VIRGINIA ELECTRIC AND POWER COMPANY
RICHMOND, VIRGINIA 23261

August 31, 2001

U.S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, D.C. 20555

Serial No.	01-490
NL&OS/GDM	R3'
Docket Nos.	50-338/339
	50-280/281
License Nos.	NPF-4/7
	DPR-32/37

Gentlemen:

VIRGINIA ELECTRIC AND POWER COMPANY
NORTH ANNA POWER STATION UNITS 1 AND 2
SURRY POWER STATION UNITS 1 AND 2
RESPONSE TO NRC BULLETIN 2001-01 CIRCUMFERENTIAL CRACKING OF REACTOR VESSEL HEAD PENETRATION NOZZLES

On August 3, 2001 the NRC issued NRC Bulletin 2001-01, "Circumferential Cracking of Reactor Pressure Vessel Head Penetration Nozzles," requesting information regarding the structural integrity of the reactor pressure vessel head penetration (VHP) nozzles including the extent of VHP nozzle leakage and cracking that has been found to date, the inspections and repairs that have been completed to satisfy applicable regulatory requirements, and the basis for concluding that plans for future inspection will ensure compliance with the applicable regulatory requirements.

Both North Anna Unit 1 and Surry Unit 1 have VHP inspections scheduled during the upcoming Fall 2001 refueling outages. The attachment to this letter provides the requested information for North Anna and Surry Power Stations.

If you have any further questions or require additional information, please contact us.

Very truly yours,

Leslie N. Hartz
Vice President – Nuclear Engineering

Attachment

A088

Commitments made in this letter:

1. It is our intention to perform an effective visual inspection (VT-2) of the reactor vessel heads under the insulation for North Anna Unit 1 and Surry Unit 1 to inspect for signs of leakage around each of the control rod drive housings and the reactor head vent where they penetrate the head during the Fall 2001 refueling outages.

2. It is our intention to perform additional inspections from under the head of North Anna Unit 1 (or Surry Unit 1 if qualification is delayed) with an eddy current procedure capable of detecting small surface connected flaws on the inner diameter (ID) of the housings, on the outer diameter (OD) of the housings below the inside surface of the head, and on the J-groove attachment welds. The inspections are contingent upon the availability and acceptable performance of the necessary equipment and personnel to accomplish the inspections.

3. The NRC will be contacted prior to the evaluation or repair of any identified circumferential flaws.

4. It is our intention to perform effective visual (VT-2) inspections of the reactor vessel heads under the insulation for North Anna Unit 2 and Surry Unit 2 during their respective refueling outages (i.e., Spring 2002 for Surry Unit 2 and Fall 2002 for North Anna Unit 2).

5. If any of the visual inspections discover evidence of leakage at the junction of the CRDM housings or head vent and the vessel head, it is our intention to perform supplemental inspections from under the vessel head using eddy current and ultrasonic inspection procedures, as appropriate, to locate the source of the leakage and to characterize any flaws that are found. In addition, it is our intention to perform eddy current and/or ultrasonic inspections, as appropriate, of an additional number of housings based on statistical determination of a relevant sample size. The inspections are contingent upon the availability and acceptable performance of the necessary equipment and personnel to accomplish the inspections.

6. Any axial indications discovered and sized by the combination of eddy current and ultrasonic inspection will be evaluated in accordance with requirements consistent with ASME Section XI and as delineated in the ASME paper entitled, "Inspection and Evaluation of the Reactor Vessel Head Penetrations at D. C. Cook Unit 2," by W. H. Bamford, et al., 1994.

7. We anticipate that the statistical analysis for determining appropriate scope and schedule for future inspection activities for North Anna Unit 2 and Surry Unit 2 will be completed and communicated to the NRC by mid-November of this year, along with the inspection results from the under the head inspections of North Anna Unit 1 or Surry Unit 1.

cc: U.S. Nuclear Regulatory Commission
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COMMONWEALTH OF VIRGINIA)
)
COUNTY OF HENRICO)

The foregoing document was acknowledged before me, in and for the County and Commonwealth aforesaid, today by Leslie N. Hartz, who is Vice President - Nuclear Engineering, of Virginia Electric and Power Company. She has affirmed before me that she is duly authorized to execute and file the foregoing document in behalf of that Company, and that the statements in the document are true to the best of her knowledge and belief.

Acknowledged before me this 31st day of August, 2001.

My Commission Expires: 3-31-04

Maggie McClure
Notary Public

(SEAL)

ATTACHMENT

Response to NRC Bulletin 2001-01
Circumferential Cracking of Reactor Pressure Vessel Head Penetration Nozzles

North Anna Power Station Units 1 and 2
Surry Power Station Units 1 and 2

Virginia Electric and Power Company
(Dominion)

Response to NRC Bulletin 2001-01
Circumferential Cracking of Reactor Pressure Vessel Head Penetration Nozzles
North Anna and Surry Power Stations Units 1 and 2

North Anna and Surry Power Stations Units 1 and 2 have been categorized with susceptibility rankings within 5 Effective Full Power Years (EFPY) of Oconee Nuclear Station Unit 3 (ONS3). Therefore, the following information is provided for North Anna and Surry Units 1 and 2 in response to the requested information specified in the NRC Bulletin:

NRC requested information

1. *All addressees are requested to provide the following information:*

   a. *the plant-specific susceptibility ranking for your plant(s) (including all data used to determine each ranking) using the PWSCC susceptibility model described in Appendix B to the MRP-44, Part 2, report;*

Response:

North Anna Units 1 and 2 and Surry Units 1 and 2 have been evaluated for susceptibility to PWSCC relative to Oconee 3 using the aforementioned Materials Reliability Program (MRP) time-at-temperature PWSCC susceptibility model. The key parameters are listed for each station in the attached Table entitled, "Key Parameters Utilized in MRP Ranking and Other NRC Requested Information."

This evaluation showed that North Anna Unit 1 will take 2.3 EFPYs of additional operation from March 1, 2001, to reach the same time-at-temperature as ONS3 at the time that leaking nozzles were discovered in March 2001. The evaluation further determined that North Anna Unit 2 will take 3.4 EFPYs, Surry Unit 1 will take 3.4 EFPYs, and Surry Unit 2 will take 3.5 EFPY to reach the same time-at-temperature.

Therefore, North Anna Units 1 and 2 and Surry Units 1 and 2 are in the same NRC category of plants within 5 EFPYs of ONS3.

   b. *a description of the VHP nozzles in your plant(s) including the number, type, inside and outside diameter, material of construction, and the minimum distance between VHP nozzles;*

Response:

North Anna Units 1 and 2 and Surry Units 1 and 2 each have 65 VHP Alloy 600 Control Rod Drive Mechanism (CRDM) nozzles plus one Alloy 600 head vent nozzle. Each CRDM nozzle and head vent nozzle was attached to the head by an Alloy 182 J-groove weld. The head arrangement and other requested nozzle details are provided in the table and Figure A-2 attached. It should be noted that the design interference fit for the vessel head penetrations is 0.0004 to 0.0012 inches which is less than the interference fit for the Oconee units which have exhibited visible leakage on the heads.

c. *a description of the RPV head insulation type and configuration;*

Response:

The reactor heads at North Anna Units 1 and 2 and at Surry Units 1 and 2 are insulated with stepped reflective stainless steel insulation as depicted in Figure 1.

d. *a description of the VHP nozzle and RPV head inspections (type, scope, qualification requirements, and acceptance criteria) that have been performed at your plant(s) in the past 4 years, and the finding. Include a description of any limitations (insulation or other impediments) to accessibility of the bare metal of the RPV head for visual examinations;*

Response:

In the last four years, visual inspections have been performed on each of the four units to address concerns raised by Generic Letter 88-05, "Boric Acid Corrosion of Carbon Steel Reactor Pressure Boundary Components in PWR Plants." The most recent visual inspections were North Anna Unit 2 in the spring of 2001, Surry Unit 2 in the fall of 2000, and North Anna Unit 1 and Surry Unit 1 in the spring of 2000. The inspections are performed by VT-2 qualified personnel each refueling outage with the vessels depressurized. The inspections place particular emphasis on evidence of boric acid accumulation and are conducted with the insulation on the head. No evidence of leakage has been detected. In addition, Westinghouse performed a best effort under head nondestructive examination (NDE) inspection at North Anna Unit 1 in February 1996, examining the two outermost rows of CRDMs. The inner diameter (ID) of 20 of 65 CRDM penetration tubes was characterized by eddy current (EC). No reportable indications were found; however, the thermal sleeves in some penetrations interfered with the EC blade probe, thus limiting the extent of the exam in those cases. The EC technique was only qualified to characterize axial ID cracks.

e. *A description of the configuration of the missile shield, the CRDM housing and their support/restraint system, and all components, structures, and cabling from the top of the RPV head up to the missile shield. Include the elevations of these items relative to the bottom of the missile shield.*

Response:

The service structure (called the reactor vessel lifting rig) bolts directly to the upper head of the reactor vessel for North Anna Units 1 and 2 and Surry Units 1 and 2. Forced air channeled through ductwork penetrating the lower portion of the service structure at 120 degree intervals provide cooling to the CRDMs. The lower part of the service structure is also provided with ledges to support the stepped RPV head insulation. A work platform on top of the service structure provides access to the upper CRDM housings. Also on top of the service structure is the CRDM seismic support platform. In the small gap between the missile shield and the top of the CRDM housings are electrical trays for CRDM power and instrumentation cabling. The missile shield for North Anna Units 1 and 2 is a single 24 inch thick slab of concrete faced with steel plate, where as the missile shield for Surry Units 1 and 2 is composed of three adjoining 24 inch thick concrete slabs, each faced with steel plate. (See attached drawings: 11715-FM-1E and 11715-FM-56A-2 for North Anna Unit 1, 12050-FM-1E and 12050-FM-56A-2 for North Anna Unit 2, 11448-FM-1E and 11448-FM-43A for Surry Unit 1 and 11548-FM-1E and 11548-FM-43A for Surry Unit 2.) The first of the pair of drawings is an overall view of the RV and missile shield in relation to the entire containment structure, and the second is a detailed view of the RV head-lifting rig (service structure) with regard to the missile shield.

Elevations (at ambient temperature and mean sea level) are as follows:

1. RV flange:
   a. North Anna Units 1 and 2: 262 feet 10 inches
   b. Surry Units 1 and 2: 18 feet 4 inches

2. Top of CRDM housing:
   a. North Anna Units 1 and 2: 290 feet 2 inches
   b. Surry Units 1 and 2: 45 feet 1 inch

3. Bottom of missile shield
   a. North Anna Units 1 and 2: 291 feet 8 5/8 inches
   b. Surry Units 1 and 2: 49 feet 8 7/8 inches

As can be seen, the gap between the upper tip of the CRDM housing and the bottom of the missile shield at ambient temperature is less than two feet for North Anna Units 1 and 2 and less than five feet for Surry Units 1 and 2. The gap at operating temperature is reduced due to thermal expansion of the CRDMs.

2. *Specific information is requested for plants that have previously experienced either leakage from or cracking in VHP nozzles.*

Response:
North Anna and Surry Units 1 and 2 have not previously experienced VHP nozzle cracking or leakage.

