Virginia Electric and Power Company
Richmond, Virginia 23261

January 12, 2001

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

Serial No. 01-013
NL&OS/ETS
Docket No. 50-339
License No. NPF-7

Gentlemen:

VIRGINIA ELECTRIC AND POWER COMPANY
NORTH ANNA POWER STATION UNIT 2
ASME SECTION XI INSERVICE INSPECTION PROGRAM
RELIEF REQUEST NDE-47

North Anna Power Station Unit 2 is presently in the second ten-year inservice inspection interval and examinations are conducted in accordance with the requirements of 1986 Edition of ASME Section XI. In accordance with ASME Section XI, the pressurizer surge line nozzle-to-vessel weld and nozzle inner radius welds require examination. These examinations have been determined to be a hardship without a compensating increase in safety based on the radiation dose required to perform the examinations.

Therefore, pursuant to 10 CFR 50.55a (a)(3)(ii), relief is requested from certain ASME Section XI Code examination requirements associated with the surge line nozzle-to-vessel weld and nozzle inner radius welds. Relief request NDE-47 is attached and provides the bases for the relief request.

A similar relief was granted for North Anna Power Station Unit 1 for the second and third intervals by NRC letters dated April 4, 1992 and April 25, 2000, respectively. Similar reliefs were also granted for Byron Station by NRC letter dated December 30, 1998, Beaver Valley by NRC letter dated October 8, 1997, and Surry Power Station Units 1 and 2 by letters dated July 19, 1995 and August 30, 1995, respectively.

This relief request has been approved by the Station Nuclear Safety and Operating Committee.

In order to support the upcoming Unit 2 refueling outage, we request that the relief request be approved by February 25, 2001. Due to the limited time available for NRC staff review and the significant outage impact, we also request a meeting with the staff to discuss the proposed relief request at your earliest convenience.

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If you have questions concerning these requests, please contact us.

Very truly yours,

Leslie N. Hartz
Vice President - Nuclear Engineering and Services

Attachment

Commitments made in this letter: None

cc: U. S. Nuclear Regulatory Commission
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Attachment

Second Ten Year Interval
Relief Request NDE-47

Virginia Electric and Power Company
(Dominion)
North Anna Power Station Unit 2

RELIEF REQUEST NDE-47

I. IDENTIFICATION OF COMPONENTS:

Pressurizer (2-RC-E-2) nozzle-to-vessel weld 9 and nozzle inner radius section 9NIR (Figures NDE-47-1 and 2). These welds are shown on drawing 12050-WMKS-RC-E-2.

II. IMPRACTICAL CODE REQUIREMENTS:

Section XI of the ASME Boiler and Pressure Vessel Code, 1986 Edition, requires in Category B-D (Item Nos. B3.110 and B3.120) that the pressurizer surge line nozzle-to-vessel weld and nozzle inside radius section be volumetrically examined.

III. BASIS FOR RELIEF:

Access to the North Anna Unit 2 pressurizer surge line nozzle is obstructed by multi-layered, stainless steel mirror insulation and the cables for the pressurizer heaters. Removal of the insulation and cables would be difficult as well as labor and time intensive. It is also likely that cable or heater pin damage would occur during removal. In addition it is possible that the impingement shield would have to be removed to gain access to the examination area.

It is almost certain that some, and possibly all, heater cables would have to be disconnected so that the cables can be pulled back to allow access for removing insulation and doing the exam. The exact scope of work to gain access cannot be fully determined until the unit is shutdown for the next refueling outage. Dose rates are predicted using a step approach to build the total projected exposure. There are four options possible. The worst case option assumes that all 78 heater cables have to be disconnected and pulled back. These cables have brazed connections that will be time consuming to remove and replace following the exam. This option carries a dose estimate of 56 rem. If the outer ring of heaters can be left intact during the examination (disconnect/reconnect 46 heaters), then the dose estimate is 35.2 rem. If only the first ring of heaters has to be dealt with (20 heaters), then the dose estimate is 18.3 rem. For the highly unlikely scenario of not having to disconnect any heater cables, the dose estimate is 5.3 rem. Separately, if the impingement shield has to be removed, then an additional 5.8 rem must be added to all these totals. It should be noted that the amount of heater cable work expected is likely to have a significant impact on overall outage manning requirements to accommodate the anticipated high dose.

