VIRGINIA POWER

DESIGN CHANGE - NUCLEAR POWER STATION

STD-GN-0001 Rev. 10 POW 11

1. Design Change Title/Station/Unit		2. Design Change Number
STEAM GENERATOR REPLACEMENT / NORTH ANNA / UNIT 1		DC 90-13-1
3. Component Mark No. 1-RC-E-1A, 1-RC-E-1B, 1-RC-E-1C		4. W.R./W.O.

5. Safety Evaluation Req'd	6. Involves an Unreviewed Safety Question?	7. Technical Specification Change Required	8. Approval Level	9. Q. A. Category
[X] Yes [ ] No See Appendix 4-1	[ ] Yes [X] No	[ ] Yes [X] No	[X] Station [ ] MSRC/NRC	[X] SR [ ] NSQ [ ] NS

Name/Affiliation (Print)	Signature	Date
10. Preparing Engineer/Affiliation (Print) MARK S. BARTH / BECHTEL	11. Signature MS Barth	12. Date 9/18/92
13. Mechanical Reviewer/Affiliation (Print) * ROBERT A. BAIN / SWEC MARK A. SMITH / BECHTEL	14. Signature Robert A. Bain M.A. Smith	15. Date 9/21/92 9/18/92
16. Electrical Reviewer/Affiliation (Print) WOLF-DIETER FREY / Bechtel DEBORAH A. SMITH / BECHTEL	17. Signature W. Frey / Bechtel Deborah A. Smith	18. Date 9/18/92 9/18/92
19. Civil Reviewer/Affiliation (Print) * ROBERT A. BAIN / SWEC DEBRA NEVERGOLD / BECHTEL	20. Signature Robert A. Bain Debra Nevergold	21. Date 9/21/92 9/18/92
22. NAPS Reactor Engineer (Print) RG MCANDREW VA PWR	23. Signature RG McAndrew	24. Date 9-21-92
25. Project Engineer (Print) S.L. Wilke	26. Signature S.L. Wilke	27. Date 9-21-92
28. Supervisor of Testing (Print) D.L. SNODGRASS FOR S.L. HARVEY	29. Signature D.L. Snodgrass	30. Date 9-21-92
31. Design Control Engineer (Print) W W CHAISSON WWC 9-30-92	32. Signature W.W. Chaisson	33. Date 9-24-92

34. SNSOC Review:

[X] Yes [ ] No - SNSOC to review the Operational Readiness Reviews?

Remarks:

INFORMATION ONLY

35. SNSOC Chairman (Print)	36. Signature	37. Date
JA STALL	J.A. Stall	9/30/92

*Indicates review of [illegible] pipe stress analysis, and critical commodities for cold measurement [illegible]

9210280294 921023
PDR ADOCK 05000338
P PDR

DCHG9013

1. Design Change Title/Station/Unit	2. Design Change Number
STEAM GENERATOR REPLACEMENT / NORTH ANNA / UNIT 1	DC 90-13-1

TABLE OF CONTENTS

1. Design Change Title/Station/Unit	2. Design Change Number
STEAM GENERATOR REPLACEMENT / NORTH ANNA / UNIT 1	DC 90-13-1

TABLE OF CONTENTS

VIRGINIA POWER

INTERIM REVISION - NUCLEAR POWER STATION

STD-GN-0001 Rev. 10 POW 11

1. Design Change Title/Station/Unit		2. Design Change No.
STEAM GENERATOR REPLACEMENT / NORTH ANNA / UNIT 1		DC 90-13-1
3. Preparing Engineer/Affiliation (Print) MARK S. BARTH / BECHTEL	4. Signature MS Barth	5. Date 9/29/92
6. Mechanical Reviewer/Affiliation (Print) MARK D SMITH / BECHTEL	7. Signature Mark D Smith	8. Date 9/29/92
9. Electrical Reviewer/Affiliation (Print) MARK D SMITH / BECHTEL FOR D FREY / BECHTEL PER TELECON	10. Signature Mark D. Smith FOR D FREY PER TELECON	11. Date 9/29/92
12. Civil Reviewer/Affiliation (Print) DEBRA NEVERGOLD / BECHTEL	13. Signature Debra Nevergold	14. Date 9/29/92
15. Other Discipline Reviewer/Affiliation (Print) N/A	16. Signature N/A	17. Date N/A
18. Project Engineer or DEO Supervisor (Print) Sheri Lyn Wilkie	19. Signature Sheri Lyn Wilkie	20. Date 9-30-92

21. Description of Change	22. Reason for Change
ER&D Section 2.2.7.2. Revised the 2nd and 4th paragraphs to discuss (1) containment purge system operation in accordance with Operating Procedure 21.2, (2) radiation monitoring, and (3) a temporary cover for the hatch opening when the equipment hatch is removed. Also, revised the wording to address the radiological consequences of a postulated drop of an old steam generator lower assembly inside the containment via reference to Calculation 21809-M-02.	To acknowledge purge system operation in accordance with the station procedure, to reflect radiation monitoring available to detect releases, to identify the presence of the temporary cover for the hatch opening and to discuss the radiological consequences of a release.
ER&D Section 2.3.3.c. Revised the last sentence of this implementation requirement to read: "While fuel is in the containment, materials and tools associated with the SGR shall not be handled or transported within the refueling barrier."	To clarify the description of areas where handling and transporting materials and tools will be prohibited.
ER&D Section 2.3.3.g. Revised this implementation requirement to read: "Coincident with equipment hatch removal, containment purge system alignment and operation shall be in accordance with Operating Procedure 21.2 so as to minimize any air leakage out of the containment. In addition to the stack monitors and the containment area radiation monitors, a continuous air monitor shall be in use adjacent to the equipment hatch.	To acknowledge purge system operation in accordance with the station procedure, to reflect radiation monitoring available to detect releases and to identify the presence of the temporary cover for the hatch opening.

VIRGINIA POWER

INTERIM REVISION (SUPPLEMENT) - NUCLEAR POWER STATION

1. Design Change Title/Station/Unit	2. Design Change No.
STEAM GENERATOR REPLACEMENT / NORTH ANNA / UNIT 1	DC 90-13-1

21. Description of Change (continued)	22. Reason for Change (continued)
In accordance with Drawing N-9013-1-M-406, a temporary cover shall also be available to isolate the hatch opening in case of loss of the containment purge system."	
ER&D Section 2.3.4. Added the following implementation requirement: "To accommodate any leakage from the loop isolation valves, two primary drain transfer tank pumps shall be available and associated drain valves on the hot and crossover legs shall be tagged open from the time the loop isolation valves are shut until reactor vessel level has been drained to below the reactor vessel nozzles."	To add the requirement to provide a means to control loop isolation valve leakage and provide this capability so long as the reactor vessel water level remains above the reactor vessel nozzles.
ER&D Section 3.2. Revised the last sentence of the discussion for Technical Specification 3/4.7.2, "Steam Generator Pressure/Temperature Limitation," to read: "The requirements of this Technical Specification requirement will continue to be met."	To ensure that the requirements of this technical specification are maintained during the steam generator replacement, when applicable.
ER&D Section 3.2. Added a discussion of Technical Specification 3/4.7.8.1, "Safeguards Area Ventilation Systems."	To ensure that the requirements of this technical specification are maintained during the steam generator replacement, when applicable, and that the surveillance requirements will be invoked, if required.
ER&D Section 3.14. Revised the 1st and 5th paragraphs of this section to discuss (1) containment purge system operation in accordance with Operating Procedure 21.2, (2) radiation monitoring, and (3) a temporary cover for the hatch opening when the equipment hatch is removed. Also, revised the wording to address the radiological consequences of a postulated drop of an old steam generator lower assembly inside the containment via reference to Calculation 21809-M-02.	To acknowledge purge system operation in accordance with the station procedure, to reflect radiation monitoring available to detect releases, to identify the presence of the temporary cover for the hatch opening and to discuss the radiological consequences of a release.
Table of Contents. Revised the total number of pages for the Engineering Review and Design and Appendix 4.1, Safety Evaluation; added this Interim Revision.	To reflect the current number of pages for these documents and this Interim Revision.

1. Design Change Title/Station/Unit	2. Design Change No.
STEAM GENERATOR REPLACEMENT / NORTH ANNA / UNIT 1	DC 90-13-1

21. Description of Change (continued)	22. Reason for Change (continued)
Drawing Revision Record. Drawing N-9013-1-M-406 has been revised to delete Note 8 and revise the reference drawings. The latest revision of Drawing N-9013-1-M-406 is now Revision 1.	To delete an incorrect note and to indicate the correct drawing references.
Appendix 4.1, Safety Evaluation. Incorporate changes described above into the applicable responses of the safety evaluation.	To make Appendix 4.1, Safety Evaluation, consistent with the ER&D.

ENGINEERING REVIEW AND DESIGN AND PROGRAM REVIEW CHECKLIST

1. Design Change Title/Station/Unit		2. Design Change Number
STEAM GENERATOR REPLACEMENT / NORTH ANNA / UNIT 1		DC 90-13-1
3. Preparing Engineer/Affiliation (Print) MARK BARTH/BECHTEL	4. Signature MSBarth	5. Date 9/18/92
6. Reviewing Engineer/Affiliation (Print) MARK D. SMITH /BECHTEL	7. Signature M.D Smith	8. Date 9/18/92

9. Engineering Review and Design:

TABLE OF CONTENTS

1. Design Change Title/Station/Unit	2. Design Change Number
STEAM GENERATOR REPLACEMENT / NORTH ANNA / UNIT 1	DC 90-13-1

9. Engineering Review and Design:

1. Design Change Title/Station/Unit	2. Design Change Number
STEAM GENERATOR REPLACEMENT / NORTH ANNA / UNIT 1	DC 90-13-1

9. Engineering Review and Design:

1. Design Change Title/Station/Unit

STEAM GENERATOR REPLACEMENT / NORTH ANNA / UNIT 1

2. Design Change Number

DC 90-13-1

9. Engineering Review and Design:

1.0 STATEMENT OF PROBLEM

The three Westinghouse Model 51 steam generators, 1-RC-E-1A, 1B, and 1C, have experienced corrosion-related degradation that require periodic inspection and plugging of the steam generator (SG) tubes to ensure their continued safe and reliable operation. Despite improvements in secondary water chemistry, tube degradation has continued in the steam generators. Consequently, the lower tube bundle assemblies will be replaced.

The new North Anna Unit 1 Model 51F steam generators discussed throughout the text of this ER&D are actually composites of the existing Westinghouse Model 51 steam domes and new Westinghouse Model 51F lower assemblies. Thus, the new North Anna Unit 1 Model 51F steam generators are different than a complete Westinghouse Model 51F steam generator.

1.1 OPERATIONAL CONDITIONS

In the current plant configuration, the plant is limited to 95% power. In addition, the current status of the existing steam generators will require extensive tube inspections which will result in significant dose to personnel.

1.2 SYSTEM/EQUIPMENT PROBLEMS

The problem areas in the original steam generator design are as follows:

a. Stress or intergranular corrosion in the tubing in the tubesheet region and at support plate intersections

b. Mechanical wear of the tubing at tube-to-anti-vibration bar intersections

c. Primary side stress cracking of tight radius U-bends in the tubing

1.3 DESIGN BASIS CRITERIA

The steam generator replacement (SGR) will be performed in accordance with the requirements of the ASME Code, Section XI, 1983 Edition and Summer 1983 Addenda which allows use of the original design codes, later editions of the original design codes, or ASME Section III for replacement activities.

The original design codes are as follows:

- Steam Generator Assemblies - ASME Section III, 1968 Edition through Winter 1968 Addenda, code case 1498, and "N" stamped.
- Reactor Coolant Piping - USAS B31.7-1969 through 1970 Addenda and applicable Code Cases
- Main Steam Piping - USAS B31.7-1969 through 1970 Addenda and applicable Code Cases
- Feedwater Piping - USAS B31.7-1969 through 1970 Addenda and applicable Code Cases
- Blowdown/Drains - USAS B31.7-1969 through 1970 Addenda and applicable Code Cases

1. Design Change Title/Station/Unit	2. Design Change Number
STEAM GENERATOR REPLACEMENT / NORTH ANNA / UNIT 1	DC 90-13-1

9. Engineering Review and Design:

- Wet Layup Piping - USAS B31.7-1969 through 1970 Addenda and applicable Code Cases
- Chemical Feed Piping - USAS B31.7-1969 through 1970 Addenda and applicable Code Cases
- Sample System Piping - USAS B31.7-1969 through 1970 Addenda and applicable Code Cases
- SG Level Instrument Tubing - USAS B31.7-1969 through 1970 Addenda and applicable Code Cases

The new steam generator lower assemblies have been fabricated in accordance with the ASME Code, Section III, 1986 edition and "NPT" stamped. The closure (girth) weld will be performed to meet ASME Section III, 1986 edition in accordance with ASME Section XI requirements and will not be "NA" s[illegible]d. Therefore, the new steam generators will be non "N" stamped vessels as allowed per AS[illegible]ection XI.

New piping materials, including valves and the new SG lower assemblies, will be procured in accordance with ASME, Section III, 1986 Edition, due to the unavailability of components fabricated in accordance with ANSI B31.7 requirements. Components currently available in existing inventory may be utilized to facilitate steam generator replacement provided these components meet or exceed the existing design code requirements. Deviations to the original design codes have been reconciled in accordance with the requirements of ASME Section XI (see Appendix 4-33).

1.4 DEFICIENCY TO BE CORRECTED

The deficiency to be corrected is to provide a steam generator with enhanced material to resist previously exhibited degradation mechanisms. In addition, portions of the feedwater lines, steam generator blowdown lines and the steam generator shell drain lines will be replaced with chrome-moly material to provide improved erosion/corrosion characteristics.

1.5 TECHNICAL ISSUES

The following technical issues have been identified:

- Steam Generator Corrosion/Material Problems

  The following features of the original steam generators have made them susceptible to intergranular stress corrosion cracking (IGSCC):

  1. The tubing, mill annealed ASTM SB-163 Alloy 600, has a microstructure that can experience IGSCC in an alkaline secondary side environment. Additionally, this material is susceptible to primary water stress corrosion cracking.

  2. The initial expansion process mechanically rolled the bottom 2.5 inches of the tubes, leaving a crevice between the tubes and the tubesheet on the secondary side which allowed concentration of impurities. After one cycle of operation, the tubes were explosively expanded (WEXTEX) over the thickness of the tubesheet. This expansion closed the crevice. However, this expansion either created an area

1. Design Change Title/Station/Unit	2. Design Change Number
STEAM GENERATOR REPLACEMENT / NORTH ANNA / UNIT 1	DC 90-13-1

9. Engineering Review and Design:

of overexpansion or left a smaller crevice at the top of the tubesheet.

3. The 0.75-inch-thick carbon steel tube support plates have round tube holes with a nominal radial clearance of 8 mils. This support plate design allows concentration of impurities and can provide a localized aggressive environment at the tube-to-tube support plate intersections. The presence of flow holes in the plates tends to further minimize flow around the tubes, and allows the aggressive environment to exist in the support plate area. This aggressive local environment can lead to corrosion of the tube and/or support plates. Corrosion products from the corrosion of the carbon steel plates completely fills the gap in some locations and can deform the tube, thereby adding stress and contributing to IGSCC.

4. Another localized aggressive environment just above the tubesheet surface is provided when thermal-hydraulic characteristics result in sludge accumulation and local dryout on the tubesheet.

IGSCC in the tubesheet region and at the support plate intersections is a pervasive problem in the original steam generators and is the primary reason for replacing the steam generators.

- Containment Modifications

Temporary runway beams will be installed to support loads from the movement of steam generator lower assemblies in and out of the equipment hatch. Steel decking will temporarily be installed over the open sections of the reactor refueling cavity, and portions of the decking will be designated as construction laydown areas. Temporary laydown supports for the steam domes will be provided. A jib crane and an auxiliary crane will be temporarily installed to support handling small loads. Temporary modifications will be made to containment purge to allow for smoke removal from the containment dome area.

- Removal of Interferences

Interferences to SG movement between their permanent locations and the hatch will be removed. These items include portions of the SG bioshield walls, the operating floor at the hatch, equipment hatch barrel floor, SG valve access platforms, tubing and tubing supports, and mechanical commodities/equipment. The engineering required to support interference removal is included in DC 90-15-1 or work plans, as applicable.

- Separation of Steam Generator Assemblies to Accommodate Removal

The separation of the steam domes and removal/replacement of the steam generator lower assemblies through the existing containment equipment hatch is the method selected to replace the steam generators. This method was selected due to limitations on the diameter of equipment that can be moved through the hatch. Implementation of the replacement will be accomplished by severing the reactor coolant and all other attached piping at the SG nozzles, severing the SGs within the transition cone and removing the lower assemblies from the containment. The steam domes will remain in containment and

1. Design Change Title/Station/Unit

STEAM GENERATOR REPLACEMENT / NORTH ANNA / UNIT 1

2. Design Change Number

DC 90-13-1

9. Engineering Review and Design:

will be modified by the installation of a flow restrictor in the steam outlet nozzle, the existing downcomer flow resistance plates on the wrapper plate removed, and the shell prepared for welding to the lower assemblies. The new shop-fabricated SG lower assemblies will be rigged into the containment through the equipment hatch and connected to the original steam domes and reactor coolant piping. Welding will be performed in compliance with ASME Section XI requirements and a Virginia Power approved QA program. All work associated with the SG vessel is safety-related.

- RCS Pipe Cuts

  During offloading of the reactor core to the spent fuel pool, cutting operations for the main steam, feedwater, and other steam generator connected piping are planned to be performed for all three steam generators. Additionally, cutting of the hot and cold leg RCS piping on the B steam generator will also be initiated and may be completed prior to fuel offload. A seismic evaluation has been performed on the RCS, main steam, feedwater and other plant secondary-side piping systems to verify the acceptability of cutting prior to the completion of defueling. These evaluations are documented in Calculations 02072.13-NP(B)-008-XE and 02072.13-NP(B)-009 XE (References 6.12.21 and 6.12.22).

  The loop isolation valves will be isolated prior to making any RCS piping cuts. Cutting of the Loop A and C RCS piping will not be performed until all fuel has been removed from the containment, the reactor cavity drained, and the fuel transfer tube gate valve closed and blind flange installed.

  Normal defueling practices will not change.

- Heavy Loads Inside the Containment

  Movement of heavy loads inside the containment will be in accordance with the station heavy loads procedure 0-MCM-1303-01 until all fuel has been removed from the containment, the reactor cavity drained, and the fuel transfer tube gate valve closed and the blind flange installed. During the defueled condition, movement of heavy loads will be in accordance with approved project procedures. Refer to Section 3.14 for a further discussion of heavy loads.

- Insulation Removal/Replacement

  The existing thermal insulation on the steam generators will be replaced with new fiberglass blanket type insulation. The quilted, light density, semi-rigid fibrous glass blankets will utilize stainless steel hook-and-loop fasteners and will be covered by a stainless steel jacket.

- Accident Analysis Reevaluation

  The acc ents analyzed in the UFSAR have been reviewed to determine if they are impacted by the design characteristics of the new steam generators. The results of the accident evaluation show that, because of the essential duplication of safety-related parameters, the new steam generators will not result in violation of any accident acceptance criteria. The LOCA containment response was reanalyzed and the results demonstrate acceptable analysis margins. The evaluation confirms that the conclusions for the accident analyses presented in the UFSAR remain valid.

- Impact Upon Permanent Plant Equipment

  Other temporary modifications to permanent plant equipment that will be performed as part of the SGR include:

  - Buried facilities along the haul route will be temporarily protected for the loads from the steam generator transporter/prime mover and load test.
  - The containment purge exhaust system will be temporarily modified to allow for smoke removal from the containment dome area.
  - To facilitate the installation of temporary air chiller units to support the SGR, the security door (A-95-1) located at the entrance to the MCC room roof will be temporarily removed and replaced with a modified security door which will allow passage of air hoses/pipes from the roof down to the containment personnel hatch.
  - The power feeder to RCP-1B will be utilized to supply power to a temporary construction power distribution system that will be installed for in-containment SGR work activities.

  These temporary modifications are described in Section 2.2.3.

  In addition, rigging loads have been evaluated for their effects on permanent plant structures and equipment and have been determined to be acceptable. See Section 2.2.7.1.

1.6 SHORT AND LONG TERM CONSEQUENCES

The short-term consequences of not performing the steam generator replacement include increased radiation exposure and expense due to increased tube inspection and replacement activities, and further unit power reduction.

The long-term consequence is a potential loss of unit availability.

The documents listed below are applicable to the steam generator primary-to-secondary leakage concerns associated with tube degradation. These documents, however, do not directly specify a condition or a number of tubes plugged when the steam generators must be repaired or replaced. The original steam generators were provided with more tubes than were required for the design load operation. When this margin has been utilized by plugging the degraded tubes, further plugging results in load reduction. Therefore, continued operation with the existing steam

1. Design Change Title/Station/Unit	2. Design Change Number
STEAM GENERATOR REPLACEMENT / NORTH ANNA / UNIT 1	DC 90-13-1

9. Engineering Review and Design:

generators would be an economic and a reliability concern.

a. NRC Bulletin 88-02, Rapidly Propagating Fatigue Cracks in Steam Generator Tubes

b. NRC Information Notice 90-49, Stress Corrosion Cracking in PWR Steam Generator Tubes

c. NRC Regulatory Guide 1.83, Inservice Inspection of Pressurized Water Reactor Steam Generator Tubes

d. NRC Regulatory Guide 1.121, Bases for Plugging Degraded PWR Steam Generator Tubes

e. ASME Code Section XI, Rules for Inservice Inspection Nuclear Power Plant Components

f. ASME Code Section III, Nuclear Power Plant Components, Div. 1.

g. North Anna Unit 1 Technical Specifications, Sections 3/4.4.5, 3/4.4.6

2.0 PROPOSED RESOLUTION

2.1 EXECUTIVE SUMMARY

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

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

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

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

The steam generator lower assemblies will be removed and replaced through the existing containment equipment hatch. This replacement process is commonly referred to as the two-piece replacement method. The two-piece replacement through the equipment hatch was determined to be the best overall method for North Anna Unit 1 due to limitations on the diameter of equipment that can be moved through the equipment hatch. The containment equipment hatch is large enough to allow passage of the steam generator lower assemblies only, not the steam domes. The existing SG lower assemblies will be removed by severing the reactor coolant and all other attached piping at the steam generator nozzles, severing the steam generators within the transition cone, and removing the lower assemblies from the containment. Following removal of the old lower assemblies, the new shop-fabricated steam generator lower assemblies will be transported into the containment through the equipment hatch and connected to the original steam domes and reactor coolant piping. Use of the equipment hatch eliminates the need to modify the containment wall concrete or pressure boundary.

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

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

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

- The steam generator and adjacent piping will be covered with new blanket insulation which has stainless steel jacketing. The replacement blanket insulation will meet or exceed the design requirements of the existing insulation.

- The steam generator upper restraints will be replaced with an equivalent restraint. Demolition of the existing upper restraints is required due to difficulties associated with their removal from the steam generators and the effort involved to reuse them.

2.2 DESIGN DESCRIPTION

2.2.1 <u>Design Changes and Enhancements in the New Steam Generator Lower Assemblies</u>

As specified in the Westinghouse safety evaluation, SECL-90-113 (Appendix 4-19), a number of design enhancements have been incorporated into the North Anna Unit 1, Series 51 replacement steam generators. The replacement steam generators have been designed to comply with the original design bases and safety analyses as currently documented in the UFSAR.

The existing steam generators were fabricated in accordance with Section III, 1968 Edition through Winter 1968 Addenda of the ASME Boiler and Pressure Vessel Code and

1. Design Change Title/Station/Unit

STEAM GENERATOR REPLACEMENT / NORTH ANNA / UNIT 1

2. Design Change Number

DC 90-13-1

9. Engineering Review and Design:

"N" stamped.

The replacement steam generator components have been fabricated in compliance with the 1986 Edition of ASME Section III and the new lower assembly has been "NPT" stamped. The Stress Report is based on the 1968 Edition of the ASME Section III, including all addenda through the Winter of 1968. For materials whose strength properties do not exist in the earlier edition, the material strength properties used in the Stress Report were determined from Section III, 1986 Edition. Additionally, in cases where the later edition specifies more conservative material strength properties than the earlier edition, the more conservative values were used in the Stress Report. Hence, the replacement steam generators are fabricated and analyzed to standards which are, at a minimum, equivalent to the existing units. Welding of the new steam generator lower assemblies to the existing upper assemblies will be performed to meet ASME Section III, 1986 edition, in accordance with the requirements of the 1983 edition through Summer 1983 addenda of Section XI of the ASME Code and the weld will not be "NA" stamped. Therefore, the new steam generators will be non "N" stamped vessels as allowed per ASME Section XI.

The design changes made to the new steam generator lower assemblies to address the problems identified in Section 1.0 as well as other design changes and enhancements are summarized below and are evaluated in the Westinghouse Safety Evaluation (Appendix 4-19). A new technical manual for the complete steam generator assembly is being submitted (Reference Appendix 4-22). This manual describes in detail the design changes and enhancements associated with the complete steam generator assemblies.

2.2.1.1 Forged Channel Heads

The one-piece SA-508, Class 3 forged channel head will have the same dimensions as the original SA-216, Grade WCC channel head casting except that the outside diameter of the primary nozzle ends will begin tapering into the channel head bowl at a point axially closer to the bowl. This results in more straight length from the nozzle ends inward and will facilitate inservice inspection (ISI). Forging the channel head instead of casting results in lower porosity and thereby enhances nondestructive examination. Additionally, a thicker primary divider plate and [redacted] diameter primary manway diaphragm screws [redacted] will be provided. Use of a thicker primary divider plate will result in reduced stresses at the plate-to-tubesheet interface and stiffening of the channel head. The use of [redacted] diameter primary manway diaphragm screws will allow for easier and quicker manway seal plate removal/reinstallation and thereby reduce personnel exposure during maintenance. The inside surface of the channel head will be provided with a 63 RMS finish. Providing a smooth finish on the inside of the channel head makes adherence of surface contamination more difficult, which should reduce personnel exposure during future maintenance activities in the channel head.

W Prop.

1. Design Change Title/Station/Unit

STEAM GENERATOR REPLACEMENT / NORTH ANNA / UNIT 1

2. Design Change Number

DC 90-13-1

9. Engineering Review and Design:

2.2.1.2 Forged Lower Shell Barrels and Transition Cone

The lower shell barrels will be fabricated as machined seamless forgings from SA-508, Class 3 low alloy steel ring forging. The new lower assembly design incorporates the stub barrel into the lower shell which will eliminate inservice inspection (ISI) of one weld.

The transition cone between the steam dome and lower assembly of the new steam generator will be fabricated as a machined seamless forging with a fully developed conical profile with [redacted] downstand. This downstand relocates the weld away from the actual transition points on the Sgs thereby resulting in lower discontinuity stresses at the girth weld. This relocation of the weld results in the elimination of this weld from the ISI program. The material chosen for the transition cone forging is a low alloy steel, SA-508, Class 3, which has the same "P Group" numbers as the SA-533, Grade A, Class 1, plate used to form the original transition cone, making the juncture readily weldable in the field. The original transition cone is made from rolled plate and has three longitudinal weld seams. W Prop.

The upper surface of the transition cone of the new steam generator lower assemblies will be supplied with a girth weld preparation to facilitate attachment of the steam dome to the new lower assembly. The transition cone finished diameter is based on the diametrical clearance requirements of the containment building equipment hatch. This dimension is also the basis for determining the diameter and elevation of the weld preparation on the interfacing steam dome.

2.2.1.3 Tube Support Plates

The quatrefoil tube support plate hole design features four flow lobes and four support lands. The lands laterally support the tubing and the lobes provide a path for water to flow adjacent to the tube. The quatrefoil design directs the flow along the tubes in a way that minimizes steam formation and chemical concentration at the tube-to-tube support plate intersections. The quatrefoil support plate results in higher average velocities adjacent to the tubes than the original lower assembly support plates, which feature circulation flow holes between tube holes. The combination of high velocities in the tube support plate region and the use of corrosion resistant material (SA-240, Type 405 ferritic stainless steel instead of the existing SA-285, Grade C carbon steel) minimizes corrosion in the vicinity of the support plates. The original lower assembly tube support arrangement results in peripheral flow around the tube and between the circular tube hole in the tube support plate.

2.2.1.4 Flow Distribution Baffle

A flow distribution baffle of SA-240, Type 405 ferritic stainless steel is located [redacted] above the tubesheet. The flow distribution baffle has W Prop

1. Design Change Title/Station/Unit

STEAM GENERATOR REPLACEMENT / NORTH ANNA / UNIT 1

2. Design Change Number

DC 90-10-1

9. Engineering Review and Design:

[redacted] broached tube holes [redacted] surrounding an open central cutout. The flow distribution baffle directs flow radially across the tubesheet and up the center of the bundle through the central cutout thus increasing lateral flow velocities across the tubesheet. W Prop.

The flow distribution baffle in conjunction with the tubelane blocks causes sludge to preferentially deposit near the center of the bundle near the blowdown intake. In addition, the octafoil hole design allows controlled flow through the flow distribution baffle, minimizes steam formation, and reduces the potential for sludge accumulation at the tubesheet.

2.2.1.5 **Support Plate Stayrods**

The support plate stayrods and spacers are SA-696, Grade C carbon steel bar and SA-106, Grade B carbon steel pipe, respectively. A total of seven stayrods with spacer pipes will be provided on each new steam generator compared to five stayrods on the original steam generators.

2.2.1.6 **Tubelane Blocking Devices**

[redacted] tubelane blocking devices are provided in the new lower assemblies. [redacted] They are positioned to restrict the flow of recirculating water along the tubelane thereby improving water flow patterns into the remainder of the tube bundle. The tubelane blocks are designed to be compatible with the blowdown system and will not interfere with sludge lancing and/or inspection efforts. W Prop.

2.2.1.7 **Handholes/Inspection Ports**

The new lower assemblies will include additional handholes. Four 6-inch handholes are located approximately 14 inches above the tubesheet (two centered with the tubelane and two 90 degrees to the tubelane). Two additional 6-inch handholes are aligned with the tubelane above the flow distribution baffle, bringing the total number of handholes to six. The handhole arrangement will enhance inspections and facilitate sludge lancing.

Two 4-inch inspection ports are located on the transition cone lower shell at an elevation slightly above the top tube support plate and aligned with the tubelane. This location facilitates inspection of the top support plates and Row 1 tube U-bends. The U-bend region of the Row 1 tubes are directly observable through these inspection ports.

1. Design Change Title/Station/Unit

STEAM GENERATOR REPLACEMENT / NORTH ANNA / UNIT 1

2. Design Change Number

DC 90-[illegible]-1

9. Engineering Review and Design:

2.2.1.8 Blowdown System

The design of the blowdown system in the new lower assemblies is similar to that used in the original steam generators. The new steam generators will be equipped with a pair of blowdown pipes located just above the surface of the tubesheet, with one pipe located on either side of the central stayrod. The pipes are not connected within the steam generator. Each pipe is connected to a separate blowdown coupling which is welded to the outside diameter surface of the steam generator tubesheet. [redacted]

W Prop.

The new steam generator blowdown system will accommodate higher blowdown flow rates and velocities than the original steam generator's system. This is accomplished by increasing the internal header pipe size [redacted] The external blowdown nozzle coupling size has also increased from 2 to 2.5 inches. However, the original portion of the blowdown system that remains currently limits the flowrate. Later, if additional blowdown piping changes are made outside of containment, the new steam generator blowdown system design will be capable of responding more rapidly to steam generator chemical imbalance conditions. The new steam generators are designed so that they can be operated at a blowdown flow rate of 3 percent of steam generator maximum flow per blowdown line (or a total of 6 percent of maximum steam flow if both blowdown lines are used) for a cumulative period of 1 year and at a flow rate of 1.5 percent of steam generator maximum steam flow per blowdown line (or a total of 3 percent of maximum steam flow if both blowdown lines are used) for the remainder of the steam generator service life. However, the existing blowdown flowrates will be maintained at the present values.

W Prop.

2.2.1.9 Heat Transfer Tubes

The number of tubes in the new lower assemblies has been increased from 3,388 to 3,592. The pitch of the tubes in the new lower assembly is slightly reduced from 1.281 inches in the original assemblies to 1.225 inches. This allows an increase in the number of tubes and hence an increase in the heat transfer area from 51,500 ft^2 to 54,500 ft^2.

To allow optimal thermal-hydraulic performance, the heat transfer tubing is procured with a strict wall thickness control. A nominal wall thickness of 0.050 inches will be provided for each new lower assembly tube set; this is unchanged from the existing steam generator tubes. The new lower assemblies feature a minimum U-bend radius [redacted] Additionally, the U-bend regions of the inner [redacted] rows of tubes are heat treated after bending to further reduce residual stresses.

W Prop.

The use of SB-163 Alloy 690 TT (Code Case N-20) tube material with optimum

1. Design Change Title/Station/Unit

STEAM GENERATOR REPLACEMENT / NORTH ANNA / UNIT 1

2. Design Change Number

DC 90-13-1

9. Engineering Review and Design:

final mill anneal at about [redacted] followed by thermal treatment for chromium replenishment at [redacted], has been shown to be several orders of magnitude [redacted] mill annealed SB-163 Alloy 600 tube material in the original steam generators, with respect to stress corrosion, particularly that originating from the primary side, and is more resistant to caustic induced intergranular attack (IGA). W Prop

2.2.1.10 Anti-Vibration Bars

Three sets of anti-vibration bars (AVBs) instead of the two in the original design are installed in between adjacent columns of tubes for tube support in the U-bend region. The tubes and AVBs are dimensionally controlled to minimize tube-to-AVB clearances and to provide close support of the tubes against fluid-induced motion which can lead to tube wear. The AVBs are wider than the original design to provide greater contact area between the tubes and AVBs. In addition, the AVBs are inserted to uniform depths to provide consistent support. [redacted]

The AVB material is [redacted] stainless steel which features enhanced wear characteristics when compared to the original chrome plated Alloy 600 AVB material. [redacted] W Prop.

2.2.1.11 Tube-to-Tubesheet Juncture

The new steam generator tubes will be inserted into the tubesheet holes, tube leg end hydraulic tack expanded, tube end welded, followed by a helium leak test. The tube leg ends are then hydraulically expanded for essentially the full thickness of the tubesheet. This process will provide better tube expansion control.

The tube ends in the new lower assemblies will be flush with the tube hole opening when welded to the tubesheet cladding. By eliminating the protruding tube ends and tube fillet welds of the original design, entry pressure losses are reduced, resulting in a lower pressure drop in the primary loop. Flush welding of the tube ends also minimizes possible locations for crud buildup. This loss of crud buildup locations is expected to reduce radiation levels within the channel head.

2.2.1.12 Tubesheet [redacted] Primary and Secondary Sides

The tubesheet will be of the same dimensions as the original tubesheet, [redacted] The tubesheet will be marked on a row by column pattern to facilitate tube identification during primary side maintenance activities. W Prop

1. Design Change Title/Station/Unit

STEAM GENERATOR REPLACEMENT / NORTH ANNA / UNIT 1

2. Design Change Number

DC 90-13-1

9. Engineering Review and Design:

[redacted]

W Prop.

The tubesheet forging material has been changed from SA-508, Class 2 to SA-508, Class 3a. [redacted]

W Prop.

2.2.1.13 Round Wrapper with Jacking Stud Support System

The original steam generators achieve lateral support of the tube bundle wrapper by using tube support plate wedges to force contact between the wrapper lateral supports and the inside surface of the steam generator shell. The new steam generators will use a method wherein [redacted] in the wrapper until they develop contact with the steam generator shell.

W Prop.

[redacted]

W Prop

2.2.1.14 Primary Nozzle Closure Rings

The new steam generators will be supplied with primary nozzle closure rings designed to interface with Westinghouse-designed primary nozzle closure covers (nozzle dams). The closure rings are permanently welded to the inside surface of the steam generator channel head, concentric with each primary nozzle. The nozzle closure covers are installed, as required, prior to maintenance activities within the channel head, and are removed prior to replacement of the channel head manway covers.

The additional primary nozzle closure rings will provide an option/backup to loop stop valve closure which will allow primary side maintenance tasks to be conducted simultaneously with reactor vessel refueling outage activities that require a flooded cavity.

The primary nozzle closure ring around each of the channel head bowl nozzle openings has no impact (i.e., no measurable pressure drop increase or flow rate decrease) due to its presence; nor does it prevent ultrasonic in-service inspection of the primary nozzle internal corner junction from outside the head.

2.2.1.15 Forged Primary Nozzle Safe-Ends

Forged stainless steel safe ends will be welded onto the primary nozzles to

1. Design Change Title/Station/Unit

STEAM GENERATOR REPLACEMENT / NORTH ANNA / UNIT 1

2. Design Change Number

DC 90-13-1

9. Engineering Review and Design:

provide for improved weld compatibility and field fit-up with the reactor coolant piping.

The primary side safe-end material has been changed from a stainless steel type 309L/308L weld deposited buildup tc an ASME SA-336, Class F 316 LN forging. The safe-end forging is provided with up to 2 inches of additional material, without end weld prep, which can be machined at the site to a length suitable for fit-up to the reactor coolant system piping.

2.2.2 Other Changes and Enhancements

2.2.2.1 Steam Generator Upper Domes

A flow limiter [redacted] will be installed in the main steam nozzle of the steam dome in accordance with Procedure STD-FP-1991-5563 (see Appendix 4-16) and the Special Processes Manual (Reference 6.5). This flow limiter is a design enhancement that reduces the pressure drop across the SG internal components during a postulated main steam line break and reduces the rate of energy released to the containment for this postulated accident. The original steam generators do not have this device. w Prop.

The purpose of the existing downcomer flow resistance plates was to reduce the flow-induced stresses on some of the tubes by creating better flow characteristics in the downcomer annulus. The re-designed steam generator lower assemblies eliminate the need for the downcomer flow resistance plates. Therefore, these plates will be removed in accordance with drawing N-9013-1-M-405.

Disconnection of the steam domes from the old lower assemblies and their reconnection to the new lower assemblies will be performed in accordance with drawing N-9013-1-M-405 and includes the following activities:

- In order to obtain access to the I.D. transition cone girth weld, two access ways will be cut in the swirl vane transition wrapper.
- Prior to severance of the transition cone and lifting of the steam dome, the primary moisture separator (PMS) will be secured to allow the PMS to be rigged with the steam dome.
- Severance of both the steam generator shell and the wrapper plate will be accomplished by using a track-mounted, tractor-driven, oxygen-fuel cutting torch. Two circumferential cuts will be made in the SG shell resulting in a band of shell material that will be removed as part of the construction effort. Removal of this band will provide an access for the cut to be performed on the wrapper plate.
- The 3" x 6" x 1/2" support bars located between the wrapper and

steam separator assembly will be removed by grinding existing welds. If grinding proves not to be a viable means of removing the support bars, an access hole will be cut in the wrapper using a torch, and the support bars removed. If this method becomes necessary, reinstallation will require a patch plate be applied to the wrapper to return it to its original condition. Prior to removal of the support bars, as-built details of the gap measurements will be taken for reinstallation to the existing arrangement.