3. *If the susceptibility ranking for your plant is within 5 EFPY of ONS3, addressees are requested to provide the following information:*

   a. *your plans for future inspections (type, scope, qualification requirements, and acceptance criteria) and the schedule;*

Response:

North Anna Unit 1 is scheduled for refueling in mid-September to mid-October of 2001. Surry Unit 1 is scheduled for refueling from mid-October to mid-November of 2001. It is our intention to perform an effective visual inspection (VT-2) of the reactor vessel heads under the insulation for both of these units to inspect for signs of leakage around each of the control rod drive housings and the reactor head vent where they penetrate the head during the Fall 2001 refueling outages. The inspections will be performed using remote robotic equipment that has been demonstrated capable of detecting the small amounts of boric acid residue accumulation that would be indicative of through-wall cracking in the CRDM housings or head vent such as was seen at Oconee and Arkansas Nuclear One (ANO). In addition, it is our intention to perform additional inspections from under the head of North Anna Unit 1 with an eddy current procedure capable of detecting small surface connected flaws on the ID of the housings, on the outer diameter (OD) of the housings below the inside surface of the head, and on the J-groove attachment welds. Surface breaking indications discovered by EC will be further investigated using an ultrasonic inspection technique capable of sizing the indications contingent upon qualification of a suitable inspection technique. These procedures are currently being developed. It is our intention to demonstrate the capabilities of the procedures prior to the North Anna Unit 1 outage on specimens with stress corrosion cracks (grown in a doped steam environment), on an EPRI sample intended to simulate a J-groove weld with a stress corrosion crack, and possibly on specimens removed from Oconee housings which contain actual PWSCC depending upon their availability. It is also our intention to invite participation of the NRC and other interested parties in these demonstrations. It is highly unlikely that we will be able to perform operator proficiency demonstrations that would satisfy Appendix VIII requirements in the time available because of the lack of appropriate specimens; however, it is anticipated that the same operators involved in the capability demonstration will perform the inspections during the outage. If it is not possible to develop a qualified method by the North Anna Unit 1 refueling outage, it is our intention to qualify the procedure and perform the under the head eddy current and ultrasonic examinations during the Surry Unit 1 refueling outage instead.

Any axial indications discovered and sized by the combination of eddy current and ultrasonic inspection will be evaluated in accordance with requirements consistent with ASME Section XI and as delineated in an ASME paper entitled, "Inspection and Evaluation of the Reactor Vessel Head Penetrations at D. C. Cook Unit 2," by W. H. Bamford, et al., 1994. NRC acceptance of the requirements provided in the ASME paper was noted in a letter dated March 9, 1994, from Mr. Allen G. Hansen of the NRC to Mr. Robert E. Link of the Wisconsin Electric Power Company, Docket No. 50-266. While it would be technically possible to evaluate ID initiated circumferential flaws and OD circumferential flaws initiated below the J-groove weld with the same criteria, evaluations of circumferential indications will only be undertaken after consultation with the NRC.

Surry Unit 2 is scheduled for refueling in the Spring of 2002. North Anna Unit 2 is scheduled for refueling in the Fall of 2002. At this time, it is our intention to perform effective visual inspections (VT-2) of the reactor vessel heads under the insulation for these units during their respective refueling outages using the same remote robotic technology to be employed during the Fall 2001 outages. In conjunction with Westinghouse, we intend to develop a statistical basis for determining appropriate scope and schedule for future inspection activities for North Anna Unit 2 and Surry Unit 2. The evaluation will be based on the inspection experience to date for Alloy 600 head penetrations and will include the results obtained this fall for North Anna Unit 1 and Surry Unit 1. The first goal of the work will be to calculate the number of flaws of a specified limiting size which could be left in the head without repair for a specific time period with a 95% confidence level of acceptable crack size. Then, given the inspection results from the upcoming outage, the number of flaws to be expected in the head of each of the uninspected units could be calculated with a 95% confidence level. These results would form the basis for future inspection decisions on North Anna Unit 2 and Surry Unit 2.

We anticipate that the statistical analysis will be completed and communicated to the NRC by mid-November of this year, and that favorable inspection results from the under the head inspections of North Anna Unit 1 or Surry Unit 1 will further substantiate the acceptability of performing the inspections of the North Anna Unit 2 and Surry Unit 2 reactor vessel heads during their next scheduled refueling outages. This is based on the similarity in design, material, manufacture, and operating conditions of the Surry and North Anna reactor vessel heads. Admittedly, there is the potential that inspection results from the two scheduled fall refueling outages would indicate the necessity of an accelerated schedule for the inspection of North Anna Unit 2 and Surry Unit 2, and we are preparing for that contingency.

It should be recognized that the specialized tools to perform these inspections are currently under development within the industry at this time. Furthermore, personnel must be trained and qualified to perform the inspections. Consequently, we are planning the inspections as discussed above contingent upon and in anticipation of the availability and acceptable performance of the necessary equipment and personnel to accomplish the inspections.

b. *your basis for concluding that the inspections identified in 3.a will assure that regulatory requirements are met (see Applicable Regulatory Requirements section). Include the following specific information in this discussion:*

   (1) *If your future inspection plans do not include performing inspections before December 31, 2001, provide your basis for concluding that the regulatory requirements discussed in the Applicable Regulatory Requirements section will continue to be met until the inspections are performed.*

Response:
The Applicable Regulatory Requirements section of NRC Bulletin 2001-01 lists the following regulatory requirements and plant commitments as providing the basis for the bulletin assessment:

- Appendix A to 10 CFR Part 50, "General Design Criteria for Nuclear Power Plants"

  Criteria 14 – "Reactor Coolant Pressure Boundary"

  Criteria 31 – "Fracture Prevention of Reactor Coolant Pressure Boundary", and

  Criteria 32 – "Inspection of Reactor Coolant Pressure Boundary"

- Plant Technical Specifications
- 10 CFR 50.55a, Codes and Standards, which incorporates by reference Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components, of the ASME Boiler and Pressure Vessel Code"
- Appendix B of 10 CFR Part 50, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," Criteria V, IX, and XVI

The following discussion addresses each of these criteria and demonstrates how our inspection plans insure that the criteria will continue to be met until visual inspections have been performed on each unit.

<u>Design Requirements: 10 CFR § 50, Appendix A – General Design Criteria</u>

The Bulletin states:

> "The applicable GDC include GDC 14, GDC 31, and GDC 32. GDC 14 specifies that the reactor coolant pressure boundary (RCPB) have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture; the presence of cracked and leaking VHP nozzles is not consistent with this GDC. GDC 31 specifies that the probability of rapidly propagating fracture of the RCPB be minimized; the presence of cracked and leaking VHP nozzles is not consistent with this GDC. GDC 32 specifies that components which are part of the RCPB have the capability of being periodically inspected to assess their structural and leak tight integrity; inspection practices that do not permit reliable detection of VHP nozzle cracking are not consistent with this GDC."

The three referenced General Design Criteria (GDC) state the following:

- Criterion 14 – Reactor Coolant Pressure Boundary

  "The reactor coolant pressure boundary shall be designed, fabricated, erected and tested so as to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture."

- Criterion 31 – Fracture Prevention of Reactor Coolant Pressure Boundary

  "The reactor coolant pressure boundary shall be designed with sufficient margin to assure that when stressed under operating, maintenance, testing, and postulated accident conditions (1) the boundary behaves in a non-brittle manner, and (2) the probability of rapidly propagating fracture is minimized. The design shall reflect consideration of service temperatures and other conditions of the boundary material under operating, maintenance, testing and postulated accident conditions and the uncertainties in determining (1) material properties, (2) the effects of irradiation on material properties, (3) residual, steady state and transient thermal stresses, and (4) size of flaws."

- Criterion 32 – Inspection of Reactor Coolant Pressure Boundary

  "Components which are part of the reactor coolant pressure boundary shall be designed to permit (1) periodic inspection and testing of important areas and features to assess their structural and leak tight integrity, and (2) an appropriate material surveillance program for the reactor pressure vessel."

During the initial plant licensing of North Anna and Surry Power Stations Units 1 and 2, it was demonstrated that the design of the reactor coolant pressure boundary met the regulatory requirements in place at that time. The GDC included in Appendix A to 10 CFR Part 50 did not become effective until May 21, 1971. The Construction Permits for Surry Units 1 and 2 and North Anna Units 1 and 2 were issued prior to May 21, 1971; consequently, these units were not subject to GDC requirements. (Reference SECY-92-223 dated September 18, 1992.) However, the following information demonstrates compliance with the design criteria relative to the cracking of RPV top head nozzles:

- Pressurized water reactors licensed both before and after issuance of Appendix A to 10 CFR Part 50 (1971) complied with these criteria in part by: 1) selecting Alloy 600 or other austenitic materials with excellent corrosion resistance and extremely high fracture toughness, for reactor coolant pressure boundary materials, and 2) following ASME Codes and Standards and other applicable requirements for fabrication, erection, and testing of the pressure boundary parts. NRC reviews of operating license submittals subsequent to issuance of Appendix A included evaluating designs for compliance with the General Design Criteria. The standard review plans (SRPs) in effect at the time of licensing did not address the selection of Alloy 600. They only required that ASME code requirements be satisfied.

- Although stress corrosion cracking of primary coolant system penetrations was not originally anticipated during plant design, it has occurred in the RPV top head nozzles at some plants. The robustness of the design has been demonstrated by the small amounts of the leakage that has occurred and by the fact that none of the cracks in Alloy 600 reactor coolant pressure boundary materials has rapidly propagated or resulted in catastrophic failure or gross rupture. The suitability of the originally selected materials has been confirmed. Given the inherently high fracture toughness and flaw tolerance of the Alloy 600 material, there is in fact an extremely low probability of a rapidly propagating failure and gross rupture. It should be noted that earlier versions of the GDCs are in terms of extremely low probability of gross rupture or significant leakage throughout design life.

- The ASME requirement for the J-groove CRDM welds is for a visual examination of 25% of the penetrations for leakage during pressure testing. The component was designed for that inspection. That examination, which at least for the near future will be conducted on the bare vessel head, is capable of assessing the structural and leak tight integrity of the head penetrations. NDE and enhanced visual examination can be performed using specialized methods.

As described above, the requirements established for design, fracture toughness, and inspectability in GDC 14, 31, and 32 respectively were satisfied during each plant's initial licensing review, and continue to be satisfied during operation, even in the presence of a potential for stress corrosion cracking of the RPV top head penetrations. It should be noted that there is no existing plant specific evidence that any of the VHP nozzles at North Anna or Surry is cracked or leaking.

Operating Requirement: 10 C.F.R. § 50.36 - Plant Technical Specifications

The Bulletin states:

> "Plant technical specifications pertain to the issue of VHP nozzle cracking insofar as they require no through-wall reactor coolant system leakage."

Title 10 of the Code of Federal Regulations, Part 50.36 (10 CFR 50.36) contains requirements for Plant Technical Specifications. Paragraphs 2 and 3 of 10CFR Part 50.36 are particularly relevant:

- 10 CFR 50.36 (2) Limiting Conditions for Operation

  "Limiting conditions for operation are the lowest functional capability or performance levels of equipment required for safe operation of the facility. When a limiting condition for operation of a nuclear reactor is not met, the licensee shall shut down the reactor or follow any remedial action permitted by the technical specifications until the condition can be met. A technical specification limiting condition for operation of a nuclear reactor must be established for each item meeting one of the following criteria:

  Criterion 3: A structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

Criterion 4: A structure, system or component which operating experience or probabilistic risk assessment has shown to be significant to public health and safety."

- 10 CFR 50.36 (3) Surveillance Requirements

  "Surveillance requirements are requirements relating to test, calibration, or inspection to assure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the limiting conditions will be met."

