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Distribution w/encl:

Central File	D. Ross
LWR #3 File	D. Vassallo
A. Dromerick	D. Swanson, OELD
O. Parr	
M. Rushbrook	

MAY 0 2 1979

NOTE TO: John Guibert, Technical Assistant to Commissioner Kennedy
Hugh Thompson, Technical Assistant to Commissioner Bradford
George Eysymontt, Technical Assistant to Commissioner Gilinsky
George Sauter, Technical Assistant to Commissioner Ahearne

FROM: D. F. Ross, Jr., Deputy Director, Division of Project Management

We have received a request from the applicant regarding an exemption to certain requirements related to Appendices G, H & J of 10 CFR 50. Enclosed is a draft SER on these exemptions for North Anna 2. We intend to issue a supplement which includes these exemptions at the time an operating license is issued for North Anna Unit 2.

The exemptions requested for North Anna Unit 2 related to Appendices G, H & J of 10 CFR 50 are similar to those granted for the McGuire Nuclear Station.

It is anticipated that Unit 2 will be ready for fuel loading by June 1979. However, our review of North Anna Unit 2 is not complete, and it will be some time before it is completed. Nonetheless, we would appreciate an expeditious review of these exemptions and your advising me by telephone (extension 27373) of any comments you may have in this regard so that work can continue on the SER supplement.

Original Signed By
Roger S. Boyd

D. F. Ross, Jr., Deputy Director
Division of Project Management
Office of Nuclear Reactor Regulation

Enclosure:
As Stated

POOR ORIGINAL

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OFFICE	LWR #3:LPM	LWR #3:BC	LWR:AD	DPM:DD		
SURNAME	ADromerick/LM	ODParr	DBVassallo	DFRoss		
DATE	4/ /79	4/ /79	4/ /79	4/ /79		

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VIRGINIA ELECTRIC AND POWER COMPANY
NORTH ANNA UNIT NO. 2 (OL)
DOCKET NUMBER 50-339
COMPLIANCE WITH APPENDICES G AND H, 10 CFR PART 50

MATERIALS ENGINEERING BRANCH
MATERIALS INTEGRITY SECTION

Compliance with Appendices G and H, 10 CFR Part 50

The reactor vessel for North Anna Unit No. 2 was manufactured by Rotterdam Dry Dock Company of Netherlands. The purchase order was issued on May 1, 1969. The vessel was fabricated to 1968 Winter Addenda of the ASME Section III Boiler and Pressure Vessel Code. Since the ASME Code editions defined in 10 CFR 50.55a preceded the publication of Appendices G and H, 10 CFR Part 50, some of the fracture toughness tests for the ferritic materials in the primary coolant pressure boundary were not conducted to demonstrate explicit compliance with the current requirements of Appendices G and H.

Virginia Electric and Power Company stated that the fracture toughness requirements of Appendices G and H, 10 CFR Part 50 were met for North Anna Unit No. 2 except for the specific requirements of Section IV.A.4 of Appendix G and Section II.C.2 of Appendix H.

Alternate methods for compliance with Appendices G and H, 10 CFR Part 50 were proposed by Virginia Electric and Power Company and exemptions were requested from the identified requirements. VEPCO also provided additional information in support of their methods of compliance with Appendix G.

We have concluded from our review of information submitted that exemptions to some of the specific requirements of Appendices G and H, 10 CFR Part 50 are required, and we have determined that the identified exemptions are justified. The bases for justification are discussed in the subsequent paragraphs of this report.

Evaluation of Compliance with Appendix G

Based on our review of the applicant's submittal for compliance with Appendix G, 10 CFR Part 50, we have determined that the requirements of Appendix G have been met for North Anna Power Station, Unit No. 2 except for Section IV.A.4.

Section IV.A.4 of Appendix G requires that a Charpy V-notch test program be conducted for the primary coolant pressure boundary ferritic bolting exceeding one inch in diameter to demonstrate that the bolting material exhibits the minimum requirements of 25 mils lateral expansion and

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45 ft.-lbs. impact energy at the lower of either the preload temperature or the lowest service temperature. The North Anna Unit No. 2 reactor vessel bolting material tests were performed in accordance with the ASME Code, 1968 Edition, including the Winter 1969 Addendum of Section III. These codes have no lateral expansion measurement requirements and specify that an average impact energy of 30 ft.-lbs. be obtained at a temperature 60°F lower than either the hydrotest or the lowest service temperature.

The impact test results for North Anna Unit No. 2 indicate that the material impact energy exceeded all the ASME Code requirements at a test temperature of 10°F. The test results further show that at 10°F, which is more conservative than that required by Appendix G, the average impact energy is approximately 42 ft.-lbs. To provide assurance that the bolting material fracture toughness complies with the requirements of Appendix G, additional impact testing was conducted at 32°F, 50°F, and 68°F. The respective average impact energies at these three test temperatures was equal to or greater than 45 ft.-lbs.