- The wrapper plate will be prepared by welding the fit-up ring with its top horizontal backing bar on the existing transition barrel. The bottom horizontal backing bar will be welded on the landing ring of the new lower assembly outside the containment.

- Once in position for welding, the transition cone shell will be preheated and the joint welded using the shielded metal arc welding process and post-weld heat treated.

- The wrapper plate will be joined by welding new radial plates on the horizontal backing bars. Vertical backing bars will be used between the radial plates.

- The support bars will be reinstalled using the as-built details and drawing N-9013-1-M-405.

- After secondary side inspection, the access ways in the swirl vane transition wrapper will be welded shut and the secondary manways will be closed.

Weld preparation, welding, and NDE will be in accordance with the Special Processes Manual, Appendix 4-21, and drawing N-9013-1-M-405, as applicable. All work performed on the steam generators is classified as safety-related.

2.2.2.2 Reactor Coolant System Piping

To permit removal and re-installation of the SG lower assemblies, the RCS hot and crossover leg piping will be cut at the steam generator nozzles as shown on drawing N-9013-1-M-401. In the event of unacceptable fit-up per the ASME Code, provisions have also been included to allow for removal of the RCS crossover leg elbow adjacent to the steam generator by making an additional cut. Weld end preparation, welding, and NDE will be in accordance with the Special Processes Manual, NAS-1009, drawing N-9013-1-M-401, and Appendix 4-21, as applicable. All work performed on the RCS piping is classified as safety-related.

The reactor coolant system hot and crossover leg pipe ends remaining after

1. Design Change Title/Station/Unit	2. Design Change Number
STEAM GENERATOR REPLACEMENT / NORTH ANNA / UNIT 1	DC 90-13-1

9. Engineering Review and Design:

removal of the Sgs will be decontaminated to reduce radiation levels in the area of the SG replacement activities. See Appendix 4-15 for Specification 21809-M-001 for procurement of decontamination services.

2.2.2.3 Main Steam Piping

As shown on drawing N-9013-1-M-402, the main steam piping will be cut at the centerline of the existing weld at the SG nozzle and on the horizontal straight section of pipe approximately 10' from the existing weld at the main steam elbow. In the event of difficulties involved with the fit-up of the main steam piping, a new spool piece section, procured in accordance with the requirements of Specification NAS-1009, is available. The temporary main steam piping configuration has been evaluated in Calculation 02072.13-NP(B)-009-XE (Reference 6.12.22) to document the acceptability of this interim system configuration. Weld end preparation, welding, and NDE will be in accordance with the Special Processes Manual, NAS-1009, and Appendix 4-21, as applicable. All work performed on the main steam piping is classified as safety-related. Material reconciliations are provided in Appendix 4-33.

2.2.2.4 Feedwater Piping

Each of the existing feedwater piping loop seals from SG nozzle up to and including the fourth elbow will be replaced with the new chrome-moly steel (piping class 601C) material to provide better erosion/corrosion resistance. The loop drain valves will also be replaced with new valves with live load packing. The piping materials and the valves will meet the requirements of Specifications NAS-1009 and NAP-0023, respectively. The modifications are shown on drawing N-9013-1-M-403. Temporary feedwater piping configuration have been evaluated in Calculation 02072.13-NP(B)-009-XE (Reference 6.12.22) to document the acceptability of this interim system configuration. Weld end preparation, welding, and NDE will be in accordance with the Special Processes Manual, NAS-1009, and Appendix 4-21, as applicable. All work performed on the feedwater piping is classified as safety-related. Material reconciliations are provided in Appendix 4-33.

The line designation tables for the feedwater system have been marked-up to reflect the material changes to the piping (see Appendix 4-30).

2.2.2.5 Blowdown and Steam Generator Shell Drain Piping

The blowdown and shell drain piping will be cut at the SG socket weld nipples and at the 3-inch header, as shown on drawing N-9013-1-M-404, sheet 1. Prior to cutting, the piping will be properly supported, including the use of temporary supports, if required. The temporary steam generator blowdown piping configurations have been evaluated in Calculation 02072.13-NP(B)-009-XE (Reference 6.12.22) to document the acceptability of the interim system configurations. The cut sections of piping, including the manual valves, will be removed, discarded, and replaced with new chrome-moly steel material (piping class 601C) to provide better erosion/corrosion protection. The new sections will be fabricated per drawings N-9013-1-M-601, 602, and 603 from materials procured by Specifications NAP-0017 and NAP-0023 and meeting the requirements of Specification NAS-1009 for reinstallation to the new Sgs. New/modified hangers will be installed in accordance with the drawings included in the Drawing Revision Record. Weld end preparations, welding, and NDE will be performed in accordance with the Special Processes Manual, NAS-1009, and Appendix 4-21, as applicable. An exception to NAS-1009 has been approved to allow welding pipe saddles to the blowdown piping (Appendix 4-34). Material reconciliations are provided in Appendix 4-33.

The new steam generator lower assemblies are provided with 2½" nozzle couplings for the blowdown piping instead of the 2" socket weld connection on the existing steam generators. The pipe size will be increased from 2 in. to 2½ in. to match the size of the nozzle on the new replacement SG lower assembly.

All work performed on the blowdown and shell drain piping is classified as safety-related.

The line designation tables for the blowdown system have been marked-up to reflect the size and material changes to the piping (see Appendix 4-30).

2.2.2.6 Sample System Piping

The sample system piping will be cut at the SG socket weld nipples as shown on drawing N-9013-1-M-404. The second cut location will be as shown on drawing N-9013-1-M-404 to provide adequate clearance for SG removal activities. Since the cut sections of piping will be removed and discarded, new sections will be fabricated from materials meeting the requirements of Specification NAS-1009 for reinstallation to the new Sgs. The replacement sections of piping will be of identical size and configuration as the removed sections and the material will meet the requirements of NAS-1009. The weld end preparations, welding, and NDE will be in accordance with the Special Processes Manual, NAS-1009, and Appendix 4-21, as applicable. All work performed on the sample system piping is classified as safety-related. Material reconciliations are provided in Appendix 4-33.

2.2.2.7 Chemical Feed Tubing

The chemical feed tubing will be disconnected at the feedwater loop seal to provide clearance to make the second cut on the feedwater pipe and to remove and replace the feedwater loop piping. See drawings N-9013-1-M-604 through 606. The second cut location will be made in accordance with drawings N-9013-1-M-604 through 606 to provide adequate clearance for SG removal activities. The chemical feed tubing will be disconnected from the feedwater pipe by cutting upstream of the manual isolation valves 1-WT-39, 51, and 67.

Existing manual valves and removed sections of pipe will be saved and reinstalled, if possible. Some of the tubing supports may be disconnected to provide adequate flexibility. Any supports that are disconnected will be reconnected in accordance with drawings N-9013-1-M-604 through 606. The replacement sections of tubing will be of identical size, material (ICN9, see DCP 80-S-82), and configuration as the removed section. Weld end preparation, welding, and NDE will be in accordance with the Special Processes Manual, NAS-1009, and Appendix 4-21, as applicable. The modification is shown on drawing N-9013-1-M-403. All work performed on the chemical feed tubing is classified as safety-related.

2.2.2.8 Wet Layup System Piping

The wet layup system piping will be disconnected at the steam generator flange in accordance with drawings N-9013-1-M-607, 608, and 609. The second cut location will be in accordance with drawings N-9013-1-M-607 through 609 to provide adequate clearance for SG removal activities. Any supports that are disconnected will be reconnected in accordance with drawings N-9013-1-M-607 through 609. Weld end preparation, welding, and NDE will be in accordance with the Special Processes Manual, NAS-1009, and Appendix 4-21, as applicable. All work performed on the wet layup system piping is classified as safety-related. Material reconciliations are provided in Appendix 4-33.

2.2.2.9 Steam Generator Level Instrumentation

To provide clearance for the removal and reinstallation of the SGs, the instrument tubing, condensate pots, root/vent valve assemblies, unistrut stanchions, and supports associated with the wide range and narrow range SG level sensing lines will be disconnected from the SGs and removed. These components are located in the immediate vicinity of the SGs, from elevation 259'-11" to 315'-0". Refer to drawings N-9013-1-FK1A and N-9013-1-FK1B for specific line and mark numbers.

Piping and tubing will be removed and either reused or discarded. If required, new materials will be procured per Specifications NAI-0001 and NAP-0023 and will meet the requirements of NAS-1009 as applicable. The existing carbon steel condensate pots, root valves and vent valves will be discarded and stainless steel replacements will be installed. Refer to drawing N-9013-1-M-801 for fabrication details of the replacement condensate pots. Refer to drawings N-9013-1-M-801, N-9013-1-FK1A and N-9013-1-FK1B for removal and reinstallation details.

Following the installation of the new SGs, all tubing, piping, valves, and supports will be installed to the original pre-SG replacement configuration. Lower root valve assemblies and condensate pot assemblies will be socket welded to the appropriate tap connections. Stanchions and supports will be attached to the insulation support rings as in the original configuration. The tubing and tubing supports will be reinstalled to meet plant specific seismic requirements per Specification NAI-0001.

Weld end preparation, welding, and NDE will be in accordance with the Special Processes Manual, NAS-1009, and Appendix 4-21, as applicable. All work performed on the steam generator level instrumentation is classified as safety-related. Material reconciliations are provided in Appendix 4-33.

2.2.2.10 Vibration and Loose Parts Monitoring System

The existing vibration and loose parts monitor assemblies will be removed from the old SG and reinstalled at different locations on the new steam generators as prescribed by Westinghouse in the Steam Generator Technical Manual (Reference 6.2). Two accelerometers are used on each steam generator. Each of the new steam generators will be provided with four sensor mounting holes (two each, at elevations 261'- 7-1/2" and 266'- 7-1/2"). The non-scheduled conduit required to complete installation of the accelerometers will be replaced, if necessary, in accordance with drawings N-9013-1-1FE57E, N-9013-1-1FE57F, and N-9013-1-1FE3HY. All work associated with the vibration and loose parts monitoring system is classified as non-safety related with special regulatory significance (NSQ).

2.2.2.11 Steam Generator Upper Restraints

The steam generator upper lateral restraints will be replaced with equivalent restraints procured in accordance with Specification NAP-0033. Demolition of the existing upper restraints is required due to difficulties associated with their removal from the steam generators and the effort involved to reuse them. Refer to drawings N-9013-1-1FV17L, M, N, and P and N-9013-1-S-018 for the new upper restraint design and installation. All work associated with the removal and installation of the permanent upper lateral restraints is classified as safety-related. Material reconciliations are provided in Appendix 4-33.

Temporary supports will be installed to facilitate installation of the new upper restraints. The temporary support arrangement does not result in any unacceptable loads being placed upon the embedded plate or clevis. See Calculation 20559-C131-08 (Reference 6.12.13).

2.2.2.12 Steam Generator Lower Supports

As part of the SGR, various components (e.g., vertical support plates, shims, socket head cap screws, hex nuts, etc.) associated with the steam generator lower supports will be removed and replaced per drawing N-9013-1-1FV17G. Material reconciliations are provided in Appendix 4-33.

2.2.2.13 Steam Generator Thermal Insulation

The existing thermal insulation on the steam generators will be replaced with new fiberglass blanket type insulation. The replacement blanket insulation will meet or exceed the design requirements of the existing insulation. The insulation will be procured by Specification NAP-0047 and will be installed in accordance with the Vendor Technical Manual (Reference 6.3). Removable panels will be designed to facilitate inservice inspection. The new insulation is seismically designed to ensure that the insulation remains in place in the event of a seismic occurrence. All work associated with the thermal insulation is classified as non-safety related with special regulatory significance (NSQ).

An insulation debris analysis has been performed (Appendix 4-28). This analysis confirmed that longterm blockage of the sump suction screens will not occur and that adequate NPSH margin is available for the low head safety injection pumps and the inside and outside recirculation spray pumps. As a result of the NPSH analysis, the setpoint for manual RWST switchover to recirculation mode will change to 22.8% RWST level (see Appendices 4-7, 4-23, and 4-28).

2.2.2.14 Optical Templating Bracket Removal

Optical templating brackets were installed as permanent plant equipment during the implementation of DC 92-006-1. These brackets will be removed at the end of the SGR outage under this DC by unbolting them. The expansion anchor bolts will be cut off flush to the surface of the wall and then driven into the wall slightly to allow for smooth finish grouting of the anchor bolt holes. Refer to drawing N-9013-1-1FM1C. Grouting of the holes following removal of the optical templating brackets will be performed in accordance with NAI-0014 and is classified as safety-related.

2.2.3 <u>Temporary Modifications</u>

2.2.3.1 Steam Generator Haul Routes

Transportation of the original and replacement steam generator lower assemblies and a load test of the haul route are required to facilitate steam generator replacement. An evaluation of the steam generator haul routes has been performed and is provided in Appendix 4-18. The objective of the haul route inspection and evaluations was to ensure that the grades and turning radii of the haul route can accommodate the transporter/prime mover and load test, that slopes and embankments will remain stable when subjected to the transporter/prime mover wheel loads, that the right-of-way has sufficient strength to support the transporter/prime mover wheel loads without excessive settlement, and that buried utilities and facilities can withstand the overburden pressures and remain within their allowable load limits. Road improvements and requirements for protection of buried utilities are provided in Appendix 4-

1. Design Change Title/Station/Unit

STEAM GENERATOR REPLACEMENT / NORTH ANNA / UNIT 1

2. Design Change Number

DC 90-13-1

9. Engineering Review and Design:

18. Limitations on transport of the new and old steam generator lower assemblies and the load test include:

- The maximum transporter speed is 5 mph.
- The transporter leveling capability will be used to maintain the transporter bed level within 5° during movement.
- In accordance with N-9013-1-M-800, the transition cone cover plate "T-sections" will be installed before the old SG lower assemblies are moved outside the protected area.
- The centerline of the steam generator will be limited to a maximum of 12'-3" off the ground.

In addition, modification of the west end security fence will provide the transporter a passage through the security fence. This modification will be accomplished in accordance with DC 92-01-3, West End Security Access Facility.

2.2.3.2 Temporary Containment HVAC Modifications for Ventilation and Smoke Removal

The containment purge exhaust system will be temporarily modified as shown on drawing N-9013-1-M-406 to allow for smoke removal from the containment dome area. The majority of the smoke generated due to the cutting and welding of the SG upper dome section and the main steam and feedwater piping will be collected locally with portable vent/filter equipment. The smoke that is not collected locally will tend to collect in the containment dome area. If not removed, the smoke will reduce visibility and air quality. Therefore, the blind flange on the 18" diameter manhole (inspection port) located upstream of the containment penetration isolation valve on the purge exhaust (return) duct, will be removed and stored. A manual volume control damper and a 90 degree elbow will be installed at the open manhole. An 18" diameter flexible duct will be connected to the other end of the elbow. The flexible duct will be run vertically up along the containment wall to elevation 345' 0" (approximately). The temporary volume control damper will be adjusted to limit the suction pressure in the flexible duct to maximum of -1" W. G. in order to prevent the duct from collapsing. It is estimated that up to 5,000 cfm can be expected through the 18" diameter flexible duct at -1" W. G. differential pressure.

All temporary HVAC modification work covered in this section is non-safety related with the exception of the flexible duct attachment to the safety-related containment purge system. Temporary supports will be provided to ensure safety-related ducts are not affected as a result of the temporary change. The containment purge system will be restored to its original pre-outage condition upon completion of SGR work.

1. Design Change Title/Station/Unit

STEAM GENERATOR REPLACEMENT / NORTH ANNA / UNIT 1

2. Design Change Number

DC 90-13-1

9. Engineering Review and Design:

2.2.3.3 Security Door Modifications

To facilitate the installation of temporary air chiller units to support the SGR, the security door (A-95-1) located at the entrance to the MCC room roof will be temporarily removed and replaced with a modified security door which will allow passage of air hoses/pipes from the roof down to the containment personnel hatch. The modified security door will meet all requirements of the North Anna Security Plan and will be removed and replaced with the original security door upon completion of the SGR work. Site security personnel will post a security guard at the door while the modified door is being installed and again when the original door is being replaced.

All work associated with these temporary modifications is classified as non-safety related with special regulatory significance (NSQ).

2.2.3.4 Temporary Construction Power

A construction power distribution system will be temporarily installed to support the in-containment SGR work activities. The power feeder to RCP-1B will be utilized to provide power to the primary side of the steam generator replacement dry type transformer. The power feeder to RCP-1B will be disconnected at the motors main terminal box and temporary feeder cables will be spliced directly to that feeder. Refer to Drawings N-9013-1-1FE1B and N-9013-1-1FE8N.

The temporary construction power demand will be less than half the RCP-1B motor load. The RCP-1B feeder is therefore acceptable for the power distribution to the SGR electrical loads. However, three General Electric IAC53B805A relays will be temporarily installed in the Unit 1 RCP-1B feeder breaker 15B3 to provide the temporary construction transformer overload protection as well as three-phase and phase-to-phase 480 v fault protection. These relays will be set in accordance with Calculation EE-0490 (Reference 6.12.101). Calculation 20559-SGR-E-001 (Reference 6.12.14) determines the SGR electrical loads.

RCP-1B feeder breaker 15B3 will be temporarily modified to jumper out pump motor protective interlocks. Installation and removal of jumpers are required to accomplish the modification and will be performed by Virginia Power. Installation of the power feeder splice and the jumpers will either be in accordance with North Anna administrative procedure ADM-14.1 "Jumpers (temporary modifications)" (VPAP-1403 "Temporary Modifications") or temporary procedures will be written and approved to perform the modification. Refer to drawings N-9013-1-1ESK5AJ, N-9013-1-1FE8N, and N-9013-1-1FE1B for details.

All temporary power activities covered in this section are non-safety related and are installed non-seismic. All temporary equipment and cables will be removed upon completion of the SGR work.

2.2.3.5 Reactor Cavity Cover

Steel decking will be installed over the open sections of the reactor refueling cavity after all fuel is removed from the containment, the reactor cavity drained, and the fuel transfer tube gate valve closed and blind flange installed. The reactor cavity cover will provide additional construction laydown area and prevent materials from entering the reactor refueling cavity. The reactor cavity cover is not required to support the rigging of the steam generator lower assemblies. In addition, the reactor cavity cover has been designed to have no adverse impact on the existing EL. 291'-10" floor slab or the manipulator crane rail. See Calculation 20559-C110-02 (Reference 6.12.7). All work associated with the reactor cavity cover is non-safety related. The reactor cavity cover will be removed prior to fuel load.

2.2.3.6 Auxiliary Crane

A non-safety related auxiliary crane will be installed in the containment on the El. 291'-10" operating floor slab to support the handling of loads during the steam generator replacement outage. The support tower for the crane will be attached to the concrete slab using anchor bolts that were installed under design change package DC 90-15-1. Loads on the El. 291'-10" floor slab associated with the auxiliary crane have been evaluated and found to be acceptable. See Calculation 20559-C109-01 (Reference 6.12.5). The auxiliary crane and support tower will not be installed until all fuel has been removed from the containment, the reactor cavity drained, and the fuel transfer tube gate valve closed and blind flange installed. The auxiliary crane and support tower will be removed prior to plant refueling.

2.2.3.7 Jib Crane

A 3-ton capacity jib crane will be temporarily installed inside the containment for use during defueling and refueling operations. As per the guidelines of NUREG-0612, physical limitations will be placed on the swing radius of the crane to ensure approved load paths are followed as designated in the station heavy loads procedure 0-MCM-1303-01 and that no adverse interaction with spent fuel or fuel handling activities will occur. This crane will be replaced, in the same physical location, by the auxiliary crane when defueling is completed. The jib crane will be reinstalled at the end of the outage to assist in containment demobilization.

2.2.4 Stress Analysis

Stress analysis has been performed for components affected by the steam generator replacement project. These components include:

- Reactor coolant loop piping
- Primary system branch piping and associated supports including the charging, letdown, loop fill, loop drain, cold leg stop valve, pressurizer spray and surge, safety injection and residual heat removal systems
- Secondary system branch piping and associated supports including main steam, feedwater, steam generator blowdown, chemical feed and wet layup systems
- Sample and steam generator instrumentation tubing systems
- Upper steam generator support
- Lower steam generator support
- Reactor coolant pump support

These component qualifications are documented in Calculations (References 6.12.1 through 6.12.4 and 6.12.15 through 6.12.88).

In order to avoid reanalysis of affected piping, the existing arrangement/layout and locations of pipe supports will not be changed beyond tolerances shown on applicable drawings. Cold gap measurements for critical components (as listed in Appendix 4-27) will be obtained before piping/support removal and again following reinstallation of piping. The selection of components including the basis for cold gap measurement verification is documented in Calculation 02072.22-NP(B)-001-X (Reference 6.12.88). These cold gap measurements will be verified to be in accordance with design requirements. During this time, any requirements for specific hot gap measurements will be determined by Engineering. Piping system cold condition is considered to be less than 150°F.

2.2.5 UFSAR Chapter 15 Accident Analysis

Technical Report NE-883, "Safety Analyses and Evaluations Supporting North Anna 1 Operation Following Steam Generator Replacement," documents the safety analysis evaluations performed for the steam generator replacement effort (see Appendix 4-23). The scope of the assessment involved the Chapter 15 UFSAR events and the Chapter 6 mass and energy and containment response analysis.

NE-883 evaluates the performance of the replacement steam generators under postulated accident conditions. Two bounding transients (Loss of Load and Excessive Load Increase) were analyzed to compare the heat transfer characteristics of the existing and replacement steam generators under transient conditions. The analysis examined seven

1. Design Change Title/Station/Unit	2. Design Change Number
STEAM GENERATOR REPLACEMENT / NORTH ANNA / UNIT 1	DC 90-13-1

9. Engineering Review and Design:

transient parameters and demonstrated that there is an insignificant difference in transient behavior between the Model 51 and 51F steam generator designs. NE-883 supports the general conclusion that the Model 51F steam generator may be considered a "replacement" component for the Model 51 steam generator from a safety analysis perspective.

For the purpose of the steam generator replacement 10 CFR 50.59 evaluation, each UFSAR Chapter 15 accident was categorized according to how the applicable transient analysis supports the basis for operation following the steam generator replacement. In most cases, the licensing basis analysis following replacement will be that which supported operation prior to the N1C9 restart. In some cases, however, the N1C9 restart reanalysis (extended steam generator tube plugging) is retained as the licensing basis analysis. For some other events (e.g., LOCA containment response), the appropriate analysis is incorporated into the North Anna analysis basis via the current steam generator replacement evaluation. For each of the accident analyses, it was concluded that all acceptance criteria would continue to be met for operation following steam generator replacement and that the UFSAR conclusions for each event would remain valid.

The analyses evaluated included mass and energy releases for the main steam line break accident and the large break LOCA. The smaller break area of the steam nozzle integral flow restrictors limits the rate at which mass and energy enter the containment. In addition, there is a small reduction in the secondary steam generator mass that, although unquantified, provides an additional margin. Therefore, the current containment analyses for the main steam line break remain bounding.

Initial assessments were performed to determine the effects of SG replacement upon the existing LOCA containment integrity and NPSH analyses. The results indicated that while there was margin to the acceptance criterion for each key parameter, certain analysis margins were decreased with respect to that in the existing analysis. In no case was the margin of safety, identified by the acceptance criteria, reduced. It was concluded, however, that the best means to confirm the acceptability of analysis margins for key containment analysis parameters, while implementing the SG replacement via the 10 CFR 50.59 process, was to use the Westinghouse mass and energy release model to reanalyze the LOCA containment releases. The Westinghouse models have been reviewed and approved by the NRC for use on PWRs with dry containment designs, such as North Anna.

The revised mass and energy release data were used in the containment response analysis with the Stone and Webster LOCTIC computer code. Analyses were performed to obtain limiting results using revised data for each of these key parameters: containment peak pressure, containment depressurization time and subatmospheric peak pressure, available NPSH (for low head safety injection, inside and outside recirculation spray pumps). Acceptable analysis margins were confirmed for each key analysis parameter.

The steam generator replacement involves specific physical plant changes which have effects on plant operation and are evaluated in Appendix 4-23:

1. Design Change Title/Station/Unit

STEAM GENERATOR REPLACEMENT / NORTH ANNA / UNIT 1

2. Design Change Number

DC 90-13-1

9. Engineering Review and Design:

- Confirmation of Reactor Protection System setpoints, Emergency Safety Features Actuation System (ESFAS) setpoints, and Technical Specifications values.
- Assessment of increased RCS flow rate
- Evaluation of steam generator replacement impact on emergency operating procedures and boration/dilution nomographs.
- Recommended changes to periodic test procedures and NCRODPs.

2.2.6 Operational Changes

The new SGs will be operated in the same manner as the existing SGs. The new SGs will have slightly different operating parameters which have been reviewed with respect to all of the applicable instrument recalibrations and procedures (see section 3.19 and the Controlled Document Summary). When all of the adjustments have been made to the instrumentation, setpoints and procedures, the SGs will function in the same manner as the existing SGs.

2.2.7 Other Technical Issues

2.2.7.1 Steam Generator Rigging Loads

A special over rated-load lift qualification for the polar crane was performed in accordance with ASME B30.2, Section 2-3.2.1.1, "Special Over Rated-Load Lifts," which concluded that the existing polar crane and the crane supporting structure have sufficient capacity to handle a 280-ton special over rated-load lift (see Appendix 4-31). The 280-ton capacity is greater than the maximum load (including rigging) to be handled by the polar crane during the SGR. See Calculation 21809-C-05 (Reference 6.12.12).

The capability of the 291'-10" floor slab to withstand the loads associated with rigging and handling the steam generators has been confirmed. See Calculation 20559-C110-01 (Reference 6.12.6).

The capability of the 291'-10" floor slab to support the steam domes during the SGR has also been confirmed. See Calculation 20559-C131-04 (Reference 6.12.10).

2.2.7.2 Drop Analyses

Movement of heavy loads inside the containment will be in accordance with the station heavy loads procedure 0-MCM-1303-01 until all fuel has been removed from the containment, the reactor cavity drained, and the fuel transfer tube gate valve closed and the blind flange installed. During the defueled condition, movement of heavy loads will be in accordance with approved

1. Design Change Title/Station/Unit

STEAM GENERATOR REPLACEMENT / NORTH ANNA / UNIT 1

2. Design Change Number

DC 90-13-1

9. Engineering Review and Design:

project procedures.

After defueling is completed, those structures, systems, and components inside the containment which could be impacted by a heavy load drop will not be required to perform a safety function at that time. To preclude any possible adverse effects on stored spent fuel or systems shared with Unit 2 as a result of a postulated steam generator drop, all component cooling water, service water, fuel pit cooling and refueling purification system, and instrument air lines to/from the containment will be isolated during rigging operations for the steam domes and lower assemblies. Thus, no adverse impacts on stored spent fuel or spent fuel cooling would result from a postulated heavy load drop inside the containment.

Prior to removing the old steam generator lower assemblies from the containment for storage, the lower assemblies will be drained and all openings sealed to prevent any leakage of radioactive contamination (see drawing N-9013-1-M-800). In addition, a coating of A-B-C Encapsulant will be applied to the exterior of the lower assemblies to prevent the spread of any surface contamination that may exist.

Coincident with equipment hatch removal, containment purge system alignment and operation will be in accordance with Operating Procedure 21.2 so as to minimize any air leakage out of the containment. In addition to the vent stack monitors and the containment area radiation monitors, a continuous air monitor will be in use adjacent to the equipment hatch and periodic air sampling will be performed. A temporary cover will also be available to isolate the hatch opening in case of loss of the containment purge exhaust system. For these reasons, there is a negligible potential for any unmonitored leakage out of the equipment hatch. This notwithstanding, the radiological consequences of a postulated drop of an old steam generator lower assembly inside the containment, during transfer from the equipment hatch platform to the transporter, or during movement within the protected area have been evaluated (see Calculation 21809-M-02 which is Reference 6.12.9) and determined to be within applicable regulatory guidelines and less than the limiting case events of the same class of accidents currently evaluated in the UFSAR.

A postulated drop of a steam generator lower assembly adjacent to the equipment hatch could result in damage to the equipment hatch or locally to the containment structure, but significant structural damage to the containment would not occur. Moreover, containment leaktight integrity is not required at this time. Further, no safety-related equipment could be impacted by a postulated drop at this location.

During transport of the new and old lower assemblies, the centerline of the steam generator lower assembly will be limited to a maximum of 12'-3" above grade. The closure plates for the old steam generator lower assemblies have been designed to withstand a load drop from the transporter. See Calculation

21809-C-02 (Reference 6.12.11). This ensures that in the event of a drop of the old steam generator from the transporter during relocation to the steam generator storage facility or during offload at the storage facility, the integrity of the steam generator shell will not be breached. No buried or adjacent facilities will be adversely impacted as a result of a postulated lower assembly or test load drop as concluded in Appendix 4-18. Road improvements and requirements for protection of buried utilities are provided in Appendix 4-18.

2.2.7.3 Timing of Piping Cuts

The reactor coolant loops A, B, and C will have temporary shielding installed on the hot leg, cold leg, crossover leg, and bypass line prior to the RCS piping cuts. RC Loop B will be severed first, prior to the completion of fuel offload. This configuration will exist with either the hot or crossover leg severed, or with both lines severed. The acceptability of these configurations is documented in Reference 6.12.21. The loop isolation valves will be isolated prior to making any RCS piping cuts. Cutting of the Loop A and C reactor coolant piping will not be performed until all fuel has been removed from the containment, the reactor cavity drained, and the fuel transfer tube gate valve closed and blind flange installed.

The acceptability of the RC loops A, B, and C piping severed from the steam generators after fuel is removed from the reactor is also documented in Reference 6.12.21.

Secondary plant piping systems including main steam, feedwater, feedwater drain, wet layup, steam generator blowdown, chemical feed, sampling, and steam generator level instrumentation is also planned to be severed from the steam generators while fuel is in the reactor vessel. Prior to severance, an appropriate support configuration will be established that ensures no adverse effects, including gravity missiles and sway interaction concerns, will result to any safety-related components. Potential loads considered include deadweight and seismic effects. This evaluation is documented in Reference 6.12.22.

2.3 SPECIAL IMPLEMENTATION REQUIREMENTS

2.3.1 General Description of Activities to be Performed

Implementation of the steam generator replacement effort will initially depend upon the removal of interferences, the connection of temporary power supplies and the installation of temporary runway beams and a reactor cavity cover. All piping connections to the steam domes will be severed, including main steam, feedwater, level instrumentation, and sample system lines. The steam domes will be separated from the lower assembly, lifted with the polar crane, and stored in stands supported by the containment floor.

All piping connections to the lower assemblies will be severed, including RCS, blowdown, drains, and level instrumentation. After removal of the lateral restraints, the lower

1. Design Change Title/Station/Unit	2. Design Change Number
STEAM GENERATOR REPLACEMENT / NORTH ANNA / UNIT 1	DC 90-13-1

9. Engineering Review and Design:

assembly will be rigged to elevation 291'-10". The lower assembly will be pivoted from a vertical to a horizontal position and lowered onto the transfer carriage. The transfer carriage moves the lower assembly out of the containment through the equipment hatch, onto the equipment hatch platform, where it is lowered onto the transporter and ultimately transported to the steam generator storage facility.

After removing the old lower assemblies, the new lower assemblies will be transported to the containment and installed in the reverse order sequence of that described above. While the piping and tubing is being reconnected, the steam dome (with steam flow limiter installed) will be reinstalled. Piping will be reconnected to the steam dome, new thermal insulation will be installed and all temporary installations can be removed.

Work activities covered by this DCP will primarily be performed inside the containment. Outside containment activities include preparatory work and the temporary modifications to the security door, RCP-1B feeder breaker, and the haul route.

All construction procedures are being generated in accordance with the Project Manual.

2.3.2 Safety Classification

The steam generator components and supports installed and/or modified by this design change are safety-related.

Activities associated with the removal and replacement of the reactor coolant system piping, main steam piping, feedwater piping, steam generator blowdown shell and drain piping, sample system piping, chemical feed piping, wet layup piping, and the steam generator level instrumentation tubing are safety-related. Decontamination of the RCS is safety-related.

Activities associated with the removal and replacement of insulation and the vibration and loose parts monitoring system are non-safety related with special regulatory significance (NSQ). The new insulation is seismically designed to ensure that the insulation remains in place in the event of a seismic occurrence.

All activities associated with transport of the steam generator lower assemblies and preparation of the haul route, including road improvements and requirements for protection, are non-safety related.

All temporary HVAC modifications are non-safety related with the exception of the flexible duct attachment to the safety-related containment purge system.

All work associated with modifying the security room door is non-safety related with special regulatory significance (NSQ).

All temporary power modifications, the reactor cavity cover, and the auxiliary and jib cranes are non-safety related.

1. Design Change Title/Station/Unit	2. Design Change Number
STEAM GENERATOR REPLACEMENT / NORTH ANNA / UNIT 1	DC 90-13-1

9. Engineering Review and Design:

Grouting of the holes following removal of the optical templating brackets is safety-related.

2.3.3 Overall Implementation Requirements

Special provisions, precautions, and/or inspections that are to be taken during the implementation of this DCP are as follows:

a. Station Tagging Procedures, Administrative Procedure VPAP-1402, shall be observed at all times. No work shall be performed on systems until they are released by Operations.

b. General in-containment preparatory activities may be performed as soon as the plant has been depressurized and while defueling/refueling is in progress.

c. During fuel movement, appropriate precautions shall be taken to ensure no debris enters the reactor vessel as a result of SGR activities and that defueling/refueling operations are not impacted. While fuel is in the containment, materials and tools associated with the SGR shall not be handled or transported within the refueling barrier.

d. Prior to defueling, while loading temporary equipment into containment, any items that penetrate the equipment hatch shall be installed such that the hatch can be closed within 4 hours in accordance with station procedures. During core alterations or during movement of irradiated fuel, the hatch shall remain closed.

e. Movement of heavy loads inside the containment shall be in accordance with the station heavy loads procedure 0-MCM-1303-01 until all fuel has been removed from the containment, the reactor cavity drained, and the fuel transfer tube gate valve closed and blind flange installed. During the defueled condition, movement of heavy loads shall be in accordance with approved project procedures.

f. After defueling, the reactor upper internals shall be placed back into the reactor vessel and the reactor vessel head shall be placed back on the reactor vessel for shielding and storage of both the internals and the vessel head during the SGR outage.

g. Coincident with equipment hatch removal, containment purge system alignment and operation shall be in accordance with Operating Procedure 21.2 so as to minimize any air leakage out of the containment. In addition to the vent stack monitors and the containment area radiation monitors, a continuous air monitor shall be in use adjacent to the equipment hatch. In accordance with Drawing N-9013-1-M-406, a temporary cover shall also be available to isolate the hatch opening in case of loss of the containment purge exhaust system.

h. For ALARA purposes, the severance of the main steam and feedwater pipes, unbolting of the wet lay-up connection, and severance of the upper dome and

1. Design Change Title/Station/Unit	2. Design Change Number
STEAM GENERATOR REPLACEMENT / NORTH ANNA / UNIT 1	DC 90-13-1

9. Engineering Review and Design:

wrapper plate shall be done with the secondary side of the steam generator flooded to a minimum level just below the transition cone cut location.

i. Certain preparatory work is required to support the completion of this Design Change Package. These activities are as follows:

- Construction of the old steam generator storage facility ready to accept the replaced steam generator lower assemblies.
- Selection, preparation, and load testing of the haul route.
- Receipt inspection, offload, and transport of the new steam generator lower assemblies.
- Modification of the west end security fence in accordance with DC 92-01-3, West End Security Access Facility.
- Equipment hatch platform modifications, electrical modifications, SG bioshield wall and operating floor elevation 291'-10" modifications, etc., as covered by DC 90-15-1.
- Procurement, fabrication, and delivery of the new steam generator replacement thermal insulation.

j. The steam generator replacement will be performed in accordance with the requirements of ASME Code, Section XI, 1983 Edition and Summer 1983 Addenda. Weld preparation, welding, postweld heat treatment, non-destructive examination (NDE), and baseline inservice inspection will be performed in accordance with the Special Processes Manual, NAS-1009, and Appendix 4-21, as applicable.

k. A fire watch shall be posted during all welding or burning operations and when any fire barriers are breached. Temporary construction and cleaning materials shall be controlled in accordance with approved project procedures. Portable fire extinguishers shall be provided, as required. Further discussion is provided in Section 3.3.

l. To maintain personnel radiation exposure as low as reasonably achievable (ALARA), the recommendations outlined in Section 3.23 shall be followed.

m. Jumpers to be installed in the RCP breaker and the temporary feeder cable splice shall be performed in accordance with North Anna Administrative Procedure ADM-14.1, "Jumpers (temporary modifications)," (VPAP-1403 "Temporary Modifications"), or temporary procedures will be written and approved to perform the modification.

n. Exercise caution when working in the vicinity of electrical circuits, which may be energized.

o. The remote indicating lights at the main control board and local indicating lights at the reactor coolant pump, RCP-1B, breaker shall be tagged to indicate power available for SGR temporary construction power.

p. Testing shall be performed in accordance with the Testing Procedure (see Tab 5 of this DC) which meets the functional testing requirements provided in Appendix 4-12.

q. Unfinished Calculations

Addenda to Class 1 Stress Reports (References 6.12.89 through 6.12.100) will be issued for all affected Class 1 piping upon completion of this Design Change. These stress reports are required to be completed prior to declaring the systems operable.

Due to the internal characteristics of the repaired steam generators, the steam generator level uncertainty calculation EE-0492 (Reference 6.12.102) will be generated based on revised Process Measurement Accuracies which will be provided by Westinghouse. This calculation is required to be completed prior to unit startup.

2.3.4 Piping Removal/Installation

Special provisions, precautions, and/or inspections applicable to these activities are identified in Sections 2.3.2, 2.3.3, and as follows:

- All piping attached to the SGs will be severed and portions removed to provide clearance for SG removal. This piping will be reconnected to the new SGs in the original configuration. In some cases, the removed piping sections and associated fittings may be discarded and replaced with new piping and fittings. The new piping and fittings will be procured per Specifications NAP-0014 and NAS-1009.

- Reinstallation will be performed per DC 90-13-1 drawings which comply with approved station drawings.

