VIRGINIA ELECTRIC AND POWER COMPANY
RICHMOND, VIRGINIA 23261

October 3, 1988

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

Serial No. 88-467
PES/ISI/DJF
Docket No. 50-339
License No. NPF-7

Gentlemen:

<u>VIRGINIA ELECTRIC AND POWER COMPANY</u>
<u>NORTH ANNA POWER STATION UNIT 2</u>
<u>ASME SECTION XI RELIEF REQUESTS</u>

In accordance with 10 CFR 50.55a paragraph (g)(5), we are requesting relief from various examination requirements of ASME Section XI for North Anna Unit 2. The particular relief requests are identified in Attachments 1 and 2. Attachment 1 addresses relief requests from hydrostatic pressure testing requirements, and Attachment 2 addresses relief requests from NDE examinations.

The relief requests for NDE examinations are to the 1974 Edition of ASME Section XI through the Summer 1975 Addenda. The hydrostatic pressure tests are performed in accordance with the 1977 Edition of the Code through the Summer 1979 Addenda as documented in the NRC letter to Virginia Electric and Power Company dated June 7, 1982. Relief is requested from this edition of the code for hydrostatic pressure testing.

North Anna Unit 2 will begin its first interval ten year ISI examinations during the scheduled February 1989 outage and complete these examinations during the next outage presently scheduled for August 1990. Review of these relief requests is desired in time to support the 1989 outage.

If you have any questions or require further information, please advise.

Enclosed is a check in the amount of $150 for the review application fee.

Very truly yours,

RJ Saunders for

W. R. Cartwright
Vice President - Nuclear

A047 w/check $150 # 018698
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Attachments

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PDR ADOCK 05000339
P PNU

cc: U.S. Nuclear Regulatory Commission
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Mr. J. L. Caldwell
NRC Senior Resident Inspector
North Anna Power Station

COMMONWEALTH OF VIRGINIA)
)
COUNTY OF HENRICO)

The foregoing document was acknowledged before me, in and for the County and Commonwealth aforesaid, today by W. R. Cartwright who is Vice President - Nuclear, of Virginia Electric and Power Company. He is duly authorized to execute and file the foregoing document in behalf of that Company, and the statements in the document are true to the best of his knowledge and belief.

Acknowledged before me this 3 day of October, 19 88.

My Commission expires: February 25, 19 90.

Vicki L. Hull
Notary Public

(SEAL)

RELIEF REQUESTS FOR 10-YEAR HYDROSTATIC TESTS

RELIEF REQUEST SPT-1 - REQUEST TO TEST AT LOWER PRESSURE

1. System - Chemical and Volume Control

2. Components - Piping on drawing 12050-FM-95C between the pumps and first flange from pumps.

PUMPS	LINE
2-RC-P-1A	2"-CH-414-1502
2-RC-P-1B	2"-CH-415-1502
2-RC-P-1C	2"-CH-416-1502

3. Code Requirements - Class 1 System Hydrostatic Test per IWB-5222, 1977 Edition through Summer 1979 Addenda of Section XI, 1.10 times system nominal operating pressure.

4. Proposed Alternative Examination

   The normal system leakage test after each refueling is an adequate examination.

5. Reason for Relief

   Pressurizing the piping listed above will also pressurize the number one seal of the reactor coolant pumps. This could potentially damage the number one seal.

   Relief was granted to Surry Power Station Unit 2 per Safety Evaluation Report dated January 24, 1986 for the same situation as described above. Relief was also granted to North Anna Power Station Unit 1 per Safety Evaluation Report dated July 13, 1987 (TAC No. 64718) for the same situation as described above.

RELIEF REQUEST SPT-2 - REQUEST TO TEST AT LOWER PRESSURE

1. System - Chemical and Volume Control

2. Component - Piping located on drawing 12050-FM-95C between the valves listing below:

Valves	Line
HVC-2311 and 2-CH-341	2"-CH-468-1502
2-CH-358 and 2-CH-340	3"-CH-401-1502

3. Code Requirements - Class 1 System Hydrostatic Test per IWB-5222. Po=2500 psig, To=496°F, Test Pressure is 2550 psig per IWB-5222.

4. Proposed Alternative Examination

   As an alternative, the Reactor Coolant System will be pressurized to a pressure as close as practical to 2335 psig but not less than 2300 psig while the reactor is in a shutdown condition to create a pressure boundary at check valves 2-CH-341 and 2-CH-358. The components listed above will then be tested to a pressure (2300 psig < test pressure < 2335 psig) as close as practical to the Reactor Coolant System pressure using a charging pump.

5. Reason for Relief

   Check valves 2-CH-341 and 2-CH-358 prevent the components listed above from being pressurized without pressurizing the Reactor Coolant System. The code required test pressure of 2550 psig will overpressurize the Reactor Coolant System.

   Also, the power operated relief valves (PCV-2456 and PCV-2455C) of the Reactor Coolant System are designed to limit the pressurizer pressure to a value below the fixed high-pressure reactor trip setpoint (2385 psig). The relief valve setpoints are 2335 psig. It is not desirable to take the Reactor Coolant System above the power operated relief valve setpoint.

   Similar relief was granted to Surry Power Station Unit 2 per Safety Evaluation Report dated January 24, 1986 (Docket 50-281) for components with the same design configuration. Relief was also granted to North Anna power Station Unit 1 per Safety Evaluation Report dated July 13, 1987 (TAC No. 64718) for the same situation as described above.