Other personnel safety concerns potentially involved in this examination include the increased risk for an unplanned exposure event and prevention of contamination with personnel wedged between the surge line and the exposed portion of the pressurizer heaters. While actions would be taken to prevent any such events, the large dose rate gradients in the under-pressurizer area would challenge even the protection afforded by the best available technology. Temporary shielding is considered impractical in this regard because placement of the shielding material would obstruct and potentially preclude accessibility to the examination surface. Other issues include actual accessibility after removal of all interferences and the likelihood of difficulties in replacing the insulation to its original configuration. Furthermore, the amount of examination coverage would be dependent on the overall accessibility obtained.

In conjunction with license renewal, Westinghouse has performed an evaluation to address the impact of operational transients for North Anna Unit 2, to account for insurge/outsurge transients in addition to design transients in the pressurizer lower head. The results of the evaluation show that the Cumulative Usage Factor (CUF), after service equivalent to 60 years of operation for the lower head to nozzle weld, is 0.32 for the inside surface and 0.07 for the outside surface. The CUFs for the nozzle inner radius are 0.17 (inside surface) and 0.09 (outside surface). These CUFs are considerably less than the design limit of 1.0 and are lower in magnitude than other locations on the pressurizer that are currently being inspected. For instance, the spray nozzle to safe-end weld has a CUF of 0.848 and the 6" safety and relief nozzle inside radius welds have a CUF of 0.148.

There are several uncertainties regarding an alternative examination of the inside surface of the pressurizer surge line area. An inspection may be able to be performed in which a boroscope could be fed through the manway and down through the middle of the pressurizer. Adding to the difficulty in performing such an exam, there is a screen device on the outlet of the surge line inner radius to control in-surge to the pressurizer. The boroscope would be positioned through the support plates, and then threaded through a screen inlet orifice, if possible, to the pressurizer surge line area. This examination could be partially obscured by the thermal sleeve. Furthermore, the resulting examination would only be of the cladding that covers the inside radius of the nozzle, which is considered to be only marginally beneficial in determining the structural integrity of the nozzle. Additionally, performing the visual inspection requires opening the reactor coolant system and establishing access and foreign material exclusion controls. The boroscope itself has the potential to become lodged inside the inlet screen device or behind a pressurizer heater support plate. This inspection effort and the significant potential risk associated with it are not commensurate with the limited benefit that may be obtained by the inspection itself.

As such, we are also applying for relief per 10 CFR 50.55a(a)(3)(ii) due to the fact that compliance with the specified requirements would result in hardship or unusual difficulties without a compensating increase in the level of quality and safety. A similar relief for North Anna Power Station Unit 1 was granted for use during the second interval by NRC Letter No. 92-255 dated 4/7/92, and during the third interval by NRC Letter No. 00-240 dated 4/25/00. Similar relief was also granted for Surry Power Station Unit 1, Letter No. 95-404, dated 7/19/95, Surry Power Station Unit 2 Letter No. 95-480, dated 8/30/95, Byron Station dated 12/30/98, and Beaver Valley dated 10/8/97.