We have reviewed the test data obtained to qualify A 540 Grade B24 bolting material used at North Anna Power Station Unit No. 2. The test data consisted of Charpy V-notch energy values obtained at 10°F on 30 test specimens, representing two heats of steel. These heats had average Charpy impact energies of 41.0 and 43.0 ft.-lbs. at 10°F. Some tests also were conducted at 32°F, 50°F, and 68°F. The average Charpy V-notch impact values were 44.7, 46.7 and 49.0 ft.-lbs. respectively.

We have also reviewed similar tests results for A 540 Grade B24 bolting material reported for several heats of steel in Electric Power Research Institute Report, EPRI NP-121, Volume II, Part One, April 1976. These data were reviewed to provide additional assurance that Charpy specimens having a minimum impact energy of 45 ft.-lbs. also had a minimum lateral expansion of 25 mils. Our review of these data indicated that specimens with impact energies greater than 45 ft.-lbs. did have lateral expansions greater than 25 mils.

Based on our evaluation of the test data, we conclude that an exemption for the area of noncompliance of Appendix G is justified. Our conclusion is based on the following:

Appendix G requires the measurement of both lateral expansion and absorbed energy to provide additional assurance that the material has adequate fracture toughness. However, absorbed impact energy and lateral expansion are very closely related criteria and provide an almost identical indication of the material quality and the toughness level. Consequently, we have determined that the measurement of the absorbed energy, in accordance with the ASME Boiler and Pressure Vessel Code requirements, is sufficient to demonstrate acceptable fracture toughness properties. Added assurance of our conclusion is supported by our review of additional test data obtained from an EPRI research program conducted for similar bolting material. As indicated previously, some of the impact specimens tested at the ASME Code 10°F test temperature had absorbed energies less than the 45 ft.-lbs. required by Appendix G. However, the 10°F test temperature specified by the ASME Code is more conservative than the test temperature required by Appendix G. To provide additional assurance that the bolting material has adequate fracture toughness some tests were conducted at higher temperatures representative of the Appendix G requirements. The results from these tests indicate that the fracture toughness for the bolting material meet the Appendix G requirement of 45 ft.-lbs. impact energy at the lower of either the preload temperature or lowest service temperature. The impact tests performed according to the ASME Code requirement and the additional tests performed at higher temperatures are sufficient to indicate that the bolting materials were manufactured properly, are of acceptable quality and have adequate fracture toughness to provide reasonable assurance that adequate safety margins will be obtained and maintained during operation as required by Appendix G.

We have evaluated the data presented in the FSAR and based on the results of our evaluation we have determined that sufficient information has been provided to demonstrate that the safety margins required by Appendix G, 10 CFR Part 50 have been achieved.

Evaluation of Compliance with Appendix H

Based on our review of VEPCO's submittal for compliance with Appendix H, 10 CFR Part 50, we have determined that the requirements of Appendix H have been met for North Anna Unit No. 2, except for Section II.C.2.

Section II.C.2 of Appendix H was not complied with for Unit No. 2 to the extent that six of the eight specimen capsules are located in areas where the lead factor is less than one. The lead factor is the ratio of the neutron flux at the specimen capsule to the maximum neutron flux at the vessel inner surface. Consequently, the neutron flux received by the capsules will be less than that received by the inner surface of the reactor vessel. The purpose of the Appendix H limitation that the surveillance capsule lead factor be in the range from one to three is to ensure that reduction in reactor vessel material toughness resulting from neutron irradiation is monitored in advance of actual vessel conditions and to minimize calculational uncertainties in extrapolating the surveillance measurements from the specimens to the reactor vessel wall. As an alternative to maintaining the required lead factor at a fixed location VEPCO has suggested a schedule for rotation of the specimen capsules at different locations such that when a capsule is removed for testing it will have a lead factor greater than one.

We have reviewed the alternate method proposed by VEPCO to achieve the required lead factor and conclude that this alternative is equivalent to the Appendix H requirement and no inaccuracy will result from the capsule rotation. Adequate data are assured because the proposed capsule rotation will provide the required total accumulated neutron fluence without significant changes in temperature or neutron fluence rate relative to the vessel inner surface during the total time of irradiation.

Based on our review and evaluation we conclude that an exemption from Section II.C.2 is justified because the alternate method proposed by VEPCO is equivalent to the Appendix H requirement and will provide adequate data to monitor the reduction in material fracture toughness and ensure that adequate safety margins are maintained during operation.