RELIEF REQUEST SPT-3 - REQUEST TO TEST AT LOWER PRESSURE

1. System - Chemical and Volume Control

2. Components - Piping between the valves listed below located on drawings 12050-FM-95C and 12050-FM-95B.

Valves	Line
2-CH-358, 2-CH-HCV-2311, and 2-CH-MOV-2289A	3/4"-CH-640-1502 2"-CH-468-1502 3"-CH-401-1502 3"-CH-479-1502 3"-CH-819-1502

3. Code Requirements - Class 2 System Hydrostatic Test per IWC-5222. Since there are no relief valves for the above components, test pressure per IWC-5222 is 3419 psig.

4. Proposed Alternative Examination

   As an alternative, the Reactor Coolant System will be pressurized to a pressure as close as practical to 2335 psig but not less than 2300 psig while the reactor is in a shutdown condition to create a pressure boundary at check valves 2-CH-341 and 2-CH-358. The components listed above will then be tested to a pressure (2300 psig < test pressure < 2335 psig) as close as practical to the Reactor Coolant System pressure using a charging pump.

5. Reason for Relief

   Check valves 2-CH-341, 2-CH-340 and 2-CH-358 prevent the components listed above from being pressurized without pressurizing the Reactor Coolant System. The Code required test pressure of 3419 psig will overpressurize the Reactor Coolant System. Also, the power operated relief valves (PCV-2456 and PCV-2455C) of the Reactor Coolant System are designed to limit the pressurizer pressure to a value below the fixed high-pressure reactor trip setpoint (2385 psig). The relief valve setpoints are 2335 psig. It is not desirable to take the Reactor Coolant System above the power operated relief valve setpoint.

   Relief was granted to North Anna Power Station Unit 1 per Safety Evaluation Report dated July 13, 1987 (TAC No. 64718) for the same situation as described above.

RELIEF REQUEST SPT-4 - REQUEST TO TEST IN ACCORDANCE WITH WESTINGHOUSE TECHNICAL MANUAL

1. System - Feedwater

2. Components - Piping between the valves listed below located on drawing 12050-FM-74A.

Valve	Connecting Line	Valve
2-FW-64	3"-WAPD-410-601 to 3"-WAPD-409-601	2-FW-66
2-FW-66	3"-WAPD-409-601	2-FW-70
2-FW-96	3"-WAPD-412-601 to 3"-WAPD-411-601	2-FW-98
2-FW-98	3"-WAPD-411-601	2-FW-102
2-FW-128	3"-WAPD-414-601 to 3"-WAPD-413-601	2-FW-130
2-FW-130	4"-WAPD-413-601	2-FW-134
2-FW-278	4"-WAPD-439-601 to 3"-WAPD-409-601	2-FW-66

3. Code Requirements - Class 2 System Hydrostatic Test per IWA 5213(d) and IWC-5222. P_d=1400 psig, T_d<200°F, Test Pressure is 1540 psig per IWC-5222.

4. Proposed Alternative Examination

   Since the components listed above cannot be pressurized without pressurizing the steam generator, they must be tested per the required manufacturer's hydrostatic test method for the steam generators. Therefore, the proposed alternative examination is the examination described in the Westinghouse Technical Manual for the secondary side of the steam generator. The examination is to pressurize the secondary side of the steam generator to 1356 psig, hold for 30 minutes, and then reduce to the design pressure (1085 psig) for 3 1/2 hours. A VT-2 examination will then be performed.

5. Reason for Relief

   Due to check valves 2-FW-134, 2-FW-102, and 2-FW-70 the piping listed above cannot be pressurized without pressurizing the steam generators. The code required test pressure of 1540 psig would overpressurize the steam generator.

   Relief was Granted to North Anna Power Station Unit 1 per Safety Evaluation Report dated July 13 1987 (TAC No. 64718) for the same situation as described above.

RELIEF REQUEST SPT-5 - REQUEST TO TEST AT LOWER PRESSURE

1. System - Feedwater, Chemical and Volume Control and Safety Injection

2. Components - Centrifugal pumps and discharge piping to first isolation valve on drawings 12050-FM-74A, 12050-FM-95B, 12050-FM-96A.

Pumps

2-FW-P-2, 2-FW-P-3A, 2-FW-P-3B
2-CH-P-1A, 2-CH-P-1B, 2-CH-P-1C
2-SI-P-1A, 2-SI-P-1B

3. Code Requirements - Class 2 System Hydrostatic Tests per IWC-5222 and IWD-5223.

4. Proposed Alternative Examination

   As an alternative, the test pressure for the pump discharge and associated piping extending to the first shutoff valve on the discharge side of the pump shall be the same as that required for the piping and components on the suction side of the pump. In support of this alternative, the 1980 Edition, Winter of 1981 Addenda of ASME Section XI (approved by the NRC) paragraph IWA-5224(d) allows the system test boundary interface to be the first shutoff valve on the discharge side of the centrifugal pump when the primary system pressure ratings on the suction and discharge sides differ.

5. Reason for Relief

   Centrifugal pumps and the portions of the pump discharge lines up to the first isolation valve cannot be isolated from the pump suction. If the discharge piping were pressurized to the required test pressure the suction piping would be subjected to a pressure far in excess if its design with the potential for permanent damage to piping and components.

