification to the valve operator does not create a different type of accident that has not been analyzed. The activity does not create any accident scenario that has not been previously analyzed. Therefore, there is no possibility of generating a different type of accident than previously evaluated. No Tech Spec stroke time or ESF response time requirements exists for 2-CH-MOV-2373 so increasing the valves stroke time does not affect system operation.

C) Reduce the margin of safety as defined in the basis of any Technical Specification.

Modification to the operator does not adversely affect operation or valve function. Since the valve thermal overloads are not affected by the activity, the Tech Specs margin of safety associated with MOV thermal overloads is not impacted by this modification.

DCP 95-179
ACTUATOR GEAR REPLACEMENT
2-RC-MOV-2536
NAPS UNIT 2

DESCRIPTION

Rate of Loading (ROL) effects were considered while performing Coefficient of Friction (COF) testing on MOVs. ROL assumes that the torque required to operate a valve under dynamic system conditions will be greater than that required under static conditions. To ensure adequate torque for PORV block valve, 2-RC-MOV-2536, the motor actuator overall gear ratio (OGR) was changed by replacing the motor pinion gear and the actuator worm shaft gear. This increased the torque available and decreased stem speed. The stroke time for the valve increased from approximately 7 seconds to approximately 12 seconds.

The valve is normally open and is used to isolate the PORV if it does not fully reseat after opening or experiences leakage. If the block valve is closed, it may be required to open as the PORVs may be used under certain conditions to control pressurizer pressure. The increased stroke time did not affect these functions of the block valve.

SUMMARY OF SAFETY ANALYSIS (95-SE-MOD-038)

Transients and accidents which were considered were Loss of Electrical Load or Turbine Trip, Loss of Normal FW, Major Rupture of Main FW Pipe, Steam Generator Tube Rupture, LOCA and Single RCP Locked Rotor. No previously evaluated malfunctions were applicable to this change.

This design change did not create an unreviewed safety question as defined by 10CFR50.59.

1) Accident probability was not increased as the PORVs are used after initiation of a transient or accident to reduce pressurizer pressure and have no role in their initiation. The block valve is used to mitigate the consequences of the PORV failing to reseat.

2) Accident consequences were not affected. The block valve will still operate to allow use of or to isolate the PORV. One PORV and associated safety valves are adequate for pressurizer overpressure protection.

3) No unique accident possibilities were created. Valve operation was not affected as the valve will still function to isolate a leaking PORV or to open and allow use of the PORV.

4) Margin of Safety was maintained because this change did not affect the design requirements of the RC system.

DC 95-193
INSTALL LADDER TO ACCESS UNIT 2 SAFEGUARDS ROOF
NORTH ANNA / UNIT 2

DESCRIPTION

Plant personnel have in the past gained access to the Unit 2 Safeguards Building roof by climbing on the Safeguards Area Ventilation supply duct on the north side of the building. This practice has caused inadvertent damage to the ventilation duct and insulation. A permanent ladder was installed adjacent to the Safeguards Area Ventilation supply duct to eliminate the tendency for climbing on the duct to access the Safeguards Building roof. The ladder extends from elevation 273'-0" to roof elevation 279'-6" and is attached to the exterior north wall of the Unit 2 Safeguards Building.

SUMMARY OF SAFETY ANALYSIS

An unreviewed safety question did not exist because:

A. The implementation of this DCP did not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety and previously evaluated in the UFSAR because structural integrity of the Safeguards Building concrete wall is maintained since no rebar was cut to install the ladder anchor bolts. The exterior ladder is located away from any Safety Related equipment housed inside the Safeguards building that could be damaged by failure of the ladder.

B. The implementation of this DCP did not create a possibility for an accident or a malfunction of a different type than any evaluated previously in the UFSAR because failure of the ladder during an earthquake can not damage any Safety Related equipment or components required for mitigation of a LOCA or MSLB. Structural integrity of the Safeguards Building is maintained to resist an earthquake or tornado missile impact since no rebar was cut to install the ladder anchor bolts. The amount of concrete removed by drilling for installation of the anchor bolts is very small compared to the overall size of the massive 2'-0" thick wall.

C. The implementation of this DCP did not reduce the margin of safety as defined in the basis of any Technical Specification because the exterior ladder is located away from any Safety Related equipment housed inside the Safeguards building that could be damaged by failure of the ladder. Safety Related equipment housed inside the Safeguards building remains Operable as required by Technical Specification.

VIRGINIA POWER

VPAP-3001 GOV 02

1. Safety Evaluation Number	2. Applicable Station	3. Applicable Unit
95-SE-ST-04	[X] North Anna Power Station [ ] Surry Power Station	[ ] Unit 1 [X] Unit 2 [ ] Unit 1 [ ] Unit 2

PART A - Resolution Summary Report

4. List the governing documents for which this safety evaluation was performed.

Special Test 2-ST-97

5. Summarize the change, test, or experiment evaluated.

The CHEMTRAC system(ABB/CE) will be utilized to quantify the amount of moisture carryover occcuring in Unit 2 using Special Test 2-ST-97. The system entails injecting an enriched Li 6 isotope(non-radioactive) into the feedwater train via an injection pump cart through normal system drain connections and then drawing downstream samples to determine the resultant concentrations. The moisture carryover samples are drawn from the OLCMS sample points.

6. State the purpose for this change, test, or experiment.

The purposee of this test is to quantify the amount of moisture entrained in the steam header.

7. List all limiting conditions and special requirements identified or assumed by this safety analysis. For each item, indicate the formal tracking mechanism that will be used to ensure that these conditions and/or requirements will be met.

1. Unit power must be stable and at 100%. This will be controlled under the Special Test procedure by having the Operator perform 2-PT-24.1, Computer Calorimetric Heat Balance after the tracer is injected and just prior to taking samples.

8. Will the proposed activity/condition result in or constitute an unreviewed safety question, an unreviewed environmental question, a change to the Fire Protection Program that affects the ability of the station to achieve and maintain safe shutdown in the event of a fire, or require a license amendment or Technical Specifications change? [ ] Yes [X] No

9. Preparer Name (Print)	10. Preparer Signature	11. Date
J. C. Temple - System Engineering	[signature]	9-12-95
12. Cognizant Supervisor Name (Print)	13. Cognizant Supervisor Signature	14. Date
R. C. Sturgill	Robert C. Sturgill	10-4-95

15. Disposition

[X] Approved [ ] Disapproved [ ] Approved As Modified [ ] Requires Further Evaluation

16. SNSOC Chairman Signature	17. Date
DATted	10-17-95

Comments

Note: Attach a Copy of Part A, Resolution Summary Report, to the Change/Activity Documentation Package.
Send a Copy of Part A to Licensing for Submittal to the NRC in Accordance With VPAP-2802, Reporting Requirements.
Send a copy of the completed Safety Evaluation to the Independent Review Coordinator (for the MSRC).
Send the completed Safety Evaluation Original to Records Management.
Use "Safety Evaluation, Supplemental Page" Form No. 730928, if additional Space is Needed.

Key: MSRC-Management Safety Review Committee

Part A - Resolution Summary Report

18. Summarize from Part D, Unreviewed Safety Question Determination, the major issues considered; state the reason the change, test, or experiment should be allowed; and state why an unreviewed safety question does or does not exist (a simple conclusion statement is insufficient).

The test involves injecting a measured amount of Li 6 into each Main Feedwater line through drain valves located in the Mechanical Equipment Room of Unit 2. After injection, samples will be drawn at the On Line Chemistry Monitoring Panel to determine if the concentration is sufficient to enable a valid test. Sampling will then be performed for off-site analysis. Li 6 is non volaitile so any of the material in the Main Steam samples has to be carried by actual moisture droplets(or carryover). Thus an actual measurement of total carryover can be obtained.

1. The following issues were considered:

A. Main Steam Line rupture and Main Feed line break. This test only collects data and has no effect on plant protective circuitry. If a steam or a feed line break were to occur, the test creates no restrictions or obstacles to plant protective features. This test will create no adverse chemical condition for the Unit 2 SG's or associated piping(CME N-93-082) nor will there be a significant personnel hazard. In addition to the CME requirements precautions such as whip restraints will be used on the temporary connections to the plant components to prevent injury.

The test only collects samples from normal sample connections and imposes no operational restrictions on plant operation. Should plant conditions require power reduction, the test will be suspended until plant conditions are again stable at full power.

Data will be recorded by the P-250 and GETARS which will not present the possibility of feedback to any protective circuitry. The special test has a provision to reset GETARS after the test to ensure the normal trip function continues to be available.

Therefore, based on the above discussion there is no unreviewed Safety Question since there is no increased likelihood of an unanalyzed event nor is there any impact on the plant margin of safety.

Part B - Applicable References

19. Identify applicable Updated Final Safety Analysis Report (UFSAR) sections.

UFSAR Sections 7.2.2.3.5, 10.3.1, 10.4.3.1, 10.4.3.1 and 15.2.8

20. Identify applicable Technical Specifications sections.

Unit 2 TS section 2.2.1, 3/4.4.5 and EPP sect. 3.1

21. Identify any other references used in this review.

CME N-93-082, Combustion Engineering CHEMTRAC procedures STD-500-010

Part C - Items Considered By This Safety Evaluation

Insert Supplemental Pages if Additional Space is Needed For Explanations. Use "Safety Evaluation, Supplemental Page" Form No. 730928. Double Asterisk (**) Items Answered "Yes" Require Design Authority Approval (on Page 12 of 12).

22. Will the operation of any system or component as described in the Safety Analysis Report be altered? This includes abandonment of equipment or extended periods of equipment out of service. Explain. [ ] Yes ** [X] No

The plant will be operated at normal 100% power. Injection and sampling occur from LMC drain connections and the OLCMS and pose no restriction on normal operation of the Feedwater System. Data collection is from the GETARS and P-250 systems only and causes no loss or degradation of normal protective circuitry, indication or operation. Although Li6 is not addressed by UFSAR table 5.5-4, the CME analysis validates that the test concentrations will not adversely affect SG chemistry and will not enhance corrosion of Feedwater components.

23. Will the activity alter the performance characteristics of any safety related system or component? Action statements, jumpers, and temporary modifications should be reviewed. Explain. [ ] Yes ** [X] No

The performance characteristics of all Feedwater, Main Steam and protective indications will remain as normal. The volume of chemical injected will have no effect on SG level(10 to 20 gallons). No alteration of normal plant control systems occurs.

24. Will the ability of operators to control or monitor the plant be reduced in any way? Explain. [ ] Yes [X] No

The operators will maintain full control of the FW and MS systems and their components throughout the test. The GETARS system will be used to collect data and would already be printing plant data should a transient occur.

25. Is a temporary modification involved? [Commitment 3.2.15] [ ] Yes ** [X] No

Are testing requirements as stated for the temporary modification adequate to ensure operabililty after installation, as well as after removal of the temporary modification? Explain. [ ] Yes [X] N/A [ ] No

Temporary test connections will be used which are installed and removed under control of the Special Test procedure.

Part C - Items Considered By This Safety Evaluation (Continued)

26. Could the proposed activity affect reactivity? Reactivity is affected by such items as: RCS temperature, dilution or flow; boric acid concentrations or volumes; RWST or accumulator boron concentration; main steam flow or instruments that measure main steam flow; main steam pressure; nuclear instrumentation; calorimetric power monitoring; rod control system; fuel, and fuel components. Explain. The Reactor Engineer must approve the explanation for "Yes" answers. [Commitments 3.2.9 & 3.2.14] [ ] Yes [X] No **

This test will occur during 100% power and remain so during the test, no effect on reactivity occurs. This test has no effect on any Primary plant system. Plant monitoring systems are used by this test(GETARS and P-250) but the data collection groups are normal plant monitoring groups.

26A. Reactor Engineer Signature | 26B. Date

27. Will the activity significantly increase the potential for personnel injury or equipment damage? [ ] Yes [X] No

The test involves connection to high energy systems but provides steps to control isolation valves and also provides warnings to slowly pressurize components to prevent rupture. No significant increase of the probability of personnel injury is created by this process.

28. Will the activity create or increase the levels of radiation or airborne radioactivity? [ ] Yes [X] No

Will that change result in a significant unreviewed environmental impact, a significant increase in occupational exposure, or significant change to the dose to operators performing tasks outside the filtered air boundary during a design basis accident (GDC-19)? Explain. The Superintendent-Radiological Protection must approve the explanation for "Yes" answers. [ ] Yes [X] No

All test connections are to normally non-radioactive secondary test collection points(MS and FW header, SG blowdown). The chemical tracer material is non-radioactive so no increase in airborne radioactivity will result from this test.

28A. Superintendent Radiological Protection Signature | 28B. Date

29. Could the activity change or decrease the effectiveness of the emergency plan? Explain. The Coordinator-Emergency Preparedness must approve the explanation for "Yes" answers. [ ] Yes [X] No

The Emergency plan will not be affected.

29A. Emergency Preparedness Coordinator Signature | 29B. Date

30. Will the consequences of failure for this activity affect the ability of systems or components to perform safety functions? Describe the modes and consequences of failure considered during this evaluation. [ ] Yes [X] No

This test involves data collection only and therefore no failure consequences exist in the performance of this test. Loss of data will only require repetition of a selected test step or subsection.

Key RCS-Reactor Coolant System; RWST-Refueling Water Storage Tank

Part C - Items Considered By This Safety Evaluation (Continued)

31A. Will the activity cause equipment to be exposed (or potentially exposed) to adverse conditions, including those created by temperature, pressure, humidity, radiation or meteorological conditions? [Commitment 3.2.11] [ ] Yes [X] No **

31B. If "Yes," could these conditions lead to equipment failure or a dangerous atmosphere? Explain. [ ] Yes [X] No

This test will not cause equipment to be exposed to adverse conditions. All test functions are performed within the design limits of the system and its components. The ABB/CE test components are rated for connection to Main Feedwater piping.

32. Could failure of the activity feed back into protective circuitry? Explain. [ ] Yes [X] No **

Plant operating data is collected from the GETARS and P-250 system. Thus, Feedback to protective circuitry is no more likely than would occur during normal plant operation.

33. Could failure of the activity feed back into control circuits important to stable plant operation (e.g., feedwater control, control rods)? [Commitment 3.2.12] [ ] Yes [X] No **

This test does not involve an electrical jumper or circuitry, only data collection and injection of a chemical tracer in a manually operated drain valve.

34. Could the activity affect emergency diesel generator sequencing logics (including testing logics), or other logics important to safety. [Commitment 3.2.8] [ ] Yes [X] No **

This test does not involve the EDG, its logic testing, electrical jumpers or circuitry.

35. Could the activity cause a loss of separation of instrument channels/trains or electrical power supplies? Explain. [ ] Yes [X] No **

This test does not involve electrical jumpers or circuitry and no instrument channels or power supplies will be affected.

36. Will the activity involve the addition or deletion of any loads on the Class 1E electrical distribution system? Explain. [ ] Yes [X] No **

This test requires no loads to be added or removed.
A small pump will be connected to normal 110V power supply which is not associated with any Safety Related circuits.

Part C - Items Considered By This Safety Evaluation (Continued)

37. Will the activity adversely affect the ability of a system or component to maintain its integrity or code requirements? Explain. [ ] Yes [X] No

Connections to the Feedwater header are through LMC threaded drain connections and will be attached to ABB/CE components rated to withstand full Feedwater header pressure.

38. Will the activity reconfigure, eliminate, or add components and/or piping to the single or two-phase erosion/corrosion piping inspection program? Explain. [ ] Yes [X] No **

All connections are temporary for the short duration of the test. No components are being added or altered which are subject to the erosion-corrosion process.

39. Will additional surveillance requirements, as defined in the Technical Specifications, be necessitated by the activity? Explain. [ ] Yes [X] No

Operating Technical Specifications will be in effect during performance of the test. Since no permanent modifications are being made, no additional surveillance requirements will be required.

40. Will the applicable Technical Specification basis description be altered by the activity? Explain. [ ] Yes [X] No

The Tech Specs and Bases are not affected by the performance of this test. Normal operating Tech. Specs. are not affected by this evolution.

41. Will the activity result in a violation of any Limiting Condition for Operation (LCO), as defined in the Technical Specifications? Explain. [ ] Yes [X] No

Entry into a Tech. Spec. LCO is not required for performance of this test. The Unit will continue to operate at 100% and Tech. Specs. for this mode of operation will apply.

42. Were any other concerns or items identified during this review? If "Yes," explain. [ ] Yes [X] No

No other items or concerns were identified during this review.

Part C - Items Considered By This Safety Evaluation (Continued)

Items 43 through 66 consider potential impacts. VPAP-3001 provides engineering evaluation guidelines for these items.

If the answer to any of the questions for these items is "Yes," a detailed engineering review must be performed. The results of the detailed review should be documented on a supplemental page, identified by this safety evaluation number and Part C Item number.

43. Station Security

Will the activity deactivate a security-related system or breach a security barrier?	[ ] Yes	[X] No

44. Fire Protection/Appendix R

A. Will the activity add or eliminate a significant amount of combustible material from plant areas?	[ ] Yes	[X] No
B. Will the activity change or affect any plant structure or barrier that acts as a fire barrier?	[ ] Yes **	[X] No
C. Will the activity impact the performance of an existing fire protection or detection system?	[ ] Yes	[X] No
D. Will the activity involve modifying any component required for Appendix R, or any Appendix R support system such as emergency lighting or emergency power supplies?	[ ] Yes **	[X] No
E. Will the activity change or affect system flow paths shown on Appendix R flow diagrams (North Anna Power Station - 11715/12050-DAR-Series and Surry Power Station - 11448/11548-DAR-Series)?	[ ] Yes	[X] No
F. Will the activity change station equipment arrangement drawings that show Appendix R equipment (North Anna Power Station - 11715-FAR-Series and Surry Power Station - 11448-FAR-Series)?	[ ] Yes	[X] No

45. Equipment Qualification/Classification

A. Will the activity adversely affect any Class 1E electrical equipment located in a potentially harsh environment (as designated by the Environmental Zone Description)?	[ ] Yes **	[X] No
B. Will the activity have the potential to alter any of the environmental parameters identified in the Environmental Zone Description?	[ ] Yes **	[X] No
C. Will the activity have the potential to affect any of the Class 1E electrical distribution systems (e.g., 4KV, 480V, 120V(AC))?	[ ] Yes **	[X] No
D. Will the activity add, eliminate, or have the potential to affect ASME Section XI equipment adversely?	[ ] Yes	[X] No
E. Will the activity change a setpoint in the Precautions, Limitations, and Setpoints (PLS) Document?	[ ] Yes	[X] No
F. Will the activity adversely affect equipment on the EQML or Q-List?	[ ] Yes	[X] No

46. Seismic

Could the activity be adversely affected by a seismic event, or could the activity affect surrounding equipment during a seismic event?	[ ] Yes **	[X] No

47. Human Factors

A. Will the activity change instrumentation or controls in the Control Room or on the auxiliary shutdown panel?	[ ] Yes	[X] No
B. Will the activity alter the Control Room or the auxiliary shutdown panel?	[ ] Yes	[X] No

Part C - Items Considered By This Safety Evaluation (Continued)		
48. Safety Parameter Display System/Emergency Response Facility		
Will the activity change any of the equipment associated with the SPDS/ERF, including SPDS/ERF computer inputs?	[ ] Yes **	[X] No
49. Station Computers		
Will the activity have a significant potential to modify or add software to station computers?	[ ] Yes **	[X] No
50. Environmental Impact/Flooding		
A. Will the activity impact more than one-fourth of an acre of land, work in navigable waters, wells, dams, or wetlands, and/or involve any wastes or discharges?	[ ] Yes	[X] No
B. Will the activity involve changes to site terrain, features, or structures?	[ ] Yes	[X] No
C. Will the activity have a significant potential to expose safety related equipment to flooding via fluid system equipment/piping malfunction or failure?	[ ] Yes **	[X] No
51. Regulatory Guide 1.97		
Will the activity have a significant potential to modify equipment and/or instrumentation associated with Regulatory Guide 1.97 variables?	[ ] Yes **	[X] No
52. Heating/Ventilation/Air Conditioning		
A. Will the activity have a significant potential to increase the heating or cooling loads in plant areas and/or to plant equipment?	[ ] Yes **	[X] No
B. Will the activity change the existing ventilation system in any way?	[ ] Yes **	[X] No
C. Will the activity change any building walls, ceilings, windows, doors, or floors, in a way that may affect existing HVAC systems?	[ ] Yes **	[X] No
53. Heavy Loads		
Will the activity involve heavy loads (including the transfer of heavy loads in areas housing safety related equipment)?	[ ] Yes	[X] No
54. Materials		
Will detrimental materials be introduced into the containment or other plant areas?	[ ] Yes	[X] No
55. As Low As Reasonably Achievable (ALARA)		
Have ALARA concepts been included? (Detailed explanation not required.)	[X] Yes	[ ] No
56. Cumulative Effects		
Will the proposed change adversely impact the current system/component capacities or design performance?	[ ] Yes	[X] No
57. Design Basis Document		
Will the activity change applicable sections of a system design basis document?	[ ] Yes	[X] No
58. Simulator Impact		
If a change to the Control Room or Safe Shutdown Panel is considered, will the change need to be replicated in the simulator?	[ ] Yes	[X] No

Part C - Items Considered By This Safety Evaluation (Continued)	
59. Nuclear Materials Control	
Will the activity result in the procurement of special nuclear materials or change the handling or storage of special nuclear materials?	[ ] Yes [X] No
60. Masonry Block Walls	
Will the activity affect a masonry block wall in any way, either through addition, removal, mounting of equipment, or location of safety related equipment within the vicinity of a block wall?	[ ] Yes [X] No
61. Hazard/Chemical Release	
Will the activity create a potential hazard/chemical release?	[ ] Yes [X] No
62. Labeling	
Will the activity affect station labeling?	[ ] Yes [X] No
63. Management Oversight	
Is Management oversight of infrequent tests or evolutions (as defined by VPAP-0108, Infrequently Conducted or Complex Tests or Evolutions) recommended?	[ ] Yes [X] No
64. Contaminated Non-radioactive Systems	
Will the activity result in a non-radioactive system having a detectable level of contamination?	[ ] Yes [X] No
65. Defeating Equipment or Automatic Safety System Functions	
Will the activity defeat automatic safety functions of any system or equipment?	[ ] Yes [X] No
66. Modifications to Radioactive Waste Systems (IEC 80-18)	
Will the activity involve a change to a radioactive waste system?	[ ] Yes [X] No
Part D.1 - Unreviewed Safety Question Determination	
Part D is based on the results of the items considered in Part C. Part C must be completed first.	
67. Which accidents previously evaluated in the Safety Analysis Report were considered? All chapter 15 events were considered, particularly MSLB and Feedwater line rupture.	
A. Could the activity increase the probability of occurrence for the accidents identified above? State the basis for this conclusion. The probability of an accident is not increased for performance of this test because: The test will be performed during normal 100% operation and no alteration of plant indication or control will occur. All connections are to normal plant fittings which are designed for the full test pressures. No plant protective circuitry is disabled or altered.	[ ] Yes [X] No
B. Could the activity increase the consequences of the accidents identified above? State the basis for this conclusion. The consequences of an accident are not affected by performance of this test. The test will be conducted at 100% power and no components which are used for accident mitigation are affected.	[ ] Yes [X] No

Part D.1 - Unreviewed Safety Question Determination (Continued)

C. Could the activity create the possibility for an accident of a different type than was previously evaluated in the Safety Analysis Report? State the basis for this conclusion. [ ] Yes [X] No

The possibility of a new accident is not credible since the systems affected are unchanged from their normal 100% Power condition.

68. What malfunctions of equipment related to safety, previously evaluated in the Safety Analysis Report, were considered?

The test will not impact the MSLB accident since the test equipment will only be tied into the Feedwater system in the Mechanical Eqipment Room. The only affect of this test on a Feedwater Line break is that 3/4" drain lines on each of the Main Feed Lines will be unisolated for a short time during the Chemical injection. The injection skid will pressurize the line slightly above the Feedwater line pressure to inject the tracer material. Injection pressures will remain below the design pressure of the line. The injection point is upstream of the manual Feedwater isolation valves but downstream of the Main Feed Reg. valves.

A. Could the activity increase the probability of occurrence of malfunctions identified above? State the basis for this conclusion. [ ] Yes [X] No

The probability of a malfunction is not increased since the affected systems are operated within their normal design conditions. No permanent change is being made and the tracer material is compatible with the System materials such that no lingering long term affect such as corrosion degradation will occur. The test has been performed numerous times at Nuclear and Fossil Power plants with no adverse affects.

B. Could the activity increase the consequences of the malfunctions identified above? State the basis for this conclusion. [ ] Yes [X] No

Only possible consequence of a malfunction which could be worsened by the test is the Feedwater Line Rupture. This accident is not affected by this test since normal operating valves will be used to tie in the injection pump one at a time. Should a test line fail the manual isolation valve can be closed to stop the leak. Additional personnel safety measures will be used such as whip restraints on the test hose and shielding where possible.

C. Could the activity create the possibility for a malfunction of equipment of a different type than was previously evaluated in the Safety Analysis Report? State the basis for this conclusion. [ ] Yes [X] No

The FW and MS systems will be operated IAW their approved operating procedures. As the equipment is operated within design conditions and by normal operating procedures, a malfunction not previously analyzed is not presented by performing this test.

69. Has the margin of safety of any part of the Technical Specifications as described in the bases section been reduced? Explain. [ ] Yes [X] No

The test involves manipulating manual drain valves in the Feedwater system and removing samples from the Main Steam and Blowdown systems. Since no systems will be operated outside their design bases, the margin of safety is not degraded. The purpose of the test is to verify that the replacement Steam Generators are operating within their guaranteed moisture carryover limits to ensure that problems related to excessive moisture such as steam erosion will not occur.

Part D.1 - Unreviewed Safety Question Determination (Continued)

70. Does the proposed change, test, or experiment require a change to the Operating License or Technical Specifications? Explain. [ ] Yes [X] No

This test will occur during normal Unit operation and does not require any change to the Operating license or the Technical Specifications.

Part D.2 - Operating License Condition Question

71. Does the proposed change adversely affect the ability of the station to achieve and maintain safe shutdown in the event of a fire? Explain. [ ] Yes [X] No

The ability of the station to mitigate the effects of a fire and safely shutdown the unit is unchanged.

Part D.3 - Unreviewed Environmental Question (North Anna)

72. Does the proposed change, test, or experiment involve a significant increase in any adverse environmental impact previously evaluated in the Final Environmental Statement (FES)? State the basis for this conclusion. [ ] Yes [X] No [ ] N/A

This test does not increase the potential for affecting the environment.

73. Does the proposed change, test, or experiment involve a significant change in effluents or power level (in accordance with 10CFR 51.5(b)(2)? State the basis for this conclusion. [ ] Yes [ ] No [X] N/A

This test will not affect station effluents or operating power level.

74. Does the proposed change, test, or experiment involve a matter not previously reviewed and evaluated in the Final Environmental Statement which may have a significant adverse environmental impact? State the basis for this conclusion. [ ] Yes [X] No [ ] N/A

This test will not affect the environment or the EPP.

75. Does the proposed change, test, or experiment involve a change to the Environmental Protection Plan? State the basis for this conclusion. [ ] Yes [X] No [ ] N/A

This test will not affect the environment nor the EPP. A small quantity of Li-6 will be injected in the Feedwater system which will be controlled in accordance with the MSDS for this material. The material is non-radioactive and will be removed over time through the blowdown system which is processed through the liquid waste system. This ensures that the effluent is sampled prior to release in accordance with the station release program such that there is no adverse impact on the environment.

VPAP-3001 GOV 02

[redacted]

Part D.4 - Surry ISFSI	
Questions 76 and 77 Apply Only to 10 CFR 72.48 Safety Evaluations	
76. Does the proposed change, test, or experiment involve a significant unreviewed environmental impact? Explain.	[ ] Yes [ ] No [X] N/A
77. Does the proposed change, test, or experiment involve a significant increase in occupational exposure? State the basis for this conclusion.	[ ] Yes [ ] No [X] N/A

If all responses are "No" or N/A to Questions 67 through 77, the proposed activity may be implemented following SNSOC approval. All related documentation must be retained.

If any response is "Yes" to Questions 67 through 77, the MSRC must review and NRC must approve (which may include a Technical Specifications change or an Operating License Amendment) the change before it is implemented.

78. Reviewer Name (Print)	79. Reviewer Title
B.E STANDLEY	SHIFT TECHNICAL ADVISOR
80. Reviewer Signature	81. Date
B Standley	1 OCT 1995

Concurrence Documentation for ** Items Answered "Yes" in Part C - The Preparer or Reviewer may also concur as Design Authority Reviewer if a member of Nuclear Engineering Services.

82. Design Authority Reviewer Name (Print)	83. Design Authority Reviewer Title
ROBERT C. STURHILL	SUPV. SYSTEM ENGR.
84. Design Authority Reviewer Signature	85. Date
Robert C. Sturhill	10-4-95

Form No. 730916 (Oct 94)

LEVEL 2
CONTROLLED WORKING COPY

VIRGINIA POWER	PROCEDURE NO: 2-ST-97
NORTH ANNA POWER STATION	REVISION NO: 0
PROCEDURE TYPE: SPECIAL TEST	UNIT NO: 2
PROCEDURE TITLE: STEAM GENERATOR MOISTURE CARRYOVER MEASUREMENT USING CHEMTRAC CHEMICAL TRACER METHOD	EFFECTIVE DATE: 10-23-95
	EXPIRATION DATE: (Temporary Procedures Only)

REVISION SUMMARY:

- New procedure
- To provide instruction for performing Moisture Carryover testing for verifying performance of the new Unit 2 Replacement Steam Generators

TEST EXPIRATION DATE: 1-15-96 DAHeal SNSOC

RECOMMENDED APPROVAL (Superintendent Operations):	DATE: 10/17/95
RECOMMENDED APPROVAL (Superintendent Engineering):	DATE: 10/16/95
APPROVAL (SNSOC): DAHeal	DATE: 10-17-95

PROBLEMS ENCOUNTERED: ☐ Yes ☐ No **Note:** If yes, note problems in remarks.

REMARKS: U-2 SPDS INSTIATED TRIP AT 10:25 A.M.
STEP 7.3.5 REQUIRES PROCEDURE SUBMITTED T. NSEL FOR MONITORING REPORT. PER DISCUSSION WITH SNS PERSONNEL ONLY NAPS REQUIREMENT IS FOR ANNUAL REPORT.

TABLE OF CONTENTS

1.0 PURPOSE

To provide instructions for performing Moisture Carryover Testing for verifying performance of the new Unit 2 Replacement Steam Generators.

The following synopsis is designed as an aid to understanding this procedure and is not intended to alter or take the place of the actual purpose, instructions, or text of the procedure itself.

The amount of Moisture Carryover that occurs at 100-percent power will be quantified by injecting a lithium trace element into each Steam Generator's feed header and then analyzing samples drawn from the Main Steam Blowdown and Main Feedwater Lines for the concentration of moisture-transported lithium.

2.0 REFERENCES

2.1 Source Documents

None

2.2 Technical Specifications

2.2.1 Unit 2 Tech Spec 2.2.1

2.3 Technical References

2.3.1 12050-FM-074A, Sheet 1, Feedwater System

2.3.2 12050-FM-070B, Sheets 1, 2, and 3, Main Steam System

2.3.3 ABB Combustion Engineering procedure STD-500-010, Rev 05, Procedure for the Field Operations and Sampling for Steam Generator Moisture Carryover

2.3.4 VPAP-1101, Test Control

2.3.5 2-PT-24.1, Calorimetric Heat Balance (Computer Calculation)

2.4 **Commitment Documents**

None

Init Verif

3.0 INITIAL CONDITIONS

3.1 Unit 2 is at 100-percent Reactor power and has been stable for at least 2 hours.

3.2 The P-250 Computer and GETARS are operational.

4.0 PRECAUTIONS AND LIMITATIONS

4.1 Comply with the following guidelines when marking steps N/A:

- IF the conditional requirements of a step do not require the action to be performed, THEN mark the step N/A.
- IF any other step is marked N/A, THEN have the Test Director (or designee) approve the N/A and submit a Procedure Action Request (PAR).

4.2 The following parameters must be maintained within the specified bands while data is collected at the various power levels. Stability must be maintained for 15 minutes before data is collected.

- Reactor Power: ± 0.5 percent
- RCS T_{avg} ± 1°F
- Steam Generator Level ± 5 percent

4.3 Required data must be recorded as quickly as possible to minimize the effects of changes in process parameters.

4.4 IF this test is suspended, THEN the Initial Conditions in Section 3.0 must be verified before the test is resumed.

4.5 All chemical tracer injection and sampling equipment will be operated by Combustion Engineering (CE) personnel. Connections to Station Service equipment (such as hoses and power supplies) will be performed by Operations or Engineering personnel. Operation of plant component connections (such as valves and circuit breakers) must be performed by Operations personnel.

4.6 IF any personnel join this test after the pre-test briefing has taken place, THEN the Test Director must inform those personnel of the current test status and any immediate precautions or limitations that may affect test performance.

4.7 IF this test has a duration of more than one shift, THEN a thorough and concise shift turnover MUST be performed to maintain test continuity and personnel safety.

4.8 Handle all chemicals used for performance of this Special Test in accordance with the Material Safety Data Sheet (MSDS).

5.0 SPECIAL TOOLS AND EQUIPMENT

- PG and SA connection hoses
- CHEMTRAC Injection and Sampling System (diagrammed on Attachments 1 and 2)

6.0 INSTRUCTIONS

6.1 Moisture Carryover Measurement Preparation

6.1.1 Obtain the Shift Supervisor's permission for this Special Test to proceed.

Signature: _____ Date: 11-1-95
Shift Supervisor

6.1.2 Perform a pre-test briefing as follows:

a. Perform a pre-test briefing and review responsibilities using the, Detailed Pre-Test Briefing Checklist of VPAP-1101, Test Control.

b. Attach the completed briefing checklist to this procedure.

c. Ensure the identity of the Test Director is known at all times to all participants in this Special Test.

6.1.3 Verify GETARS is operational.

STA

6.1.4 Obtain the Shift Technical Advisor's permission to use GETARS for collection of test data.

6.1.5 Establish a P-250 data trend block that includes, as a minimum, the data points listed on Attachment 2 and a sample interval specified by the Test Director.

6.1.6 Align the P-250 trend data output to the log copier.

6.1.7 Verify Unit 2 Condensate Polishing System is secured.

6.1.8 Verify Unit 2 SG Blowdown system is in operation.

6.1.9 Have Chemistry establish approximately 800 ml/minute flow rate at the Unit 2 OLCMS at the following locations and maintain this flow for at least 2 hours:

Chem a. A Main Steam Header

Chem b. B Main Steam Header

Chem c. C Main Steam Header

Chem d. Main Feedwater Header

Chem e. A Steam Generator Blowdown

Chem f. B Steam Generator Blowdown

Chem g. C Steam Generator Blowdown

6.1.10 Establish communications between Unit 2 Control Room, Mechanical Equipment Room Injection Pump, TSC, and the OLCMS.

6.1.11 Collect complete sets of background samples from all three SGs and associated headers from the OLCMS as directed by CE personnel.

6.2 SG Tracer Injection

6.2.1 Set up the CE injection pump in the Mechanical Equipment Room or Turbine Building using Attachment 1 as a guide.

6.2.2 Connect Service Air to injection cart metering pump. 2-SA-1088

6.2.3 Activate the P-250 normal trend block established in Step 6.1.5.

6.2.4 IF injecting the tracer into the A Feedwater line is desired, THEN have an Operator do the following in the Unit 2 Mechanical Equipment Room:

Ops

a. Verify 2-FW-48, A Main Feedwater HDR 2-FW-FCV-2478 Outlet DR, is closed.

Ops

b. Remove the drain cap from 2-FW-48.

Ops

c. Attach the injection pump high-pressure discharge whip to 2-FW-48.

6.2.5 Have CE personnel place the injection cart on line using CE STD-500-010.

* *

WARNING: Opening the FW line drain valves pressurizes the CHEMTRAC sampling and injection equipment with high-temperature, high-pressure Feedwater. A burn and scalding hazard could result in the event of a leak or line rupture. Care must be used in opening the specified FW line drain valves.

* *

Ops

6.2.6 Have an Operator slowly open 2-FW-48.

6.2.7 Initiate tracer injecion into A Feedwater Header as follows:

a. Note the time at the beginning of the test injection.

09:01

≈ 900#
PRESSURE

b. Have CE inject trace element into A Feedwater Header to achieve required concentration in accordance with CE STD-500-010.

Ops

6.2.8 WHEN injection is complete, THEN have an Operator close 2-FW-48 .

6.2.9 Have CE personnel depressurize the injection cart and remove injection hose in accordance with CE STD-500-010.

6.2.10 IF injecting the tracer into the B Feedwater line is desired, THEN have an Operator do the following in the Unit 2 Mechanical Equipment Room:

Ops a. Verify 2-FW-80, B Main Feedwater HDR 2-FW-FCV-2488 Outlet DR, is closed.

Ops b. Remove the drain cap from 2-FW-80.

Ops c. Attach the injection pump high-pressure discharge whip to 2-FW-80.

6.2.11 Have CE personnel place the injection cart on line using CE STD-500-010.

Ops 6.2.12 Have an Operator slowly open 2-FW-80.

6.2.13 Initiate tracer injecion into B Feedwater Header as follows:

a. Note the time at the beginning of the test injection.

09:14

b. Have CE inject trace element into B Feedwater Header to achieve required concentration in accordance with CE STD-500-010.

Ops 6.2.14 WHEN injection is complete, THEN have an Operator close 2-FW-80 .

6.2.15 Have CE personnel depressurize the injection cart and remove injection hose in accordance with CE STD-500-010.

6.2.16 IF injecting the tracer into the C Feedwater line is desired, THEN have an Operator do the following in the Unit 2 Mechanical Equipment Room:

BP Ops a. Verify 2-FW-113, C Main Feedwater HDR 2-FW-FCV-2498 Outlet DR, is closed.

BP Ops b. Remove the drain cap from 2-FW-113.

BP Ops c. Attach the injection pump high-pressure discharge whip to 2-FW-113.

6.2.17 Have CE personnel place the injection cart on line using CE STD-500-010.

BP Ops 6.2.18 Have an Operator slowly open 2-FW-113.

6.2.19 Initiate tracer injecion into C Feedwater Header as follows:

a. Note the time at the beginning of the test injection.

09:26

b. Have CE inject trace element into C Feedwater Header to achieve required concentration in accordance with CE STD-500-010.

BP Ops 6.2.20 WHEN injection is complete, THEN have an Operator close 2-FW-113.

6.2.21 Have CE personnel secure and depressurize the injection cart and remove injection hose in accordance with CE STD-500-010.

6.2.22 WHEN 30 minutes have elapsed since securing injection to the last SG, THEN have Chemistry draw a minimum 125 ml blowdown samples at the A, B, and C SG OLCMS sample points.

16:00

6.2.23 Have Chemistry analyze the 125 ml sample for lithium trace element concentration.

6.2.24 Record the sample lithium concentration result:

A SG lithium concentration: 14 ppb Time: 10:10

B SG lithium concentration: 16 ppb Time: 10:10

C SG lithium concentration: 16 ppb Time: 10:10

N/A

6.2.25 IF any sample result in Step 6.2.24 is less than 5 ppb, THEN repeat steps to re-inject tracer for deficient SGs. Re-initial steps that are repeated.

6.2.26 Have an Operator do the following in the Unit 2 Mechanical Equipment Room:

Ops Ops a. Verify 2-FW-48, A Main Feedwater HDR 2-FW-FCV-2478 Outlet DR, is closed.

Ops Ops b. Verify the injection discharge whip is removed from 2-FW-48.

Ops Ops c. Install the pipe cap on 2-FW-48.

6.2.27 Have an Operator do the following in the Unit 2 Mechanical Equipment Room:

Ops Ops a. Verify 2-FW-80, B Main Feedwater HDR 2-FW-FCV-2488 Outlet DR, is closed.

Ops Ops b. Verify the injection discharge whip is removed from 2-FW-80.

Ops Ops c. Install the pipe cap on 2-FW-80.

6.2.28 Have an Operator do the following in the Unit 2 Mechanical Equipment Room:

Ops Ops a. Verify 2-FW-113, C Main Feedwater HDR 2-FW-FCV-2498 Outlet DR, is closed.

Ops Ops b. Verify the injection discharge whip is removed from 2-FW-113.

Ops Ops c. Install the pipe cap on 2-FW-113.

Ops 6.2.29 Have an Operator perform 2-PT-24.1, Calorimetric Heat Balance (Computer Calculation).

6.2.30 Review the calorimetric and verify Reactor Power is at least 99.8 percent. 99.95%

6.2.31 Simultaneously record the following times to determine the clock differences:

Control Room clock: 10:21:56 WATCH 10:21:00

P-250 clock: 10:22

GETARS clock: 10:21

OLCMS clock 10:22

6.2.32 Do the following:

a. Assign a GETARS pre-trip data sample time of 5 minutes.

b. Trip GETARS.

c. Record the GETARS trip time: 10:30

d. Verify GETARS data printout.

6.2.33 Have CE and Chemistry sample-takers verify they are ready to begin sampling the following OLCMS sample points:

- A, B, and C SG Blowdown
- A, B, and C Main Steam Line
- Unit 2 Feedwater Header

NOTE: Six sample sets will be drawn for each of the 7 locations listed above (42 samples total) with a sample interval of 5 minutes.

6.2.34 Begin drawing moisture carryover samples.

6.2.35 Record the moisture carryover sample times below:

Sample Set	Sample Time OLCMS Clock Time
No. 1	10:38:15
No. 2	10:43:20
No. 3	10:48:15
No. 4	10:53:20
No. 5	10:58:20
No. 6	11:04:20

6.2.36 Secure drawing of moisture carryover samples .

STA

6.2.37 IF other SG carryover data collection is NOT in progress, THEN, after data printout is complete, have the STA reset GETARS .

6.3 Test Recovery

6.3.1 Remove all CE equipment from Unit 2 Mechanical Equipment Room and the Turbine Building and prepare for HP release from the plant.

6.3.2 Assemble all chemical tracer and sample bottles for monitoring as directed by the HP Shift Supervisor.

6.3.3 Have the System Engineer collect all P-250, GETARS, and computer calorimetric printouts and attach to this procedure.

6.3.4 Notify Unit SRO to re-establish Blowdown, Condensate Polishing, and Condensate Makeup.

6.3.5 Have Chemistry re-establish normal OLCMS sample flows.

7.0 FOLLOW-ON

7.1 Acceptance Criteria

Sufficient data has been collected to determine Unit 2 moisture carryover quantitatively.

NOTE: The Final Analysis Report shall be forwarded to the Project Engineer to ensure compliance with Steam Generator specification.

Note in the comment section reference to a System Engineer task item for final review and disposition of the Analysis Report.

7.2 Follow-On Tasks

7.2.1 IF the acceptance in Step 7.1 is met, THEN indicate (√) SAT below.

Test result: ☑ SAT ☐ UNSAT

7.2.2 IF the acceptance in Step 7.1 cannot be met, THEN indicate (√) UNSAT in Step 7.2.1 and do the following:

N/A a. Notify the Shift Supervisor.

N/A b. Notify System Engineering of unsatisfactory condition and record name of person notified.

System Engineer: _____

N/A c. Submit a Deviation Report and record DP number below.

DR No.: _____

N/A d. IF required, THEN submit a Work Request and record WR number below.

WR No.: _____

7.3 **Completion Notification**

7.3.1 Notify the Shift Supervisor that this test is complete.

7.3.2 Record any comments applicable to performance of this Special Test:

Comments: • GETARS WAS TRIPPED SUCCESSFULLY BY THE STA. TEST WAS COMPLETED AND UPON ATTEMPTING TO COMMIT DATA TO TAPE, GETARS DATA WAS LOST.
• MCT 95-7000 WAS ASSIGNED TO INSURE FINAL REPORT IS FORWARDED TO SGR PROJECT ENG. + GREATER THAN .1% MOISTURE CARRYOVER IS ANALYZED (IF RESULTS WERE TO INDICATE GREATER THAN .1%).

Completed: _____ Test Director Date: 11/1/95

7.3.3 Forward this procedure to System Engineering for review.

Comments: ALL PROCEDURE STEPS HAVE BEEN PROPERLY SIGNED; NO DISCREPANCIES FOUND.

Reviewed by: W E Thomas Date: 11/1/95
System Engineer

7.3.4 Forward this procedure to the Supervisor of System Engineering for review.

Comments: None

Reviewed by: [illegible] Date: 11/3/95
Supv of System Engineering

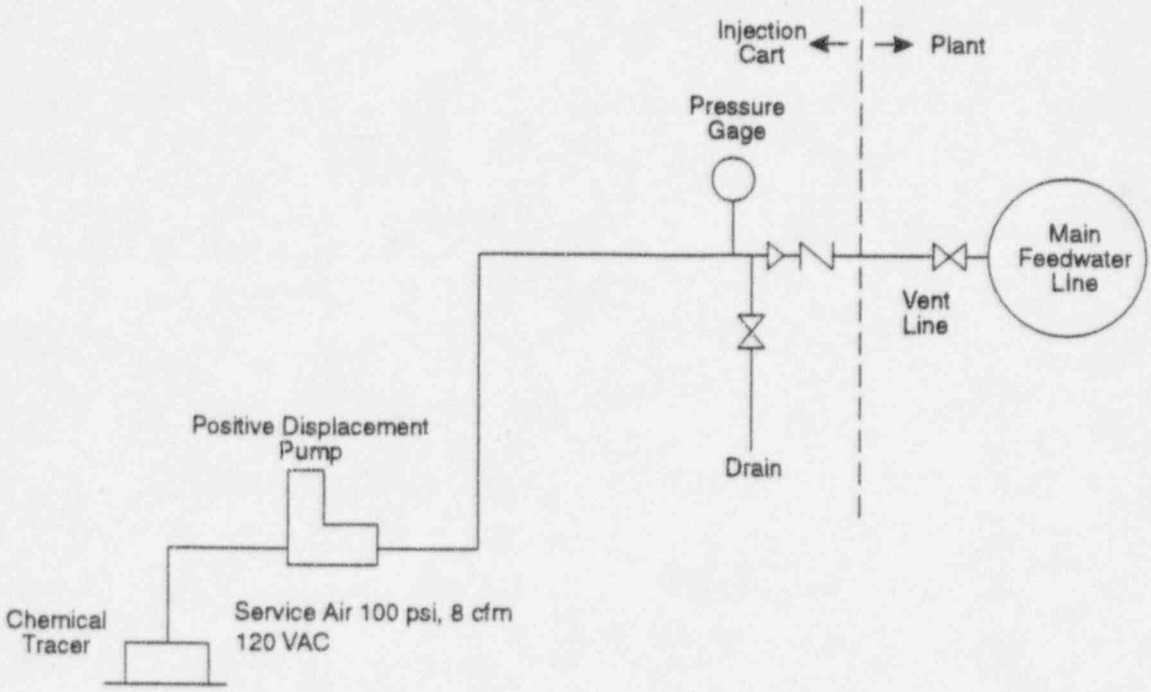
7.3.5 Forward this procedure to NS&L Administrative Assistant for collection of data necessary to complete monthly report.

7.3.6 Forward this procedure to Station Records in accordance with administrative procedures.

ATTACHMENT 1

(Page 1 of 1)

LITHIUM INJECTION SYSTEM



Services Needed at Injection Point

- De-ionized water
- Service Air at 100 psi, 8 cfm
- Drain location

Graphics No: CB2266B

LITHIUM INJECTION

ATTACHMENT 2

(Page 1 of 1)

P-250 DATA COLLECTION PARAMETERS

Description	P-250 Point No.
Steam Flow Protection Ch III – Loop A	F0405A
Steam Flow Protection Ch IV – Loop A	F0406A
Steam Flow Protection Ch III – Loop B	F0425A
Steam Flow Protection Ch IV – Loop B	F0426A
Steam Flow Protection Ch III – Loop C	F0445A
Steam Flow Protection Ch IV – Loop C	F0446A
Steam Pressure CH III – Loop A	P0401A
Steam Pressure CH III – Loop B	P0421A
Steam Pressure CH III – Loop C	P0441A
Steam Pressure CH IV – Loop A	P0402A
Steam Pressure CH IV – Loop B	P0422A
Steam Pressure CH IV – Loop C	P0442A
Turbine First Stage Pressure – CH III	P0398A
Turbine First Stage Pressure – CH IV	P0399A
Steam Generator Level CH III – Loop A	L0402A
Steam Generator Level CH III – Loop B	L0422A
Steam Generator Level CH III – Loop C	L0442A

Safety Evaluation Number
94-SE-OT-027, Rev. 1

Description Of Activity
UFSAR change request to store irradiated components in the Spent Fuel Pool in locations other than spent fuel assemblies or spent fuel rack locations.
Remove irradiated fuel assembly inserts from certain spent fuel storage locations and store irradiated components in the space between the fuel racks and the spent fuel pool wall.

Safety Evaluation Summary
Spent fuel pool storage locations that are under the path used to move the transfer canal gates are prohibited from receiving spent fuel. Tech Specs prohibit movement of any heavy load over irradiated fuel assemblies. These prohibited locations are currently being used to store irradiated fuel inserts (burnable poison rods, control rods, and thimble plugs). A Tech Spec change is being processed that would allow movement of the transfer canal gates over irradiated fuel. If the irradiated inserts are moved to acceptable alternate locations, additional spent fuel cells will be made available.