Our technical evaluation has not identified any practical method by which the existing North Anna Unit No. 2 reactor vessel can comply with the specific requirements of Section IV.A.4 of Appendix G and Section II.C.2 of Appendix H, 10 CFR Part 50. Requiring compliance with the identified specific requirements would delay the startup of the units due to the need to complete the following actions: (1) retest the bolting materials to confirm compliance with Appendix G, and (2) relocate the installed material surveillance specimens.

Based on the foregoing, pursuant to 10 CFR Section 50.12, exemption to the specific requirements of Appendices G and H of 10 CFR Part 50 as discussed above is authorized by law and can be granted without endangering life or property or the common defense and security and is otherwise in the public interest. We conclude that the public is served by not imposing certain provisions of Appendices G and H of 10 CFR Part 50 that have been determined to be either impractical or would result in hardship or unusual difficulties without a compensating increase in the level of quality and safety.

Furthermore, we have determined that the granting of this exemption does not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. We have concluded that this exemption would be insignificant from the standpoint of environmental impact and pursuant to 10 CFR 51.5(d)(4) that an environmental impact statement, or negative declaration and environmental impact appraisal, need not be prepared in connection with this action.

Evaluation of Proposed Exemption
from Appendix J Requirements
North Anna Power Station, Unit 2

In the Technical Specifications for North Anna Unit 2 the applicant describes its proposed leak testing procedure for the containment airlocks, and proposes an exemption from the associated requirements of Appendix J to 10 CFR Part 50. Based on our review, we find the proposed leak testing procedures and the proposed exemption to Appendix J acceptable. The rationale for our finding acceptable the applicant's proposed leak testing practices for the personnel airlocks and the proposed exemption from the associated requirements of Appendix J to 10 CFR 50, is discussed below.

Appendix J to 10 CFR 50 requires the containment personnel airlocks to be leak tested at six-month intervals and after each opening during such intervals (III.D.2). Appendix J further requires that the test be conducted at the peak calculated containment pressure related to the design basis accident; i.e., Pa, (III.B.2).

Considering that a full pressure airlock test is to be performed every six months, it is our judgment that testing airlocks within three days after each opening or after the initial opening in a series of openings, at Pa, will adequately demonstrate the continuing integrity of the airlock door seals such that the public health and safety will be

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ensured. The effect on accident consequences of testing after each opening versus testing within three days of an opening is judged to be insignificant. Furthermore, if an airlock door seal is damaged, it will be manifested during testing at Pa.

This is an adequate demonstration of continuing airlock integrity for the period between the six-month tests.

We find that leak testing an airlock in the manner described above is an acceptable alternative to the requirements of Appendix J. Accordingly, the proposed exemption from the requirements of Appendix J is acceptable.

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Response to IE Bulletin 79-06A
North Anna Power Station Unit No. 1

Actions taken or planned in response to each item of IE Bulletin 79-06A are as follows. Number sequence is the same as in the bulletin.

1a. A detailed review of this event, by the appropriate personnel has been completed. The station training group conducted this review with Station Supervision and the Station Nuclear Safety and Operating Committee.

1b. Operational personnel were instructed in the specific concerns of item 1b in a briefing held on April 21, 1979.

1c. Station Supervision and Operations personnel have received a briefing on the Three Mile Island incident. This briefing was conducted by NRC personnel on April 21, 1979. Those individuals needing this briefing who were not in attendance will receive this information as soon as possible.

2a. and b.
The primary operator action required to prevent the formation of voids is to insure the proper initiation and continuing performance of the engineered safety features. Present procedures require this verification. Procedure changes to prevent premature or inappropriate shutdown of engineered safety features will be made as explained in our response to items 7a and b. A procedure change to insure forced flow by reactor coolant pumps will be made as explained in our response to item 7c. These procedure changes will be completed by May 4, 1979.

2c. Procedural changes to provide additional guidance on enhancing core cooling in the event of void formation are under review at this time.

3. North Anna Unit No. 1 uses pressurizer water level coincident with pressurizer pressure for automatic initiation of safety injection. The low pressurizer level bistables for all three channels will be tripped, such that low pressurizer pressure only will initiate safety injection.

During the performance of pressurizer pressure channel functional surveillance tests, all three level channel bistables will be returned to normal.

In the event that one pressurizer pressure channel becomes inoperable, its associated level channel bistable will be returned to normal. This will provide a 1 of 2 low pressurizer pressure safety injection from the 2 remaining operable channels.

A standing order has been issued requiring operators to manually initiate safety injection when the pressurizer pressure indication reaches the actuation setpoint whether or not the level indication has dropped to the actuation setpoint.

4. We have completed our review of containment isolation initiation design and procedures and have determined that no changes are required.

5. North Anna Unit No. 1 has automatic auxiliary feedwater initiation. Per Technicial Specification 3/4.7.1.2, limiting conditions for operation and surveillance requirements have been established to maintain operability of the system.