   Relief was granted to North Anna Power Station Unit 1 per Safety Evaluation Report dated July 13, 1987 (TAC No. 64718) for the same situation as described above.

RELIEF REQUEST SPT-6 - REQUEST NOT TO PERFORM VISUAL EXAMINATION

1. System - Reactor Coolant

2. Component - Bottom of reactor vessel.

3. Code Requirements - Visual Examination per IWA-5240 and IWB-5222 during System Hydrostatic Test.

4. Proposed Alternative Examination

   The proposed alternative examination is to examine the bottom of the reactor vessel for evidence of leakage during the 10 year vessel inspection, and to perform a visual inspection of the lower head.

5. Reason for Relief

   The system hydrostatic test for the Reactor Coolant System is performed at Hot Standby, MODE 3. The bottom of the reactor vessel is inaccessible due to temperature and radiological concerns.

   Relief was granted to North Anna Power Station Unit 1 per Safety Evaluation Report dated July 13, 1987 (TAC No. 64718) for the same situation as described above.

RELIEF REQUEST SPT-7 - REQUEST TO TEST AT LOWER PRESSURE

1. System - Safety Injection

2. Components - Piping between the following sets of valves located on station print 12050-FM-96B

Valves	Lines
2-SI-92, 2-SI-90, and 2-SI-91	6"-SI-531-1502
2-SI-100, 2-SI-98, and 2-SI-99	6"-SI-533-1502
2-SI-106, 2-SI-104, and 2-SI-105	6"-SI-532-1502
2-SI-125, 2-SI-123, and 2-SI-112	6"-SI-416-1502 2"-SI-463-1502
2-SI-113, 2-SI-111, and 2-SI-117	6"-SI-419-1502 2"-SI-459-1502
2-SI-113, 2-SI-116, and 2-SI-124	6"-SI-421-1502 2"-SI-461-1502

3. Code Requirements - Class 1 System Hydrostatic Test per IWB-5222. Po=2235 psig, To=160°F, Test Pressure per IWB-5222 is 2432 psig.

4. Proposed Alternative Examination

   As an alternative, the Reactor Coolant System will be pressurized to a pressure as close as practical to 2335 psig but not less than 2300 psig while the reactor is in a shutdown condition to create a pressure boundary at the first valve of each set listed above. These components will then be tested to a pressure (2300 psig < test pressure < 2335 psig) as close as practical to the Reactor Coolant System pressure using a charging pump. The Reactor Coolant System will be borated equal to or greater than cold shutdown boron concentration.

5. Reason for Relief

   The first valve listed in each set prevents the components listed above from being pressurized without pressurizing the Reactor Coolant System. The power operated relief valves - (PCV-2456 and PCV-2455C of the Reactor Coolant System) are designed to limit the pressurizer pressure to a value below the fixed high-pressure reactor trip setpoint (2385 psig). The relief valve setpoints are 2335 psig which is below the test pressure of 2432 psig. It is not desirable to take the Reactor Coolant System above the power operated relief valve setpoint.

   Similar relief was granted to Surry Power Station Unit 2 per Safety Evaluation Report dated January 24, 1986 (Docket 50-281) for components with the same design configuration. Relief was granted to North Anna Power Station Unit 1 per Safety Evaluation Report dated July 13, 1987 (TAC No. 64718) for the same situation as described above.

RELIEF REQUEST SPT-8 - REQUEST TO TEST AT LOWER PRESSURE

1. System - Safety Injection

2. Components - Piping and valves listed below and located on drawings 12050-FM-96A and 12050-FM-96B.

Valve	Connecting Line	Valve
MOV-2890C and MOV-2890D	10"SI-418-1502/10"-SI-624-1502	2-SI-91
	to 6"-SI-531-1502	2-SI-105
	to 6"-SI-532-1502	2-SI-99
	to 6"-SI-533-1502	
MOV-2890A	10"-SI-415-1502	2-SI-112
	to 6"-SI-416-1502	2-SI-117
	to 6"-SI-419-1502	
	to 6"-SI-530-1502	2-SI-124
	to 6"-SI-421-1502	
MOV-2890B	10"-SI-540-1502	
	to 6"-SI-421-1502	2-SI-124
2-SI-89	2"-SI-451-1502	2-SI-90
2-SI-97	2"-SI-453-1502	2-SI-98
2-SI-103	2"-SI-455-1502	2-SI-104

3. Code Requirements - Class 2 System Hydrostatic Test per IWC-5222. P_d= 2485 psig, Design Temperature is less than 200°F, Test pressure is 2733.5 psig.

4. Proposed Alternative Examination

   As an alternative, the Reactor Coolant System will be pressurized to a pressure as close as practical to 2335 psig but not less than 2300 psig while the reactor is in a shutdown condition to create a pressure boundary at check valves 2-SI-92, 2-SI-100, 2-SI-106, 2-SI-125, 2-SI-113, and 2-SI-118. These components will then be tested to a pressure (2300 psig < test pressure < 2335 psig) as close as practical to the Reactor Coolant System pressure using a test pump.

5. Reason for Relief

   Check valves 2-SI-92, 2-SI-100, 2-SI-106, 2-SI-125, 2-SI-113 and 2-SI-118 prevent the components listed above from being pressurized without pressurizing the Reactor Coolant System. The Code required test pressure of 2733.5 psig will overpressurize the Reactor Coolant System.