This proposed activity will be the storage of irradiated components (but not fuel) between the storage racks and the spent fuel pool wall. The irradiated components will be located on the floor of the fuel pool, in available space between the fuel racks and the pool wall. The components stored in these locations will not be re-used in any future core. The design of the spent fuel pool systems, including but not limited to, the cooling system, the storage racks, the pool foundation, and the pool walls, are not affected. The irradiated components do not contribute to the heat load of the pool. Their mass is insignificant by comparison to the mass of the fuel assemblies and storage racks. The irradiated components can not interfere with the flow of water into and out of the cooling system.

Moving irradiated fuel components within the boundaries of the spent fuel pool does not affect overall pool reactivity. Reactivity controls of the spent fuel pool are provided by the Boraflex fuel racks, the fuel rack spacing, and the spent fuel pool boron concentration. These are not affected by this activity.

Safety Evaluation Number
95-SE-JCO-01

Description Of Activity
JCO entitled "Justification for Interim Use of a Station Battery Swing Charger"
This safety evaluation addresses the potential use of a swing charger in the event of loss of a normal charger. The intent is to use the swing charger without entering the Tech Spec LCO (3.8.2.3) while the unit is at power operation.

Safety Evaluation Summary

Major Issues

This safety evaluation addresses the safety impact of interim use of a swing charger until permanent modifications (via an upcoming DCP) are made. (The DCP will correct circuit color-coding deficiencies.) The AC power cables to the swing chargers are coded neutral, which compromises the safety-related functional capability of the swing chargers. Also, other color-coding deficiencies were identified on Unit 2's swing charger DC output cables and have been addressed in the JCO and this safety evaluation. This safety evaluation is performed for a JCO which provides justification for use of a swing charger at unit power operation without entry into the Tech Spec LCO.

Reason for Change to be Allowed

The JCO provides sufficient justification for interim use of a swing charger. This safety evaluation does not identify any adverse impact to plant safety as a result of interim use of a swing charger.

Unreviewed Safety Question

Neither the probability of occurrence nor the consequences of an accident have increased as a result of the proposed use of a swing charger. The emergency DC power system consists of two redundant trains divided into four channels with each channel consisting of a battery, DC switchboard bus, and a charger. The batteries are designed to provide power for a two-hour accident duty cycle. The emergency DC power system is designed to meet the single failure criteria such that a failure of one train (DC Buses 1-I and 1-II or 1-III and 1-IV; similar on Unit 2) will not impair the operability of the other train. Therefore, a failure with use of a swing charger will be limited to a single train, and the opposite train will still be available.

Failures associated with use of a swing charger may result in loss of charger capability to two DC buses. However, one bus will be available without its charger to support its two-hour accident duty cycle and the other two DC buses (fed from the opposite AC train) will also be available with their chargers (which are powered by an EDG) to support their two-hour accident duty cycles. Therefore, a single failure (loss of a DC switchboard) could result in loss of charger capability to another DC bus. The consequences of this failure will be limited to a single train, and the opposite train (i.e., two DC buses) will still be available. The DC system is also designed so that it does not have any automatic connections between trains.

The use of a swing charger does not create the possibility for an accident or malfunction of a different type than any evaluated previously in the SAR.

This activity (use of a swing charger) does not reduce the margin of safety. Use of a swing charger will not adversely affect the ability to achieve and maintain safe shutdown.

Safety Evaluation Number
95-SE-JCO-02

Description Of Activity
JCO 95-02, Rev 0, Excessive Overthrust of 2-CH-MOV-2286B and 2-CH-MOV-2287B
2-CH-P-1B discharge valves 2-CH-MOV-2286B and 2287B, which experienced excessive overthrust while being closed during a maintenance evolution, have been evaluated in Engineering Transmittal CME 95-071, Rev 0. The transmittal recommends that inspection of the valve disc and stem be performed and repaired/replaced if necessary, and evaluates the risks involved if this work is delayed until an outage of suitable length. The ET has determined that the valves are OPERABLE.

Special Conditions
The subject valves will be returned to service. Appropriate administrative controls will be established to require that VOTES testing and evaluation be performed any time these valves are opened during normal, non-emergency conditions, until an outage of sufficient length will accommodate valve inspection, repair, or replacement.

Safety Evaluation Summary
This Safety Evaluation examines the acceptability of returning 2-CH-P-1B discharge isolation valves to service after experiencing excessive overthrust while being closed during a maintenance evolution. See DR N95-1316 and 1318. This event and resolution are summarized in Engineering Transmittal CME 05-071, Rev 0, which is attached. Based on manufacturers recommendations and engineering judgment, the ET prescribes various inspections of the valve and actuator externals and internals, with repairs as needed. Since the valve stem, wedge, and seat can not be examined at power, inspection of these components will be delayed until an outage of sufficient duration. This JCO documents this course of action and requires the compensatory action of VOTES testing of the subject valves when opened under normal, non-emergency conditions. This will allow any possible changes or anomalies to be identified, and ensure that the valves fully reopen. This safety evaluation does not attempt to further quantify the status or OPERABILITY of the valves, it seeks to evaluate the potential impact on plant operation and safety, considering the potential failure modes.

ET CME 95-071 postulates the following potential damage and effect:

Wedge and seat deformation but no damage to underlying base metal. Possible stellite hardfacing cracking, but no separation from base metal. This condition did not affect the valves when VOTES tested, and is likely only to result in increased valve leakage past the seat. Debris intrusion from the cracked stellite is not credible.

Minor stem deformation with no impact on valve operation. Possible cracking along axis of stem due to exceeding yield strength. The cracking, if present is not likely to propagate since the valves do not normally experience any but low thrust cycles. Since maximum imposed loads are experienced at the end of a closed stroke (compressive force-additional crack propagation) and at the start of an opening stroke tensile force-possible stem to wedge separation), it is postulated that if the stem and disc remain intact (and attached) upon fully opening, they will remain attached until the next opening cycle is attempted. Spurious valve closure is not anticipated.

Note that 95-SE-ST-03, approved 8/15/95, evaluated the impact of opening and flowing the subject valves as part of a test for flow blockage. It evaluated the impact of debris upon

downstream components and flow blockage, thus, many of the responses used in that evaluation are applicable to this JCO and will be repeated accordingly, herein.

Safety Evaluation Number
95-SE-JCO-03

Description Of Activity
The spare SW pump is being manufactured at this time. The pump Net Positive Suction Head (NPSH) Required test was performed in accordance with the Hydraulic Institute Standard. Certified test data was received from Johnston Pumps on 11-15-95. The data indicates that the required NPSH at the upper flow range (above 14000 gpm) is higher than minimum available NPSH value of 36.9 for post LOCA conditions as defined in Calculation ME-305. Also this required NPSH is above the required NPSH of Z4 specified in the original Specification NAS-98. Note, that the spare pump is being fabricated per the original Specification NAS-98. The only difference between the new pump and the existing pumps is that the new pump has SS impellers instead of the bronze impellers of the existing pumps. Therefore, the test data is applicable to the existing pumps.

Special Conditions
1. If only two SW pumps are running following a SI/CDA initiation (DBA conditions), SW flow to two out of four operating RSHXs should be isolated to reduce flow on the running pumps. This action should occur after the containment pressure is stabilized at subatmospheric conditions, about one hour, but no longer than two hours after the SI/CDA initiation.

Standing Order No. 213, Rev.0 has been issued to direct operations to implement isolation of two RSHXs after one hour but no longer than two hours after the SI/CDA initiation. The RSHXs which are secured shall be one RSHX associated with one inside RS pump and one RSHX associated with one outside RS pump, if possible, to maintain a full coverage spray pattern.

This Action will justify continued operation of existing SW pumps and in-kind replacement pumps with the NPSH required in accordance with the test results dated 11-09-95. If the new pumps are manufactured with lower than 36.9 required NPSH the above requirement may be eliminated after the installation of all four SW pumps having acceptable NPSH required values.

2. Prior to summer operation (by 3/31/96) procedural controls will be instituted to limit summer time unthrottled CCHX operation after a DBA to two hours on the non-accident unit. This limitation is not expected to deviate from current operating practices, but to prevent a future deviation from current operating practice which could invalidate assumptions used in the Technical Evaluation for this JCO. This should be tracked by the Station CTS system to ensure this action is completed.

3. Prior to declaring a SW pump which has been replaced with either a new or refurbished SW pump OPERABLE, full flow performance test results will be evaluated for Strong / Weak pump interaction. This evaluation is required by the PMT matrix.

Safety Evaluation Summary
The maximum summer load case is based on the flow model results showing that one pump supplies two CCHXs with no throttling of SW to CCHXs that pump flow approaches 15100 gpm. While this is not acceptable for long term operation it is acceptable for two hour period after which SW flow to CCHXs should be throttled. Formal controls had not been required due to inherit margins until this NPSHR issued appeared. During

winter operation this is not a concern; however formal operating controls need to be put in place before summer. These controls will be based on a formal KYPIPE calculation of this 3 pump case scenario and this JCO will be revised when this calculation has been competed. This control will potentially require throttling SW flow to CCHXs not longer than two hours into this event and/or some limitations on summer time unthrottled operation of the CCHXs. The final SW KYPIPE model run will be made for this case. The calculational results can then be used to clearly delineate the operational restrictions and the limitations will be proceduralized by 3/31/96.

Prior to declaring a SW pump which has been replaced with either a new or refurbished SW pump OPERABLE full flow performance test results will be evacuated for Strong / Weak pump interaction. This evaluation will be required by the PMT matrix.

Safety Evaluation Number
95-SE-OT-01

Description Of Activity
UFSAR Change Request # FN 94-039
NAPS UFSAR Table 3.8-12 - Geotechnical Instrumentation Summary, items 2 thru 4 [Rev 26, Page 3.8-158]
NAPS UFSAR 3.8.3.5 - Instrumentation of Dam
DR # N-94-1123, dated 08-19-94.

1) Revise the following UFSAR Table 3.8-12, Lake Anna Main Dam instrumentation monitoring frequencies: a) Drainage Collector System (item # 2) from 2 to 3 months, b) Relief Wells (item # 3) from 2 to 6 months and c) Piezometers (item # 4) from 2 to 6 months 2) Revise footnote (a): NOTE: Monitoring frequencies may be modified as recommended by Engineering evaluation. This includes recommendations resulting from each 18 CFR1 2.37(c)independent consultant report performed every five years.

Safety Evaluation Summary
As noted in Deviation Report # N 94-1123, dated 08-19-94, Main Dam Periodic Test frequencies specified in UFSAR Table 3 8-12, Geotechnical Instrumentation Summary, were found to be inconsistent with those identified in 0-PT-2, 0-PT-3 and 1-PT-6. This inconsistency was attributed to a failure to submit a revision to the UFSAR simultaneous with revisions to 1-PT-2, 1-PT-3 and 1-PT-6, which were SNSOC approved 09-10-90, 09-08-90 and 09-10-90, respectively.

A review of supporting documentation for the above noted revisions indicates that various surveillance frequencies were changed from 2 to 3 and 2 to 6 months, based upon Stone & Webster independent Engineering Report recommendations contained in S&W Initial Safety Inspection Report, NA Hydroelectric Project, dated September 1986 and S&W Second Periodic Inspection Report, North Anna Hydroelectric Project (FERC License no. 6335), dated September 1988, (see revision dated 12-06-91).

In view of this, UFSAR Table 3.8-12, Main Dam instrumentation monitoring frequencies need to be revised, as follows:

- **Drainage Collector System** (item # 2) from 2 months to 3 months
- **Relief Wells** (item # 3) from 2 months to 6 months
- **Piezometers** (item # 4) from 2 months to 6 months

Also, the footnote (a) is being revised to read: **NOTE: Monitoring frequencies may be modified as recommended by Engineering evaluation. This includes recommendations resulting from each 18 CFR12.37(c) independent consultant report performed every five years.**

An unreviewed safety question does not exist because: 1) Change is administrative. (Revised PT monitoring frequencies were SNSOC approved, based on the NES Technical Report # CE-0034, [Type 1 Study # NP 2153], Rev 0, Review of Civil Related PT's, approved 04-20-90. It implemented the recommendation of the independent consultant, Stone & Webster Geotechnical Engineers, who reviewed historical data taken since the reservoir was filled in June 1973.) 2) VPAP 2802, Section 6.17, addresses implementation of 18 CFR 12 Water Power Project Safety (Lake Anna Dam) requirements into the Emergency Action Plan for Lake Anna Dam. 3) The Drainage Collector System continues to monitor seepage flow quantities on a quarterly basis and therefore serves as an indicator

of the need to monitor relief wells and piezometers on a more frequent basis if flow quantities exceed 10 gpm or other trends indicate the need to evaluate the phreatic surface in the embankment. Further, no indication has been provided since the Periodic Test frequencies were revised in 1990 to indicate that the consequences of a accident have increased.

Safety Evaluation Number
95-SE-OT-02

Description Of Activity
TRM Revision 7.
Correct administrative typographical errors in Revision 2 to the TRM that resulted in relocating selected component lists from TS to the TRM and correct noun names associated with the lists in the TRM Tables.

Safety Evaluation Summary
This TRM Revision will correct administrative typographical errors associated with the relocation of selected component lists from Technical Specifications to the TRM as per TS Change #300. Additionally, several labeling errors associated with these lists of components are being corrected by this revision to the TRM.

This TRM Revision does not present an unreviewed safety question based on the following:

1. This revision does not involve any increase in the probability or consequences of any previously evaluated accident. This TRM revision will only correct administrative errors in TRM tables. This revision does not require any modifications to plant hardware or operating practices. Therefore, this revision has no affect on any previously analyzed accidents.

2. This revision does not create the possibility of a new or different kind of accident from any accident previously evaluated. The revision does not affect any operating, maintenance, or surveillance practices; also, there are no design or hardware modifications associated with this revision. Therefore, the possibility of a malfunction or failure that results in a different kind of accident is not created.

3. This revision does not involve a reduction in the margin to safety. This TRM revision will only correct administrative errors in TRM tables and has no impact on the performance of any plant equipment. Therefore, there is not a reduction to any safety margins due to this revision.

Safety Evaluation Number
95-SE-OT-03

Description Of Activity
This safety evaluation describes the basis for changes to Section 12.1 (snubbers) of the Technical Requirements Manual for North Anna Units 1 and 2.

Remove snubbers 1-RC-HSS-825, 1-RC-HSS-113, 1-RC-HSS-114A, 1-RC-HSS-114B, 1-RC-HSS-118 and 2-RC-HSS-104 from Section 12.1 of the Technical Requirements Manual(TRM). Add snubbers 1-MS-HSS-278 and 1-RC-HSS-918. Change the designations of snubber 2-CH-HSS-492-24A to 2-CH-HSS-492-42A and snubber 2-CH-HSS-492-24B to 2-CH-HSS-492-42B.

Safety Evaluation Summary
Section 12.1 (snubbers) of the Technical Requirements Manual is being revised to ensure consistency with the ISI Engineering snubber data base as reflected in surveillance procedures 1/2-PT-79.3, 1/2-PT-79.3(T), and 1/2-PT79.7. Appropriate changes have been made to 1/2-PT-79.7 to ensure a periodic comparison between the TRM and the surveillance PT s. Typographical errors are being corrected in this revision, two snubbers are being eliminated from the TRM listing because they do not exist in the Station, and snubbers are being deleted and added in accordance with DCP 93-019. The snubber listings in the PT s are utilized by the ISI Engineering group to ensure compliance with the requirements of Technical Specification 3/4.7.10 for both units. The TRM has been previously modified without review by the ISI Engineering group, thus creating some inconsistencies between the Manual and the PT s. A link now has been created between these two types of documents by requiring that the ISI engineer compare the contents of the Technical Requirements Manual to the PT s prior to performing the PT s.

Ensuring that the TRM snubber data base accurately reflects the plant design is advantageous, rather than detrimental, for pipe stress and compliance with ASME XI requirements.

Snubber 1-RC-HSS-825 is being eliminated from the Technical Requirements Manual (TRM) because its inclusion in the original TRM is a typographical error. This snubber does not exist in the ISI data base and is not found in the Station.

Snubber 1-MS-HSS-278 was inadvertently deleted form the snubber PT s during a previous revision. A subsequent review of the PT s confirms that no other snubbers have been overlooked in developing the PT tables.

Snubbers 1-RC-HSS-113, 114A, 114B, and 118 were deleted by DCP 93-019 (PSARV Pipe Support Modifications) but were not addressed in the TRM as part of the DCP. Similarly, 1-RC-HSS-918 was added by DCP 93-019, but not included in the TRM. Linkage between the TRM and the DCP process has been addressed by Engineering. Appropriate changes have been made to checklists in VPAP-0301 and STD-GN-0001.

Snubber 2-RC-HSS-104 is being eliminated from the TRM because it does not exist in the ISI snubber data base, or in the Station. Snubbers 2-RC-HSS-492-24A and 2-RC-HSS-492-24B are being changed to 2-RC-HSS-492-42A and 2-RC-HSS-492-42B, respectively, to correct a previous transposition error.

None of these changes involves an unreviewed safety question. Ensuring that the TRM accurately reflects the snubber data base helps ensure that the technical specification

surveillance is performed to verify snubber operability. The proper functioning of snubbers is important to maintaining system integrity. There is no increased probability of occurrence or consequences of an accident or malfunction of equipment important to safety as previously evaluated in the safety analysis report due to ensuring the accuracy of the TRM snubber listing to reflect as-built plant conditions.

The possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report is not created by ensuring that the TRM snubber listing accurately reflects plant design.

The margin of safety as defined in the Technical Specification Bases is not reduced by having the TRM snubber listing revised to include all installed snubbers.

The TRM change does not adversely affect the test frequency, test conditions or testing method utilized to ensure compliance with Technical Specification 3/4.7.10.

Safety Evaluation Number
95-SE-OT-04

Description Of Activity
Request for Temporary License Amendment for Service Water Preservation - Phase II (Repair/Replacement of Exposed SW piping to/from CCHXs to be performed under DCP-94-010).

The SW piping is experiencing internal wall degradation due to general corrosion and relatively rapid wall loss in localized areas (under pits) due to microbiologically influenced corrosion (MIC). Investigation on selected pipes around CCHXs shows in some cases minimum wall thickness below 0.250 inches. In order to perform repair/replacement of the CCHX exposed piping a one time amendment of the TS will be required to allow 49 day isolation of one of the two 24" SW headers to the CCHXs (two periods for 49 days, one for each header). Also NRC approval is required for the temporary crossconnect installations and defeating the automatic closing of SW MOVs to CCHXs on a CDA signal during the time of CCHX operation from one SW header. Availability of four SW Pumps in the pre-accident condition and throttling of the bypass valve will substitute for the automatic closing of SW motor operated valves to the CCHXs of the affected unit on a CDA signal and the required SW flow will be delivered to the RSHXs of the affected unit (Calculation ME-0420).

Safety Evaluation Summary
The SW piping is experiencing internal wall degradation due to general corrosion and relatively rapid wall loss in localized areas (under pits) due to microbiologically influenced corrosion (MIC). Investigation on selected pipes around CCHXs shows in some cases minimum wall thickness below 0.250 inches. Under this Design Change exposed piping within the Auxiliary Building to/from CCHXs will be replaced or repaired.

This DC does not involve an unreviewed safety question:
Repair of the exposed SW lines to/from CCHXs will temporarily limit SW Supply to CCHXs since while one SW header is out of service and one CCHX is being repaired, the three remaining CCHXs will be supplied from one SW header. The CC water system (CCWS) is an intermediate cooling system which transfers heat from heat exchangers containing reactor coolant or other radioactive liquids to the SW system. The design basis of the CC water system is a fast cooldown of one unit while maintaining normal loads on the other unit. The CCWS is not a system which functions to mitigate a DBA or presents a challenge to the integrity of a fission product barrier. Calculation ME-0420, Rev.1, Add. B and C were performed to verify operation of three CCHXs on one SW header. It was assumed operation of two SW Pumps on one SW header to which three heat exchangers are connected. Calculation shows that to satisfy design basis requirement for component cooling system (fast cooldown of one unit while maintaining normal loads on the other unit) SW temperature in the reservoir should not be above 75°F due to SW flow limitation. This factor limits implementation of the SW piping replacement around CCHXs to time frame between October and April when temperature in the SW reservoir can be maintained below 75 degrees F. If SW temperature is between 75°F and 78.5 F, CCHX will not be able to supply cooling water of sufficiently low temperature to the RHR heat exchangers to meet the fast cooldown requirements of one unit while other unit is in operation. If the SW temperature exceeds 78.5 F, three SW pumps should be aligned to the header supplying the CCHXs while one SW pump operates on another header. This realignment is required only during RHR operation, i e. during unit shutdown. During installation of the blind flanges on the 24 SW lines to CCHXs and crossconnects between the supply and return lines on each main SW header, six TS 168

hour Section 3/4.7.4.1.d Action Statements (AS) will be required. Repeated isolation of the SW headers was previously analyzed and found to have a small effect (1.7-E8) on the probability of a core damage (CD) event for both Units. After installation of the blind flanges (during repair work) two SW headers will be available to supply RSHXs, CR Chillers and Charging Pumps. Repair of the 24 diameter SW headers to CCHXs and SW lines to individual CCHXs will require isolation of the 24 SW to CCHX for 49 days. A one time amendment of the TS will be required to allow 49 day isolation of one of the two 24 SW headers to CCHXs (two periods for 49 days, one for each header). During isolation of one SW header to CCHXs no SW pump maintenance or testing will be planned. This activity does not change SW or CCW system configuration (except for the temporary crossconnects), does not create the possibility of the accident of a different type than was previously evaluated in the Safety Analysis Report and insignificantly increases the probability of occurrence of previously analyzed accident.

The total effect of this DCP on core damage frequency (CDF) was estimated by a sensitivity analysis combining both the change in the reactor trip initiating event frequency and the increased failure probability of RHR resulting in less than a 1E-6 increase in CDF. This increase was calculated based on 140 days supply of the CCHXs from one SW header (2*49+12*3.5=140). This DCP does not effect the containment systems and there would not be any significant change in off-site dose since the increase in CDF is insignificant. The increase in CDF and resulting increase in off-site dose are less than what is defined as risk significant in the NEI (Nuclear Energy Institute, formerly NUMARC) draft PSA Applications Guide and they are considered insignificant. The evaluation of the total effect of this DCP includes a failure analysis of the reactor cooling pump and motor in case of loss of the CCW.

The assumption that neither unit is utilizing RHR while only one SW header is available, is due to the inability to quantify CDF associated with shutdown conditions. PSA experience indicates that this could be an increase in risk. If a steam generator tube rupture occurs, the unit should be placed on RHR as necessary. If RHR is needed for any other reason, then the best course of action for restoring the second SW header and utilizing RHR cooling should be evaluated based on the DCP status. It may be determined based on the DCP status and the need for RHR cooling that the best course of action is to initiate RHR while only one SW header is operable. The evaluation should include consideration of stopping all SW work to minimize potential damage to the only operable header. It should also consider expediting restoration of the second header and temporarily delaying further DCP work until the unit is off of RHR. This is not intended to allow beginning DCP work when it is known that the unit is scheduled for an outage requiring RHR operation.

Installation of crossconnects between supply and return SW headers with a manual valve on the crossconnect will ensure normal operation of the SW pumps and satisfy GDC-5, Sharing of Structures, Systems, and Components between both units. When one out of two 24" diameter SW header to CCHXs is being repaired the crossconnect valve will be throttled to ensure SW pump flow of approximately 7200 gpm. It is assumed that all four SW pumps are available in the pre-accident condition and three during the accident. In case of CDA, the throttled bypass valve will satisfy the condition of delivering through the SW headers the necessary flow to RSHXs on the accident unit (above 4500 gpm to each of four RSHXs). Availability of four SW pumps in the pre-accident condition and the throttling of the bypass valve will Substitute for the automatic closing of motor operated valves to CCHXs of affected unit on CDA signal which will be temporary disabled for the time of CCHX operation from one SW header. NRC approval is required to implement this temporary crossconnect installation and for defeating the automatic closing of SW motor operated valves (MOVs) to CCHXs on a CDA signal during the time of CCHX operation from one SW header.

Safety Evaluation Number
95-SE-OT-04, Rev. 1

Description Of Activity
Request for Temporary License Amendment for Service Water Preservation - Phase II (Repair/Replacement of Exposed SW piping to/from CCHXs to be performed under DCP-94-010).

The SW piping is experiencing internal wall degradation due to general corrosion and relatively rapid wall loss in localized areas (under pits) due to microbiologically influenced corrosion (MIC). Investigation on selected pipes around CCHXs shows in some cases minimum wall thickness below 0.250 inches. In order to perform repair/replacement of the CCHX exposed piping a one time amendment of the TS will be required to allow 49 day isolation of one of the two 24" SW headers to the CCHXs (two periods for 49 days, one for each header). Also NRC approval is required for the temporary crossconnect installations and defeating the automatic closing of SW MOVs to CCHXs on a CDA signal during the time of CCHX operation from one SW header. Availability of four SW Pumps in the pre-accident condition and throttling of the bypass valve will substitute for the automatic closing of SW motor operated valves to the CCHXs of the affected unit on a CDA signal and the required SW flow will be delivered to the RSHXs of the affected unit (Calculation ME-0420).

Safety Evaluation Summary
The SW piping is experiencing internal wall degradation due to general corrosion and relatively rapid wall loss in localized areas (under pits) due to microbiologically influenced corrosion (MIC). Investigation on selected pipes around CCHXs shows in some cases minimum wall thickness below 0.250 inches. Under this Design Change exposed piping within the Auxiliary Building to/from CCHXs will be replaced or repaired.

This DC does not involve an unreviewed safety question:
Repair of the exposed SW lines to/from CCHXs will temporarily limit SW Supply to CCHXs since while one SW header is out of service and one CCHX is being repaired, the three remaining CCHXs will be supplied from one SW header. The CC water system (CCWS) is an intermediate cooling system which transfers heat from heat exchangers containing reactor coolant or other radioactive liquids to the SW system. The design basis of the CC water system is a fast cooldown of one unit while maintaining normal loads on the other unit. The CCWS is not a system which functions to mitigate a DBA or presents a challenge to the integrity of a fission product barrier. Calculation ME-0420, Rev.1, Add. B and C were performed to verify operation of three CCHXs on one SW header. It was assumed operation of two SW Pumps on one SW header to which three heat exchangers are connected. Calculation shows that to satisfy design basis requirement for component cooling system (fast cooldown of one unit while maintaining normal loads on the other unit) SW temperature in the reservoir should not be above 75°F due to SW flow limitation. This factor limits implementation of the SW piping replacement around CCHXs to time frame between October and April when temperature in the SW reservoir can be maintained below 75 degrees F. If SW temperature is between 75°F and 78.5 F, CCHX will not be able to supply cooling water of sufficiently low temperature to the RHR heat exchangers to meet the fast cooldown requirements of one unit while other unit is in operation. If the SW temperature exceeds 78.5 F, three SW pumps should be aligned to the header supplying the CCHXs while one SW pump operates on another header. This realignment is required only during RHR operation, i e. during unit shutdown. During installation of the blind flanges on the 24 SW lines to CCHXs and crossconnects between the supply and return lines on each main SW header, six TS 168

hour Section 3/4.7.4.1.d Action Statements (AS) will be required. Repeated isolation of the SW headers was previously analyzed and found to have a small effect (1.7-E8) on the probability of a core damage (CD) event for both Units. After installation of the blind flanges (during repair work) two SW headers will be available to supply RSHXs, CR Chillers and Charging Pumps. Repair of the 24 diameter SW headers to CCHXs and SW lines to individual CCHXs will require isolation of the 24 SW to CCHX for 49 days. A one time amendment of the TS will be required to allow 49 day isolation of one of the two 24" SW headers to CCHXs (two periods for 49 days, one for each header). Also, the NRC will be requested to exempt the provisions of TS Section 3.0.4 during the two 49 days isolation of one SW header to the CCHXs. During isolation of one SW header to CCHXs no SW pump maintenance or testing will be planned. This activity does not change SW or CCW system configuration (except for the temporary crossconnects), does not create the possibility of the accident of a different type than was previously evaluated in the Safety Analysis Report and insignificantly increases the probability of occurrence of previously analyzed accident.

The total effect of this DCP on core damage frequency (CDF) was estimated by a sensitivity analysis combining both the change in the reactor trip initiating event frequency and the increased failure probability of RHR resulting in less than a 1E-6 increase in CDF. This increase was calculated based on 140 days supply of the CCHXs from one SW header (2*49+12*3.5=140). This DCP does not effect the containment systems and there would not be any significant change in off-site dose since the increase in CDF is insignificant. The increase in CDF and resulting increase in off-site dose are less than what is defined as risk significant in the NEI (Nuclear Energy Institute, formerly NUMARC) draft PSA Applications Guide and they are considered insignificant. The evaluation of the total effect of this DCP includes a failure analysis of the reactor cooling pump and motor in case of loss of the CCW.

The assumption that neither unit is utilizing RHR while only one SW header is available, is due to the inability to quantify CDF associated with shutdown conditions. PSA experience indicates that this could be an increase in risk. If a steam generator tube rupture occurs, the unit should be placed on RHR as necessary. If RHR is needed for any other reason, then the best course of action for restoring the second SW header and utilizing RHR cooling should be evaluated based on the DCP status. It may be determined based on the DCP status and the need for RHR cooling that the best course of action is to initiate RHR while only one SW header is operable. The evaluation should include consideration of stopping all SW work to minimize potential damage to the only operable header. It should also consider expediting restoration of the second header and temporarily delaying further DCP work until the unit is off of RHR. This is not intended to allow beginning DCP work when it is known that the unit is scheduled for an outage requiring RHR operation.

Installation of crossconnects between supply and return SW headers with a manual valve on the crossconnect will ensure normal operation of the SW pumps and satisfy GDC-5, Sharing of Structures, Systems, and Components between both units. When one out of two 24" diameter SW header to CCHXs is being repaired the crossconnect valve will be throttled to ensure SW pump flow of approximately 7200 gpm. It is assumed that all four SW pumps are available in the pre-accident condition and three during the accident. In case of CDA, the throttled bypass valve will satisfy the condition of delivering through the SW headers the necessary flow to RSHXs on the accident unit (above 4500 gpm to each of four RSHXs). Availability of four SW pumps in the pre-accident condition and the throttling of the bypass valve will Substitute for the automatic closing of motor operated valves to CCHXs of affected unit on CDA signal which will be temporary disabled for the time of CCHX operation from one SW header. No more than three CCHXs will operatre simultaneously on one SW header. NRC approval is required to implement this temporary

crossconnect installation and for defeating the automatic closing of SW motor operated valves (MOVs) to CCHXs on a CDA signal during the time of CCHX operation from one SW header.

Safety Evaluation Number
95-SE-OT-04, Rev. 2

Description Of Activity
Request for Temporary License Amendment for Service Water Preservation - Phase II (Repair/Replacement of Exposed SW piping to/from CCHXs to be performed under DCP-94-010).

The SW piping is experiencing internal wall degradation due to general corrosion and relatively rapid wall loss in localized areas (under pits) due to microbiologically influenced corrosion (MIC). Investigation on selected pipes around CCHXs shows in some cases minimum wall thickness below 0.250 inches. In order to perform repair/replacement of the CCHX exposed piping a one time amendment of the TS will be required to allow 49 day isolation of one of the two 24" SW headers to the CCHXs (two periods for 49 days, one for each header). Also NRC approval is required for the temporary crossconnect installations and defeating the automatic closing of SW MOVs to CCHXs on a CDA signal during the time of CCHX operation from one SW header. Availability of four SW Pumps in the pre-accident condition and throttling of the bypass valve will substitute for the automatic closing of SW motor operated valves to the CCHXs of the affected unit on a CDA signal and the required SW flow will be delivered to the RSHXs of the affected unit (Calculation ME-0420).

Safety Evaluation Summary
The SW piping is experiencing internal wall degradation due to general corrosion and relatively rapid wall loss in localized areas (under pits) due to microbiologically influenced corrosion (MIC). Investigation on selected pipes around CCHXs shows in some cases minimum wall thickness below 0.250 inches. Under this Design Change exposed piping within the Auxiliary Building to/from CCHXs will be replaced or repaired.

This DC does not involve an unreviewed safety question:
Repair of the exposed SW lines to/from CCHXs will temporarily limit SW Supply to CCHXs since while one SW header is out of service and one CCHX is being repaired, the three remaining CCHXs will be supplied from one SW header. The CC water system (CCWS) is an intermediate cooling system which transfers heat from heat exchangers containing reactor coolant or other radioactive liquids to the SW system. The design basis of the CC water system is a fast cooldown of one unit while maintaining normal loads on the other unit. The CCWS is not a system which functions to mitigate a DBA or presents a challenge to the integrity of a fission product barrier. Calculation ME-0420, Rev.1, Add. B and C were performed to verify operation of three CCHXs on one SW header. During the repair/replacement work there will be two SW pumps on one SW header to which three or four heat exchangers are connected. Calculation shows that to satisfy design basis requirement for component cooling system (fast cooldown of one unit while maintaining normal loads on the other unit) SW temperature in the reservoir should not be above 75°F due to SW flow limitation. This factor limits implementation of the SW piping replacement around CCHXs to time frame between October and April when temperature in the SW reservoir can be maintained below 75 degrees F. If SW temperature is between 75°F and 78.5 F, CCHX will not be abie to supply cooling water of sufficiently low temperature to the RHR heat exchangers to meet the fast cooldown requirements of one unit while other unit is in operation. If the SW temperature exceeds 78.5 F, three SW pumps should be aligned to the header supplying the CCHXs while one SW pump operates on another header. This realignment is required only during RHR operation, i e during unit shutdown. During installation of the blind flanges on the 24 SW lines to CCHXs and

crossconnects between the supply and return lines on each main SW header, six TS 168 hour Section 3/4.7.4.1.d Action Statements (AS) will be required. Repeated isolation of the SW headers was previously analyzed and found to have a small effect (1.7-E8) on the probability of a core damage (CD) event for both Units. After installation of the blind flanges (during repair work) two SW headers will be available to supply RSHXs, CR Chillers and Charging Pumps. Repair of the 24 diameter SW headers to CCHXs and SW lines to individual CCHXs will require isolation of the 24 SW to CCHX for 49 days. A one time amendment of the TS will be required to allow 49 day isolation of one of the two 24" SW headers to CCHXs (two periods for 49 days, one for each header). Also, the NRC will be requested to exempt the provisions of TS Section 3.0.4 during the two 49 days isolation of one SW header to the CCHXs. During isolation of one SW header to CCHXs no SW pump maintenance or testing will be planned. This activity does not change SW or CCW system configuration (except for the temporary crossconnects), does not create the possibility of the accident of a different type than was previously evaluated in the Safety Analysis Report and insignificantly increases the probability of occurrence of previously analyzed accident.

The total effect of this DCP on core damage frequency (CDF) was estimated by a sensitivity analysis combining both the change in the reactor trip initiating event frequency and the increased failure probability of RHR resulting in less than a 1E-6 increase in CDF. This increase was calculated based on 140 days supply of the CCHXs from one SW header (2*49+16*3.5=140). This DCP does not effect the containment systems and there would not be any significant change in off-site dose since the increase in CDF is insignificant. The increase in CDF and resulting increase in off-site dose are less than what is defined as risk significant in the NEI (Nuclear Energy Institute, formerly NUMARC) draft PSA Applications Guide and they are considered insignificant. The evaluation of the total effect of this DCP includes a failure analysis of the reactor cooling pump and motor in case of loss of the CCW.

The assumption that neither unit is utilizing RHR while only one SW header is available, is due to the inability to quantify CDF associated with shutdown conditions. PSA experience indicates that this could be an increase in risk. If a steam generator tube rupture occurs, the unit should be placed on RHR as necessary. If RHR is needed for any other reason, then the best course of action for restoring the second SW header and utilizing RHR cooling should be evaluated based on the DCP status. It may be determined based on the DCP status and the need for RHR cooling that the best course of action is to initiate RHR while only one SW header is operable. The evaluation should include consideration of stopping all SW work to minimize potential damage to the only operable header. It should also consider expediting restoration of the second header and temporarily delaying further DCP work until the unit is off of RHR. This is not intended to allow beginning DCP work when it is known that the unit is scheduled for an outage requiring RHR operation.

Installation of crossconnects between supply and return SW headers with a manual valve on the crossconnect will ensure normal operation of the SW pumps and satisfy GDC-5, Sharing of Structures, Systems, and Components between both units. When one out of two 24" diameter SW header to CCHXs is being repaired the crossconnect valve will be throttled to ensure SW pump flow of approximately 7400 gpm. It is assumed that all four SW pumps are available in the pre-accident condition and three during the accident. In case of CDA, the throttled bypass valve will satisfy the condition of delivering through the SW headers the necessary flow to RSHXs on the accident unit (above 4500 gpm to each of four RSHXs). Availability of four SW pumps in the pre-accident condition and the throttling of the bypass valve will Substitute for the automatic closing of motor operated valves to CCHXs of affected unit on CDA signal which will be temporary disabled for the time of CCHX operation from one SW header. No more than three CCHXs will operatre

simultaneously on one SW header. NRC approval is required to implement this temporary crossconnect installation and for defeating the automatic closing of SW motor operated valves (MOVs) to CCHXs on a CDA signal during the time of CCHX operation from one SW header.

Safety Evaluation Number
95-SE-OT-05

Description Of Activity
UFSAR - Updated Final Safety Analysis Report
TRM - Technical Requirements Manual
SER - Safety Evaluation Report, Dated March 1979
Station Appendix R Report

The SER identifies a commitment to provide water extinguishers in three areas. This evaluation is for replacing 8 - Class 2A water extinguishers in the following Unit 1 & 2 areas: Cable Vault and Tunnel, Emergency Switchgear Room and Instrument Rack Room with 8 - Class 4A60BC dry chemical extinguishers. Class A extinguishers are listed for extinguishment of ordinary combustibles such as wood, paper, and cloth and not for use on electrical equipment.

Safety Evaluation Summary
The major issued considered was a fire in either the Emergency Switchgear Room, Cable Vault/Tunnel, or Instrument Rack Room. These are the three areas in which the fire extinguishers are being replaced.

The change should be allowed since the replacement of the Class 2A water extinguishers with Class 4A60BC extinguishers improves the fire extinguishing capabilities for these areas. The water extinguisher originally installed for the ordinary Class A combustibles in the area is not rated for used on electrical fires and could create an electrical hazard if used. The introduction of dry chemical extinguishers provides twice the Class A extinguishing capability and also provides the proper extinguisher for extinguishing electrical fires. Dry chemical is nontoxic and is electrically nonconductive so that there is no personnel hazard or adverse effect on equipment.

An unreviewed safety question does not exist since fire extinguishers are used for manual fire fighting and are not required for the operation of any safety or nonsafety related systems or components necessary to operated the plant. The extinguishers will be used to extinguish a fire in the area. In the event the fire is not extinguished, Appendix R has established that alternate shutdown equipment is available outside these fire areas to safely shutdown the plant The fire extinguishers are not governed by the Operating License (OL) or Technical Specifications (TSs) therefore changing the extinguishers does not change the OL or TSs.

Safety Evaluation Number
95-SE-OT-06

Description Of Activity
Technical Specifications Change Request No. 325, Appendix J to Part 50, Type A Testing Requirements. Exemption requests from the requirements of Appendix J to Part 50, Sections III.D.1.(a) and IV.A.

Exemptions from 10 CFR 50, Appendix J schedular requirements for a Type A test that would permit delaying the Unit 2 Type A test until the next refueling outage, currently scheduled for October 1996 and eliminate the Type A test required following steam generator replacement. Include a statement in TS 4.6.1.2.a to permit exemptions from the Type A testing requirements in Appendix J as approved by the NRC.

Safety Evaluation Summary
This activity includes:

1) An exemption from the 10 CFR 50, Appendix J, Section III.D.1(a), "Periodic Retest Schedule for Type A Tests," requirements which would permit a schedular delay in the performance of the Unit 2 Type A test by one operating cycle.
2) An exemption from the 10 CFR 50, Appendix J, Section IV.A, "Special Testing Requirements for Containment Modifications," requirements which would eliminate the post-modification Type A test following the Unit 2 steam generator replacement
3) A TS change to include a statement in TS 4.6.1.2.a to permit changes in the retest schedule if an schedular exemption to Appendix J is approved by the NRC.

"Type A Tests" are defined in Section II.F of Appendix J to 10 CFR Part 50 as "tests intended to measure the primary reactor containment overall integrated leakage rate (1) after the containment has been completed and is ready for operation. and (2) at periodic intervals thereafter." Section IIID. 1(a) of Appendix J, "Periodic Retest Schedule for Type A Tests," further requires "a set of three Type A tests shall be performed at approximately equal intervals during each 10-year service period. The third test of each set shall be conducted when the plant is shutdown for the 10-year plant inservice inspections.

In addition, a special post-modification Type A test is required by Section IV.A of Appendix J, "Special Testing Requirements for Containment Modifications," which states that "any major modification, replacement of a component which is part of the primary reactor containment boundary, or resealing a seal-welded door, performed after the preoperational leakage rate test shall be followed by either a Type A, Type B, or Type C test as applicable for the area affected by the modification.

An exemption is requested from the portion of Section m.D.1(a) that requires periodic Type A retests to be performed at approximately equal intervals during each 10-year service period. In addition, an exemption is requested from the portion of Section IV.A that requires a Type A test to be performed following a major modification or replacement of a component which is part of the primary reactor containment boundary. Specifically, the post-modification exemption is requested from performing a Type A test due to the activities associated with the upcoming steam generator replacement. The basis for the post-modifications exemption request is that. in this case, the ASME Section XI inspection and testing requirements more than fulfill the intent of the requirements of Section IV.A of Appendix J.

10 CFR 50 Appendix J states that the purpose of the regulation is to assure that leakage through primary containment and systems and components penetrating containment does not exceed allowable values, as specified in the Technical Specifications or associated bases, and that proper maintenance and repair are performed throughout the service life of the containment boundary components. The integrated leakrate test (ILRT) history for North Anna Unit 2 indicates that the containment structure itself has not experienced any leakage. Additionally, the pretest containment inspections have not identified deterioration of the containment liner.

The NRC has performed a detailed study of ILRTs performed from 1987 to 1993. This study, documented in NUREG-1493, "Performance Based Containment Leak-Test Program," Draft Revision 3, March 31, 1994, determined that 97% of all leakrates that exceed the acceptance criteria are identified by the ILRT test programs. Therefore, as indicated in the NRC's study, and as evidenced by our containment performance history, postponing the ILRT by one refueling cycle remains consistent with the intent of the regulation and will not present any undue risk to the public health and safety.

The exemption requests and TS change do not alter the operation of the containment or any containment isolation valve. The operability requirements and leakage requirements for the containment, containment isolation valves, and containment systems as defined in TS 3/4.6.1 are not being changed. However, the exemption will change the actual surveillance interval for the upcoming Unit 2 Type A test. The change in surveillance interval does not affect the operation of the containment, containment isolation valves, or containment systems. No new methods of operation or accident precursors are being introduced by this activity. Therefore, this activity does not increase the probability of occurrence for the accidents identified in the SAR.

Each penetration will be tested in accordance with TS 4.6.1.2.b and Appendix J during the steam generator replacement outage with the total Type B and C as-left leakage < 0.6 La as required by TS. The previous three Type A tests have demonstrated the integrity of the containment. The containment will be maintained operable to mitigate the consequence of an accident as assumed in the accident analysis with overall leakage < La. Therefore, this activity does not increase the consequences of the accidents identified in the SAR or increase the consequences of a malfunction.

The proposed exemptions and the proposed TS change do not alter the operation of the containment or any containment isolation valve. Leakage requirements for the containment, containment isolation valves, and containment systems are not being changed. No new methods of operation or accident precursors are being introduced by this activity. Therefore, this activity does not create the possibility for an accident or malfunction of a different type than was previously evaluated.

The operability requirements and the leakage requirements for the containment, containment isolation valves, and containment systems are not being changed As stated above, each penetration will be tested in accordance with TS 4.6.1.2.b and Appendix J during the upcoming steam generator replacement outage. Any maintenance on a containment isolation valve requires a local leak rate test prior to returning to the valve to service to ensure the "as-left" leakage is within the specified TS limit. Therefore, this activity does not reduce the margin of safety of any part of the Technical Specifications as described in the bases section.

Based on the above, these exemption requests and the proposed TS change do create an unreviewed safety question.

Safety Evaluation Number
95-SE-OT-07

Description Of Activity

Temporary Shielding Request (TSR's) Nos 95-TSR-007, 008, 009, 010, and 032

Lead blanket shielding is to be placed over selected RCS piping during the 1995 NAPS Unit 2 Steam Generator Replacement Project/Refueling Outage (SGRP/RO); namely the horizontal segments of the 2" diameter loop bypass lines, the 2" diameter flow element bypass line , the downstream cold leg lines, for all three loops, as well as PZR spray lines, 4"-RC-414 & 415-1502-Q1 These shielding packages will be in place while fuel is loaded in the reactor vessel. Other RCS SGRP/RO TSR's will not be in place while there is fuel in the reactor vessel, and still others will be strategically located so that no operable Safety-Related equipment exists within their collapse envelope, hence these other "defueled" and "strategically located" TSR's are not the central issue of this 10 CFR 50.59 Safety Evaluation. While not the central issue of this safety evaluation, these "other" TSR's are still discussed within this safety evaluation for coordination purposes only

Special Conditions
North Anna Unit 2 shall be in either Mode 5 or 6 while lead blanket shielding, associated with these TSR's, is wrapped around the specified RCS piping. The RCS may be either filled or empty, upstream and/or downstream of the loop stop valves. Design calculations, referenced within, have demonstrated that the associated RCS piping can withstand the additional dead load of the lead blanket shielding, under all applicable Mode 5 and 6 normal and dynamic load conditions. Administrative control for these load conditions are implemented via Engineering, SNSOC, ALARA and OPS signoffs in the individual TSR's (Reference Parts 3 & 4 of the Temporary Shielding Engineering Evaluation and Parts 1-7 of the Temporary Shielding Installation, Inspection, and Removal)

Safety Evaluation Summary
MAJOR ISSUES:

The main issue associated with this safety evaluation is the concern for the structural integrity of the RCS piping due to the additional dead weight of lead blanket shielding from these TSR's. The five (5) TSR's for which this safety evaluation was written, directly load RCS piping while fuel is still loaded in the reactor vessel. Direct loaded lines include: PZR spray line, 4"-RC-414-1502-Q1 (95-TSR-007) ; the 2" diameter RCS loop flow element bypass lines (95-TSR-008); downstream portion of the loop cold legs (95-TSR-009); PZR spray line, 4"-RC-415-1502-Q1 (95-TSR-010), and the horizontal segments of the 8" diameter loop bypass lines (95-TSR-032). Although the RCS will be drained and isolated from the reactor vessel, at the loop stop valves, for some of these TSR's, any added dead weight to the main RCS loop piping will still be tributary to the rest of the loop. Without a special pipe stress evaluation, this additional tributary dead weight constitutes an unreviewed safety question. Hence, all TSR's that directly load RCS piping, while fuel is loaded in the reactor vessel, pose an unreviewed safety question which must be addressed by pipe stress analysis, in a 10 CFR 50.59 Safety Evaluation (Reference Engineering Transmittal No. CEM-95 -0008, attached). Administrative control (i.e. installation, temporary supports, removal, etc) for these TSR's will be achieved as previously mentioned, via Engineering, SNSOC, ALARA and OPS reviews, on Page 1 of 12, Block 7, for each individual TSR.

As mentioned on Page 1 of 12, Block 5, a certain number TSR's will be installed during this outage that will directly load RCS and other Safety-Related piping, but they have been scheduled to be in place only during the window of time that the reactor is defueled (i.e. 95-TSR-012, 035, 036, 039, 041, 043, 044, 045, 047, 049, and 051). The normal DL + LL stresses, associated with these "defueled" TSR's have been analyzed and compared against the normal DL + LL pipe allowable stress limits. Their individual analyses are referenced on each TSR. A similar analysis will be performed for any additional "defueled" TSR's that may be required. These analyses will not affect the analyses performed for the five (5) "fueled" TSR's of this safety evaluation. Since the reactor will be defueled when these other TSR's are in place, and since RCS pipe stresses still remain within allowable limits, no unreviewed safety question exists. As such, these "defueled" TSR's are not the central issue of this safety evaluation, but are mentioned here for coordination purposes only. Administrative control (i.e. installation, temporary supports, removal, etc.) for these "defueled" TSR's is still coordinated, as previously mentioned, on Page 1 of 12, Block 7, via Engineering, SNSOC, ALARA and OPS reviews for each individual TSR.

Also mentioned on Page 1 of 12, Block 5, other TSR's will be installed, during Modes 5 and 6, in the vicinity of RCS and other Safety-Related piping while fuel may or may not be loaded in the reactor vessel (i.e. 95-TSR-001, 002, 003, 004, 005, 006, 011, 031 033, 037, 038, 040, 042, 046, and 050). The location and configuration of these TSR's have been strategically selected so that no operable Safety-Related equipment will exist within their collapse envelopes. An analysis of their collapse envelopes are provided within each individual TSR. Hence, their potential for adversely affecting operable Safety-Related equipment has been eliminated. Any additional TSR's of this type may be added, as required, without affecting the analyses performed for the five (5) "fueled" TSR's of this safety evaluation. As such this group of TSR's are not the central issue of this safety evaluation, but have been mentioned for coordination purposes only. However, like all other TSR's, administrative control (i.e. installation, temporary supports, removal etc.) for this class of TSR's is handled in the manner as described on Page 1 of 12, Block 7.