The reactor coolant pressure boundary is one of the three physical barriers to the release of radioactivity to the environment. Therefore, our plant Technical Specifications (TS) include a requirement and associated action statements addressing reactor coolant pressure boundary leakage. The limits for reactor coolant pressure boundary leakage at North Anna and Surry are 1 gallon per minute for unidentified leakage, 10 gpm for identified leakage, and no leakage from a non-isolable fault in the reactor coolant system pressure boundary.

Leaks observed in other plants from Alloy 600 reactor vessel head penetrations due to PWSCC have been well below the sensitivity of on-line leakage detection systems. These plants have evaluated the condition and have determined that appropriate inspections are bare-metal visual inspections of the reactor head for boric acid deposits during plant shutdowns and/or NDE examination of the CRDMs. If leakage or unacceptable indications are found, then the defect must be repaired before the plant returns to power operations. Hypothetically, if a through-wall boundary leak of CRDMs increases to the point that the leakage is picked up by the on-line leak detection systems, then the leak must be evaluated per the specified TS acceptance criteria, and the plant shut down if the leak is determined to be a non-isolable reactor coolant system pressure boundary fault. Plant TS requirements continue to be met.

Inspection Requirements: 10 C.F.R. § 50.55a and ASME Section XI

The Bulletin states:

> "NRC regulations at 10 CFR 50.55a state that ASME Class 1 components (which include VHP nozzles) must meet the requirements of Section XI of the ASME Boiler and Pressure Vessel Code. Table IWA-2500-1 [IWB-2500-1[1]] of Section XI of the ASME Code provides examination requirements for VHP nozzles and references IWB-3522 for acceptance standards. IWB-3522.1(c) and (d) specify that conditions requiring correction include the detection of leakage from insulated components and

[1] An erratum appears to exist in the Bulletin. Table IWA-2500-1 is cited, but does not exist. It appears that the citation should have been IWB-2500-1.

discoloration or accumulated residues on the surfaces of components, insulation, or floor areas which may reveal evidence of borated water leakage, with leakage defined as 'the through-wall leakage that penetrates the pressure retaining membrane.' Therefore, 10 CFR 50.55a, through its reference to the ASME Code, does not permit through-wall cracking of VHP nozzles.

For through-wall leakage identified by visual examinations in accordance with the ASME Code, acceptance standards for the identified degradation are provided in IWB-3142. Specifically, supplemental examination (by surface or volumetric examination), corrective measures or repairs, analytical evaluation, and replacement provide methods for determining the acceptability of degraded components."

Title 10 of the Code of Federal Regulations, Part 50.55a requires that inservice inspection and testing be performed per the requirements of the ASME Boiler and Pressure Vessel Code, Section XI, "Inservice Inspection of Nuclear Plant Components." Section XI contains applicable rules for examination, evaluation and repair of code class components, including the reactor coolant pressure boundary.

Requirements for partial penetration welds attaching CRDM housings to the reactor vessel head are contained in Table IWB-2500-1, Examination Category B-E, "Pressure Retaining Partial Penetration Welds in Vessels," Item Numbers: B4.10, "Partial Penetration Welds;" B4.11, "Vessel Nozzles;" B4.12, "CRDM Nozzles;" and B4.13, "Instrumentation Nozzles." The Code requires a VT-2 visual examination of 25% of the CRDM nozzles from the external surface. Since the head is insulated, and the nozzles do not represent a bolted flange, paragraph IWA-5242(b) permits these inspections to be performed with the insulation left in place.

North Anna and Surry perform visual inspections for evidence of leakage by examining the RPV top head surface or the insulation pursuant to the requirements of NRC Generic Letter 88-05, "Boric Acid Corrosion of Carbon Steel Reactor Pressure Boundary Components in PWR Plants." North Anna Unit 1 conducted previous NDE examinations in response to NRC Generic Letter 97-01, "Degradation of Control Rod Drive Mechanisms Nozzle and Other Vessel Closure Head Penetrations," and found no indications of VHP cracking. For the next outage at each unit, North Anna and Surry will conduct these inspections under the insulation on the bare heads. We intend to conduct additional NDE examinations at North Anna Unit 1 (or at Surry Unit 1 if qualification is delayed) as discussed previously in this submittal.

Other plants have also conducted inspections beyond those required by ASME Section XI and NRC Generic Letter 88-05. These inspections have included visual examinations of the bare metal surfaces of the reactor head, eddy current and liquid penetrant surface examinations, and volumetric examinations of the nozzles. These supplemental inspections coupled with evaluations of the cracking found are considered to have provided a defense-in-depth approach for investigating and resolving this issue. As also discussed previously, additional work is underway for developing alternative inspection and analysis tools, both at Dominion and in conjunction with other industry initiatives.

The acceptance standard for the visual examination is found in paragraphs IWA-5250, "Corrective Measures" and IWB 3522, "Standards for Examination Category B-E, Pressure Retaining Partial Penetration Welds in Vessels, and Examination Category B-P, All Pressure Retaining Components." Paragraph IWA-5250 requires repair or replacement of the affected part if a through-wall leak is found and requires an assessment of damage, if any, associated with corrosion of steel components by boric acid. No plant has returned to service after finding a leak from a RPV top head nozzle without first having repaired the nozzle.

Flaws identified by NDE methods which are not addressed by specific ASME Section XI acceptance criteria are evaluated in accordance with the flaw evaluation rules for piping contained in Section XI of the ASME Code. This approach has been accepted by the NRC. Any flaw not meeting requirements for the intended service period would be repaired before returning it to service.

Repairs to RPV top head nozzles will be performed in accordance with Section XI requirements, NRC-approved ASME Code Case requirements, or an alternative repair or replacement method approved by the NRC.

North Anna and Surry comply with these ASME Code requirements through implementation of their inservice inspection programs. If a VT-2 examination detects the conditions described by IWB-3522.1(c) and (d), then corrective actions per IWB-3142 will be performed in accordance with the plant's corrective action program. No new plant actions are necessary to satisfy the cited regulatory criteria.

Quality Assurance Requirements: 10 C.F.R. § 50, Appendix B

The Bulletin states:

> "Criterion IX of Appendix B to 10 CFR Part 50 states that special processes, including nondestructive testing, shall be controlled and accomplished by qualified personnel using qualified procedures in accordance with applicable codes, standards, specifications, criteria, and other special requirements. Within the context of providing assurance of the structural integrity of VHP nozzles, special requirements for visual examination would generally require the use of a qualified visual examination method. Such a method is one that a plant-specific analysis has demonstrated will result in sufficient leakage to the RPV head surface for a through-wall crack in a VHP nozzle, and that the resultant leakage provides a detectable deposit on the RPV head. The analysis would have to consider, for example, the as-built configuration of the VHPs and the

capability to reliably detect and accurately characterize the source of the leakage, considering the presence of insulation, preexisting deposits on the RPV head, and other factors that could interfere with the detection of leakage. Similarly, special requirements for volumetric examination would generally require the use of a qualified volumetric examination method, for example, one that has a demonstrated capability to reliably detect cracking on the OD of the VHP nozzle above the J-groove weld."

As discussed previously in this submittal, the designed range of interference fit of the VHP nozzles in the North Anna and Surry vessel heads is very similar to, but slightly less than, the designed range of interference shrink fit of the Oconee units (0.4 to 1.2 mils versus 0.5 to 1.5 mils) indicating that through-wall cracking of the housings of the magnitude seen at Oconee should produce visually detectable evidence of leakage on top of the heads. While no specific analysis of the potential for detection of leakage has been done for North Anna or Surry, the discussion in Section 3 of "PWR Materials Reliability Program Interim Alloy 600 Safety Assessments for US PWR Plants (MRP-44) Part 2: Reactor Head Penetrations," indicates a leak path would exist and leakage would be detectable. The visual inspection technology that North Anna and Surry will employ uses a remote robotic video system for most of the housings and a boroscope with video camera for any housings that cannot be accessed by the robot. This type of video technology has been demonstrated to be effective at detecting small amounts of boric acid accumulation on the vessel head with sufficient resolution and sensitivity to distinguish between leakage occurring at VHP nozzles versus leakage from other sources. The inspections will be recorded on videotape. Personnel involved with the evaluation of the inspections will be VT-2 qualified and familiar with the anticipated type of indication that leakage would cause.

Additionally, qualification of the eddy current and ultrasonic inspection procedures we intend to use for the North Anna Unit 1 (or Surry Unit 1 if qualification is delayed) under head NDE will be demonstrated prior to use. Due to the generic application of this qualification, we intend to involve the NRC in the demonstration process.

The Bulletin further states:

> "Criterion V of Appendix B to 10 CFR Part 50 states that activities affecting quality shall be prescribed by documented instructions, procedures, or drawings, of a type appropriate to the circumstances and shall be accomplished in accordance with these instructions, procedures, or drawings. Criterion V further states that instructions, procedures, or drawings shall include appropriate quantitative or qualitative acceptance criteria for determining that important activities have been satisfactorily accomplished. Visual and volumetric examinations of VHP nozzles are activities that should be documented in accordance with these requirements."

Any of the work undertaken to inspect, evaluate, and/or repair the North Anna and Surry reactor vessel head penetrations will be conducted and documented in accordance with existing or new procedures which comply with the Company's Quality Assurance (QA) Topical Report, the QA program, and Criterion V of Appendix B to 10 CFR Part 50.

The last Appendix B criterion cited in the bulletin is:

> "Criterion XVI of Appendix B to 10 CFR Part 50 states that measures shall be established to assure that conditions adverse to quality are promptly identified and corrected. For significant conditions adverse to quality, the measures taken shall include root cause determination and corrective action to preclude repetition of the adverse conditions. For cracking of VHP nozzles, the root cause determination is important to understanding the nature of the degradation present and the required actions to mitigate future cracking. These actions could include proactive inspections and repair of degraded VHP nozzles."

Criterion XVI contains two important attributes pertinent to the potential for reactor vessel head penetration cracking.

The first of these is "...that measures shall be established to assure that conditions adverse to quality are promptly identified and corrected." This criterion infers a licensee's responsibility to be aware of industry experience, and has been interpreted in this manner in most plant's corrective action programs. A licensee should determine if industry experience applies to its plant and what, if any, corrective actions are appropriate. This approach is consistent with the NRC's generic communication process for an Information Notice, which reports industry experience, but does not require a response to the NRC. Licensees are expected to evaluate the applicability of the occurrence to their plant and document a record of the plant specific assessment for possible NRC review during inspections.

Criterion XVI provides the objectives and goals of the corrective action program, but licensees are responsible for determining a specific process to accomplish these goals and objectives. With regard to the bulletin response, Criterion XVI does not provide specific guidance as to what is an appropriate response, but rather, the licensee is responsible for determining actions necessary to maintain public health and safety. Specifically, in this case, the licensee must justify its actions for addressing the potential of stress corrosion cracking of vessel head penetrations. Furthermore, the regulatory criteria of 10 CFR 50.109(a)(7), provides supporting evidence when it states that "...if there are two or more ways to achieve compliance . . . then ordinarily the applicant or licensee is free to choose the way which best suits its purposes."

The second attribute of Criterion XVI that should be considered is that for "... significant conditions adverse to quality, the measures taken shall include root cause determination and corrective action to preclude repetition of the adverse conditions." The bulletin suggests that for cracking of vessel head penetrations, the root cause determination is important in understanding the nature of the degradation and the required actions to mitigate future cracking. As part of its corrective action program, a licensee, through its own efforts or as part of an industry effort, would determine the cause of cracks in the vessel head penetration, if they were detected. However, if no known cracks in the heads are identified through reasonable quality assurance measures or inspection and monitoring programs, this criterion would not require specific action on the part of a licensee for remaining in compliance with the regulation.