The power operated relief valves (PCV-2456 and PCV-2455C) of the Reactor Coolant System are designed to limit the pressurizer pressure to a value below the fixed high-pressure reactor trip setpoint (2385 psig). The relief valve setpoints are 2335 psig which is below the test pressure of 2733.5 psig. It is not desirable to take the Reactor Coolant System above the power operated relief valve setpoint.

Similar relief was granted to Surry Power Station Unit 2 per Safety Evaluation Report dated January 24, 1986 (Docket 50-281) for components with the same design configuration. Relief was also granted to North Anna Power Station Unit 1 per Safety Evaluation Report dated July 13, 1987 (TAC No. 64718) for the same situation as described above.

RELIEF REQUEST SPT-9 - REQUEST TO TEST AT LOWER PRESSURE

1. System - Safety Injection

2. Component - Piping between the sets of valves listed below located on drawing 12050-FM-96B.

Valves	Line
MOV-2865A and 2-SI-151	12"-SI-523-1502
2-SI-151 and 2-SI-149	3/4"-SI-478-1502
MOV-2865B and 2-SI-168	12"-SI-524-1502
2-SI-168 and 2-SI-166	3/4"-SI-484-1502
MOV-2865C and 2-SI-185	12"-SI-525-1502
2-SI-185 and 2-SI-183	3/4"-SI-480-1502

3. Code Requirements - Class 2 System Hydrostatic Test per IWC-5222. P_d=2485 psig, T_d<200°F, Test pressure per the code is 2733.5 psig since there is no over-pressure protection for the above components.

4. Proposed Alternative Examination

   As an alternative it it requested that the Class 2 components listed above be tested per IWB-5222. The nominal operating pressure is 660 psig and temperature is 120°F. Thus, testing per IWB-5222 would require a test pressure of 724 psig. This should be adequate considering the nominal operating conditions.

5. Reason for Relief

   Check valves 2-SI-151, 2-SI-168, and 2-SI-185 at the class 1 and 2 system boundaries prevent the pressurization of the above components without pressurizing the primary system. The required test pressure is 2733.5 psig as stated above, which would over-pressurize the primary system.

   Relief was granted to North Anna Power Station Unit 1 per Safety Evaluation Report dated July 13, 1987 (TAC No. 64718) for the same situation as described above.

RELIEF REQUEST SPT-10 - REQUEST TO TEST IN ACCORDANCE WITH WESTINGHOUSE TECHNICAL MANUAL

1. System - Main Steam
   (Steam) Decay Heat Release
   Feedwater
   Chemical Feed
   Blowdown

2. Components - Steam generator and piping located on drawings 12050-FM-70B, 12050-FM-74A, 12050-FM-89B, 12050-FM-98A, and 12050-FM-102A.

Component	Connected Piping	Component
2-RC-E-1A	32"-SHP-401-601	SV-MS-201A
	to 32"-SHP-422-601	SV-MS-202A
		SV-MS-203A
		SV-MS-204A
		SV-MS-205A
	to 6"-SHP-437-601/1"-SHP-484-601	PCV-MS-201A
	to 3"-SHP-464-601/1"SHP-478-601	2-MS-335
		2-MS-18
	to 1 1/2"-SHPD-406-601	2-MS-22
	to 1/2"-SHPD-471-601	2-MS-26
2-RC-E-1A	32"-SHP-401-601	2-MS-35
		2-NRV-MS-201A
	3"-SHP-460-601	2-MS-344
2-RC-E-1A	32"-SHP-401-601	
	to 32"-SHP-422-601	
	to 3"-SHP-445-601	2-MS-344
	to 3"-SHP-562-601	NRV-MS-203A
	to 1"-SHP-571-601/1"-SHP-555-601	2-MS-346
		2-MS-348
2-RC-E-1A	2"-SS-620-601/1"-SS-753-601	2-SS-83
2-RC-E-1A	32"-SHP-401-601	
	to 32"-SHP-422-601	
	to 3"-SDHV-401-601	
	to 4"-SDHV-404-601	2-MS-20
2-RC-E-1A	16"-WFPD-424-601	2-FW-62
		2-FW-70
	to 3/4"-CFPD-401-902	2-WT-42
2-RC-E-1A	2"-WGCB-404-601	2-BD-4
	2"-WGCB-405-601	2-BD-1
	1"-WGCB-406-601	2-BD-2

COMPONENTS	CONNECTED PIPING	COMPONENTS
2-RC-E-1B	32"-SHP-402-601	SV-MS-201B
	to 32"-SHP-423-601	SV-MS-202B
		SV-MS-203B
		SV-MS-204B
		SV-MS-205B
	to 6"-SHP-438-601/1"-SHP-485-601	PCV-MS-201B
	to 3"-SHP-465-601	2-MS-333
		2-MS-57
	to 1 1/2"-SHPD-408-601	2-MS-60
	to 1/2"-SHPD-473-601	2-MS-64
2-RC-E-1B	32"-SHP-402-601	2-MS-73
		NRV-MS-201B
	3"-SHP-461-601	2-MS-353
2-RC-E-1B	32"-SHP-402-601	
	to 32"-SHP-423-601	
	to 3"-SHP-446-601	NRV-MS-203B
	to 3"SHP-461-601	2-MS-353
	to 3"-SHP-563-601	2-MS-356
	to 1"-SHP-503-601	2-MS-357
2-RC-E-1B	2"-SS-625-601/1"-SS-754-601	2-SS-84
2-RC-E-1B	32"-SHP-402-601	
	to 32"-SHP-423-601	
	to 3"-SDHV-402-601	
	to 4"-SDHV-404-601	2-MS-20
2-RC-E-1B	16"-WFPD-423-601	2-FW-102
	to 3/4"-CFPD-402-902	2-FW-94
		2-WT-54
2-RC-E-1B	2"-WGCB-407-601	2-BD-13
	2"-WGCB-408-601	2-BD-10
	1"-WGCB-409-601	2-BD-11
2-RC-E-1C	32"-SHP-403-601	
	to 32"-SHP-424-601	SV-MS-201C
		SV-MS-202C
		SV-MS-203C
		SV-MS-204C
		SV-MS-205C
	to 6"-SHP-439-601/1"SHP-486-601	PCV-MS-201C
	to 3"-SHP-466-601	2-MS-95
		2-MS-331
	to 1 1/2"-SHPD-407-601	2-MS-98
	to 1/2" - SHPD-475-601	2-MS-102
2-RC-E-1C	32"-SHP-403-601	2-MS-111
		NRV-MS-201C