- In order to avoid reanalysis of affected piping, the existing arrangement/layout and locations of pipe supports will not be changed beyond tolerances shown on applicable drawings. Cold gap measurements for critical components (as listed in Appendix 4-27) shall be obtained before piping/support removal and again following reinstallation of piping. These cold gap measurements will be verified to be in accordance with design requirements. During this time, any requirements for specific hot gap measurements will be determined by Engineering. Piping system cold condition is considered to be less than 150°F.

- Prior to cutting, piping shall be temporarily supported for the dead weight of the remaining pipe and travel stops will be provided for spring hangers affected in accordance with Appendix 4-27.

- Openings of disconnected piping shall be closed-off and sealed sufficiently to prevent intrusion of contaminants until their use is required.

- The reactor coolant loops A, B, and C will have temporary shielding installed on the hot leg, cold leg, crossover leg, and bypass line prior to the RCS piping cuts. Cutting operations for the Loop B RCS piping may be performed during defueling operations. The acceptability of the various Loop B cut configurations is documented in Reference 6.12.21. The loop isolation valves shall be isolated prior to making any RCS piping cuts. Cutting of the Loop A and C RCS piping shall not be performed until all fuel has been removed from the containment, the reactor cavity drained, and the fuel transfer tube gate valve closed and blind flange installed.

  The acceptability of the RC loops A, B, and C piping severed from the steam generators after fuel is removed from the reactor is also documented in Reference 6.12.21.

- To accommodate any leakage from the loop isolation valves, two primary drain transfer tank pumps shall be available and the associated drain valves on the hot and crossover legs shall be tagged open from the time the loop isolation valves are shut until reactor vessel level has been drained to below the reactor vessel nozzles.

- Cutting operations for secondary plant piping systems including main steam, feedwater, feedwater drain, wet layup, steam generator blowdown, chemical feed, sampling, and steam generator level instrumentation may be performed during defueling operations. Prior to severance, an appropriate support configuration shall be established that ensures no adverse effects, including gravity missiles and sway interaction concerns, will result to any safety-related components. Loads to be considered include deadweight and seismic effects.

The following sections discuss additional requirements applicable to each piping system.

2.3.4.1 RCS Piping

The RCS piping (hot and crossover leg) will be cut using the two-cut method for severance and fit-up of the piping to the new SGs. In this method, cuts are made at the SG nozzles. The three-cut method will be used only if the two-cut method results in unacceptable fit-up per the ASME Code. The three-cut method entails the removal of the RCS crossover leg elbow adjacent to the SG by making an additional pipe cut. Mechanical cutting equipment will be used. Optical templating will be used to transfer measurements from the new SG nozzles to the RCS hot and crossover leg elbows. See drawing N-9013-1-M-

1. Design Change Title/Station/Unit	2. Design Change Number
STEAM GENERATOR REPLACEMENT / NORTH ANNA / UNIT 1	DC 90-13-1

9. Engineering Review and Design:

401 for RCS cut locations. Weld end preparation, welding, and NDE will be in accordance with the Special Processes Manual, NAS-1009, drawing N-9013-1-M-401, and Appendix 4-21, as applicable.

The RCS hot and crossover leg pipe ends remaining after removal of the SGs will be decontaminated to reduce radiation levels in the area of the SG replacement activities. (See Appendix 4-15 for Specification 21809-M-001 for procurement of decontamination services). RCS decontamination shall not be performed until all fuel has been removed from the reactor vessel.

Following weld end preparation, a visual inspection for any cutting debris shall be performed. The extent of this inspection will be, at a minimum, to the base of the crossover leg and to the loop isolation valve on the hot leg.

2.3.4.2 Main Steam Piping

The main steam piping will be cut at two locations:

- At the centerline of the existing weld at the SG nozzle, and
- On the horizontal straight section of pipe approximately 10'(horizontal) from the centerline of the existing weld at the main steam elbow as shown on drawing N-9013-1-M-402.

The piping will be mechanically cut to minimize material loss and debris and slag falling into the SG through the nozzle, since both the pipe elbow and steam dome will be reused. In the event of difficulties involved with fit-up of the main steam piping, a new spool piece section, procured in accordance with the requirements of NAS-1009, is available. Weld end preparation, welding, and NDE will be in accordance with the Special Processes Manual, NAS-1009, and Appendix 4-21, as applicable.

2.3.4.3 Feedwater Piping

The feedwater piping will be cut (either mechanical or flame cut) on the elbow side of the existing weld between the feedwater nozzle and the 90 degree elbow, and inside the fourth 90 degree elbow as shown on drawing N-9013-1-M-403. The severed pipe loop seal including the loop drain valves will be discarded and replaced with new chrome-moly steel material (piping class 601C) which meets the requirements of specification NAS-1009. The new valves with live load packing will be procured by specification NAP-0023. Weld end preparation, welding, and NDE will be in accordance with the Special Processes Manual, NAS-1009, and Appendix 4-21, as applicable.

1. Design Change Title/Station/Unit

STEAM GENERATOR REPLACEMENT / NORTH ANNA / UNIT 1

2. Design Change Number

DC 90-13-1

9. Engineering Review and Design:

2.3.4.4 Blowdown and Steam Generator Shell Drain Piping

The blowdown and shell drain piping will be cut (either mechanical or flame cut) at the SG socket weld nipples and at the 3-inch header, as shown on drawing N-9013-1-M-404, sheet 1. Prior to cutting, the piping will be properly supported, including the use of temporary supports if required. The cut sections of piping including the manual valves will be removed, discarded and replaced with new chrome moly steel material (piping class 601C). New sections will be fabricated per drawings N-9013-1-M-601, 602, and 603 from materials procured by Specifications NAP-0017 and NAP-0023 and meeting the requirements of Specification NAS-1009 for reinstallation to the new SGs. New/modified hangers will be installed in accordance with the drawings included in the Drawing Revision Record. The weld end preparations, welding, and NDE will be performed in accordance with the Special Processes Manual, NAS-1009, and Appendix 4-21, as applicable. An exception to NAS-1009 has been approved to allow welding pipe saddles to the blowdown piping (Appendix 4-34).

The new steam generator lower assemblies are provided with 2½" nozzle couplings for the blowdown piping instead of the 2" socket weld connection on the existing steam generators. The pipe size will be increased from 2 in. to 2½ in. to match the size of the nozzle on the new replacement SG lower assembly.

2.3.4.5 Sample System Piping

The sample system piping will be cut (either mechanical or flame cut) at the SG socket weld nipples as shown on drawing N-9013-1-M-404. Prior to cutting, the piping will be properly supported, including the use of temporary supports if required. The second cut location will be as shown on drawing N-9013-1-M-404 to provide adequate clearance for SG removal activities. Since the cut sections of piping will be removed and discarded, new sections will be fabricated from materials meeting the requirements of Specification NAS-1009 for reinstallation to the new SGs. The replacement sections of piping will be of identical size and configuration as the removed sections and the material will meet the requirements of NAS-1009. Weld end preparations, welding, and NDE will be in accordance with the Special Processes Manual, NAS-1009, and Appendix 4-21, as applicable.

2.3.4.6 Chemical Feed Tubing

The chemical feed tubing will be disconnected at the feedwater loop seal to provide clearance to make the second cut on the feedwater pipe and to remove and replace the feedwater loop piping. See drawings N-9013-1-M-604 through 606. The second cut location will be made in accordance with drawings N-9013-1-M-604 through 606 to provide adequate clearance for SG removal activities. The chemical feed tubing will be disconnected from the feedwater pipe by cutting upstream of the manual isolation valves 1-WT-39, 51, and 67.

Existing manual valves and removed sections of pipe will be saved and reinst[illegible] if p[illegible]ssible. Some of the tubing supports may be disconnected to provi[illegible] [illegible]ate flexibility. Any supports that are disconnected will be re[illegible]nec[illegible] in accordance with drawing N-9013-1-M-604 through 606. The [illegible]cement sections of tubing will be of identical size, material (ICN9, see DCP [illegible]-82) and configuration as the removed section. Weld end preparation, welding, and NDE will be in accordance with the Special Processes Manual, NAS-1009, and Appendix 4-21, as applicable. The modification is shown on drawing N-9013-1-M-403.

2.3.4.7 Wet Layup System Piping

The wet layup system piping will be disconnected at the steam generator flange in accordance with drawings N-9013-1-M-607, 608, and 609. The second cut location will be in accordance with drawings N-9013-1-M-607, 608, and 609 to provide adequate clearance for SG removal activities. Any supports that are disconnected will be reconnected in accordance with drawings N-9013-1-M-607 through 609. Weld end preparation, welding, and NDE will be in accordance with the Special Processes Manual, NAS-1009, and Appendix 4-21, as applicable.

2.3.4.8 Steam Generator Level Instrument Tubing

To provide clearance for the removal and reinstallation of the SGs, the instrument tubing, condensate pots, root/vent valve assemblies, unistrut stanchions, and supports associated with the wide range and narrow range SG level sensing lines will be disconnected from the SGs and removed. These components are located in the immediate vicinity of the SGs, from elevation 259'-11" to 315'-0". Refer to drawings N-9013-1-FK1A and N-9013-1-FK1B for specific line and mark numbers.

Piping and tubing will be removed and either reused or discarded. If required, new materials will be procured per Specifications NAI-0001 and NAP-0023 and will meet the requirements of NAS-1009 as applicable. The existing carbon steel condensate pots, root valves and vent valves will be discarded and stainless steel replacements will be installed. Refer to drawing N-9013-1-M-801 for fabrication details of the replacement condensate pots. Refer to drawings N-9013-1-M-801, N-9013-1-FK1A and N-9013-1-FK1B for removal and reinstallation details. Level transmitters shall be isolated prior to performing cuts on instrument tubing.

Following the installation of the new SGs, all tubing, piping, valves, and supports will be installed to the original pre-SG replacement configuration. Lower root valve assemblies and condensate pot assemblies will be socket welded to the appropriate tap connections. Stanchions and supports will be attached to the insulation support rings as in the original configuration. The tubing and tubing supports will be reinstalled to meet plant specific seismic

1. Design Change Title/Station/Unit	2. Design Change Number
STEAM GENERATOR REPLACEMENT / NORTH ANNA / UNIT 1	DC 90-13-1

9. Engineering Review and Design:

requirements per Specification NAI-0001.

The upper tap and condensate pots shall be installed with the root valve stem rotated 90 degrees to the horizontal position to maintain open channel flow from the condensate pot to the steam generator.

The elevation of the upper and lower level taps shall be surveyed after installation on the new steam generators for use in the calibration of the level instruments.

Weld end preparation, welding, and NDE will be in accordance with the Special Processes Manual, NAS-1009, and Appendix 4-21, as applicable.

2.3.5 Vibration and Loose Parts Monitoring System

The existing vibration and loose parts monitor assemblies will be removed from the old SG and reinstalled at different locations on the new steam generators as prescribed by Westinghouse in the Steam Generator Technical Manual (Reference 6.2). Two accelerometers are used on each steam generator. Each of the new steam generators will be provided with four sensor mounting holes (two each, at elevations 261'- 7-1/2" and 266'- 7-1/2"). The non-scheduled conduit required to complete installation of the accelerometers will be replaced, if necessary, in accordance with drawings N-9013-1-1FE57E, N-9013-1-1FE57F, and N-9013-1-1FE3HY.

Removal and replacement of the vibration and loose parts monitoring system may be performed during defueling and refueling operations.

2.3.6 Steam Generator Upper Restraints Removal/Installation

Virginia Power will remove and re-install the affected snubbers and struts in accordance with Procedure WPC-19. The existing upper lateral restraint will be cut and removed. To facilitate installation of the new SG upper restraints, a temporary support made of structural steel framing will be installed. The new upper restraint is bolted together per drawings N-9013-1-1FV17L, 17M, 17N, 17P, and N-9013-1-S-018.

Prior to removing the upper lateral restraints (including snubbers and struts), all fuel shall be removed from the containment, the reactor cavity drained, and the fuel transfer tube gate valve closed and blind flange installed. The temporary support shall be removed prior to fuel load.

2.3.7 Steam Generator Separation and Replacement

Since the steam generator steam dome is too large to pass through the equipment hatch, it will be severed from the lower assembly and stored inside the containment for reuse.

Several modifications are required within the steam dome to support the removal and reinstallation construction activities. In addition, a flow limiter will be installed in the main

1. Design Change Title/Station/Unit	2. Design Change Number
STEAM GENERATOR REPLACEMENT / NORTH ANNA / UNIT 1	DC 90-13-1

9. Engineering Review and Design:

steam nozzle of the steam dome and the existing resistance plates will be removed from the wrapper plate and discarded.

Prior to installation in the containment, the new SG lower assemblies will be prepared for joining to the steam domes. The tube bundles shall be adequately protected from damage during transport and rigging activities.

Disconnection of the steam domes from the old lower assemblies and their reconnection to the new lower assemblies will be performed in accordance with drawing N-9013-1-M-405 and includes the activities outlined below. Except for preparation of the new SG lower assemblies, construction activities associated with this design scope will be performed inside the containment.

- In order to obtain access to the I.D. transition cone girth weld, two access ways will be cut in the swirl vane transition wrapper.
- Prior to severance of the transition cone and lifting of the steam dome, the primary moisture separator (PMS) will be secured. This will allow the PMS to be rigged with the steam dome.
- Severance of both the steam generator shell and the wrapper plate will be accomplished by using a track-mounted, tractor-driven, oxygen-fuel cutting torch. Two circumferential cuts will be made in the SG shell, resulting in a band of shell material that will be removed as part of the construction effort. Removal of this band will provide an access for the cut to be performed on the wrapper plate.
- The 3" x 6" x 1/2" support bars located between the wrapper and steam separator assembly will be removed by grinding existing welds. If grinding proves not to be a viable means of removing the support bars, an access hole will be cut in the wrapper using a torch, and the support bars removed. If this method becomes necessary, reinstallation will require a patch plate be applied to the wrapper to return it to its original condition. Prior to removal of the support bars, as-built details of the gap measurements will be taken for reinstallation to the existing arrangement.
- For reinstallation, the girth weld area will be prepared with the steam domes in an upright position in the storage area. While the steam dome is in this position, installation of the main steam flow limiter will be performed and the existing downcomer flow resistance plates will be removed and discarded.
- The wrapper plate will be prepared by welding the fit-up ring with its top horizontal backing bar on the existing transition barrel. The bottom horizontal backing bar will be welded on the landing ring of the new lower assembly outside the containment.
- Once in position for welding, the transition cone shell will be preheated and the joint welded using the shielded metal arc welding process and post-weld heat

treated.

- The wrapper plate will be joined by welding new radial plates on the horizontal backing bars. Vertical backing bars will be used between the radial plates.
- The support bars will be reinstalled using the as-built details and drawing N-9013-1-M-405.
- All temporary interior structures in the shell will be removed.
- After secondary side inspection, the access ways in the swirl vane transition wrapper will be welded shut and the secondary manways will be closed.
- The final close-out inspection will include a visual inspection of the steam generator steam domes and secondary side tubesheet to ensure that no foreign objects are present.

Weld preparation, welding, and NDE will be in accordance with the Special Processes Manual, Appendix 4-21, and drawing N-9013-1-M-405, as applicable. Installation/welding of the main steam flow restrictor will be in accordance with Procedure STD-FP-1991-5563 (Appendix 4-16) and the Special Processes Manual. Removal of the existing downcomer flow resistance plates will be performed in accordance with drawing N-9013-1-M-405.

Various components (e.g., vertical support plates, shims, socket head cap screws, hex nuts, etc.) associated with the steam generator lower supports will be removed and replaced per drawing N-9013-1-1FV17G.

2.3.8 Steam Generator Handling and Rigging

As part of the SG replacement, heavy loads must be rigged into, out of, and within the containment building. To promote the safe performance of these rigging activities, a detailed design effort was performed to determine the most efficient and safe methods for handling the SG components, and was followed by the design of specialty items required for implementation. For further discussion, see Section 3.14.

Construction activities associated with this design scope will be performed inside the containment building, at the equipment hatch platform, along the SG haul routes, and at the SG storage facilities. A description of these activities is as follows:

- An upper swivel lifting beam (SLB) will be installed between the two main hooks of the containment polar crane.
- A lower SLB will be attached to the steam domes via a pair of link plates. As each steam dome is cut free from its lower assembly, it will be rigged to a storage area for preparation for reinstallation. The steam dome will remain in the upright position during this preparation phase.

1. Design Change Title/Station/Unit	2. Design Change Number
STEAM GENERATOR REPLACEMENT / NORTH ANNA / UNIT 1	DC 90-13-1

9. Engineering Review and Design:

- As they become available for removal, each existing lower assembly will be attached to the lower SLB using the same lifting rods as were used for the steam domes. The lower assembly will be lifted, trolleyed toward the center of containment until it clears the bioshield wall, then moved directly to the pivot stand, set in the pivot stand and downended onto the lower assembly carriage. Upon completion of the downending sequence, the pivot stand will be removed in preparation for lower assembly removal from containment.

- During the downending sequence, the lower assembly will be set on the lower assembly carriage (LAC) for removal from the containment. The LAC will be supported on a set of runway beams which will be placed parallel to the centerline of the equipment hatch. Support stands with integral hydraulic jacks will be used as the supports connecting the LAC to the tracks. The bottom of the support stands are outfitted with roller assemblies which will enable it to be pulled along the support beams. The support stands are sized and located such that the weight of the lower assembly can be transferred from inside the containment to a similar set of support beams on the equipment hatch platform outside the containment. The LAC will remain with the lower assembly throughout the rigging-out sequence.

- Once outside the containment, the lower assembly will be removed from the equipment hatch platform by a gantry crane and set on a site transporter that has hydraulic jacking capabilities. The transporter will be pulled to the storage facility, backed into the facility, and the lower assemblies offloaded onto support beams and columns utilizing the jacking capabilities of the transporter.

- The new lower assemblies will be rigged from storage into place in the containment in the reverse order of the removal process. The steam domes will then be set in place on the lower assemblies for welding, and the SLBs will be removed from the polar crane hooks.

Special provisions, precautions, and/or inspections applicable to these activities are identified in Sections 2.3.2, 2.3.3, and as follows:

- The maximum load to be lifted by the containment polar crane, including all rigging, shall be 280 tons or less, with the load divided evenly between the two main hoists spaced no closer than 1[illegible]'-3" (see Appendix 4-31). The maximum load to be handled is estimated in Calculation 21809-C-05 (Reference 6.12.12).

- To preclude any possible adverse effects on stored spent fuel or systems shared with Unit 2 as a result of a postulated steam generator drop, all component cooling water, service water, fuel pit cooling and refueling purification system, and instrument air lines to/from the containment shall be isolated during rigging operations for the steam domes and lower assemblies.

- In accordance with Item (f) of Section 2-3.2.1.1 of ASME B30.2, the polar crane shall be load tested by lifting the first SG lower assembly to be handled approximately 6 inches and suspending the load for approximately 5 minutes. See Appendix 4-31.

- The old SG vessel trunnions and lifting lugs shall be nondestructively examined (MT) prior to use.

- Prior to removing the old steam generator lower assemblies from the containment for storage, the lower assemblies shall be drained and all openings sealed to prevent any leakage of radioactive contamination (see drawing N-9013-1-M-800). A coating of A-B-C Encapsulant shall be applied to the exterior of the lower assemblies to prevent the spread of any surface contamination that may exist.

- The Haul Route Evaluation (Appendix 4-18) shall be followed and the requirements for protection of buried utilities and a load test of the haul routes performed.

- Limitations on transport of the new and old steam generator lower assemblies and the load test are as follows:

  - The maximum transporter speed is 5 mph.

  - The transporter leveling capability will be used to maintain the transporter bed level within 5° during movement.

  - In accordance with N-9013-1-M-800, the transition cone cover plate "T-sections" shall be installed before the old SG lower assemblies are moved outside the protected area.

  - The centerline of the steam generator will be limited to a maximum of 12'-3" above grade.

2.3.9 Insulation Removal and Replacement

The existing thermal insulation on the steam generators will be replaced with new fiberglass blanket type insulation. The replacement blanket insulation will meet or exceed the design requirements of the existing insulation. The insulation will be procured by Specification NAP-0047 and will be installed in accordance with the Vendor Technical Manual (Reference 6.3). Removable panels will be designed to facilitate inservice inspection. The new insulation is seismically designed to ensure that the insulation remains in place in the event of a seismic occurrence.

Removal and re-installation of insulation may be performed during defueling/refueling and will be performed in accordance with the station heavy loads procedure 0-MCM-1303-01. Instrumentation tubing which could be damaged during insulation removal shall be tagged out prior to removal of the insulation.

2.3.10 Temporary Containment HVAC Modifications for Ventilation and Smoke Removal

This DCP will temporarily modify the containment purge system so that it will be more effective during the replacement outage. The blind flange on the 18" diameter manhole (inspection port) located upstream of the containment penetration isolation valve on the purge exhaust (return) duct will be removed and stored. A manual volume control damper and a 90 degree elbow will be installed at the open manhole. An 18" diameter flexible duct will be connected to the other end of the elbow. The flexible duct will be run vertically up along the containment wall to elevation 345'-0" (approximately).

Prior to fuel load, the temporary damper, elbow and flexible duct will be removed and the blind flange reinstalled.

In accordance with Technical Specification 3/4.9.9, the containment purge system (supply and exhaust), including the standby HEPA filters and the containment ventilation system must be operable and modifications can only be performed after the plant has been defueled. After the modifications are completed, the system is to be returned to service.

2.3.11 Temporary Modified Security Door

To facilitate installation of the temporary air chiller units, the security door (A-95-1) located at the entrance to the MCC room roof will be temporarily removed and replaced with a modified security door which will allow passage of air hoses/pipes from the roof down to the containment personnel hatch. The modified security door will meet all requirements of the North Anna Security Plan and will be removed and replaced with the original security door upon completion of the SGR work. Site security personnel will post a security guard at the door while the modified door is being installed and again when the original door is being replaced.

2.3.12 Temporary Construction Power

A construction power distribution system will be temporarily installed to support the in-containment SGR work activities. The power feeder to RCP-1B will be utilized to provide power to the primary side of the steam generator replacement dry type transformer. The power feeder to RCP-1B will be disconnected at the motors main terminal box and temporary feeder cables will be spliced directly to that feeder. Refer to Drawings N-9013-1-1FE1B and N-9013-1-1FE8N.

Three General Electric IAC53B805A relays will be temporarily installed in the Unit 1 RCP-1B feeder breaker 15B3 to provide the temporary construction transformer overload protection as well as three-phase and phase-to-phase 480 v fault protection. These relays will be set in accordance with Calculation EE-0490 (Reference 6.12.101).

RCP-1B feeder breaker 15B3 will be temporarily modified to jumper out pump motor protective interlocks. Installation and removal of jumpers are required to accomplish the modification and will be performed by Virginia Power. Installation of the power feeder splice and the jumpers will either be in accordance with North Anna administrative

1. Design Change Title/Station/Unit	2. Design Change Number
STEAM GENERATOR REPLACEMENT / NORTH ANNA / UNIT 1	DC 90-13-1

9. Engineering Review and Design:

procedure ADM-14.1 "Jumpers (temporary modifications)" (VPAP-1403 "Temporary Modifications") or temporary procedures will be written and approved to perform the modification. Refer to drawings N-9013-1-1ESK5AJ, N-9013-1-1FE8N, and N-9013-1-1FE1B for details.

All temporary power covered in this section is non-safety related and is installed non-seismic. All temporary equipment and cables will be removed upon completion of the SGR work.

2.3.13 Reactor Cavity Cover

Steel decking will be installed over the open sections of the reactor refueling cavity after all fuel is removed from the containment, the reactor cavity drained, and the fuel transfer tube gate valve closed and blind flange installed. The reactor cavity cover will provide additional construction laydown area and prevent materials from entering the reactor refueling cavity. The reactor cavity cover is not required to support the rigging of the steam generator lower assemblies. The reactor cavity cover shall be removed prior to fuel load.

2.3.14 Auxiliary Crane

A non-safety related auxiliary crane will be installed in the containment on the El. 291'-10" operating floor slab to support the handling of loads during the steam generator replacement outage. The support tower for the crane will be attached to the concrete slab using anchor bolts that were installed under design change package DC 90-15-1. The auxiliary crane and support tower shall not be installed until all fuel has been removed from the containment, the reactor cavity drained, and the fuel transfer tube gate valve closed and blind flange installed. The auxiliary crane and support tower shall be removed prior to plant refueling.

2.3.15 Jib Crane

A 3-ton capacity jib crane will be temporarily installed inside the containment for use during defueling and refueling operations. As per the guidelines of NUREG-0612, physical limitations shall be placed on the swing radius of the crane to ensure approved load paths are followed as designated in the station heavy loads procedure 0-MCM-1303-01 and that no adverse interaction with spent fuel or fuel handling activities will occur. This crane will be replaced, in the same physical location, by the auxiliary crane when defueling is completed. The jib crane will be reinstalled at the end of the outage to assist in containment demobilization.

2.3.16 Optical Templating Bracket Removal

Optical templating brackets were installed as permanent plant equipment during the implementation of DC 92-006-1. These brackets will be removed at the end of the SGR outage under this DC by unbolting them. The expansion anchor bolts will be cut off flush to the surface of the wall and then driven into the wall slightly to allow for smooth finish

1. Design Change Title/Station/Unit	2. Design Change Number
STEAM GENERATOR REPLACEMENT / NORTH ANNA / UNIT 1	DC 90-13-1

9. Engineering Review and Design:

grouting of the anchor bolt holes. Refer to drawing N-9013-1-1FM1C. Grouting will be performed in accordance with NAI-0014.

3.0 PROGRAMS REVIEW

3.1 UPDATED FINAL SAFETY ANALYSIS REPORT

A review of the Updated Final Safety Analysis Report has been performed. This was based on review of sections of the UFSAR describing systems and components affected by this design change modification. Appropriate sections were identified using the UFSAR Keyword Index. The results of the review are as follows:

UFSAR Section 3.7 discusses the seismic design of structures, systems and components. Section 3.7.2.6.1 contains a listing of computer programs that are used by SWEC for the analysis of seismic class 1 components. This section needs to be updated to include the NUPIPE-SW, STRUDL-SW and STEHAM computer programs.

UFSAR Section 3.7.3.1 describes the SWEC Analyses and Design Criteria of Seismic Class 1 Piping. This section needs to be updated to include details related to the NUPIPE-SW computer program.

UFSAR Tables 3.7-11 and 3.7-12 contain a listing of seismic design margins for various components installed at North Anna Units 1 and 2. These tables need to be updated to reflect the new stress levels and component loads for the Unit 1 steam generator and reactor coolant pump equipment supports.

UFSAR Section 5.2.1 discusses the integrity of the RCS boundary and will be revised to reflect the analysis of the replacement steam generator tubesheet complex.

UFSAR Tables 5.2-4 through 5.2-8 will be revised to show new Unit 1 changes to the SG Primary-Secondary Boundary Components stress analysis.

UFSAR Tables 5.2-9 and 5.2-10 will be revised to show new Unit 1 changes in SG usage factors, individual transients, primary and secondary boundary components.

UFSAR Table 5.2-11 will be revised to show changes in the new Unit 1 tubesheet stress analysis results.

UFSAR Table 5.2-12 will be revised for the new Unit 1 limit analysis calculation results.

UFSAR Table 5.2-14 is being revised for clarity.

UFSAR Table 5.2-17 contains faulted condition loads for the reactor coolant pump feet. This table needs to be updated to reflect the new loads on the Unit 1 reactor coolant pump feet.

UFSAR Table 5.2-22 will be revised to show correct material for the new Unit 1 SGs.

1. Design Change Title/Station/Unit	2. Design Change Number
STEAM GENERATOR REPLACEMENT / NORTH ANNA / UNIT 1	DC 90-13-1

9. Engineering Review and Design:

UFSAR Figures 5.2-1 through 5.2-4 will be revised to correspond to appropriate unit descriptions for SG design information.

UFSAR Section 5.5.2 will be revised to reflect the new SG lower assembly design details.

UFSAR Section 5.5.9 describes the RCS equipment supports. Section 5.5.9.2.2 discusses specifically the steam generator and reactor coolant pump supports. This section needs to be updated to incorporate the steam generator upper support ring cold condition radial gap tolerance of 0.050 to 0.060 inches. Section 5.5.9.3.1 discusses the dynamic analyses that determines loads on the subject equipment supports. This section needs to be updated to include reference to the NUPIPE-SW computer program.

UFSAR Table 5.5-3 will be revised to show changes in the new Unit 1 SG design data.

UFSAR Figure 5.5-3 will be revised to show the replacement steam generator diagram for unit 1.

UFSAR Figure 5.5-17 provides the dynamic model for the reactor coolant loop including the steam generator and reactor coolant pump supports. This figure (to be identified as figure 5.5-17B) will remain valid for the Unit 2 dynamic model only. An additional figure (to be identified as figure 5.5-17A) for the modified Unit 1 dynamic model needs to be incorporated into the UFSAR.

UFSAR Tables 5.5-18, 5.5-19 and 5.5-20 contain loads for the steam generator and reactor coolant pump equipment supports. These tables need to be revised to incorporate the new Unit 1 equipment support loads.

UFSAR Figures 5.5-24 and 5.5-25 provide the identification of steam generator and reactor coolant pump support members respectively. These figures were previously utilized with UFSAR Tables 5.5-21 and 5.5-22 which have been deleted from the UFSAR. UFSAR figures 5.5-24 and 5.5-25 are no longer required and should be deleted.

UFSAR Section 6.2 discusses Containment Systems. This section is updated to reflect the revised analysis of peak pressure and depressurization following a LOCA based on the mass and energy releases for the replacement steam generators. Where information for Unit 1 differs from the current analysis for Unit 2, clear references to applicable units are added.

UFSAR Section 6.2 also addresses the impact of the new insulation on the containment response following a LOCA. The impact of insulation debris on sump NPSH is evaluated.

UFSAR tables 6.2-2,5,13-17, 47 53, and 77 are revised to reflect the differences between the Unit 1 and Unit 2 analyses.

UFSAR Section 6.3 discusses the Emergency Core Cooling System. This section is updated to reflect the change in the NPSH values and the revision to the RWST switchover setpoint for the LHSI recirculation mode for Unit 1.

UFSAR Figure 6.3-11 is revised to reflect the change in the Unit 1 RWST switchover setpoints.

1. Design Change Title/Station/Unit	2. Design Change Number
STEAM GENERATOR REPLACEMENT / NORTH ANNA / UNIT 1	DC 90-13-1

9. Engineering Review and Design:

UFSAR Figures 6.3-12 through 6.3-17 are revised to show the Unit 1 vs. Unit 2 NPSH analysis differences.

UFSAR Section 10.3.2 contains the description of the main steam system. This section needs to be revised to describe the new flow limiting device having a flow area of 1.4 ft^2 to be installed in each of the Unit 1 main steam nozzles of the steam generator dome.

UFSAR Figure 10.4-7 and 10.4-8 detail the chemical feed piping and indicate the pipe class at the feedwater elbow as pipe class 601. The installed piping was changed from pipe class 601 to class IC-N-9 as part of DC-80-S82, but these figures were not revised to reflect the change. The replacement chemical feed piping will be IC-N-9 to match the existing installation. These figures will be revised to indicate use of class IC-N-9.

UFSAR Figures 10.4-8 (sheet 1), 10.4-11, 10.4-13 and 10.4-14 show the configuration of the main feedwater system and a portion of the chemical feed system in the area of the feedwater loop seal piping. Each of these figures needs to be updated to reflect the change in pipe class for the new class 601C feedwater loop seal piping replacement.

UFSAR Figure 10.4-17 details the steam generator blowdown piping system arrangement. This figure needs to be updated to reflect the system modifications being implemented by this Design Change. Specifically, the changes in pipe size and class need to be incorporated in this figure.

All UFSAR changes to Section 15 are identified in Appendix C of the NAF report NE-883, Rev. 1 which is included in Appendix 4-2 . A summary page of the affected sections of Chapter 15 is also provided in the report.

The Updated Final Safety Analysis Report Change Request form has been completed and included as Appendix 4-2.

3.2 TECHNICAL SPECIFICATIONS

After review of Safety Evaluations performed by Westinghouse (SECL 90-113, Appendix 4.19) and Virginia Power (NAF Technical Report NE-883, Rev. 1, Appendix 4-23), no changes to the Station Technical Specifications are required as a result of this design change. This conclusion is based on a review of the following Technical Specifications:

Section 2.1 Safety Limits

The Reactor Core Safety Limits for three loop operation were modified by Technical Specification Amendment No. 154 to include an applicable graph (Figure 2.1-1a) for use prior to replacement of the Unit 1 steam generators. The previous limits remained in place for use after the SG replacement activities. Therefore, no change is needed as a result of this design change.

Section 2.2 Limiting Safety System Settings

This section, as with Section 2.1 above, included provisions for conditions prior to and after SG replacement. Therefore, no change is needed as a result of this design change.

1. Design Change Title/Station/Unit	2. Design Change Number
STEAM GENERATOR REPLACEMENT / NORTH ANNA / UNIT 1	DC 90-13-1

9. Engineering Review and Design:

Section 4.0.5 Applicability-Surveillance Requirements

This Technical Specification addresses the ISI requirements stated in ASME Section XI. The replacement steam generators are Class I components and, therefore, are covered by this standard.

Section 3/4.1.1 Boration Control

The Bases for Technical Specification 3/4.1.1.3.1 indicate that this flow rate provides adequate mixing, prevents stratification and ensures that reactivity changes will be gradual during boron concentration reductions in the reactor coolant system. The Bases further state that this flow rate will circulate an equivalent reactor coolant system volume of 9957 ft^3 in approximately 30 minutes. A change to this Technical Specification (proposed change 276) has been submitted to the NRC via letter No. 92-467, dated July 16, 1992, which deletes the specific volume value from the bases section. Therefore, no change to the Technical Specifications is required as a result of this design change.

The following Technical Specifications deal with reactivity control relative to the coolant boron concentration:

Section 3/4.1.1.3.1 Reactor Coolant Flow
Section 3/4.1.1.3.2 Valve Position
Section 3/4.1.2.1 Flow Path-Shutdown
Section 3/4.1.2.3 Charging Pump-Shutdown
Section 3/4.1.2.5 Boric Acid Transfer Pump-Shutdown
Section 3/4.1.2.7 Borated Water Sources-Shutdown

In each case, these Technical Specifications will be met while fuel is still in the reactor vessel. Once the vessel is defueled, these Technical Specifications do not apply until the beginning of fuel on-loading.

Section 3/4.2.5 DNB Parameters

This section provides the DNB related parameters which include minimum allowable RCS total flow. Technical Specification amendment No. 154 provided a reduced allowable minimum RCS flow value for operation with extended SG tube plugging. The previous total RCS minimum flow limits remained in the Technical Specifications and will again be applicable following the SG replacement. Therefore, there is no change to this section as a result of this design change.

Section 3/4.3.1 Reactor Trip System Instrumentation

This section provides the requirements for the reactor trip system with regard to instrumentation operability, interlocks, response times, and surveillances. These requirements are not affected by this design change, therefore, no change is required.

1. Design Change Title/Station/Unit	2. Design Change Number
STEAM GENERATOR REPLACEMENT / NORTH ANNA / UNIT 1	DC 90-13-1

9. Engineering Review and Design:

Section 3/4.3.2 Engineered Safety Feature Actuation System Instrumentation

This Technical Specification provides ESFAS operability requirements, trip setpoints, interlocks, response times, and surveillance requirements. This design change does not affect ESFAS instrumentation requirements and/or setpoints, therefore, no change is required.

Section 3/4.3.3 Monitoring Instrumentation

Technical Specifications 3/4.3.3.5, 3/4.3.3.6, and 3/4.3.3.9 require various monitoring instrumentation on the steam generators. All required instrumentation will be in-service following the steam generator replacement activities.

Section 3/4.4.5 Steam Generators

This section contains the pre-service eddy current testing requirements for the new SG tables. This will be performed at the fabrication shop which is allowable as the result of Tech. Spec. amendment No. 151. Therefore, no change is required as a result of this design change.

Section 3/4.5.2 ECCS Subsystems - $T_{ave} \geq 350$ °F

This section provides the operability requirements for the ECCS system above 350 °F. Additional equipment operability requirements were added to item A by Tech. Spec. amendment No. 153. These requirements applied to the case of a LHSI pump being inoperable and will not be in effect following the SG replacement. Therefore, no change to this section of the Tech. Spec. is necessary as a result of this design change.

Section 3/4.7.2 Steam Generator Pressure/Temperature Limitation

This Technical Specification requires that the temperatures of both the primary and secondary coolant in the steam generators be $\geq$ 70 °F when the pressure of either coolant in the steam generator is $\geq$ 200 psig. This requirement ensures that pressure-induced stresses in the steam generators do not exceed the maximum allowable fracture toughness stress limits. The requirements of this Technical Specification will continue to be met.

Section 3/4.7.8.1 Safeguards Area Ventilation System

This Technical Specification ensures the operability of the auxiliary building HEPA filter and charcoal adsorber assembly. After all fuel has been removed from the containment, to preclude any potential for damage to the charcoal filters, purge flow will normally not be directed through the filters. Should the containment purge exhaust be aligned through these filter assemblies while the purge system is removing smoke from the containment, then the surveillance requirements of 4.7.8.1.b will be invoked and additional testing of the filter assemblies will be required. No change to this Technical Specification is required by this design change; however, Station Operations must be aware of the possible implications of the purge exhaust alignment through these filters.

SWW
1-30-92

1. Design Change Title/Station/Unit

STEAM GENERATOR REPLACEMENT / NORTH ANNA / UNIT 1

2. Design Change Number

DC 90-13-1

9. Engineering Review and Design:

Section 3/4.8 Electrical Power Systems

Technical Specifications 3.8.1.2, 3.8.2.2, and 3.8.2.4 specify the minimum electrical equipment to be operable during Modes 5 and 6. The intent of these Technical Specifications is to provide power to safety-related equipment required for safe shutdown, accident mitigation, and monitoring. However, only those portions of systems required to maintain and monitor spent fuel pool cooling and makeup capability and containment purge are required during steam generator activities. Any electrical equipment inside the containment which must be moved because of interference problems will not be necessary or required to support spent fuel pool cooling and makeup or containment purge.

Section 3/4.9 Refueling Operations

Technical Specification 3.9.1 requires that the boron concentration in the reactor coolant system and refueling canal be sufficient to ensure that K_{eff} is $\leq$ 0.95 or the boron concentration is $\geq$ 2300 ppm. The provisions of this Technical Specification will be met until the reactor is defueled. At this point, the requirements of this Technical Specification are no longer necessary to ensure plant safety.