In summary, the integrated industry approach to inspection, monitoring, cause determination, and resolution of the identified CRDM nozzle cracking is clearly in compliance with the performance-based objectives of Appendix B.

(2) *If your future inspection plans include only visual inspections, discuss the corrective actions that will be taken, including alternative inspection methods (for example, volumetric examination) if leakage is detected.*

Response:

If any of the visual inspections discussed above discover evidence of leakage at the junction of the CRDM housings or head vent and the vessel head, it is our intention to perform supplemental inspections from under the vessel head using eddy current and ultrasonic inspection procedures, as appropriate, to locate the source of the leakage and to characterize any flaws that are found. In addition, it is our intention to perform eddy current and/or ultrasonic inspections, as appropriate, of an additional number of housings based on statistical determination of a relevant sample size. Any additional unacceptable indications would likely result in inspection of all of the housings on that reactor vessel head.

As discussed above, any axial indications discovered and sized by the combination of eddy current and ultrasonic inspection will be evaluated in accordance with requirements consistent with ASME Section XI and as delineated in the ASME paper entitled, "Inspection and Evaluation of the Reactor Vessel Head Penetrations at D. C. Cook Unit 2," by W. H. Bamford, et al., 1994. While it would be technically possible to evaluate ID initiated circumferential flaws and OD circumferential flaws initiated below the J-groove weld with the same criteria, evaluations of any circumferential flaw will only be undertaken after consultation with the NRC.

Repairs, if required, may include complete removal of the flaw and repair welding with Alloy 52 filler metal, partial removal of the flaw and welding with Alloy 52 (an embedded flaw technique and subject of a relief request currently being reviewed by the NRC), or other approaches as dictated by circumstances.

Information gathered from the Fall 2001 inspections of North Anna Unit 1 and Surry Unit 1, the inspections of North Anna Unit 2 and Surry Unit 2 in 2002 and future inspections will provide input to Dominion's determination of the appropriate inspection activities to implement in the future to provide early indication of VHP leakage should it occur. Factors such as inspection methodology, sample size, and schedule will be considered. In addition, the information gathered from the inspections will assist in assessing plans for future mitigation, repair, or replacement activities. As noted above, the specialized tools to perform these inspections are currently under development within the industry at this time. Furthermore, personnel must be trained and qualified to perform the inspections. Consequently, we are planning the inspections as discussed above contingent upon and in anticipation of the availability and acceptable performance of the necessary equipment and personnel to accomplish the inspections.

4. *Information is requested for plants with susceptibility rankings greater than 5 EFPY and less than 30 EFPY of ONS3.*

Response:

Since North Anna and Surry Units 1 and 2 have susceptibility rankings of less than 5 EFPY of ONS3, this item is not applicable.

5. *Addressees are requested to provide the following information within 30 days after plant restart following the next refueling outage:*

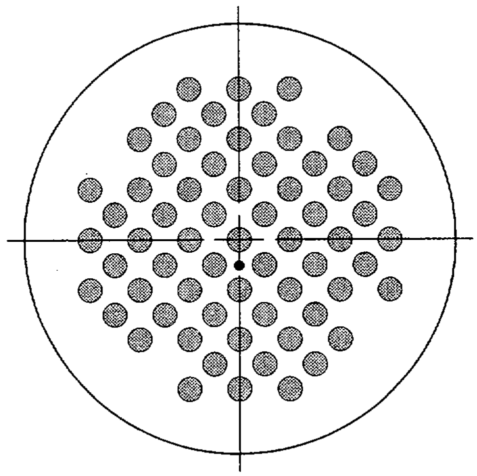
   a. *a description of the extent of VHP nozzle leakage and cracking detected at your plant, including the number, location, size, and nature of each crack detected;*

   b. *if cracking is identified, a description of the inspections (type, scope, qualification requirements, and acceptance criteria), repairs, and other corrective actions you have taken to satisfy applicable regulatory requirements. This information is requested only if there are any changes from prior information submitted in accordance with this bulletin.*

Response:

The requested information will be provided for North Anna and Surry Power Stations Units 1 and 2 should VHP nozzle cracking or leakage be identified during any of the upcoming refueling outage inspections.

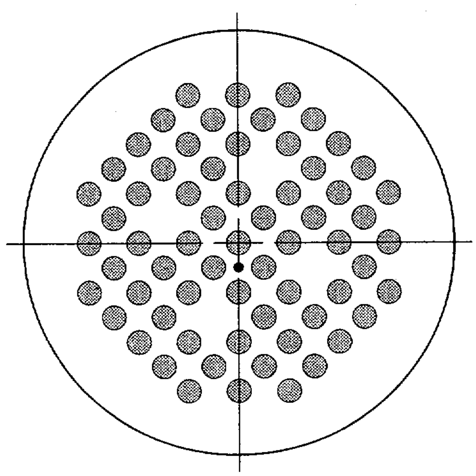
TABLE: Key Parameters Utilized In MRP Ranking and Other NRC Requested Information				
	North Anna Unit 1	North Anna Unit 2	Surry Unit 1	Surry Unit 2
Design and Fabrication				
NSSS Design	Westinghouse	Westinghouse	Westinghouse	Westinghouse
Nozzle Material Supplier	Sandvik	Sandvik	Huntington	B&W Tubular & Sandvik
Head Fabricator	Rotterdam	Rotterdam	Rotterdam/ B&W	Rotterdam/ B&W
Insulation Type/Configuration	Reflective/ Stepped	Reflective/ Stepped	Reflective/ Stepped	Reflective/ Stepped
J-Groove Type RVH Nozzle Information				
Head Map Configuration (see attached Figure)	Figure A-2b	Figure A-2b	Figure A-2a	Figure A-2a
CRDM Number	65	65	65	65
CRDM Nozzle Outside Diameter	4.000 in	4.000 in	4.000 in	4.000 in
CRDM Nozzle Inside Diameter	2.750 in	2.750 in	2.750 in	2.750 in
Min. Center to Center Distance Between CRDMs	11.97 in	11.97 in	11.97 in	11.97 in
Design Diametral Nozzle Interference Fit	0.4-1.2 mils (<Oconee)	0.4-1.2 mils (<Oconee)	0.4-1.2 mils (<Oconee)	0.4-1.2 mils (<Oconee)
Head Vent Number	1	1	1	1
Head Vent Outside Diameter (After Machining)	1.276 in	1.276 in	1.276 in	1.276 in
Head Vent Inside Diameter	0.815 in	0.815 in	0.815 in	0.815 in
Operating Time and Temperature				
MRP Ranking (out of 69 Domestic Plants)	5	8	9	10
Histogram Group Relative to Oconee where EFPYs= Effective Full Power Years	<3 EFPYs	3-6 EFPYs	3-6 EFPYs	3-6 EFPYs
Head Temperature/ Operating Time (Period #1)	600.1 F 2.9 EFPYs	600.1 F 2.0 EFPYs	597.8 F 4.6 EFPYs	597.8 F 3.8 EFPYs
Head Temperature/ Operating Time (Period #2)	607.1 F 6.9 EFPYs	607.1 F 4.7 EFPYs	599.8 F 10.0 EFPYs	597.8 F 10.6 EFPYs
Head Temperature/ Operating Time (Period #3)	600.1 F 7.3 EFPYs	600.1 F 10.0 EFPYs	597.8 F 4.9 EFPYs	597.8 F 5.0 EFPYs
Current Head Temperature/ Total Operating Time (Through February 2001)	600.1 F 17.1 EFPYs	600.1 F 16.7 EFPYs	597.8 F 19.5 EFPYs	597.8 F 19.4 EFPYs
Operating Time Normalized to 600 F	19.4 EFPYs	18.3 EFPYs	18.6 EFPYs	18.6 EFPYs
Remaining Time to Reach Oconee 3 from 3/1/01	2.3 EFPYs	3.4 EFPYs	3.4 EFPYs	3.5 EFPYs
Previous Inspection Status				
Type	Visual: GL 88-05	Visual: GL 88-05	Visual: GL 88-05	Visual: GL 88-05
	NDE: Nozzle ID	---	---	---
Date	Every RFO	Every RFO	Every RFO	Every RFO
	Feb-96	---	---	---
Extent	100%	100%	100%	100%
	31%	---	---	---
Result	No Leakage Detected	No Leakage Detected	No Leakage Detected	No Leakage Detected
	No Reportable Indications	---	---	---
Location of Examination	Top of Insulation	Top of Insulation	Top of Insulation	Top of Insulation
	Below Head	---	---	---
Next Scheduled Refueling Outage	Sept. 2001	Sept. 2002	Oct. 2001	Mar. 2002



65 CRDM Nozzles
1 Head Vent Nozzle

Figure a

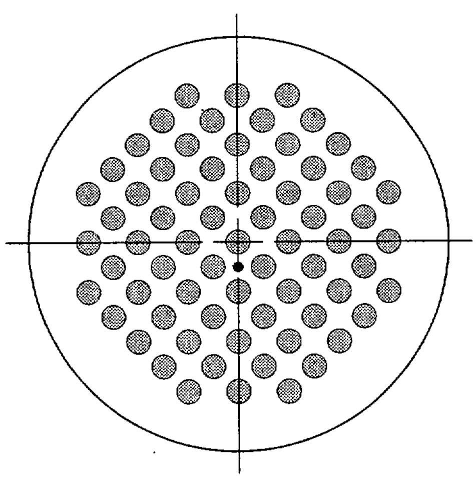
(Surry Units 1 & 2)



65 CRDM Nozzles
1 Head Vent Nozzle

Figure b

(North Anna Units 1 & 2)