All TSR's are designed to perform their radiological shielding function without adversely affecting the nearby equipment, especially operable Safety-Related equipment. Additionally, these TSR's are administratively controlled under VPAP-2105, installed in accordance with approved Health Physics work procedures, and provide an effective ALARA benefit to the personnel that will perform work in these areas during the 1995 NAPS Unit 2 SGRP/RO. ALARA, in coordination with OPS; selects the most appropriate locations for TSR packages, determines the shielding quantities, installs the shielding, monitors its effectiveness, and removes the shielding. Design Engineering evaluates the supporting structure for the shielding (i.e. pipe, Tub-Lok, or other plant structures), ensures that it is properly attached and braced (i.e. Calculation No. DEO-0102, R/0, and other referenced calculations) to prevent adverse shifting/falling, and performs a field investigation, as required, to ensure that all design requirements of each TSR have been properly installed in the field. All of these design features are verified and documented on each individual TSR to ensure that no unreviewed safety questions exist.

UNREVIEWED SAFETY QUESTION ASSESSMENT:

1. All TSR's have been analyzed for the additional dead weight of lead shielding that they directly impart to the RCS piping or have been strategically scheduled, located and/or configured to ensure that there will be no operable Safety-Related components within their collapse envelopes. Administrative control over each TSR is maintained to ensure all intended design features of each TSR get implemented in the field. Since the addition of lead shielding has no affect upon the frequency of previously analyzed

dynamic events (i.e. seismic, thermal, hydraulic, etc), there is no increase in the probability of experiencing any of the above mentioned dynamic events.

2. Since the evaluations, mentioned above, have yielded stresses below the design allowable limits and/or were verified that no adverse interaction will occur with operable Safety-Related components, there is no net increase in the design maximum stresses that can occur under the applicable Mode 5 & 6 load condition. Hence, there is no expected increase in the consequence for dynamic events with these TSR's in place, as described.

3. Given the discussions, listed above, in items #1 and #2, and considering the requirement that all critical TSR's are field-verified to ensure their intended design features are implemented, accidents of a different kind than were previously analyzed are not expected to occur.

Safety Evaluation Number
95-SE-OT-08

Description Of Activity
UFSAR Section 8.1

The UFSAR is being reviewed over the next few years to enhance its accuracy. The description of the utility grid in Section 8.1.1 was expanded to reflect the Gordonsville 230 Kv line and to better describe the connections between the switchyard and the plant. Section 8.1.2 was revised to reflect the correct number of normal 480-v buses per unit. In addition, several administrative/editorial changes were made. Also, Figure 8.2-15 should be upgraded to reflect the 230 Kv Gordonsville line. Drawing 1171 5-FE-1 BB could be used as a replacement.

Safety Evaluation Summary
The UFSAR is being reviewed over the next few years to enhance its accuracy and to better reflect actual plant operations. The description of the utility grid in Section 8.1.1 was expanded to reflect the Gordonsville 230 Kv line and to better describe the connections between the switchyard and the plant. Section 8.1.2 was revised to reflect the correct number of normal 480-v buses per unit. In addition, several administrative / editorial changes were made. Also, Figure 8.2-15 should be upgraded to reflect the 230 Kv Gordonsville line. Drawing 11715-FE-1BB could be used as a replacement.
The proposed changes do not present on unreviewed safety question because:
1) The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report was not increased. No changes are being made to electrical power systems that could increase the probability of occurrence or the consequences of accidents described in the safety analysis report. Safety related equipment will continue to function as assumed in the SAR. The UFSAR is being revised to enhance its accuracy and to reflect actual approved plant operations. No changes to plant procedures or equipment operation are being made as a result of the proposed changes.
2) The possibility for an accident or malfunction of a different type than any evaluated in the safety analysis report has not been created. No changes are being made to electrical power systems. Safety related equipment will continue to function as assumed in the SAR. The UFSAR is being revised to enhance its accuracy and to reflect actual approved plant operations. No changes to plant procedures or equipment operation are being made as a result of the proposed changes.
3) The margin of safety as defined in the basis for any technical specification was not reduced. No changes are being made to electrical power systems. Safety related equipment will continue to function as assumed in the SAR and the Technical Specifications. The UFSAR is being revised to enhance its accuracy and to reflect actual approved plant operations. No changes to plant procedures or equipment operation are being made as a result of the proposed changes.

Safety Evaluation Number
95-SE-OT-09

Description Of Activity
Technical Requirements Manual Revision 9.
Change General Requirement 1.0.10 to have original written Special Report submitted to the NRC Document Control Desk.

Safety Evaluation Summary
This TRM Revision deals with the distribution of Special Reports to the NRC as per NRC Administrative Letter 94-17. NRC Special Reports are to be sent to the Document Control Desk and a copy to the Regional Administrator, Region 2.

This change does not result in an unreviewed safety question based on the following:

1. This revision does not involve any increase in the probability or consequences of any previously evaluated accident. This TRM revision will only ensure NRC Special Reports are distributed as per NRC Administrative Letter 94-17. This revision does not require any modifications to plant hardware or operating practices. Therefore, this revision has no affect on any previously analyzed accidents.

2. This revision does not create the possibility of a new or different kind of accident from any accident previously evaluated. The revision does not affect any operating, maintenance, or surveillance practices; also, there are no design or hardware modifications associated with this revision. Therefore, the possibility of a malfunction or failure that results in a different kind of accident is not created.

3. This revision does not involve a reduction in the margin to safety. This TRM revision will only ensure NRC Special Reports are distributed as per NRC Administrative Letter 94-17 and has no impact on the performance of any plant equipment. Therefore, there is no reduction to any safety margins due to this revision.

Safety Evaluation Number
95-SE-OT-10

Description Of Activity
UFSAR Change Request FN 95-005
DR N95-272
1-PT-64.8 & 2-PT-64.8

The UFSAR change request revises the description of how the temporary dike around the Inside Recirculation Spray pumps(IRSP) in containment is filled and drained for the performance of the Inservice Inspection flow test of the Inside Recirculation Spray pumps. Specifically, it deletes the following sentence: "Water is supplied to the sump from the refueling cavity by a portable pump and piping and is returned to the cavity following completion of the test."

Safety Evaluation Summary
This safety evaluation is for a UFSAR Change Request which revises the description of the Inservice Inspection flow test of the inside Recirculation Spray(RS) pumps. Specifically, the change request revises how the temporary dike built around the Inside Recirculation Spray pumps(IRSP) in containment is filled with water and drained following completion of the test. The UFSAR states that the dike is filled with water from the refueling cavity and the water is pumped back to the refueling cavity after the test is complete. The change request deletes this description of filling and draining the dike.
Use of water from the refueling cavity is not considered good ALARA practice as this water could contain highly radioactive contaminants from failed fuel. Returning the water to the refueling cavity from the sump dike after the test is complete is not desired as this water is usually a dirty, rust color and may contain particulates that could enter the Reactor Coolant System(RCS).
Use of Primary Grade(PG) water to fill the dike reduces occupational exposure from what could occur if water from the refueling cavity is used. By draining the PG water from the dike into the containment sump after completing the test, intrusion of foreign material into the RCS is eliminated.
The residual PG water left in the lower sump after the test would decrease the sump boron concentration by less than 1 ppm during a LOCA. This was evaluated and found to not affect reactivity during a LOCA. The residual PG water will also tend to offset the effect of the sodium hydroxide from the Quench Spray Chemical Addition Tank on sump pH. This was evaluated and found to be an insignificant change.
There is no impact of the test results due to the use of PG water instead of the borated water from the refueling cavity. The density of the PG water is within the range of densities of the borated water for the expected sump temperatures during a Loss of Coolant Accident.
This change is a document change and it does not alter the design, operation or performance of the RS system. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety is not increased by this change. Therefore, an unreviewed safety question does not exist.

Safety Evaluation Number
95-SE-OT-11

Description Of Activity

Safety Evaluation Summary
North Anna Independent Spent Fuel Storage Installation (ISFSI) Safety Analysis Report and Technical Specifications

Submittal to the NRC of ISFSI license application documents, including a License Application, Environmental Report, Technical Specifications and Safety Analysis Report

Safety Evaluation Summary
This change involves submittal of license application documents to the NRC for construction and operation of an ISFSI at North Anna. The scope of this safety evaluation extends only to licensing and operation of the ISFSI. It does not include construction of the ISFSI. which will be evaluated under a DCP. It does not cover station modifications for cask handling, which will also be evaluated under a DCP. It does not cover the handling of casks in the station, which will be evaluated under a separate safety evaluation (10 CFR Part 50).

Licensing and operation of an ISFSI will not create an unreviewed safety question under 10 CFR Part 50. However, since the design calculations and accident analyses have not yet been reviewed by the NRC, construction and operation of an ISFSI constitutes an unreviewed safety question. Submittal and NRC approval of the licensing application documents is therefore required.

The licensing requirements of 10 CFR Part 72 have been addressed by the licensing application documents. The ISFSI facility and the cask designs which are specified in the licensing application documents comply with the applicable requirements of Part 72. The results of the accident evaluations documented in the Safety Analysis Report demonstrate that the ISFSI complies with the applicable limits of Part 72.

Safety Evaluation Number
95-SE-OT-12

Description Of Activity
Temporary Shielding Request (TSR's) Nos. 95-TSR-035, 049
Lead blanket shielding is to be placed over selected RCS piping during the 1995 NAPS Unit 2 Steam Generator Replacement Project/Refueling Outage (SGRP/RO): namely the hot leg and crossover leg. These shielding packages will be in place while fuel is in the reactor vessel.

Special Conditions
North Anna Unit 2 shall be in either Mode 5 or 6 while lead blanket shielding, associated with these TSR's, is wrapped around the specified RCS piping. The RCS may be either filled or empty, upstream and/or downstream of the loop stop valves. Design calculations, referenced within, have demonstrated that (1) the associated RCS piping can withstand the additional dead load of the lead blanket shielding, under all applicable Mode 5 and 6 normal and dynamic load conditions and (2) that the methods of attaching lead blanket shielding to pipe will be adequate to keep the blankets from falling off in a seismic event. Administrative control for these load conditions are implemented via Engineering, SNSOC, ALARA and OPS signoffs in the individual TSR's (Reference Parts 3 & 4 of the Temporary Shielding Engineering Evaluation and Parts 1-7 of the Temporary Shielding Installation, Inspection, and Removal).

Safety Evaluation Summary
MAJOR ISSUES:

The main issue associated with this safety evaluation is the concern for the structural integrity of the RCS piping due to the additional dead weight of lead blanket shielding from these TSR-s. The two (2) TSR's for which this safety evaluation was written, directly load RCS piping while fuel is still in the reactor vessel. Direct loaded lines include RCS hot leg and crossover leg. Although the RCS will be drained and isolated from the reactor vessel, at the loop stop valves, for some of these TSR's, any added dead weight to the main RCS loop piping will still impose load to the rest of the loop. Without a special pipe stress evaluation, this additional dead weight constitutes an unanalyzed condition. Hence, all TSR's that directly load RCS piping, while fuel is in the reactor vessel, must be addressed by review of pipe stress analysis, in a 10 CFR 50.59 Safety Evaluation. Administrative control (i.e. installation, temporary supports, removal, etc...) for these TSR's will be achieved as previously mentioned, via Engineering, SNSOC, ALARA and OPS reviews on Page 1 of 12, Block 7, for each individual TSR.

All TS is designed to perform their radiological shielding function without adversely affecting the nearby equipment, especially operable Safety-Related equipment. Additionally, these TSR's are administratively controlled under VPAP-2105 installed in accordance with approved Health Physics work procedures, and provide an effective ALARA benefit to the personnel that will perform work in these areas during the 1995 NAPS Unit 2 SGRP/RO. ALARA, in coordination with OPS; selects the most appropriate locations for TSR packages, determines the shielding quantities, installs the shielding, monitors its effectiveness, and removes the shielding. Design Engineering evaluates the supporting structure for the shielding (i.e. pipe, Tub-Lok, or other plant structures), ensures that it is properly attached and braced (i.e. Calculation No. DEO-0102, R/0 and Bechtel Calculation 22462-C-033) to prevent adverse shifting/falling, and performs a field investigation, as required, to ensure that all design requirements of each TSR have been

properly installed in the field. All of these design features are verified and documented on each individual TSR to ensure that no unreviewed safety questions exist.

UNREVIEWED SAFETY QUESTION ASSESSMENT:

1. All TSR's have been analyzed for the additional dead weight of lead shielding that they directly impose on the RCS piping and have been analyzed to ensure that the methods of attaching lead blanket shielding to pipe will be adequate to keep the blankets from falling off in a seismic event. Administrative control over each TSR is maintained to ensure all intended design features of each TSR get implemented in the field. Since the addition of lead shielding has no affect upon the frequency of previously analyzed dynamic events (i.e. seismic, thermal, hydraulic, etc...), there is no increase in the probability of experiencing any of the abovementioned dynamic events.

2. Since the evaluations, mentioned above, have determined that stresses are below the design allowable limits and were verified that no adverse interaction will occur with operable Safety-Related components, there is no net increase in the design maximum stresses that can occur under the applicable Mode 5 & 6 load conditions. Hence, there is no expected increase in the consequences for dynamic events with these TSR's in place.

3. Given the discussions, listed above, in items #1 and #2, and considering the requirement that all critical TSR's are field-verified to ensure their intended design features are implemented, accidents of a different kind than were previously analyzed are not expected to occur

Safety Evaluation Number
95-SE-OT-13

Description Of Activity
UFSAR Section 12.2
The UFSAR is being reviewed over the next few years to enhance its accuracy. The description of the design objectives for the ventilation system in Section 12.2.1 refers to the wrong table in 10CFR20, Appendix B and it should be corrected. The design description of the fuel building ventilation system in Section 12.2.2.4 refers to a Table 11.1-5 that provides the design and expected escape rate coefficients for failed fuel in the spent fuel pit. Table 11.1-5 does not provide this information, so this information is being deleted. In addition. Section 12.2.2.4 is being enhanced to describe a safety analysis assumption that fuel in the spent fuel pit decays for 100 hours, the minimum time before fuel can be transferred from the core to the spent fuel pit. This section is being also being expanded to describe that the Technical Specifications require 150 hours as the minimum time prior to movement of irradiated fuel from the core. Lastly, throughout Section 12.2.4, references to other sections in the UFSAR need to be updated to the proper reference.

Safety Evaluation Summary
The UFSAR is being reviewed over the next few years to enhance its accuracy. The description of the design objectives for the ventilation system in Section 12.2.1 refers to the wrong table in 10CFR20, Appendix B and it should be corrected. The design description of the fuel building ventilation system in Section 12.2.2.4 refers to a Table 11.1-5 that provides the design and expected escape rate coefficients for failed fuel in the spent fuel pit. Table 11.1-5 does not provide this information, so this information is being deleted. In addition, Section 12.2.2.4 is being enhanced to describe a safety analysis assumption that fuel in the spent fuel pit decays for 100 hours, the minimum time before fuel can be transferred from the core to the spent fuel pit. This section is being also being expanded to describe that the Technical Specifications require 150 hours as the minimum time prior to movement of irradiated fuel from the core. Lastly, throughout Section 12.2.4, references to other sections in the UFSAR need to be updated to the proper reference.

The proposed changes do not present an unreviewed safety question because:

1) The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report was not increased. No changes are being made to the actual operation or methods of operating the ventilation systems in the plant. Therefore, ventilation systems will continue to operate as designed and maintain the radiological consequences of an accident within the10 CFR 20 Appendix B and 10 CFR 100 requirements. The UFSAR is being revised to enhance its accuracy and to reflect actual approved plant operations.

2) The possibility for an accident or malfunction of a different type than any evaluated in the safety analysis report has not been created. No changes are being made to the actual operation or methods of operating the ventilation systems in the plant. Therefore, they will continue to operate as designed. The UFSAR is being revised to enhance its accuracy and to reflect actual approved plant operations.

3) The margin of safety as defined in the basis for any technical specification was not reduced. Safety related equipment will continue to function as assumed in the SAR and Technical Specifications. The UFSAR is being revised to enhance its accuracy and to reflect actual approved plant operations. No changes to plant procedures or equipment operation are being made as a result of the proposed changes

Safety Evaluation Number
95-SE-OT-13 Rev. 1

Description Of Activity
UFSAR Section 12.2
The UFSAR is being reviewed over the next few years to enhance its accuracy. The description of the design objectives for the ventilation system in Section 12.2.1 refers to the wrong table in 10CFR20, Appendix B and it should be corrected. The design description of the fuel building ventilation system in Section 12.2.2.4 refers to a Table 11.1-5 that provides the design and expected escape rate coefficients for failed fuel in the spent fuel pit. Table 11.1-5 does not provide this information, so this information is being deleted. In addition. Section 12.2.2.4 is being enhanced to describe a safety analysis assumption that fuel in the spent fuel pit decays for 100 hours, the minimum time before fuel can be transferred from the core to the spent fuel pit. This section is being also being expanded to describe that the Technical Specifications require 150 hours as the minimum time prior to movement of irradiated fuel from the core. Lastly, throughout Section 12.2.4, references to other sections in the UFSAR need to be updated to the proper reference.

Safety Evaluation Summary
The UFSAR is being reviewed over the next few years to enhance its accuracy. The description of the design objectives for the ventilation system in Section 12.2.1 refers to the wrong table in 10CFR20, Appendix B and it should be corrected. The design description of the fuel building ventilation system in Section 12.2.2.4 refers to a Table 11.1-5 that provides the design and expected escape rate coefficients for failed fuel in the spent fuel pit. Table 11.1-5 does not provide this information, so this information is being deleted. In addition, Section 12.2.2.4 is being enhanced to describe a safety analysis assumption that fuel in the spent fuel pit decays for 100 hours, the minimum time before fuel can be transferred from the core to the spent fuel pit. This section is being also being expanded to describe that the Technical Specifications require 150 hours as the minimum time prior to movement of irradiated fuel from the core. Lastly, throughout Section 12.2.4, references to other sections in the UFSAR need to be updated to the proper reference.

The proposed changes do not present an unreviewed safety question because:

1) The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report was not increased. No changes are being made to the actual operation or methods of operating the ventilation systems in the plant. Therefore, ventilation systems will continue to operate as designed and maintain the radiological consequences of an accident within the10 CFR 20 Appendix B and 10 CFR 100 requirements. The UFSAR is being revised to enhance its accuracy and to reflect actual approved plant operations.

2) The possibility for an accident or malfunction of a different type than any evaluated in the safety analysis report has not been created. No changes are being made to the actual operation or methods of operating the ventilation systems in the plant. Therefore, they will continue to operate as designed. The UFSAR is being revised to enhance its accuracy and to reflect actual approved plant operations.

3) The margin of safety as defined in the basis for any technical specification was not reduced. Safety related equipment will continue to function as assumed in the SAR and Technical Specifications. The UFSAR is being revised to enhance its accuracy and to reflect actual approved plant operations. No changes to plant procedures or equipment operation are being made as a result of the proposed changes

Safety Evaluation Number
95-SE-OT-14

Description Of Activity
UFSAR Section 15.2.13, Accidental Depressurization of the Main Steam System
UFSAR Section 15.4.2.1, Rupture of a Main Steam Line
Implement a revised Main Steamline Break accident analysis for North Anna Units 1 and 2.

Special Conditions
This analysis assumes the following conditions apply:
1. Limiting MSLB inside containment of 1.4 ft2, consistent with steam outlet flow limiter flow area in Model 51F steam generators. Therefore, this analysis applies to Unit 2 only after steam generator replacement.
2. Operation in accordance with the North Anna Units 1 and 2 Technical Specifications.

Safety Evaluation Summary
This safety evaluation implements a revised analysis of the Main Steamline Break (MSLB) event. The MSLB event is characterized by an increase in heat removal by the secondary system, decreasing steam pressure, decreasing primary core average temperature and RCS pressure and increasing power and/or decreasing shutdown margin. It is assumed to include any accident which results in an uncontrolled steam release from a steam generator. The release can occur due to a break in a steamline or a valve malfunction. Under the Westinghouse methodology followed for analysis of this event, the bounding MSLB case assumes a double-ended steamline break with a flow area equivalent to that of the steam outlet flow limiter for the Model 51 F steam generators (1.4 ft2) at hot zero power (HZP) conditions. In the presence of a negative moderator coefficient, the cooldown results in a positive reactivity insertion. If the most reactive rod cluster control assembly (RCCA) is assumed stuck in its fully withdrawn position after reactor trip, the possibility that the core will become critical and return to power Increases.

This analysis of the MSLB event contains three significant assumptions that make it consistent with current and anticipated design basis:

1 . The analysis assumes a bounding break size of 1.4 ft2, consistent with the steam outlet flow limiter flow area for Model 51 F steam generators in operation in Unit 1 and scheduled for installation in Unit 2 during the current outage.

2. The boron injection tank (BIT) has a 0% boron concentration at the onset of the transient. Use of a 0% BIT boron concentration provides a bounding analysis to support either a reduced BIT boron concentration or BIT removal.

3. The lead/lag time constants in the steamline pressure protection loop is modified to a 12 second lead and a 2 second lag, versus a 50 second lead and a 5 second lag currently in use in the plant. These modified time constants will reduce the sensitivity of the protection loop to large steam pressure changes while providing adequate core protection. This analysis will support operation with either set of time constants.

The analysis of a main steamline break must satisfy the following criteria:

1. Assuming a stuck rod cluster control assembly, with or without offsite power, and assuming a single failure in the engineered safeguards, there is no consequential damage to the primary system, and the core remains in place and intact.

2. Energy released to containment from the worst steam pipe break does not cause failure of the containment structure.

Reactor protection in the MSLB analysis is provided by negative reactivity feedback effects, steamline isolation on low core average temperature coincident with high steam flow, and safety injection actuation on the following trips:

1. Low steam pressure coincident with high steam flow.

2. Low core average temperature coincident with high steam flow.

3. Low pressurizer pressure.

The action of these mitigating functions prevents the core from reaching the minimum DNBR limit during the cooldown and resulting return to power, The North Anna mass and energy release and containment response analyses bound the current set of assumptions for MSLB by assuming a bounding break size of 4.6 ft2 versus the 1.4 ft2 break in this analysis.

The reanalysis of the MSLB assumes hot zero power initial conditions and the failure of one safeguards train (minimum safeguards), consistent with the Westinghouse MSLB methodology. The analysis also includes steamline isolation and safety injection time delays consistent with Technical Specifications, and conservative safety injection flow rates.

As documented in Technical Report NE-1016, Revision 0, the MSLB analysis produced limiting conditions minimum DNBR results that were bounded by the correlation minimum DNBR limit. Thus, assuming nucleate boiling (and fuel damage) occurs at the correlation minimum DNBR limit, the analysis meets the acceptance criterion outlined above by assuring no damage occurs to the primary system or the core during a MSLB. Containment structural integrity is confirmed by continued use of the mass and energy releases associated with the 4.6 ft2 break.

The reanalysis of the MSLB event assuming a 1.4 ft2 limiting break, a 0% BIT boron concentration, and the revised steam generator protection lead and lag time constants produced acceptable results that support station operation. The 0% BIT boron concentration and the revised steam generator protection time constants represent bounding conditions which support other proposed changes to the plant. No unreviewed safety question is created by implementing this revised MSLB analysis as evidenced by the following:

1. No increase in the probability of occurrence or consequence of an accident or malfunction of equipment will result from this change since the requirements for the MSLB analysis acceptance criteria and the methodology for analyzing the transient remain unchanged. The use of a 1.4 ft2 limiting break size mitigates the consequences of the MSLB; the 0% BIT boron concentration and the revised lead/lag time constants provide bounding conditions for the MSLB analysis and do not represent physical changes to plant equipment.

2. Implementing this analysis does not create the possibility of an accident or malfunction of equipment of a different type than any which have been evaluated previously in the Safety Analysis Report. No new or unique accident precursors have been introduced into the analysis itself or into the design basis as a result of implementing this analysis.

3. The margin of safety as defined in the basis for any technical specification is not reduced by implementation of this change. The requirements for the acceptance criteria and the analysis methodology remain unchanged.

Safety Evaluation Number
95-SE-OT-15

Description Of Activity
UFSAR Chapter 4. Chapter 15
VEP-FRD-42 Rev l-A. "Reload Nuclear Design Methodology" Reload and operation of North Anna Unit 2 Cycle 11. This reload incorporates the following mechanical features described in Technical Report NE-I019 Rev 0 for North Anna Unit 2:
1. Vibration suppression damping assemblies (VSDA) are being placed in V5H fuel assemblies used in baffle locations.
2. Rotation of alternate mixing vane grids in the fresh fuel to suppress assembly vibration.
3. Minor modifications to the fresh fuel to enhance debris resistance including a new protective grid being added directly above the bottom nozzle.

In addition, starting with the fresh feed assemblies for Cycle 11, all North Anna Unit 2 fuel assemblies will be fabricated with fuel cladding, guide thimble tubes, instrumentation tubes and mixing vane grids made from Westinghouse's advanced zirconium alloy, ZIRLO.

Special Conditions
This safety analysis assumes the operation of North Anna Unit 2 Cycle 11 in accordance with the North Anna Power Station Safety Analysis Report (SAR) as updated by the Reload Safety Evaluation of Technical Report NE-1019.

In addition. this safety analysis assumes:

1. Steam generator replacement for North Anna Unit 2 as described in Design Change Package 93-011 has been performed.

2. Implementation of revised main steamline break analysis as documented in Nuclear Analysis & Fuel Safety Evaluation No. 950006, April 1995.

3. Operation at a maximum core thermal power not exceeding 2893 MWt.

4. Cycle 11 burnup will not exceed 21,100 MWD/MTU for EOC10 = 17,600 MWD/MTU, or 19,700 MWD/MTU for EOC10 = 21,100 MWD/MTU. This limit includes approximately 3,000 MWD/MTU of power coastdown from end of full power reactivity. The actual EOC10 burnup is 19,263 MWD/MTU.

5. There is adherence to plant operating limitations as stated in the Technical Specifications and the Core Operating Limits Report.

6. Maximum FQ of 2.19 as modified by K(Z) is not exceeded.

7. RCCA fully withdrawn position of 225 steps.

8. Use of ZIRLO material in place of Zircaloy-4 in fresh fuel cladding and skeleton components.

Safety Evaluation Summary
A safety evaluation has been performed to determine whether an unreviewed safety question will result from the refueling and operation of North Anna Unit 2 Cycle 11. In this evaluation. reload cycle parameters have been calculated and compared to the existing

safety analysis assumptions. These parameters have been shown to be either 1) explicitly bounded. or 2) accommodated by existing safety analysis margin and/or conservatism.

The North Anna main steamline break event has been reanalyzed recently, and documented under a separate 50.59 evaluation. The results of the MSLB reanalysis are being implemented starting with this reload cycle.

This safety evaluation assumes steam generator replacement in Unit 2 as per DCP 93-011. It has been determined that the current analysis basis does not change as a result of steam generator replacement as the Technical Specification minimum measured flow will remain the same as in the previous
cycle.

The impact of the following mechanical design changes and use of ZIRLO material in the reload fuel have been accounted for in the appropriate evaluations performed for Cycle 11:

1. Vibration suppression damping assemblies being placed in Vantage 5H fuel assemblies used in core baffle locations.

2. Rotation of alternate mixing vane grids for the fresh fuel to suppress assembly vibration.

3. Minor modifications to the fresh fuel to enhance debris resistance, including a new protective grid being added directly above the bottom nozzle.

4. Minor dimensional changes in the fuel rod and fuel assembly design due to the use of ZIRLO material for fuel cladding and skeleton components.

The following reload cycle parameters were found to be outside the range of the current safety analysis input assumptions: (i) The most negative Doppler power coefficient vs. power is not bounded by the analysis assumptions below approximately 16% rated power (ii) For operation in a control rod urgent failure condition the predicted maximum FdeltaH for the cycle could be slightly above the normal operation FdeltaH limit for Cycle 11. In accordance with VEP-FRD-42 Rev 1-A, "Reload Nuclear Design Methodology," an evaluation was performed to determine the impact of the parameters on the currently applicable safety analyses.

The Doppler power coefficient (DPC) provides a measure of reactivity feedback that affects transients sensitive to changes in heat flux and reactor power after trip and transients where large power spikes may occur. The unbounded DPC was accommodated by showing that the power-integrated reactivity contribution - the Doppler defect - which is used as input in the safety analyses remains bounded by the input assumptions,

An evaluation performed to address operation in a control rod urgent failure condition shows that, based on known sensitivities between FdeltaH and DNBR margin, the DNBR criterion for the ANS condition II accidents initiated from the given conditions, continues to be met

The results of this evaluation can be summarized as follows:

1. No increase in the probability of occurrence or consequences of an accident will result from this core reload, The reload creates only incremental chances in the values of parameters previously shown to be significant in determining core response to known accidents. Since the currently applicable safety analyses remain bounding for North Anna

Unit 2 Cycle 11, it is concluded that operation with the proposed reload will neither increase the probability of occurrence nor the consequences of initiating events for any known accident.

2. N2C11 reload fuel incorporates changes in mechanical design and in the material used for cladding and skeleton components. However, there have been no changes which would violate currently applicable safety analysis limits. Further, the effects upon system accident response are fully described by the parameters evaluated; and therefore, operation with this core does not create the possibility of an accident of a different type than any which has been evaluated previously in the Safety Analysis Report.

3. The effects of reload parameter variations were accommodated within the conservatism of the assumptions used in the applicable safety analyses. These analyses have demonstrated that calculated results meet all design acceptance criteria as stated in the UFSAR. Therefore, the margin of safety is not reduced for the North Anna Unit 2 Cycle 11 reload.

The conclusions stated above are based in the following:

1. Operation at a maximum core thermal power not exceeding 2893 MWt.

2. Cycle 11 burnup will not exceed 21,100 MWD/MTU for EOC10 = 17,600 MWD/MTU, or 19,700 MWD/MTU for EOC10 = 21,100 MWD/MTU. This limit includes approximately 3,000 MWD/MTU of power coastdown from end of full power reactivity. The actual EOC10 burnup is 19,263 MWD/MTU.

3. There is adherence to plant operating limitations as stated in the Technical Specifications and the Core Operating Limits Report.

4. Maximum FQ of 2.19 as modified by K(Z) is not exceeded.

5. RCCA fully withdrawn position of 225 steps.

6. Use of ZIRLO material in place of Zircaloy-4 in fresh fuel cladding and skeleton components.

7. The North Anna Unit 2 steam generator replacement as described in DCP 93-011 has been performed.

Safety Evaluation Number
95-SE-OT-16

Description Of Activity
Proposed Maintenance on 1-FW-MOV-154A, 154B, and 154C
The fast-acting main feedwater MOVs (1-FW-MOV-154A/B/C) will be de-energized one at a time with the unit at power for modification to the MOV torque switches.

Safety Evaluation Summary
The fast-acting main feedwater MOVs (1-FW-MOV-154A/B/C) will be de-energized one at a time with the unit at power for modification to the MOV torque switches. The torque switches may need resetting to assure proper valve operation due to high measured coefficients of friction. This activity will increase the reliability of the fast-acting MOVs to be able to close during design bases conditions in the future.

The UFSAR describes the operation of the MFW MOVs to isolate feedwater flow to the S/G on an SI signal or a high-high S/G level. De-energizing the MOVs will prevent them from automatically closing and performing their intended isolation function. Only one MOV will be disabled at a time for a short period of time required to perform the torque switch adjustment. Redundancy exists to stop MFW flow by closing the MFRVs and / or tripping the MFW pumps, both of which take place automatically on a feedwater isolation signal.

An SI or Hi-Hi S/G level signal will still isolate main feedwater by tripping the MFW pumps and closing the MFRVs. In addition, the MFRVs will automatically close following a reactor trip and RCS low temperature to limit the RCS cool down. Main feedwater isolation will still occur when required and will occur within the proper time response assumed in the accident analysis. Therefore, no unreviewed safety question exists.

Safety Evaluation Number
95-SE-OT-17

Description Of Activity
Technical Requirements Manual Revision 10.
A. Incorporate CO2 tank requirements, address specific ACTIONS for CO2 tanks, and add new NOTE to 7.1.2.
B. Change ACTION A.1 to hourly firewatch on 7.1.3, delete NOTE on 7.1.4, add halon storage tank pressure to SR 7.1.4.1, add manual actuation to SR 7.1.4.3, delete "For Unit 2" on ACTION A.1.1 on 7.1.5, add ESW-H-2 to table 7.1.5-1.
C. Add table 7.1.6-1 to 7.1.6, add "GR 1.0.3 and GR 1.0.4 are not applicable" to 7.1.6, 7.1.8, 7.2, 7.3, 7.4, 7.5, 7.6, 7.7, 7.8, 7.9, and 7.10.
D. Add two additional wet pipe sprinkler systems and foam systems to table 7.1.8-1.
E. Add table 7.10-1 to 7.10.
F. Make editorial changes for clarification and renumber Section 7 pages to eliminate blank pages and subscripted page numbers.

Safety Evaluation Summary
These proposed Technical Requirements Manual changes will update the Fire Protection requirements to reflect actual plant configuration, add a table for the list of fire protection passive equipment, update surveillance requirements, add that GR 1.0.3 and GR 1.0.4 are not applicable to all the requirements of section 7, correct typographical errors, renumber section 7 pages, provide specific guidance to differentiate when an hourly and continuous firewatch is required in areas protected by gaseous suppression systems that are inoperable, update ACTIONS to be consistent between Unit 1 and Unit 2

It has been determined that an unreviewed safety question does not exist based on the following:

1. Involve an increase in the probability of occurrence of an accident previously evaluated.
There are no hardware modifications associated with these changes. The GR 1.0.3 and 1.0.4 additions are administrative in nature and do no affect system operation. There are no requirements imposed by any regulatory agencies that require the fire protection equipment to be tested or operable for MODE changes to occur. The current requirements ensure that adequate compensatory measures are in affect to provide adequate fire protection. Therefore, these changes do not increase the probability of any accident to occur.

2. Increase the consequences of an accident previously evaluated.
There are no hardware modifications associated with these changes. The GR 1.0.3 and 1.0.4 additions are administrative in nature and do no affect system operation. There are no requirements imposed by any regulatory agencies that require the fire protection equipment to be tested or operable for MODE changes to occur. The current requirements ensure that adequate compensatory measures are in affect to provide adequate fire protection. Therefore, these changes do not increase the consequences of any accident that could occur.

3. Create the possibility for an accident of a different type than was previously evaluated.
No hardware modifications are involved nor is the operation of any equipment affected by the proposed changes. No new accident precursors are being generated by these changes; therefore, these changes do not create the possibility for an accident of a different type than has been previously evaluated.

4. Decrease the margin of safety.
Current fire protection requirements in the Technical Requirements Manual ensure that adequate compensatory measures are in affect to provide adequate fire protection. The margin of safety for the plant is not reduced.

Safety Evaluation Number
95-SE-OT-18

Description Of Activity

North Anna UFSAR
The UFSAR needs to be changed for the following:
1. To provide flexibility to engineers in utilizing computer codes verified in accordance with NRC requirements and are not identified in sections 3.7.3.1, 3.7.3.2 and 3.7.3.3 of UFSAR.
2. To state that personnel other than Stone & Webster have performed analysis of safety related piping systems and/or may perform such analysis in the future.

Safety Evaluation Summary
The reasons for this change are as follows:

1. To provide flexibility to engineers in utilizing computer codes verified in accordance with NRC requirements and are not identified in sections 3.7.3.1, 3.7.3.2 and 3.7.3.3 of UFSAR.

2. To recognize that personnel other than Stone ~ Webster may have performed analysis of safety related piping systems and/or may perform such analysis in the future.

The changes should be allowed for the following reasons:

1. It is necessary to have an alternative computer code(s) to perform analysis of safety related piping systems because NUPIPE - the presently used code - will not be available on a time-sharing basis in near future. Even though we are planning to install NUPIPE on an in-house work station, the availability of additional codes will enhance flexibility and reliability.

2. It will be economical to perform analysis with in-house computer codes.

3. The analyses meeting NRC requirements have been performed by personnel other than Stone and Webster in the past. This will also allow flexibility and will be economical.

Based on Part D it can be seen that all safety questions are reviewed and concluded that they are not affected by

Safety Evaluation Number
95-SE-OT-19

Description Of Activity
Technical Specification Change Request Number 310 (Surveillance Requirements 4.7.1.7.2 for Units 1 and 2) UFSAR Change Request (Chapter 10 / Section 10.2, "Turbine Generator")
The proposed changes would revise Technical Specifications Surveillance Requirements 4.7.1.7.2 to increase the surveillance test interval for the reheat stop and intercept valves from at least once per 31 days to at least once every 18 months and would extend the visual and surface inspection of these valves to 60 months provided there are no indications of operational distress. Also, the criteria for expanding the visual and surface inspections of the turbine control valves based on unacceptable inspection results will be clarified to eliminate unnecessary inspections.

Safety Evaluation Summary
Technical Specifications Surveillance Requirements 4.7.1.7.2 require monthly testing of the turbine overspeed protection system control valves to ensure their operability to prevent overspeeding of the turbine. The proposed changes would reduce the required frequency for testing the turbine reheat stop and intercept valves from at least
- once every 31 days to at least once every 18 months and would extend the visual and surface inspection of these valves to 60 months providing these are no indications of operational distress. Also, the criteria for performing expanded visual and surface inspections of the turbine control valves when unacceptable inspection results are found would be clarified to eliminate unnecessary testing.

A turbine overspeed condition significantly increases the probability of turbine missile generation relative to operation at normal speed due to increased stress in the turbine rotor at higher operating speeds. Regular testing and inspection of the turbine control valves reduces the probability of their failure and the probability of turbine overspeed.

An evaluation of the effects of extending the test and inspection intervals of the reheat stop and intercept valves on turbine missile ejection probability for North Anna Units 1 and 2 was conducted by Westinghouse using the fault tree models and methodology from the Westinghouse Report WCAP-11525, "Probabilistic Evaluation of Reduction in Turbine Valve Test Frequency." WCAP-11525 has been accepted by the NRC. Based upon the results of the site-specific evaluation in their report, "Evaluation of Turbine Missile Ejection Probability Resulting from Extending the Test Interval of Interceptor and Reheat Stop Valves at North Anna Units 1 and 2," it was determined that the total turbine missile ejection probability meets the NRC acceptance criteria. The proposed changes do not increase the overall probability of missile generation or missile damage as described in the UFSAR. Additionally, operating experience and testing have disclosed no significant problems related to the proper operation of the overspeed protection system, and no incidents of turbine reheat stop or intercept valve stem sticking have occurred while the units were carrying load.

The proposed changes have been evaluated and determined to have no adverse impact upon the probability or consequences of any accident previously identified. The proposed changes apply only to the testing and inspection requirements and do not result in any physical alteration to any plant system. Therefore, no new or unique accident precursors are being introduced by this change. The design and operation of the turbine overspeed protection and turbine control systems are not being changed, and the operability of the reheat stop and intercept valves will be demonstrated on a refueling outage basis. The

results of the accident analysis in the UFSAR continue to bound operation under the proposed changes so that there is no safety margin deduction. Based upon the above, it is concluded that the proposed changes do not involve an unreviewed safety question.

Safety Evaluation Number
95-SE-OT-20
Description Of Activity
Technical Specification Change Request No. 328, "Operation at Reduced Power Levels with Inoperable MSSVs"
The proposed change would revise the existing power range neutron flux high setpoints specified in Technical Specifications Table 3.7-1, "Maximum Allowable Power Range Neutron Flux High Setpoints With Inoperable Steam Line Safety Valves During 3 Loop Operation" with conservative setpoints, and delete the information in Table 3.7-1 associated with 2 Loop Operation. The Bases will also be revised to reflect the methodology used.

Safety Evaluation Summary
Technical Specifications Table 3.7-1, "Maximum Allowable Power Range Neutron Flux High Setpoint with Inoperable Steam Line Safety Valves During 3 Loop Operation" allows the operation of the plant with a reduced number of operable Main Steam Safety Valves (MSSVs) at a reduced power level, as determined by the high neutron flux trip setpoints. Westinghouse Electric Corporation identified a potential safety concern in their Nuclear Safety Advisory Letter NSAL-94-001 regarding plant operation within the limits established in Table 3.7-1.

The loss of load/turbine trip (LOL/TT) analysis from full power bounds the case where all MSSVs are operable and no overpressure condition would occur. The analysis was performed with the assumption that futher analysis would not be required for individual cases of inoperable MSSVs since the original full power analysis was determined to bound such an event. This was based upon the assumption that a linearly reduced high flux setpoint would limit the heat addition below the removal capacity of the remaining operable MSSVs. Westinghouse has determined this assumption as invalid, therefore, the LOL/TT analysis may not be bounding for the current allowable operating configurations of Table 3.7-1 when MSSVs are inoperable.

In order to resolve Westinghouse's concern, new calculations were performed such that the maximum power levels allowed for operation with inoperable MSSVs is below the heat removing capability of the operable MSSVs. This represents the most conservative methodology by setting the power range high neutron flux setpoints to this level, thus ensuring that the actual power level cannot exceed this value.

The information in Technical Specifications Table 3.7-1 associated with two loop operation has been deleted since Virginia Electric and Power Company will not operate in this condition.

The proposed changes in Technical Specifications Table 3.7-1 have been evaluated and determined to have no adverse impact upon the probability or consequences of any accident previously identified. No new or unique accident precursors are being introduced by these changes. Since the changes involve revised setpoints only, there are no physical alterations to any plant system and does not create the possibility of any new or different kind of accident that has been evaluated in the UFSAR. The proposed changes will provide conservative power range neutron flux high setpoints that utilized the Westinghouse recommended conservative methodology. Additionally, the design and operation of the main steam system will not be changed. Based upon the above, it is concluded that the proposed changes do not involve an unreviewed safety question.

Safety Evaluation Number
95-SE-OT-20, Rev 1.

Description Of Activity
Technical Specification Change Request No. 328, "Operation at Reduced Power Levels with Inoperable MSSVs"
The proposed change would revise the maximum allowable power range neutron flux high setpoints specified in Technical Specifications Table 3.7-1, "Maximum Allowable Power Range Neutron Flux High Setpoints With Inoperable Steam Line Safety Valves During 3 Loop Operation" with conservative setpoints, and delete the information in Table 3.7-1 and Limiting Condition for Operation 3.7.1.1 Action "b" associated with 2 Loop Operation. The Bases will also be revised to reflect the methodology used.

Safety Evaluation Summary
Technical Specifications Table 3.7-1, "Maximum Allowable Power Range Neutron Flux High Setpoint with Inoperable Steam Line Safety Valves During 3 Loop Operation" allows the operation of the plant with a reduced number of operable Main Steam Safety Valves (MSSVs) at a reduced power level, as determined by the high neutron flux trip setpoints. Westinghouse Electric Corporation identified a potential safety concern in their Nuclear Safety Advisory Letter NSAL-94-001 regarding plant operation within the limits established in Table 3.7-1.

The loss of load/turbine trip (LOL/TT) analysis from full power bounds the case where all MSSVs are operable and no overpressure condition would occur. The analysis was performed with the assumption that futher analysis would not be required for individual cases of inoperable MSSVs since the original full power analysis was determined to bound such an event. This was based upon the assumption that a linearly reduced high flux setpoint would limit the heat addition below the removal capacity of the remaining operable MSSVs. Westinghouse has determined this assumption as invalid, therefore, the LOL/TT analysis may not be bounding for the current allowable operating configurations of Table 3.7-1 when MSSVs are inoperable.

In order to resolve Westinghouse's concern, new calculations were performed such that the maximum power levels allowed for operation with inoperable MSSVs is below the heat removing capability of the operable MSSVs. This represents the most conservative methodology by setting the power range high neutron flux setpoints to this level, thus ensuring that the actual power level cannot exceed this value.

The information in Technical Specifications Table 3.7-1 and Limiting Condition for Operation 3.7.1.1 Action "b" associated with two loop operation will be deleted since Virginia Electric and Power Company will not operate in this condition.

The proposed changes in Technical Specifications Table 3.7-1 have been evaluated and determined to have no adverse impact upon the probability or consequences of any accident previously identified. No new or unique accident precursors are being introduced by these changes. Since the changes involve revised setpoints only, there are no physical alterations to any plant system and does not create the possibility of any new or different kind of accident that has been evaluated in the UFSAR. The proposed changes will provide conservative power range neutron flux high setpoints that utilized the Westinghouse recommended conservative methodology. Additionally, the design and operation of the main steam system will not be changed. Based upon the above, it is concluded that the proposed changes do not involve an unreviewed safety question.

Safety Evaluation Number
95-SE-OT-21

Description Of Activity
Technical Specification Change Request Number 330, Associated UFSAR Change Number FN 95-017. and Associated Technical Requirements Manual Change Revision 11.

This change increases the pressurizer safety valve (PSV) lift setpoint tolerance. Technical Specification 3.4.2 and 3.4.3.1 currently specify a PSV lift setpoint tolerance of +1% in both the as-found and as-left conditions. The TS change increases the Mode 4 PSV lift setpoint tolerance (TS 3.4.2) from +1% as-found and +1% as-left to +3% as-found and +1% as-left. The TS change increases the Modes 1, 2, and 3 PSV lift setpoint tolerance (TS3.4.3.1) from +1% as-found and +1% as-left to +2%/-3% average as-found with no single valve outside +3% as-found and ±1% per valve as-left. The change also reduces the TS requirement for High Pressurizer Pressure Reactor Trip from its current value of 2385 psig setpoint and 2395 allowable value to 2360 psig setpoint and 2370 psig allowable value. This affects TS Table 2.2-1 item 10. In addition, a reduction of the Technical Requirements Manual Table 6.2-1 item 10 (Pressurizer High pressure Reactor Trip Response Time) from 2.0 seconds to 1.0 seconds is required by this change.

Safety Evaluation Summary
The purpose of this safety evaluation is to support an increased pressurizer safety valve (PSV) lift setpoint tolerance. Technical Specification 3.4.2 and 3.4.3.1 specify a PSV lift setpoint tolerance of ± 1% in both the as-found and as-left conditions. It is proposed that the Mode 4 PSV lift setpoint tolerance (TS 3.4.2) be increased from + 1% as-found and + 1% as-left to +3% as-found and + 1% per valve as-left. It is proposed that the Modes 1, 2, and 3 PSV lift setpoint tolerance (TS 3.4.3.1) be increased from ± 1% as-found and ± 1% as-left to + 2 % /-3 % average as-found with no single valve outside ± 3 % as-found and ± 1% per valve as-left. A 3 % tolerance is the maximum permitted by ASME Code Section m, Division 1, Subsection NB, Part 7513, for code safety valves. This increased setpoint tolerance will decrease the likelihood that pertinent accident analyses will have to be reevaluated in the future to resolve violations of Tech Spec setpoint tolerance requirements.

In order to meet the accident analysis acceptance criteria with the revised TS tolerance, two other changes are required:

I) Reduce the TS requirement for High Pressurizer Pressure Reactor Trip from its current value of 2385 psig setpoint and 2395 allowable value to 2360 psig setpoint and 2370 psig allowable value. This affects TS Table 2.2-1 item 10.

2) Reduce the response time requirement for the Pressurizer High Pressure Reactor Trip from 2 seconds to 1 second. This affects the Technical Requirements Manual Table 6.2-1 item 10.

An increased PSV lift setpoint tolerance affects the maximum pressure that will be attained in a system transient which challenges the PSV lift setpoint. Of the North Anna UFSAR Chapter 15 accidents, the Loss of External Load, the Locked Reactor Coolant Pump Rotor, and Rod Withdrawal events (from subcritical and at-power) are the most limiting events in terms of RCS pressurization. Therefore, these events were reanalyzed to determine the impact of an increased PSV lift setpoint tolerance on the overpressure results of these transients. The results of the analyses show acceptable Mode 1,2,3,and 4 overpressure protection, DNB protection, and insignificant impact on normal operation.

In the analyses described above, the PSVs were assumed to open in accordance with the Reference (I) and (2) pressurizer safety valve model, hereafter termed the Westinghouse model. To support the proposed PSV lift setpoint tolerance increase, the PSVs were assumed to begin to open at a pressure 2% above the nominal lift plus an additional 1% 'medium shift' to account for the effects of setting the valves on steam while installing them on a water-filled loop seal installed on a loop seal in accordance with the Westinghouse model. In addition, from Reference 10, an additional delay was assumed for the opening of the valve to simulate purging of the loop seal. Lastly, the time required for the PSV to 'pop' completely open was simulated by application of an additional 0.1% tolerance to the assumed +2% lift setpoint tolerance and + 1% medium shift. The PSVs are assumed to close at a pressure 3 % below the lift pressure (3 % blowdown).

In past analyses, the three PSVs have been modeled as one valve with pressure relief characteristics equivalent to the three valves opening in tandem. In order to confirm the appropriateness of an average PSV tolerance specification, the Loss of Load sensitivity studies were performed modeling three separate valves and using several combinations of valve tolerances averaging +2% . The results of these sensitivities showed that a tolerance of +2%/+2%/+2% is more conservative than any other combination averaging +2%. Therefore, the past practice of modeling the three valves as one valve will remain appropriate under a requirement of a +2% average tolerance.