COMPONENTS	CONNECTED PIPING	COMPONENTS
2-RC-E-1C	3"-SHP-462-601	2-MS-362
	32"-SHP-403-601 to 32"-SHP-424-601 to 3"-SHP-447-601 to 3"-SHP-462-601 to 3"-SHP-564-601	2-MS-362
	to 1"-SHP-504-601/1"-SHP-557-601	NRV-MS-203C 2-MS-365 2-MS-366
2-RC-E-1C	2"-SS-627-601/1"-SS-755-601	2-SS-85
2-RC-E-1C	32"-SHP-403-601 to 32"-SHP-424-601 to 3"-SDHV-403-601 to 4"-SDHV-404-601	2-MS-20
2-RC-E-1C	16"-WFPD-422-601 to 3/4"-CFPD-403-902	2-FW-126 2-FW-134 2-WT-70
2-RC-E-1C	2"-WGCB-410-601 2"-WGCB-411-601 1"-WGCB-412-601	2-BD-22 2-BD-19 2-BD-20

3. Code Requirements - Class 2 System Hydrostatic Test per IWA-5213(d) and IWC-5222. For Feedwater components P_d=1100 psig, T_d-200-F, test pressure per IWC-5222 would be 1375 psig. For the Chemical Feed Components P_d=1775 psig, T_d-200-F, test pressure per IWC-5222 would be 1952.5 psig. The remaining components have P_d=1085 psig, T_d-200-F, test pressure per IWC-5222 would be 1356 psig.

4. Proposed Alternative Examination

The Westinghouse Technical Manual for the Steam Generator requires the secondary side to be pressurized to 1356 psig, held for 30 minutes and then reduced to the design pressure (1085 psig) for a sufficient time to permit proper examination of welds, closures and surfaces for leakage or weeping.

The secondary side will be held at 1356 psig for 30 minutes and then at 1085 psig for a minimum of 3 1/2 hours in accordance with the Code. A VT-2 examination will then be performed.

5. Reason for Relief

Westinghouse, the manufacturer of the steam generators, gives specific testing requirements for the steam generator which must also be applied to the components listed above due to the fact that the components cannot be isolated from the steam generators.

Relief was granted to North Anna Power Station Unit 1 per Safety Evaluation Report dated July 13, 1987 (TAC No. 64718) for the same situation as described above.

RELIEF REQUEST SPT-11 - REQUEST TO TEST AT LOWER PRESSURE

1. System - Component Cooling, Chemical and Volume Control, Fuel Pit Cooling, Safety Injection, Quench Spray, Recirc Spray, Service Water, and Sampling.

2. Components - Piping and components included in the system hydrostatic test boundary.

3. Code Requirement - Per IWA-5265(b)..."the imposed pressure on any component, including static head, will not exceed 106% of the specified test pressure for the system."

4. Proposed Alternative Examination

   Hydrostatic testing of systems that cannot be isolated to meet the system test pressure at the test boundary high point and the 106% system test pressure maximum at the test boundary low point shall be conducted by pressurizing to the system test pressure at the low point in the test boundary.

5. Reason for Relief

   Unisolable portions of the various systems within the system hydrostatic test boundary are located throughout the plant such that there are variations in elevation within the boundaries that would result in imposed pressure in excess of six percent of the specified test pressure. It is Virginia Electric and Power Company's desire to limit the test pressure imposed on system components to 106% of the specified test pressure (as required by paragraph IWA-5265(b)). Thus, due to the effects of static head, portions of the piping at higher elevations will be subjected to a test pressure lower than that specified. There is no practical method for isolating the piping segments to achieve the required test pressure at all elevations.

   In a Safety Evaluation Report on Duane Arnold Energy Center (Docket No. 50-331) dated March 31, 1986, relief was granted from IWA-5265(b) for situations as described above. Relief was also granted to North Anna Power Station Unit 1 per Safety Evaluation Report dated July 13, 1987 (TAC No. 64718) for the same situation as described above.

RELIEF REQUEST SPT-12 - REQUEST TO TEST AT LOWER PRESSURE

1. System - Residual Heat Removal

2. Component - Piping located on drawing 12050-FM-94A between valves listed below.

Valves	Line
MOV-2701 and MOV-2700	14"-RH-401-1502

3. Code Requirements - Class 1 System Hydrostatic Test IWB-5222. Po=2235 psig, To=650 degrees F, Test pressure per IWB-5222 is 2280 psig.