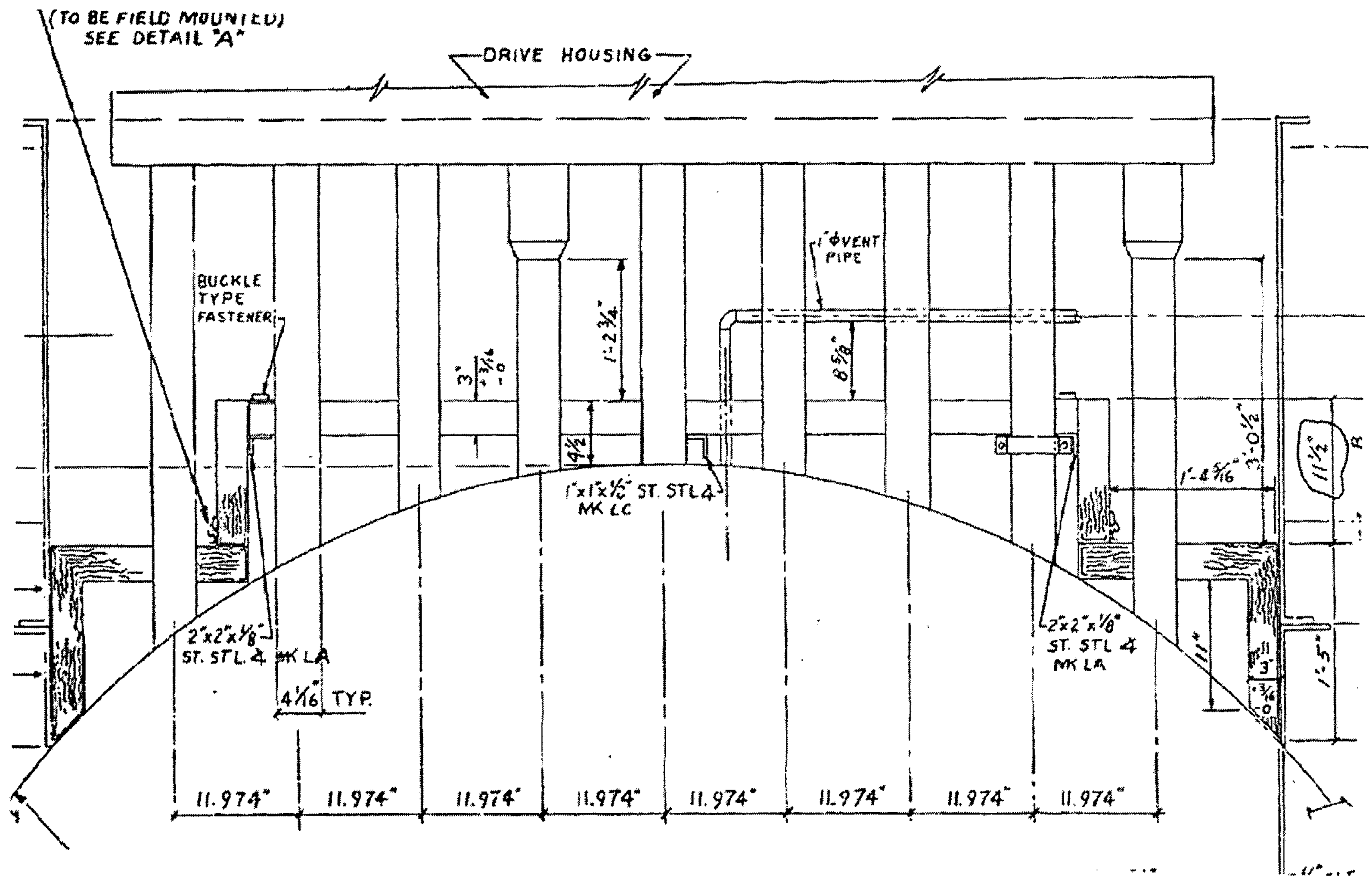


69 CRDM Nozzles
1 Head Vent Nozzle

Figure c

Figure A-2
Penetration Locations—Westinghouse 3-Loop Plants

Figure 1: Stepped Insulation Arrangement at Both North Anna Units 1 and 2 and Surry Units 1 and 2

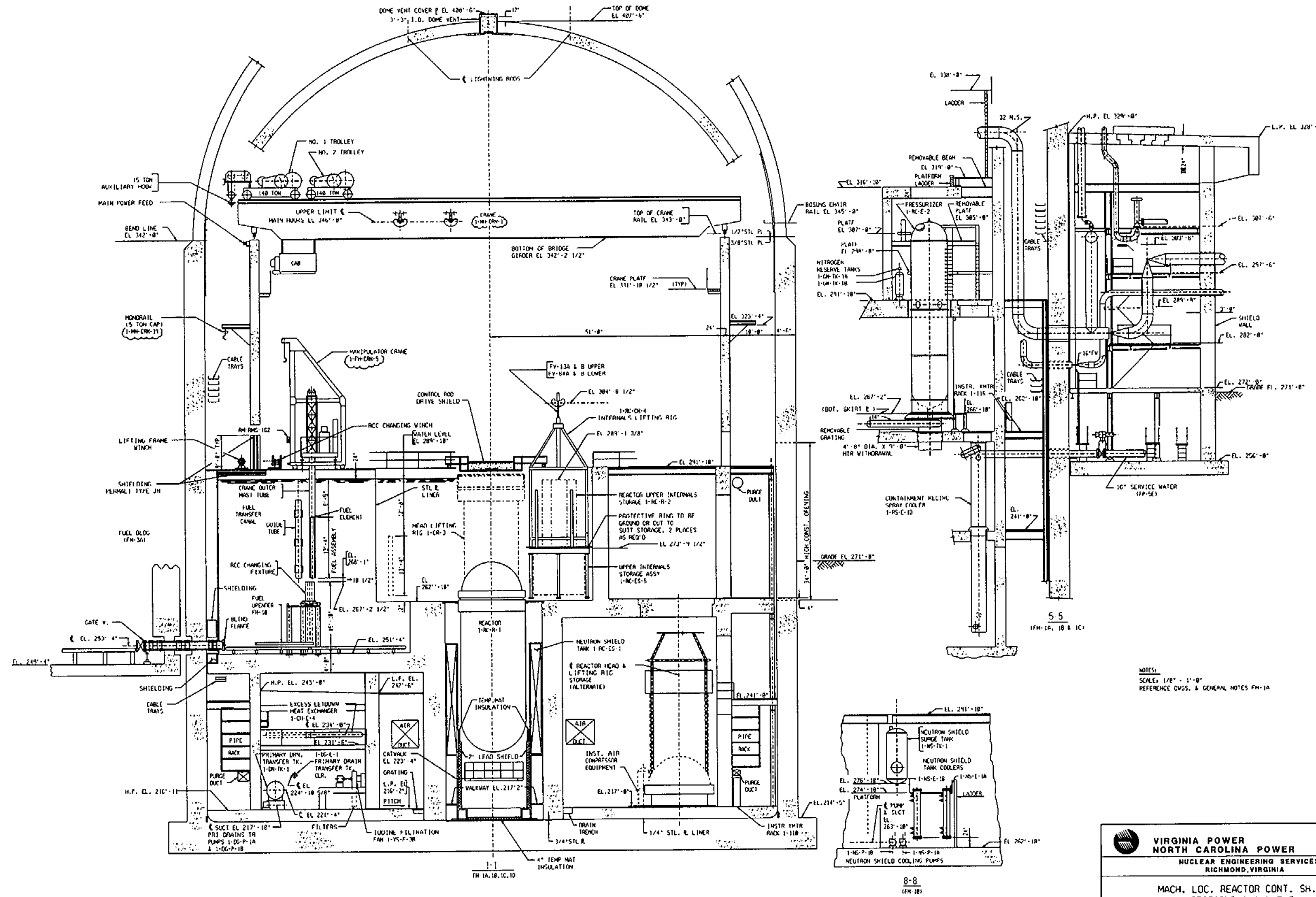


Drawings:

11715-FM-1E and 11715-FM-56A-2 for North Anna Unit 1
12050-FM-1E and 12050-FM-56A-2 for North Anna Unit 2
11448-FM-1E and 11448-FM-43A for Surry Unit 1
11548-FM-1E and 11548-FM-43A for Surry Unit 2

(Reference response to NRCB 2001-01 Item 1.e)

Virginia Electric and Power Company
(Dominion)



NOTES:
SCALE: 1/8" = 1'-0"
REFERENCE DWGS. & GENERAL NOTES FH-1A

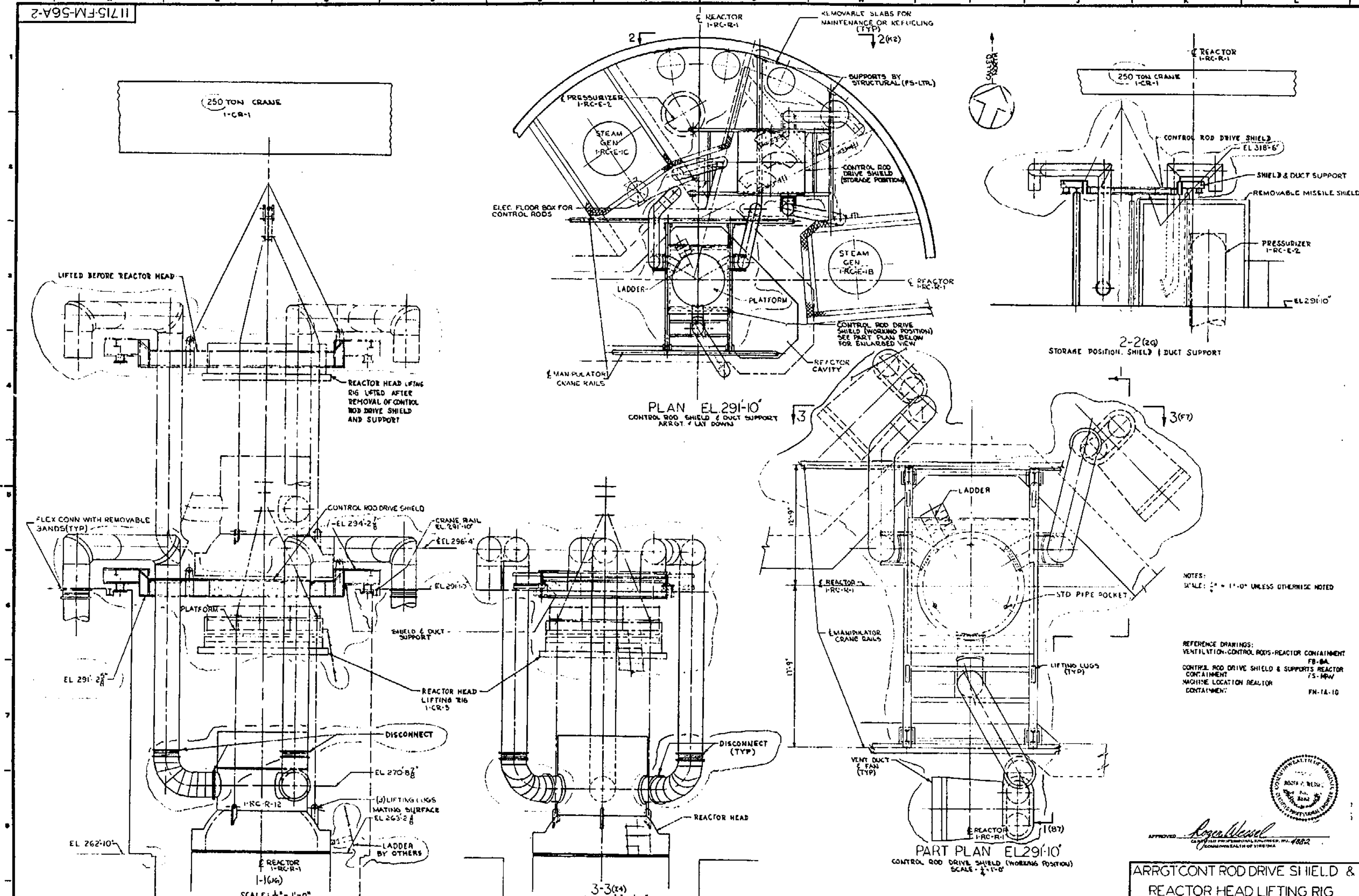
13	REVISED PER DUR 91-1079 THIS DRAWING SUPERSEDES THE REV. 12 ORIGINAL	
12	REDRAWN FROM 11715-FM-1E REV. 11 PER NP 1113	12-21-87

VIRGINIA POWER
NORTH CAROLINA POWER

NUCLEAR ENGINEERING SERVICES
RICHMOND, VIRGINIA

MACH. LOC. REACTOR CONT. SH. 5
SECTIONS 1-1 & 5-5
NORTH ANNA POWER STATION

DSGN	DSGN SUPV	CAD NO. /usr/dgn/1E1FM00Wdj1	
DRWN RH	ENGR SUPV	DRAWING NO. 11715-FM-1E	REV. 13
CHKD D. GODFREY			
DISPL ENGR JAR	SCALE AS NOTED	UNLESS OTHERWISE NOTED	SH 1 OF 1



NOTES:
SCALE: 1/4" = 1'-0" UNLESS OTHERWISE NOTED

REFERENCE DRAWINGS:
VENTILATION-CONTROL RODS-REACTOR CONTAINMENT FB-8A
CONTROL ROD DRIVE SHIELD & SUPPORTS REACTOR CONTAINMENT FS-14W
MACHINE LOCATION REACTOR CONTAINMENT FM-1A-10