The RETRAN code was employed in the performance of these transient analyses.

The proposed changes have been reviewed against the criteria of 10 CFR 50.59. This review concluded that these changes raise no unreviewed safety questions. The basis for this determination is as follows:

I) The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated are not increased. Affected safety related parameters were analyzed for a change to North Anna I and 2 Technical Specifications 3.4.2 and 3.4.3 and Table 2.2-1 item 10. It was determined that the overpressure safety limits would not be exceeded in the most limiting overpressure transients (Loss of Load, Locked Rotor, and Rod Withdrawal events) with the pressurizer safety valve lift setpoint positive side tolerance increased to an average of + 2 % . The DNBR results of transients impacted by the proposed setpoint tolerance increase meet the acceptance criterion after accounting for the impact of the proposed changes. The increased setpoint tolerance will not result in an inadvertent opening of the pressurizer safety valves. Mode 4 overpressure protection is adequate with one PSV with a ± 3 % tolerance. Since the affected accidents have been evaluated and found to meet their acceptance criteria with the revised PSV tolerance, the probability of occurrence of an accident or malfunction of equipment important to safety previously evaluated is not increased.

2) The possibility of an accident or malfunction of a different type than any evaluated previously in the safety analysis report is not created. The proposed change to North Anna I and 2 Technical Specifications 3.4.2 and 3.4.3 and Table 2.2-1 item 10 does not involve any alterations to the physical plant which would introduce any new or unique operational modes or accident precursors. Only the allowable as-found tolerance about the existing PSV lift setpoint will be changed, along with a reduction in the pressurizer high pressure reactor trip setpoint.

3) The margin of safety as defined in the basis of the technical specifications is not reduced. It was determined that the most limiting overpressure transients do not result in maximum pressures in excess of the overpressure safety limits. The DNBR results of affected

transients are not made more limiting by the proposed setpoint tolerance increase. Therefore, the margin of safety is unchanged by the proposed increase in the safety valve setpoint tolerance.

Safety Evaluation Number
95-SE-OT-22

Description Of Activity
Technical Specification Change #320
Allow a maximum of two charging pumps to be operable during pump switching operations when RCS temperature is less than 235°F (Unit 1) and 270°F (Unit 2).
Allow methods other than "pull-to-lock" switch position to be used to render a HHSI/LHSI inoperable for the above plant conditions.

Safety Evaluation Summary
These proposed Technical Specifications changes will incorporate SNSOC approved positions on the following items:

1. Methods used to render HHSI/LHSI pumps inoperable during operation in Mode 4 and below.
2. Allow a maximum of two charging pumps to be operable and capable of injecting into the RCS for pump switching during RCS low temperature operations, and
3. Modify the the appropriate basis section of Technical Specifications for the above two items.

Current Technical Specifications are too specific in that they state that the HHSI/LHSI pumps be rendered inoperable by placing the control switches for the pumps in the Pull-to-Lock position. The intent is to place the pump(s) in a condition where they cannot inject into the RCS. Other methods such as tagging the supply breaker or ensuring the discharge valves are closed will also place the pump(s) in the desired condition. The proposed changes will reflect these methods.

During RCS low temperature operations Technical Specifications require that only one charging pump be operable and capable of injecting into the RCS. This would require securing the operable charging pump before placing another pump in service. In this case you violate the Technical Specification requirement for minimum RCS boration flow paths and interrupt the RCS make-up and RCP seal injection. The equivalent specification in NUREG-1431, Standard Technical Specifications - Westinghouse Plants permit a maximum of two charging pumps be operable and capable of injecting into the RCS for pump switching. The proposed changes will allow this same flexibility.

It has been determined that an unreviewed safety question does not exist based on the following:

1. Involve an increase in the probability of occurrence of an accident previously evaluated.
For short periods of time during pump switching operations there will be two charging pumps capable of injecting into the RCS. This evolution will be under the direct administrative control of the operator and will be able to immediately respond to any excessive RCS mass addition (including Safety Injection actuation); therefore, there is no significant increase in the probability for any accident previously evaluated.

2. Increase the consequences of an accident previously evaluated.
For short periods of time during pump switching operations there will be two charging pumps capable of injecting into the RCS. This evolution will be under the direct administrative control of the operator and the operator will be able to immediately respond to any excessive RCS mass addition (including Safety Injection actuation). The ability of

the PORVs to actuate for this accident are not affected; therefore, there is no significant increase in the consequences of any accident previously evaluated.

3. Create the possibility for an accident of a different type than was previously evaluated.
The proposed changes do not involve any hardware modifications or changes in operation of any plant equipment. No new accident precursors are being generated by these proposed changes. These changes are in accordance with NUREG-1431, Standard Technical Specifications - Westinghouse Plants which has been reviewed and approved by the NRC; therefore these changes will not create the possibility for an accident of a different type than was previously evaluated.

4. Decrease the margin of safety as described in the bases section of Technical Specifications.
The basis section of Technical Specifications will be modified to allow two charging pumps to be operable and capable of injecting into the RCS during pump switching operations. These changes are in accordance with NUREG-1431, Standard Technical Specifications - Westinghouse Plants which has been reviewed and approved by the NRC; therefore the margin of safety is not decreased.

Safety Evaluation Number
95-SE-OT-23

Description Of Activity
WO# 321552-01 for 1-TV-IA-102B
An air jumper will be installed around the SOV for 1-IA-TV-102B to maintain the valve open.

Safety Evaluation Summary
While performing 1-PT-213.14, containment isolation Instrument Air valve 1-IA-TV-102B failed to stroke CLOSE on demand. It was determined by Maintenance that the SOV was sticking and needs to be replaced. An air jumper is to be used to maintain the valve OPEN while the SOV is being replaced by Maintenance. Without the jumper installed to maintain IA to containment, IA pressure will decrease. As the air pressure decreases, the inside containment air compressors will attempt to supply adequate discharge pressure. However, there is concern that the compressors can not maintain adequate air pressure so that other containment isolation valves may start to go shut.

The "A" train containment isolation valve, 01-IA-TV-102A, is operable and will provide isolation of the containment penetration in the unlikely event a Phase "B" containment isolation signal is received. The T.S. action for an inoperable containment isolation valve provides for a four hour time period to restore the valve to operable status while maintaining the other penetration isolation valve operable. This requirement will be maintained during the period the block is installed.

No unreviewed safety question exists because one isolation valve for the affected penetration is maintained operable. Since the T.S. LCO will not be exceeded while replacing the SOV, this activity should be allowed.

Note that this safety evaluation is being prepared after the maintenance activity. This complies with the requirements of VPAP-3001 which allows for safety evaluations to be orally approved when a time critical condition exists and time for a formal safety evaluation writeup does not exist. The purpose of documenting this safety evaluation is to comply with the VPAP requirement to complete the paperwork as soon as practical.

Safety Evaluation Number
95-SE-OT-24

Description Of Activity
Operational Quality Assurance Program Topical Report, VEP 1-5A North Anna Power Station Updated Final Safety Analysis Report, Chapter 17
Routine changes to Virginia Power's Operational Quality Assurance Program for North Anna and Surry Power Station. The changes include revisions to the nuclear organizational structure (including some title changes), clarification of record retention requirements for certain procurement documents, and clarifications of QA management's ability to certify personnel assigned to perform QA functions.

Safety Evaluation Summary
A change is being made to the Operational Quality Assurance Program (QA Program) to reflect the current structure and titles within Virginia Power's nuclear organization, clarify that procurement documents with information equivalent to "Purchaser Order (Unpriced) Including Amendments" may be retained in lieu of purchase orders, and clarify that the Managers - Quality Assurance, as well as the Supervisors - Quality, are authorized to certify personnel performing QA functions.

Changes being incorporated into the Operational QA Program also include those associated with Technical Specification changes for the MSRC review of safety evaluations and the SNSOC/management review of procedure changes. The Technical Specification changes (Change Package No. 294A)were already approved by the NRC (refer to TS Amendment 191 for North Anna Unit I and Amendment 172 for North Anna Unit 2, as approved in letter Serial No. 95-243, dated May 2, 1995)

This change neither modifies the nuclear plants (North Anna and Surry) nor changes the procedures which govern the operation and maintenance of those plants.

The likelihood that an accident will occur is neither increased nor decreased by this change to the QA Program. The change (described above) would not be a precursor to or cause of an accident or other previously analyzed accident in the UFSAR. Further, the consequences of a malfunction of equipment important to safety previously evaluated in the UFSAR are not increased by this change, because the programmatic controls associated with the operation, maintenance, and management of the nuclear facilities will continue to meet the obligation to 10 CFR 50, Appendix B. As such, this change to the QA Program does not impact the design or operation of plant equipment.

This change to the QA Program will not produce a new accident scenario or produce a new type of equipment malfunction.

The change does not contribute positively or negatively to the margin of safety. The margin of safety is not impacted by such administrative changes to the QA Program.

Safety Evaluation Number
95-SE-OT-25

Description Of Activity
10CFR Appendix R Report.
This evaluation is being performed to assess the 1995 update of the North Anna 10CFR50 Appendix R Report. It incorporates DCPs completed in 1994 through May 1995, and information concerning other station changes as they pertain to Appendix R. New engineering evaluations were added that address fire barriers. Changes to Appendix R exemption requests were performed to reflect current plant configurations.

Safety Evaluation Summary
The Appendix R Report update is a document update only. The report serves as a mechanism to show compliance with 10CFR50 Appendix R.

The changes to the report fall into three categories editorial or changes to clarify the report changes due to design changes and new and changed engineering evaluations and exemption requests. Editorial changes are not addressed in the safety evaluation since they have no safety impact. Design change package (DCP) changes have already been reviewed during the design change process. This update merely documents previously approved DCP changes. New and changed engineering evaluations and exemption requests are addressed in this evaluation.

The two most significant changes to the Appendix R Report are listed as follows. Evaluation 28 in Chapter 10 was added that addresses the acceptability of 6 foam penetration seals in the floor of the Normal Switchgear Room. Evaluation 29 Chapter 10 was added to address a steel plug fire barrier configuration in the Charging Pump Cubicles. Both of these new fire barrier configurations are location specific and cannot be applied to other areas of the plant without further evaluation.

A change was made to one long standing Appendix R assumption that loss of offsite power should be assumed for any station fire. It has been clarified by the NRC that loss of offsite power need only be considered if the fire scenario has the potential to cause a loss of offsite power or if the fire will require alternate shutdown measures to be imposed. Although this assumption is being removed no other changes are being made to the report that are due to this assumption change.

A change was made to the Preface of Chapter 7 Exemption Requests to allow plenum rated cable to be used in the plant. Exemption 3 was revised to discuss that ECST level indication could also be obtained at the turbine driven auxiliary feedwater pump. Exemption 28 was corrected to state that roof fans were between one and seven feet apart. Exemption 32 was modified to discuss that the service building stairwell is now considered its own fire area.

Minor changes made to Chapter 10 Engineering Evaluations include enhancing table 1-1 in evaluation 1 to include all doors that have been modified that are discussed in exemption requests and engineering evaluations. Evaluation 3 was revised by adding a statement about the classification of containment penetrations Evaluation 22 was clarified to state that there is some safe shutdown equipment located in the Service Building stairwell

A minor change was made to Chapter 11 which addresses penetration seal configurations. References were added to two new configurations that have been approved by Underwriters Laboratory an approved independent test laboratory.

This update does not create any condition not previously analyzed because it is simply documenting commitments and is consistent with information in the current UFSAR and Technical Requirements Manual. Evaluations are performed in accordance with GL 86-10. The level of fire protection for the station is not being diminished and the change will not adversely affect the capability to achieve and maintain safe shutdown in the event of a fire

Safety Evaluation Number
95-SE-OT-26

Description Of Activity
PPR 95-009
The fourth paragraph on page 10.4-15 of the UFSAR concerning operation of the Motor-Driven AFW Pumps and associated NPSH will be removed.

Safety Evaluation Summary
The major issue considered was the capability of the AFW system during specific design basis accidents.

The accidents, previously evaluated in the SAR, and discussed in the subject paragraph are the Loss of Normal Feedwater (LONF) and a Main Steam Line Break (MSLB) on "A" Steam Generator (SG).

The proposed change will remove from the NAPS UFSAR the detailed discussion of the ability of the MDAFWPs to deliver the required flow to the SGs following a MSLB. The paragraph to be deleted includes the predicted AFW flows and the required and available Net Positive Suction Head (NPSHR, NPSHA) to the MDAFWPs when the discharge Pressure Control Valves (PCVs) fail to maintain pump discharge pressure above 750 psig.

The nominal setpoint for the PCVs is currently 900 psig and applicable calculation indicates satisfactory pump performance with this PCV setpoint. Therefore, a detailed discussion of this issue in the UFSAR is unnecessary.

The proposed change will not involve a physical change to plant systems, procedures, or setpoints. The performance of the AFW system will not change. Therefore, the proposed change will not increase the probability or consequences of accidents previously evaluated in the SAR, will not create the possibility of an accident not previously evaluated in the SAR. Further, the proposed change will neither create the possibility of malfunctions of a different type than have been previously evaluated in the SAR, nor increase the probability or consequences of malfunctions previously evaluated in the SAR.

Safety Evaluation Number
95-SE-OT-27

Description Of Activity
UFSAR Change Package for Section 11.3.
Delete the references and description of operation of the Waste Gas Catalytic Recombiner.

Safety Evaluation Summary
The UFSAR currently describes, in detail, the operation of the Waste Gas Decay Tank Thermal Hydrogen Recombiner System. This system, as described, has not been utilized for its intended function of reducing the quantity of gases and the type of gases in the decay tanks. Technical Specifications were amended as requested in Technical Specification change #230 by license amendments 148 and 132 issued by the NRC on 9-25-91 to delete the Technical Specification limitation on hydrogen concentration in the Waste Gas Decay Tanks. The intent of this change was to limit the explosive content of the gas mixture by limiting the oxygen concentration instead of limiting the hydrogen concentration as was the original design concept of the Waste Gas Hydrogen Recombiner. Since the Waste Gas Hydrogen Recombiner is not used, references to the operation of this system should be deleted from the UFSAR.

It is determined that an unreviewed safety question does not exist based on the following:

1. Increase the probability of occurrence for any evaluated accident?

Deleting the reference to the Waste Gas Catalytic Recombiner from the UFSAR is an administrative change and will not result in any plant hardware modifications nor change the operation of any plant system or component. Administrative controls are in place that will take the appropriate corrective action in the advent that a combustible mixture of hydrogen/oxygen are present in the Waste Gas Decay Tanks. Therefore, the probability of occurrence for any analyzed accident is not increased.

2. Increase the consequences for any evaluated accident?

The worst case loading of fission gases is already assumed to exist in the Waste Gas Decay Tanks accident analysis. Deleting the reference to the Waste Gas Catalytic Recombiner from the UFSAR will not have any affect on the quantity of fission gases (e.g., curie content of the tank). Therefore, the consequences of any evaluated accident is not increased
.

3. **Create the possibility for an accident of a different type than was previously evaluated?**

Deleting the reference to the Waste Gas Catalytic Recombiner from the UFSAR is an administrative change and will not result in any plant hardware modifications nor change the operation of any plant system or component. Administrative controls are in place that will take the appropriate corrective action in the advent that a combustible mixture of hydrogen/oxygen gases are present in the Waste Gas Decay Tanks. Therefore, this does not create the possibility for an accident of a different type than has been previously evaluated.

4. **Result in a reduction in the margin of safety as described in the basis section of Technical Specification?**

Technical Specifications addresses the explosive content of the gases contained in the waste gas decay tanks but does not address the waste gas hydrogen recombiner. Although the recombiner was initially installed to prevent an explosive mixture from forming in the decay tanks by minimizing the hydrogen concentration, current operating procedures and Technical Specification limitations on the decay tank oxygen concentration act to prevent and take corrective actions for a potential explosive mixture.

Safety Evaluation Number
95-SE-OT-28

Description Of Activity
UFSAR SECTION 9.2.1.3.2
Delete discussion of Service Water (SW) check valves in the Recirculation Spray Heat Exchanger (RSHX) supply header preventing an operable header from back-feeding an inoperable header. Replace with discussion of SW-MOVs in the cross-ties being powered from different emergency busses.

Safety Evaluation Summary
The Service Water piping upstream of 1-SW-1 14 & 116 and 2-SW-68 & 70 is ASME Class 3, safety related seismic piping is generally considered not subject to a passive failure. This piping is routinely tested in accordance with the ASME Section XI System Pressure Test Program. Catastrophic failure of the pressure boundary is highly unlikely.

Abnormal Procedure 0-AP-12, Loss of Service Water, places no reliance on the check valves at the cross-tie to close. In the unlikely event of the loss of a SW header, the AP directs operators to completely isolate the affected header and place any operating pump(s) on the unaffected header.

The activity is a wording change to UFSAR Section 9.2.1.3.2 in which the discussion of the passive failure of the parallel service water header is being revised. The passive failure discussed placed reliance on check valves installed at the cross-tie when in fact no reliance is placed on these valves. The discussion is to be replaced with a text that discusses the two MOVs in the upstream and downstream crossties around the RSHXs. And how each valve is powered from separate emergency busses, thereby providing redundant isolation in the unlikely event of a header failure.

This activity does not increase the chances of safety related malfunction. The Tech Spec Bases does not include a discussion of the passive failure of the service water pipe. Furthermore, the Single Active Failure Analysis of the Service Water System did not include passive failure, nor did the discussion of UFSAR Chapter 15.

The proposed activity is a wording change and does not involve a test or experiment, nor the ability of the station to achieve and maintain safe shutdown. The activity does not increase the consequences or probability of a Safety Related malfunction. The margin of safety as defined in the Bases Section of the Technical Specifications has not been reduced. Service Header operability is not affected by the UFSAR change.

Safety Evaluation Number
95-SE-OT-29

Description Of Activity
Technical Specifications, Unit 1 and Unit 2; Technical Requirements Manual.
Change the A.C. Sources - Operating specification to permit a single fourteen day outage, once every 18 months, to perform the emergency diesel generator (EDG) preventive maintenance inspection using the alternate A.C. diesel (AAC DG) in place of the EDG during this time. The operability of the AAC DG is defined in the Technical Requirements Manual.

Safety Evaluation Summary
Virginia Electric and Power Company (Virginia Power) proposed to change the North Anna Units 1 and 2 Technical Specifications to allow a single, fourteen (14) day outage for each EDG once every eighteen (18) months during any operating mode (i.e., Modes 1 to 6). The proposed change permits this outage to perform preventive maintenance inspection, appropriate for diesels used for this class of standby service, which requires disassembly of the diesel generator. Currently this inspection is performed during Refueling Outages as required by Technical Specifications Surveillance Requirement 4.8.1.12.d.1. Only one EDG is required to be OPERABLE during Mode 5 and 6.

Virginia Power proposes to change Technical Specification 3.8.1.1 to permit this particular EDG inspection during any operating mode (i.e., Modes 1 to 4). A probabilistic safely assessment (PSA) has been performed which demonstrates that a single outage, fourteen (14) days in length, once every eighteen (18) m0onths for each EDG results in no significant change in core damage frequency assuming adequate compensatory measures are in place. In particular, the AAC DG will be OPERABLE and serve as a replacement source of power during each fourteen day EDG outage when one of the EDGs is out of service.

The compensatory measures include the administrative requirement that the other EDGs and the off-site power supply be OPERABLE during the outage. The AAC DG must also be OPERABLE during the maintenance inspection outage.

Technical Specification 4.0.2, which allows up to 25 % extension on the surveillance interval, is acceptable from a risk perspective. However, performing the surveillance more frequently than once every 18 months while the unit is in modes 1 to 4 is not acceptable from a risk perspective.

The Technical Specifications changes have been reviewed against the criteria of 10 CFR 50.59, and have been determined to result in the finding that an unreviewed safety question does not exist resulting from a violation of any of the three criteria.

a. Operation under the proposed Technical Specifications changes does not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report. The PSA performed to support the proposed Technical Specifications changes show no significant change in the core damage frequency as a result of this modification to permit a single, fourteen (14) day diesel generator outage once every eighteen (18) months per EDG. The proposed changes permit a limited, specific increase in the time that operation in Modes 1, 2, 3 or 4 can occur with only one EDG OPERABLE. As a compensatory measure the AAC DG is assumed to be

OPERABLE during this time. The PSA found no significant increase in core damage frequency resulting from operation in this configuration.

b. The proposed Technical Specifications do not create the possibility of an accident or malfunction of a different type than any evaluated previously in the safety analysis report. The proposed Technical Specifications changes only modify the operability of an EDG for a limited and defined period of time. The UFSAR accidents are analyzed assuming that the EDG is the worst single failure. This assumption is more severe than the proposed Technical Specifications changes which replace the EDG with the AAC DG. Similarly, the PSA performed to evaluate the proposed Technical Specifications changes considered all of the initiating events defined for the PSA performed for the Individual Plant Examination. No new initiations were defined as a result of a review of the PSA model. Therefore, it is concluded that no new or different kind of accident or malfunction from any previously evaluated has been created.

c. The proposed Technical Specifications changes do not result in a reduction in margin of safety as defined in the basis for any Technical Specifications. The PSA were performed to evaluate the concept of a one time outage. The result of the analysis show no significant change in the core damage frequency. As described above, the proposed Technical Specifications changes only modify the operability of an EDG for a limited and defined period of time. Thus, operation with slightly increased EDG unavailability due to maintenance AND the AAC DG OPERABLE is acceptable. The PSA found no significant increase in core damage frequency resulting from operation in this configuration.

Safety Evaluation Number
95-SE-OT-30

Description Of Activity
UFSAR Change Request Engineering Transmittal ME-95-044
UFSAR Table 6.3-2 is to be revised. The stroke time requirements for ECCS MOVs not required to operate on an SI signal are to be changed.

Safety Evaluation Summary
UFSAR Table 6.2-3 lists the stroke time requirements for ECCS valves not required to operate on a safety injection signal. These stroke time requirements are not the same and are more restrictive than the original Westinghouse purchase specification. The purchase specification separated MOVs into slow acting or fast acting with the slow acting valves including some of the ECCS valves not required to operate on an SI. The valves affected by the discrepancy were determined to be the LHSI suction from the containment sump, SI-MOV-1860/2860, LHSI suction from the RWST, SI-MOV-1862/2862, LHSI discharge to the charging pumps, SI-MOV-1863/2863 and the cold lea loop injection isolation valves, SI-MOV-1864/2864. These valves do not meet the requirements of the UFSAR and the table is to be changed to agree with the Westinghouse specification stroke requirements.

The potential for the stroke time for the valves to be increased more without exceeding the revised stroke time limits was evaluated. The cold lea injection valve is normally open which is its safe position in the event of a LOCA. The valve is only operated when swapping from cold lea injection to hot leg injection. The increased stroke time limits will not affect the ability of these valves to perform their safety function.

The other three MOVs are used during the swap over from LHSI pump suction from the RWST to taking suction from the containment sump. The UFSAR specifies a swap over time of less than 240 seconds. (Note: Unit 2 swap over time is currently stated as less than 480 seconds. The unit 1 swap over time was revised after its steam generator replacement. Now that the steam generators have been replaced on unit 2, UFSAR change request #93-011 will revise the unit 2 swap over time to be the same as unit 1.) These valves operate in series so that it is possible to exceed the 240 second swap over time without exceeding the individual valve limits. To prevent this, a 90 second administrative limit is to be applied, via the IST program, to SI-MOV-1860/2860 and SI-MOV-1862/2862.

All accidents which require SI actuation were considered, although long term cooling is only required after a LOCA. Other accidents requiring SI actuation are Steam Generator Tube Rupture, Main Steam Line Break and CRDM Rupture. Malfunctions considered were valve malfunction, pipe failure in suction line and LHSI pump failure to start.

UNREVIEWED SAFETY QUESTION ASSESSMENT

1) Accident probability will not be increased as the SI system is for accident mitigation and has no role in their initiation.

2) Accident consequences are not affected. UFSAR change will not affect the ability of the valves to perform their accident function.

3) No unique accident possibilities are created. The valves are only used in an accident situation and can not create the possibility for one of a different type.

4) Margin of Safety is maintained because the modification does not affect the design requirements of the SI system for accident mitigation.

Safety Evaluation Number
95-SE-OT-31

Description Of Activity
This safety evaluation provides the basis for the documentation revisions needed to support the Casing Cooling tank low level setpoint and safety analysis volume requirement changes.
The minimum required volume in the Casing Cooling tank will be reduced to 100,000 gallons. The Technical Specifications minimum volume of 116,500 will NOT be changed. The EOP setpoint for securing the Casing Cooling system will be increased from a tank level of 3% to 4%. The MOV auto-closure setpoint for 1(2)-RS-MOV-1(2)00A/B will be changed from a tank level of 2.6% to 3.5%.

Safety Evaluation Summary
MAJOR ISSUES

A review of the North Anna Casing Cooling tank level Technical Specification. safety analysis volume requirement, high and low level setpoints and the setpoint uncertainties identified deficiencies in the low level setpoints. The current setpoints are inadequate to ensure containment isolation, via this leakage path, following a CDA.

JUSTIFICATION

Revised low level setpoints have been developed which ensure auto-closure of the pump discharge MOV's on low level and also ensure that the indication will allow operator closure via the EOP's. These new setpoints accommodate the applicable uncertainties. In support of these setpoint revisions, NAF has documented a reduction in the safety analysis required volume from the current 110,000 gallons to 100,000 gallons in Reference 2. Margin analysis has verified that even while ensuring containment isolation on Casing Cooling tank low level, the current Technical Specifications minimum volume will ensure that the Safety Analysis required volume remains available.

UNREVIEWED SAFETY QUESTION ASSESSMENT

The proposed changes to the North Anna Technical Specifications do not pose an unreviewed safety question as defined by 10 CFR 50.59 for the following reasons:

Accident Probability and Consequences. UFSAR Chapter 15 accident probability is not increased. The setpoint and documentation changes have no impact upon the accident precursors. Potential accident consequences are not increased because the containment depressurization criterion has been shown to be satisfied and containment isolation will be ensured.*

Unique Accident Probability. No hardware or procedural changes are made which generate unique accident risk. Thus no new accident probability is created: the scope of the current Chapter 15 accidents remains fully bounding. The proposed setpoint changes simply ensure containment isolation while meeting the depressurization criterion.

Margin of Safety. The revised setpoints will still maintain the required safety analysis volume while ensuring containment isolation: thus the margin of safety as defined in the Technical Specifications will be maintained.

*While these setpoint changes will ensure that the Casing Cooling System is secured on low level, thus isolating a potential leakage path out of containment following a CDA, this requirement has been reviewed and found to be unnecessary. This issue was reviewed at length and documented via memo, T P John (NAPS System Engineering) to J T. Benton, **RS Design Basis Document Revision,** Reference No TJ/0594/018A, NAPS RS File, 5/24/94, and the closure documentation for deviation reports N-94-0038, -0151, -0181, -0189 and -0338.

Safety Evaluation Number
95-SE-OT-32

Description Of Activity
Technical Specification Change Package No. 323.
The proposed changes provide for separate actions for PORVs which are inoperable by reason of a) seat leakage; b) an inoperable automatic pressure control system: c) an inoperable backup nitrogen supply system; and d) all others. A 14 day AOT and a pressure surveillance requirement are added for the backup nitrogen accumulators. Surveillance requirements for the PORV backup nitrogen system are being added.

Safety Evaluation Summary
At North Anna Power Station, the control and indication circuits for the pressurizer power operated relief valves (PORVs) are powered from redundant safety grade emergency buses. Normal motive power for the PORVs is provided by Containment Instrument Air, a highly reliable non-safety-grade system. The PORV's are opened by energizing two solenoid valves which admit air to the actuator to open the valve. De-energizing or loss of power to the solenoid valves causes the PORV to close. The respective solenoid valves are powered from separate 125 VDC emergency buses. High pressure, seismically supported nitrogen tanks (one per PORV) provide a redundant source of motive power to the PORVs.

Each PORV has associated with it a motor operated block valve. The block valves are powered from 480 VAC emergency buses which are not associated with the 125 VDC buses which feed the associated PORV solenoid valves.

On June 25, 1990, the USNRC issued Generic Letter 90-06 (Ref. 1), which addressed the resolution of two Generic Issues. (1) Generic Issue 70, "Power-Operated Relief Valve and Block Valve Reliability" arose out of the awareness that, as operational experience accumulated in the Nuclear Industry, the pressurizer power operated relief valves (PORVs) were being relied on to perform certain safety related functions, e.g. for coping with design basis accidents, although the PORVs were originally designed as non-safety components. (2) Generic Issue 94 involved the reliability of the Low Temperature Overpressure Protection System (LTOPS). The generic letter provided modified Technical Specifications to be used by utilities as a model for enhancing the LTOPS operability requirements.

The technical findings and analyses related to GI 70 were documented in NUREG-1316 (Reference 2). Refs. I & 2 concluded that the use of pressurizer PORVs has evolved such that they are now relied upon to perform one or more of the following safety related functions:

1. Mitigation of a design-basis steam generator tube rupture accident,
2. Low-temperature overpressure protection of the reactor vessel during startup and shutdown, or
3. Plant cooldown in compliance with Branch Technical Position RSB 5-1 to SRP 5.4.7, "Residual Heat Removal (RHR) system.

Item 1 pertains to the ability of the operator to depressurize the RCS in a timely manner in the event of a tube rupture event. Items 2 and 3 relate to overpressure protection of the reactor vessel and RHR systems, respectively.

Based on the analysis and findings for Gl-70, the USNRC staff concluded that while it would not be cost-effective to upgrade existing nonsafety-grade PORVs and associated

control systems to safety grade, certain actions should be taken to improve the reliability of PORVs and block valves. These actions fall in three areas. Utilities should:

1. Include PORVs and block valves within the scope of an operational quality assurance program that is in compliance with 10 CFR Pan 50, Appendix B.
2. Include PORVs, valves in PORV control air systems, and block valves within the scope of a program covered by Subsection IWV, "Inservice Testing of Valves in Nuclear Power Plants," of Section XI of the ASME Boiler and Pressure Vessel Code.
3. For operating PWR plants, modify the limiting conditions of operation (LCO) of PORVs and block valves in the Technical Specifications for Modes 1, 2 and 3 to incorporate the most recent staff licensing position.

   These modified LCOs require a 72 hour shutdown action for one PORV inoperable (except due to excessive seat leakage) and a I hour shutdown action for two PORVs inoperable (except due to excessive seat leakage). For one or both PORV's inoperable due to excessive seat leakage, closure of the associated block valve(s) with power maintained to the block valve(s) is an acceptable action.

The nitrogen accumulators, 1/2-GN-TK-IA and B, for the pressurizer PORVs were initially installed to provide NDT overpressurization protection when the reactor coolant system is water-solid during modes 5 and 6.[3] In response to the events at Three Mile Island, the NRC required a safety backup motive power supply to the PORVs during modes 1-4.[4] The nitrogen accumulator system was modified to meet this commitment by providing a backup to the instrument air system, which normally powers the PORVs.[5]

Thus in the context of the North Anna design, the emergency (backup) power source for the PORVs referred to in the model bases of GL 90-06 is the nitrogen accumulator system (major component: I/2-GN-TK-1A,B).

The GL-90-06 requirements were addressed in a set of proposed Technical Specification changes which were submitted to the NRC in Letter 94-238, dated April 15, 1994. The NRC approved these changes in License Amendments No. 189 and 170 for Units I and 2, respectively, which were issued on October 5, 1994.

The current changes are proposed to address several weaknesses in the existing Specifications. For example, the current specifications require closure of the PORV blocks valves in the case of an inoperable automatic pressure control system, even if there is no seat leakage and manual control remains available This is an inappropriate response. In the proposed specification this has been corrected by providing for separate actions for seat leakage, an inoperable backup nitrogen supply, an inoperable automatic control system, and all other causes.

In addition, provision has been made for a 14 day allowed outage time for PORVs inoperable by virtue of an inoperable backup nitrogen supply in Modes 1-3. The proposed AOT is supported by a probabilistic safety assessment which demonstrates that the impact on overall PORV availability is negligibly small.

Surveillance requirements on the backup nitrogen supply pressure are added based on the recent engineering study of requirements for these components, summarized in NE Technical Report No. 1027, Technical and Operational Basis for Pressurizer PORV

Nitrogen Accumulators, North Anna Power Station Units I and 2, dated June 1995. These requirements were developed based on consideration of the following:

a: the performance requirements [number of strokes and stroke time] established by the three safety related functions discussed above.
b: performance testing performed at NAPS Unit 2 in May 1995 to establish the impact of stroking the valves on nitrogen supply pressure and the impact of supply pressure on valve stroke time.
c: the accuracy of available instrumentation.

A review of the design and performance characteristics of the pressurizer PORV nitrogen accumulators at North Anna has been performed. Major conclusions of this study are as follows:

1. For operational Modes 1-3. the pressurizer PORV performance requirements should be established based on the operator's ability to successfully perform Emergency Operating Procedure E-3, Steam Generator Tube Rupture.

2. To successfully cope with a tube rupture, the nitrogen accumulators should support a minimum of 15 cycles of each PORV. No specific stroke time requirements apply in Modes 1-3.

3. The currently available instrumentation for the nitrogen accumulators has a design accuracy of +/- 85 psi..

4. Based on considerations 1-3, the nitrogen accumulators should be maintained at a minimum pressure of 305 psig for Modes 1-3.

5. In the shutdown modes (4-5), the nitrogen accumulator pressure limits should continue to be set by the Low Temperature Overpressure Mitigating System (LTOPS) design basis.

6. Analyses have confirmed that the original accumulator design basis of 120 strokes is appropriate for performing the LTOPS function for 10 minutes without operator intervention.

7. Test data have shown that the LTOPS stroke open time limit of 2.14 seconds is met for accumulator pressures in excess of 280 psig.

8. Based on the above, for shutdown modes (4-5) where LTOPS is required to be operable, a minimum accumulator pressure of 1660 psig should be maintained .

9. Probabilistic safety assessment techniques have been used to demonstrate that overall pressurizer PORV reliability is relatively insensitive to the presence of the backup motive power source (i.e. nitrogen accumulators), and therefore a realistic allowed outage time of 14 days has been proposed for these components. Based on this assessment, a 14 day allowed outage time for inoperable nitrogen accumulators will be proposed.

REFERENCES:

1) USNRC, Generic Letter 90-06, Resolution of Generic Issue 70, "Power Operated Relief Valve and Block Valve Reliability," and Generic Issue 94, "Additional

Low-Temperature Overpressure Protection for Light Water Reactors", Pursuant to 10 CFR 50.54(f), June 25, 1990.

2) USNRC, NUREG-1316, Technical Findings and Regulatory Analysis Related to Generic Issue 70, Evaluation of Power-Operated Relief Valve and Block Valve Reliability in PWR Nuclear Power Plants, (Enclosure C to GL-90-06).

3) Design Change 7844, Reactor Coolant System Overpressurization Protection Modification.

4) NUREG~0737, Post TMI Requirements, Item II.G.I, Emergency Power for Pressurizer Equipment.

5) Design Change 79-S53, Nitrogen Supply to PORVs Modification.

Safety Evaluation Number
95-SE-OT-33

Description Of Activity
Proposed Technical Specifications Change Request No. 329, Steam Generator Inspection Scope Reduction.
The proposed change affects Table 4.4-1 of Technical Specification Surveillance Requirement 4.4.5.1. The proposed change reduces from two to one the minimum number of steam generators required to be opened and inspected during the first refueling outage after steam generator replacement. The change also removes extraneous information from Table 4.4-1, and removes Note 2 and renumbers the table notes in the associated Table Notations.

Safety Evaluation Summary
The purpose of this change request is to reduce from two to one the minimum number of steam generators required to be opened for steam generator tube examination during the first refueling outage following the Unit 2 steam generator replacement. The minimum number of tubes required to be inspected has not changed, i.e., the minimum tube sample size for one steam generator is twice the minimum required to be inspected in each of two steam generators. Although the actual number of tubes required to be examined remains the same, opening of a second steam generator is considered unnecessary after an operating period of only one fuel cycle of operation.

The requested change in inspection scope will not increase the probability of occurrence for any previously analyzed accident. Each of the replacement steam generators is manufactured and operated essentially identically. The performance history of Westinghouse steam generators (especially the Model F components) has demonstrated that degradation of the tubes is not significant in the early cycles of operation.

The proposed change does not increase the consequences of any accident because the change only reduces the scope of the first inspection after steam generator replacement. Reducing from two to one the minimum number of steam generators required to be opened for tube inspections during a single outage when the steam generators are relatively new cannot contribute to primary-to-secondary leakage during normal or accident conditions. Therefore, any hypothetical tube leak or failure is bounded by the consequences of a postulated steam generator tube rupture event.

Structural integrity of the steam generator tubes is maintained during all plant conditions. The revised inservice inspection program for steam generator tubing will continue to provide adequate detection of steam generator tube degradation and provide assurance of steam generator tube integrity. The proposed change does not introduce any new failure mechanisms or postulated accident affects, i.e., changes to the scope of tube inspection will not act as an accident initiator.

The proposed change to the Technical Specifications does not involve modifications to any of the existing equipment or affect the operation of any existing systems. The current reactor coolant system reliability and operation are maintained in accordance with the descriptions found in the UFSAR. Further, the proposed change does not affect the assumptions, design parameters, or results of any UFSAR accident analysis.

The operability of each steam generator will continue to be verified by the inservice inspection required by the Technical Specifications. The size of the inspected tube

population has not changed and there is no increase in the probability of occurrence of a tube leak or failure. This is not an operability concern for the replacement steam generators.

The margin of safety has not been reduced. Inservice inspection of the same number of tubes is still required -- only the number of steam generators required to be opened for inspection is changed. Structural and leakage integrity of the tubes continues to be verified by inspection during outages and monitoring of the leakage detection systems during operation.

The proposed change would affect only the scope of the first inservice inspection of the steam generator tubes. This proposed change does not affect or change any limiting conditions for operation (LCO) or any other surveillance requirements in the Technical Specifications for North Anna Unit 2.

Safety Evaluation Number
95-SE-OT-34

Description Of Activity
Technical Specifications 3.9.4, 4.9.4. Bases of 3/4.8.1 and 3/4.8.2, Bases of 3/4.9.4 and Bases of 3/4.9.12, facility operating license conditions 2.G for Unit 1 and 2.I for Unit 2 and Virginia Electric and Power Company Response to NRC question 6.72, Amendment 34 to the FSAR, dated March 14. 1975.
The Technical Specifications are being changed to allow both containment personnel airlock doors to remain open during CORE ALTERATIONS or movement of irradiated fuel within containment and to clarify the electrical power system requirements for mitigating the consequences of a Fuel Handling Accident. License conditions 2.G and 2.I are being eliminated because they are no longer required. In addition, the Virginia Electric and Power Company response to NRC question 6.72 is being clarified.

Special Conditions
The Fuel Handling Accident (FHA) safety analysis makes the following assumptions that will have to be confirmed and may require changes to plant operating and surveillance procedures and operator training requirements:

(1) Procedures for handling spent fuel inside containment will have to require continuous communication between the fuel handling coordinator and the control room operator.

(2) The fuel handling coordinator will have to be instructed to immediately notify the control room of any FHA with the potential for release of radioactive material.

(3) An Abnormal Procedure will have to be developed for a FHA that requires immediate operator action upon verbal notification by the fuel handling coordinator of a FHA inside containment or Hi-Hi radiation alarm from the manipulator crane radiation monitor to manually isolate the control room and initiate bottled air.

(4) Operator training for response to a FHA will need to emphasize the importance of isolating the control room within two minutes and evacuating personnel promptly from containment.

(5) Whenever both airlock doors are open during refueling operations, procedures will have to ensure that an operator is stationed near the airlock, aware of the requirements for OPERABILITY of an airlock door, and available to close one personnel airlock door after the containment is evacuated in the event of a FHA.

(6) Surveillance procedures will have to be established to verify operability of an airlock door consistent with the revised specification.

These necessary actions will be assigned through the Station Commitment Tracking System to ensure that they will be met.

Safety Evaluation Summary
North Anna Technical Specifications section 3.9.4 requires that one of the containment personnel airlock doors be closed during core alterations or movement of irradiated fuel in containment. This requires cycling the personnel airlock doors for each containment entry. Frequent containment entries are required while core alterations or fuel movement is in progress and the resulting heavy use of the personnel airlock produces wear and high maintenance requirements. There could be a large number of personnel in containment

during refueling operations and it may take several cycles of the airlock to evacuate personnel from containment if a Fuel Handling Accident were to occur. The time required for these cycling operations would increase personnel doses. A change is being proposed to Technical Specifications section 3.9.4 to allow both doors to remain open during fuel movements or core alterations provided that one door is operable and an individual is available to close the airlock door after personnel are evacuated if a Fuel Handling Accident should occur. This would reduce the maintenance requirements for the airlock doors and the dose to personnel in containment in the event of a Fuel Handling Accident.

While reviewing the licensing basis for the Fuel Handling Accident. it was determined that Virginia Power's response the NRC Question 6.72 discussing conformance with the recommendations in Regulatory Guide 1.52, "Design, Testing, And Maintenance Criteria For Atmosphere Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants," requires clarification. Specifically, Virginia Power indicated that the North Anna Auxiliary Building Ventilation System complies with position C.2.h of Regulatory Guide 1.52. This position states that the power supplies and electrical distribution systems should be designed in accordance with IEEE-308, "Criteria for Class 1E Electrical Systems of Nuclear Power Generation Station." Contrary to this, the fans (HV-F-7A and 7B) in the Fuel Building exhaust system, which are considered by Regulatory Guide 1.52 to be part of the filtration system, are non-safety related and are powered from non-safety related power supplies. In addition, Virginia Power stated that the system complied with Position C.2.c. which states that all components of an engineered-safety-feature system should be designed to Seismic Category I if failure of the component would lead to release of fission products that would result in potential offsite exposures comparable to 10 CFR 100 limits. While some portions of the Fuel Building Ventilation system required to support filtration of the exhaust is not Seismic Category 1, exhaust filtration are not required to ensure that doses are less than 10 CFR 100 limits.

The dose consequences resulting from a Fuel Handling Accident for the North Anna Power Station have been reanalyzed to support the proposed Technical Specifications changes and to help clarify requirements for systems to mitigate a Fuel Handling Accident. The Fuel Handling Accident was evaluated without credit for iodine removal by the charcoal filtration system provided for exhaust gases from the fuel building or credit for isolation of the containment. The calculated dose consequences for the Fuel Handling Accident are within the applicable regulatory limits of GDC 19 and 10 CFR 100 and also meet the SRP 15.7.5 guideline of well within 10 CFR 100 limits (less than 25 % of these limits). If credit is taken for the fuel building filtration system. the thyroid doses for a Fuel Handling Accident in the fuel building would be reduced to 15% of the values calculated without filtration. Isolation of the containment would reduce doses for a Fuel Handling Accident inside containment below the values calculated assuming no isolation. Although it has been determined that some of the responses to 6.72 must be clarified, the as-built plant configuration is consistent with the safety analysis and the revised Fuel Handling Accident analysis shows that all regulatory limits and NRC Standard Review Plan guidelines are met without credit for filtration or radionuclide retention in the fuel building or containment.

Facility license operating conditions 2.G and 2.1 were added to define the analysis submittal establishing the licensing basis for control room doses. These license conditions were added because no other changes were identified as required to resolve an unreviewed safety question identified on control room doses. While this submittal still defines the limiting doses for the control room, these license conditions are no longer necessary because the limiting doses are defined in the UFSAR and in the NRC Safety Evaluation Report. Therefore, the requested changes include elimination of these license conditions.

These activities to support the proposed Technical Specifications changes, to revise the Fuel Handling Accident and to clarify the Virginia Power responses to NRC Question 6.72 do not result in an unreviewed safety question as defined in 10 CFR 50.59 for the following reasons:

No increase in the probability of occurrence of an accident will be created due to any changes in the analysis assumptions or methodologies, or due to the proposed Technical Specifications change to allow both containment personnel airlock doors to remain open during core alterations or fuel movement inside containment. Allowing both personnel airlock doors to remain open during core alterations or fuel movement inside containment will not have any impact on the probability of a Fuel Handling Accident either in containment or in the fuel building. The consequences of the EAB and LPZ doses for a Fuel Handling Accident without credit for iodine filtration have not changed significantly from those shown in the UFSAR for a less conservative analysis that assumes filtration by the fuel building ventilation system. The doses remain well within (<25~.) of the NRC regulatory limits of 10 CFR 100. The predicted control room operator doses have increased from those previously reported to the NRC for a Fuel Handling Accident, but remain bounded by the limiting case for control room doses and within the regulatory limits of General Design Criterion 19. In addition, the action to clarify the responses to NRC question 6.72 will not increase the probability or consequences of the Fuel Handling Accident.

No new accident types or equipment malfunction scenarios are introduced as a result of the clarification to the Virginia Power response to 6.72 or as a result of these changes in analysis methods or the proposed Technical Specifications changes to allow both personnel airlock doors to remain open during core alterations or fuel movement inside containment. Therefore, there is no possibility of an accident of a different type than any previously evaluated in the North Anna UFSAR.

The margin of safety is not reduced. An evaluation of the Fuel Handling Accident doses at the EAB. the LPZ and to control room operators has been performed and it has been concluded that the acceptance criteria defined by GDC-19, 10 CFR 100, and the NRC Standard Review Plan will continue to be met. As noted in NSAC/125, Guidelines for 10 CFR 50.59 Safety Evaluations. the margin of safety is the range between the acceptance limits as defined in the licensing basis and a design failure or other unacceptable condition. As long as the doses remain below the NRC acceptance limits. there is no reduction in the margin of safety. For the control room, operator doses are calculated for five accidents and compared to GDC 19 criteria and SRP section 6.4 guidelines. Even though the control room doses for the Fuel Handling Accident have increased, the limiting case for control room doses is still either a Main Steam Line Break or Steam Generator Tube Rupture. It should be noted that the NRC Safety Evaluation Report (letter from Leon B. Engle to W. L. Stewart dated March 6, 1990) establishes the acceptance limits for control room doses with the following statement:

> "On the basis of the above evaluation, the staff (NRC) concludes that the multiple entries to the NA-1&2 control room will still leave the control room in a safe and habitable condition during and following a design basis accident and provide adequate protection against radiation so that the radiological exposure to the control room operator will be within the limits specified in GDC 19 and within the guidelines provided in SRP Section 6.4. "

For offsite Fuel Handling Accident doses, the NRC Safety Evaluation Report (NUREG-0053, Supplement Number 7, August, 1977) states that "The doses are seen to be well within the guideline values of 10 CFR Pan 100, and are acceptable."

Thus, based on the analysis results for a Fuel Handling Accident we conclude that the calculated doses are within the acceptance limits and that there is no reduction in the margin of safety.

Safety Evaluation Number
95-SE-OT-35

Description Of Activity
North Anna Power Station, UFSAR Sections 15.3.1 and 15.4.1.
Reanalysis of the Large Break Loss of Coolant Accident (LBLOCA) and the Small Break Loss of Coolant Accident (SBLOCA) are being incorporated into the licensing analysis basis for North Anna Units 1 and 2. These reanalysis accommodate a rated core power of 2893 MWt, steam generator tube plugging (SGTP) levels up to 7% in any steam generator, safety analysis FQ limits of 2.19 (LBLOCA) and 2.32 (SBLOCA), safety analysis FdeltaH limits of 1.60 (LBLOCA) and 1.65 (SBLOCA), and safety analysis hot assembly average power limit of 1.45 (LBLOCA).

Safety Evaluation Summary
Reanalysis of the Large Break Loss of Coolant Accident (LBLOCA) and the Small Break Loss of Coolant Accident (SBLOCA) have been performed to support the Upflow Conversion project at North Anna Unit I and documented in Reference 1. The Upflow Conversion project is being performed under a separate safety evaluation/50.59 review. The reanalysis are applicable to both Units 1 and 2.

The reanalysis of the Large Break Loss of Coolant Accident (LBLOCA) has been performed using the Westinghouse LOCA-ECCS Evaluation Model denoted as the 1981 Model with BASH (WCAP-10266-P-A, Reference 2). This model has been used extensively in the North Anna LBLOCA analysis. The analytical techniques are in full compliance with 10CFR50, Appendix K. This evaluation model is an approved methodology in the COLR list (T.S. 6.9.1.7). This analysis employs a revised spacer grid heat transfer model and a hot leg nozzle gap model. Westinghouse has submitted documentation of these model changes to the NRC (References 3 and 4) and is taking credit for these model changes without prior NRC review and approval. Westinghouse has informed NRC of this approach for such model changes in a topical report defining the application of 10CFR50.46 model reporting requirements (Reference 5). In addition, this analysis incorporates an explicit analysis of skewed power shapes (Reference 6).

The reanalysis of the Small Break Loss of Coolant Accident (SBLOCA) has been performed using the Westinghouse LOCA-ECCS SBLOCA Evaluation Model denoted as the 1981 Model with NOTRUMP (WCAP-10054-P-A, Reference 7). This model has been used extensively in the North Anna SBLOCA analysis. The analytical techniques are in full compliance with 10CFR50, Appendix K. This evaluation model is an approved methodology in the COLR list (T.S. 6.9.1.7.). This analysis incorporates safety injection (SI) in the broken loop and the COSI condensation model). Westinghouse has submitted documentation of these model changes to the NRC (WCAP-10054, Addendum 2; Reference 8) and is taking credit for these model changes without prior NRC review and approval. Westinghouse has informed NRC of this approach for such model changes in a topical report defining the application of 10CFR50.46 model reporting requirements (Reference 5).