4. Proposed Alternative Examination

   As an alternative, the components listed above will be tested in accordance with IWC-5222. The test pressure will be 584 psig as determined by the setpoints of relief valves RV-2721A and RV-2721B (467 psig). This alternative is considered sufficient since the relief valves are set at 467 psig. As a result, line 14"-RH-401-1502 should not see a pressure significantly higher than 467 psig. In addition, MOV-2700 and MOV-2701 will not open if the Reactor Coolant pressure is >660 psig.

5. Reason for Relief

   During the system hydrostatic test of the primary system, MOV-2700 is closed in addition to MOV-2701 in order to prevent possible over-pressurization of the Residual Heat Removal System. Thus, the portion of the RHR system identified above cannot be pressurized with the primary system and due to design, it cannot be pressurized without opening one of the MOV's.

   Relief was granted to North Anna Power Station Unit 1 per Safety Evaluation Report dated July 13, 1987 (TAC No. 64718) for the same situation as described above.

ATTACHMENT II

Attached are the requests for relief from impractical Code requirements as stated in Articles IWB and IWC of the 1974 Edition of ASME Section XI with addenda through the Summer 1975 Addendum. Extent of coverage obtainable is not presently available for these components. This information will not be known until the components are examined in the future.

A process of identifying partial examination coverage will be instituted at North Anna Power Station. This process will involve alternate angle and alternate method evaluation. Once an examination of these components has been performed, the process will identify the total amount of coverage on all components in which 100% of the code required volume is not examined.

RELIEF REQUESTS FOR NDE EXAMINATIONS

RELIEF REQUEST NDE-1

I. IDENTIFICATION OF COMPONENTS

Control Rod Drive Housing Welds

II. IMPRACTICAL CODE REQUIREMENTS

Section XI of the ASME Boiler and Pressure Vessel Code, 1974 Edition, through the Summer 1975 Addendum, Table IWB-2500 states that the examination areas shall include essentially 100% of the weld metal and base metal for one wall thickness beyond the edge of the weld in the installed peripheral control rod drive housings only.

III. BASIS FOR RELIEF

Several of the peripheral housings are not accessible for ultrasonic examinations per the requirements of IWB-2500 due to the closeness of the non-removable insulation around and under the reactor vessel and instrumentation penetrations.

IV. ALTERNATE EXAMINATION

Housings on the inner portion of the head which are accessible will be substituted for the peripheral housings which are not accessible.

A volumetric examination will be performed to the substituted housings on the inner portion of the head.

RELIEF REQUEST NDE-2

I. IDENTIFICATION OF COMPONENTS

Reactor Vessel (2-RC-R-1): Head-To-Flange Weld

II. IMPRACTICAL CODE REQUIREMENTS

Section XI of the ASME Boiler and Pressure Vessel Code, 1974 Edition through the Summer 1975 Addendum, requires the Category B-C reactor vessel head-to-flange weld have a volumetric examination each inspection interval in accordance with subsection IWB-2500. A full volumetric examination is not practicable.

III. BASIS FOR RELIEF

The geometric configuration of the reactor vessel head-to-flange weld limits the extent to which ultrasonic examinations can be performed from the flange side of the weld.

IV. ALTERNATE EXAMINATION

A volumetric examination will be performed to the extent practicable.

RELIEF REQUEST NDE-3

I. IDENTIFICATION OF COMPONENTS

Pressurizer (2-RC-E-2): Shell-To-Head Weld

II. IMPRACTICAL CODE REQUIREMENTS

Section XI of the ASME Boiler and Pressure Vessel Code, 1974 Edition through the Summer 1975 Addendum, requires the pressurizer shell-to-head weld of Category B-B have a volumetric examination each inspection interval in accordance with Subsection IWB-2500. A full volumetric examination is not practicable.

III. BASIS FOR RELIEF

Examination of the vessel shell-to-head weld is limited by the vessel configuration. The vessel configuration restricts the scan and prevents complete examination of the volume as required by Table IWB-2500.

IV. ALTERNATE EXAMINATION

A volumetric examination will be performed to the extent practicable.

RELIEF REQUEST NDE-4

I. IDENTIFICATION OF COMPONENTS

Pressurizer (2-RC-E-2): Nozzle-To-Safe End Welds

II. IMPRACTICAL CODE REQUIREMENTS

Section XI of the ASME Boiler and Pressure Vessel Code, 1974 Edition through the Summer 1975 Addendum, requires the Category B-F pressurizer nozzle-to-safe end welds have a surface and volumetric examination each inspection interval in accordance with subsection IWB-2500. A full volumetric examination is not practicable.

III. BASIS FOR RELIEF

Examination of the pressurizer nozzle-to-safe end welds is limited by the geometry and surface condition of the nozzle. The physical geometry of the nozzle-to-safe end weld permits ultrasonic examination of only the weld and the pipe side of the weld.

IV. ALTERNATE EXAMINATION

The surface examination will be performed in accordance with the Code requirements and a volumetric examination will be performed on the weld and the base metal on the pipe side of the weld to the extent practicable.