APPROVED Roger Wessel

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ARRGT CONT ROD DRIVE SHIELD & REACTOR HEAD LIFTING RIG

NORTH ANNA POWER STATION
VIRGINIA ELECTRIC AND POWER COMPANY
STONE & WEBSTER ENGINEERING CORPORATION
BOSTON, MASS.
DRAWING NUMBER 11715-FM-56A-2

ISSUE	DESCRIPTION	CHKD	CORR	APPR	DATE
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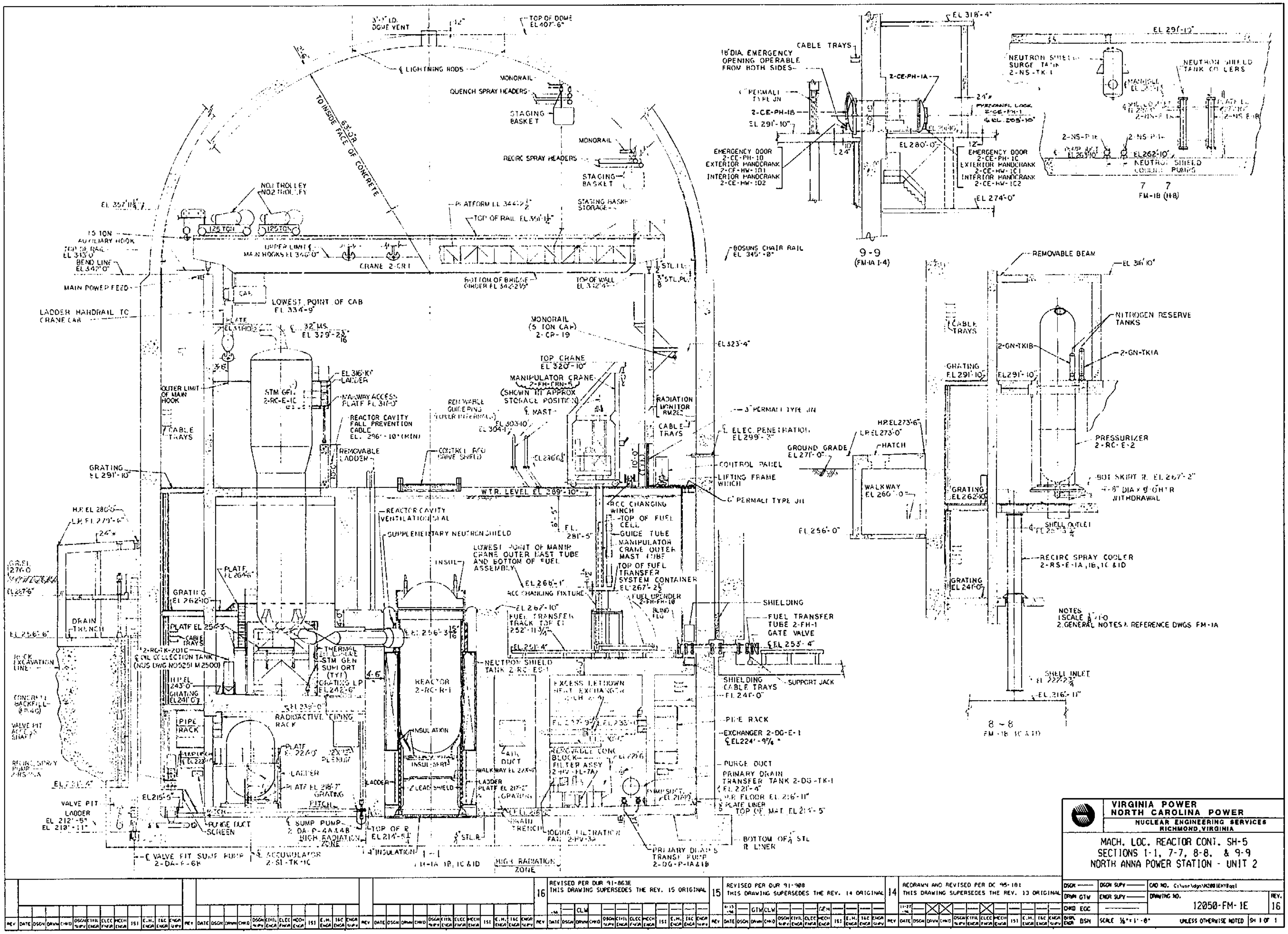
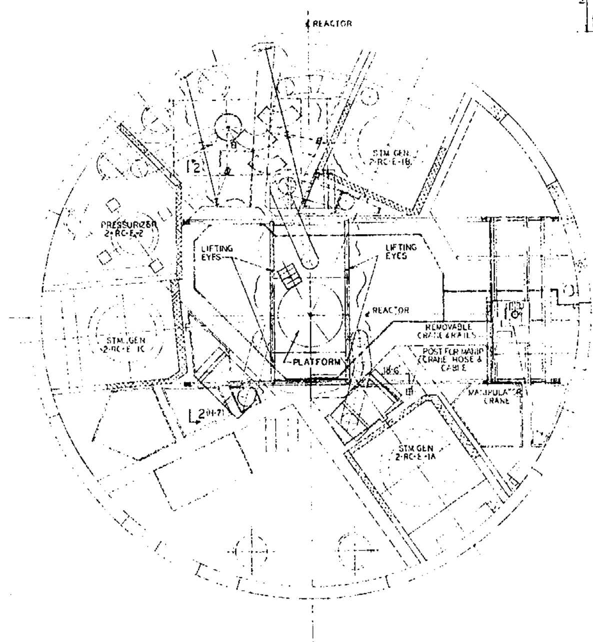
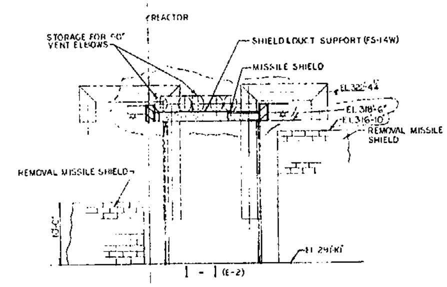
1	ORIGINAL ISSUE				

11715-FM-56A-2

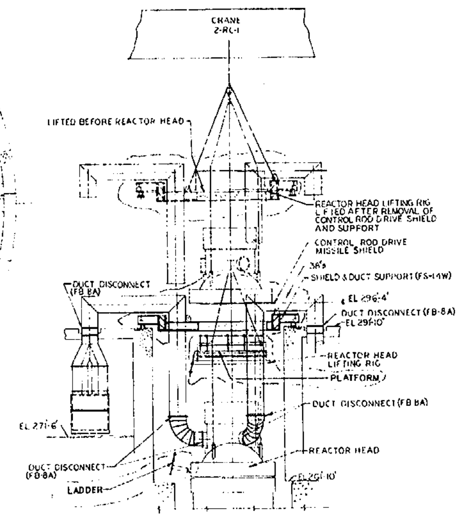

VIRGINIA POWER
NORTH CAROLINA POWER
NUCLEAR ENGINEERING SERVICES
RICHMOND, VIRGINIA
MACH. LOC. REACTOR CONT. SH-5
SECTIONS 1-1, 7-7, 8-8, & 9-9
NORTH ANNA POWER STATION - UNIT 2
12050-FM-1E
16
SCALE 1/4"=1'-0"
UNLESS OTHERWISE NOTED
16 REVISED PER DUR 91-863E THIS DRAWING SUPERSEDES THE REV. 15 ORIGINAL
15 REVISED PER DUR 91-988 THIS DRAWING SUPERSEDES THE REV. 14 ORIGINAL
14 REDRAWN AND REVISED PER DC 95-101 THIS DRAWING SUPERSEDES THE REV. 13 ORIGINAL
NOTES
1. SCALE 1/4"=1'-0
2. GENERAL NOTES & REFERENCE DWGS FM-1A
9-9 (FM-1A 1-4)
7 7 FM-1B (H8)
8-8
1-1
TOP OF DOME EL 407'-6"
LIGHTNING RODS
TO INSIDE FACE OF CONCRETE
MONORAIL
QUENCH SPRAY HEADERS
STAGING BASKET
RECIRC SPRAY HEADERS
NO1 TROLLEY
NO2 TROLLEY
CRANE 2-CR-1
LOWEST POINT OF CAB EL 334'-9"
MAIN POWER FEED
LADDER HANDRAIL TO CRANE CAB
STM GEN 2-RC-E-1C
REACTOR 2-RC-R-1
INSULATION
PRESSURIZER 2-RC-E-2
NITROGEN RESERVE TANKS
2-GN-TK1B
2-GN-TK1A
REMOVABLE BEAM
RECIRC SPRAY COOLER 2-RS-E-1A, 1B, 1C & 1D
SHELL INLET
SHELL OUTLET
WALKWAY EL 260'-0"
GROUND GRADE EL 271'-0"
HATCH
CABLE TRAYS
EMERGENCY DOOR 2-CE-PH-1D
EMERGENCY DOOR 2-CE-PH-1C
2-CE-PH-1A
2-CE-PH-1B
18" DIA. EMERGENCY OPENING OPERABLE FROM BOTH SIDES
NEUTRON SHIELD SURGE TANK 2-NS-TK-1
NEUTRON SHIELD TANK COOLERS
NEUTRON SHIELD COOLING PUMPS
WTR. LEVEL EL 289'-10"
RCC CHANGING WINCH
FUEL TRANSFER TUBE 2-FH-1
GATE VALVE
SUPPORT JACK
SHIELDING
PIPE RACK
PURGE DUCT
PRIMARY DRAIN TRANSFER TANK 2-DG-TK-1
PRIMARY DRAIN TRANSF PUMP 2-DG-P-1A & 1B
BOTTOM OF 1/4" STL LINER
SUMP PUMP 2-DA-P-4A & 4B
ACCUMULATOR 2-SI-TK-1C
VALVE PIT SUMP PUMP 2-DA-P-6A
HIGH RADIATION ZONE
NEUTRON SHIELD TANK 2-RC-ES-1
REACTOR CAVITY VENTILATION SEAL
SUPPLEMENTARY NEUTRON SHIELD
MANIPULATOR CRANE 2-FH-CRN-5
TOP CRANE EL 320'-10"
MONORAIL (5 TON CAP) 2-CP-19
RADIATION MONITOR
CONTROL PANEL
LIFTING FRAME WINCH
GRATING EL 291'-10"
DRAIN TRENCH



PLAN EL 291'-10"



1 - 1 (E-2)



2-2 (D-6)

NOTES:
1. SCALE ⅛" = 1'-0"

REFERENCE DWGS

VENTILATION-CONTROL RODS-REAC. CONT.	FB-8A
CONTROL ROD DRIVE SHIELD SHIELD & SUPPORTS REACTOR CONTAINMENT	FS-14W
MACHINE LOCATION REAC. CONT.	FM-1A-1F