These reanalysis assumed a rated core power of 2893 MWt, steam generator tube plugging (SGTP) levels up to 7% in any steam generator, safety analysis FQ limits of 2.19 (LBLOCA) and 2.32 (SBLOCA), safety analysis FdeltaH limits of 1.60 (LBLOCA) and 1.65 (SBLOCA), and safety analysis hot assembly average power limit of 1.45 (LBLOCA), and conservative assumptions for accumulator temperature (100°F), and high head safety injection (HHSI) and low head safety injection (LHSI) flow. These

assumptions are equivalent to or conservative with respect to existing limits and plant capabilities.

The analysis results show that the emergency core cooling system will meet the acceptance criteria as presented in 10 CFR 50.46 as follows:

1. The calculated peak fuel rod clad temperature for the limiting case is below the requirements of 2200°F.

2. The amount of fuel element cladding that reacts chemically with water or steam does not exceed 1% of the total amount of Zircaloy in the reactor.

3. The clad temperature transient is terminated at a time when the core is still amenable to cooling. The localized cladding oxidation limit of 17% is not exceeded during or after quenching.

4. The core remains amenable to cooling during and after the break.

5. The core temperature is reduced and the long-term heat is removed for an extended period of time.

The reanalysis demonstrates that the LBLOCA and the SBLOCA meet the requirements of 10CFR50.46. This ensures that the peak clad temperature will be less than the ZIRLO melting point and the core will remain in place and substantially intact with its essential heat transfer geometry preserved. It can be concluded that implementing these analyses will not increase the radiological consequences for this event.

All analysis parameters were equivalent to, or conservative with respect to, those allowed by Technical Specifications limits. All analysis parameters are expected to be conservative with respect to actual plant conditions for North Anna Units I and 2. Continued operation in accordance with the Technical Specifications will not violate the design basis of plant safety equipment. Therefore, no unreviewed safety question is created by continued operation of the North Anna Power Station, Units I and 2 as evidenced by the following:

1. No increase in the probability of occurrence or consequences of an accident or malfunction will result from the implementation of the LBLOCA and SBLOCA analyses since there are no changes to the plant configuration or operating procedures. Furthermore, the reanalysis have demonstrated that calculated results meet all acceptance criteria as established in 10 CFR 50.46. Therefore, implementation of these analyses will not result in more severe consequences than those considered in the SAR.

2. The implementation of these analyses into the North Anna Power Station, Units 1 and 2 design basis can not create the possibility of an accident of a different type than was previously evaluated in the SAR. No changes to plant configuration or mode of operation are implemented by the reanalysis; therefore, no new mechanisms for the initiation of accidents are created by the implementation of these analyses.

3. The reanalysis demonstrated that operation within the assumed conditions will not result in more severe consequences than those considered in the SAR. The reanalysis has demonstrated that calculated results meet all acceptance criteria as established in 10 CFR 50.46. Therefore, the margin of safety has not been reduced by the implementation of these analyses.

REFERENCES

1. Virginia Power Technical Report NE-1044, "Effect of Reactor Vessel Upflow Modification Upon NSSS Accident Analyses, North Anna Power Station Unit 1," October, 1995.

2. WCAP-10266-P-A, "The 1981 Version of the Westinghouse ECCS Evaluation Model Using the BASII Code", March, 1987.

3. Lener from Nick Liparulo (Westinghouse-Manager, Nuclear Safety & Regulatory Activities) to USNRC, "Notification of Changes to the Westinghouse Large Break LOCA ECCS Evaluation Model," ET-NRC-92-3787, December 22, 1992; transmits WCAP-10484, Addendum 1, "Spacer Grid Heat Transfer Effects During Reflood".

4. Lener from Nick Liparulo (Westinghouse-Manager, Nuclear Safety & Regulatory Activities) to USNRC, "Transmittal of Topical Reports WCAP-14404-P and WCAP-14405-NP, 'Methodology for Incorporation Hot Leg Nozzle Gaps into Bash Model'," July, 26, 1995.

5. Lener from N. I. Liparulo (Westinghouse-Manager, Nuclear Safety & Regulatory Activities) to USNRC, "Westinghouse Methodology for Implementation of 10CFR50.46 Reporting," ET-NRC-92-3755, October 30, 1992; transmits WCAP-13451, "Westinghouse Reporting Methodology for Implementation of 10 CFR 50.46 Reporting."

6. Lener from Nick Liparulo (Westinghouse-Manager, Nuclear Safety & Regulatory Activities) to USNRC, "Withdrawal of WCAP-12909-P on Power Shape Sensitivity Model (PSSM)," NTD-NRC-954518, August 7, 1995.

7. WCAP-10054-P-A, "Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code", August, 1985.

8. WCAP-10054, Addendum 2, "Addendum to the Westinghouse Small Break; ECCS Evaluation Model Using the NOTRUMP Code: Safety Injection into the Broken Loop and COSI Condensation Model", August, 1994.

Safety Evaluation Number
95-SE-OT-36

Description Of Activity
Technical Specification Change No. 333
Remove the prescriptive leakage rate testing requirement of 10 CFR 50, Appendix J from the TS Establish a performance-based testing program in accordance with Option B of 10 CFR 50 Appendix and RG 1.163 (which endorses NEI 94-10, July 1995~. Remove the TS reporting requirements in accordance with Option R

Safety Evaluation Summary
The proposed TS change will permit using Option B, of the revised 10 CFR 50, Appendix J . Option B permits performance base testing of the containment and containment penetrations. Performance based testing will significantly reduce the amount of testing and personnel exposure each outage. In addition, removing the reporting will reduce administrative burden. The proposed changed has been reviewed against the criteria of 10 CFR 50.59 and it has been determined that an unreviewed safety questions does not exist for the following reasons:

Plant systems and components will not be operated in a different manner as a result of the proposed TS changes. The proposed changes permit a performance based approach to determining the leakage-rate test frequency for the containment and containment penetrations (Type A, B, and C tests). There are no plant modifications, or changes in methods of operation. Therefore, the change in testing intervals for the containment and containment penetrations has no affect on the probability of occurrence of a LOCA.

The Limiting Conditions for Operations and the leakage-rate acceptance criteria are not being changed. The containment will be operable as assumed in the accident analysis to mitigate the consequences of an accident. Therefore, the consequences of evaluated accidents are not being increased by the change in tests.

Changing the test interval for the containment and containment penetrations does not create any new accident precursors. There are no plant modifications, or changes in methods of operation. Therefore, the possibility for an accident of a different type than was previously evaluated in the Safety Analysis Report is not created by the proposed TS change.

The proposed changes will increase the overall risk to the public by approximately 0.7% and 2.2% (for changes in the frequency of Type A tests and Type B and C tests, respectively) due to the longer intervals between tests. However, this increase in the probability of a malfunction has been judged to be insignificant. This increase has been reviewed and judged to be acceptable by the NRC as documented in NUREG-1493 and the recent change to 10 CFR 50 Appendix J.

Plant systems and components will not be operated in a different manner . The Limiting Conditions for Operation for the containment and the containment penetrations are not changed as a result of the proposed TS changes. The containment and containment penetrations will remain operable (leakage rate less than La and 0.6 La for the containment and the containment penetrations, respectively) to mitigate the consequence of an accident. Therefore, the accident analysis assumption for DBA are unaffected and the margin of safety is not decreased by the proposed TS change.

Safety Evaluation Number
95-SE-OT-37

Description Of Activity
Operational Quality Assurance Program Topical Report, VEP 1-5A.
North Anna Unit I and Unit 2 Updated Final Safety Analysis Report, Chapter 17, Quality Assurance Program.
Surry Unit 1 and Unit 2 Updated Final Safety Analysis Report, Chapter 17, Quality Assurance Program.
The Nuclear Organization has been reorganized. Specifically, the Quality Assurance Department has been eliminated and replaced by Nuclear Oversight, certain Corporate positions have consolidated, quality control inspections and assessments have been relegated to the line organization (rather than being treated as an oversight function), and the description of the Nuclear Organization has been modified to reflect these changes.

Safety Evaluation Summary
A proposed change, test, or experiment shall be deemed to involve an unreviewed safety question (i) if the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report may be increased; or (ii) if a possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report may be created; or (iii) if the margin of safety as defined in the basis for any technical specification is reduced.

The likelihood that an accident will occur is neither increased nor decreased by this change to the Operational Quality Assurance Program. Restructuring the Nuclear Oversight organization and streamlining the independent oversight function to focus on safety significant activities during the operations phase would not be a precursor to nor cause of an accident or other previously analyzed accident in the UFSAR. The consequences of an accident or of a malfunction of equipment important to safety previously evaluated in the UFSAR are not increased by this change, since this change does not reduce the effectiveness of the Operational QA Program. Instead, the change allows Nuclear Oversight (which replaces the Quality Assurance Department) to focus on more safety significant issues during the operations phase through audits and oversight activities as assigned by management. The line organization (Station Nuclear Safety, Maintenance, and Nuclear Site Services) will also focus on the more safety significant issues through nuclear safety assessments and by promoting greater personal accountability in the inspection process. The proposed changes to the Operational QA Program continue to meet the requirements of 10 CFR 50, Appendix B. Therefore, this change to the Operational Quality Assurance Program does not adversely impact the design or operation of plant equipment.

This proposed change to the Operational Quality Assurance Program will not produce a new accident scenario or produce a new type of equipment malfunction since no physical modifications are being made and station operations are not affected.

The Operational Quality Assurance Program does not impact any margin of safety described in the Safety Analysis Report (SAR) documents (i.e., Technical Specifications or UFSAR). The proposed Operational Quality Assurance Program change reflects a change in management philosophy for addressing safety significant issues associated with the operations phase of the nuclear facilities. The philosophy change and restructuring of independent oversight and quality assurance line functions (assessments and inspections) do not reduce the effectiveness of the Operational QA Program (a SAR document which does not describe any margins of safety). Therefore, no margin of safety is impacted by this proposed change.

Safety Evaluation Number
95-SE-OT-38

Description Of Activity
DR N95-1586
Operation without continuous radiation monitoring of Service Water effluent from the four Component Cooling Heat Exchangers as described in UFSAR 11.4.2.8.

Safety Evaluation Summary
UFSAR Section 11.4.2.8 states that the CC Heat Exchanger SW RM continually monitors the SW effluent from the heat exchangers utilizing samples that are individually pumped to the monitor. Currently, and in the past, there have been sample pump and sample line clogging problems which have resulted in poor or no sample flow to the RM, thus rendering the RM inoperable. Efforts are in progress by Engineering to restore the RM, sample lines, and pumps to operable status. The purpose of this Safety Evaluation is to determine whether operation for extended periods of time with this RM out of service creates an Unreviewed Safety Question.

The SW RM was designed to detect radioactive CC leakage from the CC system to the SW system through tube leakage in the CCHXs. Some possible sources of radioactive contamination of the CC system are RHR, Letdown and the RCP thermal barrier. During a CDA, these sources are isolated. Otherwise, the CC system is continuously monitored, and it is normally not contaminated. CCHX tube leakage is maintained at a minimum. CC system leakage is monitored via the CC surge tank level and sampling by 0-OP-51.6 (Sampling For Small Outleakage From The Component Cooling System).

Operation with the RM out of service will not introduce new malfunction scenarios that could lead to contamination of CC or leakage into the SW system. The subject RM is not required to support accident mitigation. Therefore, operation with the RM out of service will not increase the consequences of any accidents. Removing the RM from service will not affect SW effluent to Lake Anna or otherwise affect the environment. Other SW effluent RMs together with sampling capability provide the means to monitor SW effluent and determine any radioactive leakage sources should contamination of the SW system occur.

For these reasons, an unreviewed safety question is not created by operation in this condition.

Safety Evaluation Number
95-SE-OT-39

Description Of Activity
Technical Requirements Manual Change Request 15.
Section 12.2 "EQ Doors" of the Technical Requirements Manual (TRM) shall be modified to clarify the necessary compensatory actions that must be taken when an EQ door will not function as an environmental barrier automatically or with the assistance of the EQ Watch. The TRM does not presently state what actions must be taken in the event that an EQ door will not perform as an environmental barrier. This condition and the required actions shall be added. The existing condition in the TRM that requires an EQ watch to be initiated shall be broadened to include situations where the EQ door will not perform as an environmental barrier automatically.

Safety Evaluation Summary
MAJOR ISSUES

In section 50.49 of title 10 of the Code of Federal Regulations, the NRC established requirements for the environmental qualification of electrical equipment important to safety. Because it was assumed that mild environments do not change into harsh environments, equipment that operates in a mild zone was not included in the NRC equipment qualification requirements.

In July of 1992, the NRC issued Information Notice 92-52. This notice alerted addressees to problems that could result from missing or deteriorated barriers or seals. These seals protect equipment designed to operate in a mild environment from areas that could have a harsh environment following a design basis accident. Information Notice 92-52 also suggested that some licensees may not have procedures to ensure the integrity of barriers between harsh and mild environments.

In an effort to address IN 92-52, Virginia Power classified various doors as EQ doors. These doors function as environmental barriers between areas that could have a harsh environment during a design basis accident and areas that will have mild environments. The changes to the TRM clarify what compensatory measures must be taken if a EQ door cannot provide this barrier function or will not perform this function automatically.

The clarification to the TRM does allow equipment behind an inoperable EQ environmental barrier to remain in operable status if the barrier is restored within the time allotted by a Probabilistic Safety Assessment (PSA). The reason for this is that the increase in core damage frequency is negligible within this time frame.

REASON FOR CHANGE TO BE ALLOWED

The change to the TRM is a clarification only. This change is not a result of new requirements, commitments or a change in the EQ program. It is intended to give the reader of the TRM a clearer understanding of what actions must be taken in the event that an EQ door will not function as an environmental barrier as assumed in the Environmental Zone Description (EZD).

UNREVIEWED SAFETY QUESTION

An unreviewed safety question does not exist for the following reasons:

The probability of occurrence and consequences of a malfunction of equipment in a mild environment has not been increased as a result of this clarification of the TRM. This change to the TRM is not the result of any physical changes to the station or equipment. This clarification does not provide any additional concessions concerning compensatory measures related to EQ doors. The increase in core damage frequency is negligible if an environmental barrier is restored within the allotted time given by a PSA.

This clarification of the TRM has not created the possibility of an accident of a different type than evaluated previously in the SAR. Neither the EQ program nor any EQ documents require revision due to this TRM change.

No margin of safety is being reduced. This change to the TRM is not a result of any physical changes to the station or equipment.

Safety Evaluation Number
95-SE-OT-40

Description Of Activity
Bearing Cooling Water Tower Sludge Classification of Material, Memo from W.A. Thornton to A. H. Stafford dated 11/9/95
Radioactive contaminants have been found in sludge in the Bearing Cooling Water Tower.

Safety Evaluation Summary
Cesium-137 and Cobalt-60 radioactive contamination has been found in sludge in the Bearing Cooling Water Tower. These isotopes are not naturally occurring and, with the exception of some Ce-137 from the fallout of nuclear testing, must have come from operations at North Anna since there are no nuclear sites upstream. The amounts and concentrations of radioactive material in the sludge are low but greater than the calculated concentrations in the lake based on the allowed releases since North Anna went into operation. This is consistent with the most credible mechanism for how the radioactive material got into the Bearing Cooling system: by the addition of lake water as makeup to the Bearing Cooling system. The concentration of the radionuclides increases due to the radionuclides being basically nonvolatile and not being carried away with vapor, and thus building up within the sludge in the BC Tower basin. There are no system interfaces with the Bearing Cooling system other than makeup from the lake which have the potential to carry radioactive contaminants into the Bearing Cooling system in the amounts now detected.

The amount of Cesium-137 is within the range of background levels due primarily from nuclear weapons testing fallout. Pre-operational sediment samples show Cesium-137 levels consistent with fallout. Thus, another possible source for at least the Cesium-137 is fallout and soil runoff into the lake and then into the Bearing Cooling system.

The BC system is systematically removing radioactive material from the lake by acting as the "bottoms tank" in an evaporative process. Thus, the total amount of radioactive material released from the station over the entirety of its operation which is left in the uncontrolled area around the station has actually decreased. The increasing concentration of the radionuclides in the BC shows that the vapor effluent from the BC Tower carries away radioactive material at a rate which is less than the rate of addition into the BC Tower. Since the rate of addition is dependent on the concentrations in the lake, and the concentrations in the lake are dependent on the concentrations of discharges, and the average discharge over the last 17 years has been within limits, the molecular concentration of radioactive material in the vapor effluent from the BC Tower must be less than the molecular concentration in lake water.

Furthermore, the current concentrations of radioactive material in the sludge are less than the 10CFR20 discharge concentration limits: 5.0E-8 uCi/ml vs. 3.0E-6uCi/ml or a fraction of one-sixtieth (1/60) for Co-60 and 1.4E-7 uCi/ml vs. 1.0E-6 uCi/ml or a fraction of one-seventh (1/7) for Ce-137. Thus, if the BC Tower basin ruptured causing a spill which made its way directly into the lake, the effect would be less than that of a maximum concentration discharge.

Clearly, if the process continues the concentration of radioactive material in the BC system will continue to increase. This increase will be monitored by the continued periodic performance of the PT which samples the BC Tower sludge. At some threshold the concentration will have reached a level which will require action such as isolation or removal. This threshold level will be provided by the existing Radiation Protection

Program. However, this threshold level for action will not be reached for some time. Therefore, on both a total amount basis and on a concentration basis, operating the BC Tower in this manner with the buildup of radioactive material poses no additional risk to the health and safety of the public.

The station has been in operation for over 17 years discharging effluent into the lake. Dividing the current concentration of Cobalt-60 in the BC Tower sludge by the concentration in the lake based on the average effluent over the 17 year period, the BC Tower sludge Co-60 concentration is approximately 90 times the lake concentration: 5.0E-8 uCi/ml vs. 5.6E-10 uCi/ml. Likewise, for Cesium-137 the ratio is 110 times: 1.4E-7 uCi/ml vs. 1.3E-9 uCi/ml. Assuming that the tower basin has not been cleaned in the past, it has taken approximately 17 years to increase the concentration in the BC Tower sludge 100-fold. It can be expected to take no more than another 17 years for it to increase another 100-fold.

Safety Evaluation Number
95-SE-OT-41

Description Of Activity
DR 95-1820, UFSAR Change Notice FN 92-130
An evaluation of the condition identified in DR 95-1820 to perform continuity testing of the MSTV SOV's in accordance with Tech Spec requirements which are less frequent than originally written in the UFSAR. This evaluation supports the change to continuity test the MSTV SOV's on a refueling cycle interval versus the original UFSAR stated monthly testing.

Safety Evaluation Summary

An unreviewed Safety Question does not exist for the intent of UFSAR change request No. FN 92-130 (SOV continuity check interval shift from monthly to MSTV functional testing every refueling) for the following reasons:

a) The arrangement of the MSTV actuation SOV's presents multiple paths for initiation of the ESF Main Steam isolation. The arrangement incorporates both Vital Bus powered, dual train initiation. Additionally, redundant initiation methods are available using the 'Appendix R' isolation switch.

b) Since implementation of the refueling interval surveillance to stroke the MSTV and verification of the 5 second stroke time, no failures of the SOV's to actuate have occurred. This demonstrates greater reliability than that which had been seen by the industry in the late 1970's which prompted the 31 day interval requirement.

c) The only viable source of a common mode SOV failure has been eliminated at North Anna with the upgrade of the IA system. Also, no "generic" concerns have been identified with ASCO SOV's that energize to actuate.

d) Safe core operation has been enhanced by the valve testing on a refueling interval by eliminating a likely source of inadvertent or personnel error initiated Reactor trip and Safety Injection from full power operation. The risk of creating this severe transient and its potential detriment to Reactor Safety outweigh any perceived benefit of frequent SOV continuity testing.

In summary, this UFSAR change created no Unreviewed Safety Question in that the change in test interval created no appreciable improvement in the safe and reliable operation of an already highly redundant and reliable ESF which does have great potential for creating a challenge to safe core operation if a testing error were to occur during plant operation.

Safety Evaluation Number
95-SE-OT-42

Description Of Activity
Operational Quality Assurance Program Topical Report, VEP 1-5A Chapter 17 of the NAPS and SPS UFSARs, "Quality Assurance Program".
Revise the statements on pages 17.2-2 and 17.2-34 concerning maintenance and modification activities that are conducted during the operations phase and are comparable to activities conducted during the construction phase. The revision will clarify that the operational phase QA program assures that the technical requirements for these activities will be at least equivalent to the original technical requirements, so that structures, systems, and components will perform satisfactorily in service.

Safety Evaluation Summary
This change clarifies the requirement for determining the technical requirements for maintenance and modifications that are performed during the operational phase and are similar to activities conducted during the construction phase. It requires that the technical requirements be at equivalent to, but not necessarily the same as, the original requirements. For example, this change will allow a reduction in the number of QC hold points from those established during the construction phase. However, the change requires that equivalent means be provided to ensure the performance of the affected structures, systems, and components (such as testing, the use of DR trending results, or the use of equipment reliability data).

This change is consistent with the Operational QA Program, which is defined as those managerial and administrative policies and controls used to assure the safe and reliable operation of the nuclear facilities. It is also consistent with our commitment to ANSI N18.7-1976, "Administrative Controls and Quality Assurance for the Operational Phase of Nuclear Power Plants," Sections 5.2.7 and 5.2.17. The QA Program continues to meet the requirements of 10 CFR 50, Appendix B.

This revision does not involve an assumption, precursor, or a contributing factor for any of the accidents previously analyzed in the Safety Analysis Report. The proposed changes to the Operational Quality Assurance Program do not change the design or operation of any plant system. The program changes are administrative in nature and do not create an unreviewed safety question.

The probability that an accident or a malfunction of equipment will occur is not increased by the proposed QA program changes. This is an administrative clarification that does not change any component, structure, or system operation as described in the UFSAR. The consequences of any accident or malfunction of equipment are not increased by these clarification's to the QA Program.

The probability that a new or different accident will occur is not increased by the proposed QA program changes. The clarification does not create any new method of operation or precursors to any accident identified in the UFSAR.

The proposed changes to QA Program do not impact any margin of safety as described in the Safety Analysis Report (i.e., Technical Specifications, Operating License, or UFSAR). The proposed changes are a clarification to administrative controls. Therefore, no margin of safety is impacted by this change.

Safety Evaluation Number
95-SE-OT-43

Description Of Activity
1) NAPS UFSAR Section 6.2 2) NAPS TS Section 3.6
Removal of containment floor plugs; Reduced casing cooling flowrate; Reduced casing cooling volume; Extended RMT time for LHSI switch over. and provide a sensitivity of the ORS pump NPSH to casing cooling temperature.

Safety Evaluation Summary
A containment sensitivity analysis (Ref 2) has been done to evaluate the impact of the following key changes:

I) Removal of concrete floor plugs
2) Reduced casing cooling available volume to 100,000 gal
3) Reduced casing cooling pump flow to 600 gpm
4) Increased maximum RMT time from 4 min to 5 min
5) Applied TS uncertainties to IRS/ORS timer delays
6) Use TS Hi Hi CLS pressure setpoint of 30 psia
7) Increased casing cooling water temperature to 55 F for the ORS pump NPSH sensitivity

Although the sensitivity analysis used a conservative value of 600 gpm for the casing cooling flowrate, a calculation (Reference 3) has determined that the casing cooling pump can deliver approximately 760 gpm flow to the suction of the ORS pumps. The current Technical Specification allows a maximum of 50 F for the casing cooling temperature however a sensitivity study for ORS pump NPSH impact was done using a 55 F casing cooling temperature. This 55 F casing cooling temperature sensitivity was included to accommodate future increase in the TS limit should the need arise. The sensitivity analysis specifically included the Technical Specification IRS/ORS timer uncertainties which were evaluated previously in Reference 1. The previous Reference I analysis used a casing cooling tank volume of 110,000 gal whereas the new Ref 2 analysis used 100,000 gallons. This reduced volume has no impact on the containment peak pressure and pump NPSH calculations.

The Reference 2 containment sensitivity analysis is not comprehensive but supplements the Reference 1 analysis of record by evaluating the limiting cases which are impacted by the above changes. The results continue to be bounded by the design criteria. The results of these sensitivity cases are listed below.

	Current Values (Ref 2)	Previous Values (Ref 1)	Acceptance Criteria
Peak containment pressure =	43.99 psig	43.93 psig	45 psig
Subatm. peak pressure =	-0.20 psig	-0.04 psig	0.0 psig
ORS pump NPSH margin =	1.67	3.93 ft	0.0 ft
LHSI pump NPSH margin =	1.09 ft	1.2 ft	0.0 ft

From the above results it is seen that the acceptance criteria continue to be met. In addition, the peak containment pressure remained below the Technical Specification value of 44.1 psig previously established for leak rate testing. The analysis margins lost due to the above changes were partially offset by using a higher calculated recirculation spray effectiveness in the Reference 2 analysis. An IRS NPSH calculation was not necessary since there is sufficient margin in the Ref I analysis and the changes made would have little impact on this parameter. The above results are based on the LBLOCA which is more limiting than the MSLB and the SBLOCA for containment peak pressure and pump NPSH calculations. Although the MSLB is used to establish the EQ containment temperature profile, the changes made will have an insignificant impact on the temperature profile because these changes have no impact on the mass/energy blowdown or heat removal systems. In addition the MSLB results are very conservative because the steam generator outlet nozzle Integrated Flow Restrictors were not credited in calculating the EQ envelope.

In conclusion, this change does not constitute an unreviewed safety question because:

1) No increase in the probability of occurrence or consequence of an accident or malfunction of equipment will result from the changes listed above. There are no changes to the plant systems resulting from this reanalysis and the future removal of the concrete floor plugs from the containment will not impact operation of any systems. The consequences are acceptable because the acceptance criteria are met.

2) The implementation of the changes do not create the possibility of an accident or malfunction of equipment of a different type than any which have been evaluated previously in the Safety Analysis Report. No new or unique accident precursors have been introduced.

3) The margin of safety as defined in the basis of the Technical Specifications is not reduced by implementation of this change. The Technical Specifications ensure that the plant conditions will be such that these changes will not challenge the containment integrity. Analysis has shown that the acceptance criteria for the most limiting containment analyses will not be exceeded.

Safety Evaluation Number
95-SE-OT-44

Description Of Activity
UFSAR change
Deleting portion of section 9.5.8.2 referencing the requirement for the diesel generator room logs to require an operator to issue a work order to clean the diesel generator room when necessary.

Safety Evaluation Summary
This UFSAR change deletes a portion of section 9.5.8.2 referencing the requirement for the diesel generator room logs to require an operator to issue a work order to clean the diesel generator room when necessary.

The requirement as it exists is obsolete. General area walkdown requirements for operators are included in OPAP-0006. The following direction is provided: "Abnormal conditions shall be corrected on the spot or the proper corrective action initiated and reported to the Shift Supervisor." The plant housekeeping program is described in VPAP-1301 and requires a Supervisor walkdown of the area at least once per month. Not all occasions require a work order to perform cleaning.

Current plant administrative requirements require that the diesel room be kept clean. These requirements are located in VPAP-1301 and OPAP-0006. Since the diesel rooms will continue to be kept in a high state of cleanliness, diesel operation and performance will not be affected. Therefore, this UFSAR change does not present any unreviewed safety questions and should be allowed.

Safety Evaluation Number
95-SE-OT-45

Description Of Activity
Deviation Report N-95-1933
During implementation of DCP 95-131 for repairing building waterstops to eliminate ground water intrusion, concrete was found deep within the 2" nominal width rattlespace joint between the Auxiliary Building basement floor slab and Unit 1 Quench Spray Pump House (OSPH). The concrete was found at elevation 242'-6", 3" above the expected elevation of the water stop to be repaired by DCP 95-131. The rattlespace is located between column lines G and H outside the "B" Gas Stripper Cubicle at Auxiliary Building elevation 244'-6". This condition is believed to be the result of improper installation of foam board formwork for concrete placement during original plant construction.

Safety Evaluation Summary
The design of expansion joint / rattlespaces between Safety Related Buildings is intended to allow differential seismic and thermal movement between massive concrete structures without the buildings contacting. The rattlespace gap is intended as an analytical simplification to allow buildings to be seismically qualified independent of possible interaction with adjacent buildings.

The existence of non-compressible material (concrete) in the rattlespace between the Auxiliary Building basement floor slab and Unit 1 QSPH between elevation 242'-3" and 246'-6" is acceptable because:

The location of non-compressible material in the rattlespace is approximately 1 foot above the base of the Auxiliary Building and QSPH. At the building foundation, seismic movement of adjacent buildings is the same as the vibratory ground motion during an earthquake. Therefore, differential seismic movement of the buildings is negligible at the location of the concrete in the expansion joint. The ability of the Auxiliary Building and QSPH to withstand an OBE/DBE as stated in the UFSAR is maintained.

The probability of occurrence of an accident or equipment malfunction are not increased. The existence of noncompressible material in the rattlespace between the Auxiliary Building basement floor slab and QSPH has no effect on the probability of an earthquake occurring. The probability of malfunction of Safety Related equipment due to ground water intrusion flooding in the Auxiliary Building has not increased.

The consequences of an accident or equipment malfunction are not increased since structural integrity of the Auxiliary Building and QSPH is maintained. Safety Related components housed within the buildings will be protected and will remain available to function as designed.

The possibility for an accident or equipment malfunction of a different type than was previously evaluated can not be attributed to the existence of non-compressible material in the rattlespace between the Auxiliary Building basement floor slab and QSPH. Overall structural integrity of the Auxiliary Building and QSPH will be maintained. Safety Related components housed within the buildings will be protected and will remain available to function as designed.

Safety Evaluation Number
95-SE-OT-46

Description Of Activity
TEMPORARY SHIELDING REQUEST NOS.96-TSR-005, 96-TSR-010, 96-TSR-012 & 96-TSR-013

Lead blanket shielding is to be placed over selected RCS piping during the 1996 NAPS Unit 1 Refueling Outage;
1. Local segments of PZR spray line, 4 -RC-15-1502-Q1, near PZR spray valves, 1-RC-PCV-1455A & 1455B, (ref. 96-TSR-005) 2. horizontal segments of the 8 diameter loop bypass lines, 8 -RC-11, 12 & 13-1502-Q1, (ref. 96-TSR-010)
3. 2 diameter flow element bypass lines, 2 -RC-44, 45 & 45-1502-Q1, (ref. 96-TSR-012)
4. PZR top nozzle spray line, 4 -RC-15-1502-Q1, (ref. 96-TSR-013)
The above reactor coolant system Temporary Shielding Request (TSR) packages will be in place while fuel is loaded in the reactor vessel. Other RCS TSR's may be in place only while the reactor vessel has been defueled and still other RCS TSR's will be strategically located so that no operable Safety-Related equipment will exist within their collapse envelope, hence these other defueled and strategically located TSR's are not the central issue of this 10 CFR 50.59 Safety evaluation. While not the central issue of this safety evaluation, these other TSR's are still discussed within this safety evaluation for coordination purposes only.

Special Conditions
North Anna Unit 1 shall be in either Mode 5 or 6 while lead blanket shielding, associated with these TSR's, is wrapped around the specified RCS piping. The RCS may be either filled or empty, upstream and/or downstream of the loop stop valves. Design calculations, referenced within, have demonstrated that the associated RCS piping can withstand the additional dead load of the lead blanket shielding, under all applicable Mode 5 and 6 normal and dynamic load conditions, in accordance with NAPS Unit 1 Tech. Spec. 3/4.4.10. Administrative control for these load conditions are implemented via Engineering, SNSOC, ALARA and OPS signoffs in the individual TSR s (Reference Parts 3 & 4 of the Temporary Shielding Engineering Evaluation and Parts 1-7 of the Temporary Shielding Installation, Inspection, and Removal).

Safety Evaluation Summary
MAJOR ISSUES:

The main issue associated with this safety evaluation is the concern for the structural integrity of the RCS piping due to the additional dead weight of lead blanket shielding from these TSR's, in accordance with NAPS Unit 1 Technical Specification 3/4.4.10. Material compatibility of the lead blankets in the normally ambient containment environment is another issue. The four (4) TSR's for which this safety evaluation was written, directly load RCS piping while fuel is still loaded in the reactor vessel. Directly loaded lines include the following:

1. local segments of PZR spray line, 4"-RC-15-1502-Q1, near PZR spray valves, 1-RC-PCV-1455A & 1455B, (ref. 96-TSR-005),
2. horizontal segments of the 8" diameter loop bypass lines, 8"-RC-11, 12 & 13-1502-Q1, (ref. 96-TSR-010),
3. 2" diameter flow element bypass lines, 2"-RC-44, 45 g 45-1502-Q1, (ref. 96-TSR-012), and
4. PZR top nozzle spray line, 4"-RC-15-1502-Q1, (ref. 96-TSR-013).

The RCS may be drained or filled, as desired. Without a special pipe stress evaluation, this additional tributary dead weight constitutes an unreviewed safety question. Hence, all TSR's that directly load RCS piping, while fuel is loaded in the reactor vessel, pose an unreviewed safety question which must be addressed by pipe stress analysis, in a 10 CFR 50.59 Safety Evaluation (Reference Engineering Transmittal No. CEM-95-0060, Rev 0, attached). Administrative control (i.e. installation, temporary supports, removal, etc...) for these TSR's will be achieved as previously mentioned, via Engineering, SNSOC, ALARA and OPS reviews, on Page 1 of 12, Block 7, for each individual TSR.

As mentioned on Page 1 of 12, Block 5, a certain number TSR's may be installed during this outage that will directly load RCS and other Safety-Related piping, but they have been scheduled to be in place only during the window of time that the reactor is defueled. The normal DL + LL stresses, associated with these "defueled" TSR's have been analyzed and compared against the normal DL + LL pipe allowable stress limits. Their individual analyses are referenced on each TSR. A similar analysis will be performed for any additional "defueled" TSR's that may be required. These analyses will not affect the analyses performed for the four (4) "fueled" TSR's of this safety evaluation. Since the reactor will be defueled when these other TSR's are in place, and since RCS pipe stresses still remain within allowable limits, no unreviewed safety question exists. As such, these "defueled" TSR's are not the central issue of this safety evaluation, but are mentioned here for coordination purposes only. Administrative control (i.e. installation, temporary supports, removal, etc...) for these "defueled" TSR's is still coordinated, as previously mentioned, on Page 1 of 12, Block 7, via Engineering, SNSOC, ALARA and OPS reviews for each individual TSR.

Also mentioned on Page 1 of 12, Block 5, other TSR's may be installed, during Modes 5 and 6, in the vicinity of RCS and other Safety-Related piping while fuel may or may not be loaded in the reactor vessel. The location and configuration of these TSR's have been strategically selected so that no operable Safety-Related equipment will exist within their collapse envelopes. An analysis of their collapse envelopes are provided within each individual TSR. Hence, their potential for adversely affecting operable Safety-Related equipment has been eliminated. Any additional TSR's of this type may be added, as required, without affecting the analyses performed for the four (4) "fueled" TSR's of this safety evaluation. As such, this group of TSR's are not the central issue of this safety evaluation, but have been mentioned for coordination purposes only. However, like all other TSR's, administrative control (i.e. installation, temporary supports, removal, etc...) for this class of TSR's is handled in the very same manner as described on Page 1 of 12, Block 7.

All TSR's are designed to perform their radiological shielding function without adversely affecting the nearby equipment, especially operable Safety-Related equipment. Additionally, these TSR's are administratively controlled under VPAP-2105, installed in accordance with approved Health Physics work procedures, and provide an effective ALARA benefit to the personnel that will perform work in these areas during the 1996 NAPS Unit 1 Refueling Outage. ALARA, in coordination with OPS, selects the most appropriate locations for TSR packages, determines the shielding quantities installs the shielding, monitors its effectiveness, and removes the shielding. Design Engineering evaluates the supporting structure for the shielding (i.e. pipe, Tub-Lok, or other plant structures), ensures that it is properly attached and braced (i.e. Calculation No. DEO-0102, R/0, and other referenced calculations) to prevent adverse shifting/falling, and performs a field investigation, as required, to ensure that all design requirements of each TSR have been properly installed in the field. All of these design features are verified and documented on each individual TSR to ensure that no unreviewed safety questions exist.

Safety Evaluation Number
95-SE-OT-47

Description Of Activity
UFSAR Change Request for Section 7.6.7: Pressurizer Relief Valve Flow Indication.
Revise UFSAR Section 7.6.7 to describe an environmentally and seismically qualified acoustic Technology for Energy Corporation (TEC) Valve Monitoring (VMS) System installed per DCP 84-17 & 18 (installed 07-17-86 & 03-28-86 respectively) to replace the previously installed, unqualified, Babcock and Wilcox acoustic VMS System.

Safety Evaluation Summary
This UFSAR Change documents the fact that a seismically and environmentally qualified Technology for Energy (TEC) (Reg. Guide 1.97 (Rev 3) Post Accident Monitoring) Acoustic VMS System, was installed in Units 1& 2 on 07-17-86 and 03-28-86, respectively, replacing an unqualified Babcock & Wilcox Acoustic Monitoring (VMS) System (NUREG-0578), however, the UFSAR Change Notification, included as Appendix 8-3 to DCP 84-17 & 84-18 failed to get incorporated into the UFSAR.

The subject change described the "like-for-like" replacement of a qualified acoustic VMS System protecting the RCS which provided Control Room indication and alarm for Pressurizer Power operated Relief Valves (PCV-1455C/1456) and Safety Valves (SV-1551A, B, and C) during and after accident conditions. It also provides the range of "closed", "not closed" position for the above noted valves.

Since no physical changes to the RCS System or its pressure boundary were required to be made, neither the design, operation, or margin of safety of the RCS System has been affected. This meant that the existing Operating License and Technical Specifications were also not affected as a result of this change.

This change did not constitute an unreviewed safety question since it had not increased the probability or the consequences of accidents previously evaluated, nor had new or unique accident precursors been introduced related to Pressurizer Relief Valve Flow Indication or Safety Valve failure. It had not created the possibility of an accident of a different type, rather, it had improved the reliability of the VMS System to provide visual valve indication based on flow noise in the piping upstream of the SV's or downstream in the case of the PORV's.

This modification was basically a replacement of the existing system, therefore, it could not and has not caused equipment to be exposed to adverse environmental conditions nor has it significantly impacted the FES during the past 9 years that it has been installed. (All inside reactor containment replacement items were environmentally qualified, as required by IEEE-232-1974 and seismically qualified, as required by IEEE-344-1975).

Safety Evaluation Number
95-SE-PROC-01

Description Of Activity
PAR to 1-ECM-2303-01, Rev. 3, Installation and Removal of Temporary Circuit Modifications to Test Run Unit 1 Equipment.
Add an additional Temporary Modification to Attachment 8 that will allow running both 1-FW-P-1A1 and 1-FW-P-1A2 simultaneously.

Safety Evaluation Summary
It is desired to run both motors for the A main feedwater pump simultaneously while uncoupled from the pump. In its present form, 1-ECM-2303-01 does not address this mode of testing. In the maintenance procedure, other electrical jumpers are installed to allow the motors to be individually run uncoupled. If the motors are run uncoupled simultaneously, interlock logic will be made up to open the main feed water recirc flow control valves. This interlock is designed to prevent dead-heading the pump during conditions of low flow. Since the pump and motors are uncoupled, this interlock is not required.

Operation of the main feed recirc valves is not specifically address in the UFSAR but the operation is implied by various figures and design data. This TM will not affect the operation or design of the recirc valves since each main feed pump will be removed from service and recirc flow is not required for safe pump operation. This TM will prevent the recirc valves from opening inadvertently during the testing. Inadvertent opening of the recirc valves could cause the in-service pumps to run out and result in a partial or complete loss of flow to the steam generators.

Feed water pump trips, feed isolation signals, and heat removal capability are not affected in any way by this TM. This TM will prevent the recirc valves from opening inadvertently during testing.

Uncoupled test runs of equipment removed from service is a normal maintenance evolution. Operation of the feed water system will not be affected. This TM does not affect the operation of the recirc valves as is applies to the in-service pumps. Safety functions of the feed water system are not affected in any way. Feed isolation, pump trip logic, and heat removal is not affected.

Safety Evaluation Number
95-SE-PROC-01, Rev 1

Description Of Activity
PAR to 1-ECM-2303-01, Rev. 3, Installation and Removal of Temporary Circuit Modifications to Test Run Unit 1 Equipment, and PAR to 2-ECM-2303-01, Rev. 4, Installation and Removal of Temporary Circuit Modifications to Test Run Unit 2 Equipment.
Add stepson attachments to each Unit's procedure which will allow running the "A", "B", and "C" Main Feedwater inboard and outboard motors coupled to each other but not coupled to their respective pump. Each new attachment is essentially a combination of existing attachments for running an inboard or outboard motor uncoupled individually with an added lead lifted. Therefore the only new jumpers are those lifted leads added to prevent the opening of a feedwater recirc flow control valve and thus prevent a feedwater transient.

Safety Evaluation Summary
It is desired to run both motors for the Unit 1 and Unit 2 "A", "B", and "C" main feedwater pumps simultaneously while uncoupled from the pump. In their present form, 1-ECM-2303-01, Rev. 2 and 2-ECM-2303-01, Rev. 3, do not address this mode of testing. In the maintenance procedure, other electrical jumpers are installed to allow the motors to be individually run uncoupled. If the motors are run uncoupled simultaneously, interlock logic will be made up to open the main feed water recirc flow control valves. This interlock is designed to prevent dead-heading the pump during conditions of low flow. Since the pump and motors are uncoupled, this interlock is not required. Nor is it desired as it will cause a feed water transient.

Operation of the main feed recirc valves is not specifically address in the UFSAR but the operation is implied by various figures and design data. This TM will not affect the operation or design of the recirc valves since each main feed pump will be removed from service and recirc flow is not required for safe pump operation. This TM will prevent the recirc valves from opening inadvertently during the testing. Inadvertent opening of the recirc valves could cause the in-service pumps to run out and result in a partial or complete loss of flow to the steam generators.

Feed water pump trips, feed isolation signals, and heat removal capability are not affected in any way by this TM. This TM will prevent the recirc valves from opening inadvertently during testing.

Uncoupled test runs of equipment removed from service is a normal maintenance evolution. Operation of the feed water system will not be affected. This TM does not affect the operation of the recirc valves as is applies to the in-service pumps. Safety functions of the feed water system are not affected in any way. Feed isolation, pump trip logic, and heat removal is not affected.

Safety Evaluation Number
95-SE-PROC-02

Description Of Activity
1/2-AP-33.1, "Reactor Coolant Pump Seal Failure", Revision 1
The philosophy for dealing with a RCP number 1 seal failure has changed such that a reactor trip, RCP shutdown, and seal leak off valve closure is required within 5 minutes of seal failure. This is different than the previous philosophy of allowing 30 minutes to ramp down the unit prior to tripping the reactor and shutting down the RCP.

Safety Evaluation Summary
The philosophy for dealing with a RCP number 1 seal failure has changed such that a reactor trip, RCP shutdown, and seal leak off valve closure is required within 5 minutes of seal failure. This is different than the previous philosophy of allowing 30 minutes to ramp down the unit prior to tripping the reactor and shutting down the RCP. The AP is being revised to incorporate the suggestions given in NRC Information Notice 93-84, "Determination of Westinghouse Reactor Coolant Pump Seal Failure". This identified flaws with the Braidwood Station's abnormal procedures.

The performance characteristics of the plant on a reactor trip with loss of one RCP is not changed. The procedure change that is being evaluated requires more restrictive guidelines for manually tripping the reactor than were previously given.

Safety Evaluation Number
95-SE-PROC-03

Description Of Activity
1-OP-7.13 R2-P1 Venting Low Head Safety Injection Pump Discharge Lines.
2-OP-7.13 Venting Low Head Safety Injection Pump Discharge Lines. (when revised)
The PAR provides an attachment to the base procedure to conduct depressurized venting of the headers. One pump and header at a time is isolated from the RWST and vented. In the depressurized condition, the air bubbles will expand and migrate to the vent points. Repressurization of the headers is accomplished by a red rubber hose connected from a drain valve on the isolated header's recirculation and test line to a drain valve on the pump discharge. The drain valve on the recirculation and test line remains shut except when actually repressurizing the isolated header.

Safety Evaluation Summary
The safety evaluation is performed for revisions to 1/2-OP-7.13 Venting Low Head Safety Injection Pump Discharge Lines. The PAR provides an attachment to the base procedure to conduct depressurized venting of the headers. One pump and header at a time is isolated from the RWST and vented. In the depressurized condition, the air bubbles will expand and migrate to the vent points. Repressurization of the headers is accomplished by a red rubber hose connected between a drain valve on the isolated header's recirculation and test line and a drain valve on the pump discharge. The drain valve on the recirculation and test line remains shut except when actually repressurizing the isolated header.

The purpose of the PAR is to scavenge as much residual entrained air as possible from the headers to minimize pressure spikes in the header during pump starts.

A Temporary Modification is installed for the repressurization of the vented LHSI header. A temporary hose is connected between the recirculation and test line and the pump discharge header to facilitate repressurization of the isolated header following the venting process. The procedure requires a leak check of the hose and fittings upon installation. Steps exist to shut the affected valves and restore the pipe caps upon removal of the hose.

The activity cannot increase the probability of an accident or malfunction because this activity has no effect on the integrity of the RCS or Main Steam systems.

The activity cannot increase the consequences of an accident or malfunction because one train of LHSI will be maintained available at all times.

The activity cannot create a new accident or malfunction because the activity does not reconfigure any system, structure, or component, such that the design bases for the plant are altered.

The activity should be allowed because it is prudent to remove as much entrained air as possible from the LHSI headers so that pressure spiking will be minimized on pump starts.

Safety Evaluation Number
95-SE-PROC-04

Description Of Activity
1-MOP-55.82 "Turbine Auto Sop Oil Low Pressure Instrumentation"
A one time only PAR was written to verify that Channel II of Auto Stop Oil was in "TRIP" per the Instrument Department's Troubleshooting Sheet for WO 00310389-01.

NOTE: The procedure was performed without prior written approval. The PAR and safety evaluation are being written to document the actions taken.

Safety Evaluation Summary
A one time only PAR was written for 1-MOP-55.82, "Turbine Auto Sop Oil Low Pressure Instrumentation", to verify that Channel II of Auto Stop Oil was in trip per the Instrument Department's Troubleshooting Sheet for WO 00310389-01. The procedure was performed without prior written approval. The PAR and safety evaluation are being written to document the actions taken. These actions were taken in compliance with VPAP-0502 and VPAP-3001.

During performance of Solid State power supply modification DCP 95-118, an unexpected alarm was received when the input breaker from Vital bus II to the Train A solid state cabinets was opened. The Channel II Auto Stop Oil low pressure status light lit and the Auto Stop Oil Pressure Low alarm annunciated. Because the Train A of Solid State was in INHIBIT per the DCP implementing procedure, the alarm was not expected since the pressure circuitry for Train B should have been energized from the Train B input relay cabinet.

AP-3 for instrumentation failures was initiated and the procedure required placing the key switch located at the turbine in TRIP for Channel II of Auto Stop Oil. There was concern that a wire mix-up existed in the field which resulted in the Train A powered pressure switch feeding the Train B Solid State Cabinet. Because of the uncertainty in the configuration of the field wiring, it was decided to verify that the Train B input relay from Auto Stop Oil was actually in the trip condition instead of using the key switch as required by the existing MOP. The Train B input relay was verified to be in the tripped condition within the TS required one hour time frame. Because Train A of Solid State was in INHIBIT, then only Train B was required to be in trip.

The intent of the procedural requirement of AP-3 for placing the "failed" channel in the TRIP condition was met. A low pressure signal from either channel I and/or III of Auto Stop Oil would have resulted in a Reactor Trip as designed. The probability of any malfunctions or accidents were not increased because the plant was operated within its designed parameters at all times. The consequences of any malfunctions or accidents previously analyzed were not affected because all required safety systems were available. The possibility of creating a different malfunction or accident than previously evaluated did not exist because the circuit would have performed as designed upon the receipt of a channel I or III auto stop oil low pressure signal. Therefore, an unreviewed safety question did not exist.

Safety Evaluation Number
95-SE-PROC-05

Description Of Activity
Mechanical Maintenance Procedures 0-MCM-1303-01, 0-MCM-1106-01, 0-MCM-1106-04
The heavy loads handling procedure is being revised to permit use of an auxiliary crane in support of the Unit 2 steam generator replacement (SGR) outage. The proposed procedure change is applicable only during Modes 5 and 6 of the SGR outage. Use of the auxiliary crane during the defueled condition will be in accordance with approved work procedures. The auxiliary crane will be removed from containment at the end of the Unit 2 SGRO.