IV. ALTERNATE REQUIREMENTS:

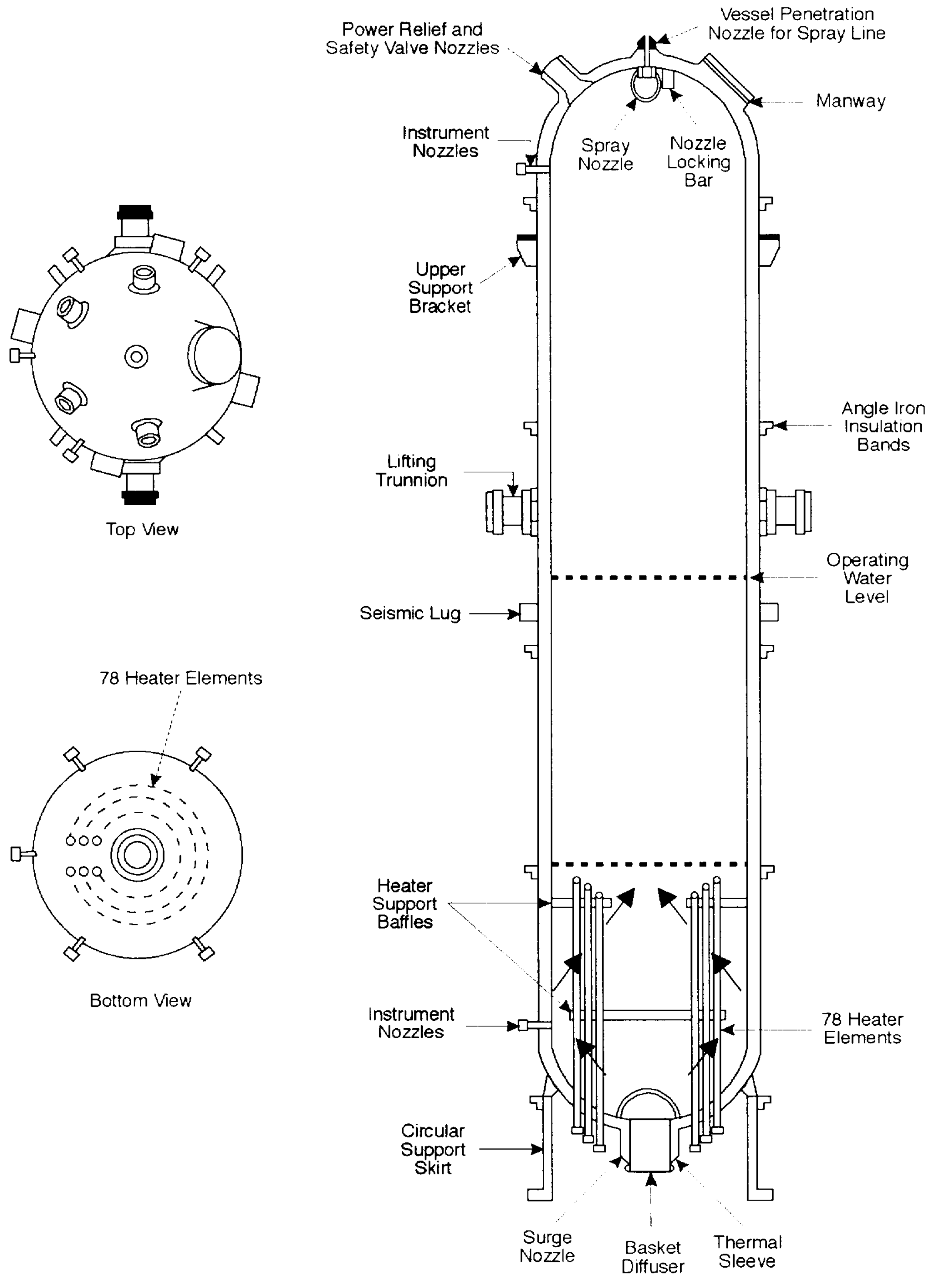
A visual (VT-2) examination of the pressurizer surge line nozzle-to-vessel weld will be performed during the normally scheduled system leakage test each refueling and will provide continued assurance of component integrity. In addition:

1. Technical Specifications require that the Reactor Coolant System Leak Rate be limited to one gallon per minute unidentified leakage. This value is calculated at least once per 72 hours; and

2. The containment atmosphere particulate radioactivity is checked every 12 hours.

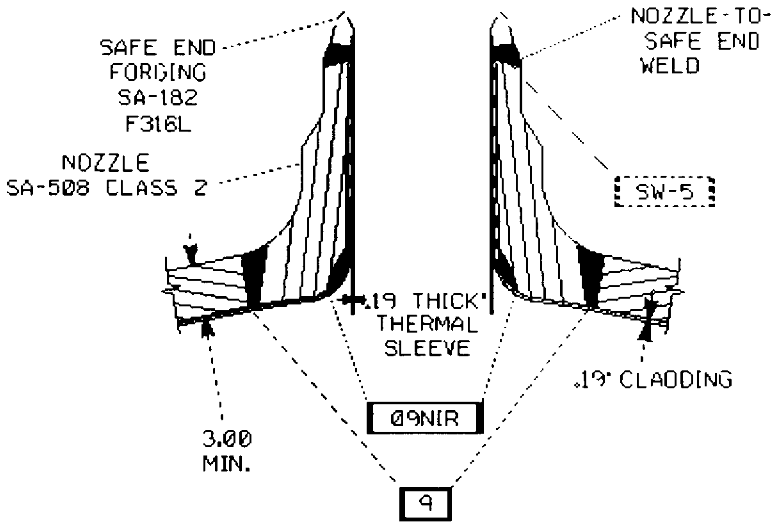
The proposed alternatives stated above will ensure that the overall level of plant quality and safety will not be compromised.

FIGURE NDE-47-1



PRESSURIZER

FIGURE NDE-47-2



SURGE NOZZLE

PACKAGE DIVIDER

DRP-003