Technical Specification 3.9.2 requires that two source range neutron flux monitors be operable during Mode 6. The provisions of this Technical Specification will be met until the reactor is defueled. At this point, the requirements of this Technical Specification are no longer necessary to ensure plant safety.

Technical Specification 3.9.4 specifies the status of containment penetrations during movement of irradiated fuel inside containment. This Technical Specification will continue to be met until the reactor is defueled. At this point, the requirements of this Technical Specification are no longer necessary to ensure plant safety.

Technical Specification 3.9.5 requires that direct communications between the control room and personnel at the refueling station be maintained during core alterations. This Technical Specification will continue to be met until the reactor is defueled. At this point, the requirements of this Technical Specification are no longer necessary to ensure plant safety.

Technical Specification 3.9.8 specifies operability requirements for the RHR system during Mode 6. The requirements of this Technical Specification will be met until the reactor is defueled. At this point, the requirements of this Technical Specification are no longer necessary to ensure plant safety.

Technical Specification 3.9.10 specifies the required water levels in the reactor vessel during movement of fuel assemblies and control rods within the reactor. These requirements will be met during defueling activities. Once the reactor is defueled, the requirements of this Technical Specification are no longer necessary to ensure plant safety.

Section 3/4.9.9 Containment Purge

This design change will make temporary changes to the containment purge and exhaust system. This system is required to be operable during Mode 6, "refueling." Modification to this system will not be performed until all defueling operations are completed. The temporary changes will be removed prior to fuel loading operations, therefore, no change to this section of the Technical Specifications is required as a result of this design change.

Section 5.4.2 Reactor Coolant System Volume

This Technical Specification Design Features section indicates that the total water and steam volume of the reactor coolant system is 9957 ± 10 ft^3 at a nominal T_{avg} of 525°F. The volume of the reactor coolant system will increase approximately 132 ft^3 as a result of the steam generator replacement. This is due to the increase in the number of tubes in the new steam generator compared to the original design.

No change is needed to the Technical Specification regarding this section since Technical Specification Change Package No. 276 proposes an "administrative change" correcting the modifying the Technical Specification description for RCS volume. Various types of systems modifications have affected this value which in itself is not a significant number. It is proposed that the RCS volume be changed to "approximately 10,000 ft^3." This proposed Technical Specification change has been submitted to the NRC via letter No. 92-467, dated July 16, 1992.

Section 3/4.4.1.3 Reactor Coolant System
Section 3/4.9.8 Residual Heat Removal and Coolant Circulation

Technical Specification 3/4.4.1.3 and 3/4.9.8 specify minimum number of loops and/or RHR systems to be operable. The availability of RHR systems will allow for the severance of RCS Loop B piping before the completion of fuel offload. Accordingly, these sections of the Technical Specifications do not require revision or exception.

Future Technical Specification Revisions

Although this design change does not require a revision to the current Technical Specifications, implementation of the design change will make some information contained in the Technical Specifications obsolete. This information exists in sections 2.1, 2.2, 3/4.2.5, and 3/4.5.2 as discussed above. Following SG replacement, this information will be removed as part of an administrative change to the Technical Specifications.

3.3 FIRE PROTECTION/APPENDIX R

The electrical components and cables added under this design change are for temporary service for the SGR and are non-safety related. This equipment performs no safe shutdown or fire protection function. Except for the cables and electrical components, no combustibles are introduced to the plant. As this equipment is temporary and required only during the SGR, no combustible material is permanently added to or deleted from the plant.

Temporary construction and cleaning materials will be controlled in accordance with approved project procedures.

The instrument tubing, upper condensate pot, and root/vent valve assemblies are associated with the wide- and narrow-range steam generator level transmitters. Also, the steam generators are part of

1. Design Change Title/Station/Unit	2. Design Change Number
STEAM GENERATOR REPLACEMENT / NORTH ANNA / UNIT 1	DC 90-13-1

9. Engineering Review and Design:

both main steam and reactor coolant systems which are safe shutdown systems. These systems are required for Appendix R. However, since the equipment is reinstalled to its original configuration, there is no effect on Appendix R.

During maintenance and refueling outages, fire suppression is provided by hose stations and portable fire extinguishers brought in and stationed throughout the containment. Weld and flame permits will be issued as required. A fire watch will be posted during all welding or burning operations. After the design change is implemented, special fire protection in the affected areas will not be needed.

This modification does not delete any emergency lights, nor does it obstruct or change illumination levels from the emergency lighting system.

This DCP does not add new components required for safe shutdown.

For the previously stated reasons, it is concluded that the proposed modification does not adversely impact the station's design basis for compliance with Appendix R to 10 CFR 50. The Appendix R Design Summary Sheet is included as Appendix 4-3.

3.4 ENVIRONMENTAL QUALIFICATION

Plant conditions which affect environmental qualification considerations for electrical equipment and instrumentation does not change as a result of the steam generator replacement. This evaluation is based on the following:

a. No new fluid systems will be added to containment or other plant locations.

b. No changes in categorization (i.e., high, moderate, or low energy) will be made to existing fluid systems.

c. No changes to the locations of existing fluid systems will be made that will add high or moderate energy systems to areas or compartments which currently have no such systems.

d. Peak and long-term containment conditions such as pressure, temperature and humidity following design basis accidents do not increase as a result of the internal design changes in the new steam generators (see Appendix 4-23)

Existing equipment removed as interferences will be properly restored using plant procedures.

As a result of the steam generator replacement, the vibration and loose parts monitors will be relocated to El. 261'-7 1/2" and 266'-7 1/2". The affected monitors are as follows:

Steam Generator A (1-RC-E-1A)
- YE-LPM-100C
- YE-LPM-100H

1. Design Change Title/Station/Unit	2. Design Change Number
STEAM GENERATOR REPLACEMENT / NORTH ANNA / UNIT 1	DC 90-13-1

9. Engineering Review and Design:

Steam Generator B (1-RC-E-1B)
YE-LPM-100D
YE-LPM-100J

Steam Generator C (1-RC-E-1C)
YE-LPM-100E
YE-LPM-100K

These monitors are classified as non-safety related with special regulatory significance (NSQ) and will be located in a harsh environment as designated in the Environmental Zone Descriptions. 10CFR50.49(b)(2) requires that non-safety related electrical equipment be reviewed to determine if failure of this equipment under postulated environmental conditions could prevent satisfactory accomplishment of a different component's safety function. The mountings for the loose parts monitor accelerometers are depicted on drawing N-9013-1-E-3500. These mounts have been designed to ensure that the accelerometers will remain in place under postulated environmental conditions.

There are no new safety-related electrical components installed by this design change. Therefore, the Environmental Qualification Master List is not affected.

3.5 SECURITY REVIEW

A temporary door will be installed and locked at the exit to the Rod Control MCC roof in accordance with the security plan. In addition, the West End Security Access Facility will be installed in accordance with DC 92-01-3.

3.6 ELECTRICAL SYSTEM ANALYSIS

All loads added are temporary and are required only during the steam generator replacement outage. Temporary loads have been compared to existing equipment ratings to ensure that no limits are exceeded. Temporary relays will provide overload protection as well as 3-phase and phase-to-phase fault protection. Bus and feeder modifications to support the temporary needs will be restored to their pre-SGR outage configuration. There will be no change to any permanent electrical loads, therefore the Station Electrical Load List (SELL) will remain unchanged.

3.7 INSERVICE INSPECTION

The steam generator lower tube bundle assembly, upper shell, and attachment welds will be baseline preservice inspected to ASME Section XI, 1983 Edition, Summer 1983 Addenda, requirements. This inspection will include:

- Steam generator tubing
- Feedwater nozzle to piping
- Main steam nozzle to piping
- Sample system piping
- Chemical feed injection
- Blowdown piping and valves

In addition to the above baseline inspection, the following items are to be added to the NAPS-1 ISI program:

- Safe end to RC piping
- Channel head nozzle to safe end
- Primary nozzle knuckle radius

The following ISI joints have been eliminated due to the design of the steam generators:

- Lower transition cone to shell barrel
- Stub barrel to shell barrel
- Transition cone girth weld (see Appendix 4-24)

Piping system NDE requirements for this DC are identified in the Special Processes Manual and Appendix 4-21, as applicable.

3.8 SEISMIC

The replacement steam generator lower assemblies are qualified as seismic Category I.

The entire reactor coolant loop, including piping, components, component supports, branch lines, and branch supports, has been evaluated to determine the impact of a small increase in the normal operating weight of the new steam generators on its seismic qualification.

Westinghouse performed an evaluation which concluded that the effect of the weight increase on the RCL piping and equipment (including equipment nozzle loads and Class 1 branch line nozzles) is acceptable. The Westinghouse evaluation is contained in Letter No. VRA-91-022, dated March 5, 1991.

SWEC performed several calculations which show that the RCL component supports, SG and reactor coolant pump (RCP) feet, RCL branch lines, and branch line supports remain acceptable after replacement of the steam generators. The SWEC calculations are summarized below:

- Calculation 02072.13-NP(B)-001-X (Reference 6.12.15) generated new loads on the component supports, component feet, and new displacements at the RCL branch lines.
- Calculation 02072.13-NP(B)-002-BA (Reference 6.12.16) evaluated the lower SG and RCP support frames and generated acceptable stresses and factors of safety for all components.
- Calculation 02072.13-NP(B)-003-BA (Reference 6.12.17) evaluated the upper SG restraint and generated acceptable factors of safety for all components.
- Calculation 02072.13-NP(B)-005 (Reference 6.12.19) reviewed applied loads for the steam generator and RCP support feet. Bending moments at various points on the SG shell were also reviewed. Acceptable factors of safety were generated for all items.
- Calculation 02072.13-NP(B)-004-X (Reference 6.12.18) is a summary of all RCL branch line

1. Design Change Title/Station/Unit	2. Design Change Number
STEAM GENERATOR REPLACEMENT / NORTH ANNA / UNIT 1	DC 90-13-1

9. Engineering Review and Design:

and branch line support evaluations which concluded that all branch lines and branch line supports remain acceptable. The evaluation of each branch and its affected supports are shown to be acceptable in their respective calculations (References 6.12.24 through 6.12.68).

- The piping and instrument tubing connected to the steam generators will be replaced in the same configuration as they were prior to the replacement. Reference 6.12.18 performed an evaluation of the new seismic RCL movements at the connection points on the SG and determined seismic reanalysis of these components and their supports is not necessary. Calculation 02072.13-NP(B)-007 (Reference 6.12.20) reviewed the structures which support the SG level instrumentation tubing to determine the maximum reactions at the structure's support points. These structures are supported by the SG insulation support bands (supplied by the SG insulation supplier) and the bands are designed to withstand the maximum anticipated loads.

- The feedwater loop seal piping will be replaced with new chrome-moly steel material and the drain valves will be replaced by new valves with live load packing. These changes are evaluated, along with the seismic movements of the SG at the feedwater connection, in References 6.12.26, 6.12.42, and 6.12.43.

- The chemical feed tubing which connects to the feedwater lines at the loop seals will be replaced exactly as they were prior to replacement. References 6.12.47, 6.12.48, and 6.12.49 performed an evaluation of the new seismic RCL movements at the connection points and determined seismic reanalysis of these components and their supports is not necessary.

- The SG blowdown piping will be modified to accommodate a larger pipe size and new chrome-moly steel material. Also, the valves will be replaced with new live load packing valves. The modified blowdown piping/support system has been seismically qualified by SWEC and documented in References 6.12.1 through 6.12.4.

The electrical components installed under this design change are temporary and are required only during the SGR. All electrical components installed will be removed after the SGR.

The temporary construction power supply requirements will be fed from non-Class 1E power sources. As adequate isolation exists between these non-Class 1E power sources and any Class 1E sources or loads, no Class 1E power systems will in any way be degraded or jeopardized during a seismic event.

The replacement SG insulation is seismically qualified per the Procurement Specification NAP-0047.

The acceptability of interim configurations of the RCL, main steam, feedwater, and other plant secondary side piping systems is documented in Calculations 02072.13-NP(B)-008-XE and 02072.13-NP(B)-009-XE (References 6.12.21 and 6.12.22, respectively).

Based on the previously stated reasons and the calculations referenced above, this modification does not adversely impact seismically qualified safety-related equipment.

1. Design Change Title/Station/Unit	2. Design Change Number
STEAM GENERATOR REPLACEMENT / NORTH ANNA / UNIT 1	DC 90-13-1

9. Engineering Review and Design:

3.9 HUMAN FACTORS

This design change results in a change to Emergency Operating Procedures. The setpoint for manual RWST switchover to recirculate mode will change from 29% to 22.8% RWST level. The change provides additional water to the sump prior to switchover, which results in increased NPSH for the LHSI pump. This increase was necessary to overcome an NPSH penalty imposed by Regulatory Guide 1.82 (see Appendices 4-7, 4-23, and 4-28).

The manual switchover setpoint is referenced in the following North Anna Unit 1 EOPs:

1-E-1, Continuous Action Page and Step 21
1-ES-1.1, Continuous Action Page
1-ES-1.2, Continuous Action Page
1-E-3, Continuous Action Page
1-ES-3.1, Continuous Action Page
1-ES-3.2, Continuous Action Page
1-ES-3.3, Continuous Action Page
1-ECA-0.2, Step 2
1-ECA-1.1, Step 9
1-ECA-2.1, Continuous Action Page
1-ECA-3.1, Continuous Action Page
1-ECA-3.2, Continuous Action Page
1-FR-C.1, Caution prior to Step 1
1-FR-C.2, Caution prior to Step 1
1-FR-C.3, Caution prior to Step 1
1-FR-H.1, Continuous Action Page and Caution prior to Step 24

This design change does not add to or modify instruments at the auxiliary shutdown panel or in the control room and, therefore, no further human factors review is required.

Compliance with NUREG-0700, Guidelines for Control Room Design Reviews, is not affected.

3.10 IN-CONTAINMENT BANNED/RESTRICTED MATERIALS

The design change replaces valves in the feedwater and blowdown systems. Both the original and replacement valves have seats containing stellite, a cobalt containing alloy, which is designated as a banned engineered material in the In Containment Materials Standard (STD-MAT-0006). Valves with stellite seats will be utilized in this design because of their availability. Activation of the cobalt that results when the stellite valve seats erode is not a concern because these valves are being installed in the secondary system with no flow path into the primary system.

Temporary construction and cleaning materials will be controlled in accordance with approved project procedures.

3.11 STATION COMPUTER SOFTWARE/HARDWARE

No hardware changes to the station process computer (P250) will result from this design change.

1. Design Change Title/Station/Unit	2. Design Change Number
STEAM GENERATOR REPLACEMENT / NORTH ANNA / UNIT 1	DC 8J-13-1

3. Engineering Review and Design:

P250 software changes identified as a direct result of this design change are as follows:

U0980 PERCENT FULL POWER
Change high limit to 100.4
Change low limit to 99.0

U0981 PERCENT FULL POWER LAST 10 MINUTES
Change high limit to 100.2
Change low limit to 99.0

U0982 CALORIMETRIC - TOTAL THERMAL POWER
Change high limit to 2900.0

U0983 CALORIMETRIC - LOOP A THERMAL POWER
Change high limit to 969.0

U0984 CALORIMETRIC - LOOP B THERMAL POWER
Change high limit to 969.0

U0985 CALORIMETRIC - LOOP C THERMAL POWER
Change high limit to 969.0

U0986 RUNNING 12 HOUR SHIFT AVERAGE POWER
Change high limit to 100.0
Change low limit to 99.0

T0410A RCL1 OVERTEMP DT 1 SP
Change high limit to 149.0

T0430A RCL1 OVERTEMP DT 2 SP
Change high limit to 149.0

T0450A RCL1 OVERTEMP DT 3 SP
Change high limit to 149.0

K0316 SG Q TILT LIM PWR TOLERANCE
Change to 10

C00001A thru C0064A Control Rod Bank Positions
Change high limit to new rod withdrawn positions

3.12 EMERGENCY RESPONSE FACILITIES SYSTEM

The design change does affect the ERF computer system. ERF Design Controlled Documents are affected. The controlled document UFSAR and the North Anna Unit 1 PLS document will require updating.

1. Design Change Title/Station/Unit	2. Design Change Number
STEAM GENERATOR REPLACEMENT / NORTH ANNA / UNIT 1	DC 90-13-1

9. Engineering Review and Design:

The ER Design Checklist form has been completed and included as Appendix 4-5.

3.13 PLANT FLOODING

None

3.14 HEAVY LOADS

Movement of heavy loads inside the containment will be in accordance with the station heavy loads procedure 0-MCM-1303-01 until all fuel has been removed from the containment, the reactor cavity drained, and the fuel transfer tube gate valve closed and the blind flange installed. Movement of heavy loads during the defueled condition will be in accordance with approved project procedures. After defueling is completed, those structures, systems, and components inside the containment which could be impacted by a heavy load drop will not be required to perform a safety function at that time. To preclude any possible adverse effects on stored spent fuel or systems shared with Unit 2 as a result of a postulated steam generator drop, all component cooling water, service water, fuel pit cooling and refueling purification system, and instrument air lines to/from the containment will be isolated during rigging operations for the steam domes and lower assemblies. Thus, no adverse impacts on stored spent fuel or spent fuel cooling would result from a postulated heavy load drop inside the containment.

A non-safety related auxiliary crane will be installed in the containment on the El. 291'-10" operating floor slab to support the handling of loads during the steam generator replacement outage. The auxiliary crane and support tower will not be installed until all fuel has been removed from the containment, the reactor cavity drained, and the fuel transfer tube gate valve closed and blind flange installed. The auxiliary crane and support tower will be removed prior to plant refueling.

A 3-ton capacity jib crane will be temporarily installed inside the containment for use during defueling and refueling operations. As per the guidelines of NUREG-0612, physical limitations will be placed on the swing radius of the crane to ensure approved load paths are followed as designated in the station heavy loads procedure 0-MCM-1303-01 and that no adverse interaction with spent fuel or fuel handling activities will occur. This crane will be replaced, in the same physical location, by the auxiliary crane when defueling is completed. The jib crane will be reinstalled at the end of the outage to assist in containment demobilization.

A special over rated-load lift qualification for the polar crane was performed in accordance with ASME B30.2, Section 2-3.2.1.1, "Special Over Rated-Load Lifts," which concluded that the existing polar crane and the crane supporting structure have sufficient capacity to handle a 280-ton special over rated-load lift (see Appendix 4-31). The 280-ton capacity is greater than the maximum load (including rigging) to be handled by the polar crane during the SGR. See Calculation 21809-C-05 (Reference 6.12.12).

Coincident with equipment hatch removal, containment purge system alignment and operation will be in accordance with Operating Procedure 21.2 so as to minimize any air leakage out of the containment. In addition to the vent stack monitors and the containment area radiation monitors, a continuous air monitor will be in use adjacent to the equipment hatch and periodic air sampling will be performed. A temporary cover will also be available to isolate the hatch opening in case of loss of the containment purge exhaust system. For these reasons, there is a negligible potential for any unmonitored leakage out of the equipment hatch. This notwithstanding, the radiological consequences of a postulated drop of an old steam generator lower assembly inside the containment, during movement from the equipment hatch platform to the transporter, or during movement within the protected area have been evaluated (see Calculation 21809-M-02 which is Reference 6.12.9) and determined to be within applicable regulatory guidelines and less than the limiting case events of the same class of accidents currently evaluated in the UFSAR.

A postulated drop of a steam generator lower assembly adjacent to the equipment hatch could result in damage to the equipment hatch or locally to the containment structure, but significant structural damage to the containment would not occur. Moreover, containment leaktight integrity is not required at this time. Further, no safety-related equipment could be impacted by a postulated drop at this location.

Transportation of the original and replacement steam generator lower assemblies and a load test of the haul routes are required to facilitate steam generator replacement. An evaluation of the steam generator haul routes has been performed and is provided in Appendix 4-18. The objective of the haul route inspection and evaluations was to ensure that the grades and turning radii of the haul route can accommodate the transporter/prime mover, that slopes and embankments will remain stable when subjected to the transporter/prime mover wheel loads, that the right-of-way has sufficient strength to support the transporter/prime mover wheel loads without excessive settlement, and that buried utilities and facilities can withstand the overburden pressures and remain within their allowable load limits. Road improvements and requirements for protection of buried utilities are provided in Appendix 4-18. Limitations on transport of the new and old steam generator lower assemblies and the load test include:

- The maximum transporter speed is 5 mph.
- The transporter leveling capability will be used to maintain the transporter bed level within 5° during movement.
- In accordance with N-9013-1-M-800, the transition cone cover plate "T-sections" shall be installed before the old SG lower assemblies are moved outside the protected area.
- The centerline of the steam generator will be limited to a maximum of 12'-3" off the ground.

The closure plates for the old steam generator lower assemblies have been designed to withstand a load drop from the transporter. See Calculation 21809-C-02 (Reference 6.12.11). This ensures that in the event of a drop of the old steam generator from the transporter during relocation to the steam generator storage facility or during offload at the storage facility, the integrity of the steam generator shell will not be breached. No buried or adjacent facilities will be adversely impacted as a result of a postulated lower assembly or test load drop as concluded in Appendix 4-18.

1. Design Change Title/Station/Unit	2. Design Change Number
STEAM GENERATOR REPLACEMENT / NORTH ANNA / UNIT 1	DC 90-13-1

9. Engineering Review and Design:

3.15 POST-ACCIDENT MONITORING

This design change results in a change to Emergency Operating Procedures. The setpoint for manual RWST switchover to recirculation mode will change to 22.8% RWST level. This setpoint change (see Appendix 4-7) will require a revision to various EOPs, but none will impact the Station's compliance with Regulatory Guide 1.97. The Post-Accident Monitoring Design Checklist is attached as Appendix 4-6.

3.16 HEATING, VENTILATION, AND AIR CONDITIONING (HVAC)

This design change does not involve any new permanent heat loads; therefore, the existing HVAC system is not impacted. However, this DCP will temporarily modify the containment purge system so that it will be more effective during the replacement outage. This modification will not adversely affect the ability of the purge system to perform its design function. Prior to fuel load, the modification will be removed.

3.17 THE SIMULATOR

The Model 51F steam generators have been designed to perform as equivalent replacements for the existing Model 51 steam generators. It has been demonstrated that the behavior of the two SG designs during either transient heatup or cooldown events is essentially identical. For full power steady-state operation at any given RCS average temperature, Westinghouse projects that the replacement SGs will produce greater SG outlet pressure than the existing SGs. The magnitude of this difference will depend upon the value of RCS T_{ave} at which the system is operated following SGR. Unit 1 will operate at an RCS T_{ave} of 580.8 °F after SGR, which is equal to the Unit 2 value. At this T_{ave}, it is expected that steam generator pressure will be approximately equal to the design value of 850 psia. Therefore, it is expected that both North Anna Units would have similar characteristics which approximate the original design conditions, such that no changes to simulator steady-state inputs are required.

License Amendments 153 and 154 provide the values of several setpoints which were revised for operation with extended SG tube plugging, but which revert to the prior value for operation after SGR. The temporary changes each have a note which indicates that the change is effective for the period of operation until SGR. The simulator setpoints should be verified (and changed if needed) to be consistent with the Technical Specification setpoints applicable for operation after SGR.

In addition, setpoints listed in the PLS document were altered for extended plugging operation. Each of these will also revert to the prior value for operation after SGR. These values are provided in Appendix 4-7.

The LOCA containment analysis has been revised for operation with the replacement SGs. The revised analysis requires a change in the low head safety injection recirculation transfer setpoint, used to initiate switchover from RWST suction to sump suction. It was necessary to make this change in order to confirm the acceptability of analysis margins for key containment analysis parameters in the revised analysis. The analysis assumed that the existing setpoints for both manual and automatic switchover are reduced by approximately 5.5% of the span from current values. We have determined setpoint values, based upon the existing setpoint calculation and our safety analysis

1. Design Change Title/Station/Unit	2. Design Change Number
STEAM GENERATOR REPLACEMENT / NORTH ANNA / UNIT 1	DC 90-13-1

9. Engineering Review and Design:

values. The values are provided in Appendix 4-7.

It should be noted that after this change, Unit 1 and 2 will have different setpoint values for LHSI recirculation transfer. Typically, the North Anna simulator represents the Unit 1 operations. When differences between units exist, they are handled through training. It is anticipated that this approach will also apply here and that the simulator values will be changed to reflect the Unit 1 parameters.

The PLS and setpoint documents in Appendix 4-7 will be reviewed for impact on the simulator.

3.18 NUCLEAR PLANT RELIABILITY DATA SYSTEM (NPRDS)

NPRDS reviews were conducted as part of the equipment selection process. Replacement steam generator lower assemblies and various secondary system valves were reviewed to determine if these components will be installed in compatible systems and environments.

No failure reports were filed against the selected engineered equipment for this DCP. Neither of the components were identified as having unresolved 10 CFR 21 problems. The final NPRDS printout is included as Appendix 4-29.

3.19 SET POINTS, INSTRUMENT ACCURACY, AND SCALING

Several reactor trip setpoints were modified to implement NA-1 Technical Specification Amendment No. 154. These changes were made and documented in accordance with EWR 92-092 and included revisions to an Overpower Delta T Coefficient (K_1), an Overtemperature Delta T Coefficient (K_4), the Power Range High Level Trip Setpoint and the Power Range High Flux Level Rodstop Alarm. Technical Specification Amendment No. 154 did not replace the previous setpoints with revised values, but instead placed conditional alternate setpoint values into the Technical Specification. The condition necessary to invoke the alternate setpoints values will remain in effect until the SGs are replaced. This approach permits the SG replacement to proceed without an additional Technical Specification amendment. However, upon completion of the SG replacement activity, these setpoints must return to the original values that existed prior to Amendment No. 154. The Setpoint Change Form is included as Appendix 4-7.

To confirm the existence of adequate analysis margin in available NPSH for the LHSI pumps, it was necessary to reduce the RWST setpoints for LHSI recirculation transfer. The setpoints for both manual and automatic recirculation transfer were reduced by 5.5% from their existing values. This reduction was implemented in conjunction with a revised assumption for duration of the automatic LHSI recirculation transfer sequence. The RWST volume changes and reduced automatic transfer duration are documented in Appendix 4-23. The total sequence duration was selected to be 240 seconds, which has been confirmed to exceed the summation of individual stroke time limits for the valves involved in the transfer sequence, as contained in Periodic Test procedure 1-PT-213.8, "Valve Inservice Inspection - Safety Injection System, Rev. 4." No change to 1-PT-213.3 is required since the stroke times in the PT are more restrictive (smaller) than the revised analysis values. This is acceptable, since the procedure requires that any valve which cannot meet its acceptance criterion be declared inoperable, and the required Technical Specification actions taken. Calculation EE-0093, "RWST Level Calibration Values," has been revised to reflect this setpoint change for Unit 1. There is no setpoint change for Unit 2. The Set Point Change Form is included as Appendix 4-7.

1. Design Change Title/Station/Unit	2. Design Change Number
STEAM GENERATOR REPLACEMENT / NORTH ANNA / UNIT 1	DC 90-13-1

9. Engineering Review and Design:

Due to the internal characteristics of the repaired steam generators, the steam generator level uncertainty calculation EE-0492 (Reference 6.12.102) will be generated based on revised Process Measurement Accuracies which will be provided by Westinghouse. This calculation will be completed prior to unit startup.

The new steam generators will utilize the same type and configuration of instrumentation as was used with the original SGs. Data from the Westinghouse Thermal-Hydraulic Design Data Report (see Appendix 4-26) was used to determine scaling factors. These scaling factors are documented in Technical Report EE-0085 (Reference 6.6) and will be incorporated into upgraded Instrument Calibration Procedures (ICPs). These ICPs are normally performed during each refueling outage, however, they have also been included in the testing section of this DC as they will be validated with startup measurements.

Listed below are the instrumentation loops that are affected by the new Model 51F Steam Generators. Please refer to North Anna Power Station Drawings NA-DW-108D014 series and the relevant drawings in the NA-DW-6007 & 6008 series for further information.

T_{AVE} and ΔT Loops

The coefficients associated with the OTΔT and OPΔT setpoint equations are to be returned to their original values. Note that the T_{AVE} and ΔT loops are being scaled for the RTD Bypass Elimination up to the point where the new Median / High T_{AVE} and ΔT values are developed. This Median / High T_{AVE} is used to various control functions to be compared with a reference value, T_{REF}, to produce a control signal. These control functions are Rod Speed and Direction, Pressurizer Level Control and Steam Dump control. The new T_{AVE} value of 580.8 °F is also used in the Control programs for the above functions.

Turbine Impulse Pressure

Present First Stage Impulse Pressure (TIP) values for 100% power level are presently in the region of approximately 590 psig. The present span for PT-1446 and PT-1447 is 600 psig. Mechanical Engineering have estimated the new 100% power TIP value to be between 595 and 600 psig. A change in the 100% power value for TIP will affect the following instrument loops: Rod Speed and Direction, Steam Dump Control, High Steam Flow Setpoint Summer and Steam Generator Level Control. The 100% TIP value is used to produce the various control or setpoint programs for the above functions.

Rod Speed and Direction

The T_{REF} program is an input to Rod Speed and Direction and uses the 100% power level values of TIP and T_{AVE} to define it's program. This program will be developed with the "preliminary TIP value" provided by Mechanical Engineering and may have to be refined with actual as measured plant data. The P_{REF} program for the power mismatch circuit also uses the 100% TIP value. The preliminary scaling provided for both T_{REF} and P_{REF} may need to be revised according to data obtained during plant start up.

1. Design Change Title/Station/Unit	2. Design Change Number
STEAM GENERATOR REPLACEMENT / NORTH ANNA / UNIT 1	DC 90-13-1

9. Engineering Review and Design:

Steam Dump Control

The steam dump control program for the Load Rejection mode of T_{AVE} control is similar to that of Rod Speed and Direction. This will require testing to ensure correct functional operation of this instrument loop.

Pressurizer Level Control

The L_{REF} program requires the 100% T_{AVE} value of 580.8 °F, the 64.5 % level value (full power) and the 28.4% level value (zero power) given by the Precautions, Limitations and Setpoints (PLS) Document to define the level program.

Steam Flow/Feed Flow

Revised scaling will be provided for start-up and may require revision as a result of the data obtained from the flow test.

High Steam Flow Setpoint

This requires an input from Turbine First Stage Pressure to generate the setpoint, i.e. ΔP_{REF}. The actual ΔP value obtained from the steam flow transmitters is compared to this ΔP_{REF} value. Preliminary scaling will be provided for the ΔP_{REF} program and may need to be revised according to data obtained during the flow test.

Steam Generator Level Control

Preliminary scaling based on the data obtained from the revised Model 51F Steam Generator Thermal and Hydraulic Design Data Report for Virginia Power Company, North Anna Unit 1, WNEP 9110, for T_{AVE} = 580.8 °F will be provided for this control system.

Steam Pressure

A preliminary value for steam pressure obtained from the revised WNEP 9110 will be used for the scaling of the steam flow NMD cards. This parameter provides the density compensation so as to obtain a true steam flow value on the output of this card. This preliminary scaling may need to be revised according to data obtained from a flow test.

RCS Flow

Replacing the steam generator lower assemblies will cause the RCS flow rate to increase over the current value. This increase is a result of tube plugging in the existing steam generators as compared to no plugged tubes in the new steam generator lower assemblies. RCS flow instruments are scaled to read approximately 100% flow at 100% power; hence, best estimate RCS flow scaling will be provided prior to startup which re[illegible]nts the increase in RCS flow rate. This re-scaling will also preclude changing the low flow [illegible]oint which is a percentage of 100% flow.

3.20 SECONDARY PIPING AND COMPONENT INSPECTION PROGRAM

The components and piping for the SGR project are similar to the original configuration. However, the existing feedwater loop piping will be removed and replaced with new chrome-moly piping (piping class 601C) to minimize erosion/corrosion effects. Also, the SG blowdown piping from SG nozzles to the existing 3 in. header will be replaced with new chrome-moly piping (piping class 601 . The erosion/corrosion rates will not be adversely affected.

The main steam, feedwater, and blowdown systems are currently being tracked for erosion/corrosion concerns as part of the Secondary Piping and Component Inspection Program. The existing erosion/corrosion isometric drawings and database for the steam generator feedwater, main steam, and blowdown piping systems will be updated to reflect the system enhancements to be implemented by this Design Change. Mark-ups of the affected erosion/corrosion drawings and the required baseline inspections to be performed are included as Appendix 4-36.

Piping material and size changes in the feedwater and blowdown systems have required that the line designation tables be updated. Mark-ups of the line designation tables are included as Appendix 4-32.

3.21 RADIO FREQUENCY INTERFERENCE REVIEW

This design change does not involve the addition of any new electronic equipment, nor does it modify any existing electronic equipment; therefore, a radio frequency interference review is not required.

3.22 Q-LIST

This design change affects the steam generators and other components on the Q-List. It revises data for certain components but does not add any new components to the Q-List. The feedwater, blowdown, and chemical feed sections of the Q-List will require revision to incorporate the data related to the thirty-six (36) new replacement valves being added by this Design Change.

A Q-List change request has been prepared and is included as Appendix 4-4 to the DCP.

3.23 ALARA ANALYSIS

Various techniques will be employed during the steam generator replacement outage to ensure that personnel exposures are maintained as low as reasonably achievable (ALARA). In general, the objectives of each technique are to reduce the time required to perform the task, increase the distance between radiation sources and workers, and/or decrease the dose rates to workers by providing intervening shielding and/or by decontaminating exposed surfaces.

The ease of accessibility and the effective control and utilization of Radiological Control Area (RCA) work-spaces are essential for minimizing exposure while working in potentially high radiation areas. Onsite investigations by design engineering teams will be made to confirm adequate clearances for the movement of tools and equipment, to identify potential physical interferences, and to ensure that there is adequate space for the ingress and egress of personnel and for the efficient performance of required tasks.

1. Design Change Title/Station/Unit

STEAM GENERATOR REPLACEMENT / NORTH ANNA / UNIT 1

2. Design Change Number

DC 90-13-1

9. Engineering Review and Design:

Temporary shielding will be used to maintain exposures from nearby components ALARA. Decisions involving the use of temporary shielding will be made on a case-by-case basis, weighing man-rem saved against man-rem expended to shield to maintain dose rates at acceptably low levels.

The secondary side of the original steam generators will be kept full of water as long as possible prior to lifting to provide shielding from the radioactive material contained inside the primary tubes. The steam generator replacement outage will involve significant personnel exposure due to the general level of radioactive material contained inside the primary tubes and the radiation levels in the immediate areas where work is to be performed.

Exposure will be minimized by performing as much work as possible in low or nonradiation areas and by decontaminating the work area.

To minimize total personnel contamination events resulting from the steam generator replacement, a general decontamination of the containment building will be performed during the initial phases of the outage.

The reactor refueling cavity will be decontaminated to minimize the potential for airborne radioactivity generation in accordance with station-approved procedures.

After the initial decontamination, additional surface contamination generated from the replacement process will be removed as needed by an ongoing decontamination program.

Decontamination of the reactor coolant system pipe ends will also be performed. An abrasive-blast decontamination of the internal surfaces of the reactor coolant system hot/cold leg pipe ends will reduce overall exposure during the machining and welding processes.

Temporary shielding of residual hot spots will be used if it will result in a significant exposure reduction after decontamination. Protective clothing will be required and respirators will be used as necessary.

In addition to the pipe decontamination described previously, shielding of reactor coolant piping will be provided to reduce dose rates to workers involved in the cutting, pipe end preparation, and welding of the reactor coolant pipe ends.

Mock-up training and specialized tools will be used in a number of instances to minimize exposures. Automated cutting, milling, and welding machines will be used, to the extent practical, to reduce doses in high radiation areas. These machines typically run on tracks attached to the pipes and allow personnel to stand back in a lower radiation area to observe the correct operation of the machines.

These ALARA considerations are taken into account in the implementation and work procedures. At the time of the steam generator replacement outage, health physics personnel will review the work areas and access paths to identify radiation hot spots and to determine whether use of temporary shielding is required. Low-radiation working areas will also be identified. Radiation work permits will be issued as required and prework briefings will be conducted as necessary. Latest radiation surveys will be reviewed to ensure that radiation levels in the work areas have not significantly increased from those reported previously. Station health physics personnel will establish the acceptability of radiation levels upon completion of decontamination.

Dose rates for the various work areas must be determined at the time of actual installation.

The estimated total radiation exposure to perform the replacement on the steam generators is 571 man-rem, which does not include the dose estimate associated with DC 89-40-1 which removes the RTD piping but the estimate assumes the RTD piping has been removed early in the SGR outage.

The Engineering ALARA Evaluation and Design Checklist has been completed for this design change (see Appendix 4-8).

3.24 CUMULATIVE EFFECTS ON PLANT SYSTEMS

None

3.25 RECENT NRC AND INDUSTRY CONCERNS

Recent NRC and industry concerns and a brief discussion of these concerns applicable to the proposed modification are as follows:

a. Generic Letter 86-16, Westinghouse ECCS Evaluation Models

Generic Letter 86-16 identifies licensee actions required for PWR plant whose ECCS analyses use either the 1978 or 1981 versions of the Westinghouse ECCS Evaluation Model that contain the WREFLOOD or BART codes. The NA-1 large break LOCA ECCS analysis, being implemented for operation following steam generator replacement, uses the 1981 version of the Westinghouse ECCS Evaluation Model containing the BART code.

The analysis to be implemented for Unit 1 operation after steam generator replacement incorporates all GL 86-16 required actions pertaining to the large break LOCA ECCS analysis.

b. NRC Bulletin 88-02, Rapidly Propagating Fatigue Cracks in Steam Generator Tubes

The purpose of Bulletin 88-02 was to request that holders of operating licenses or construction permits for Westinghouse designed nuclear power reactors with steam generators having carbon steel support plates implement actions to minimize the potential for a steam generator tube rupture caused by a rapidly propagating fatigue crack such as that which occurred at NA-1 on July 15, 1987.

No analysis is required for the replacement steam generators since they have stainless steel tube support plates.

c. Information Notice 88-06, Foreign Objects in Steam Generators

The purpose of Information Notice 88-06 was to alert addressees to a potentially generic problem with foreign objects on the secondary side of steam generators in PWRs and the potential for failure of steam generator tubes as a result of fretting. The information notice was issued following the discovery of foreign metal objects in steam generators at Catawba 2 during the plant's first refueling outage in early 1988. Although the information notice requires no formal response and no specific actions are recommended, Virginia Power is sensitive to the potential problems that could be caused by loose foreign objects in the steam generators. As such, the following measures will be taken to avoid this problem during the steam generator replacement.