Special Conditions
The requirements regarding auxiliary crane use and operator qualifications shall be in accordance with those required by ANSI B30.4, as invoked by Construction Procedure C-6. Also, the proposed changes to 0-MCM-1303-01, 1106-01, and 1106-04 will require that the auxiliary crane boom be in a retracted, horizontal position, and the boom directed away from the cavity when the crane is unmanned. Additionally, no loads shall be lifted over the refueling barricade without approval from the Refueling SRO. Formal tracking will be ensured by the existing measures in this procedure such as Crane Operator Qualification/Training, Crane and Hoist Usage Log, Crane/Hoist Operating Permit, Crane/Hoist Operator's Daily Checklist, Pre-Lift Briefing regarding "Restricted Areas", Sign-Off Sheet (at the completion of the job). The proposed procedure change is applicable only during Modes 5 and 6 of the SGR outage. The auxiliary crane will be removed from containment at the end of the Unit 2 SGRO. No additional Limiting conditions or special requirements, specific to the auxiliary crane use, are necessary.

Safety Evaluation Summary
To support steam generator replacement, numerous miscellaneous heavy load handling activities will be performed by the auxiliary crane. The issues considered in this safety evaluation include:

• Could lifting and handling activities performed by the auxiliary crane with fuel in containment result in unacceptable consequences?

• Could a seismic event affect the auxiliary crane resulting in unacceptable consequences?

Upon evaluation of these issues, it was concluded that the auxiliary crane can be used without undue risk to the health and safety of the public and that the activities do not create an unreviewed safety question as defined by 10 CFR 50.59. These conclusions rest on the following major points:

• The auxiliary crane has been designed to meet the Crane Manufacturers Association of America (CMAA) Publication #70 (1983 Edition) and ANSI B30.4-1973. This satisfies NUREG 0612 (Section 5.1.1) regarding the crane design.

The support tower for the auxiliary crane has been designed in accordance with the AISC Manual of Steel Construction, and ACI 349, Appendix B. The crane-tower assembly has been analyzed in an unloaded condition with a retracted and horizontal crane boom to demonstrate that it is capable of withstanding a seismic event without any failure (Bechtel Calculation 22462-C-019). The floor slab at El. 291 -10 has also been evaluated (SWEC Calculation 02072.88-S-4) to be adequate for the loads transferred by the crane-tower assembly.

• The existing requirements of 0-MCM-1303-01, 1106-01, and 1106-04, such as Crane Operator Qualification Sheet, Operator Training, Crane and Hoist Usage Log, Crane/Hoist Operating Permit, Crane/Hoist Operator s Daily Checklist, Pre-Lift Briefing regarding Restricted Areas , and Sign-Off Sheet (at the completion of the job) provide adequate assurance that the crane will be operated in a safe manner. These requirements are in place to meet Virginia Power s commitments to NUREG 0612 (Section 5.1.1). The procedures will require that the auxiliary crane boom be in a retracted, horizontal position, pointed away from the cavity when the crane is unmanned. Additionally, no loads shall be lifted over the refueling barricade without approval from the Refueling SRO. The proposed procedure change is applicable only during Modes 5 and 6 of the SGR outage. The auxiliary crane will be removed from containment at the end of the SGRO. No additional requirements, specific to the auxiliary crane usage, are necessary for miscellaneous heavy load handling operations.

• By complying with the requirements for restricted regions (safe load paths), as specified in procedures 0MCM-1303-01, 1106-01, and 1106-04, the potential risk of impacting equipment important to fuel movement or decay heat removal as a result of a seismic event is minimized.

• The auxiliary crane is equipped with a hydraulic overload safety system which prevents boom lowering or winching up (with any given load) at a radius that creates an overload condition. When this happens, the hydraulic system automatically locks the boom in place and the winch in a brake condition. The operator retains the ability to raise the boom to reduce radius, or to lower the load with the winch.

Safety Evaluation Number
95-SE-PROC-05, Rev. 1

Description Of Activity
Mechanical Maintenance Procedures 0-MCM-1303-01, 0-MCM-1106-01, 0-MCM-1106-04
The heavy loads handling procedure is being revised to permit use of an auxiliary crane in support of the Unit 2 steam generator replacement (SGR) outage. The proposed procedure change is applicable only during Modes 5 and 6 of the SGR outage. Use of the auxiliary crane during the defueled condition will be in accordance with approved work procedures. The auxiliary crane will be removed from containment at the end of the Unit 2 SGRO.

Special Conditions
The requirements regarding auxiliary crane use and operator qualifications shall be in accordance with those required by ANSI B30.4, as invoked by Construction Procedure C-6. Also, the proposed changes to 0-MCM-1303-01, 1106-01, and 1106-04 will require that the auxiliary crane boom be in a retracted position, and the boom directed away from the cavity when the crane is unmanned. Additionally, no loads shall be lifted over the refueling barricade without approval from the Refueling SRO. Format tracking will be ensured by the existing measures in this procedure such as Crane Operator Qualification/Training, Crane and Hoist Usage Log, Crane/Hoist Operating Permit, Crane/Hoist Operator's Daily Checklist, Pre-Lift Briefing regarding "Restricted Areas", Sign-Off Sheet (at the completion of the job). The proposed procedure change is applicable only during Modes 5 and 6 of the SGR outage. The auxiliary crane will be removed from containment at the end of the Unit 2 SGRO. No additional limiting conditions or special requirements, specific to the auxiliary crane use, are necessary.

Safety Evaluation Summary
To support steam generator replacement, numerous miscellaneous heavy load handling activities will be performed by the auxiliary crane. The issues considered in this safety evaluation include:

• Could lifting and handling activities performed by the auxiliary crane with fuel in containment result in unacceptable consequences?

• Could a seismic event affect the auxiliary crane resulting in unacceptable consequences?

Upon evaluation of these issues, it was concluded that the auxiliary crane can be used without undue risk to the health and safety of the public and that the activities do not create an unreviewed safety question as defined by 10 CFR 50.59. These conclusions rest on the following major points:

• The auxiliary crane has been designed to meet the Crane Manufacturers Association of America (CMAA) Publication #70 (1983 Edition) and ANSI B30.4-1973. This satisfies NUREG 0612 (Section 5.1.1) regarding the crane design.

The support tower for the auxiliary crane has been designed in accordance with the AISC Manual of Steel Construction, and ACI 349, Appendix B. The crane-tower assembly has been analyzed in an unloaded condition with a retracted crane boom to demonstrate that it is capable of withstanding a seismic event without any failure (Bechtel Calculations 22462-C-019 and 22462-C-032). The floor slab at El. 291 -10 has also been evaluated (SWEC Calculation 02072.88-S-4) to be adequate for the loads transferred by the crane-tower assembly.

• The existing requirements of 0-MCM-1303-01, 1106-01, and 1106-04, such as Crane Operator qualification Sheet, Operator Training, Crane and Hoist Usage Log, Crane/Hoist Operating Permit, Crane/Hoist Operator s Daily Checklist, Pre-Lift Briefing regarding Restricted Areas , and Sign-Off Sheet (at the completion of the job) provide adequate assurance that the crane will be operated in a safe manner. These requirements are in place to meet Virginia Power s commitments to NUREG 0612 (Section 5.1.1). The procedures will require that the auxiliary crane boom be in a retracted position, pointed away from the cavity when the crane is unmanned. Additionally, no loads shall be lifted over the refueling barricade without approval from the Refueling SRO. The proposed procedure change is applicable only during Modes 5 and 6 of the SGR outage. The auxiliary crane will be removed from containment at the end of the SGRO. No additional requirements, specific to the auxiliary crane usage, are necessary for miscellaneous heavy load handling operations.

• By complying with the requirements for restricted regions (safe load paths), as specified in procedures 0MCM-1303-01, 1106-01, and 1106-04, the potential risk of impacting equipment important to fuel movement or decay heat removal as a result of a seismic event is minimized.

• The auxiliary crane is equipped with a hydraulic overload safety system which prevents boom lowering or winching up (with any given load) at a radius that creates an overload condition. When this happens, the hydraulic system automatically locks the boom in place and the winch in a brake condition. The operator retains the ability to the boom to reduce radius, or to lower tha load with the winch.

Safety Evaluation Number
95-SE-PROC-06

Description Of Activity
PAR to 2-OP-2.2
Number 1 Governor valve to be closed by isolating EHC to 2-MS-GOV-1A.

Safety Evaluation Summary
At a power level of approximately 83 to 86 percent power, the #1 governor valve is 15 to 17 percent open with the #2 and #3 valves in the full open position. At this power level and valve position, number 1 bearing vibrations in the range of 10 to 14 mils have been experienced. Past experience has shown that this vibration can be eliminated by closing the #1 governor valve and opening the #4 governor valve.

This change in valve sequencing can normally be accomplished by using the governor valve test circuitry. By depressing the GV 1 CLOSE pushbutton, the #1 governor valve will go closed and the #4 valve will open to control turbine load.

After the #1 governor valve is closed with the test circuitry, EHC will be isolated from the governor valve to ensure that the valve remains closed. Westinghouse turbine experts have reviewed the turbine design and have determined that operation of the turbine with the #1 governor valve closed is acceptable. There are no safety concerns with long-term operation of the turbine with the #1 governor valve in the closed position.

Safety Evaluation Number
95-SE-PROC-07
Description Of Activity
2-PT-83.5J - Under Voltage / Degraded Voltage Functional Test of Miscellaneous Equipment on 2J Bus.
The procedure is being revised to install a Temporary Modification that will energize the BAST Heater, 1-CH-EHR-8B, to support UV/DV testing.

Safety Evaluation Summary
One purpose of the procedure (2-PT-83.5J) is to verify the load shed/sequence function of the "C" Boric Acid Tank Heater. Previously, this test required an operator to manually override the associated temperature controller such that the heater would be energized when a UV/DV signal is input for the test. The energized heater would then trip off-line during the load shed testing, verifying the proper operation of the circuit. The Temporary Modification proposed for incorporation into the procedure will perform this same function without the need for Operations Department support.

The TM will be installed for a short duration during the performance of the test, and can only result in an increase in "C" BAST temperature. This has the same affect as manually overriding the TC. This result is identical whether the action is performed by jumper or by manual Operator action. This temporary modification does not defeat any portion of the UV/load shed circuitry; it merely enhances the testing process. This TM does not adversely impact any portion of the 1E distribution system, boric acid storage tank, or related systems.

Safety Evaluation Number
95-SE-PROC-08

Description Of Activity
MDAP-0019 for WO-00303634-01-01 (1-MS-IV-1A Test SOV Sticking)
MDAP-0019 for WO-00303635-01-01 (1-MS-IV-1C Test SOV Sticking)
This activity will isolate one line at a time of the LP turbine crossover steam supply piping by closing the Intercept Valve (IV). The IV will be maintained closed by manually closing the IV's high pressure EHC supply isolation valve. The associated IV Test SOV will be repaired / replaced as necessary.

Safety Evaluation Summary
The Test SOVs for the 'A' and 'C' LP Turbine Intercept Valves (IVs) have been sticking during previous Turbine Valve Freedom Testing. The proposed MDAP-0019 is a contingency action if the Test SOVs fail during future Turbine Valve Freedom Testing. This activity will isolate one line at a time of the LP turbine crossover steam supply piping by closing the Intercept Valve (IV). The IV will be maintained closed by manually closing the IV's high pressure EHC supply isolation valve. The associated IV Test SOV will be repaired / replaced as necessary.

Performing this evolution will require that the unit commence a ramp down to 50 percent power. Westinghouse also requires that the cumulative time the unit is in the IV closed condition is less than 80 hours in any one calender year. (See attached correspondence from Westinghouse.)

The turbine valve freedom test is referenced in UFSAR 10.2, stating that detailed testing procedures are given in the Westinghouse Tech Manual. The proposed activity is outside of the test procedure as given in the Tech Manual. Performance of this task requires that the IV is closed and maintained that way by isolation of the EHC to the valve. This change in system status is considered a Temporary Modification to the Turbine Overspeed Protection System.

The major issues considered were the effect of the activity on turbine integrity and the effect of the activity on the turbine overspeed protection. By operating within the rules outlined by Westinghouse in the attached memo, the integrity of the LP turbine assemblies is not adversely impacted. Because the activity will place an IV in the closed condition, the safety function of isolating steam to the LP turbine to prevent overspeed has been accomplished. Therefore, there were no technical problems with the proposed activity.

An unreviewed safety question was determined not to exist because:
(1) The probability of causing an accident or malfunction previously evaluated in the SAR was not increased. The unit will be operated as required by the equipment manufacturer and the ability to prevent overspeed of the turbine has not been decreased.
(2) The consequences of any accident or malfunction described in the SAR are not increased because no fission product barriers are being affected. The possibility of missile generation is not affected because the probability of overspeed was not affected and therefore, there is no increased potential for a missile generated from turbine overspeed to violate one of the fission product barriers.
(3) The probability of creating a different accident or malfunction than previously evaluated in the SAR is not affected because the activity will be performed in a controlled manner utilizing simultaneous verification where appropriate and all required safety functions are being retained during the activity.

Safety Evaluation Number
95-SE-PROC-09

Description Of Activity
Procedure IMP-C-2-RVLIS-01
A temporary spool piece is being placed in the RVLIS system.

Safety Evaluation Summary
A temporary spool piece is being placed in the RVLIS system. This temporary spool piece will allow the RVLIS system to remain operable for a longer time in the outage, therefore allowing RVLIS to be used to monitor RCS level as a backup to the standpipe level indicator which is presently used to monitor RCS level.

The Spool piece will be sized to accommodate similar system pressures and temperatures as the normal RVLIS piping. The operation of the RVLIS system will be unchanged, except to allow use of the system to monitor vessel level for a longer period during the outage.

Since the system design is not being changed and this spool piece will be removed prior to taking the unit above cold shutdown, this activity does not involve any unreviewed safety question.

Safety Evaluation Number
95-SE-PROC-10

Description Of Activity
See the attached list of Unit 1 and Unit 2 ICP's and Channel Functional PT's for Steam Generator Level Protection, Steam Flow/Feed Flow Protection, and Turbine First Stage Pressure as well as the AMSAC ICP, Logic Test, and Functional Test. There are 35 such procedures on each unit, 70 overall.
In each of these procedures, a signal to AMSAC from the channel being tested is rendered inoperable when the channel is placed in test. The change will be to always place AMSAC in manual bypass whenever one of these procedures is performed. Placing AMSAC in manual bypass will make AMSAC inoperable. Placing the system in manual bypass is currently a feature of the AMSAC procedures. However, these procedures are still being included in the scope of this safety evaluation.

Safety Evaluation Summary
See the attached list of Unit 1 and Unit 2 ICP's and Channel Functional PT's for Steam Generator Level Protection, Steam Flow/Feed Flow Protection, and Turbine First Stage Pressure as well as the AMSAC ICP, Logic Test, and Functional Test. There are 35 such procedures on each unit, 70 overall, which are covered by this safety evaluation.

In each of these procedures, a signal to AMSAC from the channel being tested is rendered inoperable when the channel is placed in test. Although the station has not always done so in the past, the change will be to always place AMSAC in manual bypass whenever one of these procedures is performed. Placing AMSAC in manual bypass will make AMSAC inoperable. Placing the system in manual bypass is currently a feature of the AMSAC procedures. However, these procedures are still being included in the scope of this safety evaluation.

For the SG Level and Turbine First Stage Pressure procedures and for the Steam Flow/Feed Flow PT's, AMSAC will be bypassed at the beginning of the procedure and returned to normal at the end of the procedure. However, for the Steam Flow/Feed Flow ICP's AMSAC will be placed in bypass only during the performance of the section of the ICP in which Turbine First Stage Pressure is in test.

Currently, these procedures do not consistently place AMSAC in manual bypass even though they affect system operability. Thus, the twofold purpose of this safety analysis is to establish consistency in actions in response to the inoperability of the system and to demonstrate that no unreviewed safety issue exists for this activity.

AMSAC is described in UFSAR 7.7.1.14. AMSAC is enabled by 2/2 first stage pressures greater than 38 percent. Losing one first stage pressure input when it is put in test would therefore disable AMSAC.

The AMSAC circuitry is designed to be bypassed for the purposes of testing / maintenance on the system. The proposed activity will bypass the circuitry for brief periods while input channels are being tested. SSPS and RPS systems are unaffected (except for the channels being worked, which will be placed in the tripped state) and will continue to be available to initiate automatic reactor trips and ESF actuations.

The system will be bypassed for short periods of time (total time of approximately 10 hours/unit/month). There is no affect on the operability or function of the SSPS / RPS.

For the reasons listed above no unreviewed safety question exists and this activity should be allowed.

Safety Evaluation Number
95-SE-PROC-11

Description Of Activity
MDAP-0019 Supplemental Work Instructions for WO 00315173-01 (1-EP-MT-1A)
The supplemental work instructions are for troubleshooting the 43 selector switch for the 1A main transformer cooling circuit. The steps require removal of the switch cover and manipulation of the switch as necessary to ensure positive switch position. The MDAP also contains contingency step to energize all cooling circuits if needed and this requires installation of a Temporary Modification.

Safety Evaluation Summary
The 43 selector switch on the 1A main transformer cooling circuit is a three position switch that can be selected to "1+3" or "2+4" or "MAN". When the switch was selected to "1+3", the Bank #4 fans would not start. This appears to be a switch problem and troubleshooting is required to identify what (if any) repairs need to be complete.

The troubleshooting procedure removes the cover of the 1A main transformer cooling circuit selector switch. Manipulation of the switch may be required to verify appropriate relays energized or to verify continuity. The procedure also contains contingency steps in case of complete failure of the switch during the troubleshooting evolution. The contingency steps provide for installation of an electrical point-to-point jumper which would energize all cooling circuits. This Temporary Modification requires a Safety Evaluation.

The jumper is to be installed in non-safety related components with no effect on any safety related components. Failure of the jumper could result in gradual overheating of the 1A main transformer which in a worst case would require unit load reduction and opening of the generator output breaker. The jumper will be installed by Control Operations personnel who are familiar with the transformer cooling circuitry. There is no difficult or unusual requirements for installation of the jumper. The jumper will only be used as a contingency action.

Based on the above major issues considered, there is no unreviewed safety question. The ability of the unit to shutdown and remain shutdown in the event of a fire is not affected.

Safety Evaluation Number
95-SE-PROC-12

Description Of Activity
IMP-C-FP-01, Robertshaw FMS-1000 Fire Monitoring System
Procedural steps are being added to remove, and subsequently return to service, smoke and heat detectors in the Unit 2 Containment.

Safety Evaluation Summary
Technical Requirements Manual Section 6.1 and Table 6.1-2 require that heat and smoke detectors be operable in the reactor containment whenever equipment protected by the fire detection instrument is required to be operable. The detection instruments are installed for the reactor coolant pumps, RHR pump area, cable penetration area, and recirc air system. The RHR pumps, cable penetration area, and recirc air system are required to be operable whenever fuel is in the reactor vessel.

During the course of the Unit 2 steam generator replacement outage, a significant amount of cutting, grinding, and welding will be performed in containment. This activity will create numerous smoke and heat alarms in the neighboring areas. The high number of alarms that are expected would decrease the control room Operator's sensitivity to any valid alarms.

The Safety and Loss Prevention Group, as well as the Station Appendix R coordinator, have confirmed that removing the Unit 2 containment smoke and heat alarms from service during the majority of generator replacement is acceptable. All required fire detection will be implemented by fire watches. The revised Robertshaw procedure includes instructions to ensure that the affected detection instrumentation is placed in action, and that they will be returned to service prior to fuel on-load.

The equipment protected by the fire detection instrumentation will not be required to be operable and the probability of accidents is not affected. The provisions of Technical Requirement 6.1 (14 day Action) will be adhered to. No other personnel or equipment safety issues are introduced per Safety & Loss Prevention and the Fire Protection Coordinator. Since there is no unreviewed safety question, and the activity will help the control room Operators maintain their sensitivity to valid fire alarms, this activity should be allowed.

Safety Evaluation Number
95-SE-PROC-13

Description Of Activity
D-NAT-91-010-2-1, Revision 2
Final Design Test Procedure for "Repair / Replacement of 24" Headers To / From Unit 2 RSHX's".
This procedure incorporates the use of a Temporary Modification to hydrostatically test the Service Water headers following installation and removal of welded plugs. The welded plugs are installed / removed under DCP 91-010-2 to allow repair / replacement of the RSHX supply and return headers during the current Unit 2 refueling outage.

Safety Evaluation Summary
Service Water (SW) system corrosion is a known problem at NAPS. As a result, a Service Water piping repair and replacement program has been underway for some time. The Unit 2 RSHX Service Water piping repair / replacement evolution was previously evaluated under DCP 91-010-2. The purpose of this Safety Evaluation is to evaluate the use of a Temporary Modification (hose jumper) to supply the high pressure source for the required hydrostatic testing associated with the DCP activities.

The Condensate (CN) header that runs into the Auxiliary Building will be used as the high pressure supply for the SW header hydrostatic tests. This CN header is normally used for Component Cooling system head tank makeup. This proposed use of the CN system does not present any significant hazard to station operation. The potential impact would be a decrease in Unit 1 CN header pressure. This could result in automatic starting of the standby CN pump. This could also initiate a feedwater train transient. The procedure has precautions about the potential affect on operation. Operations personnel are notified prior to the use of the proposed TM. The possibility that the proposed evolution will alter CN discharge header pressure / flow significantly is remote because the maximum expected makeup flow rate from the CN header will be less than 200 gpm and the normal CN pump flow rates are approximately 8000 gpm per pump (2 normally running).

No unreviewed Safety Question exists because:
1) The probability of occurrence for any accident or malfunction previously analyzed has not been affected. The CN system does not perform any safety related functions and the affected SW system header will be removed from service with appropriate Tech Spec compliance.
2) The consequences of any accident or malfunction previously analyzed are not altered. The ability for the Unit 1 RSHXs to function to maintain containment integrity is not affected. Unit 2 will be shut down during this evolution. CN is not used to maintain any fission product barriers.
3) No new accidents or malfunctions are created because the evolution will be performed in a controlled manner in accordance with a written procedure. The pressures and conditions that are being placed on the affected systems and components are within their design capabilities.

Technical Specifications will be complied with and no Tech Spec basis will be altered by the proposed activity.

Safety Evaluation Number
95-SE-PROC-14

Description Of Activity
2-OP-8.2 (Rev. 14 P1) "Demineralizer Operation"
2-CH-I-3B, Deborating Demineralizer, will be placed in service for RCS purification during start-up with a previously utilized lithium form mixed bed resin until RCS temperature exceeds 350°F.

Safety Evaluation Summary
The CVCS must be maintained capable of reducing the concentration of ionic isotopes in the RCS as required in the design basis. Normally, this is accomplished by passing letdown flow through the mixed bed demineralizers; however, 2-CH-I-1A, Mixed Bed Demineralizer, lithium form mixed bed resin is exhausted, and 2-CH-I-1B is not available. Additionally, use of the new lithium form resin is prohibited until the RCS temperature exceeds 350°F.

2-CH-I-3B, Deborating Demineralizer, will be placed in service for RCS purification during start-up with a previously utilized lithium form mixed bed resin until RCS temperature exceeds 350°F and RCS cleanup has been completed per the Chemistry Department.

The UFSAR discusses operation of the mixed bed and deborating demineralizers; however, it does not address operation of the deborating demineralizer during startup with mixed bed type resin. The demineralizers are provided for use near the end of the core cycle, but they can be used at any time using a hydroxl-based ion exchange resin. The deborating demineralizer will not perform its design function while loaded with mixed bed type resin; however, it is not provided for use during startup operations. Following startup activities, the deborating demineralizer will be returned to its normal standby condition.

The procedure adequately controls the evolution. The greatest concern with this evolution is the potential to dilute the RCS if the resin removes boron form the letdown stream and passes this diluted water back into the RCS. The procedure prevents this from happening in step 4.9.5, which directs the procedure performer to contact Chemistry to determine if the resin bed needs to be flushed to borate the bed. If Chemistry requires a flush (which will be the case the first time the bed is flowed - according to Operations), the bed will be flushed to the stripper until such time that boron concentration in the influent and effluent of the bed is equalized. This will prevent dilution by adequately borating the bed and flushing the piping with borated RCS water.

The operation and design of the CVCS is not altered in such a way as to affect the probability or consequences of any accident. The CVCS is being altered as described in the UFSAR to enhance the RCS cleanup effort during startup activities. Accident response is unaffected. Therefore, no unreviewed safety question exists.

Safety Evaluation Number
95-SE-PROC-14 Rev. 1

Description Of Activity
2-OP-8.2 (Rev. 14 P1) "Demineralizer Operation"
2-CH-I-3B, Deborating Demineralizer, will be placed in service for RCS purification with mixed bed resin.

Safety Evaluation Summary
The CVCS must be maintained capable of reducing the concentration of ionic isotopes in the RCS as required in the design basis. Normally, this is accomplished by passing letdown flow through the mixed bed demineralizers; however, it is not desired to use this lineup at this time.

2-CH-I-3B, Deborating Demineralizer, will be placed in service for RCS purification with a mixed bed resin.

The UFSAR discusses operation of the mixed bed and deborating demineralizers; however, it does not address operation of the deborating demineralizer with mixed bed type resin. The demineralizers are provided for use near the end of the core cycle, but they can be used at any time using a hydroxl-based ion exchange resin. The deborating demineralizer will not perform its design function while loaded with mixed bed type resin; however, it is not provided for use during operation except at end-of-cycle. Prior to reaching end-of-cycle, the deborating demineralizer will be returned to its normal condition, if it is needed.

The procedure adequately controls the evolution. The greatest concern with this evolution is the potential to dilute the RCS if the resin removes boron from the letdown stream and passes this diluted water back into the RCS. The procedure prevents this from happening in step 4.9.5, which directs the procedure performer to contact Chemistry to determine if the resin bed needs to be flushed to borate the bed. If Chemistry requires a flush (which will be the case the first time the bed is flowed - according to Operations), the bed will be flushed to the stripper until such time that boron concentration in the influent and effluent of the bed is equalized. This will prevent dilution by adequately borating the bed and flushing the piping with borated RCS water.

The operation and design of the CVCS is not altered in such a way as to affect the probability or consequences of any accident. The CVCS is being altered as described in the UFSAR to enhance the RCS cleanup effort during startup activities. Accident response is unaffected. Therefore, no unreviewed safety question exists.

Safety Evaluation Number
95-SE-PROC-15

Description Of Activity
DC 92-012-3, Station Blackout Diesel Generator Tie-in to Station
This Safety Evaluation reviews the scope required to verify proper operation after tying the SBO Alternate AC diesel generator and related equipment in to existing plant equipment. The voltage profile test will be run after the 3A relay, which indicates mode of operation to the diesel (parallel or isochronous), has been moved from breaker 05L2 in the Unit 1 normal switchgear room to 0-AAC-PNL-03 in the SBO diesel building. This evaluation supplements prior Safety Evaluations (94-SE-MOD-009, 94-SE-MOD-038, and 95-SE-MOD-019) which were approved and addressed installation of new equipment and tie in. Since all equipment will have been installed, tied into the station, and tested with acceptable results, prior to this evolution, this evaluation addresses ONLY the movement of the 3A relay and the performance of voltage profile acceptance testing in accordance with Test Procedure D-NAT-92-012-3-10.

Special Conditions
Due to the rapid progress of the Unit 2 SGRP outage and the delay in the return to service of the charging system from its original target date, a charging pump was not available to perform this test when the tie ins of SBO to the 'E- Transfer Bus and 'F' Transfer Bus were completed. Additionally since the service water header outages are scheduled to occur beginning about May 7 and charging is due back about May 4, this test may need to be coordinated with the Service Water (SW) outages to assure that this test is not run during an unacceptable alignment in an SW header outages. This is reflected in the initial conditions of the Test Procedure D-NAT-92-012-3-'0. In addition, since this evolution is being addressed as an INFREQUENTLY CONDUCTED OR COMPLEX EVOLUTION ON TEST, Category III per VPAP 0108, a Senior Operations Manager or Operations Manager on call will exercise control to assure compliance. This evolution will only be performed with Unit 2 in an appropriate mode (5, 5, or defueled) and with the other Unit emergency bus operable and carrying all essential loads. The test is based upon testing on 2H with 2J operable. Verify no large 4 KV motor loads operating on station service bus fed from RSST B.

Safety Evaluation Summary
The purpose of this Safety Evaluation is to document that the performance of the voltage profile test utilizing the Unit 2 "H" emergency bus during an acceptable window in the SGRP refueling outage will not cause an unreviewed safety question and will not result in an unacceptable plant operating condition. The evolution addressed is: 1. All SBO related tie ins to the station at Transfer Busses "D", "E" and "F" have been completed, tested, and been found acceptable.

2. The 3A - 0AACN27 relay, which indicates to the diesel its mode of operation (parallel or isochronous), has been moved from breaker 05L2 to 0-AAC-PNL-03 in the SBO diesel building in accordance with DCPs 92-011 and 92-012 and IPRs 1293, 1294, and 95-054.

3. The Unit 2 "J" emergency bus which is <u>not</u> being used for this test is operable and all necessary loads are being powered from this bus. This bus is or is capable of powering the Unit 2 service water pump for a SW header which is in service.

4. The Unit 1 circulating water pumps may be powered from RSST "C" via 15G10 or by their normal feed by RSST "B".

This test will verify acceptable bus voltage on the test Unit 2 emergency bus when: (1) off site power is interrupted to the emergency bus (2) the bus is powered by the SBO AAC diesel generator (3) the motor with the highest starting load (charging pump) is started and the feeder breaker does not trip. (4) the 2H emergency diesel generator is in maunal local, if not defeated

The motor with the highest starting load causes the greatest voltage drop on the bus and minimum bus voltage and therefore the greatest chance to trip the emergency bus undervoltage relays, which are not defeated when the bus is powered by the AAC diesel. This needs to be verified to assure that the undervoltage relay does not trip on the initial start or restart of a charging pump while an emergency bus is fed from the AAC diesel.

This evolution does not cause entry into any Limiting Condition of Operation. Failure of the test could result in: (1) trip of the undervoltage relay for Emergency Bus 2H, (2) opening of breaker 25Hll, (3) start of the 2H emergency diesel generator (EDG), (4) closure of 25H2 and loading the EDG with 2H loads, and (5) operator action to restore off-site power by (a) opening 05Ll, (b) placing the defeat switch for 15E3 in normal, (c) closing 15E1, (d) closing 15E3, and (e) synchronizing across and closing 25H11.

UFSAR Sections 6.3, 6.3.1, 6.3.1.1, 6.3.1.4, 15.2.5 and 15.2.9 were reviewed. The voltage profile testing of emergency bus 2H while fed from the SBO AAC diesel generator does not impact the design basis of the ECCS discussed in Section 6.3, since Unit 2 is in an outage and 2J is the operable bus and Unit 1 is on line with both emergency busses operable. Therefore the only accidents considered were a Partial Loss of Reactor Coolant Flow (12.5.5) and a Loss Of Off-site Power (LOOP) to station auxiliaries (15.2.9). These are not applicable to Unit 2 as the unit is shut down.

This test is being run while 2J is the operable Unit 2 Emergency Bus during the outage. The AAC system is tested by paralleling the SBO diesel to the system through E Transfer Bus, separating 2H powered by the AAC DG from the system, starting a charging pump, stopping it, synchronizing with the system, and shutting down the SBO AAC DG. Properly performed this evolution does not pose a hazard to the power supplies for Unit 1 loads or to required Unit 2 loads.

This voltage profile testing of the Alternate AC power supply will not increase the probability of occurrence of these accidents. The test involves paralleling the SBO AAC DG with the system, separating, testing, synchronizing, and returning to normal alignments.

The probability of a partial or total LOOP or of an equipment malfunction is not increased as (1) all equipment involved has previously been tested and run, (2) during the test there is separation between the test and the balance of the station electrical system, (3) plans for restoration of power to 2H in case of a problem with the test have been made and are included in the test plan even though 2J will be the operable Unit 2 emergency bus to meet Technical Specification during the test, and (4) the power supplies for auxiliaries for Unit 1 are the station service transformers for normal loads, and RSSTs A and C for emergency busses, and RSST B leads or RSST C leads via 2G for circ water (1G), not Transfer Bus E which is the off site supply to 2H which is being tested. If the station desires to further minimize possible impact on Unit 1, its circ water bus 1G may be cross tied to 2G by closing 15G10 and opening 15G1.

Should the test result in a problem which impacted RSST B, a trip of Unit 1 and failure of the generator breaker occur, then the result could be loss of forced flow in one loop per

lS.2.5. Due to the low frequency of unit trips, less than thirty in the last 7.5 years, the lack of failures of the Unit 1 backfeed due to proper operation of the generator breaker, and the limited time when a problem on a paralleled Transfer Bus E could occur, probably less than 30 minutes, the probability of occurrence of an accident is not increased.

During testing, two off site power supplies will be maintained to the emergency busses of Unit 1 which is on line and no modifications will be performed which could impact the operability of an EDG. Therefore emergency power supplies and related equipment will be available to respond to an accident should one occur, this assures that the consequences of the reviewed accidents and of malfunctions will not increase while this testing is in progress. This testing will be performed only in appropriate outage windows.

This voltage profile testing will not create the possibility for an accident or malfunction of a different type than was previously evaluated in the SAR. Since two off site power sources are maintained to the emergency busses of Unit 1 unit which is on line during the test and one independent off site power supply is maintained to the operable Unit 2 emergency bus (2J) during the test with Unit 2 in mode 5, 6 or other outage condition, the present design basis of the plant is maintained. Separation of these sources provides assurance against a malfunction or accident either new or previously evaluated.

The testing will be performed separated from the on site power system and after completion the system will be returned to normal by synchronizing across and closing 15E1, restoring off-site power to 2H and shutting down the SBO AAC DG. These activities could, if the greatest errors imaginable occurred, disrupt power on Transfer Bus E and possibly RSST B which could cause a partial LOOP if Unit 1 tripped, but this is already evaluated.

For severe adverse impacts to occur the operator would need to close 15E1 with the SBO diesel significantly out of phase with the system. Due to operator training and multiple persons reviewing such an operation this is not realistic. There is no synch check relay governing closure of 15E1 just as there is no such relay for the EDGs, however, 15El was successfully synched across on April 29. A NOTE is included in the Test Procedure to remind the operator that there is no synch check on 15El.

The testing of this non-safety related system by powering the Emergency Bus 2H during an appropriate window in the Unit 2 outage will not result in a reduction of the margin of safety. It is acceptable to have only one emergency bus and one emergency diesel operable on a unit during certain periods in a refueling outage. This test will be scheduled when the 2J Emergency Bus and its EDG are operable and supplying power to loads required to maintain Unit 2 in compliance with its Technical Specifications. Therefore, there is no reduction in the margin of safety since performance of the required safety related busses is unaffected. There is also no need for a change to the Technical Specifications.

The interlock which prevents the connection of the AAC diesel generator to an RSST while the RSST is feeding a Station Service Bus must be defeated for RSST B feeding Station Service Bus 2B during this test. This interlock may be defeated providing that there are no large 4KV motors operating on the 2B Station Service Bus.

The purpose of this interlock is to prevent potentially elevated fault currents from being available. The fault currents, which could be available as a result of contributions from the system, the AAC diesel , and the large 4 KV motors on the station service bus, could potentially be above the rating of certain existing breakers. With no 4 KV motors operating on the 2B Bus, their contribution is eliminated and the resulting fault currents are acceptable. This condition is similar to that previously analyzed as part of the Electrical System Analysis performed for DCP 92-012-3.

Safety Evaluation Number
95-SE-PROC-16

Description Of Activity
2-OP-5.1, Filling and Venting the Reactor Coolant System
Revision 23 of the procedure allows installation of a temporary modification to bypass the cold leg loop stop valve interlocks

Safety Evaluation Summary
RCS Cold Leg Loop Stop Valve interlocks are designed to ensure that an accidental startup of an unborated and/or cold, isolated reactor coolant loop results only in a relatively slow reactivity insertion rate. The interlock performs a protective function using two independent limit switches to verify that the hot leg loop stop valve is open, two independent limit switches to verify that the cold leg loop stop valve is full closed, and two independent flow switches to verify that bypass flow around the cold leg loop stop valve is greater than 125 gpm for 90 minutes. (The flow verifies that the pump is running, the bypass line is not blocked, and the valves in the bypass line are open).

Previous experience with the described protective circuitry has shown that spurious flow spikes (due to sweeping of air out of the loop or random noise) can result in restarting the 90 minute timer resulting in failure to receive the cold leg stop valve open permissive after the required 90 minute run time. This Safety Evaluation considers a Temporary Modification that would allow bypassing the protective circuitry to allow opening of the cold leg loop stop valve. This jumper is considered acceptable because all of the required conditions will be met via administrative controls. Operating Procedure 2-OP-5.1 has steps to verify that the hot leg loop stop valve is full open, verify that cold leg loop stop bypass flow is greater than 125 gpm after starting the RCP and after the 90 minute required flow time is complete, and verifying that loop temperatures are within the required band.

Because the administrative controls meet the required Technical Specification requirements and do not alter the bases of diminishing the potential for a water slug injection accident, this Temporary Modification should be allowed. In addition, the jumper will not be installed unless the installed protective circuitry fails to perform as designed.

No Unreviewed Safety Question exists because the probability of occurrence and the consequences of the startup of an inactive reactor coolant loop accident are not affected. In addition, there is no postulated accidents or malfunctions that could be generated by the proposed activity.

Additionally, the UFSAR analyzed condition for startup of an inactive loop with the cold loop stop valve initially closed states: "Even with the assumption that administrative procedures are violated to the extent that an attempt is made to open the loop stop valves with 0 ppm in the inactive loop while the remaining portion of the system is at 1200 ppm, the dilution of the boron in the core is slow. ... For these conditions, the time for shutdown margin to be lost and the reactor to become critical is 16.4 min.". As can be seen, there is plenty of time for the operator to identify the high count rate and to take appropriate actions.

Safety Evaluation Number
95-SE-PROC-17

Description Of Activity
• Final Design Test Procedure D-NAT-94-008-2-1 for "High Head Safety Injection Flow Instrumentation Upgrades - NAPS Unit 2"
• 2-PT-61.4, Rev 16 P1 "RCS Pressure Isolation Valves - Leakage Test"
System leakage testing will be performed on the hot leg and cold leg Safety Injection piping inside containment. Also, Pressure Isolation Valve (PIV) leak testing may be required for check valves 2-SI-85 and 2-SI-107. The PIV testing is only required if a jumper is required to pressurize the SI piping.

Special Conditions
• An operator is required to be stationed in the Aux. Bldg. penetration area while the containment penetration LMC valves are open.
• Communications will be established between the MCR and the operator in the Auxiliary Building penetration area.
• Containment isolation valves are conservatively assumed inoperable while the associated LMC valves are open. Therefore, the TS 3.6.3.1 LCO for Containment Isolation Valves should be entered.
These requirements are included in D-NAT-94-008-2-1 and 2-PT-61.4

Safety Evaluation Summary
The activity being evaluated is the system leakage test required as a result of DCP 94-008-2, "High Head Safety Injection Flow Instrumentation Upgrades - NAPS Unit 2" and DCP 95-158, "2-SI-100 Check Valve Replacement". Since the portions of the SI lines that need to be tested can only be isolated from the RCS via check valves, the RCS must be pressurized before this test can be accomplished. (Note that freeze seals were determined not to be a reliable means of system isolation while shutdown).

For Unit 1 the required testing was simply accomplished by cracking open the BIT bypass MOV. However, if the corresponding Unit 2 valve is flowed, TS requires leak testing to verify its isolation capability from the RCS. Since this can not be performed without making the SI system inoperable, the proposed Temporary Modification will be used to provide an SI system pressure source.

The TM consists of jumpering from downstream of MOV-2289A (normal charging line) to downstream of the BIT bypass MOV, 2836. These locations were chosen for ease of installation (the jumper is confined to a small area of the Auxiliary Building penetration area) and because of the protection provided by the automatic closure of 2-CH-MOV-2289A and 2289B. This activity will be performed in Mode 3 with the RCS borated to CSD. A small amount of charging flow will be diverted to the SI lines for a brief time to pressurize the lines. In the event of an SI actuation, MOV-2289A and B will close to isolate the installed jumper. This prevents any unanalyzed BIT bypass flow. Therefore, the TS basis for the SI system will be maintained. Containment integrity is maintained by the inside containment check valves in both the charging and SI lines. An operator will be stationed at the penetrations to provide manual isolation if required. These penetrations do not require automatic closure for containment isolation. Therefore, the TS basis will be maintained. As a conservative measure, The TS 3.6.3.1 LCO for Containment Isolation Valves will be entered while the LMC valves are open. This will limit the time that the LMC valves can be open to four hours.

There is no unreviewed Safety Question because:

• The activity involves no accident or malfunction initiators. The SI piping (and jumper hose) is designed to withstand full RCS pressure. Therefore, the probability of occurrence for any accident requiring SI and / or containment isolation is not altered.

• If an accident were to occur, the SI system would continue to function as designed. The automatic closure of MOV-2289A and B on an SI signal will provide jumper hose isolation to prevent any unanalyzed BIT bypass flow. Containment isolation is maintained by the inside containment check valves in the SI and Charging lines. Since the fission product barriers are maintained, there is no increase in the consequences of any accident or malfunction.

• The jumper hose is temporary and will be in service for a short time. The hose will be adequate to withstand the RCS pressure. The activity will be performed using an approved procedure by qualified operators. Therefore, the probability of creating a different type of accident or malfunction is not increased.

Safety Evaluation Number
95-SE-PROC-18

Description Of Activity
Procedure 0-MCM-1904-01: On-line leak repair of 1-SW-679, SW to charging pump seal cooler return header isolation valve.
On-line leak repair of 1-SW-679, SW to charging pump seal cooler return header isolation valve.

Safety Evaluation Summary
On-line leak repair of 1-SW-679, SW to charging pump seal cooler return header isolation valve is being performed because 1-SW-679 has a body-to-bonnet leak and it is not desired to permanently repair the valve at this time.

This valve has never been injected. The injection process is controlled via procedure 0-MCM-1904-01. The function of the valve and service water system will not be changed by this process, therefore, no unreviewed safety question exists or is created by this activity.

Safety Evaluation Number
95-SE-PROC-19

Description Of Activity
Procedure 0-MCM-1410-03: EHC Portable EHC Filtration Skid. This procedure provides instructions for connecting, cleaning,and flushing the Unit 1 or Unit 2 EHC fluid systems using the portable filtration skid.
A portable EHC Filtration unit will be hooked into the existing EHC fluid reservoir bypassing the existing Fullers earth filtration subsystem. This skid system will essentially be an auxiliary filtration unit substituting for the existing filtration subsystem which is ineffective.

Safety Evaluation Summary
The normal Fuller's earth filter is not working on Unit 1. Either the suction needle valve is mechanically failed completely or partially blocking flow and/or the suction line from the valve to the filter is clogged with debris. Hence, the fluid water content has gone high out of spec and the fluid acidity has gone high out of spec. The acidity going high causes viton rings in the MOOG valves to deteriorate and could result in a failure leading to a turbine trip. Therefore, the EHC system must be connected to what is essentially an auxiliary filtration unit. Guidance for shutting down and removing the skid is provided by periodic Predictive Analysis oil sample results compared to acceptance criteria for neutralization number (acidity) and particulate size which are given in the procedure.

Concerning instrumentation and control: Contact A of 1-EH-LS-100 (and 2-EH-LS-200) closes on decreasing level below the low setpoint (17.25 inches) to alarm. Contact B closes on low-low level (7.62 inches) to energize the low fluid lockout relay providing an alarm inhibiting operation of both EH Fluid High Pressure Oil Pumps, 1-TM-P-3 and 4 (and 2-TM-P-3 and 4), and providing a turbine trip (SOV-20AST-2). Contact A of 1-EH-LS-101 closes on increasing level above the high setpoint (22 inches) to activate the alarm. Contact B and B1 close on decreasing level below the low-low setpoint (11.62 inches) to activate the alarm and allow 1-EH-LS-100 to energize the low fluid lockout relay. A decrease in EHC fluid pressure below the setpoint closes contacts of 1-EH-PS-103 (and 2-EH-PS-203) to initiate pumps start in auto provided the lockout relay, 86LFT, is reset.

Normal operating height is approximately 18 inches from the bottom. The skid suction goes 3 inches into the oil. Therefore, at 15 inches from the bottom, a complete failure of the skid would leave 15 inches of fluid in the reservoir. This level is below the 17.25 inch low level alarm of 1-EH-LS-100-1 thus alerting the control room operators. However, this level is above the 11.62 inch setpoint for the low-low level alarm of 1-EH-LS-101-2. Therefore, a loss of fluid would bring in the low level alarm but not the low-low level alarm and low-low-level lockout. A barrel supply of EHC fluid will be at the reservoir per the procedure and operations standby support will be requested in order to makeup to the reservoir and skid system for the initial hookup and in case of low level.

Turbulence of flow in the reservoir should not disturb the floats in the reservoir. Although the flow will be greater than normal at 5 gpm, the discharge will be extended approximately 10 inches into the fluid (or 8 inches from the bottom) and it will come into the reservoir in the vertical direction. Additionally, the location of the discharge will be 4 to 5 inches from the ball of the level floats. The system normally flows one gpm. However, the discharge of the normal flowpath from the Fuller's earth filter comes into the side of the reservoir moving in tier horizontal direction. Thus the existing controls should not be adversely impacted by the additional flow and a spurious low-low lockout and a turbine trip is avoided. The potential of reversing the skid, that is putting the discharge to the suction and

the suction to the discharge thus removing the above conservative design, is eliminated by making the suction and discharge into the reservoir each a different type of connection. The suction is a flanged connection, whereas the discharge is a fitted swagelock coupling (see 0-MPM-1410-03 Attachment 1).

A complete test run of the skid will be performed before hooking up to the EHC reservoir in order to hydro test the skid for leaks. The design pressure of the skid is 150 psig. The operating pressure of the filtration portion of the EHC system is approximately 20 psig, that is the portion of the system downstream of the orifice which throttle s the system down from 2200 psig to approximately 20 psig and a flow of 1 gpm.

Fryquel is caustic. Warning and precaution steps have been included to cite the MSDS. Specifically, rubber gloves and safety glasses must be worn during the EHC flush and EHC fluid has the potential to be a cancer causing agent. Fryquel is a nonflammable liquid. Therefore, no additional precautions against fire are needed.

The additional load on the service air system to drive the skid pump should not effect the ability of the service air system to maintain pressure. This will be confirmed by the test to be performed before hooking up the skid to the EHC system to ensure no adverse effect on Service Air.

Safety Evaluation Number
95-SE-PROC-19 Rev. 1

Description Of Activity
Procedure 0-MCM-1410-03: EHC Portable EHC Filtration Skid. This procedure provides instructions for connecting, cleaning,and flushing the Unit 1 or Unit 2 EHC fluid systems using the portable filtration skid.
A portable EHC Filtration unit will be hooked into the existing EHC fluid reservoir bypassing the existing Fullers earth filtration subsystem. This skid system will essentially be an auxiliary filtration unit substituting for the existing filtration subsystem which is ineffective.

Safety Evaluation Summary
The normal Fuller's earth filter is not working on Unit 1. Either the suction needle valve is mechanically failed completely or partially blocking flow and/or the suction line from the valve to the filter is clogged with debris. Hence, the fluid water content has gone high out of spec and the fluid acidity has gone high out of spec. The acidity going high causes viton rings in the MOOG valves to deteriorate and could result in a failure leading to a turbine trip. Therefore, the EHC system must be connected to what is essentially an auxiliary filtration unit. Guidance for shutting down and removing the skid is provided by periodic Predictive Analysis oil sample results compared to acceptance criteria for neutralization number (acidity) and particulate size which are given in the procedure.

Concerning instrumentation and control: Contact A of 1-EH-LS-100 (and 2-EH-LS-200) closes on decreasing level below the low setpoint (17.25 inches) to alarm. Contact B closes on low-low level (7.62 inches) to energize the low fluid lockout relay providing an alarm inhibiting operation of both EH Fluid High Pressure Oil Pumps, 1-TM-P-3 and 4 (and 2-TM-P-3 and 4), and providing a turbine trip (SOV-20AST-2). Contact A of 1-EH-LS-101 closes on increasing level above the high setpoint (22 inches) to activate the alarm. Contact B and B1 close on decreasing level below the low-low setpoint (11.62 inches) to activate the alarm and allow 1-EH-LS-100 to energize the low fluid lockout relay. A decrease in EHC fluid pressure below the setpoint closes contacts of 1-EH-PS-103 (and 2-EH-PS-203) to initiate pumps start in auto provided the lockout relay, 86LFT, is reset.

Normal operating height is approximately 18 inches from the bottom. The skid suction goes 3 inches into the oil. Therefore, at 15 inches from the bottom, a complete failure of the skid would leave 15 inches of fluid in the reservoir. This level is below the 17.25 inch low level alarm of 1-EH-LS-100-1 thus alerting the control room operators. However, this level is above the 11.62 inch setpoint for the low-low level alarm of 1-EH-LS-101-2. Therefore, a loss of fluid would bring in the low level alarm but not the low-low level alarm and low-low-level lockout. A barrel supply of EHC fluid will be at the reservoir per the procedure and operations standby support will be requested in order to makeup to the reservoir and skid system for the initial hookup and in case of low level.