   Automatic initiation results from:

   1. SI,
   2. Low-Low S/G level,
   3. Loss of Main Feedwater Pumps,
   4. Loss of Offsite power

6a. We have reviewed the applicable procedures and have determined that no changes are needed for this item.

6b. This item has been covered by a Standing Order. The appropriate procedures will be revised by May 4, 1979.

7a. and b.
The applicable procedures will be revised by May 4, 1979, to prohibit overriding engineered safety features, unless continued operation of engineered safety features would result in unsafe conditions. Specifically, emergency procedures will specify that if the high pressure injection system has been automatically actuated because of a low pressure conditions, it must remain in operation until either:

1) Both low pressure injection pumps are in operation and flowing for 20 minutes or longer, at a rate which would assure stable plant behavior; or

2) The high pressure injection system has been in operation for 20 minutes and all hot and cold leg temperatures are at least 50 degrees below the saturation temperature for the existing RCS pressure. If 50 degrees subcooling cannot be maintained after high pressure injection cutoff, the high pressure injection should be reactivated. The degree of subcooling beyond 50 degrees F and the length of time high pressure injection is in operation shall be limited by the pressure/temperature considerations for reactor vessel integrity. Shutdown of the high pressure injection system prior to 20 minutes is permitted only when overpressurization of the reactor coolant system is eminent and provided the above listed margins to saturation temperature are maintained.

7c. Operating procedures will be revised by May 4, 1979 to specify that in the event of high pressure injection initiation with reactor coolant pumps (RCPs) operating, at least two RCPs shall remain operating as long as the pumps are providing forced flow.

7d. A Standing Order has been issued which precautions against overreliance on pressurizer level indications and recommends examination of other plant parameters in assessing water inventory and plant conditions. Operator training will incorporate a review of this concern.

8. Periodic tests will be revised to address this concern. Additionally, administrative procedures for shift turnover already address this item in that known unit conditions are reviewed. Maintenance Operating Procedures already address this concern.

9a. Two interlocks exist to prevent the transfer of radioactive gases when high radiation indication exists.

1) Containment purge and exhaust is secured on high containment activity
2) Containment vacuum pump operation is terminated on high activity in the Process Vent System.

9b. Transfer of potentially radioactive gases and liquids is prevented by the initiation of Phase A containment isolation.

9c. Technical Specification 4.6.3.1.2.a provides for periodic testing of Phase A containment isolation. Procedures will be revised to insure that sufficient liquid waste tank capacity is available prior to pumping the containment sump. Operating procedures will be revised to incorporate precautions in containment sump pump operations.

10a. The concern is addressed by the Shift Supervisor in reviewing the "Action Statement Status" log prior to releasing a piece of redundant equipment for testing or preventive maintenance. For corrective maintenance on engineered safety features equipment, the redundant equipment will be tested before removing from service the equipment needing maintenance. However, in cases where testing of the redundant equipment makes that equipment inoperable, it will not be tested.

10b. Maintenance Operating Procedures already cover this item.

10c. With one exception, all Periodic Tests concerning ESF equipment require Shift Supervisor notification prior to commencement of and following completion of the test. In the case of the exception, that test will be revised by May 4, 1979, to include the notification requirement.

Our maintenance reporting system involves Shift Supervisor review prior to maintenance and following completion of maintenance.

11. Existing notification procedures will be revised to specify that the NRC be notified within one hour of the time the reactor is not in a controlled or expected condition of operation. The procedure will include provisions for establishing and maintaining a continuous open channel of communication with the NRC.

12. The existing equipment for removal of hydrogen from containment consists of two identical portable skid mounted hydrogen recombiners, two hydrogen analyzers, two purge blowers and associated piping systems. Operating procedures presently exist to strip hydrogen from the primary coolant. Additional operating modes and procedures for dealing with significant amounts of hydrogen gas are under review at this time.

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VIRGINIA ELECTRIC AND POWER COMPANY
RICHMOND, VIRGINIA 23261

May 3, 1979

Mr. James P. O'Reilly, Director
Office of Inspection and Enforcement
U. S. Nuclear Regulatory Commission
Region II
101 Marietta Street, Suite 3100
Atlanta, Georgia 30303

Serial No.	274A
PO/DLB:baw	
Docket Nos.:	50-338
	50-339
License Nos.:	NPF-4
	CPPR-78

Subject: IE Bulletin 79-06A
North Anna Unit No. 2

Dear Mr. O'Reilly:

Attachment 2 to our letter of April 26, 1979 identified our actions taken or planned on North Anna Unit No. 1 in response to IE Bulletin 79-06A. This is to inform you that all commitments made for North Anna Unit No. 1 in response to the subject bulletin will also apply to North Anna Unit No. 2 and will be implemented on Unit No. 2 prior to the issuance of the operating license.

Very truly yours,

C. M. Stallings
Vice President-Power Supply
and Production Operations