Adherence to material accountability procedures was required of all major suppliers and sub-suppliers during fabrication of the new lower assemblies and assembly of the new steam generators. In addition, frequent inspections were performed throughout different fabrication and assembly stages, up to and including the final pre-service inspections of the new steam generators. Vigorous Quality Assurance oversight was maintained, including review of procedures, witnessing of inspections, and review of inspection records.

The information notice indicates that pre-service visual inspections were performed on the Catawba 2 steam generators after fabrication and prior to initial reactor operation, however, nothing of significance was found at that time. It is speculated by the NRC that the foreign objects found later on the secondary side of the tubesheet may have accumulated in the top works of the steam generators, and were later washed down to the tubesheets after the pre-service inspections. The pre-service inspections at Catawba 2 were performed according to applicable requirements of NRC Generic Letter 85-02, however, the NRC has acknowledged in the information notice that the recommended visual inspection is concentrated in the vicinity of the tubesheet and would not detect foreign objects on the top works. On this basis, the NA-1 final pre-service inspections will address the need to visually inspect the steam generator steam domes and secondary side tubesheet to ensure that no foreign objects are present.

d. <u>Information Notice 88-31, Steam Generator Tube Rupture Analysis Deficiency</u>

Information Notice 88-31 was issued to notify addressees of potential problems resulting from a non-conservatism in the safety analysis for rupture of a steam generator tube which may increase offsite dose consequences. This information notice was issued as a result of new analyses performed by Westinghouse of certain design basis events, including a steam generator tube rupture, for Surry following the July 15, 1987 tube rupture event at NA-1 (see item b.). This reanalysis of Surry found that during a steam generator tube rupture event, the water level on the secondary side could fall below the top of the steam generator tubes. This finding is significant because if the break location is uncovered, a direct path for fission products in the reactor coolant to be released to the atmosphere without partitioning by the secondary side coolant could be created.

This issue is currently being addressed by the Westinghouse Owners Group, of which Virginia Power is a member.

e. <u>NRC Bulletin 89-01, Failure of Westinghouse Steam Generator Tube Mechanical Plugs</u>

Bulletin 89-01 requested that addressees determine whether certain mechanical plugs supplied by Westinghouse are installed in their steam generators, and if so, that an action plan be implemented to ensure that these plugs will continue to provide adequate assurance of reactor coo'ınt system pressure boundary integrity under normal operating, transient, and postulated accident conditions. This request was the result of the mechanical plug failure experienced at NA-1 on February 25, 1989. The plug failure caused a 75 gpm primary-to-secondary leak. The failure mechanism involved a full circumferential severance of the top portion of the plug from the body of the plug. The top portion of the plug was propelled up the length of the affected tube by reactor coolant system pressure to a point just above the U-bend tangent point where it impacted and punctured the outer curvature of the tube and subsequently impacted and dented an adjacent tube.

The use of Westinghouse mechanical plugs fabricated from Inconel-600 material has been discontinued at both North Anna Units.

f. <u>Information Notice 89-33, Potential Failure of Westinghouse Steam Generator Tube Mechanical Plugs</u>

Information Notice 89-33 was issued to addressees to alert them to the potential for failure of Westinghouse steam generator tube mechanical plugs. This information notice describes the same problem addressed in item e.

Due to steam generator replacement for North Anna Unit 1, no further plug work is required.

g. <u>Information Notice 89-65, Potential for Stress Corrosion Cracking in Steam Generator Tube Plugs Supplied by Babcock and Wilcox</u>

Information Notice 89-65 was issued to addressees to inform them of the status of primary water stress corrosion (PWSCC) problems being experienced with steam generator tube plugs supplied by Babcock and Wilcox.

Due to steam generator replacement for North Anna Unit 1, no further plug work is required.

h. <u>Information Notice 90-04, Cracking of the Upper Shell-to-Transition Cone Girth Welds in Steam Generators</u>

Information Notice 90-04 was issued to alert addressees to continuing problems related to cracking of the upper shell-to-transition cone girth welds in the steam generators originally described in Information Notices 82-37 and 85-65 (Zion 1 and Indian Point 2). Cracks and linear indications on the inner circumference of the upper shell-to-transition cone girth weld have been detected in 18 steam generators in the U.S. These flaws have only been observed in Westinghouse Model 44 and 51 vertical recirculating U-tube steam generators with the feedwater ring design.

Inspection of the new closure weld for cracking after post weld heat treatment will be performed.

1. Design Change Title/Station/Unit

STEAM GENERATOR REPLACEMENT / NORTH ANNA / UNIT 1

2. Design Change Number

DC 90-13-1

9. Engineering Review and Design:

i. Information Notice 90-07, New Information Regarding Insulation Material Performance and Debris Blockage of PWR Containment Sumps

Information Notice 90-07 is intended to alert licensees to new information concerning performance of insulating materials in post loss-of-coolant-accident environments within PWR containment sumps when such materials become debris which may block sumps.

The existing thermal insulation on the steam generators will be replaced with new fiberglass blanket type insulation (Specification NAP-0047). The quilted, light density, semi-rigid fibrous glass blankets will utilize stainless steel hook-and-loop fasteners and will be covered by a stainless steel jacket. A debris and NPSH analysis has been performed and determined the new insulation to be acceptable (see Appendix 4-28).

j. Information Notice 90-10, Primary Water Stress Corrosion Cracking (PWSCC) of Inconel 600

Information Notice 90-10 was issued to alert addressees to potential problems related to primary water stress corrosion cracking (PWSCC) of Inconel 600 that has occurred in pressurizer heater thermal sleeves and instrument nozzles.

Inconel 690 is the tube material for the replacement lower assemblies, therefore this information notice is not applicable.

k. Information Notice 90-49, Stress Corrosion Cracking in PWR Steam Generator Tubes

Information Notice 90-49 is intended to alert addressees of recent problems involving stress corrosion cracking in PWR steam generator tubes particularly in SCC detection during inservice inspections.

The Alloy 690 tubing and the stainless steel tube support plates are expected to provide increased resistance to stress corrosion cracking.

l. The following documents discuss concerns related to the thinning of pipe walls in Nuclear Power Plant piping systems:

Information Notice 91-18, High-Energy piping Failures Caused by Wall Thinning.

Generic Letter 89-08, Erosion/Corrosion-Induced Pipe Wall Thinning.

NRC Bulletin 87-01, Thinning of Pipe Walls in Nuclear Power Plants.

These generic communications are intended to alert addressees to continuing erosion/corrosion problems affecting the integrity of high-energy piping systems and apparently inadequate monitoring programs. The piping failures at domestic plants indicate that, despite implementation of long-term monitoring programs pursuant to Generic Letter 89-08, "Erosion/Corrosion-Induced Pipe Wall Thinning," Piping failures caused by wall thinning continue to occur in operating plants.

1. Design Change Title/Station/Unit	2. Design Change Number
STEAM GENERATOR REPLACEMENT / NORTH ANNA / UNIT 1	DC 90-13-1

9. Engineering Review and Design:

Virginia Power has conducted inspections at the site as part of the Erosion/Corrosion program and no additional action was required.

All replacement piping will meet or exceed the specifications of the original piping with respect to resistance to erosion/corrosion. Portions of the blowdown and feedwater piping systems are being upgraded with a chrome-moly allow (Class 601C) to improve resistance to erosion/corrosion concerns.

m. Information Notice 91-19, Steam Generator Feedwater Distribution Piping Damage

Information Notice 91-19 is intended to alert addressees to potential problems resulting from degradation of feedwater distribution piping in steam generators due to thermal stress, cracking, erosion and corrosion. Depending on the design of the steam generator feedwater system, these problems may affect operation of the auxiliary feedwater system.

Virginia Power indicates no additional actions. Mechanical Engineering ISI/NDE Program departments will continue follow-up efforts to determine the potential for erosion/corrosion to the steam generator feedrings.

n. Information Notice 91-28, Cracking In Feedwater System Piping

Information Notice 91-28 is intended to inform addressees of the issuance of a document which closes out NRC Bulletin 79-13, "Cracking in Feedwater System Piping," and actions taken by some licensees to perform augmented inspections of their feedwater lines as part of their inservice inspection programs.

Volumetric examinations were performed on feedwater weld areas and snubbers in containment. Modifications of cracked feedwater piping has been completed.

o. Information Notice 91-38, Thermal Stratification In Feedwater System Piping

Information Notice 91-38 is intended to alert addressees to feedwater system piping that could be subjected to thermal stratification and cause unacceptable pipe movement.

The replacement feedwater piping will have the same configuration as it did before the SGR outage. Therefore, no new concerns related to stratification will be created. The material upgrade for the loop portion of the feedwater piping system should not have any effect on this issue.

p. Information Notice 91-43, Recent Incidents Involving Rapid Increases In Primary-To-Secondary Leak Rate

Information Notice 91-43 is intended to inform addressees of recent incidents involving very rapid increases in the primary-to-secondary leak rate. One of these incidents was followed by a steam generator tube rupture (SGTR). The leakage during these incidents increased at rates that were significantly higher than would be predicted on the basis of Figure 1 of

Bulletin 88-02, "Rapidly Propagating Fatigue Cracks in Steam Generator Tubes."

Current administrative limits for primary-to-secondary leakage at North Anna are far more conservative than those suggested by Figure 1 of Bulletin 88-02 and, combined with other leakage detection and monitoring methods, are adequate to ensure detection and plant shutdown prior to a large leak resulting in an SGTR.

Air ejector radiation monitors RM-SV-121/222 provide continuous readout and alarm functions. Air ejector radiation monitor data is adequately utilized for detection of primary-to-secondary leakage at North Anna.

The issue related to the use of N-16 monitors in main steam lines is applicable to North Anna. N-16 monitors 1/2-RI-190/191/192193 provide continuous readout (in gpd) and alarm functions. The N-16 alarm setpoint is administratively controlled to detect significant increases in leakage (on the order of 10 to 20 gpd above prior levels).

Due to the steam generator replacement for NA-1, no further action is required based on the actions in Bulletin 88-02.

q. Information Notice 91-67, Problems With the Reliable Detection of Intergranular Attack (IGA) of Steam Generator Tubing

Information Notice 91-67 is intended to inform addresses of recent problems experienced at the Trojan Nuclear Plant concerning the reliable detection of general intergranular attack of the steam generator tubes at the tube-to-support plate intersections. This notice compliments information notice 90-49.

There should be no effect due to the use of Alloy 690 tubing and a SS tube support plate design. The combination of high velocities in the tube support plate region and the use of corrosion resistant material would minimize the intergranular attack.

3.26 IMPACT OF/ON OTHER DESIGN CHANGES

Certain preparatory work associated with the steam generator replacement will be performed under a separate DCP (DC 90-15-1), "Prep for Steam Generator Repair." Therefore, the implementation of this DCP shall be closely coordinated with DC 90-15-1.

Removal of the floor-mounted pipe whip restraints below the steam generators are covered by DC 90-06-1, "Steam Generator Primary Coolant Pipe Whip Restraint Removal."

Design Change 89-40-1, "RTD Bypass Elimination," involves the elimination of RTD piping.

Design Change 92-006-1, "Optical Templating for SG Replacement," called for the evaluation of optical templating bracket removal in DC 90-13-1. As a result of this evaluation, the brackets will be removed and the anchor bolts cut off and grouted over (see section 2.2.2.14).

1. Design Change Title/Station/Unit

STEAM GENERATOR REPLACEMENT / NORTH ANNA / UNIT 1

2. Design Change Number

DC 90-13-1

9. Engineering Review and Design:

3.27 SUMMARY OF EQUIPMENT ADDED OR REMOVED

Table 3.27-1 Equipment Added/Removed

Mark No.	Add/Rem	Mfg	Model	Location	Function
1-BD-1 (2")	Rem	Vogt	SW-12141	Cont.	SYPB/ISO
1-BD-2 (1")	Rem	Vogt	SW-12141	Cont.	SYPB/ISO
1-BD-4 (2")	Rem	Vogt	SW-12141	Cont.	SYPB/ISO
1-BD-10 (2")	Rem	Vogt	SW-12141	Cont.	SYPB/ISO
1-BD-11 (1")	Rem	Vogt	SW-12141	Cont.	SYPB/ISO
1-BD-13 (2")	Rem	Vogt	SW-12141	Cont.	SYPB/ISO
1-BD-19 (2")	Rem	Vogt	SW-12141	Cont.	SYPB/ISO
1-BD-20 (1")	Rem	Vogt	SW-12141	Cont.	SYPB/ISO
1-BD-22 (2")	Rem	Vogt	SW-12141	Cont.	SYPB/ISO
1-BD-1 (2 1/2")	Add	Conval	2.50-10G4J-F228K	Cont.	SYPB/ISO
1-BD-2 (1")	Add	Conval	1.00-10G2J-F225S	Cont.	SYPB/ISO
1-BD-4 (2 1/2")	Add	Conval	2.50-10G4J-F228K	Cont.	SYPB/ISO
1-BD-10 (2 1/2")	Add	Conval	2.50-10G4J-F228K	Cont.	SYPB/ISO
1-BD-11 (1")	Add	Conval	1.00-10G2J-F225S	Cont.	SYPB/ISO
1-BD-13 (2 1/2")	Add	Conval	2.50-10G4J-F228K	Cont	SYPB/ISO
1-BD-19 (2 1/2")	Add	Conval	2.50-10G4J-F228K	Cont.	SYPB/ISO
1-BD-20 (1")	Add	Conval	1.00-10G2J-F225S	Cont.	SYPB/ISO
1-BD-22 (2 1/2")	Add	Conval	2.50-10G4J-F228K	Cont.	SYPB/ISO
1-FW-256 (3/4")	Rem	Vogt	SW-12141	Cont.	SYPB/DRN
1-FW-258 (3/4")	Rem	Vogt	SW-12141	Cont.	SYPB/DRN
1-FW-260 (3/4")	Rem	Vogt	SW-12141	Cont.	SYPB/DRN
1-FW-256 (3/4")	Add	Conval	0.75-11G2J	Cont.	SYPB/DRN
1-FW-258 (3/4")	Add	Conval	0.75-11G2J	Cont.	SYPB/DRN
1-FW-260 (3/4")	Add	Conval	0.75-11G2J	Cont.	SYPB/DRN
1-FW-70 (3/4")	Rem	Vogt	SW-12141	Cont.	SYPB/IISO
1-FW-71 (3/4")	Rem	Vogt	SW-12141	Cont.	SYPB/IISO
1-FW-72 (3/4")	Rem	Vogt	SW-12141	Cont.	SYPB/IISO
1-FW-73 (3/4")	Rem	Vogt	SW-12141	Cont.	SYPB/IISO
1-FW-74 (3/4")	Rem	Conval	12G2	Cont.	SYPB/IISO
1-FW-75 (3/4")	Rem	Vogt	SW-12141	Cont.	SYPB/IISO
1-FW-70 (3/4")	Add	Conval	0.75-11G3J-S16-3D	Cont.	SYPB/IISO
1-FW-71 (3/4")	Add	Conval	0.75-11G3J-S16-3D	Cont.	SYPB/IISO
1-FW-72 (3/4")	Add	Conval	0.75-11G3J-S16-3D	Cont.	SYPB/IISO
1-FW-73 (3/4")	Add	Conval	0.75-11G3J-S16-3D	Cont.	SYPB/IISO
1-FW-74 (3/4")	Add	Conval	0.75-11G3J-S16-3D	Cont.	SYPB/IISO
1-FW-75 (3/4")	Add	Conval	0.75-11G3J-S16-3D	Cont.	SYPB/IISO
1-FW-76 (3/4")	Rem	Vogt	SW 2801	Cont.	SYPB/IISO
1-FW-77 (3/4")	Rem	Vogt	SW-12141	Cont.	SYPB/IISO
1-FW-76 (3/4")	Add	Conval	0.75-11G3J-S16-3D	Cont.	SYPB/IISO
1-FW-77 (3/4")	Add	Conval	0.75-11G3J-S16-3D	Cont.	SYPB/IISO

1. Design Change Title/Station/Unit

STEAM GENERATOR REPLACEMENT / NORTH ANNA / UNIT 1

2. Design Change Number

DC 90-13-1

9. Engineering Review and Design:

Table 3.27-1 Equipment Added/Removed (cont'd)

Mark No.	Add/Rem	Mfg	Model	Location	Function
1-FW-102 (3/4")	Rem	Vogt	SW-12141	Cont.	SYPB/IISO
1-FW-103 (3/4")	Rem	Vogt	SW-12141	Cont.	SYPB/IISO
1-FW-104 (3/4")	Rem	Vogt	SW-12141	Cont.	SYPB/IISO
1-FW-105 (3/4")	Rem	Vogt	SW-12141	Cont.	SYPB/IISO
1-FW-106 (3/4")	Rem	Vogt	SW-12141	Cont.	SYPB/IISO
1-FW-107 (3/4")	Rem	Vogt	SW-12141	Cont.	SYPB/IISO
1-FW-108 (3/4")	Rem	Vogt	SW-12141	Cont.	SYPB/IISO
1-FW-109 (3/4")	Rem	Vogt	SW-12141	Cont.	SYPB/IISO
1-FW-102 (3/4")	Add	Conval	0.75-11G3J-S16-3D	Cont.	SYPB/IISO
1-FW-103 (3/4")	Add	Conval	0.75-11G3J-S16-3D	Cont.	SYPB/IISO
1-FW-104 (3/4")	Add	Conval	0.75-11G3J-S16-3D	Cont.	SYPB/IISO
1-FW-105 (3/4")	Add	Conval	0.75-11G3J-S16-3D	Cont.	SYPB/IISO
1-FW-106 (3/4")	Add	Conval	0.75-11G3J-S16-3D	Cont.	SYPB/IISO
1-FW-107 (3/4")	Add	Conval	0.75-11G3J-S16-3D	Cont.	SYPB/IISO
1-FW-108 (3/4")	Add	Conval	0.75-11G3J-S16-3D	Cont.	SYPB/IISO
1-FW-109 (3/4")	Add	Conval	0.75-11G3J-S16-3D	Cont.	SYPB/IISO
1-FW-134 (3/4")	Rem	Vogt	SW-12141	Cont.	SYPB/IISO
1-FW-135 (3/4")	Rem	Vogt	SW-12141	Cont.	SYPB/IISO
1-FW-136 (3/4")	Rem	Vogt	SW-12141	Cont.	SYPB/IISO
1-FW-137 (3/4")	Rem	Vogt	SW-12141	Cont.	SYPB/IISO
1-FW-138 (3/4")	Rem	Vogt	SW-12141	Cont.	SYPB/IISO
1-FW-139 (3/4")	Rem	Vogt	SW-12141	Cont.	SYPB/IISO
1-FW-140 (3/4")	Rem	Vogt	SW-12141	Cont.	SYPB/IISO
1-FW-141 (3/4")	Rem	Vogt	SW-12141	Cont.	SYPB/IISO
1-FW-134 (3/4")	Add	Conval	0.75-11G3J-S16-3D	Cont.	SYPB/IISO
1-FW-135 (3/4")	Add	Conval	0.75-11G3J-S16-3D	Cont.	SYPB/IISO
1-FW-136 (3/4")	Add	Conval	0.75-11G3J-S16-3D	Cont.	SYPB/IISO
1-FW-137 (3/4")	Add	Conval	0.75-11G3J-S16-3D	Cont.	SYPB/IISO
1-FW-138 (3/4")	Add	Conval	0.75-11G3J-S16-3D	Cont.	SYPB/IISO
1-FW-139 (3/4")	Add	Conval	0.75-11G3J-S16-3D	Cont.	SYPB/IISO
1-FW-140 (3/4")	Add	Conval	0.75-11G3J-S16-3D	Cont.	SYPB/IISO
1-FW-141 (3/4")	Add	Conval	0.75-11G3J-S16-3D	Cont.	SYPB/IISO

3.28 SYSTEM AND PLANT DESIGN BASIS DOCUMENTS

A review of the North Anna System and Plant Design Basis Documents has been completed and is summarized as follows:

The review of the North Anna Units 1 and 2 Plant Design Basis Documents, Revision 4, has determined that this Design Change will be accomplished in accordance with all applicable requirements and will not result in any changes to the Plant Design Basis Documents.

The review of the North Anna Power Station System Design Basis Document for the Recirculation

1. Design Change Title/Station/Unit	2. Design Change Number
STEAM GENERATOR REPLACEMENT / NORTH ANNA / UNIT 1	DC 90-13-1

9. Engineering Review and Design:

Spray System (SDBD-NAPS-RS-Revision 0) and the Safety Injection System (SDBD-NAPS-SI-Revision 0) has determined that changes to these System Design Basis Documents will be required to address the change in NPSH margin for the inside and outside Recirculation Spray Pumps and the low head safety injection pumps respectively.

An Engineering Change Notice for the design basis document change with proposed revisions has been prepared and is included as Appendix 4-9.

3.29 INSTALLATION SPECIFICATIONS

The following installation specifications and procedures and/or manufacturer's instructions are applicable to this DCP:

- STD-FP-1991-5563, Field Service Procedure - Install Steam Flow Restrictor (Appendix 4-16)
- Westinghouse Model 51F Steam Generator Technical Manual (Reference 6.2)
- Transco Insulation Technical Manual, CNT 390346 (Reference 6.3)

3.30 REMOVABLE BLOCKS AND OTHER BARRIERS

Portions of the bioshield wall will be temporarily removed during construction and will be re-installed prior to refueling. Shielding is not a problem during the time when the steam generator is removed.

3.31 ENVIRONMENTAL IMPACT (NON-RADIOLOGICAL)

None

3.32 MASONRY BLOCK WALLS

None

3.33 NUCLEAR CONTROL ROOM OPERATOR DEVELOPMENT PROGRAM (NCRODP) TRAINING MANUALS

The new SGs incorporate design changes that enhance operation. Therefore, the following NCRODPMs were reviewed for any impact as a result of this design change:

*	NCRODP-23	Main Steam System
	NCRODP-26	Feedwater System
	NCRODP-34	Secondary Sampling System
	NCRODP-37	Steam Generator Blowdown System
*	NCRODP-38	Reactor Coolant System

1. Design Change Title/Station/Unit	2. Design Change Number
STEAM GENERATOR REPLACEMENT / NORTH ANNA / UNIT 1	DC 90-13-1

9. Engineering Review and Design:

*	NCRODP-52	Safety Injection System
*	NCRODP-53	Quench Spray System
*	NCRODP-54	Recirculation Spray System
*	NCRODP-70	Vibration and Loose Parts Monitoring System

The NCRODPs with an * will be impacted by this design change and mark-ups of these sections are included in Appendix 4-10.

3.34 RECOMMENDED SPARE PARTS

A list of recommended operational spare parts is included in Appendix 4-11.

3.35 LABELLING

None

3.36 ABANDONMENT OF EQUIPMENT

None

3.37 VENDOR TECHNICAL MANUALS

The Westinghouse steam generator technical manual presently applies to the Model 51 steam generators at both the North Anna units. This DCP will revise the existing technical manual to only apply to North Anna Unit 2 steam generators (2-RC-E-1A,B and C). The new Westinghouse Model 51F technical manual will be incorporated into a Vendor Technical Manual (VTM) and will apply to the Unit 1 steam generators (1-RC-E-1A,B and C).

The Transco vendor technical manual addresses the new thermal insulation.

Appendix 4-22 contains the VTM change requests.

3.38 REACTIVITY MANAGEMENT

A reactivity management program review has been performed and the results are provided in Appendix 4-13.

3.39 OTHER CONCERNS

None

4.0 <u>FUNCTIONAL TESTING REQUIREMENTS</u>

Upon completion of the steam generator replacement, a testing program will be initiated to return the plant to service. Functional testing requirements are given in Appendix 4-12.

The test procedures and test plan will be developed in accordance with ENAP-0025.

5.0 SYSTEMS DESCRIPTIONS (Operational)

The replacement steam generators are designed to be functionally the same as the old steam generators in an undegraded condition. As such, the impact of the new steam generators will be seen by the operators as an ability to achieve design power levels while operating at a reduced T_{avg} of 580.8°F. The secondary systems are basically unaffected by this replacement, but will be verified and scaled in accordance with Section 3.19. Because of the increase in the number of SG tubes, a slightly higher reactor coolant flow is also anticipated. The Westinghouse Thermal Hydraulic Report (Appendix 4-26) for the replacement steam generators provides the anticipated operating parameters for North Anna Unit 1 following replacement of the steam generators. Using the revised operating parameters provided in Appendix 4-26, all existing station procedures have been evaluated for potential change. The Controlled Document Summary lists the procedures that require changes.

As a result of the re-evaluation to confirm adequate NPSH at the containment sump following design basis accidents, a setpoint change has been initiated. This change will lower the RWST level at which the LHSI system switches over and begins taking a suction on the containment sump. This re-evaluation was required due to the replacement of SG insulation. The setpoint change will require revisions to the ERF computer system and to station Emergency Operating Procedures (EOPs). Affected EOPs are listed in the Controlled Document Summary

There will be no changes to the system function as a result of the steam generator replacement covered under this design change. There will be changes required to the existing plant operational and maintenance procedures due to additional handholes and inspection ports. These are described in the Westinghouse Technical Manual for the steam generators.

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

While the function of the replacement steam generators is essentially the same as the original undegraded steam generators, certain enhancements have been incorporated into the new SG design which will improve resistance to future tube degradation and facilitate certain maintenance activities. These changes are summarized below:

a. The replacement steam generator blowdown system design will be capable of responding more rapidly to steam generator chemical imbalance conditions. The replaced steam generators are designed to operate at a blowdown flow rate of 3 percent of steam generator maximum steam flow per blowdown line (or a total of 6 percent of maximum steam flow if both blowdown lines are used) for a cumulative period of 1 year and at a flow rate of 1.5 percent of steam generator maximum steam flow per blowdown line (or a total of 3 percent of maximum steam flow if both blowdown lines are used) for the remainder of the steam generator service life. While the replaced steam generator

1. Design Change Title/Station/Unit	2. Design Change Number
STEAM GENERATOR REPLACEMENT / NORTH ANNA / UNIT 1	DC 90-13-1

9. Engineering Review and Design:

and inside containment blowdown piping will support increased blowdown flow, the outside containment blowdown system is unchanged. Therefore, there will be no operational change to the manner in which blowdown is currently operated.

b. The replaced steam generators will be provided with primary nozzle closure rings designed to interface with Westinghouse designed primary nozzle closure covers (nozzle dams). The closure rings are permanently welded to the inside surface of the steam generator channel head, concentric with each primary nozzle.

The addition of primary nozzle closure rings will provide a maintenance option/backup in lieu of loop stop valve closure which will allow primary side maintenance tasks to be conducted simultaneously with reactor vessel refueling outage activities that require a flooded cavity.

The primary nozzle closure ring around each of the channel head bowl nozzle openings has no impact, i.e., no measurable pressure drop increase or flow rate decrease, due to its presence; nor does it prevent ultrasonic in-service inspection of the primary nozzle internal corner junction from outside the head.

c. Four additional hand holes have been added for better access to the tube bundle. These can be used for inspections, sludge lancing and foreign object retrieval if required. Additionally, a mark has been placed at the 7th tube support plate tube lane location should we elect to install another port at this location in the future.

d. A new flow distribution baffle plate has been installed to distribute flow over the tubesheet. This should minimize the buildup of sludge in the tubesheet.

e. The tube support plate has the quatrefoil design which will minimize sludge buildup on the support plate. This enhancement reduces the possibility of future tube degradation since there is less support plate area for the sludge to buildup on and the open flow areas around the tube avoid stagnate flow locations around the tube/support plate interface.

f. The tube-to-tubesheet weld for the replacement steam generators has been performed to provide a flush tube end at the channel head. If a foreign object gets into the channel head during operations, tube end damage is avoided.

g. The tubes are fully expanded (hydraulically) into the tube sheet. This avoids the over expansion problem experienced previously and also minimizes the tube to tubesheet crevices, thus minimizing the crud build-up over time in this area.

h. Approximately 50% of the tubes have been identified by laser scribe marking on the tube sheet. The tube identification is by row and column and is in OCR characters. This tube identification information should greatly enhance the efficiency of any channel head maintenance work in the future.

i. A new fiberglass blanket type insulation is replacing the existing metal reflective and metal encapsulated fiberglass insulation on the SGs. Experience at other nuclear plants with this new insulation has demonstrated that the blanket type is more efficient and should result in decreased containment temperatures during operation. The new insulation is designed for ease of removal and reinstallation to facilitate maintenance efforts.

j. Portions of the main feedwater system piping and blowdown system piping have been changed from class 601 to class 601C (chrome-moly) as shown on the reference drawings. This material change should result in a decreased susceptibility to erosion/corrosion degradation in the areas of the material upgrade.

The above design enhancements are specifically identified and described in more detail in Section 2.2.

Two steam generator shell ISI welds have been eliminated because of the shell forgings used. However, the two nozzle safe ends have been added to the ISI weld list. Because of the reduced ISI weld sizes, there should be a net decrease in the dose received by personnel performing ISI.

6.0 REFERENCES

NOTE: (*) Indicates documents that require revision as a result of this design change.

* 6.1 Westinghouse Model 51 Steam Generator Technical Manual

* 6.2 Westinghouse Model 51F Steam Generator Technical Manual

* 6.3 Transco Insulation Technical Manual (CNT 390346)

6.4 Project Manual

6.5 Special Processes Manual

6.6 Technical Report EE-0085 "7300 Process Control Scaling Implementing Procedure for North Anna Power Station"

6.7 Quality Assurance Department Instructions Manual

6.8 Virginia Power Radiation Protection Plan

6.9 Inservice Inspection Manual

6.10 Station Drawings

6.10.1	11715-FK-1C	Instrument Piping Reactor Containment, Sh. 3
6.10.2	11715-FP-1A	Main Steam Reactor Containment
6.10.3	11715-FP-2A	Feedwater Reactor Containment
6.10.4	11715-FP-3T	Cubicle No. 1 Piping Reactor Containment

1. Design Change Title/Station/Unit	2. Design Change Number
STEAM GENERATOR REPLACEMENT / NORTH ANNA / UNIT 1	DC 90-13-1

9. Engineering Review and Design:

6.10.5	11715-FP-3U	Cubical No. 1 Piping, Reactor Containment
6.10.6	11715-FP-3X	Cubical No. 2 Piping, Reactor Containment
6.10.7	11715-FP-3Y	Cubicle No. 2 Piping Reactor Containment
6.10.8	11715-FP-3AA	Cubicle No. 3 Piping Reactor Containment
6.10.9	11715-FP-3AB	Cubical No. 3 Piping, Reactor Containment
6.10.10	11715-FP-9A	Reactor Coolant Piping
6.10.11	11715-FP-9B	Reactor Coolant Piping
6.10.12	11715-FP-19A	Sample System Reactor Containment
6.10.13	11715-FM-070B	Flow/Valve Operating Numbers Diagram, Main Steam System
6.10.14	11715-FM-098A	Flow/Valve Operating Numbers Diagram, Steam Generator Blowdown System
6.10.15	13075-FM-102C	Flow/Valve Operating Numbers Diagram, Chemical Feed System
6.10.16	11715-FV-17B	SG Lower Support Assembly
6.10.17	11715-FV-17C	SG Lower Support Weldment, Sh. 1
6.10.18	11715-6.11-21A	Thermowell Installation Detail
6.10.19	11715-6.11-29B	Thermowell Installation Detail
6.10.20	1184J12	Westinghouse, 51 Series Vertical SG Outline - North Anna Unit 1
6.10.21	1184J13	Westinghouse, 51 Series SG General Arrangement
6.10.22	1877E88	Westinghouse, Downcomer Flow Resistance Plates - Installation
6.10.23	4461D85	Westinghouse, Vertical SG Transition Assembly
6.10.24	717J561	Westinghouse, SG (Vertical) General Assembly & Fab.
6.10.25	651B261	Westinghouse, Vertical SG Transition Barrel

1. Design Change Title/Station/Unit

STEAM GENERATOR REPLACEMENT / NORTH ANNA / UNIT 1

2. Design Change Number

DC 90-13-1

9. Engineering Review and Design:

6.10.26	397A133	Westinghouse, Cover Plate
6.10.27	4462D48	Westinghouse, Vertical SG Deck Plate Assembly
6.10.28	4462D49	Westinghouse, Vertical SG Deck Plate Assembly
6.10.29	11715-ESK-5AJ	Elem. Dia. 4160V Ckts. React.Cool. Pp. 1-RC-P-1B
6.10.30	11715-FE-33G	Grdg. Plan & Details React. Containment
6.10.31	11715-FP-3AT	Reactor Containment Annulus Pipe Support Details, Rev.1
6.10.32	11715-FP-3BQ	Reactor Containment Annulus Pipe Support Details, Rev.1
6.10.33	11715-FP-3BM	Reactor Containment Annulus Pipe Support Details, Rev.1
6.10.34	11715-CBM-70B-2 Sheet 1	ISI Classification Drawing Interval 2 Main Steam System
6.10.35	11715-CBM-70B-2 Sheet 2	ISI Classification Drawing Interval 2 Main Steam System
6.10.36	11715-CBM-70B-2 Sheet 3	ISI Classification Drawing Interval 2 Main Steam System
6.10.37	11715-CBM-89B-2 Sheet 3	ISI Classification Drawing Interval 2 Sampling System
6.10.38	11715-CBM-93A-2 Sheet 1	ISI Classification Drawing Interval 2 Reactor Coolant System Loop 1
6.10.39	11715-CBM-93A-2 Sheet 2	ISI Classification Drawing Interval 2 Reactor Coolant System Loop 2
6.10.40	11715-CBM-93A-2 Sheet 3	ISI Classification Drawing Interval 2 Reactor Coolant System Loop 3
6.10.41	11715-CBM-102A-2 Sheet 1	ISI Classification Drawing Interval 2 Chemical Feed Systems
6.10.42	13075-CBM-102A-2 Sheet 1	ISI Classification Drawing Interval 2 Chemical Feed Systems
6.10.43	NA-DW-108D014 Sheets 1-17	Control Systems Block Diagrams

1. Design Change Title/Station/Unit

STEAM GENERATOR REPLACEMENT / NORTH ANNA / UNIT 1

2. Design Change Number

DC 90-13-1

9. Engineering Review and Design:

6.10.44	NA-DW-6007 D01 through D99	Protection System Drawings
6.10.45	NA-DW-6008 D01 through D87	Control Systems Drawings
6.10.46	11715-FV-20F	Reactor Coolant Pump Support Details, Sheet 2
6.10.47	11715-FV-17A	Steam Generator & Coolant Pump Supports Arrangement Plan Composite
6.10.48	11715-FV-17K	Steam Generator & Coolant Pump Supports Arrangement Plan Composite

6.11 Station Specifications

6.11.1	NAS-0014	RCS Piping Components and Elbows
6.11.2	NAS-0017	Carbon Steel and Stainless Steel Gate, Globe and Check Valves, 2-1/2" and Larger
6.11.3	NAS-0023	Manually Operated Forged Stainless Steel and Carbon Steel Gate, Globe and Check Valves and Soicket Weld Ends 2" and Larger
6.11.4	NAS-0033	Steam Generator Upper Restraints
6.11.5	NAS-0047	Steam Generator Thermal Insulation
6.11.6	NAS-232	Steam Generator and Primary Coolant Pump Support Installation
6.11.7	NAS-1009	Installation of Piping and Mechanical Equipment
6.11.8	NAI-0001	Installation of Instrumentation
6.11.9	NAS-1023	Installation of Imbedment Plates and Baseplates Using Anchors Drilled in Concrete
6.11.10	NAS-3018	Installation of Drillco Maxi-Bolt Anchor Bolts
6.11.11	NAS-3014	Specification for Installation of Electrical Equipment
6.11.12	NAS-3012	Criteria Specification for Design and Identification of Electrical Cable Systems
6.11.13	NAI-0014	Grouting and Concrete Repair

1. Design Change Title/Station/Unit	2. Design Change Number
STEAM GENERATOR REPLACEMENT / NORTH ANNA / UNIT 1	DC 90-13-1

9. Engineering Review and Design:

6.12 Station Calculations

SG Blowdown System (Stone & Webster)

6.12.1 02072.07-NP(B)-001-XE, "Pipe Stress Analysis of Steam Generator Blowdown Piping Including Modifications, Cubicle A"

6.12.2 02072.07-NP(B)-002-XE, "Pipe Stress Analysis of Steam Generator Blowdown Piping Including Modifications, Cubicle B"

6.12.3 02072.07-NP(B)-003-XE, "Pipe Stress Analysis of Steam Generator Blowdown Piping Including Modifications, Cubicle C"

6.12.4 02072.07-NP(B)-004-ZB, "Qualification of Pipe Supports for the Steam Generator Blowdown Piping Modifications"

SGR Calculations (Bechtel)

6.12.5 20559-C109-01, "Auxiliary Crane Support Tower"

6.12.6 20559-C110-01, "Runway Beams Inside Reactor Containment"

6.12.7 20559-C110-02, "Reactor Cavity Cover At El 291'-10""

6.12.8 21809-M-134-01, "Steam Generator Level Instrumentation Condensate Pot Wall Thickness"

6.12.9 21809-M-02, "Steam Generator Load Drop Analysis"

6.12.10 20559-C131-04, "Steam Dome Support Stand"

6.12.11 21809-C-02, "Steam Generator Load Drop Analysis"

6.12.12 21809-C-05, "Maximum Load on Polar Crane Hook"

6.12.13 20559-C131-08, "Temporary Support for SG Upper Restraint"

6.12.14 20559-SGR-E-001, "Temp. Power Distribution For Steam Generator Replacement"

RCL Evaluation (Stone & Webster)

6.12.15 02072.13-NP(B)-001-X, "Evaluation of the Unit 1 Reactor Coolant Loop for the Repaired Steam Generators (Repaired Generator Model 51F)"

6.12.16 02072.13-NP(B)-002-BA, "Analysis of Unit 1 Lower Steam Generator Support and Reactor Coolant Pump Support for Steam Generator Repair"

1. Design Change Title/Station/Unit	2. Design Change Number
STEAM GENERATOR REPLACEMENT / NORTH ANNA / UNIT 1	DC 90-13-1

9. Engineering Review and Design:

6.12.17 02072.13-NP(B)-003-BA, "Analysis of Unit 1 Steam Generator Upper Restraint for Steam Generator Repair"

6.12.18 02072.13-NP(B)-004-X, "Summary of Un 1 Reactor Coolant Loop Branch Line Qualifications Required for the Repaired Steam Generators"

6.12.19 02072.13-NP(B)-005, "Factor of Safety Calculation for Unit 1 RCP Feet, SG Feet and SG Bending Moments After SG Repair"

6.12.20 02072.13-NP(B)-007, "Loads Applied to the Unit 1 Steam Generator Insulation Support Bands by the Steam Generator Level Instrumentation Support Structures"

6.12.21 02072.13-NP(B)-008-XE, "Seismic Evaluation of the Unit 1 Reactor Coolant Loops and Reactor Coolant Loop Components After Cutting the Hot Leg and Crossover Legs at the Steam Generator Nozzles"

6.12.22 02072.13-NP(B)-009-XE, "Evaluation of the Unit 1 Secondary Side Reactor Coolant Loop Branch Lines After Their Severance from the Steam Generator"

RCL Branch Line Calculations (Stone and Webster)

6.12.24 11715-X1-3, "Main Steam Reactor Containment (Loop B)"

6.12.25 11715-X1-4, "Main Steam Reactor Containment (Loop C)"

6.12.26 11715-X2-16, "Steam Generator Feedwater Piping Cubicle B"

6.12.27 11715-X3-113A, "RCS Piping Cold Leg Stop Valve Bypass Lines - SSR-15, 11715-MSK-103AR, Iss. 2, -103AQ, Iss. 2, -103AS, Iss. 2"

6.12.28 11715-X3-304, "Reactor Containment Annulus Piping"

6.12.29 11715-X3-310, "Reactor Containment Annulus Piping"

6.12.30 11715-X3-316, "Reactor Containment Annulus Piping"

6.12.31 11715-X3-320, "Safety Injection Piping - Containment North Anna # 1 - SSR-3 - 11715-MSK-103AD"

6.12.32 11715-X3-321, "Safety Injection Piping - Containment North Anna # 1 - SSR-3 - 11715-MSK-103AF, Iss. 3"

6.12.33 11715-X3-323, "Safety Injection Piping - Containment North Anna # 1 - SSR-3 - 11715-MSK-103AJ"

1. Design Change Title/Station/Unit	2. Design Change Number
STEAM GENERATOR REPLACEMENT / NORTH ANNA / UNIT 1	DC 90-13-1

9. Engineering Review and Design:

6.12.34 11715-X13-280, "Reactor Coolant Loop No. 1 / LHSI & RHR (11715-SSR-2) (11715-MSK-113A, ISS. 5)"

6.12.35 11715-X13-280Z, "Low Head Safety Injection and RHR System"

6.12.36 11715-X13-281, "Reactor Coolant Loop No. 2 / LHSI & RHR (11715-SSR-2) (11715-MSK-113B, ISS. 4)"

6.12.37 11715-X13-282, "Reactor Coolant Loop No. 3 / LHSI & RHR (11715-SSR-2) (11715-MSK-113C, ISS. 4)"

6.12.38 13075.30-NP(B)-03-XE, "Steam Generator Wet Lay-Up System - Cubicle A"

6.12.39 13075.30-NP(B)-04-XE, "Steam Generator Wet Lay-Up System - Cubicle B"

6.12.40 13075.30-NP(B)-05-XE, "Steam Generator Wet Lay-Up System - Cubicle C"

6.12.41 13075.62-NP(B)-4-X2, "Evaluation of Effects Due to Multiple Structure ARS - Unit 1 Main Steam System - Problem 2"

6.12.42 13075.62-NP(B)-5-X2, "Evaluation of Effects Due to Multiple Structure ARS - Unit 1 Steam Generator Feedwater System - Problem 17 Cubicle C"

6.12.43 13075.62-NP(B)-38-X2, "Evaluation of Effects Due to Multiple Structure ARS - Unit 1 Steam Generator Feedwater Reactor Coolant System - Problem 15 Cubicle A"

6.12.44 13075.62-NP(B)-103-XC, "Class 1 Fatigue Evaluation for Piping System / Unit 1, SSR 7, Prob 7223"

6.12.45 13075.62-NP(B)-104-XC, "Class 1 Fatigue Evaluation for Piping System / Unit 1, SSR 7, Prob 7209"

6.12.46 13075.62-NP(B)-105-XC, "Class 1 Fatigue Evaluation for Piping System / Unit 1, SSR 7, Prob 7198"

6.12.47 13075.94-NP(B)-1-X6, "Replacement of Chemical Feed Lines to the Steam Generators - Stress Analysis"

6.12.48 13075.94-NP(B)-2-X6, "Replacement of Chemical Feed Lines to the Steam Generators - Stress Analysis"

6.12.49 13075.94-NP(B)-3-X6, "Replacement of Chemical Feed Lines to the Steam Generators - Stress Analysis"

1. Design Change Title/Station/Unit	2. Design Change Number
STEAM GENERATOR REPLACEMENT / NORTH ANNA / UNIT 1	DC 90-13-1

9. Engineering Review and Design:

6.12.50 14938.22-NP(B)-003-X, "Evaluation of Line 3"CH-1-1502-Q1 (Supplements Teledyne Report E-1473-1, Rev. A (SSR-12)) for Change in OBEA Branch Connection Displacements - Unit 1"

6.12.51 14938.22-NP(B)-004-X, "Evaluation of Lines 4"RC-14-1502-Q1 and 4"RC-15-1502-Q1 (Supplements Teledyne Report E-1473-10 (11715-SSR-5)) for Change in OBEA Branch Connection Displacements - Unit 1"

6.12.52 14938.22-NP(B)-005-X, "Review of Reactor Coolant Loop Branch Pipe Displacements for the Loop Drain System, Unit # 1, Loop 2 (Supplements Nuclear Services Corp. Report 11715-SSR-10"

6.12.53 14938.22-NP(B)-006-X, "Evaluation of Lines 2"CH-5-1502-Q1 (Supplements Teledyne Report E-1473-4, Rev. A (11715-SSR-13)) for Change in OBEA Branch Connection Displacements - Unit 1"

6.12.54 14938.22-NP(B)-007-X, "Review of Reactor Coolant Loop Branch Pipe Displacements for the Loop Fill System, Unit # 1 - Supplement to 11715-SSR-9, Nuclear Services Corp."