Turbulence of flow in the reservoir should not disturb the floats in the reservoir. Although the flow will be greater than normal at 5 gpm, the discharge will be extended approximately 10 inches into the fluid (or 8 inches from the bottom) and it will come into the reservoir in the vertical direction. Additionally, the location of the discharge will be 4 to 5 inches from the ball of the level floats. The system normally flows one gpm. However, the discharge of the normal flowpath from the Fuller's earth filter comes into the side of the reservoir moving in tier horizontal direction. Thus the existing controls should not be adversely impacted by the additional flow and a spurious low-low lockout and a turbine trip is avoided. The potential of reversing the skid, that is putting the discharge to the suction and

the suction to the discharge thus removing the above conservative design, is eliminated by making the suction and discharge into the reservoir each a different type of connection. The suction is a flanged connection, whereas the discharge is a fitted swagelock coupling (see 0-MPM-1410-03 Attachment 1).

A complete test run of the skid will be performed before hooking up to the EHC reservoir in order to hydro test the skid for leaks. The design pressure of the skid is 150 psig. The operating pressure of the filtration portion of the EHC system is approximately 20 psig, that is the portion of the system downstream of the orifice which throttle s the system down from 2200 psig to approximately 20 psig and a flow of 1 gpm.

Fryquel is caustic. Warning and precaution steps have been included to cite the MSDS. Specifically, rubber gloves and safety glasses must be worn during the EHC flush and EHC fluid has the potential to be a cancer causing agent. Fryquel is a nonflammable liquid. Therefore, no additional precautions against fire are needed.

The additional load on the service air system to drive the skid pump should not effect the ability of the service air system to maintain pressure. This will be confirmed by the test to be performed before hooking up the skid to the EHC system to ensure no adverse effect on Service Air.

Safety Evaluation Number
95-SE-PROC-20

Description Of Activity
North Anna Vendor Procedure 0-FH-FEB-001, "Temporary New Fuel Elevator Basket Replacement." Westinghouse is the vendor.
This procedure will he used as a procedurally controlled Temporary Modification in accordance with VPAP-1403, "Temporary Modifications," to temporarily replace the new fuel elevator basket in the spent fuel pool with a similar Westinghouse basket. The Westinghouse basket is identical in function, but has additional design features to facilitate fuel inspection and reconstitution. The replacement basket is functionally equivalent to the original equipment basket, but has additional features which accommodate Westinghouse fuel inspection equipment. The replacement is temporary, and the basket will be in service for approximately six weeks.

Special Conditions
Irradiated fuel will be inserted in the temporary fuel elevator basket. The elevator basket with irradiated fuel will normally reside in the down position except for times when the irradiated fuel assembly in the elevator basket is raised to a sate elevation to allow removal or reinstallation of the fuel assembly top nozzle or rotate the guide block orientation. During periods when the fuel building is vacated (i e.. breaks or off-shift hours), the basket with irradiated fuel will be in the down position and the knife switch for the fuel elevator power will be locked open. This is controlled by vendor procedure I P-VRA-FI1, "Fuel Inspection and Repair for North Anna Unit #l "

Safety Evaluation Summary
This safety evaluation was prepared for vendor procedure 0-FI1-FEB-001, "Temporary New Fuel Elevator Basket Replacement." This procedure will be used as a procedurally controlled temporary modification in accordance with VPAP-1403, "Temporary Modifications." The purpose of the procedure is to replace the original equipment fuel elevator basket with a functionally equivalent Westinghouse basket. The Westinghouse basket has additional design features which facilitate handiing of individual fuel rods (removing and replacing fuel rods in the fuel assembly after the fuel assembly top nozzle is removed. Handling of individual fuel rods in this manner is required for individual fuel rod inspection and or fuel reconstitution. Fuel reconstitution is the process in which damaged or otherwise failed fuel rods are replaced with fuel bearing rods with similar reactivity characteristics or solid filler rods manufactured from stainless steel or zirconium-based alloy such as Zircaloy-4 or ZIRLO.

Fuel inspection and reconstitution programs have been performed in the past by gaining access to the fuel rods by removing the fuel assembly's bottom nozzle. This requires large mechanical equipment to invert the fuel assembly in order to remove the bottom nozzle. This is a proven technique but requires large cumbersome mechanical equipment to be installed in the spent fuel pool. The principle advantages of using the new fuel elevator are: other than the replacement basket. it requires no specialized large equipment, time for installation is generally on the order of one shift as opposed to four days to a week for inverting equipment. Our fuel is designed to allow the top nozzles to be removed and replaced. and Westinghouse now has a proven record of accomplishing fuel inspections and reconstitution using the new fuel elevator at (at least) eleven other sites in the past three years including Surry in July 1994.

The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report is not increased as a result of this change. The fuel elevator basket will be temporarily replaced with a generic Westinghouse basket to perform individual fuel rod examinations The temporary replacement basket is functionally equivalent to the original equipment basket. Either basket is capable of holding an irradiated fuel assembly. Other than replacing the basket, there will be no change to any other major component of the fuel elevator, specifically the cable, winch, motor and track. The assumptions used in the analysis for the fuel handling accident in the fuel building remain bounding for handling irradiated fuel in the elevator basket. All fuel handling will be performed in accordance with existing fuel handling procedures. Fuel inspection procedures will require the mechanical stop to be installed whenever irradiated fuel is inserted in the fuel elevator basket. This will assure that a minimum of 9 feet of water shielding exists over the active fuel in a fuel assembly [UFSAR 12.1.2.5, "Fuel Handling Shielding"]

The possibility for an accident or malfunction of a different type tnan any evaluated previously in the safety analysis report is not increased. The only equipment that will be used for this change is the fuel handling tool and fuel elevator with the replacement fuel elevator basket. The basket is a temporary replacement and is functionally equivalent to the original equipment basket. The limiting failure of any fuel handling equipment is the fuel handling accident outside of containment described in Section 15.4.5 of the UFSAR.

The margin of safety as defined in the basis for any technical specification is not reduced as a result of this change. This change temporarily replaces the fuel elevator basket with a Westinghouse generic fuel elevator basket. The replacement basket is functionally equivalent to the original equipment basket with the exception that the Westinghouse basket accommodates fuel inspections with irradiated fuel in the basket. No restrictions preclude the handling of irradiated fuel in the fuel elevator basket. Fuel inspection procedures will require the mechanical stop to be installed whenever irradiated fuel is handled in the fuel elevator basket to prevent lifting the fuel assembly to unsafe elevations. All fuel handling will be performed in accordance with existing site procedures.

Safety Evaluation Number
95-SE-PROC-21

Description Of Activity
North Anna Vendor Procedure FP-VRA-FI1, "Fuel Inspection and Repair for North Anna Unit #I"
This procedure delineates the tasks and evolutions required to remove and reattach removable top nozzles on a designated fuel assembly, remove and replace designated fuel rods from the fuel assembly, and perform required inspections on individual fuel rods. Fuel assemblies and individual fuel rods are designated on fuel handling report in accordance with VPAP-1406, "Nuclear Material Control."

Special Conditions
Irradiated fuel assemblies will be inserted into the Westinghouse temporary fuel elevator basket. The new fuel elevator is required to lift the fuel assembly to a safe elevation to remove and reattach the removable top nozzle on the fuel assembly. Special mechanical stops will be installed on the elevator tracks to prevent raising the basket to an unsafe elevation with irradiated fuel in it. The temporary basket will be installed and tracked separately by way of a procedurally controlled Temporary Modification in accordance with VPAP-1403, "Temporary Modifications." Fuel assembly and individual fuel rod handling will be performed and tracked in accordance with VPAP-1406, "Nuclear Material Control."

Safety Evaluation Summary
This safety evaluation was performed to evaluate North Anna Vendor Procedure FP-VRA-FI1 which provides the instructions to remove a fuel assembly's top nozzle to gain access to the fuel rods for either individual fuel rod inspection or individual fuel rod replacement. The procedure also provides instructions for performing individual fuel rod examinations required to support the Westinghouse Advanced Material Demonstration Program. Westinghouse is the vendor who developed the procedure.

The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report is not increased as a result of this change. The assumptions used in the analysis for the fuel handling accident in the fuel building remain bounding for handling and inspecting irradiated fuel. Most of the significant isotopes used in the offsite dose analysis for the fuel handling accident are now decayed away. Assembly AM2 was last irradiated in N1C10 which ended in September 1994. The two candidate reconstitution fuel assemblies were last irradiated in N1C9 which ended in January 1993. Two additional fuel assemblies that may be inspected operated in N2C10 and have over three months of cooling. All fuel handling will t)e performed in accordance with existing fuel handling procedures. Fuel inspection procedures will require the mechanical stop to be installed whenever irradiated fuel is inserted in the fuel elevator basket. This will assure that at a safe amount of water shielding exists over an irradiated fuel assembly. With the mechanical stops installed, an irradiated fuel assembly will always remain below the maximum elevation of the fuel when the long fuel handling tool is used for fuel movement. Similarly, the handling of individual fuel rods with the fuel rod handling tool is only possible with the fuel assembly fully lowered in the fuel elevator (fully lowered means the top of the fuel elevator basket is approximately the same elevation as the top of the fuel racks). The fuel rod handling procedure requires a sling at least 9 feet long to be rigged to the fuel handling crane hook with the other end attached to the fuel rod handling tool. This limits the upward travel of the fuel rod handling tool in case of an inadvertent lift. Because of these procedural and physical limitations, an individual fuel rod cannot be raised to an unsafe elevation when using the individual fuel rod handling tool.

The possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report is not increased. Fuel handling equipment will be used for this activity. The limiting failure of any fuel handling equipment is a fuel handling accident in the fuel building as described in Section 15.4.5 of the North Anna UFSAR. All fuel handling equipment that will be used is designed specifically for handling fuel assemblies or individual fuel rods. The individual fuel rod handling tool was designed to minimize the risk of inadvertently dropping a fuel rod.

The margin of safety as defined in the basis for any technical specification is not reduced as a result of this change. This activity provides instructions for removing and reattaching removable top nozzles on fuel assemblies, inspecting individual fuel rods, and replacing individual fuel rods with solid stainless steel filler rods. The fuel inspections will not alter the handling or storage characteristics of any fuel assembly. Fuel inspections procedures will require the mechanical stop to be installed whenever irradiated fuel is inserted in the fuel elevator basket to prevent lifting the fuel assembly to unsafe elevations. All fuel handling will be performed in accordance with existing site procedures.

Safety Evaluation Number
95-SE-PROC-22

Description Of Activity
2-MOP-26.113, P1
P1 of 2-MOP-26.113 provides more specific instructions on how to prepare the plant for the removal of MCC-2Cl-3 from service.

Special Conditions
The 2-lV battery charger actual loads must be verified 90 amps or less prior to the start of 2-GM-P-8. This ensures the battery charger current is limited to Z75 amps maximum. This is accomplished in section 5.5 of 2-MOP-26.113, P 1 .

Safety Evaluation Summary
Although the results of the screening checklist for P1 of 2-MOP-26.113 do not require a safety evaluation to be performed, station management felt that a safety analysis is warranted to ensure that the running of 2-GM-P-8 does not overload the 2-lV battery charger and cause the battery to discharge unnecessarily. Running of this pump is necessary to maintain proper air side seal oil pressure since the normal pump (2-GM-P-1) will be out of service when its normal power source (MCC-2C1-3) is removed from service for a breaker replacement.

Z-GM-P-8 is intended to be used on a loss-of-station-power (which causes the loss of the normal pump) to prevent potential generator damage due to leakage of hydrogen gas from the generator. 2-GM-P-8 is normally powered from the 2-lV DC bus. The pump motor full load current was evaluated and determined to be within the charger capability as documented in Reference 2.3.5 of the procedure PAR P1. Instructions are also specified in the procedure to preclude overloading of the charger when this pump is started. The 2-1V DC bus, including its battery charger, will not be adversely affected by the starting and running of this pump and thus will be able to carry out its intended safety functions. As such, an unreviewed safety question does not exist.

Safety Evaluation Number
95-SE-PROC-23

Description Of Activity
PAR P3 to North Anna Vendor Procedure FP-VRA-FI1, "Fuel Inspection and Repair for North Anna Unit 1."
This procedure delineates the tasks and evolutions required to safely remove fuel rod AM2E07 which is partially inserted in fuel assembly AM2 to the designated fuel rod storage canister. Additionally, the procedural step(s) to install the tamper-proof seal in the fuel rod storage canister may be removed.

Safety Evaluation Summary
Fuel assembly and individual fuel rod handling will be performed and tracked in accordance with VPAP-1406, "Nuclear Material Control." Normal fuel handling procedures will ensure that fuel building ventilation is lined up to filtered exhaust. Fuel Assembly AM2 will be restricted from further use in any reload core. This is tracked by the Restricted Fuel Assembly List in accordance with the Nuclear Design Control Program's Nuclear Analysis and Fuel Implementing Procedures.

Westinghouse was in the process of examining fuel rods from test assembly AM2 which contains fuel rods dad with various types of characterized cladding material. While attempting to reinstall fuel rod AM2E07 into fuel assembly AM2, the fuel rod became unlatched from the gripper with approximately 20 inches of the fuel rod extending out of the bottom of the fuel rod handling tool into fuel assembly AM2. Repeated attempts to relatch the rod onto the gripper failed. The fuel rod in this configuration is supported at the bottom, both vertically and horizontally by being partially inserted in fuel assembly AM2. The remaining section of the fuel rod (approximately 132 inches) is laterally supported in the fuel rod handling tool's (FRHT) collet tube. PAR P3 to North Anna Vendor Procedure FP-VRA-FI1 provides the necessary instructions to safely grip fuel rod AM2E07, remove the (FRHT) collet tube, remove the fuel rod from assembly AM2, and transfer the fuel rod to the designated fuel rod storage canister.

In addition to, and unrelated to the augmented fuel rod handling instructions provided in P3 to FP-VRA-FI1, the procedural step(s) to install the
' tamper-proof seal used in the fuel rod storage canisters is not required. There is no requirement or commitment to install a tamper-proof seal other than the procedural step requiring the installation. The rationale for the tamper-proof seal was to provide tamper evidence when unirradiated fuel rods were stored in a fuel rod storage container. The unirradiated fuel rods have been removed from the fuel rod storage canister, and the significant complexity involved with handling irradiated fuel rods stored in the fuel rod storage canister inherently makes them tamper-proof.

The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report is not increased as a result of this change. The assumptions used in the analysis for the fuel handling accident in the fuel building remain bounding for handling and inspecting irradiated fuel. Most of the significant isotopes used in the offsite dose analysis for the fuel handling accident are now decayed away. Assembly AM2 was last irradiated in N1C10 which ended in September 1994. Precautions will be taken to ensure that a safe amount of shielding water will exist over irradiated fuel at all times including full time health physics coverage.

The possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report is not increased. Fuel handling equipment will be used for this activity. The limiting failure of any fuel handling equipment is a fuel handling accident in the fuel building as described in Section 15.4.5 of the North Anna UFSAR. The hand-held collet is designed to handle individual fuel rods. Air-operated vice grips in addition to air-operated tooling for axial support will be used for additional rod support during the transfer operation. As backup, the air-operated vice grips can be used as the primary gripping device. All equipment will be used in accordance with its intended purpose as stated in the procedure.

The margin of safety as defined in the basis for any technical specification is not reduced as a result of this change. This activity provides instructions for transferring a single fuel rod from the fuel rod handling tool to permanent storage in the fuel rod storage container. Only one fuel rod is being transferred by this activity whereas the fuel handling accident in the spent fuel pool assumes the simultaneous destruction of 264 fuel rods. This activity will not alter the handling or storage characteristics of any fuel assembly. Fuel rods may be stored indefinitely in a fuel rod storage canister. Proper health physics control will be maintained at all times when the fuel rod is being moved.

Safety Evaluation Number
95-SE-PROC-24

Description Of Activity
Procedure 2-OP-7.10, Operation Of Casing Cooling Subsystem Of The Recirc Spray System
The above procedure will be changed to incorporate a new section which will provide the steps to add RWST water from temporary hose connections in the Unit 2 Safeguards to add borated water to the Unit 2 Recirc Spray Sump and to monitor the effect on the sump level as the addition is being made.

Safety Evaluation Summary
The purpose of the addition is to raise the Recirc Spray Sump level above the 6 inch requirement which provides a thermal blanket to prevent thermal stresses on the Low Head Safety Injection Pump suction valves from the containment sump, possibly causing these valves to bind and not open for cold leg recirc after the RWST level has dropped.

The makeup source to the sump will be from the RWST via the suction side of the B LHSI pump. The temporary hose will be run from the suction side vent connection and will not adversely affect the operation of the pump in the event of a SI or CDA. The temporary hose will be routed to an LMC on the discharge of the B Casing Cooling pump. An operator will be standing by to isolate the makeup in the event of an SI or CDA and this action complies with the guidelines of Generic Letter 91-18. Containment isolation and LHSI pump operability will be ensured with these actions.

Because the temporary hose will be connected to the suction of the LHSI pump and the discharge of the Casing Cooling pump downstream of the RS-MOV-200 isolation, the pressures that the hose could experience are much less than the 100 psig limit of the hose. Additionally, the hose is flexible enough to not be a seismic concern.

No Tech Spec basis are adversely affected. The affected casing cooling pump will be inoperable during this evolution but the duration will be within the allowable Tech Spec LCO. The possible containment leak path from the casing cooling line connection will be isolated by the dedicated operator. The suction source from the LHSI pump will not adversely affect the LHSI pump during a postulated SI or CDA since the vent line is small in comparison to the suction line and an operator is stationed in the area to isolate the line as necessary. Additionally, the temporary connection will be made from the "B" train of LHSI to the "B" train of Casing Cooling to eliminate the possibility of losing both trains of a single component in the event of a Loss of Offsite Power.

The makeup source is borated water from the RWST. During a CDA, the recirc spray sump boron concentration will not be adversely affected. Boron concentration of the recirc spray sump during normal operation is not monitored and is typically a function of the concentration of any leakage that collects in the sump. This makeup source is conservative.

Any water that overflows the recirc spray sump will run into the containment sump and then be automatically pumped out for normal liquid waste processing. This small amount of water will not affect the leakage monitoring capabilities of the containment sump. Monitoring of the recirc spray sump level increase is by the SPDS computer system and it does not affect the accident monitoring capabilities of that system in any way.

The makeup source is from the RWST which is a low activity source of radiation. The activity is being performed exclusively in the Safeguards area which has a sump processing

system designed to receive this type of water. The piping evolution consists of a normal type of activity that is routinely performed in an RCA.

Safety Evaluation Number
95-SE-PROC-25

Description Of Activity
2-MOP-7.01 R14-P1 Low Head Safety Injection Pump 2-SI-P-1A
2-MOP-7.02 Low Head Safety Injection Pump 2-SI-P-1 B (when revised)
The PAR provides an attachment to the base procedure to conduct vacuum-assisted depressurized venting of the "A" header. The pump and header are isolated from the RWST and a vacuum is drawn on the high point vents to assist venting. In the vacuum condition, the air bubbles will expand and migrate to the vent points. Repressurization of the headers is accomplished by a red rubber hose connected from a drain valve on the isolated header's recirculation and line to a drain valve on the pump discharge. The drain valve on the recirculation line remains shut except when actually repressurizing the isolated header.

Safety Evaluation Summary
The safety evaluation is performed for revisions to 2-MOP-7.01, Low Head Safety Injection Pump 2-SI-P-1A. The PAR provides an attachment to the base procedure to conduct vacuum assisted depressurized venting of the "A" header. The pump and header are isolated from the RWST and vented. In the vacuum depressurized condition, the air bubbles will expand and migrate to the vent points. Repressurization of the headers is accomplished by a red rubber hose connected between a drain valve on the isolated header's recirculation and test line and a drain valve on the pump discharge. The drain valve on the recirculation and test line remains shut except when actually repressurizing the isolated header.

The purpose of the PAR is to scavenge as much residual entrained air as possible from the headers to minimize pressure spikes in the header during pump starts.

A Temporary Modification is installed for the repressurization of the vented LHSI header. A temporary hose is connected between the recirculation and test line and the pump discharge header to facilitate repressurization of the isolated header following the venting process. The procedure requires a leak check of the hose and fittings upon installation. Steps exist to shut the affected valves and restore the pipe caps upon removal of the hose.

The structural properties of the LHSI pump "can" have been reviewed and the structural integrity of the can will not be affected by a vacuum in the system.

The activity cannot increase the probability of an accident or malfunction because this activity has no effect on the integrity of the RCS or Main Steam systems.

The activity cannot increase the consequences of an accident or malfunction because one train of LHSI will be maintained operable at all times.

The activity cannot create a new accident or malfunction because the activity does not reconfigure any system, structure, or component, such that the design bases for the plant are altered.

The activity should be allowed because it is prudent to remove as much entrained air as possible from the LHSI headers so that pressure spiking will be minimized on pump starts.

Safety Evaluation Number
95-SE-PROC-26

Description Of Activity
Nuclear Site Services Work Procedure GMP-M-162 Rev. 1 - Installation and Repair of Underground Fire Protection Piping and Appurtenances.
Revision 1 to the stated Work Procedure directs craft personnel to contact Loss Prevention, OPS Tagging Office, the Shift Supervisor and Engineering prior to starting excavation of fire line or appurtenances in order to provide compensatory missile protection measures and temporary seismic restraints.

Special Conditions
This Safety Evaluation is being performed to address the removal of cover from the Fire Main Yard Loop and subsequent installation of temporary seismic/missile restraints/protection, as specified by Engineering. These interim restraints/protection measures constitute a Temporary Modification as determined on an Activity Screening Check List. This Temporary Modification will be procedurally controlled by GMP-M-162.

Safety Evaluation Summary
Nuclear Site Services General Work Procedure GMP-M-162 provides for excavation, repair and replacement of the Underground Fire Main Loop Piping. This procedure is used to implement DCP's and stand alone work Orders. This procedure, when used with a stand alone Work Order does not permit any modifications, with the following exception. Throughout the excavation process of Yard Loop Piping, DEO-Civil is contacted and involved in ensuring the missile protection/seismic integrity of the system. Temporary seismic restraints and missile protection compensatory measures may be specified by DEO to ensure no adverse impact to the operational portion of the system. These temporary seismic restraints and missile protection compensatory measures were determined to constitute a Temporary Modification on the Activity Screening Check List, thereby resulting in this Safety Evaluation. This Temporary Modification is procedurally controlled by GMP-M-162.

The temporary seismic restraints and compensatory missile protection measures do not create an unreviewed Safety Question for the following reasons. They do not increase the probability of occurrence or consequences of an accident previously evaluated in the SAR. The possibility for an accident or malfunction of a different type than previously evaluated in the SAR will not be created. They do not decrease the margin of safety.

The temporary seismic restraints will maintain the seismic integrity of the operational portion of the Yard Loop by ensuring that any exposed operational portion of the system is seismically restrained and the exposed piping will not impart any adverse tributary loading to the remaining buried piping. Likewise the missile protection compensatory measures will ensure that the operable Loop Piping is not vulnerable to an increased risk of damage from severe weather.

This temporary change will be procedurally controlled by GMP-M-162.

Safety Evaluation Number
95-SE-PROC-27

Description Of Activity
Procedure 1/2-OP-7.10, Operation Of Casing Cooling Subsystem Of The Recirc Spray System
The above procedure will be changed to incorporate a new section which will provide the steps to add PG water from temporary hose connections in the Safeguards area to the Recirc Spray Sump and to monitor the effect on the sump level as the addition is being made.

Special Conditions
A dedicated operator will be available in the Safeguards area to isolate the makeup in the event of an SI or CDA. This compensatory action is being added by this procedure change to both the Unit 1 and Unit 2 procedures by the addition of step 4.10 (P&L), the Caution before step 5.16.9 and step 5.16.9.

Safety Evaluation Summary
The purpose of the addition is to raise the Recirc Spray Sump level above the required level which provides a thermal blanket to prevent thermal stresses on the Low Head Safety Injection Pump suction valves from the containment sump, possibly causing these valves to bind and not open for cold leg recirc after the RWST level has dropped during a LOCA.

The makeup source to the sump will be PG water. A temporary hose will be routed from a PG source to an LMC on the discharge of the B Casing Cooling pump. An operator will be standing by to isolate the makeup in the event of an SI or CDA and this action complies with the guidelines of Generic Letter 91-18. Containment isolation will be ensured with these actions.

The pressure of the PG source is greater than the 45 psig that could be experienced during a CDA if the check valve, located downstream of the LMC connection, leaks by.

No Tech Spec basis are adversely affected. The affected casing cooling pump will be inoperable during this evolution but the duration will be within the allowable Tech Spec LCO. The possible containment leak path from the casing cooling line connection will be isolated by the dedicated operator during a postulated SI or CDA.

The sump is normally full of PG water left over from the performance of the Inside Recirc Spray Pump flow test which uses PG water (see safety evaluation # 95-SE-OT-10). This volume of PG water was evaluated and found to not effect reactivity during a LOCA. The PG water will also tend to offset the affect of the NaOH from the QS chemical addition tank of sump pH. This was evaluated and found to be insignificant.

Specifically, the results of analysis performed for 95-SE-OT-10 to use PG water instead of refueling cavity water in the Inside Recirc Spray Pump sump and dike bounds the adding of PG to the sump in this safety evaluation. In the former case, approximately 1325 gallons remain in the lower part of the sump around the RS pumps following IRSP flow tests. This volume would slightly dilute the boron concentration of the sump during a LOCA. NA&F reviewed calc #SM-468 Rev1 and determined that this would have an insignificant effect on reactivity. The 1325 gal will also tend to decrease sump pH. NA&F reviewed calc SM-415 Rev1 and determined the effect on pH was also negligible. Calc SM-592 was also reviewed which concluded that up to 26,000 gallons of unborated water may be introduced into the sump without invalidating the current safety analysis.

Any water that overflows the recirc spray sump will run into the containment sump and then be automatically pumped out for normal liquid waste processing. This small amount of water will not effect the leakage monitoring capabilities of the containment sump. Monitoring of the recirc spray sump level increase is by the SPDS computer system and it does not affect the accident monitoring capabilities of that system in any way.

Opening of "sealed valves" on an intermittent basis is allowed by the Technical Specifications as long as proper administrative controls are established, which will be accomplished by the procedure. To be conservative, however, the LCO for this T.S. will be entered. The expected duration of this evolution should not exceed the LCO time allowance.

Safety Evaluation Number
95-SE-PROC-28

Description Of Activity
MDAP-19 for 02-EP-CB-28J - Replacement of relay 27XB-2J1
The Unit 2J 27XB degraded voltage relay will be replaced and a new relay will be tested.

Safety Evaluation Summary
During the performance of the Unit 2 Undervoltage PT, 2-PT-36.9.1.J, on 11/02/95, the 27XB relay of the "B" phase degraded voltage circuitry was found to be sticking. Jumpers were installed per 2-ECM-2802-02 placing the B phase of the degraded voltage circuitry in trip within one hour per Tech Spec 3.3.2.1.

The 27XB relay needs to be replaced. This requires additional "continuity" jumpers in order to maintain both the UV and DV circuitry operable while the relay is being replaced. These "continuity" jumpers will be removed after the relay is replaced and before the "trip" jumpers are removed, the one-hour action of Tech Spec 3.3.2.1 is reentered, and the new relay is tested by a partial performance of 2-PT-36.9.1.J.

The "continuity" jumpers are placed in such a way that a UV or DV condition will be sensed and the ESF equipment will initiate as required. Thus the UV/DV ESF will be fully maintained operable throughout the evolution.

Since these "continuity" jumpers will not affect the UV/DV circuitry or EDG operability, an unreviewed safety question does not exist.

Safety Evaluation Number
95-SE-PROC-29

Description Of Activity
Procedure 0-TOP-50.6: Drawing A Sludge Sample from the BC Cooling Tower Basin Via a Passive Cyclonic Separator and a Submersible Pump. This procedure provides instructions for connecting, operating and disconnecting a rig to be used for sampling the BC basin sludge.
A portable pump and passive cyclonic separator unit will be aligned to the BC basin.

Safety Evaluation Summary
The procedure aligns the basin of the BC towers to a portable 120 VAC 1/3 HP Grinder pump having a discharge head of 45 psig and a capacity of 65 gpm. The discharge of the pump is aligned to a passive cyclonic separator. The separator has two separate discharge paths, one as a return path to the basin, and the other as a valved sample point directed to a 5 gallon sample container. All of the equipment required for this evolution is being supplied by the vendor. When the evolution is completed, the pump is rinsed off with domestic water in a Virginia Power supplied 55 gallon drum and returned to the vendor.

For personnel safety, the individual handling the submersible pump will be equipped with a floatation device or a life line & harness. Adequate protective clothing will be worn while handling hazardous materials.

Various chemicals are added to the BC system during normal operations. These include microbiocide, biocide, zinc chloride and other corrosion inhibitors. In undiluted conditions, these may be poisonous or hazardous to personnel or the environment. To ensure no leakage of BC water to the environment, temporary absorbing barriers will be placed around the rig to control any leakage. In the event any leakage develops, the pump will be stopped, and the Shift Supervisor and Environmental Compliance Coordinator will be notified. Since the procedure directs collection of a sample in a 5 gallon temporary container, the Environmental Compliance Coordinator will be notified.

Operation of the BC sytem is not expected to be altered by the activity. The BC basin level will be essentially unchanged; thus, there is no adverse affect on BC pump NPSH available. A disturbance of the sediment in the BC basin will occur as a result of operating the submersible pump which has the potential to cause blockage of the suction side of some BC cooled heat exchangers. These heat exchangers will be periodically observed to verify no adverse trends. Indication of any adverse trends will require stopping the portable pump and evaluation of the indication.

Safety Evaluation Number
95-SE-PROC-30

Description Of Activity
Procedure 0-MOP-50.1: Desludging the Bearing Cooling Tower Basin to a Sludge Separator
A portable pump (s) will be used to pump sludge from the Bearing Cooling Tower basin to a vendor's sludge separation trailer. The sludge removed will be disposed of as radioactive waste and the water returned to the BC tower basin.

Safety Evaluation Summary
The procedure is to pump the accumulated sludge from the lower basin of the Bearing Cooling Tower with the tower remaining in operation. One or more air operated diaphragm pumps will be used, with a suitable suction device, to remove the sludge from the bottom of the tower basin and discharge it into a vendor's sludge separation trailer, which uses a centrifuge to separate the sludge from the water. The sludge will be disposed of as radioactive waste and the water returned to the cooling tower basin. Temporary barriers will be erected under/around the pumping rig components to catch any spills which may occur.

For personnel safety, the individual handling the pump suction will be equipped with a flotation device or a life line & harness. Adequate protective clothing will be worn while pumping.

Various chemicals are added to the BC system during normal operations. These include microbiocide, biocide, zinc chloride and other corrosion inhibitors. In undiluted conditions, these may be poisonous or hazardous to personnel or the environment. Sludge removal personnel will not be required to handle these materials.

Operation of the BC system is not expected to be altered by the activity. The BC basin level will be essentially unchanged because the BC Makeup Pumps will provide the additional necessary makeup water to compensate for the volume removed; thus, there is no adverse affect on BC pump NPSH available. A disturbance of the sediment in the BC basin will occur as a result of operating the pumping rig which has the potential to cause blockage of the suction side of some BC cooled heat exchangers. These heat exchangers will be periodically observed to verify no adverse trends. Indication of any adverse trends will require stopping the portable pump and evaluation of the indication.

The BC water will also be sampled periodically to monitor water chemistry and to ensure that concentrations remain within acceptable limits. If any adverse trends are observed, then pumping will be stopped until an evaluation is made.

Safety Evaluation Number
95-SE-PROC-31

Description Of Activity
Procedure I /2-PT-2 10.19 SI Accumulator Discharge Check Valves Full Open Test
This test will verify that the SI accumulator discharge check valves will exhibit full open stroke during a controlled accumulator dump into the an open RCS. The test will be performed with the upper internals installed, the reactor head off, and the reactor vessel full up to at least 4' above the reactor flange. The fuel may be installed. The reactor refueling cavity will receive the accumulator inventory. No LCOs will be entered. The special requirements to ensure a safe and successful test consist of accumulator boron concentration not lower than 65 ppm below the concentration required by 1/2-PT-1().2 and that the RCS boron is at least 30 ppm above this requirement, reactor fuel is not being moved. reactor vessel upper internals have been placed in the reactor vessel, and the reactor refueling cavity is at least 4' above the vessel flange. The RCS boron concentration may be waived if the accumulator boron concentration is at or above the minimum concentration required by 1/2-PT-10.2. These requirements are incorporated into 1/2-PT-210.19.

Safety Evaluation Summary
An unreviewed safety question for this activity does not exist. The equipment and systems involved in this test will all be operated well within their design limits. This precludes the failure or cracking of any primary piping. The Containment equipment hatch or temporary equipment hatch plate will be installed and the ventilation established in accordance with Health Physics. The potential nitrogen introduced to the containment will not pose a health hazard as it represents less than 1% of the free volume of the containment.

Reactivity addition is precluded by requiring the boron concentration of each accumulator to be not lower than 65 ppm below the concentration required by 1/2-PT- 10.2 and that the RCS is at least 30 ppm above this requirement. This 65 ppm value was conservatively calculated to ensure no more than a 30 ppm swing in RCS boron concentration with the addition of the entire volume of 3 accumulators, and assuming the entire RCS volume is only that volume from mid-loop to 4' above the vessel flange. This is conservative in that the RCS volume below mid-loop was discounted and the cavity level may be well above 4' over the flange. The RCS boron concentration requirements may be waived if the accumulator boron concentration is at least the minimum concentration specified in 1/2-PT-10.2. To ensure that the boron concentration remains uniform throughout this testing, at least one train of RHR is required to be operating. Preclusion of a fuel handling accident inside containment is achieved by requiring as an initial condition that no fuel movement be in progress.

Safety Evaluation Number
95-SE-PROC-32

Description Of Activity
MDAP-0019 for Work Order 00331754-01
2-CC-TV-204C (CC supply to the 'C' RCP) will be closed for air tubing replacement. Contingency plans allow for installation of an air jumper to re-open the valve if RCP motor temperatures approach required RCP trip Setpoints.

Special Conditions
• Seal Injection is assumed to be in service to 2-RC-P-1C during the maintenance evolution.
• Containment Temperature will be monitored during the maintenance evolution.
• The P250 computer or the RCP temperature chart recorder will be monitored during the maintenance evolution.
• Tech Spec requirements for containment isolation valves govern contingency air jumper installation.
ALL REQUIREMENTS ARE CONTAINED IN THE MDAP-0019 SUPPLEMENTAL INSTRUCTIONS

Safety Evaluation Summary
An active air leak exists on the instrument air tubing for 2-CC-TV-204C (CC supply to the 'C' RCP) and future catastrophic failure of the air line is possible if corrective maintenance is not performed. Valve closure will be required for the air tubing replacement. Contingency plans allow for installation of an air jumper to re-open the valve if RCP motor temperatures approach required RCP trip Setpoints.

2-CC-TV-204C is the CC supply to the 'C' RCP Upper and Lower Lube Oil Coolers, Stator Exhaust Coolers, CRDM Shroud Coolers, and the Thermal Barrier Heat Exchanger (TBHX).

• Loss of cooling to the TBHX is of no consequence as long as adequate seal injection flow is maintained. Therefore, it will be verified that seal injection flow is in service to the RCP prior to maintenance. If a loss of seal injection flow occurs while TBHX CC flow is secured, AP-33.2 gives guidance for dealing with a complete loss of RCP seal cooling.
• Loss of cooling to the RCP motor stator coolers and CRDM shroud coolers will have no adverse affect on RCP or CRDM operation. However, the containment temperature may rise slightly and therefore, containment temperature will be monitored during the proposed evolution.
• Loss of cooling to the RCP lube oil coolers will be the limiting variable for the proposed activity. UFSAR Section 9.2.2.5.4 describes that loss of CC will result in a 5 degree F rise in bearing temperature per minute. Therefore, only about 10 minutes are available before the RCP trip Setpoints described in OP-5.2 are reached. Industry and station experience has shown that the actual time to reach a trip limit is actually closer to 20 minutes. The maximum allowed RCP motor bearing temperature is 195F. Because of the concern with motor bearing temperatures, Operations will monitor RCP motor bearing temperatures on the P250 computer or the alternate chart recorder during the maintenance activity.

An unreviewed safety question does not exist with the proposed activity because the maintenance will not place the plant outside of any design parameters. The worst case scenarios is that the loss of CC cooling results in meeting RCP manual trip criteria in which

case the unit will be tripped and the RCP secured. The consequences of any accidents or malfunctions are not altered by installation of the contingency air jumper because the inside containment isolation check valve and the closed CC system inside containment will provide adequate containment isolation. A subsequent single failure is not required to be assumed while in the allowable Tech Spec Action time (four hours) for having an inoperable containment isolation valve. No new accident or malfunction scenarios were identified. In addition, the Operating License and Tech Spec are not altered or violated by the proposed activity.

The planned maintenance activity should be allowed because failure to repair the instrument air tubing will lead to further depredation than could result in catastrophic failure of the air line. An unplanned loss of RCP motor cooling will inevitably lead to a required manual reactor trip and RCP shutdown.

Safety Evaluation Number
95-SE-ST-01

Description Of Activity
1-ST-104: Safeguards Exhaust Flow Verification of 1-HV-F-40B with the Inspection Port of 1-HV-F-40A open.
The Special Test will align the Unit 1 Safeguards Area Ventilation System (SAVS) related components in a manner that will allow flow measurements to be taken with one of the exhaust fans running and the inspection port open on the discharge of the idle exhaust fan. Delta P measurements between the Unit 1 Safeguards Area and the atmosphere will be taken during the testing as well.

Special Conditions
During a CDA on Unit 1 or 2, the Test Coordinator must perform the following (as contained in 1-ST-104):
1) Notify test personnel located at the Unit 1 Safeguards Exhaust Fans in the Aux. Bldg. to immediately reinstall the idle fan discharge inspection port.
2) Close all test measurement points.
3) Ensure at least one Unit 1 Safeguards Exhaust Fan is in service.

Safety Evaluation Summary
The document evaluated by this Safety Evaluation is Special Test, 1-ST-104: Safeguards Exhaust Flow Verification of 1-HV-F-40B with the Inspection Port of 1-HV-F-40A open.

The Special Test will align the Unit 1 Safeguards Area Related ventilation components in a manner that will allow flow measurements to be taken with an exhaust fan running and the inspection port open on the discharge of the idle fan. Delta P measurements between the Unit 1 Safeguards Area and the atmosphere will be taken during the testing as well. The purpose for these measurements is to collect data for further evaluation of this ventilation alignment, and to determine if the Unit 1 Safeguards Area is at a negative pressure with an exhaust fan running and the inspection port open on the discharge of the idle fan.

The test will be controlled by a Test Coordinator. A briefing will be held to ensure that all test personnel are familiar with the purpose of the test, the equipment alignment, individual responsibilities, and the actions required to be taken to terminate the test in the event of a Unit 1 or 2 CDA.

The 7 day Action Statement of T.S. 3.7.8.1 will be entered for the idle fan while the discharge inspection port is not fully in place. Operability of the running fan will be maintained during the test by limiting the opening of the inspection port to a position that will not cause flow to decrease below the T.S. minimum value. However, the ability to return the full complement of the Safeguards Area Ventilation System to a known T.S. operable condition will be maintained during the performance of test. The Technical Specification Basis for the Safeguards Area Ventilation System (SAVS) is that the system will function to ensure that radioactive materials leaking from the ECCS equipment within the Safeguards Area following a LOCA are filtered prior to reaching the environment. It is noted in the T.S. Bases section for this LCO, that operation of the system and the resultant effect on offsite dosage calculations was assumed in the accident analyses. Action will be taken, in the event of a CDA on Unit 1 or 2, to place the system in a known condition to ensure the proper operation of the system to mitigate the effects of any leakage into the Safeguards Area from the ECCS equipment housed in the area. It should be noted that the test in no way affects the ability of the dampers to align the system to discharge through the Charcoal Filters. That is, regardless of the test configuration, the dampers will still position to the filtration position automatically during a Unit 1 CDA. No automatic actions

are defeated by the test. The manual actions referred to above will ensure the system is properly aligned in the event of a CDA on either Unit. These actions will include the following:

During a CDA on Unit 1 or 2, the Test Coordinator must perform the following (as contained in 1-ST-104):
1) Notify test personnel located at the Unit 1 Safeguards Exhaust Fans in the Aux. Bldg. to immediately reinstall the idle fan discharge inspection port.
2) Close all test measurement points.
3) Ensure at least one Unit 1 Safeguards Exhaust Fan is in service.

These actions are considered to be reasonable and non-complicated. The exhaust fan discharge inspection port is of such dimension and weight that it can easily be installed by one person. The inspection port is equipped with four "dogs" to quickly secure the panel in place; no screws or bolts are required for installation. All flow test points in the duct work are secured with threaded caps, which can also be installed quickly and easily. Fan alignment can be performed quickly from the Control Room on the Unit 1 Backboards. It should be noted that verification of a SAVS exhaust fan running following actuation of the affected Unit's CDA system is performed independently in Attachment 2 (Verification of Phase "B" Isolation) of 1-E-0.

Performance of these actions upon the notification that a CDA has actuated on either Unit will ensure that the Safeguards Area Ventilation system will perform its design function following a Large Break LOCA (LBLOCA). It is judged that the actions can be performed in a time frame that will ensure that any radioactive leakage from components in the Safeguards Area will be properly evacuated from the area and filtered prior to release to the environment. UFSAR described leakages from the piping and components in the Safeguards area are related to potential Recirculation Loop Leakage External to the Containment and RS Leakage Outside Containment. The leakages discussed are from small assumed system leaks and from passive failure piping rupture/leaks. Radioactive release from the described leakage sources would not be expected within the first few minutes of the LBLOCA. The Outside Recirc. Spray pumps start after a time delay of 210 seconds from the CDA signal initiation. This means that the associated RS piping in the Safeguards area will not be flowing until at least 3 minutes from the initiation of the accident. The LHSI pumps start immediately upon the receipt of the SI signal, but the contents of the pump and piping for the injection phase (approximately the first 50 to 60 minutes) is from the RWST, an activity controlled water source. Passive failures are not assumed to occur (according to the UFSAR criteria) until the Long Term period of the accident; defined as the period following the first 24 hours of the accident. As such it can be assumed that the manual actions will be carried out prior to the release of any radioactivity into the Safeguards Area air space, thus ensuring that any credible radioactive release to the Safeguards Area will be filtered prior to release to the environment.

The Special Test does not introduce an Unreviewed Safety Question for the following reasons:

The special test does not create any conditions that could reduce the integrity of the Reactor Coolant System, ECCS, or CDA Systems. The test aligns the SAVS for the purpose of data collection. The test is limited to the SAVS only. No Chapter 15 Accidents could be initiated by the proposed testing.

As discussed above, accident mitigation functions provided by the system and assummed in the accident analyses will still be provided in the event the functions are required.

Action will be taken, in the event of a CDA on Unit 1 or 2, to place the system in a known condition to ensure the proper operation of the system to mitigate the effects of any leakage into the Safeguards Area from the ECCS equipment housed in the area. These procedural constraints and actions ensure that the margin of safety described in the T.S. Bases will be maintained.

Safety Evaluation Number
95-SE-ST-02

Description Of Activity
1-ST-104 (Rev 1): Safeguards Exhaust Flow Verification of 1-HV-F-40B with the Inspection Port of 1-HV-F-40A open.
The Special Test will align the Unit 1 Safeguards Area Ventilation System (SAVS) related components in a manner that will allow flow measurements to be taken with one of the exhaust fans running and the inspection port fully open on the discharge of the idle exhaust fan. Delta P measurements between the Unit 1 Safeguards Area and the atmosphere will be taken during the testing as well.

Special Conditions
During a CDA on Unit 1 or 2, the Test Coordinator must perform the following (as contained in 1-ST-104):
1) Notify test personnel located at the Unit 1 Safeguards Exhaust Fans in the Aux. Bldg. to immediately reinstall the idle fan discharge inspection port.
2) Close all test measurement points.
3) Ensure at least one Unit 1 Safeguards Exhaust Fan is in service.

Safety Evaluation Summary
The document evaluated by this Safety Evaluation is Special Test, 1-ST-104: Safeguards Exhaust Flow Verification of 1-HV-F-40B with the Inspection Port of 1-HV-F-40A open.

The Special Test will align the Unit 1 Safeguards Area Related ventilation components in a manner that will allow flow measurements to be taken with an exhaust fan running and the inspection port open on the discharge of the idle fan. Delta P measurements between the Unit 1 Safeguards Area and the atmosphere will be taken during the testing as well. The purpose for these measurements is to collect data for further evaluation of this ventilation alignment, and to determine if the Unit 1 Safeguards Area is at a negative pressure with an exhaust fan running and the inspection port open on the discharge of the idle fan.

The test will be controlled by a Test Coordinator. A briefing will be held to ensure that all test personnel are familiar with the purpose of the test, the equipment alignment, individual responsibilities, and the actions required to be taken to terminate the test in the event of a Unit 1 or 2 CDA.

The 7 day Action Statement of T.S. 3.7.8.1 will be entered for the idle fan while the discharge inspection port is not fully in place. The ability to return the full complement of the Safeguards Area Ventilation System to a known T.S. operable condition will be maintained during the performance of test. The Technical Specification Basis for the Safeguards Area Ventilation System (SAVS) is that the system will function to ensure that radioactive materials leaking from the ECCS equipment within the Safeguards Area following a LOCA are filtered prior to reaching the environment. It is noted in the T.S. Bases section for this LCO, that operation of the system and the resultant effect on offsite dosage calculations was assumed in the accident analyses. Action will be taken, in the event of a CDA on Unit 1 or 2, to place the system in a known condition to ensure the proper operation of the system to mitigate the effects of any leakage into the Safeguards Area from the ECCS equipment housed in the area. It should be noted that the test in no way affects the ability of the dampers to align the system to discharge through the Charcoal Filters. That is, regardless of the test configuration, the dampers will still position to the filtration position automatically during a Unit 1 CDA. No automatic actions are defeated by the test. The manual actions referred to above will ensure the system is properly aligned in the event of a CDA on either Unit. These actions will include the following:

During a CDA on Unit 1 or 2, the Test Coordinator must perform the following (as contained in 1-ST-104, Rev 1):
1) Notify test personnel located at the Unit 1 Safeguards Exhaust Fans in the Aux. Bldg. to immediately reinstall the idle fan discharge inspection port.
2) Close all test measurement points.
3) Ensure at least one Unit 1 Safeguards Exhaust Fan is in service.
4) Ensure 1-HV-HV-4 is in SLOW speed.

These actions are considered to be reasonable and non-complicated. The exhaust fan discharge inspection port is of such dimension and weight that it can easily be installed by one person. The inspection port is equipped with four "dogs" to quickly secure the panel in place; no screws or bolts are required for installation. All flow test points in the duct work are secured with threaded caps, which can also be installed quickly and easily. Fan alignment can be performed quickly from the Control Room on the Unit 1 Backboards. It should be noted that verification of a SAVS exhaust fan running following actuation of the affected Unit's CDA system is performed independently in Attachment 2 (Verification of Phase "B" Isolation) of 1-E-0.

Performance of these actions upon the notification that a CDA has actuated on either Unit will ensure that the Safeguards Area Ventilation system will perform its design function following a Large Break LOCA (LBLOCA). It is judged that the actions can be performed in a time frame that will ensure that any radioactive leakage from components in the Safeguards Area will be properly evacuated from the area and filtered prior to release to the environment. UFSAR described leakages from the piping and components in the Safeguards area are related to potential Recirculation Loop Leakage External to the Containment and RS Leakage Outside Containment. The leakages discussed are from small assumed system leaks and from passive failure piping rupture/leaks. Radioactive release from the described leakage sources would not be expected within the first few minutes of the LBLOCA. The Outside Recirc. Spray pumps start after a time delay of 210 seconds from the CDA signal initiation. This means that the associated RS piping in the Safeguards area will not be flowing until at least 3 minutes from the initiation of the accident. The LHSI pumps start immediately upon the receipt of the SI signal, but the contents of the pump and piping for the injection phase (approximately the first 50 to 60 minutes) is from the RWST, an activity controlled water source. Passive failures are not assumed to occur (according to the UFSAR criteria) until the Long Term period of the accident; defined as the period following the first 24 hours of the accident. As such it can be assumed that the manual actions will be carried out prior to the release of any radioactivity into the Safeguards Area air space, thus ensuring that any credible radioactive release to the Safeguards Area will be filtered prior to release to the environment.

The Special Test does not introduce an Unreviewed Safety Question for the following reasons:

The special test does not create any conditions that could reduce the integrity of the Reactor Coolant System, ECCS, or CDA Systems. The test aligns the SAVS for the purpose of data collection. The test is limited to the SAVS only. No Chapter 15 Accidents could be initiated by the proposed testing.

As discussed above, accident mitigation functions provided by the system and assummed in the accident analyses will still be provided in the event the functions are required.

Action will be taken, in the event of a CDA on Unit 1 or 2, to place the system in a known condition to ensure the proper operation of the system to mitigate the effects of any leakage

into the Safeguards Area from the ECCS equipment housed in the area. These procedural constraints and actions ensure that the margin of safety described in the T.S. Bases will be maintained.

Safety Evaluation Number
95-SE-ST-03

Description Of Activity
2-ST-98, 2-CH-P-1B Discharge Valve Flow and Differential Pressure Verification for 2-CH-MOV-2286B and 2287B.
The test will involve installation of test equipment and the collection of flow and differential pressure data while flowing 2-CH-P-1B through each of its discharge MOVs, one at a time, to the RCS via the Normal Charging Header. [2-CH-MOV-2286B will flow directly to the normal header; 2-CH-MOV-2287B will flow to the normal header indirectly through the alternate header MOV(s) on the adjacent pump(s) to the normal header MOV(s)].