APPROVED Roger P. Wessel 1982

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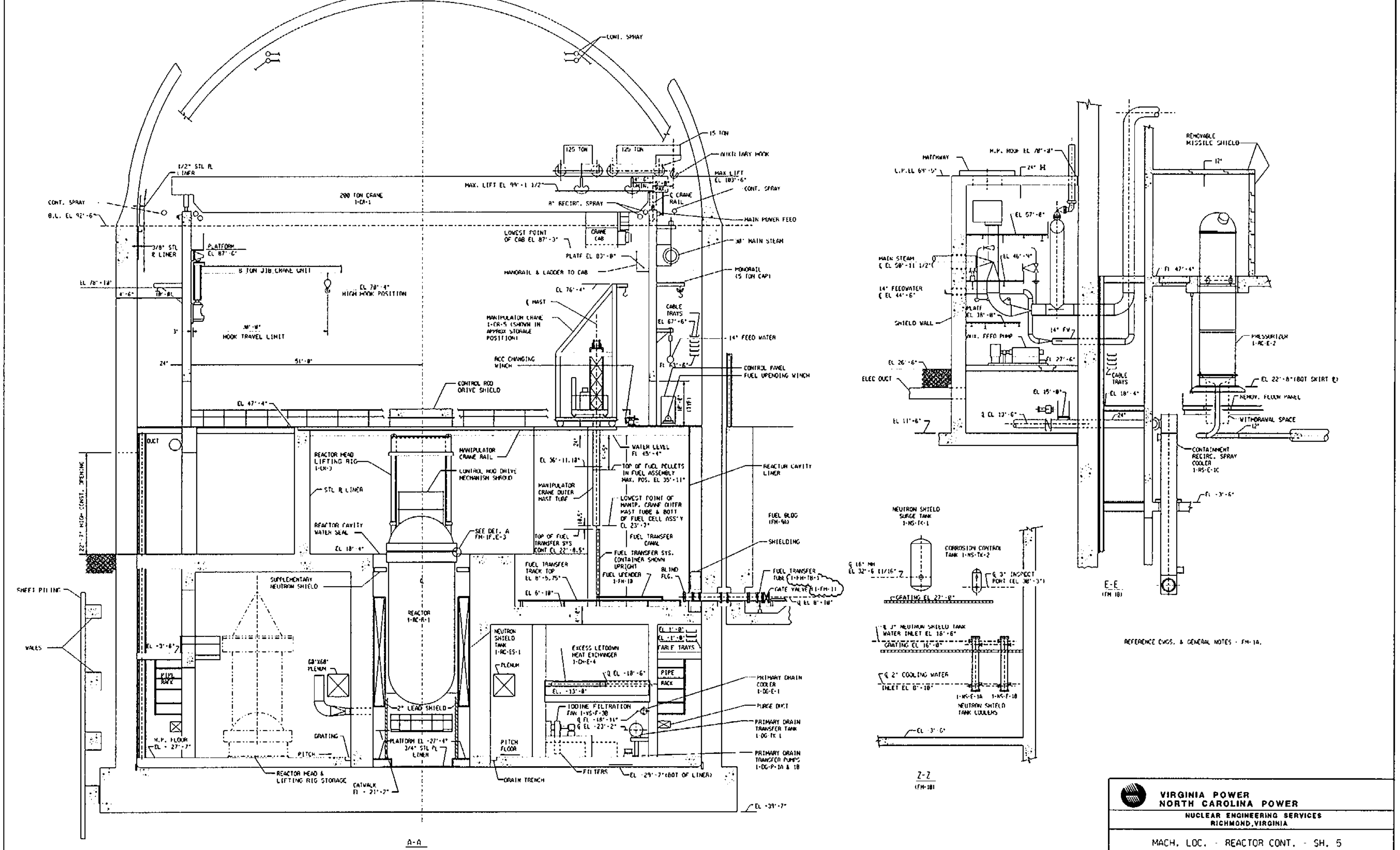
ISSUE	DESCRIPTION
2	REV'D CONTROL ROD DRIVE SHIELD (E-5, H-2, H-5) LADDER & PLATFORM ADDED (H-6)
1	ORIGINAL ISSUE

ARRANGEMENT CONTROL ROD DRIVE SHIELD & STORAGE AREA

UNIT 2 - NORTH ANNA POWER STATION
VIRGINIA ELECTRIC AND POWER COMPANY
STONE & WEBSTER ENGINEERING CORPORATION
BOSTON, MASS.
DRAWING NUMBER 12050-FM-56A 2

12050-FM-56A-2

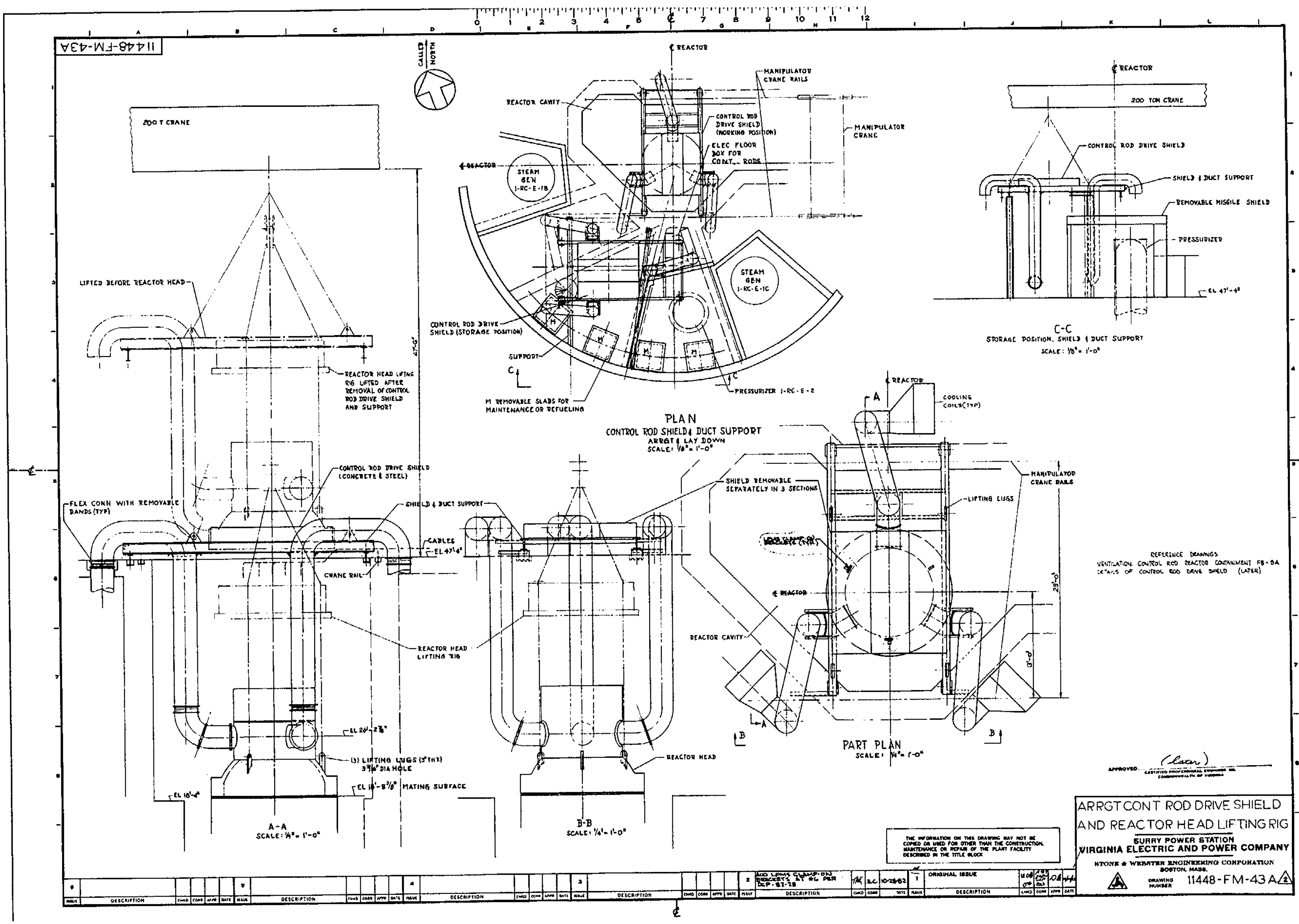


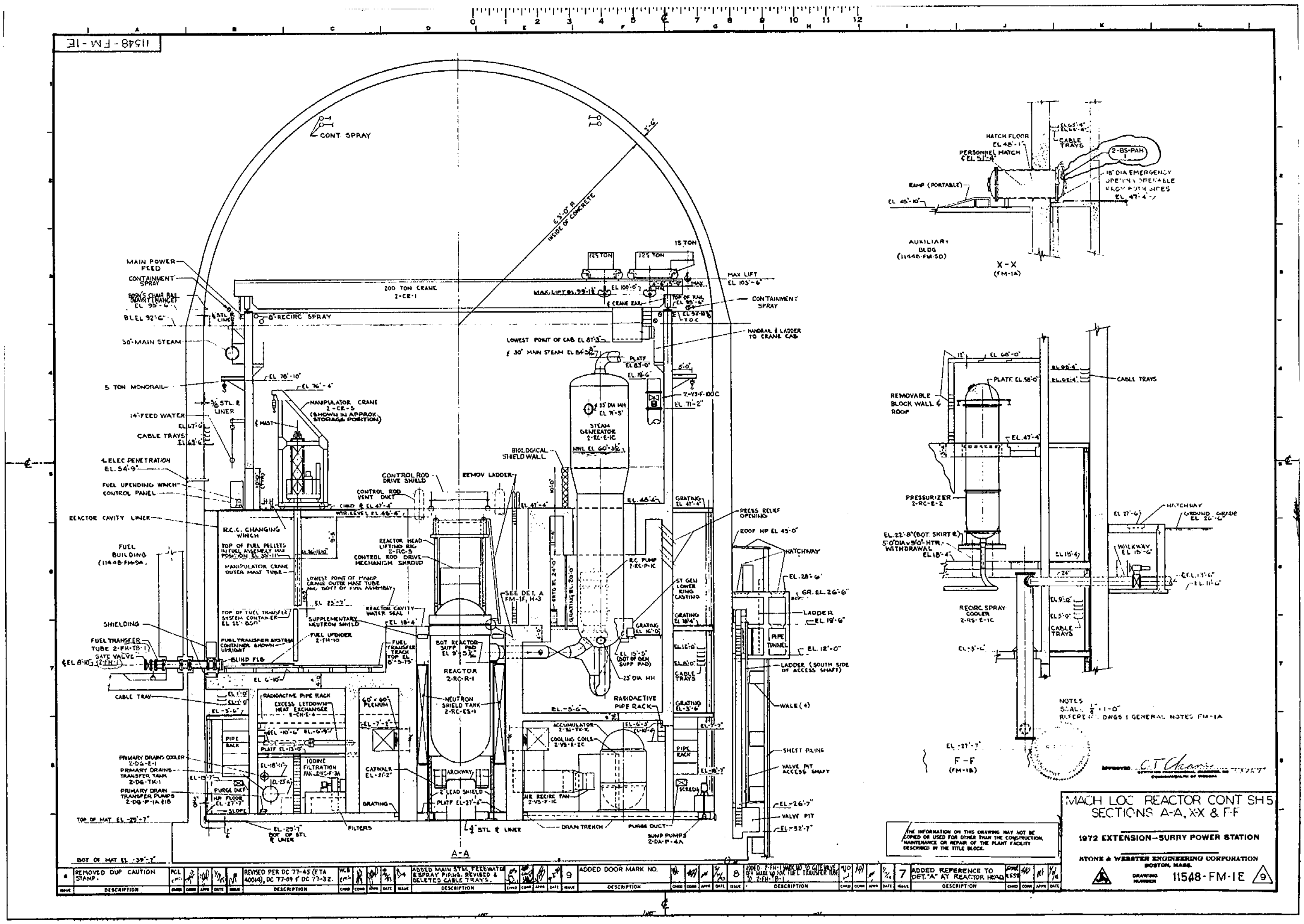
VIRGINIA POWER
NORTH CAROLINA POWER
NUCLEAR ENGINEERING SERVICES
RICHMOND, VIRGINIA