6.12.55 14938.22-NP(B)-008-X, "Review of Reactor Coolant Loop Branch Pipe Displacements for the Loop Drain System, Unit # 1, Loops 1 & 3 - Supplement to 1175-SSR-10, Nuclear Services Corp."

6.12.56 14258.32-NP(B)-01-XC, "Stress Analysis for CVCS - Seal Water Outlet System, Coupling Addition Due to Flow Splitter Removal"

6.12.57 14928.56-NP(B)-005-XC, "Pipe Stress Analysis - Effects of Thermal Stratification and Thermal Striping on the Pressurizer Surge Line"

6.12.58 02072.13-NP(B)-006-ZB, "Evaluation of Pipe Supports for Chemical and Volume Control System due to Reanalysis Required for Steam Generator Repair"

6.12.59 13075-SEO-410, "FPH-CH-14-1"

6.12.60 13075-SEO-528, "FPH-CH-14-1 & 14-3"

6.12.61 13075-SEO-546, "FPH-CH-14-4"

6.12.62 13075-SEO-613, "FPH-CH-16-6"

6.12.63 13075-SEO-1011, "FPH-CH-15-6"

6.12.64 13075-SEO-2286, "FPH-CH-14-1"

6.12.65 13075-SEO-2884, "FPH-CH-16-6"

1. Design Change Title/Station/Unit	2. Design Change Number
STEAM GENERATOR REPLACEMENT / NORTH ANNA / UNIT 1	DC 90-13-1

9. Engineering Review and Design:

6.12.66 13075-SEO-3148, "FPH-CH-14-1"

6.12.67 13075-SEO-3149, "FPH-CH-14-3"

6.12.68 13075-SEO-3358, "FPH-CH-14-1, -14-2, -14-3, -14-4, -15-6, -16-6"

Miscellaneous Affected Calculations (Stone and Webster)

6.12.69 11715-NM(B)-17-GA, "Stress Analysis of the Steam Generator Upper Restraint"

6.12.70 11715-NM(B)-20-BA, "Steam Generator and Pump Support Analysis", Dated 3-9-72

6.12.71 11715-NM(B)-109-BA, "Steam Generator and Reactor Coolant Pump Supports"

6.12.72 11715-NM(B)-147-BF, "Jet Impingement on Steam Generator and Reactor Coolant Pump Frame"

6.12.73 11715-NM(B)-211-GA, "Reactor Coolant Pump Column Buckling Analysis at Thread Relief"

6.12.74 11715-NM(B)-259-BF, "Calculation of Steam Generator Support Stresses in Short Transverse Locations"

6.12.75 11715-NM(B)-260-BF, "Steam Generator/Reactor Coolant Pump Support Stress Summary"

6.12.76 11715-NM(B)-263-BF, "Stress Analysis of RCP Support Rear Weldment Central Vertical Member with Material Specimen Cut-Out"

6.12.77 11715-NM(B)-306-BF, "SG/RCP Support Analysis for Asymmetric Pressure Due to Break 7"

6.12.78 11715-NM(B)-349-BFA, "Analysis of Jet Impingement and Pipe Whip on Steam Generator, Reactor Coolant Pump, Pressurizer and their Supports from S&W designed High Energy Line Breaks"

6.12.79 11715-NM(B)-365-BF, "SG Upper Snubber Loads Due to Main Steam Line Break and Seismic"

6.12.80 14938.22-NM(B) 60-BA, "Primary Component Support Snubber Elimination Study - Normal Operating and Seismic Loads"

6.12.81 14938.22-NM(B)-464-BA, "Dynamic Analysis of Limiting Pipe Ruptures for Leak-Before-Break (LBB) Licensing Basis"

1. Design Change Title/Station/Unit	2. Design Change Number
STEAM GENERATOR REPLACEMENT / NORTH ANNA / UNIT 1	DC 90-13-1

9. Engineering Review and Design:

6.12.82 14839.22-NM(B)-467-GX, "North Anna 1 and 2 - Steam Generator Upper Restraint Strut"

6.12.83 14938.22-NM(B)-470-BA, "North Anna Leak Before Break Evaluation: Factors of Safety for Selected Primary Loop Components"

6.12.84 14938.22-NM(B)-471-BA, "Major Equipment Snubbers Leak Before Break Modification - Factors of Safety"

6.12.85 14938.22-NM(B)-480-GA, "Design and Analysis of Existing Snubber Extension Rod Modification"

6.12.86 14938.22-NP(B)-001-X, "Reactor Coolant Loop Snubber Reduction Evaluation - Branch Displacements and Thermal Loadings"

6.12.87 02072.02-NP(B)-001-XH, "Steam Generator Repair Project - Review of the Effect of the New Steam Generator on the Reactor Loop Branch Connections and Secondary Piping"

6.12.88 02072.22-NP(B)-001-X, "Critical Commodities for Cold Gap Verification"

<u>Class 1 Stress Reports (Westinghouse)</u>

* 6.12.89 SSR-1, NAPS Unit 1 Reactor Coolant System Class 1 Stress Report

<u>Class 1 Stress Reports (Stone and Webster)</u>

* 6.12.90 11715-SSR-2, "North Anna Power Station Unit 1 Residual Heat Removal System ASME III Stress Analysis Report"

* 6.12.91 11715-SSR-3, "North Anna Power Station Unit 1 Safety Injection System ASME III Stress Analysis Report"

* 6.12.92 11715-SSR-5, "North Anna Power Station Unit 1 Pressurizer Spray System ASME III Stress Analysis Report"

* 6.12.93 11715-SSR-7, "North Anna Power Station Unit 1 Chemical Volume and Control System Seal Water Inlet Piping ASME III Stress Analysis Report"

* 6.12.94 11715-SSR-8, "North Anna Power Station Unit 1 Chemical Volume and Control System Seal Water Outlet Piping ASME III Stress Analysis Report"

* 6.12.95 11715-SSR-9, "North Anna Power Station Unit 1 Chemical Volume and Control System Loop Fill System ASME III Stress Analysis Report"

1. Design Change Title/Station/Unit

STEAM GENERATOR REPLACEMENT / NORTH ANNA / UNIT 1

2. Design Change Number

DC 90-13-1

9. Engineering Review and Design:

* 6.12.96 11715-SSR-10, "North Anna Power Station Unit 1 Chemical Volume and Control System Loop Drain System ASME III Stress Analysis Report"

* 6.12.97 11715-SSR-12, "North Anna Power Station Unit 1 Chemical Volume and Control System Charging Line Downstream of Regenerative Heat Exchanger of CVCS ASME III Stress Analysis Report"

* 6.12.98 11715-SSR-13, "North Anna Power Station Unit 1 Chemical Volume and Control System - Letdown Line Upstream of Regenerative Heat Exchanger of CVCS ASME III Stress Analysis Report"

* 6.12.99 11715-SSR-15, "North Anna Power Station Unit 1 Cold Leg Stop Valve Bypass Line ASME III Stress Analysis Report"

* 6.12.100 11715-SSR-16, "North Anna Power Station Unit 1 Pressurizer Surge Line - Thermal Stratification ASME III Stress Analysis Report"

Virginia Power Calculations

6.12.101 EE-0490, "Temporary Relay Settings for the Construction Power Transformer Being Used During the Replacement of the Steam Generator"

* 6.12.102 EE-0492, "Unit 1 Steam Generator Narrow Range Level Indication and Trips Uncertainty"

Bechtel Reconciliation Calculation

6.12.103 21809-M-03, "Code/Design Reconciliation"

PROGRAMS REVIEW CHECKLIST

DCP Number DC 90-13-1	Station NORTH ANNA UNIT 1	Page 1 of 6

Preparing Engineer/Affiliation (Print)	Signature	Date
MARK BARTH/Bechtel	MSBarth	9/18/92
Reviewing Engineer/Affiliation (Print)	**Signature**	**Date**
M.D Smith / BECHTEL	M.D Smith	9/18/92

If the answer to a question is "yes", a review must be performed for that item in accordance with the directions provided in Attachment 5. The results of the review shall be documented in Section 3.0 of the ER&D. The referenced screening questions must all be reviewed and answered "no" before the question in this checklist can be answered "no". If one or more screening questions are answered "yes", then the question in this checklist must be answered "yes".

1. **Updated Final Safety Analysis Report (UFSAR)**

List the UFSAR sections which cover the activities, structures, systems or components affected by the change.

Numerous sections, tables, and figures as listed and described in ER&D Section 3.1.

Will the DCP (1) affect the description of any structures, systems, components or activities addressed in the UFSAR or (2) add any structures, systems, components or activities for which a description must be added to the UFSAR? Yes X No ___

2. **Technical Specifications**

List the Tech. Specs. which cover the activities, systems or components affected by the change.

Numerous sections as listed and described in ER&D Section 3.2.

Will the DCP impact the Technical Specifications or involve technical specification requirements as indicated by a "yes" answer to one or more of the questions presented in Section 3.2 of Attachment 4. Yes ___ No X

3. **Fire Protection/Appendix "R"**

Could the DCP impact the Appendix "R" program as indicated by a "yes" answer to one or more of the questions presented in Section 3.3 of Attachment 4. Yes X No ___

PROGRAMS REVIEW CHECKLIST

DCP Number	Station	Page
DC 90-13-1	NORTH ANNA UNIT 1	2 of 6

		Yes	No
4.	**Equipment Qualification** Could the DCP impact the EQ program as indicated by a "yes" answer to one or more of the questions presented in Section 3.4 of Attachment 4.	X	
5.	**Station Security** Could the DCP impact security as indicated by a "yes" answer to one or more of the questions presented in Sectic 3.5 of Attachment 4.	X	
6.	**Electrical System Analysis** Could the DCP impact the Electrical System Analysis as indicated by a "yes" answer to one or more of the questions on the Electrical System Analysis Checklist (STD-EEN-0026).		X
7.	**Inservice Inspection** Will the DCP temporarily or pe ntly add, remove, repair, replace or modify ASME tion XI equipment?	X	
8.	**Seismic** Will the DCP require a seismic evaluation as indicated by a "yes" answer to one or more of the questions presented in Section 3.8 of Attachment 4.	X	
9.	**Human Factors** Could the DCI impact the Human Factors program as indicated by a "yes" answer to one or more of the questions presented in Section 3.9 in Attachment 4.	X	
10.	**In Containment Banned/Restricted Materials?** Will the DCP add, replace or remove from inside the containment any of the banned/restricted materials identified in STD-MAT-0006?	X	
11.	**Station Computer Software/Hardware** Could the DCP impact station computer software or hardware as indicated by a "yes" answer to one or more of the questions presented in Section 3.11 of Attachment 4.		X

PROGRAMS REVIEW CHECKLIST

DCP Number DC 90-13-1	Station NORTH ANNA UNIT 1	Page 3 of 6

12. **Emergency Response Facilities (ERF) System**

	Yes	No
Could the DCP impact the ERF as indicated by a "yes" answer to one or more of the questions presented on ERF Design Checklist (STD-GN-0028).	X	

13. **Plant Flooding**

	Yes	No
Could the DCP impact plant flooding as indicated by a "yes" answer to one or more of the questions presented in Section 3.13 of Attachment 4.		X

14. **Heavy loads**

	Yes	No
Could the DCP impact the Heavy Loads Program as indicated by a "yes" answer to one or more of the questions presented in Section 3.14 of Attachment 4.	X	

15. **Post-Accident Monitoring (Reg. Guide 1.97)**

	Yes	No
Could the DCP impact the station's compliance with Reg. Guide 1.97 as indicated by a "yes" answer to one or more of the questions presented on the Post-Accident Monitoring Design Checklist (STD-GN-0035).	X	

16. **Heating, Ventilation and Air Conditioning (HVAC)**

	Yes	No
Could the DCP impact the HVAC systems as indicated by a "yes" answer to one or more of the questions presented in Section 3.16 of Attachment 4.	X	

17. **The Simulator**

	Yes	No
Could the DCP impact the simulator as indicated by a "yes" answer to one or more of the questions presented in Section 3.17 of Attachment 4.	X	

18. **Nuclear Plant Reliability Data System (NPRDS)**

	Yes	No
Will the DCP require an NPRDS review as indicated by a "yes" answer to one or more of the questions presented in Section 3.18 of Attachment 4.	X	

19. **Setpoints, Instrument Accuracy and Scaling**

	Yes	No
Could the DCP impact the setpoint and/or instrument accuracy Program as indicated by a "yes" answer to one or more of the questions presented in Section 3.19 of Attachment 4.	X	

PROGRAMS REVIEW CHECKLIST

DCP Number	Station	Page
DC 90-13-1	NORTH ANNA UNIT 1	4 of 6

20. **Secondary Piping and Component Inspection Program**

	Yes	No
Could the DCP impact the Inspection Program as indicated by a "yes" answer to one or more of the questions presented in Section 3.20 of Attachment 4.	X	

21. **Radio Frequency Interference**

	Yes	No
Will the DCP involve the addition, replacement or modification of electronic equipment?		X

22. **Q-List**

	Yes	No
Will the DCP (1) impact equipment on the Q-List or (2) add SR or NSQ equipment?	X	

23. **ALARA**

	Yes	No
Will the DCP require an ALARA evaluation as indicated by a "yes" answer to one or more of the questions presented on the Preliminary ALARA Evaluation Form (STD-GN-0019)?	X	

24. **Cumulative Effects on Plant Systems**

	Yes	No
Will the cumulative effects on plant systems and components have to be considered as indicated by a "yes" answer to one or more of the questions presented in Section 3.24 of Attachment 4.		X

25. **Recent NRC Concerns**

	Yes	No
Are there any known recent NRC and/or industry concerns that relate to this DCP as indicated by a search of the operating experience database?	X	

26. **Impact on/of Other Design Changes**

	Yes	No
Will the DCP affect other ongoing design changes or have the potential to be affected by other ongoing design changes?	X	

27. **Summary of Equipment Added or Removed**

	Yes	No
Will the DCP add or remove equipment or components?	X	

PROGRAMS REVIEW CHECKLIST

DCP Number DC 90-13-1	Station NORTH ANNA UNIT 1	Page 5 of 6

28. **System and Plant Design Basis Documents**

	Yes	No
Will the DCP require a revision to one or more System and/or Plant Design Basis Documents?	X	

29. **Installation Specifications**

	Yes	No
Will the DCP require the development or revision of an installation specification?	X	

30. **Removal Blocks and Other Barriers**

	Yes	No
Will the DCP involve the removal of removable blocks or impact another barrier as indicated by a "yes" answer to one or more of the questions presented in Section 3.30 of Attachment 4.	X	

31. **Environmental Impact (non-radiological)**

	Yes	No
Will the impact of the DCP on the environment have to be addressed as indicated by a "yes" answer to one or more of the questions presented in Section 3.31 of Attachment 4.		X

32. **Masonry Block Walls**

	Yes	No
Will the DCP (1) impact masonry block walls or (2) involve the addition, removal or replacement of equipment on block walls or within the collapse envelope of block walls?		X

33. **NCRODP Training Modules**

	Yes	No
Will the DCP require a change to one or more system training modules for the Nuclear Control Room Operator Development Program (NCRODP)?	X	

34. **Recommended Spare Parts**

	Yes	No
Does the DCP involve the procurement of equipment by Engineering?	X	

35. **Labelling**

	Yes	No
Will the DCP require new equipment/component labels or tags as indicated by a "yes" answer to one or more of the questions presented in Section 3.35 of Attachment 4?		X

PROGRAMS REVIEW CHECKLIST

DCP Number DC 90-13-1	Station NORTH ANNA UNIT 1	Page 6 of 6

36. **Abandonment of Equipment**

	Yes	No
Will the DCP involve the abandonment of equipment in place?		X

37. **Vendor Technical Manuals (VTM)**

	Yes	No
Will the DCP require the development of new VTMs or the revision of existing VTMs in accordance with ENAP-0023?	X	

38. **Reactivity Management**

	Yes	No
Will the DCP have the potential to affect core reactivity as indicated by a "yes" answer to one or more of the questions presented in Section 3.38 of Attachment 4?	X	

39. **Other Concerns**

	Yes	No
Were there any other concerns or items identified during the preparation of the DCP?		X

DRAWING REVISION RECORD AND MATERIAL LIST

VIRGINIA POWER

DRAWING REVISION RECORD - NUCLEAR POWER STATION

STD-GN-0001 REV. 10 POW 11

1. Design Change Title	2. Station and Unit	3. Design Change
STEAM GENERATOR REPLACEMENT	NORTH ANNA UNIT 1	DC 90-13-1

4. DCP Drawing No. Sheet No./Rev. No. 5. Original Drawing No. Sheet No./Rev. No.	6. Drawing Title	7.2 Rev #	8. F.C #
N-9013-1-1ESK5AJ REV. 0 11715-ESK-5AJ REV. 18	ELEMENTARY DIAGRAM - 4160 CKTS - REACTOR COOLANT PUMP 1-RC-P-1B		
N-9013-1-1FE1B REV. 0 11715-FE-1B REV. 15	4160V ONE LINE DIAGRAM - SH1 - BUS 1A & 1B TRANSFER BUS D & E		
N-9013-1-1FE3HY REV. 0 11715-FE-3HY REV. 4	WIRING DIAGRAM LOOSE PARTS MONITORING & ACCOUSTICAL MONITORING PANELS		
N-9013-1-1FE57E REV. 0 11715-FE-57E REV. 25	COND. PLAN - INSTRUMENT, COMPUTER & ALARMS, REACTOR CONT. EL. 241'-0" SH. 1		
N-9013-1-1FE57F REV. 0 11715-FE-57F REV. 20	COND. PLAN - INSTRUMENT, COMPUTER & ALARMS, REACTOR CONT. EL. 241'-0". SH. 2		
N-9013-1-1FE8N REV. 0 11715-FE-8N REV. 16	WIRING DIAGRAM 4160V BUS "1B" REACTOR COOLANT PUMP 1-RC-P-1B BKR15B3		
N-9013-1-1FV17G REV. 0 11715-FV-17G REV. 11	STEAM GENERATOR LOWER SUPPORT FOOT ATTACHMENT DETAILS		
N-9013-1-1FV17L REV. 1 11715-FV-17L REV. 8	STEAM GENERATOR UPPER RESTRAINT ASSEMBLY		
N-9013-1-1FV17M REV. 1 11715-FV-17M REV. 5	STEAM GENERATOR UPPER RESTRAINT DETAILS SH. 1		
N-9013-1-1FV17N REV. 1 11715-FV-17N REV. 5	STEAM GENERATOR UPPER RESTRAINT DETAILS SH. 2		

VIRGINIA POWER

1. Design Change Title	2. Station and Unit	3. Design Change
STEAM GENERATOR REPLACEMENT	NORTH ANNA UNIT 1	DC 90-13-1

4. DCP Drawing No. Sheet No./Rev. No. / 5. Original Drawing No. Sheet No./Rev. No.	6. Drawing Title	7.2 Rev #	8. F.C #										
N-9013-1-1FV17P REV. 1 / 11715-FV-17P REV. 5	STEAM GENERATOR UPPER RESTRAINT SHIM												
N-9013-1-E-3500 REV. 1	LPM ACCELEROMETER MOUNTING BOLT												
N-9013-1-FK1A REV. 2 / 11715-FK-1A REV. 14	STEAM GENERATOR INSTRUMENT TUBING REMOVAL & REPLACEMENT												
N-9013-1-FK1B REV. 3 / 11715-FK-1B REV. 11	STEAM GENERATOR INSTRUMENT TUBING REMOVAL AND REPLACEMENT												
N-9013-1-M-401 REV. 3	REMOVAL & REPLACEMENT DRAWING - REACTOR COOLANT SYSTEM												
N-9013-1-M-402 REV. 2	REMOVAL & REPLACEMENT DRAWING - MAIN STEAM SYSTEM												
N-9013-1-M-403 REV. 4	REMOVAL & REPLACEMENT DRAWING - FEEDWATER SYSTEM												
N-9013-1-M-404 SH. 1/REV. 2	REMOVAL AND REPLACEMENT DRAWING - SG BLOWDOWN AND SHELL DRAIN PIPE												
N-9013-1-M-404 SH. 2/REV. 2	REMOVAL AND REPLACEMENT DRAWING - SAMPLE SYSTEM PIPING												
N-9013-1-M-405 REV. 4	REMOVAL AND REPLACEMENT DRAWING - SG TRANSITION CONE AND WRAPPER PLATE												

VIRGINIA POWER

DRAWING REVISION RECORD - NUCLEAR POWER STATION

STD-GN-0001 REV. 10 POW 11

1. Design Change Title		2. Station and Unit									3. Design Change		
STEAM GENERATOR REPLACEMENT		NORTH ANNA UNIT 1									DC 90-13-1		
4. DCP Drawing No. Sheet No./Rev. No. 5. Original Drawing No. Sheet No./Rev. No.	6. Drawing Title	7.2 Rev #	8. F.C #										
N-9013-1-M-406 REV. 1	TEMPORARY CONTAINMENT HVAC MODIFICATION FOR SMOKE REMOVAL												
N9013-1-M-601 SH. 1 OF 1/REV. 0	STEAM GENERATOR BLOWDOWN PIPING LOOP A - LAYOUT FOR 2 1/2" REPLACEMENT PIPING												
N9013-1-M-602 SH. 1 OF 2/REV. 0	STEAM GENERATOR BLOWDOWN PIPING LOOP B - LAYOUT FOR 2 1/2" REPLACEMENT PIPING												
N9013-1-M-602 SH. 2 OF 2/REV. 0	STEAM GENERATOR BLOWDOWN PIPING LOOP B - LAYOUT FOR 2 1/2" REPLACEMENT PIPING												
N9013-1-M-603 SH. 1 OF 2/REV. 0	STEAM GENERATOR BLOWDOWN PIPING LOOP C - LAYOUT FOR 2 1/2" REPLACEMENT PIPING												
N9013-1-M-603 SH 2 OF 2/REV. 0	STEAM GENERATOR BLOWDOWN PIPING LOOP C - LAYOUT FOR 2 1/2" REPLACEMENT PIPING												
N-9013-1-M-604 REV. 1	CHEMICAL FEEDWATER VIRGINIA POWER N. ANNA UNIT 1 LOOP A												
N-9013-1-M-605 REV. 1	CHEMICAL FEEDWATER VIRGINIA POWER N. ANNA UNIT 1 LOOP B												
N-9013-1-M-606 REV. 1	CHEMICAL FEEDWATER VIRGINIA POWER N. ANNA UNIT 1 LOOP C												
N-9013-1-M-607-0 EV. ~~1~~ 0 msb 10-14-92	STEAM GENERATOR WET LAY UP VIRGINIA POWER N. ANNA UNIT 1 LOOP A												

VIRGINIA POWER

DRAWING REVISION RECORD - NUCLEAR POWER STATION

1. Design Change Title	2. Station and Unit	3. Design Change
STEAM GENERATOR REPLACEMENT	NORTH ANNA UNIT 1	DC 90-13-1

4. DCP Drawing No. Sheet No./Rev. No. / 5. Original Drawing No. Sheet No./Rev. No.	6. Drawing Title	7.2 Rev #	8. F.C #										
N-9013-1-M-608-0 REV. ~~X~~ 0 MSB 10-14-92	STEAM GENERATOR WET LAY UP VIRGINIA POWER N. ANNA UNIT 1 LOOP B												
N-9013-1-M-609-0 REV. ~~X~~ 0 MSB 10-14-92	STEAM GENERATOR WET LAY UP VIRGINIA POWER N. ANNA UNIT 1 LOOP C												
N-9013-1-M-800 REV. 3	STEAM GENERATOR VESSEL CLOSURES FOR REMOVAL AND STORAGE												
N-9013-1-M-801 REV. 1	STEAM GENERATOR LEVEL INSTRUMENTATION CONDENSATE POT DETAILS												
N-9013-1-S-018 REV. 1	STEAM GENERATOR UPPER RESTRAINT INSTALLATION NOTES												
N9013-1-M-1MFSK1969 SH. 1/REV. 0 11715-MFSK-1969 REV. 1	STEAM GENERATOR BLOWDOWN MODIFICATION CUBICLE A PIPE SUPPORT FPH-WGCB-5-4												
N9013-1-M-1MFSK1969 SH. 2/REV. 0 11715-MFSK-1969 REV. 1	STEAM GENERATOR BLOWDOWN MODIFICATION CUBICLE A PIPE SUPPORT FPH-WGCB-5-4												
N9013-1-M-1MFSK1972 SH. 1/REV. 1 11715-MFSK-1972 REV. 1	STEAM GENERATOR BLOWDOWN MODIFICATION CUBICLE A PIPE SUPPORT FPH-WGCB-4-4												
N9013-1-M-1MFSK1972 SH. 2/REV. 1 11715-MFSK-1972 REV. 1	STEAM GENERATOR BLOWDOWN MODIFICATION CUBICLE A PIPE SUPPORT FPH-WGCB-4-4												
N9013-1-M-1MFSK1973 SH. 1/REV. 0 11715-MFSK-1973 REV. 1	STEAM GENERATOR BLOWDOWN MODIFICATION CUBICLE B PIPE SUPPORT FPH-WGCB-8-4												

1. Design Change Title		2. Station and Unit		3. Design Change
STEAM GENERATOR REPLACEMENT		NORTH ANNA UNIT 1		DC 90-13-1

4. DCP Drawing No. Sheet No./Rev. No. / 5. Original Drawing No. Sheet No./Rev. No.	6. Drawing Title	7.2 Rev #	8. F.C #
N9013-1-M-1MFSK1973 SH. 2/REV. 0 / 11715-MFSK-1973 REV. 1	STEAM GENERATOR BLOWDOWN MODIFICATION CUBICLE B PIPE SUPPORT FPH-WGCB-8-4		
N9013-1-M-1MFSK1975A SH. 1/REV. 0 / 11715-MFSK-1975A REV. 1	STEAM GENERATOR BLOWDOWN MODIFICATION CUBICLE B PIPE SUPPORT 1-WGCB-SH-MFSK-1975A		
N9013-1-M-1MFSK1975A SH. 2/REV. 0 / 11715-MFSK-1975A REV. 1	STEAM GENERATOR BLOWDOWN MODIFICATION CUBICLE B PIPE SUPPORT 1-WGCB-SH-MFSK-1975A		
N9013-1-M-1MFSK1976A SH. 1/REV. 1 / 11715-MFSK-1976A REV. 3	STEAM GENERATOR BLOWDOWN MODIFICATION CUBICLE B PIPE SUPPORT FPH-WGCB-7-4		
N9013-1-M-1MFSK1976A SH. 2/REV. 1 / 11715-MFSK-1976A REV. 3	STEAM GENERATOR BLOWDOWN MODIFICATION CUBICLE B PIPE SUPPORT FPH-WGCB-7-4		
N9013-1-M-1MFSK2277A SH. 1/REV. 0 / 11715-MFSK-2277A REV. 3	STEAM GENERATOR BLOWDOWN MODIFICATIONS CUBICLE B PIPE SUPPORT 1-WGCB-R-321		
N9013-1-M-1MFSK2277A SH. 2/REV. 0 / 11715-MFSK-2277A REV. 3	STEAM GENERATOR BLOWDOWN MODIFICATIONS CUBICLE B PIPE SUPPORT 1-WGCB-R-321		
N0913-1-M-1MFSK2285A SH. 1/REV. 1 / 11715-MFSK-2285A REV. 1	STEAM GENERATOR BLOWDOWN MODIFICATION CUBICLE A PIPE SUPPORT FPH-WGCB-5-5		
N0913-1-M-1MFSK2285A SH. 2/REV. 1 / 11715-MFSK-2285A REV. 1	STEAM GENERATOR BLOWDOWN MODIFICATION CUBICLE A PIPE SUPPORT FPH-WGCB-5-5		
N9013-1-M-1MFSK2286A SH. 1/REV. 1 / 11715-MFSK-2286A REV. 1	STEAM GENERATOR BLOWDOWN MODIFICATION CUBICLE B PIPE SUPPORT FPH-WGCB-8-5		

1. Design Change Title	2. Station and Unit	3. Design Change
STEAM GENERATOR REPLACEMENT	NORTH ANNA UNIT 1	DC 90-13-1

4. DCP Drawing No. Sheet No./Rev. No. / 5. Original Drawing No. Sheet No./Rev. No.	6. Drawing Title	7.2 Rev #	8. F.C #
N9013-1-M-1MFSK2286A SH. 2/REV. 1 / 11715-MFSK-2286A REV. 1	STEAM GENERATOR BLOWDOWN MODIFICATION CUBICLE B PIPE SUPPORT FPH-WGCB-8-5		
N9013-1-M-1MFSK2295A SH. 1/REV. 1 / 11715-MFSK-2295A REV. 1	STEAM GENERATOR BLOWDOWN MODIFICATION CUBICLE A PIPE SUPPORT FPH-WGCB-4-3		
N9013-1-M-1MFSK2295A SH. 2/REV. 1 / 11715-MFSK-2295A REV 1	STEAM GENERATOR BLOWDOWN MODIFICATION CUBICLE A PIPE SUPPORT FPH-WGCB-4-3		
N9013-1-1MFSK2296A SH. 1/REV. 1 / 11715-MFSK-2296A REV. 1	STEAM GENERATOR BLOWDOWN MODIFICATIONS CUBICLE B PIPE SUPPORT FPH-WGCB-7-3		
N9013-1-1MFSK2296A SH. 2/REV. 1 / 11715-MFSK-2296A REV. 1	STEAM GENERATOR BLOWDOWN MODIFICATIONS CUBICLE B PIPE SUPPORT FPH-WGCB-7-3		
N9013-1-M-1MFSK3040A SH. 1/REV. 1 / 11715-MFSK-3040A REV. 2	STEAM GENERATOR BLOWDOWN MODIFICATION CUBICLE C PIPE SUPPORT FPH-WGCB-10-1		
N9013-1-M-1MFSK3040A SH. 2/REV. 1 / 11715-MFSK-3040A REV. 2	STEAM GENERATOR BLOWDOWN MODIFICATION CUBICLE C PIPE SUPPORT FPH-WGCB-10-1		
N9013-1-M-1MFSK3041A SH. 1/REV. 1 / 11715-MFSK-3041A REV. 1	STEAM GENERATOR BLOWDOWN MODIFICATION CUBICLE C PIPE SUPPORT FPH-WGCB-10-2		
N9013-1-M-1MFSK3041A SH. 2/REV. 1 / 11715-MFSK-3041A REV. 1	STEAM GENERATOR BLOWDOWN MODIFICATION CUBICLE C PIPE SUPPORT FPH-WGCB-10-2		
N9013-1-M-1MFSK3044A SH. 1/REV. 1 / 11715-MFSK-3044A REV. 1	STEAM GENERATOR BLOWDOWN MODIFICATION CUBICLE C PIPE SUPPORT FPH-WGCB-12-1		

1. Design Change Title	2. Station and Unit	3. Design Change
STEAM GENERATOR REPLACEMENT	NORTH ANNA UNIT 1	DC 90-13-1

4. DCP Drawing No. Sheet No./Rev. No. / 5. Original Drawing No. Sheet No./Rev. No.	6. Drawing Title	7.2 Rev #	8. F.C #									
N9013-1-M-1MFSK3044A SH. 2/REV. 1 / 11715-MFSK-3044A REV. 1	STEAM GENERATOR BLOWDOWN MODIFICATION CUBICLE C PIPE SUPPORT FPH-WGCB-12-1											
N9013-1-M-1MFSK3047A SH. 1/REV. 1 / 11715-MFSK-3047A REV. 1	STEAM GENERATOR BLOWDOWN MODIFICATION CUBICLE C PIPE SUPPORT FPH-WGCB-10-3											
N9013-1-M-1MFSK3047A SH. 2/REV. 1 / 11715-MFSK-3047A REV. 1	STEAM GENERATOR BLOWDOWN MODIFICATION CUBICLE C PIPE SUPPORT FPH-WGCB-10-3											
N9013-1-M-1MFSK753A SH. 1/REV. 0 / 11715-MFSK-753A SH. 1/REV. 3	STEAM GENERATOR BLOWDOWN MODIFICATIONS CUBICLE B PIPE SUPPORT 1-WGCB-R-MFSK-753A											
N9013-1-M-1MFSK773A SH. 1/REV. 1 / 11715-MFSK-773A REV. 1	STEAM GENERATOR BLOWDOWN MODIFICATION CUBICLE A PIPE SUPPORT FPH-WGCB-4-2											
N9013-1-M-1MFSK773A SH. 2/REV. 1 / 11715-MFSK-773A REV. 1	STEAM GENERATOR BLOWDOWN MODIFICATION CUBICLE A PIPE SUPPORT FPH-WGCB-4-2											
N9013-1-M-1MFSK774A SH. 1/REV. 1 / 11715-MFSK-774A REV. 2	STEAM GENERATOR BLOWDOWN MODIFICATION CUBICLE B PIPE SUPPORT FPH-WGCB-7-2											
N9013-1-M-1MFSK774A SH. 2/REV. 1 / 11715-MFSK-774A REV. 2	STEAM GENERATOR BLOWDOWN MODIFICATION CUBICLE B PIPE SUPPORT FPH-WGCB-7-2											
N9013-1-M-1MFSK776A SH. 1/REV. 1 / 11715-MFSK-776A REV. 1	STEAM GENERATOR BLOWDOWN MODIFICATION CUBICLE A PIPE SUPPORT FPH-WGCB-5-3											
N9013-1-M-1MFSK776A SH. 2/REV. 1 / 11715-MFSK-776A REV. 1	STEAM GENERATOR BLOWDOWN MODIFICATION CUBICLE A PIPE SUPPORT FPH-WGCB-5-3											