RELIEF REQUEST NDE-5

I. IDENTIFICATION OF COMPONENTS

Pressurizer (2-RC-E-2) and Steam Generators (2-RC-E-1A, 2-RC-E-1B, and 2-RC-E-1C): Interior Clad Surfaces

II. IMPRACTICAL CODE REQUIREMENTS

Section XI of the ASME Boiler and Pressure Vessel Code, 1974 Edition through the Summer 1975 Addendum, requires the steam generator and pressurizer interior clad surfaces of Category B-I-2 have a visual examination of 100% of the clad patch areas each inspection interval in accordance with subsection IWC-2500. A visual examination is not practicable.

III. BASIS FOR RELIEF

Subsequent edition and addenda to the ASME Code, which have been approved by the NRC for incorporation into 10 CFR 50.55a, have deleted the cladding examination.

Recognizing this deletion and the intent of the ASME Section XI examination to provide monitoring of component degradation over the plant's service interval, it is our position that the radiation exposure and cost associated with the cladding examinations are not commensurate with the increase in safety realized. The clad examination results obtained during the first inspection interval will not be directly comparable to examination results in later intervals.

Based on the preceding factors, we request relief from the remaining Pressurizer and Steam Generator cladding examinations at North Anna Power Station Unit 2 during the first ten year inservice inspection interval.

IV. ALTERNATE EXAMINATION

No additional alternate examinations: later editions and addendum of ASME Section XI, approved by the NRC and incorporated into 10 CFR 50.55a, no longer require cladding examinations.

Relief was granted tc North Anna Power Station Unit 1 per Safety Evaluation Report dated July 13, 1987 (TAC No. 64718) for the same situation as described above.

RELIEF REQUEST NDE-6

I. IDENTIFICATION OF COMPONENTS

Reactor Coolant Pump (2-RC-P-1A, 2-RC-P 1B and 2-RC-P-1C): Pump casings and pressure retaining welds in pump casings

II. IMPRACTICAL CODE REQUIREMENTS

Section XI of the ASME Boiler and Pressure Vessel Code, 1974 Edition through the Summer 1975 Addendum, requires a volumetric examination be performed during each inspection interval on 100% of the pressure retaining welds of Category B-L-1 in at least one pump in each group of pumps performing similar functions in the system (e.g., reactor coolant pumps) and a visual examination of one pump casing of Category B-L-2 in each group of pumps performing similar functions in the system be performed.

III. BASIS FOR RELIEF

The North Anna Power Station Unit 2 reactor coolant pumps are Westinghouse Model 93 controlled leakage pumps. The Model 93 pump's casing is fabricated by welding two stainless steel castings together. Thus, there is one circumferential pressure boundary weld in the pumps that is to be examined in accordance with Category B-L-1.

Since the installation of these pumps, it has been recognized that a volumetric examination of the casing welds is not practical with today's ultrasonic techniques.

The physical properties of the stainless steel casting and weld material preclude a meaningful ultrasonic examination. The capability to examine these pump casing welds in the field did not exist until recently. In the spring of [illegible], an examination was performed on one of the reactor coolant pumps at the R.E. Ginna plant using the miniature linear accelerator (MINAC), which was built under an EPRI sponsored program. This equipment has been made available to other utilities, and currently constitutes the only viable examination method for the volumetric examination of reactor coolant pump welds.

The volumetric examination method is radiographic and is performed by placing the MINAC inside the pump casing and placing film on the outside of the pump. To perform the examination, the pump must be completely disassembled, including removal of the diffuser adapter. This amount of disassembly is far beyond the amount of disassembly performed for normal maintenance. Insulation must also be removed from the exterior of the pump casing.

The examination has been performed at four different sites, all of which have the Westinghouse Model 93 pump. The MINAC examination was performed at Ginna in the spring of 1981, at Point Beach Unit 1 in the fall of 1981, at Turkey Point Unit 3 early in 1982 and at H. B. Robinson Unit 2 later in 1982. No problems with the welds were found at any of the sites. A review of the original radiographs of the Point Beach Unit 1 pump was performed prior to the MINAC examination, and all the landmarks found were identified during field examination with no apparent change.

The successful performance of this volumetric examination using the MINAC at four different sites demonstrates that the method is capable of satisfying ASME Section XI examination requirements. However, the performance of the examination has shown there is a relatively high radiation exposure associated with it. The total exposure associated with insulation removal, disassembly, examination and reassembly of the pump has averaged about 40 man-rem per pump.

There have been no defects identified by the four examinations performed on these pumps to date. A volumetric examination was attempted at North Anna in 1982. A radioactive source was placed within the pump casing and film around the outside. The developed film did not meet the density requirements for an acceptable examination. This examination was attempted twice at Surry. Both examinations yielded similar results.

The pump casing examinations are also not justified from a cost/benefit perspective. The pump disassembly, examination and reassembly is estimated to cost $750,000.

IV. ALTERNATE EXAMINATION

A visual examination of the <u>external</u> surfaces of one pump's casing weld and a surface examination to the extent practicable of the <u>external</u> casing weld of one pump will be performed to the extent and frequency of Category B-L-1.

Relief was granted to North Anna Power Station Unit 1 per Safety Evaluation Report dated July 13, 1987 (TAC No. 64718) for the same situation as described above.

RELIEF REQUEST NDE-7

I. IDENTIFICATION OF COMPONENTS

Class 1 Valve Bodies Exceeding 4 in. Nominal Pipe Size

II. IMPRACTICAL CODE REQUIREMENTS

In Class 1 systems, valves which are greater than four inches nominal pipe size are subject to visual examination. These valves vary in size, design and manufacturer but are all manufactured from either cast stainless steel or carbon steel. None of the valve bodies are welded.