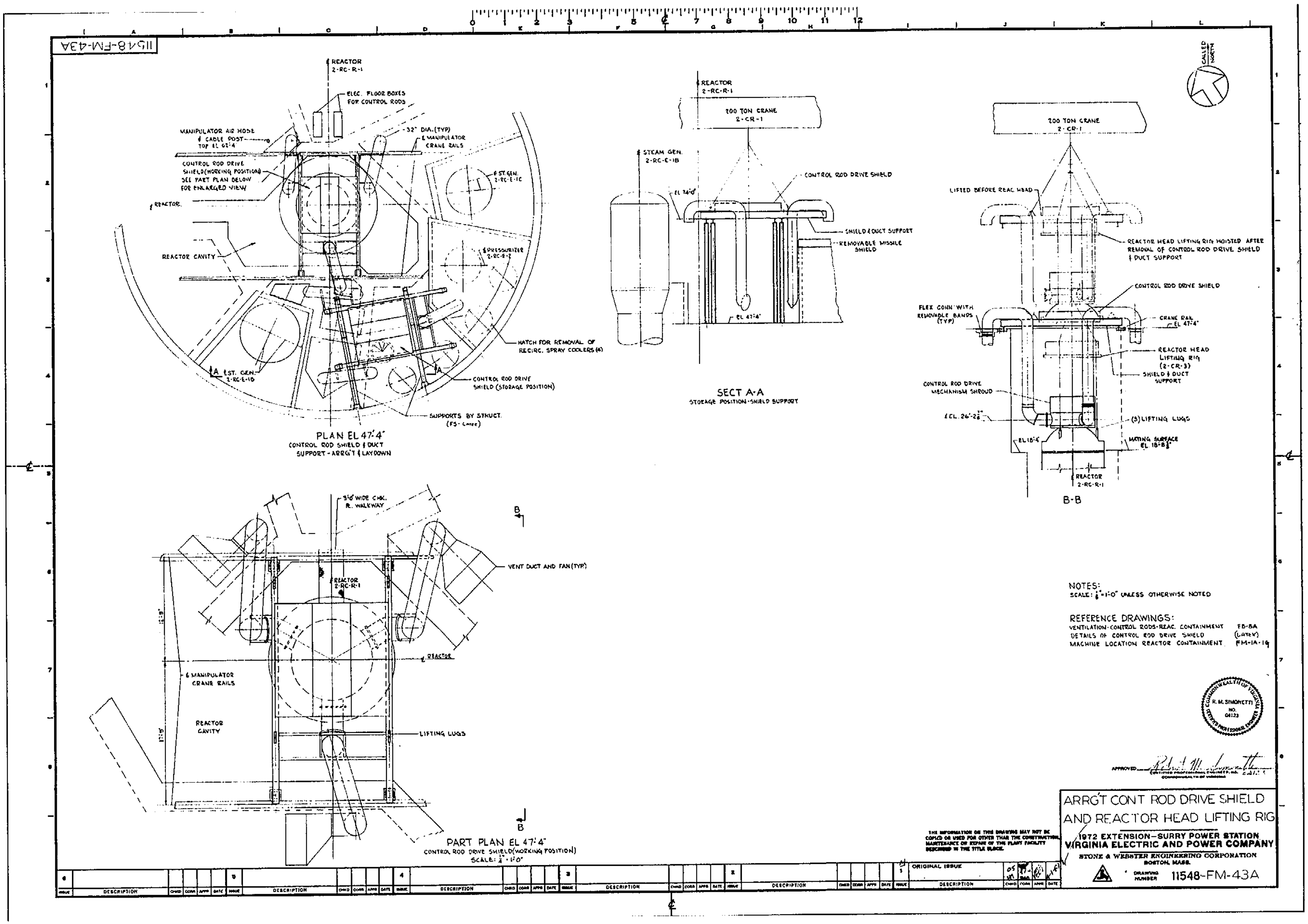
MACH. LOC. - REACTOR CONT. - SH. 5
SECTIONS "A-A", "E-E" & "Z-Z"
SURRY POWER STATION

DSGN --	DSGN SUPV	CAD NO.
DRWN A. MILLNER	ENGR SUPV	DRAWING NO. 11448-FM-1E — REV. 11
CHKD DW	PWT	
DISPL ENGR 8/24/87	SCALE 1/8"=1'-0"	UNLESS OTHERWISE NOTED — SH 1 OF 1

REV	DESCRIPTION
11	ADDED MARK NUMBER FOR FUEL TRANSFER TUBE AND GATE VALVE
10	ADDED REFERENCE TO DETAIL "A" AT REACTOR HEAD.
9	DRAWN PER REVISION 8 OF 11448-FM-1E


11448-FM-43A
CALLED NORTH
200 T CRANE
LIFTED BEFORE REACTOR HEAD
REACTOR HEAD LIFTING RIG LIFTED AFTER REMOVAL OF CONTROL ROD DRIVE SHIELD AND SUPPORT
CONTROL ROD DRIVE SHIELD (CONCRETE & STEEL)
SHIELD & DUCT SUPPORT
FLEX CONN WITH REMOVABLE BANDS (TYP)
CABLES
EL 47'-4"
CRANE RAIL
REACTOR HEAD LIFTING RIG
EL 20'-2⅞"
(3) LIFTING LUGS (3" THK) 3 9/16" DIA HOLE
EL 18'-8 3/8" MATING SURFACE
EL 18'-4"
A-A
SCALE: 1/4" = 1'-0"
B-B
SCALE: 1/4" = 1'-0"
REACTOR HEAD
℄ REACTOR
REACTOR CAVITY
MANIPULATOR CRANE RAILS
CONTROL ROD DRIVE SHIELD (WORKING POSITION)
ELEC FLOOR BOX FOR CONT. RODS
MANIPULATOR CRANE
STEAM GEN 1-RC-E-1B
STEAM GEN 1-RC-E-1C
CONTROL ROD DRIVE SHIELD (STORAGE POSITION)
SUPPORT
M REMOVABLE SLABS FOR MAINTENANCE OR REFUELING
PRESSURIZER 1-RC-E-2
PLAN
CONTROL ROD SHIELD & DUCT SUPPORT
ARRGT & LAY DOWN
SCALE: 1/8" = 1'-0"
200 TON CRANE
CONTROL ROD DRIVE SHIELD
SHIELD & DUCT SUPPORT
REMOVABLE MISSILE SHIELD
PRESSURIZER
EL 47'-4"
C-C
STORAGE POSITION, SHIELD & DUCT SUPPORT
SCALE: 1/8" = 1'-0"
COOLING COILS (TYP)
SHIELD REMOVABLE SEPARATELY IN 3 SECTIONS
MANIPULATOR CRANE RAILS
LIFTING LUGS
PART PLAN
SCALE: 1/4" = 1'-0"
REFERENCE DRAWINGS
VENTILATION CONTROL ROD REACTOR CONTAINMENT FB-5A
DETAILS OF CONTROL ROD DRIVE SHIELD (LATER)
APPROVED (later)
THE INFORMATION ON THIS DRAWING MAY NOT BE COPIED OR USED FOR OTHER THAN THE CONSTRUCTION, MAINTENANCE OR REPAIR OF THE PLANT FACILITY DESCRIBED IN THE TITLE BLOCK
ARRGT CONT ROD DRIVE SHIELD AND REACTOR HEAD LIFTING RIG
SURRY POWER STATION
VIRGINIA ELECTRIC AND POWER COMPANY
STONE & WEBSTER ENGINEERING CORPORATION BOSTON, MASS.
DRAWING NUMBER 11448-FM-43A
ADD LPMS CLAMP-ON BRACKETS AT #6 PER DCP-82-28
ORIGINAL ISSUE
DESCRIPTION


11548-FM-1E
CONT. SPRAY
63'-0" R INSIDE OF CONCRETE
MAIN POWER FEED
CONTAINMENT SPRAY
200 TON CRANE 2-CR-1
8"-RECIRC SPRAY
30"-MAIN STEAM
5 TON MONORAIL
14"-FEED WATER
CABLE TRAYS
4 ELEC PENETRATION EL. 54'-9"
FUEL UPENDING WINCH CONTROL PANEL
REACTOR CAVITY LINER
FUEL BUILDING (11448 FM-5A)
SHIELDING
CABLE TRAY
MANIPULATOR CRANE 2-CR-5 (SHOWN IN APPROX STORAGE POSITION)
R.C.C. CHANGING WINCH
CONTROL ROD DRIVE SHIELD
REMOV LADDER
BIOLOGICAL SHIELD WALL
STEAM GENERATOR 2-RC-E-1C
REACTOR 2-RC-R-1
NEUTRON SHIELD TANK 2-RC-ES-1
RADIOACTIVE PIPE RACK
R.C. PUMP 2-RC-P-1C
PRESS RELIEF OPENING
HATCHWAY
LADDER
PIPE TUNNEL
SHEET PILING
VALVE PIT ACCESS SHAFT
VALVE PIT
SUMP PUMPS 2-DA-P-4A
DRAIN TRENCH
PURGE DUCT
FILTERS
IODINE FILTRATION
CATWALK
A-A
AUXILIARY BLDG (11448-FM-50)
X-X (FM-1A)
PRESSURIZER 2-RC-E-2
RECIRC SPRAY COOLER 2-RS-E-1C
CABLE TRAYS
F-F (FM-1B)
NOTES
SCALE: 1/8" = 1'-0"
REFERENCE DWGS & GENERAL NOTES FM-1A
THE INFORMATION ON THIS DRAWING MAY NOT BE COPIED OR USED FOR OTHER THAN THE CONSTRUCTION, MAINTENANCE OR REPAIR OF THE PLANT FACILITY DESCRIBED IN THE TITLE BLOCK.
MACH LOC REACTOR CONT SH 5
SECTIONS A-A, X-X & F-F
1972 EXTENSION—SURRY POWER STATION
STONE & WEBSTER ENGINEERING CORPORATION
BOSTON, MASS.
DRAWING NUMBER 11548-FM-1E
9
REMOVED DUP CAUTION STAMP.
REVISED PER DC 77-45 (ETA 40014), DC 77-09 & DC 77-32.
ADDED MAIN STM, FEEDWATER & SPRAY PIPING, REVISED & DELETED CABLE TRAYS.
9 ADDED DOOR MARK NO.
7 ADDED REFERENCE TO DET. "A" AT REACTOR HEAD



11548-FM-43A
PLAN EL 47'-4"
CONTROL ROD SHIELD & DUCT SUPPORT - ARRG'T & LAYDOWN
SECT A-A
STORAGE POSITION - SHIELD SUPPORT
B-B
PART PLAN EL 47'-4"
CONTROL ROD DRIVE SHIELD (WORKING POSITION)
200 TON CRANE 2-CR-1
VENT DUCT AND FAN (TYP)
LIFTING LUGS
REACTOR CAVITY
NOTES:
SCALE: 1/8"=1'-0" UNLESS OTHERWISE NOTED
REFERENCE DRAWINGS:
VENTILATION-CONTROL RODS-REAC. CONTAINMENT FB-8A
DETAILS OF CONTROL ROD DRIVE SHIELD (LATER)
MACHINE LOCATION REACTOR CONTAINMENT FM-1A-19
ARRG'T CONT ROD DRIVE SHIELD AND REACTOR HEAD LIFTING RIG
1972 EXTENSION—SURRY POWER STATION
VIRGINIA ELECTRIC AND POWER COMPANY
STONE & WEBSTER ENGINEERING CORPORATION
BOSTON, MASS.
DRAWING NUMBER 11548-FM-43A
ORIGINAL ISSUE