1. Design Change Title	2. Station and Unit	3. Design Change
STEAM GENERATOR REPLACEMENT	NORTH ANNA UNIT 1	DC 90-13-1

4. DCP Drawing No. Sheet No./Rev. No. / 5. Original Drawing No. Sheet No./Rev. No.	6. Drawing Title	7. Rev #	8. F.C #										
N9013-1-M-1MFSK777A SH. 1/REV. 1 / 11715-MFSK-777A REV. 1	STEAM GENERATOR BLOWDOWN MODIFICATION CUBICLE B PIPE SUPPORT FPH-WGCB-8-1												
N9013-1-M-1MFSK777A SH. 2/REV. 1 / 11715-MFSK-777A REV. 1	STEAM GENERATOR BLOWDOWN MODIFICATION CUBICLE B PIPE SUPPORT FPH-WGCB-8-1												
N9013-1-M-1MFSK778A SH. 1/REV. 1 / 11715-MFSK-778A REV. 2	STEAM GENERATOR BLOWDOWN MODIFICATION CUBICLE C PIPE SUPPORT FPH-WGCB-11-2												
N9013-1-M-1MFSK778A SH. 2/REV. 1 / 11715-MFSK-778A REV. 2	STEAM GENERATOR BLOWDOWN MODIFICATION CUBICLE C PIPE SUPPORT FPH-WGCB-11-2												
N9013-1-M-1MFSK780A SH. 1/REV. 1 / 11715-MFSK-780A REV. 2	STEAM GENERATOR BLOWDOWN MODIFICATION CUBICLE A PIPE SUPPORT FPH-WGCB-4-1												
N9013-1-M-1MFSK780A SH. 2/REV. 1 / 11715-MFSK-780A REV. 2	STEAM GENERATOR BLOWDOWN MODIFICATION CUBICLE A PIPE SUPPORT FPH-WGCB-4-1												
N9013-1-M-1MFSK780A SH. 3/REV. 1 / 11715-MFSK-780A REV. 2	STEAM GENERATOR BLOWDOWN MODIFICATION CUBICLE A PIPE SUPPORT FPH-WGCB-4-1												
N9013-1-M-1MFSK781A SH. 1/REV. 1 / 11715-MFSK-781A REV. 3	STEAM GENERATOR BLOWDOWN MODIFICATION CUBICLE B PIPE SUPPORT FPH-WGCB-7-1												
N9013-1-M-1MFSK781A SH. 2/REV. 1 / 11715-MFSK-781A REV. 3	STEAM GENERATOR BLOWDOWN MODIFICATION CUBICLE B PIPE SUPPORT FPH-WGCB-7-1												
N9013-1-M-1MFSK781A SH. 3/REV. 1 / 11715-MFSK-781A REV. 3	STEAM GENERATOR BLOWDOWN MODIFICATION CUBICLE B PIPE SUPPORT FPH-WGCB-7-1												

1. Design Change Title	2. Station and Unit	3. Design Change
STEAM GENERATOR REPLACEMENT	NORTH ANNA UNIT 1	DC 90-13-1

4. DCP Drawing No. Sheet No./Rev. No. / 5. Original Drawing No. Sheet No./Rev. No.	6. Drawing Title	7.2 Rev #	8. F.C #										
N9013-1-M-1MFSK783A SH. 1/REV. 1 / 11715-MFSK-783A REV. 2	STEAM GENERATOR BLOWDOWN MODIFICATION CUBICLE A PIPE SUPPORT FPH-WGCB-5-1												
N9013-1-M-1MFSK783A SH. 2/REV. 1 / 11715-MFSK-783A REV. 2	STEAM GENERATOR BLOWDOWN MODIFICATION CUBICLE A PIPE SUPPORT FPH-WGCB-5-1												
N9013-1-M-1MFSK783A SH. 3/REV. 1 / 11715-MFSK-783A REV. 2	STEAM GENERATOR BLOWDOWN MODIFICATION CUBICLE A PIPE SUPPORT FPH-WGCB-5-1												
N9013-1-M-1MFSK784A SH. 1/REV. 1 / 11715-MFSK-784A REV. 3	STEAM GENERATOR BLOWDOWN MODIFICATION CUBICLE B PIPE SUPPORT FPH-WGCB-8-3												
N9013-1-M-1MFSK784A SH. 2/REV. 1 / 11715-MFSK-784A REV. 3	STEAM GENERATOR BLOWDOWN MODIFICATION CUBICLE B PIPE SUPPORT FPH-WGCB-8-3												
N9013-1-M-1MFSK784A SH. 3/REV. 1 / 11715-MFSK-784A REV. 3	STEAM GENERATOR BLOWDOWN MODIFICATION CUBICLE B PIPE SUPPORT FPH-WGCB-8-3												
N9013-1-M-1MFSK788A SH. 1/REV. 1 / 11715-MFSK-788A REV. 3	STEAM GENERATOR BLOWDOWN MODIFICATION CUBICLE A PIPE SUPPORT FPH-WGCB-5-2												
N9013-1-M-1MFSK788A SH. 2/REV. 1 / 11715-MFSK-788A REV. 3	STEAM GENERATOR BLOWDOWN MODIFICATION CUBICLE A PIPE SUPPORT FPH-WGCB-5-2												
N9013-1-M-1MFSK789A SH. 1/REV. 1 / 11715-MFSK-789A REV. 6	STEAM GENERATOR BLOWDOWN MODIFICATION CUBICLE B PIPE SUPPORT FPH-WGCB-8-2												
N9013-1-M-1MFSK789A SH. 2/REV. 1 / 11715-MFSK-789A REV. 6	STEAM GENERATOR BLOWDOWN MODIFICATION CUBICLE B PIPE SUPPORT FPH-WGCB-8-2												

1. Design Change Title	2. Station and Unit	3. Design Change
STEAM GENERATOR REPLACEMENT	NORTH ANNA UNIT 1	DC 90-13-1

4. DCP Drawing No. Sheet No./Rev. No. / 5. Original Drawing No. Sheet No./Rev. No.	6. Drawing Title	7.2 Rev #	8. F.C #										
N9013-1-M-1MFSK790A SH. 1/REV. 1 / 11715-MFSK-790A REV. 1	STEAM GENERATOR BLOWDOWN MODIFICATION CUBICLE C PIPE SUPPORT FPH-WGCB-11-1												
N9013-1-M-1MFSK790A SH. 2/REV. 1 / 11715-MFSK-790A REV. 1	STEAM GENERATOR BLOWDOWN MODIFICATION CUBICLE C PIPE SUPPORT FPH-WGCB-11-1												
N9013-1-M-1PSSK103AAD.08 SH. 1/REV. 0 / 11715-PSSK-103AAD.08 SH. 1/REV. 1	STEAM GENERATOR BLOWDOWN MODIFICATIONS CUBICLE A PIPE SUPPORT 1-WGCB-A-318												
N9013-1-M-1PSSK103AAD.08 SH. 2/ REV. 0 / 11715-PSSK-103AAD.08 SH. 1/REV. 1	STEAM GENERATOR BLOWDOWN MODIFICATIONS CUBICLE A PIPE SUPPORT 1-WGCB-A-318												
N9013-1-M-1PSSK103AAD.09 SH. 1/REV. 1 / 11715-PSSK-103AAD.09 SH. 1/REV. 1	STEAM GENERATOR BLOWDOWN MODIFICATIONS CUBICLE A PIPE SUPPORT 1-WGCB-R-319												
N9013-1-M-1PSSK103AAD.09 SH. 2/REV. 1 / 11715-PSSK-103AAD.09 SH. 1/REV. 1	STEAM GENERATOR BLOWDOWN MODIFICATIONS CUBICLE A PIPE SUPPORT 1-WGCB-R-319												
N9013-1-M-1PSSK103AAD.12 SH. 1/REV. 0 / 11715-PSSK-103AAD.12 SH. 1/REV. 1	STEAM GENERATOR BLOWDOWN MODIFICATION CUBICLE A PIPE SUPPORT 1-WGCB-SH-MFSK-1971A												
N9013-1-M-1PSSK103AAD.12 SH. 2/REV. 0 / 11715-PSSK-103AAD.12 SH. 1/REV. 1	STEAM GENERATOR BLOWDOWN MODIFICATION CUBICLE A PIPE SUPPORT 1-WGCB-SH-MFSK-1971A												
N9013-1-M-1PSSK103AAD.13 SH 1/REV. 0 / 11715-PSSK-103AAD.13 SH. 1/REV. 1	STEAM GENERATOR BLOWDOWN MODIFICATIONS CUBICLE A PIPE SUPPORT 1-WGCB-HSS-001												
N9013-1-M-1PSSK103AAD.14 SH. 1/REV. 0 / 11715-PSSK-103AAD.14 SH. 1/REV. 1	STEAM GENERATOR BLOWDOWN MODIFICATIONS CUBICLE A PIPE SUPPORT 1-WGCB-R-450												

1. Design Change Title	2. Station and Unit	3. Design Change
STEAM GENERATOR REPLACEMENT	NORTH ANNA UNIT 1	DC 90-13-1

4. DCP Drawing No. Sheet No./Rev. No. 5. Original Drawing No. Sheet No./Rev. No.	6. Drawing Title	7.2 Rev #	8. F.C #
N9013-1-M-1PSSK103BY.01 SH. 1/REV. 0 11715-PSSK-103BY.01 SH. 1/ REV.1	STEAM GENERATOR BLOWDOWN MODIFICATIONS CUBICLE C PIPE SUPPORT 1-BD-HSS-002		
N9013-1-M-1PSSK103BY.02 SH. 1/REV. 0 11715-PSSK-103BY.02 SH. 1/REV. 1	STEAM GENERATOR BLOWDOWN MODIFICATIONS CUBICLE C PIPE SUPPORT 1-WGCB-R-312		
N9013-1-M-1PSSK103BY.02 SH. 2/REV. 0 11715-PSSK-103BY.02 SH. 2/REV. 1	STEAM GENERATOR BLOWDOWN MODIFICATIONS CUBICLE C PIPE SUPPORT 1-WGCB-R-312		
N9013-1-M-1PSSK103BY.03 SH. 1/REV. 0 11715-PSSK-103BY.03 SH. 1/REV. 1	STEAM GENERATOR BLOWDOWN MODIFICATIONS CUBICLE C PIPE SUPPORT 1-WGCB-R-304		
N9013-1-M-1PSSK103BY.03 SH. 2/REV. 0 11715-PSSK-103BY.03 SH. 2/REV. 1	STEAM GENERATOR BLOWDOWN MODIFICATIONS CUBICLE C PIPE SUPPORT 1-WGCB-R-304		
N9013-1-M-1PSSK103BY.03 SH. 3/REV. 0 11715-PSSK-103BY.03 SH. 3/REV. 1	STEAM GENERATOR BLOWDOWN MODIFICATIONS CUBICLE C PIPE SUPPORT 1-WGCB-R-304		
N9013-1-M-1PSSK103BY.04 SH. 1/REV. 0 11715-PSSK-103BY.04 SH. 1/REV. 1	STEAM GENERATOR BLOWDOWN MODIFICATIONS CUBICLE C PIPE SUPPORT 1-WGCB-R-303		
N9013-1-M-1PSSK103BY.04 SH. 2/REV. 0 11715-PSSK-103BY.04 SH. 2/REV. 1	STEAM GENERATOR BLOWDOWN MODIFICATIONS CUBICLE C PIPE SUPPORT 1-WGCB-R-303		
N9013-1-M-1PSSK103BY.04 SH. 3/REV. 0 11715-PSSK-103BY.04 SH. 3/REV. 1	STEAM GENERATOR BLOWDOWN MODIFICATIONS CUBICLE C PIPE SUPPORT 1-WGCB-R-303		
N9013-1-M-1PSSK103BY.05 SH. 1/REV. 0 11715-PSSK-103BY.05 SH. 1/REV. 1	STEAM GENERATOR BLOWDOWN MODIFICATIONS CUBICLE C PIPE SUPPORT 1-WGCB-R-122		

1. Design Change Title		2. Station and Unit	3. Design Change
STEAM GENERATOR REPLACEMENT		NORTH ANNA UNIT 1	DC 90-13-1

4. DCP Drawing No. Sheet No./Rev. No. / 5. Original Drawing No. Sheet No./Rev. No.	6. Drawing Title	7.2 Rev #	8. F.C #											
N9013-1-M-1PSSK103BY.06 SH. 1/REV. 0 / 11715-PSSK-103BY.06 SH. 1/REV. 1	STEAM GENERATOR BLOWDOWN MODIFICATIONS CUBICLE C PIPE SUPPORT 1-WGCB-R-121													
N9013-1-M-1PSSK103BY.07 SH. 1/REV. 0 / 11715-PSSK-103BY.07 SH. 1/REV. 1	STEAM GENERATOR BLOWDOWN MODIFICATIONS CUBICLE C PIPE SUPPORT 1-WGCB-R-328													
N9013-1-M-1PSSK103BY.07 SH. 2/REV. 0 / 11715-PSSK-103BY.07 SH. 2/REV. 1	STEAM GENERATOR BLOWDOWN MODIFICATIONS CUBICLE C PIPE SUPPORT 1-WGCB-R-328													
N9013-1-M-1PSSK103BY.08 SH. 1/REV. 0 / 11715-PSSK-103BY.08 SH. 1/REV. 1	STEAM GENERATOR BLOWDOWN MODIFICATIONS CUBICLE C PIPE SUPPORT 1-WGCB-R-MFSK-754A													
N9013-1-M-1PSSK103BY.09 SH. 1/REV. 0 / 11715-PSSK-103BY.09 SH. 1/REV. 1	STEAM GENERATOR BLOWDOWN MODIFICATION CUBICLE C PIPE SUPPORT 1-WGCB-SH-MFSK-1979A													
N9013-1-M-1PSSK103BY.09 SH. 2/REV. 0 / 11715-PSSK-103BY.09 SH. 2/REV. 1	STEAM GENERATOR BLOWDOWN MODIFICATION CUBICLE C PIPE SUPPORT 1-WGCB-SH-MFSK-1979A													
N-9013-1-M-700.01 SH. 1/REV. 1	STEAM GENERATOR BLOWDOWN MODIFICATIONS GENERAL NOTES													
N9013-1-M-700.07 SH. 1/REV. 0	STEAM GENERATOR BLOWDOWN MODIFICATIONS 3" CLAMP ANCHOR ASSEMBLY													
N9013-1-M-700.07 SH. 2/REV. 0	STEAM GENERATOR BLOWDOWN MODIFICATIONS 3" CLAMP ANCHOR ASSEMBLY													
N9013-1-M-700.07 SH. 3/REV. 0	STEAM GENERATOR BLOWDOWN MODIFICATIONS 3" CLAMP ANCHOR ASSEMBLY													

1. Design Change Title		2. Station and Unit	3. Design Change
STEAM GENERATOR REPLACEMENT		NORTH ANNA UNIT 1	DC 90-13-1

4. DCP Drawing No. Sheet No./Rev. No. / 5. Original Drawing No. Sheet No./Rev. No.	6. Drawing Title	7.2 Rev #	8. F.C #										
N9013-1-M-700.08 SH. 1/REV. 1	STEAM GENERATOR BLOWDOWN MODIFICATIONS CUBICLE B PIPE SUPPORT 1-WGCB-R-21C												
N9013-1-M-700.08 SH. 2/REV. 1	STEAM GENERATOR BLOWDOWN MODIFICATIONS CUBICLE B PIPE SUPPORT 1-WGCB-R-21C												
N9013-1-M-700.09 SH. 1/REV. 0	STEAM GENERATOR BLOWDOWN MODIFICATIONS CUBICLE B PIPE SUPPORT 1-WGCB-A-20C												
N9013-1-M-700.09 SH. 2/REV. 0	STEAM GENERATOR BLOWDOWN MODIFICATIONS CUBICLE B PIPE SUPPORT 1-WGCB-A-20C												
N9013-1-M-700.10 SH. 1/REV. 1	STEAM GENERATOR BLOWDOWN MODIFICATIONS CUBICLE A PIPE SUPPORT 1-WGCB-PH-4.1												
N9013-1-M-700.10 SH. 2/REV. 1	STEAM GENERATOR BLOWDOWN MODIFICATIONS CUBICLE A PIPE SUPPORT 1-WGCB-PH-4.1												
N9013-1-M-700.11 SH. 1/REV. 0	STEAM GENERATOR BLOWDOWN MODIFICATIONS CUBICLE B PIPE SUPPORT 1-WGCB-R-131												
N9013-1-M-700.12 SH. 1/REV. 0	STEAM GENERATOR BLOWDOWN MODIFICATIONS CUBICLE B PIPE SUPPORT 1-WGCB-R-323												
N9013-1-M-700.12 SH. 2/REV. 0	STEAM GENERATOR BLOWDOWN MODIFICATIONS CUBICLE B PIPE SUPPORT 1-WGCB-R-323												
N9013-1-M-700.12 SH. 3/REV. 0	STEAM GENERATOR BLOWDOWN MODIFICATIONS CUBICLE B PIPE SUPPORT 1-WGCB-R-323												

1. Design Change Title	2. Station and Unit	3. Design Change
STEAM GENERATOR REPLACEMENT	NORTH ANNA UNIT 1	DC 90-13-1

4. DCF Drawing No. Sheet No./Rev. No. / 5. Original Drawing No. Sheet No./Rev. No.	6. Drawing Title	7.2 Rev #	8. F.C #								
N9013-1-M-700.13 SH. 1/REV. 0	STEAM GENERATOR BLOWDOWN MODIFICATIONS CUBICLE B PIPE SUPPORT 1-WGCB-R-322										
N9013-1-M-700.13 SH. 2/REV. 0	STEAM GENERATOR BLOWDOWN MODIFICATIONS CUBICLE B PIPE SUPPORT 1-WGCB-R-322										
N9013-1-M-700.13 SH. 3/REV. 0	STEAM GENERATOR BLOWDOWN MODIFICATIONS CUBICLE B PIPE SUPPORT 1-WGCB-R-322										
N9013-1-M-700.14 SH. 1/REV. 1	STEAM GENERATOR BLOWDOWN MODIFICATIONS CUBICLE B PIPE SUPPORT 1-WGCB-PH-7.1										
N9013-1-M-700.14 SH. 2/REV. 1	STEAM GENERATOR BLOWDOWN MODIFICATIONS CUBICLE B PIPE SUPPORT 1-WGCB-PH-7.1										
N9013-1-M-700.15 SH. 1/REV. 0	STEAM GENERATOR BLOWDOWN MODIFICATIONS CUBICLE C PIPE SUPPORT FPH-WGCB-11-3										
N9013-1-M-700.15 SH. 2/REV. 0	STEAM GENERATOR BLOWDOWN MODIFICATIONS CUBICLE C PIPE SUPPORT FPH-WGCB-11-3										
N9013-1-M-700.15 SH. 3/REV. 0	STEAM GENERATOR BLOWDOWN MODIFICATIONS CUBICLE C PIPE SUPPORT FPH-WGCB-11-3										
N9013-1-M-700.16 SH. 1/REV.0	STEAM GENERATOR BLOWDOWN MODIFICATIONS GRINNELL PIPE SADDLE DETAILS										
N-9013-1-1-CBM074A-2 SH. 1 OF 4/REV. 1 / 11715-CBM-074A-2 SH. 1 OF 4/REV.1	ISI CLASSIFICATION BOUNDARY DWG INTERVAL 2 FEEDWATER SYSTEM										

1. Design Change Title	2. Station and Unit	3. Design Change
STEAM GENERATOR REPLACEMENT	NORTH ANNA UNIT 1	DC 90-13-1

4. DCP Drawing No. Sheet No./Rev. No. / 5. Original Drawing No. Sheet No./Rev. No.	6. Drawing Title	7.2 REV #	8. F.C #									
N9013-1-1CBM098A-2 SH. 2 OF 5/REV. 0 / 11715-CBM-098A-2 SH. 2 OF 5/REV. 1	ISI CLASSIFICATION BOUNDARY DWG INTERVAL 2 STEAM GENERATOR BLOWDOWN SYSTEM											
N9013-1-1CBM098A-2 SH. 3 OF 5/REV. 0 / 11715-CBM-098A-2 SH. 3 OF 5/REV. 1	ISI CLASSIFICATION BOUNDARY DWG INTERVAL 2 STEAM GENERATOR BLOWDOWN SYSTEM											
N9013-1-1CBM098A-2 SH. 4 OF 5/REV. 0 / 11715-CBM-098A-2 SH. 4 OF 5/REV. 1	ISI CLASSIFICATION BOUNDARY DWG INTERVAL 2 STEAM GENERATOR BLOWDOWN SYSTEM											
N9013-1-1CBM102A-2 SH. 2 OF 2/REV. 0 / 11715-CBM-102A SH. 2 OF 2/REV. 0	ISI CLASSIFICATION BOUNDARY DWG INTERVAL 2 CHEMICAL FEED SYSTEM											
N9013-1-1DAR074A ~~SH. 1 OF 4~~/REV. 0 MSB 10-14-92 / 11715-DAR-74A REV. 0	APPENDIX "R" SAFE SHUTDOWN FLOW DIAGRAM FEEDWATER											
N-9013-1-1FM1C REV. 0 / 11715-FM-1C SH 1 OF 1/REV. 12	MACH LOC - REACTOR CONT - SH 3 PLAN 241' - 0"											
N9013-1-1FM-43A REV. 0 / 11715-FM-43A REV. 14	FLOW DIAGRAM STEAM GENERATOR BLOWDOWN											
N-9013-1-1FM074A SH. 1 OF 4/REV.1 / 11715-FM-074A SH. 1 OF 4/REV. 33	FLOW/VALVE OPERATING NUMBERS DIAGRAM FEEDWATER SYSTEM											
N-9013-1-1FM098A SH 2 OF 5/REV. 0 / 11715-FM-098A SH 2 OF 5/REV. 21	FLOW/VALVE OPERATING NUMBERS DIAGRAM STEAM GENERATOR BLOWDOWN SYSTEM											
N-9013-1-1FM098A SH 3 OF 5/REV. 0 / 11715-FM-098A SH 3 OF 5/REV. 20	FLOW/VALVE OPERATING NUMBERS DIAGRAM STEAM GENERATOR BLOWDOWN SYSTEM											

VIRGINIA POWER

DRAWING REVISION RECORD - NUCLEAR POWER STATION

STD-GN-0001 REV. 10 POW 11

1. Design Change Title	2. Station and Unit	3. Design Change
STEAM GENERATOR REPLACEMENT	NORTH ANNA UNIT 1	DC 90-13-1

4. DCP Drawing No. Sheet No./Rev. No. / 5. Original Drawing No. Sheet No./Rev. No.	6. Drawing Title	7.2 REV #	8. F.C #									
N-9013-1-1FM098A SH 4 OF 5/REV. 0 / 11715-FM-098A SH 4 OF 5/REV. 23	FLOW/VALVE OPERATING NUMBERS DIAGRAM STEAM GENERATOR BLOWDOWN SYSTEM											
N-9013-1-1FM102A SH. 2 OF 2/REV. 0 / 11715-FM-102A SH. 2 OF 2/REV. 15	FLOW/VALVE OPERATING NUMBERS DIAGRAM CHEMICAL FEED SYSTEMS											
N-9013-1-1SPM074A-2 SH. 1 OF 4/REV. 1 / 11715-SPM-074A-2 SH. 1 OF 4/REV. 0	SYSTEM PRESSURE TESTING DRAWING INTERVAL 2 FEEDWATER SYSTEM											
N-9013-1-1SPM-~~098-2~~ 098A-2 SH 2 OF 5/REV. 0 msB 10-14-92 / 11715-SPM-098A-2 SH 2 OF 5/REV. 0	SYSTEM PRESSURE TESTING DWG INTERVAL 2 STEAM GENERATOR BLOWDOWN SYSTEM											
N-9013-1-1SPM-098A-2 SH 3 OF 5/REV 0 / 11715-SPM-098A-2 SH 3 OF 5/REV. 0	SYSTEM PRESSURE TESTING DWG INTERVAL 2 STEAM GENERATOR BLOWDOWN											
N-9013-1-1SPM-098A-2 SH 4 OF 5/REV. 0 / 11715-SPM-098A-2 SH 4 OF 5/REV. 0	SYSTEM PRESSURE TESTING DWG INTERVAL 2 STEAM GENERATOR BLOWDOWN											
N-9013-1-1SPM-102A-2 SH 2 OF 2/REV. 0 / 11715-SPM-102A-2 SH 2 OF 2/REV 0	SYSTEM PRESSURE TESTING DWG INTERVAL - 2 CHEMICAL FEED SYSTEMS											
N-9013-1-1WMKS-RC-E-1A.1 SH. 1 OF 2/REV. 0 / 11715-WMKS-RC-E-1A.1 SH. 1 OF 2/REV. 0	INSERVICE INSPECTION DETAIL DRAWING STEAM GENERATOR: 1-RC-E-1A											
N-9013-1-1WMKS-RC-E-1A.2 SH. 1 OF 2/REV. 0 / 11715-WMKS-RC-E-1A.2 SH. 1 OF 2/REV. 0	INSERVICE INSPECTION DETAIL DRAWING STEAM GENERATOR: SECTION & DETAILS											
N-9013-1-1WMKS-RC-E-1B.1 SH. 1 OF 2/REV. 0 / 11715-WMKS-RC-E-1B.1 SH. 1 OF 2/REV. 0	INSERVICE INSPECTION DETAIL DRAWING STEAM GENERATOR: 1-RC-E-1B											

VIRGINIA POWER

DRAWING REVISION RECORD - NUCLEAR POWER STATION

STD-GN-0001 REV. 10 POW 11

1. Design Change Title	2. Station and Unit	3. Design Change
STEAM GENERATOR REPLACEMENT	NORTH ANNA UNIT 1	DC 90-13-1

4. DCP Drawing No. Sheet No./Rev. No. / 5. Original Drawing No. Sheet No./Rev. No.	6. Drawing Title	7.2 REV #	8. F.C #								
N-9013-1-1WMKS-RC-E-1B.2 SH. 1 OF 2/REV. 0 / 11715-WMKS-RC-E-1B.2 SH. 1 OF 2/REV. 0	INSERVICE INSPECTION DETAIL DRAWING STEAM GENERATOR: SECTION & DETAILS										
N-9013-1-1WMKS-RC-E-1C.1 SH. 1 OF 2/REV. 0 / 11715-WMKS-RC-E-1C.1 SH. 1 OF 2/REV. 0	INSERVICE INSPECTION DETAIL DRAWING STEAM GENERATOR: 1-RC-E-1C										
N-9013-1-1WMKS-RC-E-1C.2 SH. 1 OF 2/REV. 0 / 11715-WMKS-RC-E-1C.2 SH. 1 OF 2/REV. 0	INSERVICE INSPECTION DETAIL DRAWING STEAM GENERATOR: SECTION & DETAILS										
N-9013-1-1WMKS-0101B SH. 1 OF 3/REV. 0 / 11715-WMKS-0101B SH. 1 OF 3/REV. 0	INSERVICE INSPECTION ISOMETRIC SHP SYS: 32" PIPE RC-E-1A TO PEN 73										
N-9013-1-1WMKS-0101C SH. 1 OF 2/REV. 0 / 11715-WMKS-0101C SH. 1 OF 2/REV. 0	INSERVICE INSPECTION ISOMETRIC SHP SYS: 32" PIPE RC-E-1B TO PEN 74										
N-9013-1-1WMKS-0101D SH. 1 OF 2/REV. 0 / 11715-WMKS-0101D SH. 1 OF 2/REV. 0	INSERVICE INSPECTION ISOMETRIC SHP SYS: 32" PIPE RC-E-1C TO PEN 75										
N-9013-1-1WMKS-0102A SH. 1 OF 3/REV. 0 / 11715-WMKS-0102A SH. 1 OF 3/REV. 0	INSERVICE INSPECTION ISOMETRIC WFPD SYS: 16" PIPING TO 1-RC-E-1A										
N-9013-1-1WMKS-0102B SH. 1 OF 3/REV. 0 / 11715-WMKS-0102B SH. 1 OF 3/REV. 0	INSERVICE INSPECTION ISOMETRIC WFPD SYS: 16" PIPING TO 1-RC-E-1B										
N-9013-1-1WMKS-0102C SH. 1 OF 2/REV.0 / 11715-WMKS-0102C SH. 1 OF 2/REV. 0	INSERVICE INSPECTION ISOMETRIC WFPD SYS: 16" PIPING TO 1-RC-E-1C										

VIRGINIA POWER

MATERIAL LIST - NUCLEAR POWER STATION

STD-GN-0001 Rev. 10 RIS 14

1. Design Change Title/Station/Unit		2. Design Change No.
STEAM GENERATOR REPAIR / NORTH ANNA / UNIT 1		DC 90-13-1
3. Preparing Engineer/Affiliation (Print)	4. Signature	5. Date
6. Reviewing Engineer/Affiliation (Print)	7. Signature	8. Date

9. Item No.	10. Quantity	11. Description (include any referenced specifications)	12. QA Cat	13. Purchased By	14. Requisition Number
		ELECTRICAL			
	3	RELAY, TIME OVERCURRENT, GENERAL ELECTRIC TYPE 1AC, 60 HZ, 0.5-4 AMPS TIME OVERCURRENT RANGE AND 10-80 AMPS INSTANTANEOUS RANGE, MODEL NO. 12IAC53B805A	NS	F	40 118298
		SG LEVEL INSTRUMENTATION			
2	1500 FT	TUBING, 1/2" OD X 0.065" WALL; ASTM A213 TYPE 316; CLASS Q2; STD. 20 FT. LENGTHS (REF. DWG. N-9013-1-FK1A & -FK1B)	SR	S	36 554480 (TO BE TAKEN FROM STOCK)
3	24	VALVE, GLOBE, NPS 3/4, ANSI B16.34 CLASS 900, SOCKET WELDED; BODY MATERIAL ASTM A182 TYPE F316; SPEC. NAP-0023, PIPE CLASS Q2 (REF DWG N-9013-1-FK1A)	SR	F	CNT 396432
4	12	VALVE, GLOBE, NPS 1/2, ANSI B16.34 CLASS 900, SOCKET WELDED; BODY MATERIAL ASTM A182 TYPE F316; SPEC. NAP-0023, PIPE CLASS Q2; (REF DWG N-9013-1-FK1A)	SR	F	CNT 396432
05	120	UNION, TUBE SOCKET WELD, 1/2" X 1/2"; CAJON CATALOG NO. SS-8-TSW-6 (REF. DWG. N-9013-1-FK1A & -FK1B)	SR	S	38 757470 (TO BE TAKEN FROM STOCK)
6	30	ADAPTER, PIPE/TUBE, 3/4" IPS X 1/2" TUBE; CAJON CATALOG NO. SS-12-MPW-A-8TSW (REF. DWG. N-9013-1-FK1A & -FK1B)	SR	S	05 021540 (TO BE TAKEN FROM STOCK)
7	50	CAP, PIPE, NPS 3/4; CAJON CATALOG NO. SS-12-CP (REF. DWG. N-9013-1-FK1A & -FK1B)	SR	S	06 449240 (TO BE TAKEN FROM STOCK)
8	450	CAP, TUBING, 1/2" DIA.; SWAGELOK CATALOG NO. SS-810-C (REF. DWG. N-9013-1-FK1A &-FK1B)	SR	S	06 181870 (TO BE TAKEN FROM STOCK)

1. Design Change Title/Station/Unit	2. Design Change No.
STEAM GENERATOR REPAIR / NORTH ANNA / UNIT 1	DC 90-13-1

9. Item No.	10. Quantity	11. Description (include any referenced specifications)	12. QA Cat	13. Purchased By	14. Requisition Number
9	40 FT	PIPE, NPS 3/4, SCH. EXTRA STRONG, SEAMLESS; ASME SA106 GRADE B; PIPE CLASS Q2 (REF. DWG. N-9013-1-FK1A & -FK1B)	SR	F	41 046125
10	800 FT	CHANNEL, HOT DIPPED GALVANIZED, 1-5/8" X 3-1/4"; UNISTRUT CATALOG NO. P1001; STD. 20 FT. LENGTHS (REF. DWG. N-9013-1-FK1A & -FK1B)	SR	S	05 007840
11	100 FT	CHANNEL, HOT DIPPED GALVANIZED, 1-5/8" X 1-5/8"; UNISTRUT CATALOG NO. P1000; STD. 20 FT. LENGTHS (REF. DWG. N-9013-1-FK1A & -FK1B)	SR	S	34 201950
12	500	NUT, 1/2" DIA., WITH SPRING; UNISTRUT CATALOG NO. P1010	SR	S	34 215280
13	500	CAP SCREW, HEX HEAD, 1/2" DIA. X 1-3/16"; UNISTRUT CATALOG NO. HHCS050119EG	SR	S	06 197620
14	45	FITTING, FLAT PLATE; UNISTRUT CATALOG NO. P1067	SR	S	34 216860
15	100	FITTING, FLAT PLATE; UNISTRUT CATALOG NO. P1064	SR	S	06 213010
16	240	FITTING, 90 DEGREE ANGLE; UNISTRUT CATALOG NO. P1346	SR	F	41 046013
17	25	FITTING, "Z" SHAPE; UNISTRUT CATALOG NO. N1045EG	SR	F	NS390853
18	100	FITTING, "Z" SHAPE; UNISTRUT CATALOG NO. P1453	SR	F	NS334990
19	100	FITTING, "Z" SHAPE; UNISTRUT CATALOG NO. P5545	SR	F	NS394990
20	60	BOLT, ANCHOR, 1/2" DIA. X 4-1/2", HILTI KWIK BOLT II WITH NUT & WASHER; HILTI CATALOG NO. 000454-736	SR	S	06 242240 (TO BE TAKEN FROM STOCK)
21	250	CLAMP, TUBE SUPPORT, 1-5/8" WIDE FOR 1/2" OD TUBE; ASTM A240 TYPE 304	SR	F	NS 392558
22	500	CAP SCREW, HEX HEAD, 3/8" DIA. X 3/4"; UNISTRUT CATALOG NO. HHCS037075EC	SR	F	

1. Design Change Title/Station/Unit	2. Design Change No.
STEAM GENERATOR REPAIR / NORTH ANNA / UNIT 1	DC 90-13-1

9. Item No.	10. Quantity	11. Description (include any referenced specifications)	12. QA Cat	13. Purchased By	14. Requisition Number
23	500	NUT, 3/8" DIA., WITH SPRING; UNISTRUT CATALOG NO. P1008	SR	S	34 215270 (TO BE TAKEN FROM STOCK)
		<u>RCS PIPING</u>			
24	3	ELBOW, 31" ID X 2.985" MIN. WALL, 40-DEG; ASME SA351 GRADE CF8M; SPEC. NAP-0014, PIPE CLASS Q1 (REF. DWG. N-9013-1-M-401)	SR	E	BNT 369730
		<u>MAIN STEAM PIPING</u>			
25	30 FT	PIPE, 32" OD X 0.970" MIN. WALL, PLAIN ENDS; ASME SA691 GRADE CMS 75, CLASS 32; PIPE CLASS Q2 (REF. DWG. N-9013-1-M-402)	SR	F	CNT 384094
		<u>FEEDWATER PIPING</u>			
26	20 FT	PIPE, NPS 16", SCH. 80, PLAIN ENDS; ASTM A335 GRADE P22, PIPE CLASS Q2 (REF. DWG. N-9013-1-M-403)	SR	S	06 369860 (TO BE TAKEN FROM STOCK)
27	5	ELBOW, NPS 16, SHORT RADIUS 90-DEG., ANSI B16.9, SCH. 80; SEAMLESS, PLAIN ENDS; ASTM A234 GRADE WP22; PIPE CLASS Q2 (REF. DWG. N-9013-1-M-403)	SR	S	06 366080 (TO BE TAKEN FROM STOCK)
28	7	ELBOW, NPS 16, LONG RADIUS 90-DEG., ANSI B16.9, SCH. 80, SEAMLESS, PLAIN ENDS; ASTM A234 GRADE WP22; PIPE CLASS Q2 (REF. DWG. N-9013-1-M-403)	SR	S	06 393280 (TO BE TAKEN FROM STOCK)
29	30 FT	PIPE, NPS 3/4, SCH. EXTRA STRONG, SEAMLESS; ASME SA106 GRADE B; PIPE CLASS Q2 (REF. DWG. N-9013-1-M-403)	SR	F	CNT 384093
30	3	CAP, PIPE, NPS 3/4, ANSI B16.11 CLASS 3000; ASTM A105; PIPE CLASS Q2 (REF. DWG. N-9013-1-M-403)	SR	S	38 151430 (TO BE TAKEN FROM STOCK)
31	3	VALVE, GLOBE, NPS 3/4, ANSI B16.34 CLASS 600, SOCKET WELDED; BODY MATERIAL ASME SA105; SPEC. NAP-0023, PIPE CLASS Q2; MARK NO. VOS-60C (REF DWG. N-9013-1-M-403)	SR	S	05 004270 (TO BE TAKEN FROM STOCK)