Section XI of the ASME Boiler and Pressure Vessel Code, 1974 Edition through the Summer 1975 Addendum, requires a visual examination be performed on the internal pressure boundary surfaces of one valve in each group of valves that are of the same constructional design, manufacturing method, and manufacturer and that perform similar functions in the system (Category B-M-2).

Since these examinations must be met whether or not the valves have to be disassembled for maintenance, this requirement is considered impractical.

III. BASIS FOR RELIEF

The requirement to disassemble primary system valves for the sole purpose of performing a visual examination of the internal pressure boundary surfaces has only a very small potential of increasing plant safety margins and a very disproportionate impact on expenditures of plant manpower and radiation exposure.

The performance of both carbon and stainless cast valve bodies has been excellent in PWR applications. Based on this experience and both industry and regulatory acceptance of these alloys, continued excellent service performance is anticipated.

A more practical approach that would essentially provide an equivalent sampling program and significantly reduced radiation exposure to plant personnel is to examine the internal pressure boundary of only those valves that require disassembly for maintenance purposes. This would still provide a reasonable sampling of the primary system valves and give adequate assurance that the integrity of these components is being maintained.

IV. ALTERNATE EXAMINATION

The visual examination of the internal pressure boundary surfaces will be performed, to the extent practical, when a valve is disassembled for maintenance purposes.

RELIEF REQUEST NDE-8

I. IDENTIFICATION OF COMPONENTS

Excess Letdown Heat Exchanger (2-CH-E-4): Head-To-Flange Weld

II. IMPRACTICAL CODE REQUIREMENTS

Section XI of the ASME Boiler and Pressure Vessel Code, 1974 Edition through the Summer 1975 Addendum, requires the excess letdown heat exchanger head-to-flange weld of Category C-A have a volumetric examination each inspection interval in accordance with subsection IWC-2500. A full volumetric examination is not practicable.

III. BASIS FOR RELIEF

The configuration of the excess letdown heat exchanger head-to-flange weld is such that a full volumetric examination is impractical. The flange side slopes and the head side contains 4 nozzles such that the required volume as described in Table IWC-2520 is not practicable.

IV. ALTERNATE EXAMINATION

A volumetric examination will be performed to the extent practicable.

RELIEF REQUEST NDE-9

I. IDENTIFICATION OF COMPONENTS

Non-Regenerative Heat Exchanger (2-CH-E-2): Shell-To-Head Weld and Shell-To-Flange Weld

II. IMPRACTICAL CODE REQUIREMENTS

Section XI of the ASME Boiler and Pressure Vessel Code, 1974 Edition through the Summer 1975 Addendum, requires the non-regenerative letdown heat exchanger shell-to-head weld and shell-to-flange weld of Category C-A have a volumetric examination each inspection interval in accordance with subsection IWC-2500. A full volumetric examination is not practicable.

III. BASIS FOR RELIEF

Examination of the vessel shell-to-head weld is limited by the intergral attachments and the shell-to-flange weld is limited by the slope of the flange. The integral attachments and slope of the flange restrict the scan and prevent complete examination of the volume as required by Table IWC-2520.

IV. ALTERNATE EXAMINATION

A volumetric examination will be performed to the extent practicable.

RELIEF REQUEST NDE-10

I. IDENTIFICATION OF COMPONENTS

Seal Water Injection Filters (2-CH-FL-4A and 2-CH-FL-4B): Shell-To-Flange Welds.

II. IMPRACTICAL CODE REQUIREMENTS

Section XI of the ASME Boiler and Pressure Vessel Code, 1974 Edition through the Summer 1975 Addendum, requires the Category C-A shell-to-flange welds on the seal water injection filters be volumetrically examined each inspection interval in accordance with subsection IWC-2500. A full volumetric examination is not practicable.

III. BASIS FOR RELIEF

The configuration of the seal water injection filter head-to-flange weld is such that a full volumetric examination as required by Table IWC-2520 is impracticable.

IV. ALTERNATE PROVISIONS

A volumetric examination will be performed to the extent practicable.

RELIEF REQUEST NDE-11

I. IDENTIFICATION OF COMPONENTS

Low Head Safety Injection Pumps (2-SI-P-1A and 2-SI-P-1B): Pump Casing Welds

II. IMPRACTICAL CODE REQUIREMENTS

Section XI of the ASME Boiler and Pressure Vessel Code, 1974 Edition through the Summer 1975 Addendum, requires pump casing welds in Category C-G, Item Number C3.1, have a volumetric examination each inspection interval in accordance with subsection IWC-2500. A volumetric examination of all of the pumps casing welds is not practicable.

III. BASIS FOR RELIEF

Each of the two low head safety injection pump casings have a total of five circumferential welds and five longitudinal welds. Three of the circumferential welds and three of the longitudinal welds are completely encased in concrete and are not accessible for examination. Of the remaining two longitudinal welds, one weld is partially encased in concrete and one weld is partially covered by a vibration plate. Volumetric examinations can be performed on the accessible areas on both of these longitudinal welds. The remaining two circumferential welds are accessible for volumetric examinations.

IV. ALTERNATE PROVISIONS

A volumetric examination of the accessible circumferential and longitudinal welds will be performed to the extent and frequency described in IWC-2500. A remote visual examination of the I.D. of the pump casing welds will be performed only if the pump is disassembled for maintenance.