Special Conditions
This Safety Evaluation assumes that there will be administrative control over the test gauges and connections installed by this procedure for the duration of their installation. Administrative control is considered to consist of an individual located in the vicinity of the test equipment, in direct communication with the Main Control Room, who will manually isolate the test equipment from the system piping upon notification of a Safety Injection actuation or a seismic event. These administrative control actions and descriptions are contained in the Special Test (2-ST-98).

Safety Evaluation Summary
The document evaluated by this Safety Evaluation is Special Test 2-ST-98, "2-CH-P-1B Discharge Valve Flow and Differential Pressure Verification for 2-CH-MOV-2286B and 2287B". The purpose of the test is to provide instructions for the collection of flow and pressure data to be used in the evaluation of the flow characteristics of 2-CH-MOV-2286B and 2287B. These MOVs were inadvertently over-thrusted during a previous post-PM valve stroke.

The test will involve installation of test equipment and the collection of flow and differential pressure data while flowing 2-CH-P-1B through each of its discharge MOVs, one at a time, to the RCS via the Normal Charging Header. [2-CH-MOV-2286B will flow directly to the normal header; 2-CH-MOV-2287B will flow to the normal header indirectly through the alternate header MOV(s) on the adjacent pump(s) to the normal header MOV(s)]. Normal Charging and normal Seal Injection will remain in service during the test.

2-CH-P-1B will not be considered operable during the testing, however, the pump will be in service during the test providing Normal Charging and Seal Injection supply. As such, 2-CH-P-1B will be available for normal and emergency operation, though no credit will be taken for emergency operation of the pump. 2-CH-P-1A will be the operable Charging/HHSI and T.S. Boration Flow Path pump during the test. 2-CH-P-1C will not be operable during the test since the 2J Bus supply breaker for the pump will be racked out to support operation of 2-CH-P-1B (breaker interlock) and the 2H Bus supply breaker will be racked out to support 2-CH-P-1A operability (breaker UV interlock).

Since only 2-CH-P-1A will be operable during the test, the 72 hour Action Statement of T.S. 3.5.2 and 3.1.2.4 will be entered for no operable "B" Train Charging/HHSI pumps (ECCS) and less than two Charging pumps operable for Boration Flow Path support. It is anticipated that the data collection portion of the test will be completed prior to the expiration of the 72 hour Action Statement time frame. Regardless, compliance with the applicable T.S. LCOs and Action Statements will be maintained during the test. As such,

the margin of safety described in the bases of the T.S. will not be reduced by the proposed test.

During all portions of the test at least one Charging/HHSI pump will be operable and capable of flowing to the core as part of a Boration Flow Path or an ECCS flow path. This is acceptable since the condition will be bounded by the 72 hour limitation of the applicable T.S. Action Statements with regard to single failure application. Therefore the capability of the CVCS and ECCS to perform their designed safety function when required will be maintained.

This Safety Evaluation assumes that there will be administrative control over the test gauges and connections installed by this procedure, for the duration of their installation. Administrative control, in this case, is considered to consist of an individual located in the vicinity of the test equipment, in direct communication with the Main Control Room, who will manually isolate the test equipment from the system piping upon notification of a Safety Injection actuation on the Unit or a seismic event. It should be noted that the test equipment and associated test connections to the system piping will not be considered seismic. The purpose for their isolation from the piping system during an SI actuation on the Unit or a seismic event is to eliminate the test equipment connection site as a potential leak site for the CVCS or ECCS. This further ensures proper operation of the systems for the fulfillment of their safety function when required. These administrative control actions, as described above, are contained in the Special Test (2-ST-98). It should be noted that the actions are consistent with accepted manual action requirements described in NRC Generic Letter 91-18, regarding the use of manual action for maintaining system operability.

It should be noted that the during the previous over-thrust of the MOVs, it is possible that internal valve components may have been damaged. It is not known if any debris was generated as a result of the over-thrust condition. VOTES testing scheduled to be performed prior to the performance of the Special Test is expected to reveal any anomalies with the valve internals. Should there be debris present internal to the valve, and should that debris become dislodged while flowing the MOV, it is not expected that it would adversely obstruct the common flow path of the CVCS or ECCS. In addition, it is not expected that any potential debris would travel to the reactor core. It is expected that any potential debris would be caught in downstream components such as the Charging Flow Control Valve or the Regenerative Heat Exchanger. Debris introduction into the RCP Seals is not expected since Seal Injection Water is filtered prior to entering the seals. As such, no component safety function failures are anticipated.

The Special Test does not set or create conditions that are a precursor for an UFSAR Accident or Malfunction. The test simply places the B Charging/HHSI pump in service and collects flow and pressure data while flowing the pump's discharge MOVs one at a time. Debris release from the MOVs is considered unlikely, however, should it occur, it is not anticipated that it would adversely affect any downstream components in a manner that would create an accident precursor or affect the safety function performance of any component.

As previously described, at least one Charging/HHSI pump will be operable at all times during the test. The nature of the test is such that any data collection will not adversely affect the operable pump or flow path. Therefore, the test will not adversely affect the CVCS or ECCS from fulfilling its safety function if required. Administrative control of the test equipment will ensure that the piping system is not adversely affected by the non-seismic test equipment in the event of a Safety Injection actuation on the Unit or a seismic event. This ensures that the consequences of any accident or malfunction are not increased.

In addition, the capability of the CVCS and ECCS to mitigate the consequences of an accident will be maintained.

Since 2-CH-P-1A will be the only operable pump during the test, a 72 hour Action Statement will be entered as previously described. The Unit will maintain T.S. compliance with the applicable Action Statement during the test. This will ensure that the T.S. margin of safety is maintained commensurate with Action Statement assumptions and limitations.

For these reasons, the Special Test does not create an Unreviewed Safety Question, and the test performance should be allowed.

Safety Evaluation Number
95-SE-ST-04

Description Of Activity
Special Test 2-ST-97
The CHEMTRAC system(ABB/CE) will be utilized to quantify the amount of moisture carryover occurring in Unit 2 using Special Test 2-ST-97. The system entails injecting an enriched Li 6 isotope(non-radioactive) into the feedwater train via an injection pump cart through normal system drain connections and then drawing downstream samples to determine the resultant concentrations. The moisture carryover samples are drawn from the OLCMS sample points.

Special Conditions
1. Unit power must be stable and at 100%. This will be controlled under the Special Test procedure by having the Operator perform 2-PT-24.1, Computer Calorimetric Heat Balance after the tracer is injected and just prior to taking samples.

Safety Evaluation Summary
The test involves injecting a measured amount of Li 6 into each Main Feedwater line through drain valves located in the Mechanical Equipment Room of Unit 2. After injection samples will be drawn at the On Line Chemistry Monitoring Panel to determine if the concentration is sufficient to enable a valid test. Sampling will then be performed for off-site analysis. Li 6 is non volatile so any of the material in the Main Steam samples has to be carried by actual moisture droplets(or carryover). Thus an actual measurement of total carryover can be obtained.

1. The following issues were considered:

A. Main Steam Line rupture and Main Feed line break. This test only collects data and has no effect on plant protective circuitry. If a steam or a feed line break were to occur the test creates no restrictions or obstacles to plant protective features. This test will create no adverse chemical condition for the Unit 2 SG s or associated piping(CME N-93-D82) nor will there be a significant personnel hazard. In addition to the CME requirements precautions such as whip restraints will be used on the temporary connections to the plant components to prevent injury.

The test only collects samples from normal sample connections and imposes no operational restrictions on plant operation. Should plant conditions require power reduction the test will be suspended until plant conditions are again stable at full power.

Data will be recorded by the P-250 and GETARS which will not present the possibility of feedback to any protective circuitry. The special test has a provision to reset GETARS after the test to ensure the normal trip function continues to be available.

Therefore based on the above discussion there is no unreviewed Safety Question since there is no increased likelihood of an unanalyzed event nor is there any impact on the plant margin of safety.

Safety Evaluation Number
95-SE-TM-01

Description Of Temporary Modification
Temporary Modification # 95-1611

The vapor extractor for the turbine lube oil reservoir will be secured. A temporary extractor will be set up.

Safety Evaluation Summary
A temporary vapor extractor for the turbine lube oil reservoir will be installed while maintenance is performed on the vapor extractor demister. The drain line for the demister has become clogged, causing oil mist to be released to the atmosphere. This TM will be installed to perform maintenance on the demister and restore the system to normal operation.

Safety Evaluation Number
95-SE-TM-02

Description Of Temporary Modification
Temporary Modification 95-N2-1082
This temporary modification connects a regulated Service Air (SA) system jumper to the Condenser Air Ejector exhaust flowmeter manifold test connections at 2-VP-331 (2-CN-EJ-1A) and 2-VP-337 (2-CN-EJ-1B) and allows the installation of heaters (eg: heat lamps heat tape, etc) adjacent to AEJ exhaust piping upstream of the RM.

Safety Evaluation Summary
The Condenser Air Ejector Exhaust Radiation Monitor (2-SV-RM-221) monitors the non-condensable gases from the main condenser for the presence of radioactivity which is indicative of Steam Generator Tube Leakage. Of late, this RM has exhibited erratic spiking which does not correlate with the other independent indicators of Primary to Secondary Leakage (AEJ Grab Samples, N16 Monitors, Blowdown RMs, and RCS and SG Sampling). With detector replacement and channel calibration unsuccessful in eliminating the erroneous behavior, it has been postulated that the cause may be attributable to the following:

1. Insufficient flow to the RM (due to extremely low air inleakage to the Main Condenser)
2. Excessive water entrainment and condensation in relatively cool piping and RM assembly

The installation of a regulated SA supply and heaters (heat tape, heat lamps, etc) is expected to alleviate these problems, resulting in more accurate indication.

2-SV-RM-221 is used for the following:

1. Primary to Secondary Leakage Monitoring per TS 3.4.6.3 and 3.4.6.4.
2. Gaseous Effluent Monitoring per VPAP-2103.
3. Hi Radiation automatic swapover of AEJ exhaust from Vent Stack A to Containment via closure of 2-SV-TV-202-2 and opening of 2-SV-TV-202-1 and 203.
4. Phase A Isolation and Hi Radiation closes AEJ steam supply valve 2-AS-FCV-200A and 200B. Note that Phase A closes 2-SV-TV-202-2, 202-1, and 203 per TS 3.6.3.1 and TRM, TR 5.1, Table 5.1-Z.
5. Diagnosis and mitigation of Steam Generator Tube Rupture as discussed in UFSAR Section 15.4.3.

The count rate from the Condenser Air Ejector Radiation Monitor (2-RM-SV-221) is used to determine the primary-to-secondary leak rate by multiplying the count rate of the detector above background by a constant. This constant is calculated using a correlation developed based on the RCS activities of certain isotopes, the total flow past 2-RM-SV-221, and core power level. This correlation is already part of an existing station procedure, 2-PT-46.3A.2. while introducing a small amount of additional air flow past 2-RM-SV-221 will tend to dilute the concentration of the radioactive noncondensible gases past the detector and reduce the detector count rate, this situation is no different than when an air leak develops in the condenser which causes a similar dilution of the radioactive noncondensible gases as they flow past 2-RM-SV-221. The dilution flow to be added will increase the total flow past 2-RM-SV-221 to a value typically observed during full power operation (3 to 6 scfm). This is a range over which the correlation has historically been shown to yield accurate primary-to-secondary leak rate results. By collecting data after the modification has been performed and using the total air ejector flow past 2-RM-SV-221, a new correlation

constant will be determined and incorporated using 2-PT-46.3A.2. Therefore, this modification will not adversely impact primary-to-secondary leakage monitoring using 2-RM-SV-221.

The total air ejector flow is also used in 2-PT-46.3A.2 to determine "X-Values" which are provided to the Health Physics Department for performing 2-PT-46.3B. This procedure calculates primary-to-secondary leak rates using a grab sample of the air ejector exhaust. This grab sample is collected downstream of where the additional air will be injected; therefore, the "X-Values" calculated by 2-PT-46.3A.2 using the increased air ejector flows will be consistent with the grab samples obtained and there will be no adverse impact to this method of calculating primary-to-secondary leak rates.

Another function of 2-RM-SV-221 is to cause the Condenser Air Ejector exhaust to be diverted to containment when the high-high activity alarm is received. This high-high alarm for 2-RM-SV-221 is calculated based on the high alarm setpoint. As the concentration of radioactive noncondensible gases past the detector is reduce by the introduction of Service Air, this setpoint will need to be reduced accordingly. Part of the data collection of 2-PT-46.3A.2 mentioned above will check the validity of the high alarm setpoint and initiate a setpoint change using existing station procedure 2-PT-46.3A.3 as required. By resetting the high alarm (and the high-high alarm), 2-RM-SV-221 will be able to perform its design function in accidents such as a steam generator tube rupture (SGTR).

This modification will not detract from the ability of the operator to control and monitor the plant. Moreover, this modification will allow the AEJ RM to remain in service and accurately present the radiation level of the AEJ effluent. This ensures that its passive and active functions will be preserved. The function and operation of the air ejector RM is unchanged. The intentional introduction of filtered, dry air is similar in effect to in-leakage experienced in lesser integrity condensers. Like this normally occurring leakage, this intentional flowrate will be measured by their installed flow meters. Interestingly enough, UFSAR Section 10.4.6.1 states that the normal anticipated exhaust is 12.5 scfm per air ejector. The current total of 2.0 scfm is significantly less than this expected value. while this modification is not discussed in the UFSAR, it is not prohibited. It does not increase the likelihood or consequences of any accident. Thus, this modification does not result in a reduction of the margin of safety, nor an unreviewed safety question.

Safety Evaluation Number
95-SE-TM-03

Description Of Temporary Modification
TEMPORARY MODIFICATION #N1-95-1612
This TM jumpers out the 4 minute timer in the CW pump start/stop circuitry. This will allow starting or stopping the CW pumps as deemed necessary (operator adherence to the 4 minute delay will continue but the circuitry will be jumpered out).

Safety Evaluation Summary
This TM jumpers out the 4 minute timer in the CW pump start/stop circuitry. This will allow starting or stopping the CW pumps as deemed necessary (operator adherence to the 4 minute delay will continue but the circuitry will be jumpered out).

The 4 minute time for the "D" CW pump has failed, rendering the other pumps unable to be manually started or stopped. This TM will allow the starting or stopping of pumps as necessary under an emergency situation (e.g. a fish impingement event) until the timer can be replaced.

The 4 minute timer is installed to prevent multiple CW pump start or stops from creating a pressure wave in the CW intake tunnel which could potentially damage the tunnel. When this time delay is defeated, no significant increase in the potential for CW tunnel damage occurs due to operator knowledge of the time delay and their adherence to the delay.

This TM will not affect the ability of the CW pumps to trip during a turbine building flooding scenario, nor will it affect the ability of the pumps to trip on a electrical fault. Therefore, no unreviewed safety question exists or is created by this activity.

Safety Evaluation Number
95-SE-TM-04

Description Of Temporary Modification
Temporary Modification 95-1083
This temporary modification installs a controlled supply of Nitrogen (GN) system jumper to the Condenser Hotwell at an unused conductivity cell connection 2-CN-167 or 2-CN-168. The Nitrogen is supplied from the Feedwater Nitrogen blanket supply (2-GN-170) or high pressure Nitrogen bottle.

Safety Evaluation Summary
A discussion of the effect of the increased flow through the Air Ejector Radiation Monitor 2-SV-RM-221 can be found in Temporary Modification 95-N2-1082 and 10CFR 50.59 Safety Evaluation 95-SE-TM-02.

This Temporary Modification will test whether a 3-10 SCFM continuous flow of Nitrogen into the Main Condenser will improve the air removal efficiency and cause a reduction in dissolved Oxygen in the Condensate. Other plants have used this method to reduce their dissolved Oxygen levels. One theory for this is a sufficient flow of non-condensable is not available to allow the air removal section of the Air Ejector to function as designed. Thus the Oxygen does not get swept out of the Condenser. Therefore, if some Nitrogen were continuously introduced into the Condenser, the resulting increase in non-condensable flowrates may entrain more of the Oxygen, and sweep it out along with the Nitrogen. total Nitrogen flowrate shall not exceed 10 SCFM.

Since a continuous stream of non-condensable gas Nitrogen will be introduced into the Condenser, Logs and calculations of Condenser in-Leakage need to be adjusted by subtracting the Nitrogen in-flow, in order to properly obtain a true value for air in-leakage.

Injecting Nitrogen below the normal Condenser Hotwell water level will not affect operation of the Condenser at the rate of addition allowed. Since the maximum flow rate of the Air Ejector is approximately 30 SCFM, the nitrogen will not build up in the condenser sufficiently to interfere with heat transfer. Nitrogen gas was chosen since it is inert and will not support corrosion or other adverse affects to the system or introduce a fire hazard. It does however, present an asphyxiation hazard and can create a Life threatening atmosphere in a confined space. The station Confined Space Entry permitting program is sufficient to insure this does not pose an additional personnel hazard. Operations will need to include the Temporary Modification on the abnormal Equipment Status Log to verify the supply is isolated and verified before opening the condenser for maintenance.

As explained in the 95-SE-TM-02 this modification will allow the AEJ RM to remain in service and accurately present the radiation level of the AEJ effluent. This will ensure that its passive and active functions will be preserved. The function and operation of the Air ejector RM is unchanged. The intentional introduction of Nitrogen is similar in effect to in-leakage experienced in lesser integrity condensers. This additional flow will still be monitored by the installed flow meters. UFSAR section 10.4.6.1 states that the normal anticipated exhaust is 12.5 SCFM per air ejector. The current total of 2.0 SCFM is significantly less than this value. Therefore, this modification does not result in an unreviewed safety question, does not increase the likelihood of an event or decrease the margin of safety.

Safety Evaluation Number
95-SE-TM-05

Description Of Temporary Modification
Temporary Modification N1-1613
A Merline-Gerin (M/G) Spectral Analysis Radiation Monitor is being temporarily installed on Vent Stack 'B'. The Radiation Monitor will be observed for a period of 2-3 months to determine whether if would be an effective replacement for the current Vent Stack radiation monitoring configuration.

Safety Evaluation Summary
The M/G in-line duct radiation monitor will be installed via a temporary modification (TM) in accordance with VPAP 1403. The TM installation can be broken down to two phases: (1) Mechanical Installation and (2) Electrical Installation. During the mechanical phase, the detector assembly will be installed in the Vent stack 'B' duct-work. This will be accomplished by reducing Vent Stack 'B' flow to a minimum , removing a 27 inch x 27 inch plate on the underside of Vent Stack 'B' approximately 15 feet prior to the vertical turn (adjacent to the Turbine Building siding), drilling the plate to accept the bolt pattern on the radiation monitor's support stanchion and detector cable connector, bolting the detector stanchion to the plate, and re-installing (with the monitor mounted) the plate. The slot that will allow access to the detector cabling will then be sealed with putty. Minimizing Vent Stack 'B' flow during the intrusive installation bill minimize the potential for personnel injury or contamination since pressure and flow will be minimized. HP coverage during the mechanical installation will be necessary since the vent stack bill be open to atmosphere. This will complete the mechanical installation phase. The proposed installation site is located in the NSR, non-seismic portion of the stack. The detector has been classified NSR for purposes of this testing.

A representative from the vendor will arrive to oversee the electrical installation phase. The vendor will arrive at a pre-planned date and will bring the detector cable, the local processing unit (LPU), a R232 cable to interface the LPU with a NAPS-supplied computer, and the software that will allow the ability to perform on-line monitoring of the detector. The electrical installation will be performed by running approximately 80 feet of detector cable from the detector cable port (drilled and plugged during the mechanical phase), through the Turbine Building siding (approximately 6 inches off the Service Building roof), dropping the cable down into the Turbine building along the wall that separates the Turbine Building from the Unit 1 switchgear room (near the Powdex units control panel). The cable will then be routed under the inactive and latched side of Normal Switchgear Door S07-5 (Fire Door) to the LPU. A table will be set-up adjacent to the door and near a receptacle which will house the LPU, computer, and monitor for the 2-3 month trial application. The vendor will then load the software and start-up the system. Power for the LPU and computer will come from an installed 120V NSR receptacle and will not significantly increase any temporary power requirements since the LPU, computer, and monitor load are minimal. A source, provided by health Physics, will be used to check the sensitivity of the detector, to calibrate the detector, and to validate the software programmed geometry.

Installation and trial application of this in-line duct monitor will allow in-situ testing of the radiation monitor. Although the N-16 monitors use similar technology, in-situ testing will validate the specific application. Since a computer is being installed to monitor the LPU response, data relating to the monitor performance will be saved on disk.

No unreviewed safety question is created by installation of the TM. Additionally the probability of occurrence o the consequences of any malfunctions addressed by this safety

evaluation are not increased as a result of this TM. The basis for this is the TM does not alter the plant configuration. Specific details are listed below.

1-VG-RM-112, 113 (General Atomics), 1-VG-RM-175 (NRC), and 1-VG-RM-180 (KAMAN) will all be available after the mechanical installation phase to monitor effluent releases from Vent Stack 'B'. During the mechanical installation and removal phase, HP coverage will ensure no unmonitored or diluted release path is available. Since the Safeguard Over-pressure Relief Dampers were installed, all piping downstream is not safety-related or seismically qualified. Therefore, installation of the TM will not alter the qualification of the ~B~ Vent Stack or increase the probability of a crimped stack. The relief damper will ensure the minimum Tech Spec Safeguard flow is available under all conditions. During installation and removal, Tech Spec Table 3.3-6, Action 21 and 35 are applicable since 1-VG-RM 112, -113 and 1-VG-RM-180-1, -2 are inoperable (see Question 39 for details). Action 21 refers to the Action Statements of T.S. 3.9.12 which, if applicable (i.e., if moving fuel over the spent fuel pit or moving a load over the spent fuel pit), would require suspension (Action B) of these activities until 1-VG-RM-112 -113 were operable. Additionally, VPAP 2103, Attachment 15, Instruments 4 (A) will be inoperable during the installation/removal of this TM. Thus, the VPAP 2103 Attachment 15, Action 2 (effluent release may continue if a grab sample is taken every 12 hours and analyzed for gross or gamma activity within 24 hours) is applicable during the installation and removal of the TM.

Since the detector cable will be run under Normal Switchgear Room door S07-5 (an Appendix R fire suppression door), the possibility of the detector cable increasing the probability or consequences of an Appendix R fire in the Normal Switchgear room were evaluated. It was concluded that the detector cable will not increase the probability of a fire since the cable rating is fire resistant (i.e., not prone or subject to ignition temperatures necessary for flame growth along cable length). Additionally, a fire in the turbine building would not be expected to increase the probability of a fire in the Unit 1 Normal Switchgear Room since the inactive and latched S07-5 fire door will not be altered. Therefore, any flame growth from a Turbine Building fire would be quenched since the effective area under the door is not great enough to allow transmission from the Turbine Building to the Normal Switchgear Room. One other factor that was considered was the use of the wall receptacle to power the LPU, computer, and monitor. It was concluded and verified by Engineering that the power requirements were so small that they neither increase the probability or consequences of a fire in the Normal Switchgear room. Additionally, they are not a limiting fault o the 120V NSR, service supply breaker.

Safety Evaluation Number
95-SE-TM-06

Description Of Temporary Modification
Jumper N1-1614
This temporary modification will install an electrical jumper from the bypass transformer to distribution panel 1IHS-MO in the Security System.

Safety Evaluation Summary

MAJOR ISSUES:

The Security system has failed and a replacement component will be installed under DCP 95-117. This jumper will electrically bypass the failed system. It can then be functionally tested to determine the root cause of the failure. while this jumper is being installed the system will be out of service. The security diesel will be functionally tested during this period. Security Contingency Plan Implementing Procedure SCPIP-05 will be implemented prior to and during the installation of this jumper.

JUSTIFICATION:

This temporary modification will allow the system to be functionally tested. while this jumper is being installed the security diesel will be functionally tested in accordance with station procedures.

UNREVIEWED SAFETY QUESTION ASSESSMENT:

1) Accident probability has not been increased because this design change conforms to all applicable standards and administrative procedures. The operation of the system will not be permanently affected by this temporary modification.

2) Accident consequences are not increased. The implementation of this temporary modification will be performed and controlled using station approved procedures. The operation of the Security system will not be permanently affected by this temporary modification. Loss of the Security system will not affect the safe shutdown capability of the plant.

3) The implementation of this temporary modification will not create a possibility for an accident or a malfunction of a different type than previously analyzed in the SAR. Loss of the Security system will not affect the safe shutdown capability of the plant.

4) The Margin of Safety will not be compromised. The integrity of the Security system will be maintained during the installation of this temporary modification due to implementation of Security Contingency Plan Implementing Procedure SCPIP-05. The Security system will be out of service during jumper installation.

Safety Evaluation Number
95-SE-TM-07
Description Of Temporary Modification
Temporary Modification 95-1084
This temporary modification installs a controlled supply of Nitrogen (GN) system jumper to the Condenser Hotwell at an unused conductivity cell connection "A" and/or "B" condensers. The Nitrogen is supplied from the Feedwater Nitrogen blanket supply (2-GN-170) or high pressure Nitrogen bottle.

Safety Evaluation Summary
A discussion of the effect of the increased flow through the Air Ejector Radiation Monitor 2-SV-RM-221 can be found in temporary Modification 95-N2-1082 and 10CFR 50.59 Safety Evaluation 9S-SE-TM-02.

This Temporary Modification will test whether a 3-10 SCFM continuous flow of Nitrogen into the Main Condenser will improve the Air removA1 efficiency and cause a reduction in dissolved Oxygen in the Condensate. other plants have used this method to reduce their dissolved Oxygen levels. One theory for this is a sufficient flow of non-condensables is not available to allow the air removal section of the Air Ejector to function as designed. Thus the Oxygen does not get swept out of the Condenser. Therefore, if some Nitrogen were continuously introduced into the Condenser, the resulting increase in non-condensable flowrates may entrain more of the Oxygen, and sweep it out along with the Nitrogen. Total Nitrogen flowrate shall not exceed 10 SCFM.

Since a continuous stream of non-condensable gas Nitrogen will be introduced into the Condenser, Logs and calculations of Condenser in-leakage need to be adjusted by subtracting the Nitrogen in-flow, in order to properly obtain a true value for air in-leakage.

Injecting Nitrogen below the normal Condenser Hotwell water level will not affect operation of the Condenser at the rate of addition Allowed. Since the maximum flow rate of the Air Ejector is approximately 30 SCFM, the nitrogen will not build up in the condenser sufficiently to interfere with heat transfer. Nitrogen gas was chosen since it is inert and will not support corrosion or other adverse affects to the system or introduce A fire hazard. It does however, present an asphyxiation hazard and can create a life threatening atmosphere in a confined space. The station Confined Space Entry permitting program is sufficient to insure this does not pose an additional personnel hazard. Operations will need to include the Temporary Modification on the Abnormal Equipment Status Log to verify the supply is isolated and verified before opening the condenser for maintenance.

As explained in the 9S-SE-TM-02 this modification will allow the AEJ RM to remain in service and accurately present the radiation level of the AEJ effluent. This will ensure that its passive and active functions will be preserved. The function and operation of the Air ejector RM is unchanged. The intentional introduction of Nitrogen is similar in effect to in-leakage experienced in lesser integrity condensers. This Additional flow will still be monitored by the installed flow meters. UFSAR section 10.4.6.1 states that the normal anticipated exhaust is 12.5 SCFM per air ejector. The current total of 2.0 SCFM is significantly less than this value. Therefore. this modification does not result in an unreviewed safety question, does not increase the likelihood of an event or decrease the margin of safety.

Safety Evaluation Number
95-SE-TM-08

Description Of Temporary Modification
Temporary Modification TM-N2-1085.
Install a temporary hose between an SI accumulator vent and a drain off of the RHR relief valve discharge line.

Safety Evaluation Summary
2-SI-PCV-200 is the large pressure drop pressure control valve in the nitrogen supply line to the Pressurizer Relief Tank (PRT). This valve is currently not working and parts are not available. It is desired to provide a controlled source of nitrogen to the PRT to provide a slight overpressure to the RCS as part of the normal degassing process prior to RCS drain down. The proposed Temporary Modification will use a hose rated for at least 100 psig to supply nitrogen from a SI accumulator vent to the RHR relief valves discharge line to the PRT. This will allow the control room operator to control RCS overpressure by opening the pressurizer PORVs and controlling the makeup flow of nitrogen to the SI accumulator with its supply HCV.

Personnel safety will be maintained by maintaining the nitrogen supply pressure from accumulator at approximately 50 psi, and the hose will be physically restrained at the connections . In addition, a check valve will be provided on the jumper discharge side. This will prevent the hose from whipping and limit the amount of radioactive gas that could be released from the PRT if the jumper hose were to be cut.

Equipment safety will be provided by the fact that RCS NDT protection will be in service and with the pressurizer PORVs blocked open, the PRT rupture discs will relieve on high pressure. The nitrogen overpressure to the RCS will be limited to less than or equal to 50 psig and is typically in the 10 to 15 psig range. This overpressure will provide a back pressure to the RHR relief valves. However, as discussed above, other RCS/RHR overpressure protection is in place.

An unreviewed safety question is not created because:
(1) The probability of an accident or malfunction previously evaluated in the SAR occurring is not increased. The TM does not introduce any accident initiators. The unit is shutdown and will be in Mode 5 while this TM is active.
(2) The consequences of any accident or malfunction previously evaluated in the SAR are not increased. No fission product barriers are compromised by this TM. The unit is shutdown and will be in Mode 5 while this TM is active.
(3) The possibility of creating a new accident or malfunction has not increased. The TM will be installed by qualified personnel and using appropriate safety guidelines. The control room operator will have control of the nitrogen supply via the SI accumulator makeup HCV. A jumper hose rupture does not breach the RCS boundary because of the installed check valve.

Because the TM is not an undue risk to personnel safety or reactor safety, and because the TM will help ensure that the outage will not be unduly delayed, (remember that a shorter outage is a safer outage as long as it is properly planned), this Temporary Modification should be allowed.

Safety Evaluation Number
95-SE-TM-09

Description Of Temporary Modification
Temporary Modification N2-1086
Disable two of four parking brakes on the polar crane bridge.

Safety Evaluation Summary
The polar crane is designed in accordance with ANSI B30.2.0-1967. The ANSI Safety Code requires that the crane motion be capable of being stopped when the crane is under load. The polar crane is equipped with a hydraulic-electric brake that acts as a parking brake when the crane is de-energized. The affected portion of the brake is two of four that are used to lock the bridge portion of the crane in the rotational axis.

The parking brake is provided with a foot pedal control for stopping the crane during operation, and this is not affected. This T.M. only affects the braking action on a loss of power or when the crane is de-energized. The remaining two parking brakes will continue to operate as designed. The design of the parking break is to lock the bridge when unattended, this is not affected. It is routinely used on a linear motion overhead crane that could be mounted on an incline.

The hydraulic portion of the braking system is not affected by this activity. The parking brake is only used when the crane is de-energized and unattended. This will not occur with a load on the crane.

Operation of the crane with a load is not affected in any way by this activity. There is no requirement in the ANSI Safety Code for this parking brake to be on the crane. The parking brake action will still be provided by the remaining two parking brakes. This activity does not affect the function or operation of the crane in any way.

Safety Evaluation Number
95-SE-TM-010

Description Of Temporary Modification
Temporary Modification 95 - 1615
Calgon biocide H-900 is to be added to the bearing cooling system by using one or more plastic containers, with perforations for liquid flow through, placed in the return basin at the top of the bearing cooling tower. The container allows BC liquid to come in contact with the H-900 tablets to slowly dissolve them while preventing their direct contact with the wood structure of the tower

Safety Evaluation Summary
This analysis considers the potential adverse safety effects of adding H-900 biocide tablets to the Bearing Cooling liquid as it passes through the upper distribution plenum of the Bearing Cooling tower. This TM is needed because the normal bromination path is unavailable for adding the biocide.

An unreviewed safety question does not exist for the following reasons:

* H-900 is the biocide normally used in the Bearing Cooling System.

* Diluted concentrations of the biocide pose no threat to the integrity or proper function of the Bearing Cooling tower.

* The plastic containers allow Bearing Cooling water flow direct contact with the tablets but prevents tablet/wood structure contact. It is only the direct contact between tablets and wood that is not advisable by respective vendors (Marley for BC tower wood and Calgon for the Biocide chemicals).

* The presence of the plastic container poses no threat to the operability of the tower, its fans or the Bearing Cooling pumps. The container is far larger than the flow holes through which the BC water cascades down in the wood structure.

Safety Evaluation Number
95-SE-TM-11

Description Of Temporary Modification
To support 2-PT-87H DC Distribution System Service Test (Train A), Temporary Mod (TM) N95-1087
Temporary Modification (TM) to the supply and exhaust ductwork in the 2-I Battery Room in support of 2-PT-82H.

Safety Evaluation Summary
The TM is to be installed to support the performance of 2-PT-87H DC Distribution System Service Test during fuel movement or work over the spent fuel pool to maintain the Design Basis of the UFSAR for Control Room Habitability and Tech. Spec. compliance for Units 1 and 2.

An Unreviewed Safety Question does not exist because:
•The accidents considered were any event which releases radioactivity outside of containment for Units 1 and 2.

•The TM does not affect any accident precursors. Control Room Pressure Boundary is maintained during fuel movement or work over the Spent Fuel Pool.

•Accident mitigation equipment is not affected. Tech. Spec. compliance support control room habitability. Fire dampers in the 2-I Battery Room ductwork remains operable. The TM does not affect the normal performance of the battery PT.

•The potential consequences of control room pressure boundary issues are already addressed by habitability analysis.

•Tech. Spec. compliance and UFSAR Design Basis is being maintained for Units 1 and 2 therefore the margin of safety is not reduced.

Safety Evaluation Number
95-SE-TM-012

Description Of Temporary Modification
Temporary Modification N2-1090
The field cable for 2-RM-RMS-263 (Containment Area Radiation monitor) to return 2-RM-RMS-262 (Manipulator Crane Radiation monitor) to service.

Safety Evaluation Summary
2-RM-RMS-262, the manipulator crane radiation monitor, is required to be operable to enable containment Purge and Exhaust to be placed in service. The field cable for 2-RM-RMS-262 has been damaged in the containment and there is not enough cable available to repair the damage. The damaged section of cable is that portion that moves with the manipulator crane as it travels over the refueling cavity.

In order to return the manipulator crane radiation monitor to service, the area radiation monitor 2-RM-RMS-263 will be tagged out and its field cable used. A short piece of RMS cable will be used, together with standard cable connectors, to join the RMS-263 field cable to the detector for RMS-262. This temporary modification will only be installed after the manipulator crane is parked in its normal storage location and no additional movement is planned. This will insure that the temporary cable installation will not be damaged.

In the control room, the field cable normally used for 2-RM-RMS-263 will then be connected to the instrument drawer for 2-RM-RMS-262. 2-RM-RMS-263 is an area radiation monitor that is not required to be operable during Mode 6.

After Mode 5 is reached, containment purge is no longer required. At this time, the temporary cable will be removed, the field cable reinstalled on 2-RM-RMS-263, and the Temporary Modification removed.

No unreviewed safety question exists, the manipulator crane radiation monitor will function as designed, thus allowing containment purge to be placed in service. The area radiation monitor will not be affected in any way after the Temporary Modification is removed.

Safety Evaluation Number
95-SE-TM-13

Description Of Temporary Modification
Temporary Modification 95-1092
This temporary modification installs a controlled supply of Nitrogen (GN) system jumper to the Condenser Hotwell at an unused conductivity cell connection "A" and/or "B" condensers. The Nitrogen is supplied from the Feedwater Nitrogen blanket supply (2-GN-170) or high pressure Nitrogen bottle.

Safety Evaluation Summary
A discussion of the effect of the increased flow through the Air Ejector Radiation Monitor 2-SV-RM-221 can be found in Temporary Modification 95-N2-1082 and 10CFR 50.59 Safety Evaluation 95-SE-TM-02.

This Temporary Modification will test whether 3-10 SCFM continuous flow of Nitrogen into the Main Condenser will improve the air removal efficiency And cause a reduction in dissolved Oxygen in the Condensate. Other plants have used this method to reduce their dissolved Oxygen levels. One theory for this is a sufficient flow of non-condensables is not available to allow the air removal section of the Air Ejector to function as designed. Thus the Oxygen does not get swept out of the Condenser. Therefore, if some Nitrogen were continuously introduced into the Condenser, the resulting increase in non-condensable flowrates may entrain more of the Oxygen, and sweep it out along with the Nitrogen. Total Nitrogen flowrate shall not exceed 10 SCFM.

Since a continuous stream of non-condensable gas Nitrogen will be introduced into the Condenser, Logs and calculations of Condenser in-leakage need to be adjusted by subtracting the Nitrogen in-flow, in order to properly obtain a true value for air in-leakage.

Injecting Nitrogen below the normal Condenser Hotwell water level will not affect operation of the Condenser at the rate of addition allowed. Since the maximum flow rate of the Air Ejector is approximately 30 SCFM, the nitrogen will not build up in the condenser sufficiently to interfere with heat transfer. Nitrogen gas was chosen since it is inert and will not support corrosion or other adverse affects to the system or introduce a fire hazard. It does however, present an asphyxiation hazard and can create a life threatening atmosphere in a confined space. The station Confined Space Entry permitting program is sufficient to insure this does not pose an additional personnel hazard. Operations will need to include the Temporary Modification on the Abnormal Equipment status Log to verify the supply is isolated and verified before opening the condenser for maintenance.

As explained in the 95-SE-TM-02 this modification will allow the AEJ RM to remain in service and accurately present the radiation level of the AEJ effluent. This will ensure that its passive and active functions will be preserved. The function and operation of the Air ejector RM is unchanged. The intentional introduction of Nitrogen is similar in effect to in-leakage experienced in lesser integrity condensers. This additional flow will still be monitored by the installed flow meters. UFSAR section 10.4.6-1 states that the normal anticipated exhaust is 12.5 SCFM per air ejector. The current total of 2.0 SCFM is significantly less than this value. Therefore, this modification does not result in an unreviewed safety question does not increase the likelihood of an event or decrease the margin of safety.

Safety Evaluation Number
95-SE-TM-14

Description Of Temporary Modification
Temporary Modification 95-N1-1617
This temporary modification installs a controlled supply of Nitrogen (GN) system jumper to the Condenser Hotwell at an unused conductivity cell connection "A" and/or "B" condensers. The Nitrogen is supplied from the Feedwater Nitrogen blanket supply (1-GN-413) or high pressure Nitrogen bottle.

Safety Evaluation Summary
A discussion of the effect of the increased flow through the Air Ejector Radiation Monitor 2-SV-RM-221 can be found in Temporary Modification 95-N2-1082 and 10CFR 50.59 Safety Evaluation 95-SE-TM-02. These discussions apply to this TM for the Unit 1 AEJ as well and are provided for reference.

This Temporary Modification will test whether 3-10 SCFM continuous flow of Nitrogen into the Main Condenser will improve the air removal efficiency And cause a reduction in dissolved Oxygen in the Condensate. Other plants have used this method to reduce their dissolved Oxygen levels. One theory for this is a sufficient flow of non-condensables is not available to allow the air removal section of the Air Ejector to function as designed. Thus the Oxygen does not get swept out of the Condenser. Therefore, if some Nitrogen were continuously introduced into the Condenser, the resulting increase in non-condensable flowrates may entrain more of the Oxygen, and sweep it out along with the Nitrogen. Total Nitrogen flowrate shall not exceed 10 SCFM.

Since a continuous stream of non-condensable gas Nitrogen will be introduced into the Condenser, Logs and calculations of Condenser in-leakage need to be adjusted by subtracting the Nitrogen in-flow, in order to properly obtain a true value for air in-leakage.

Injecting Nitrogen below the normal Condenser Hotwell water level will not affect operation of the Condenser at the rate of addition allowed. Since the maximum flow rate of the Air Ejector is approximately 30 SCFM, the nitrogen will not build up in the condenser sufficiently to interfere with heat transfer. Nitrogen gas was chosen since it is inert and will not support corrosion or other adverse affects to the system or introduce a fire hazard. It does however, present an asphyxiation hazard and can create a life threatening atmosphere in a confined space. The station Confined Space Entry permitting program is sufficient to insure this does not pose an additional personnel hazard. Operations will need to include the Temporary Modification on the Abnormal Equipment status Log to verify the supply is isolated and verified before opening the condenser for maintenance.

As explained in the 95-SE-TM-02 this modification will allow the AEJ RM to remain in service and accurately present the radiation level of the AEJ effluent. This will ensure that its passive and active functions will be preserved. The function and operation of the Air ejector RM is unchanged. The intentional introduction of Nitrogen is similar in effect to in-leakage experienced in lesser integrity condensers. This additional flow will still be monitored by the installed flow meters. UFSAR section 10.4.6-1 states that the normal anticipated exhaust is 12.5 SCFM per air ejector. The current total of 3.0 SCFM is significantly less than this value. Therefore, this modification does not result in an unreviewed safety question does not increase the likelihood of an event or decrease the margin of safety.

Safety Evaluation Number
95-SE-TM-15

Description Of Temporary Modification
Temporary Modification N1-1859
Installation of a portable AC unit in the Unit 1 alleyway (located outside of the Unit 1 Rod Drive Room). Installation of a ventilation trunk from the portable AC unit to the doorway of the Unit 1 Rod Drive Room. This temporary arrangement will be used to provide additional cooling air to the Unit 1 Rod Drive Room, Cable Vault, and Cable Tunnel areas.

Safety Evaluation Summary
This evaluation is performed for the installation of a portable AC unit in the Unit 1 alleyway (located outside of the Unit 1 Rod Drive Room), and installation of a ventilation trunk from the portable AC unit to the doorway of the Unit 1 Rod Drive Room. This temporary arrangement will be used to provide additional cooling air to the Unit 1 Rod Drive Room, Cable Vault, and Cable Tunnel areas. The following concerns were addressed in the determination that an unreviewed safety question does not exist:

1) Location:

The temporary AC unit is positioned outside the Unit 1 Rod Drive Room. It is not located near any safety related equipment. Therefore, during an earthquake or tornado, no equipment required for safe shutdown would be damaged due to the temporary AC unit.

2) Affect on Control Room Envelope:

The portable AC unit will provide additional cooling air flow to the Rod Drive Room, as well as the Cable Vault and Tunnel. While it is in service, the potential exists, due to the air supply capacity of the areas vs. the air exhaust capabilities, to pressurize the Rod Drive and Cable Vault and Tunnel spaces. This might decrease the Control Room envelope DP between the Control Room and Cable Vault and Tunnel to less than allowable limits. DP will be monitored (via operator logs) periodically and following ventilation lineup changes to ensure DP is in spec. If not in spec, the CR habitability 7 day action statement will be entered, and the ventilation alignment will be corrected to restore the proper DP. This will ensure design Control Room habitability is maintained during any accident condition so that the 5 rem whole body limit is not exceeded for Control Room personnel during any postulated accident.

3) Additional Cooling to Affected Areas:

Additional cooling to the areas will aid in the reliable performance of the equipment housed in the areas. (i.e: Rod Control, MCCs)

4) Power Supply:

The temporary AC unit will be powered by a non-emergency power supply. The power will be supplied from 1C1-1A2R, which is a unit heater in the Unit 1 Aux feed pump house. This power supply will not affect the aux feed system in any way. For these reasons, an unreviewed safety question does not exist, and the temporary modification should be allowed.

Safety Evaluation Number
95-SE-TM-16
Description Of Temporary Modification
Temporary Modification #N1-95-1620
The EHC line upstream of 1-EH-12 is to be crimped to stop EHC flow to the EHC filters.

Safety Evaluation Summary
The EHC pump discharge bypass line that flows through the EHC filters has a leak at a weld union which is degrading. It is not desired to shut down the unit to repair the leak. Therefore, the EHC line upstream of 1-EH-12 is to be crimped to stop EHC flow to the EHC filters. This Temporary Modification will provide sufficient pressure drop so that the degraded section of tubing will be at close to atmospheric pressure which should eliminate concern over catastrophic failure of the EHC line.

Precautions that will be taken to ensure that the problem is corrected and that the proposed TM will not further degrade the EHC system are listed below:
• Mockup testing is required using the developed crimping tool. NDE inspection of the section will be performed to ensure that the crimping will not create a loss of tubing integrity. [These requirements are contained in the MDAP-0019 instructions.]
• The crimped / defective section of EHC tubing should be replaced during the next refueling outage. [The requirement to submit a Work Order is included in the MDAP-0019 instructions.]

This TM will isolate flow to the fuller earth filter and cellulose filter - the function of the EHC system will not be altered. Main Turbine valve control and overspeed protection systems will not be altered. EHC fluid cleanup can be performed using alternative methods if required.

The potential for equipment damage or personnel injury is lessened by this TM. This TM will prevent the catastrophic failure of the EHC piping system by isolating the leaking line when the leak becomes uncomfortably large.

The trip functions of the turbine-generator remain functional with the installation of the TM. In the unlikely event of a loss of EHC before, during, or after installation of the TM, the turbine valves will close as designed to trip the turbine.

Since this TM will not affect turbine operation (including turbine overspeed protection or trip functions), and since this TM will be used to prevent the loss of the EHC system due to a worsening leak, an unreviewed safety question does not exist.

Safety Evaluation Number
95-SE-TM-16 Rev 1.

Description Of Temporary Modification
Temporary Modification #N1-95-1620
The EHC line upstream of 1-EH-12 is to be crimped to stop EHC flow to the EHC filters.

Safety Evaluation Summary
The EHC pump discharge bypass line that flows through the EHC filters has a leak at a weld union which is degrading. It is not desired to shut down the unit to repair the leak. Therefore, the EHC line upstream of 1-EH-12 is to be crimped to stop EHC flow to the EHC filters. This Temporary Modification will provide sufficient pressure drop so that the degraded section of tubing will be at close to atmospheric pressure which should eliminate concern over catastrophic failure of the EHC line.

Precautions that will be taken to ensure that the problem is corrected and that the proposed TM will not further degrade the EHC system are listed below:
• Mockup testing is required using the developed crimping tool. NDE inspection of the section will be performed to ensure that the crimping will not create a loss of tubing integrity. [These requirements are contained in the MDAP-0019 instructions.]
• The crimped / defective section of EHC tubing should be replaced during the next refueling outage. [The requirement to submit a Work Order is included in the MDAP-0019 instructions.]

This TM will isolate flow to the fuller earth filter and cellulose filter - the function of the EHC system will not be altered. Main Turbine valve control and overspeed protection systems will not be altered. EHC fluid cleanup can be performed using alternative methods if required.

The potential for equipment damage or personnel injury is lessened by this TM. This TM will prevent the catastrophic failure of the EHC piping system by isolating the leaking line when the leak becomes uncomfortably large.

The trip functions of the turbine-generator remain functional with the installation of the TM. In the unlikely event of a loss of EHC before, during, or after installation of the TM, the turbine valves will close as designed to trip the turbine.

A Furmanite box was previously installed over the pinhole leak in the EHC line. (This is not the injection of Furmanite between crimp locations discussed in question 42.) Also, a clamp has been left on a crimp location in order to better maintain delta-p across the crimp and minimize or eliminate leakage past the crimp. Civil DEO has inspected the EHC line with the Furmanite box added, crimps made, and clamp added even though the EHC line is not seismically qualified. The determination of Civil DEO was that the additional weight of the Furmanite box and the clamp will not adversely affect the loading on the EHC line and does not pose an additional threat to the line or surrounding equipment in the event of a seismic event.

Since this TM will not affect turbine operation (including turbine overspeed protection or trip functions), and since this TM will be used to prevent the loss of the EHC system due to a worsening leak, an unreviewed safety question does not exist.

Safety Evaluation Number
95-SE-TM-17

Description Of Temporary Modification
Temporary Modification # 95-N2-1093
Temporarily remove the actuation input from the Bearing Cooling Tower Fire Protection System to the Bearing Cooling Fan trip circuitry.

Safety Evaluation Summary
The Bearing Cooling and affected Fire Protection systems are not safety related nor are they required by Technical Specifications. Installation of the jumper does not affect any safety related systems and would not affect the ability of any safety related systems to perform its function for safe shutdown of the units.

A fire watch will be posted while this Temporary Modification is installed. The purpose of the Fire Watch is to provide detection and notification of any BC Tower Cell fire. Upon notification, actions will be taken to secure BC Fans, dispatch the Fire Brigade, and attempt to initiate Tower Cell deluge.

The probability of a fire is not increased by the installation of this Temporary Modification. Temporarily defeating portions of the FP control/protection circuit during the system maintenance, while maintaining BC Fan operation, has no ability to influence the mechanisms by which a fire is generated.

The overall operational consequences of a loss of Bearing Cooling from a Tower fire remains the same - shutdown of the Secondary Side Plant.

No safety related systems or structures are altered or affected by the Temporary Modification. The TM is limited to the BC and associated FP systems and does not create the means to cause a new or different malfunction of the tower, system, or surrounding equipment.

The margin of safety for the station as described in the Technical Specification Bases is not altered since the BC system and its associated FP system are not contained or described in the Technical Specifications.

An Unreviewed Safety Question is therefore not created by the Temporary Modification. The activity should be allowed.