1. Design Change Title/Station/Unit: STEAM GENERATOR REPAIR / NORTH ANNA / UNIT 1

2. Design Change No.: DC 90-13-1

9. Item No.	10. Quantity	11. Description (include any referenced specifications)	12. QA Cat	13. Purchased By	14. Requisition Number
32	4	ELBOW, NPS 3/4, 90-DEG, ANSI B16.11 CLASS 3000, SOCKET WELDED; ASME SA105; PIPE CLASS Q2 (REF. DWG-N-9013-1-M-403)	SR	F	
33	3	SOCKOLET, 16" X 3/4", ANSI B16.11 CLASS 3000; ASME SA182 GRADE F22; PIPE CLASS Q2 (REF. DWG N-9013-1-M-403)	SR	F	CNT 384093
		CHEMICAL FEED PIPING			
34	6	UNION, TUBE SOCKET WELD, 3/4" X 3/4", CAJON CATALOG NO. SS-12-TSW-6 (REF. DWG. N-9013-1-M-403)	SR	S	38 757490 (TO BE TAKEN FROM STOCK)
35	40 FT	TUBING, 3/4" OD X 0.109 WALL; ASTM A213 TYPE 316; CLASS Q2; STD. 20 FT. LENGTHS (REF. DWG. N-9013-1-M-403)	SR	S	05 018410 (TO BE TAKEN FROM STOCK)
36	3	ADAPTER, PIPE/TUBE, 3/4" IPS X 3/4" TUBE, CAJON CATALOG NO. SS-12-MPW-A-12TSW (REF. DWG N-9013-1-M-403)	SR	S	06 063450 (TO BE TAKEN FROM STOCK)
		SAMPLE SYSTEM			
37	20 FT	PIPE, NPS 2, SCH. EXTRA STRONG, SEAMLESS, PLAIN ENDS; ASTM A106 GRADE B; PIPE CLASS Q2 (REF. DWG. N-9013-1-M-404 SH. 2)	SR	S	38 850350 (TO BE TAKEN FROM STOCK)
38	40 FT	PIPE, NPS 1, SCH. EXTRA STRONG, SEAMLESS, PLAIN ENDS; ASME SA106 GRADE B; PIPE CLASS Q2 (REF. DWG. N-9013-1-M-404 SH. 2)	SR	F	38 412500
39	3	ELBOW, NPS 1, 90-DEG., ANSI B16.11 CLASS 3000, SOCKET WELDED; ASTM A105; PIPE CLASS Q2 (REF. DWG. N-9013-1-M-404 SH. 2)	SR	S	38 412500 (TO BE TAKEN FROM STOCK)
40	3	REDUCING COUPLING, 2" X 1", ANSI B16.11 CLASS 3000; SOCKET WELD; ASME SA105; PIPE CLASS Q2 (REF. DWG. N-9013-1-M-404 SH. 2)	SR	F	CNT 384093

1. Design Change Title/Station/Unit	2. Design Change No.
STEAM GENERATOR REPAIR / NORTH ANNA / UNIT 1	DC 90-13-1

9. Item No.	10. Quantity	11. Description (include any referenced specifications)	12. QA Cat	13. Purchased By	14. Requisition Number
41	3	COUPLING, NPS 1; ANSI B16.11 CLASS 3000, SOCKET WELDED; ASTM A105; PIPE CLASS Q2 (REF. DWG. N-9013-1-M-404 SH. 2)	SR	S	06 193380 (TO BE TAKEN FROM STOCK)
		STEAM GENERATORS			
42	3	STEAM GENERATOR LOWER TUBE BUNDLE ASSEMBLY, WESTINGHOUSE	SR	E	R1412-0000057 & R1247-0002308, REV. 1
43	3	OUTLET NOZZLE FLOW LIMITER ASSEMBLY	SR	E	
44	3	STEAM GENERATOR UPPER RESTRAINT ASSEMBLY PER SPEC. NAP-0033	SR	E	BNT 367162
45	3	FIT-UP RING, 1/2" X 1-1/2", ASTM A285 GRADE C; PER DWG. N-9013-1-M-405	SR	F	R851807594
46	24	HORIZONTAL BACKING BAR SECTION, 1/2" X 1", ASTM A285 GRADE C; PER DWG. N-9013-1-M-405	SR	F	R851807594
47	12	VERTICAL BACKING BAR, 1/2" X 1" X 7-1/8", ASTM A285 GRADE C; PER DWG. N-9013-1-M-405	SR	F	R851807594
48	12	RADIAL PLATE SECTION, 3/8" X 9-1/8", ASTM A285 GRADE C; PER DWG. N-9013-1-M-405	SR	F	R851807594
49	90 LF	BACKING BAR, 3/8" X 2", ASTM A285 GRADE C; PER DWG. N-9013-1-M-405	SR	F	R851807594
50	3	GASKET, 2-1/2" DIA., CLASS 600, TYPE 304 SS, STYLE CG FLEXITALIC, 1/8" THICK RING (WET LAYUP CONNECTION)	NS	F	NT 396902
51	6	CAP SCREW, HEX HEAD, 5/16" DIA X 1/2", 24UNF-2A, FOR ACCELEROMETER MOUNTING; ASME SA193 GRADE B7 (REF. DWG N-9013-1-E-3500)	SR	F	R851808803
52	6	WASHER, 5/16" DIAMETER; ASTM F436 TYPE 1 STAINLESS STEEL (REF. DWG N-9013-1-E-3500)	SR	F	R851808803

1. Design Change Title/Station/Unit	2. Design Change No.
STEAM GENERATOR REPAIR / NORTH ANNA / UNIT 1	DC 90-13-1

9. Item No.	10. Quantity	11. Description (include any referenced specifications)	12. QA Cat	13. Purchased By	14. Requisition Number
		INSULATION			
53	3	INSULATION, BLANKET TYPE, FOR MODEL 51F SG AND ASSOCIATED PIPING IN ACCORDANCE WITH SPEC. NAP-0047	NSQ	E	CNT 390346
		HVAC			
54	1	GASKET, 18" DIA., ASTM1056-78-RE41GRB NEOPRENE RESILIENT TO 250 F, 1/4" THICK, (REF. DWG. N-9013-1-M-406)	NS	F	
		OLD SG CLOSURE PLATES			
55	6	PLATE, PRIMARY NOZZLE CLOSURE, 3" x 34" DIA.; ASTM A36; SPEC. NAP-0043, PER DWG. N-9013-1-M-800	NS	F	
56	6	PLUG, PRIMARY MANWAY SEAL, 3" X 15 15/16" DIA.; ASTM A36; SPEC. NAP-0043, PER DWG. N-9013-1-M-800	NS	F	
57	6	PLUG, SECONDARY HANDHOLE SEAL, 3" X 5 7/8" DIA.; ASTM A36; SPEC NAP-0043, PER DWG. N-9013-1-M-800	NS	F	
58	6	PLATE, RCS COLD LEG ELBOW CLOSURE; ASTM A36; SPEC. NAS-0043, PER DWG. N-9013-1-M-800	NS	F	
59	9	PLUG, BOTTOM BLOWDOWN NOZZLE AND TRANSITION CONE SEAL, 1 11/16" DIA. X 3"; ASTM A36; SPEC. NAS-0043, PER DWG. N-9013-1-M-800,	NS	F	
60	3	PLUG, SHELL DRAIN SEAL, 13/16" DIA. X 3"; ASTM A36; SPEC. NAS-0043, PER DWG. N-9013-1-M-800	NS	F	
61	3	PLUG, WIDE RANGE LEVEL TAP SEAL, 19/32" DIA. X 3"; ASTM A36; SPEC. NAS-0043, PER DWG. N-9013-1-M-800	NS	F	
62	3	PLATE, TRANSITION CONE COVER, 3 5/8" X 13'-9" DIA.; ASTM A533 CLASS 1, GRADE A; SPEC. NAP-0043, PER DWG. N-9013-1-M-800	NS	F	

1. Design Change Title/Station/Unit

STEAM GENERATOR REPAIR / NORTH ANNA / UNIT 1

2. Design Change No.

DC 90-13-1

9. Item No.	10. Quantity	11. Description (include any referenced specifications)	12. QA Cat	13. Purchased By	14. Requisition Number
		STEAM GENERATOR BLOWDOWN PIPING			
63	300 FT	PIPE, NPS 2 1/2, SCH 80, SEAMLESS; ASTM A335 P22; PIPE CLASS Q3	SR	F	CNT 384093
64	60 FT	PIPE, NPS 1, SCH EXTRA STRONG, SEAMLESS; ASTM A335 P22; PIPE CLASS Q3	SR	F	CNT 384093
65	20 FT	PIPE, NPS 3, SCH 80, SEAMLESS; ASTM A106 GRADE B; PIPE CLASS Q3	SR	F	CNT 384093
66	24	ELBOW, NPS 2 1/2, LONG-RADIUS 90-DEG, ANSI B16.9, SCH 80, BUTT-WELDED, SEAMLESS; ASTM A234 WP22; PIPE CLASS Q3	SR	F	CNT 384093
67	12	ELBOW, NPS 2 1/2, LONG RADIUS 45-DEG, ANSI 16.9, SCH 80, BUTT-WELDED, SEAMLESS; ASTM A234 WP22; PIPE CLASS Q3	SR	F	CNT 384093
68	14	ELBOW, NPS 1, 90-DEG, ANSI B16.11 CLASS 3000, FORGED STEEL, SOCKET WELDED; ASTM A182, GRADE F22; PIPE CLASS Q3	SR	F	CNT 384093
69	4	ELBOW, NPS 1, 45-DEG, ANSI B16.11 CLASS 3000, FORGED STEEL, SOCKET WELDED; ASTM A182 GRADE F22; PIPE CLASS Q3	SR	F	CNT 384093
70	12	REDUCER, CONCENTRIC, 3" X 2 1/2", ANSI B16.9, SCH 80, BUTT-WELDED, SEAMLESS; ASTM A234 WP22; PIPE CLASS Q3	SR	F	CNT 384093
71	5	TEE, REDUCING, 3" X 3" X 2 1/2", ANSI B16.9, SCH 80, SEAMLESS, BUTT-WELDED; ASME SA234 WP22; PIPE CLASS Q3	SR	F	CNT 391372
72	5	SOCKOLET, 2 1/2" X 1", ANSI B16.11 CLASS 3000; ASTM A182 GRADE F22; PIPE CLASS Q3	SR	F	CNT 384093
73	6	VALVE, GLOBE, NPS 2 1/2, ANSI B16.34 CLASS 600, BUTT WELDED, BODY MATERIAL ASTM A182 F22; PIPE CLASS Q3; MARK NO. VOS-60C	SR	F	CNT 3[illegible]737

VIRGINIA POWER

MATERIAL LIST - NUCLEAR POWER STATION

STD-GN-0001 Rev. 10 RIS 14

1. Design Change Title/Station/Unit	2. Design Change No.
STEAM GENERATOR REPAIR / NORTH ANNA / UNIT 1	DC 90-13-1

9. Item No.	10. Quantity	11. Description (include any referenced specifications)	12. QA Cat	13. Purchased By	14. Requisition Number
74	3	VALVE, GLOBE, NPS 1, ANSI B16.34 CLASS 600, SOCKET WELDED; BODY MATERIAL ASTM A182 F22; PIPE CLASS Q3; MARK NO. VOS-60U	SR	F	CNT 392293
		SG BLOWDOWN PIPE SUPPORTS			
75	3	TUBING, STRUCTURAL, 1/4" WALL THICKNESS X 4" X 4" SQ.; ASTM A500 GRADE B, 20' RANDOM LENGTH	SR	S	02 241160 (TO BE TAKEN FROM STOCK)
76	2	TUBING, STRUCTURAL, 1/4" WALL THICKNESS X 3" X 3" SQ.; ASTM A500 GRADE B, 20' RANDOM LENGTH	SR	S	02 241170 (TO BE TAKEN FROM STOCK)
77	1	TUBING, STRUCTURAL, 1/4" WALL THICKNESS X 2" X 2" SQ.; ASTM A500 GRADE B, 20' RANDOM LENGTH, WELDED, HEAT NUMBER REQUIRED	SR	S	06 199800 (TO BE TAKEN FROM STOCK)
78	1	TUBING, STRUCTURAL, 1/4" WALL THICKNESS X 4" X 2" RECT.; ASTM A500 GRADE B, 20' RANDOM LENGTH	SR	S	02 247480 (TO BE TAKEN FROM STOCK)
79	1	TUBING, STRUCTURAL, 1/4" WALL THICKNESS X 2 1/2" X 2 1/2" SQ.; ASTM A500 GRADE B, 20' RANDOM LENGTHS	SR	F	NS 390455
80	1	TUBING, STRUCTURAL, 3/8" WALL THICKNESS X 6 X 6 SQ.; ASTM A500 GRADE B, 20' RANDOM LENGTHS	SR	F	NS 390455
81	1	PLATE, STRUCTURAL, 3/4" X 48" X 96"; ASTM A36	SR	S	06 125570 (TO BE TAKEN FROM STOCK)
82	1	PLATE, STRUCTURAL, [illegible]" X 48" X 96"; ASTM A36	SR	S	06 135110 (TO BE TAKEN FROM STOCK)
83	1	PLATE, STRUCTURAL, 1/2" X 48" X 96"; ASTM A36	SR	S	06 135350 (TO BE TAKEN FROM STOCK)
84	1	PLATE, STRUCTURAL, 1/4" X 48" X 96"; ASTM A36	SR	S	40 070284
85	72	PIPE PROTECTION SADDLE, 2 1/2", FOR 2" INSULATION THICKNESS, GRINNELL, FIG. 162	SR	F	NS 396112

VIRGINIA POWER

MATERIAL LIST - NUCLEAR POWER STATION

STD-GN-0001 Rev. 10 RIS 14

1. Design Change Title/Station/Unit	2. Design Change No.
STEAM GENERATOR REPAIR / NORTH ANNA / UNIT 1	DC 90-13-1

9. Item No.	10. Quantity	11. Description (include any referenced specifications)	12. QA Cat	13. Purchased By	14. Requisition Number
86	2	PIPE CLAMP, 2 1/2", DOUBLE BOLT, CHROME MOLYBDENUM, GRINNELL, FIG. 295A	SR	F	NS 396112
87	6	PIPE PROTECTION SADDLE, 1", FOR 2" INSULATION THICKNESS, GRINNELL, FIG. 162	SR	F	NS 398017
88	12	PIPE PROTECTION SADDLE, 3", FOR 3" INSULATION THICKNESS, GRINNELL, FIG. 164	SR	F	NS 396112
89	1	HANGER, VARIABLE SPRING, SIZE 00, TYPE A, HL = 47#, CL = 51#, MARK NO. FPH-WGCB-5-4, GRINNELL, FIG. NO. B-268	SR	F	NS 396112
90	1	HANGER, VARIABLE SPRING, SIZE 00, TYPE A, HL = 59#, CL = 52#, MARK NO. FPH-WGCB-8-4, GRINNELL, FIG. NO. B-268	SR	F	NS 396112
91	1	HANGER, VARIABLE SPRING, SIZE 6, TYPE A HL = 381#, CL = 373#, MARK NO. 1-WGCB-SH-MFSK-1975A, GRINNELL, FIG. NO. B-268	SR	F	NS 396112
92	1	HANGER, VARIABLE SPRING, SIZE 4, TYPE A, HL = 233#, CL = 227#, MARK NO. 1-WGCB-SH-1971A, GRINNELL, FIG. NO. 82	SR	F	NS 396112
93	2	ROD, ALL THREAD, 5/8" DIA. X 9"; GRINNELL, FIG. NO. 146	SR	F	NS 396112
94	1	EYE NUT, 5/8" DIA., WELDLESS, GRINNELL, FIG. NO. 290	SR	F	NS 396112
95	5	NUT, HEX, 5/8" DIA.; ASTM A194 GRADE 2H	SR	S	06 189790 (TO BE TAKEN FROM STOCK)
96	12	NUT, HEAVY HEX, 3/4" DIA., 10-UNC-2B; ASTM A194 GRADE 2H	SR	S	06 117750 (TO BE TAKEN FROM STOCK)
97	12	WASHER, FLAT, 3/4", CS, ASTM F436	SR	F	NS 387995
98	12	NUT, HEAVY HEX, 1/2" DIA, 13-UNC-2B; ASTM A194 GRADE 2H	SR	S	06 125490 (TO BE TAKEN FROM STOCK)
99	12	NUT, HEAVY HEX JAM, 1/2" DIA, 13-UNC-2B; ASTM A194 GRADE 2H	SR	F	NS 387995

VIRGINIA POWER

MATERIAL LIST - NUCLEAR POWER STATION

1. Design Change Title/Station/Unit	2. Design Change No.
STEAM GENERATOR REPAIR / NORTH ANNA / UNIT 1	DC 90-13-1

9. Item No.	10. Quantity	11. Description (include any referenced specifications)	12. QA Cat	13. Purchased By	14. Requisition Number
100	12	BOLT, 1/2" DIA. X 2 1/2", 13-UNC-2A; ASTM A193 GRADE B7	SR	S	06 125490 (TO BE TAKEN FROM STOCK)
101	12	WASHER, FLAT, 1/2", CS; ASTM F436	SR	S	40 102547 (TO BE TAKEN FROM STOCK)
102	12	BOLT, 3/4" DIA. X 7", 10-UNC-2A; ASTM A193 GRADE B7	SR	F	NS 387995
		CONDENSATE POTS			
103	20 FT	PIPE, NPS 3, SCH. 40S, SEAMLESS; ASME SA312 TYPE 304; PIPE CLASS Q2	SR	F	CNS 397423
104	20 FT	PIPE, NPS 3/4, SCH. 80, SEAMLESS; ASTM A376 TYPE 316; PIPE CLASS Q2	SR	S	(TO BE TAKEN FROM STOCK)
105	28	CAP, PIPE, NPS 3, ANSI 16.9, SCH. 40S; ASTM A182 TYPE F304; PIPE CLASS Q2	SR	S	05 19720 (TO BE TAKEN FROM STOCK)
106	28	HALF COUPLING, NPS 3/4, ANSI B16.11 CLASS 3000; ASTM A182 TYPE F316; PIPE CLASS Q2	SR	S	05 019920 (TO BE TAKEN FROM STOCK)
107	14	HALF COUPLING, NPS 1/2, ANSI B16.11 CLASS 3000; ASTM A182 TYPE F316; PIPE CLASS Q2	SR	S	05 019910 (TO BE TAKEN FROM STOCK)
108	10 FT	PIPE, NPS 1/2, SCH. 80, SEAMLESS; ASME SA312 TYPE 304; PIPE CLASS Q2	SR	F	CNS 397423
109	10 FT	PIPE, NPS 3/4, SCH. 80, SEAMLESS; ASME SA106 GRADE B; PIPE CLASS Q2	SR	F	
110	28	ADAPTER, TUBE TO PIPE, 1/2 X 3/4", CAJON SS-12-MPW-A-8TSW	SR	S	05 021540 (TO BE TAKEN FROM STOCK)
		SG SUPPORTS			
111	24	LUBRITE PLATE, 9" X 4 1/2" X 1" LUBRITE ALLOY #424, MATING PIECE ASTM A36	SR	F	R851807783
112	12	LUBRITE PLATE, 19" X 8 7/8" X 1" LUBRITE ALLOY #424, MATING PIECE ASTM A36	SR	F	R851807783

1. Design Change Title/Station/Unit	2. Design Change No.
STEAM GENERATOR REPAIR / NORTH ANNA / UNIT 1	DC 90-13-1

9. Item No.	10. Quantity	11. Description (include any referenced specifications)	12. QA Cat	13. Purchased By	14. Requisition Number
113	12	PLATE, VERTICAL SUPPORT, 25" X 9 1/2" X 7"; AISI 4340, HEAT TREAT TO OBTAIN 140-160 KSI YIELD STRENGTH	SR	F	
114	72	CAP SCREW, SOCKET HEAD, 1 1/2" DIA. X 9", 12UNF2A; ASTM A574	SR	F	NS 394370
115	72	SCREW THREAD INSERT, STAINLESS STEEL TYPE 304, HELICOIL TYPE 3591, SIZE 24 CN 4.500" LONG, BLACK AND DECKER	SR	F	MSR 59007169
116	96	MACHINE SCREW, 1/2" DIA. X 1 1/2", 13UNC2A, CARBON STEEL, FLAT PHILLIPS HEAD	SR	F	CNS 394377
117	24	BOLT, HEX HEAD, 2" DIA. X 10", 8UN2A, HEAVY SEMI-FINISHED; ASTM A354 GRADE BD	SR	F	NS 394370
118	24	BOLT, HEX HEAD, 2" DIA. X 9", 8UN2A, HEAVY SEMI-FINISHED; ASTM A354 GRADE BD	SR	F	NS 394370
119	48	NUT, HEAVY HEX, 2" DIA., 8UN2B, SEMI-FINISHED; ASTM A563 GRADE DH	SR	F	NS 394381
120	48	LOCKWASHER, 2 DIA.; CARBON STEEL	SR	F	NS 394370
121	1 LOT	PLATE (SHIM MATERIAL), 13" X 16", THICKNESSES OF 5/8", 3/4",AND [illegible]; ASTM A36	SR	F	CNS 397497
122	1 LOT	PLATE (SHIM MATERIAL), 6 1/2" X 9 1/2", THICKNESSES OF 5/8", 3/4", AND 7/8"; ASTM A36	SR	F	CNS 397497
123	1 LOT	PLATE (SHIM MATERIAL), 9" X 19", THICKNESSES OF 3/16", 1/4", 3/8", 5/8", AND 1"; ASTM A36	SR	F	CNS 397497
124	1 LOT	PLATE (SHIM MATERIAL), 10" X 13", THICKNESSES OF 1/4", 1/2", 3/4", 7/8", AND 1"; ASTM A36	SR	F	CNS 397497
125	72	WASHER, 2 1/4" OD X 1 9/16" ID X 1/16", COPPER; ASTM B152, UNS NO. C10200, HOT ROLLED AND ANNEALED (025)	SR	F	MSR 59007160
126	1	SPECIAL HELICOIL INSERTING TOOL, 535-24, BLACK AND DECKER	NS	F	NS 399513

CONTROLLED DOCUMENT SUMMARY

VIRGINIA POWER

Controlled Document Summary (CDS)

VPAP-0301

POW 1

1. Station	2. Unit	3. Design Change Number
☐ Surry ☒ North Anna	☒ 1 ☐ 2	9013

4. DCP Title: STEAM GENERATOR REPAIR

Procedures

5. Item Number	6. Change Required	7. Item (Use * to Show Priority)	8. Person Contacted (If Applicable)
1	☒ Yes ☐ No	*Periodic Test Procedures	
2	☒ Yes ☐ No	*Station Operating Procedures	
3	☐ Yes ☒ No	*Annunciator Procedures	
4	☒ Yes ☐ No	*Emergency Procedures (EPAPs, EOPs, EPIPs)	
5	☒ Yes ☐ No	*[illegible]	
6	☒ Yes ☐ No	*[illegible]	
7	☒ Yes ☐ No	*Chemistry Procedures	LAMBERSON
8	☒ Yes ☐ No	*Health Physics Procedures	
9	☐ Yes ☒ No	Administrative Procedures	
[illegible]	☐ Yes ☒ No	Loss Prevention Procedures	
[illegible]	☐ Yes ☒ No	Security Procedures	
12	☒ Yes ☐ No	*Surveillance Procedures	
13	☐ Yes ☐ No		
14	☐ Yes ☐ No		

Computer Databases/Software

5. Item Number	6. Change Required	7. Item (Use * to Show Priority)	8. Person Contacted (If Applicable)
1	☒ Yes ☐ No	Chesterton Valve Packing Database	
2	☒ Yes ☐ No	P250 Computer Software	PERRINE
3	☒ Yes ☐ No	ERF Computer Software	
4	☒ Yes ☐ No	Simulator Software (Hardware modifications may also be required.)	
5	☐ Yes ☒ No	Post Maintenance Testing Database	
6	☐ Yes ☒ No	Maintenance Check Valve Database	
7	☐ Yes ☒ No	Maintenance Relief Valve Database	
8	☐ Yes ☒ No	Snubber Tracking Database	
9	☐ Yes ☒ No	Local Area Network Software	
10	☐ Yes ☒ No	MOV Database	
[illegible]	☐ Yes ☒ No	Breaker Database	
12	☐ Yes ☒ No	Vibration Analysis Database	

Key: EPAP's-Emergency Planning Administrative Procedures; EOP's-Emergency Operating Procedures; EPIP's-Emergency Planning Implementing Procedures; ERF-Emergency Response Facility

Computer Databases/Software - Continued

5. Item Number	6. Change Required	7. Item (Use * to Show Priority)	8. Person Contacted (If Applicable)
13	☐Yes ☐No		
14	☐Yes ☐No		
15	☐Yes ☐No		

Other Controlled Documents

5. Item Number	6. Change Required	7. Item (Use * to Show Priority)	8. Person Contacted (If Applicable)
1	☒Yes ☐No	*Plant Drawings	
2	☒Yes ☐No	Q-List	
3	☒Yes ☐No	Procurement Specifications	
4	☒Yes ☐No	Installation Specifications	
5	☒Yes ☐No	System and Plant Design Basis Documents	
6	☒Yes ☐No	Design Calculations	
7	☒Yes ☐No	Class 1 Stress Reports (North Anna Power Station Only)	
8	☐Yes ☒No	Appendix R Report	
	☐Yes ☒No	Station Electrical Load List	
	☐Yes ☒No	Qualifications Document Review Packages	
11	☐Yes ☒No	Environmental Zone Descriptions	
12	☐Yes ☒No	EQ Master List	
13	☐Yes ☒No	EQ Maintenance Manual	
14	☐Yes ☒No	EQ Procurement Manual	
15	☐Yes ☒No	Technical Specifications	
16	☒Yes ☐No	UFSAR	
17	☒Yes ☐No	Curve Book	
18	☒Yes ☐No	Training Manuals	
19	☒Yes ☐No	* North Anna Setpoint Document, Surry Power Station DRPs	
20	☒Yes ☐No	North Anna Precautions, Limitations and Setpoint Document	
21	☐Yes ☒No	Fire Protection Plan	
22	☐Yes ☒No	Security Plan	
23	☒Yes ☐No	Vendor Technical Manuals	
24	☐Yes ☒No	ASME XI - IWV Valve Program	
	☐Yes ☒No	ASME XI - IWP Pump Program	
26	☒Yes ☐No	ISI Plan	

Key: Q-List-Quality List; EQ-Environmental Qualification; UFSAR-Updated Final Safety Analysis Report; ASME-American Society of Mechanical Engineers; ISI-In-service Inspection; DRPs-Design Reference Procedures

VPAP-0301

Other Controlled Documents - Continued

5. Item Number	6. Change Required	7. Item (Use * to Show Priority)	8. Person Contacted (If Applicable)
27	☒Yes ☐No	Spare Part Stocking Level Requirements	
28	☐Yes ☒No	Crane Program	
29	☐Yes ☐No		
30	☐Yes ☐No		
31	☐Yes ☐No		

9. Remarks

10. Preparer/Affiliation (Print Name)	10a. Preparer (Signature)	10b. Date
MARK S. BARTH / BECHTEL	MS Barth	9/18/92
11. Reviewer/Affiliation (Print Name)	**11a. Reviewer (Signature)**	**11b. Date**
MARK A. SMITH / BECHTEL	M.A. Smith	9/18/92

VIRGINIA POWER

Controlled Document Summary (CDS) - Supplement Sheet

POW 11

VPAP-0301

1. Station	2. Design Change Number	3. Responsible Department	4.
☐ Surry ☑ North Anna			Page 1 of 1

5. DCP Title

6. Procedure Number	7. Other Referenced Documents (such as VTMs) which must be treated as priorities.

8. Comments

9. Information Provided By (Print Name)	9a. Information Provided By (Signature)	9b. Date

Key: DCP-Design Change Package; VTMs-Vendor Technical Manuals

Form No. 721804(Mar 92)

VIRGINIA POWER

Controlled Document Summary (CDS) - Supplement Sheet

POW 11

VPAP-0301

1. Station	2. Design Change Number	3. Responsible Department	4.
☐ Surry ☒ North Anna	90-13	STATION PROCEDURES	Page 1 of 11

5. DCP Title

STEAM GENERATOR REPLACEMENT PROJECT

6. Procedure Number	7. Other Referenced Documents (such as VTMs) which must be treated as priorities.
0-ECM-0205-01	
1-PT-18	
1-PT-27	
1-PT-138	
0-MCM-1109-01	
0-MCM-1109-02	
0-MCM-1110-01	
0-MCM-1110-02	
0-MCM-0504-01	
0-MCM-0504-02	
0-MCM-1303-01	
0-MCM-1208-02	

8. Comments

9. Information Provided By (Print Name)	9a. Information Provided By (Signature)	9b. Date
EDWARD J. WELLS	Edward J. Wells	9-22-92

Key: DCP-Design Change Package; VTMs-Vendor Technical Manuals

Form No. 721604(Mar 92)

VIRGINIA POWER

Controlled Document Summary (CDS) - Supplement Sheet

POW 11

VPAP-0301

1. Station	2. Design Change Number	3. Responsible Department	4.
☐ Surry ☒ North Anna	90-13	STATION PROCEDURES	Page 2 of 11

5. DCP Title
STEAM GENERATOR REPLACEMENT PROJECT

6. Procedure Number	7. Other Referenced Documents (such as VTMs) which must be treated as priorities.
0-MCM-1209-01	
0-MCM-1209-02	
1-E-1	
1-ECA-1.1	
1-FR-C.1	
1-FR-C.3	
1-ES-1.1	
1-ES-1.2	
1-E-3	
1-OP-3.4	
1-OP-5.4	
1-OP-33.2	

8. Comments

9. Information Provided By (Print Name)	9a. Information Provided By (Signature)	9b. Date
EDWARD J. WELLS	Edward J. Wells	9-22-92

Key: DCP-Design Change Package; VTMs-Vendor Technical Manuals

Form No. 721604(Mar 92)

VIRGINIA POWER

Controlled Document Summary (CDS) - Supplement Sheet

POW 11

VPAP-0301

1. Station	2. Design Change Number	3. Responsible Department	4.
☐ Surry ☒ North Anna	90-13	STATION PROCEDURES	Page 3 of 11

5. DCP Title

STEAM GENERATOR REPLACEMENT PROJECT

6. Procedure Number	7. Other Referenced Documents (such as VTMs) which must be treated as priorities.
1-ES-3.1	
1-ES-3.2	
1-ES-3.3	
1-ECA-2.1	
1-ECA-3.1	
1-ECA-3.2	
1-FR-H.1	
1-ECA-0.2	
1-FR-C.2	
1-PT-31.1.1	
1-PT-31.1.2	
1-PT-31.1.3	

8. Comments

9. Information Provided By (Print Name)	9a. Information Provided By (Signature)	9b. Date
EDWARD J. WELLS	Edward J. Wells	9-22-92

Key: DCP-Design Change Package; VTMs-Vendor Technical Manuals

Form No. 721604(Mar 92)

VIRGINIA POWER

Co[illegible] d Document Summary (CDS) - Supplement Sheet

POW 11

VPAP 0301

1. Station	2. Design Change Number	3. Responsible Department	4.
☐ Surry ☒ North Anna	90-13	STATION PROCEDURES	Page 4 of 11

5. DCP Title

STEAM GENERATOR REPLACEMENT PROJECT

6. Procedure Number	7. Other Referenced Documents (such as VTMs) which must be treated as priorities.
1-PT-30.2.1	
1-PT-30.2.2	
1-PT-30.2.3	
1-PT-30.2.4	
1-PT-45.1.1	
1-PT-45.1.2	
1-PT-45.1.3	
1-PT-45.1.4	
1-PT-32.1.1	
1-PT-32.1.4	
1-PT-32.1.7	
1-PT-32.1.2	

8. Comments

9. Information Provided By (Print Name)	9a. Information Provided By (Signature)	9b. Date
EDWARD J. WELLS	Edward J. Wells	9-22-92

Key: DCP-Design Change Package; VTMs-Vendor Technical Manuals

Form No. 721804(Mar 92)

VIRGINIA POWER

Controlled Document Summary (CDS) - Supplement Sheet

POW 11

VPAP-0301

1. Station	2. Design Change Number	3. Responsible Department	4.
☐ Surry ☒ North Anna	90-13	STATION PROCEDURES	Page 5 of 11

5. DCP Title

STEAM GENERATOR REPLACEMENT PROJECT

6. Procedure Number	7. Other Referenced Documents (such as VTMs) which must be treated as priorities.
1-PT-32.1.5	
1-PT-32.1.8	
1-PT-32.1.3	
1-PT-32.1.6	
1-PT-32.1.9	
1-PT-32.3.4	
1-PT-32.3.1	
1-PT-32.3.5	
1-PT-32.3.2	
1-PT-32.3.6	
1-PT-32.3.3	
1-PT-32.5.1	

8. Comments

9. Information Provided By (Print Name)	9a. Information Provided By (Signature)	9b. Date
EDWARD J. WELLS	Edward J. Wells	9-22-92

Key: DCP-Design Change Package; VTMs-Vendor Technical Manuals

Form No. 721604(Mar 92)

VIRGINIA POWER

Controlled Document Summary (CDS) - Supplement Sheet

POW 11

VPAP-0301

1. Station	2. Design Change Number	3. Responsible Department	4.
☐ Surry ☒ North Anna	90-13	STATION PROCEDURES	Page 6 of 11

5. DCP Title: STEAM GENERATOR REPLACEMENT PROJECT

6. Procedure Number	7. Other Referenced Documents (such as VTMs) which must be treated as priorities.
1-PT-32.5.2	
1-PT-32.5.3	
ICP-RC-1-T-1412	
ICP-RC-1-T-1422	
ICP-RC-1-T-1432	
ICP-NI-1-N-41	
ICP-NI-1-N-42	
ICP-NI-1-N-43	
ICP-NI-1-N-44	
1-IMP-MS-T-408	
1-ICP-QS-L-100A	
1-ICP-QS-L-100B	

8. Comments

9. Information Provided By (Print Name)	9a. Information Provided By (Signature)	9b. Date
EDWARD J. WELLS	Edward J. Wells	9-22-92

Key: DCP-Design Change Package; VTMs-Vendor Technical Manuals

Form No. 721804 (Apr 92)

VIRGINIA POWER

Controlled Document Summary (CDS) - Supplement Sheet

POW 11

VPAP-0301

1. Station	2. Design Change Number	3. Responsible Department	4.
☐ Surry ☒ North Anna	90-13	STATION PROCEDURES	Page 7 of 11

5. DCP Title

STEAM GENERATOR REPLACEMENT PROJECT

6. Procedure Number	7. Other Referenced Documents (such as VTMs) which must be treated as priorities.
1-ICP-QS-L-100C	
1-ICP-QS-L-100D	
1-ICP-FW-L-1474	
1-ICP-FW-L-1475	
1-ICP-FW-L-1476	
1-ICP-FW-L-1477	
1-ICP-FW-L-1484	
1-ICP-FW-L-1485	
1-ICP-FW-L-1486	
1-ICP-FW-L-1487	
1-ICP-FW-L-1494	
1-ICP-FW-L-1495	

8. Comments

9. Information Provided By (Print Name)	9a. Information Provided By (Signature)	9b. Date
EDWARD J. WELLS	Edward J. Wells	9-22-92

Key: DCP-Design Change Package; VTMs-Vendor Technical Manuals

Form No. 721604(Mar 92)

VIRGINIA POWER

Controlled Document Summary (CDS) - Supplement Sheet

POW 11

VPAP-0301

1. Station	2. Design Change Number	3. Responsible Department	4.
☐ Surry ☒ North Anna	90-13	STATION PROCEDURES	Page 8 of 11

5. DCP Title: STEAM GENERATOR REPLACEMENT PROJECT

6. Procedure Number	7. Other Referenced Documents (such as VTMs) which must be treated as priorities.
1-ICP-FW-L-1496	
1-ICP-FW-L-1497	
ICP-P-1-F-1478	
ICP-P-1-F-1488	
ICP-P-1-F-1498	
1-ICP-MS-F-1475	
1-ICP-MS-F-1474	
1-ICP-MS-F-1485	
1-ICP-MS-F-1484	
1-ICP-MS-F-1495	
1-ICP-MS-F-1494	
1-ICP-MS-F-14,4	

8. Comments

9. Information Provided By (Print Name)	9a. Information Provided By (Signature)	9b. Date
EDWARD J. WELLS	Edward J. Wells	9-22-92

Key: DCP-Design Change Package; VTMs-Vendor Technical Manuals

Form No. 721804(Mar 92)

VIRGINIA POWER

Controlled Document Summary (CDS) - Supplement Sheet

POW 11

VPAP-0301

1. Station	2. Design Change Number	3. Responsible Department	4.
☐ Surry ☒ North Anna	90-13	STATION PROCEDURES	Page 9 of 11

5. DCP Title

STEAM GENERATOR REPLACEMENT PROJECT

6. Procedure Number	7. Other Referenced Documents (such as VTMs) which must be treated as priorities.
1-ICP-MS-P-1475	
1-ICP-MS-P-1476	
1-ICP-MS-P-1486	
1-ICP-MS-P-1485	
1-ICP-MS-P-1484	
1-ICP-MS-P-1494	
1-ICP-MS-P-1495	
1-ICP-MS-P-1496	
ICP-MS-1-P-1446A	
ICP-P-1-P-447	
ICP-P-1-P-446B	
ICP-P-1-T-408	

8. Comments

9. Information Provided By (Print Name)	9a. Information Provided By (Signature)	9b. Date
EDWARD J. WELLS	Edward J. Wells	9-22-92

Key: DCP-Design Change Package; VTMs-Vendor Technical Manuals

Form No. 721804(Mar 92)

VIRGINIA POWER

Controlled Document Summary (CDS) - Supplement Sheet

POW 11

VPAP-0301

1. Station	2. Design Change Number	3. Responsible Department	4.
☐ Surry ☒ North Anna	90-13	STATION PROCEDURES	Page 10 of 11

5. DCP Title: STEAM GENERATOR REPLACEMENT PROJECT

6. Procedure Number	7. Other Referenced Documents (such as VTMs) which must be treated as priorities.
ICP-P-1-T-409A	
ICP-P-1-T-409B	
IMP-C-MISC-06	
1-AP-11	
1-AR-3	
1-AR-9	
1-OP-2.1	
1-OP-2.2	
1-OP-3.3	
1-OP-4.1	
1-OP-5A	
1-OP-33A	

8. Comments

9. Information Provided By (Print Name)	9a. Information Provided By (Signature)	9b. Date
EDWARD J. WELLS	Edward J. Wells	4-22-92

Key: DCP-Design Change Package; VTMs-Vendor Technical Manuals

Form No. 721604(Mar 92)

VIRGINIA POWER

Controlled Document Summary (CDS) - Supplement Sheet

POW 11

VPAP-0301

1. Station	2. Design Change Number	3. Responsible Department	4.
☐ Surry ☒ North Anna	90-13	STATION PROCEDURES	Page 11 of 11

5. DCP Title

STEAM GENERATOR REPLACEMENT PROJECT

6. Procedure Number	7. Other Referenced Documents (such as VTMs) which must be treated as priorities.
1-OP-30.5	
1-MOP-28.31	
1-MOP-28.32	
1-MOP-28.33	
1-MOP-28.34	
1-MOP-28.35	
1-MOP-28.36	
1-MOP-28.46	
1-MOP-5.90	
1-TOP-26.15	
1-TMOP-26.15	
1-MOP-33.4	

8. Comments

9. Information Provided By (Print Name)	9a. Information Provided By (Signature)	9b. Date
EDWARD J. WELLS	Edward J. Wells	9-22-92

Key: DCP-Design Change Package; VTMs-Vendor Technical Manuals

Form No. 721804(Mar 92)

VIRGINIA POWER

Controlled Document Summary (CDS) - Supplement Sheet

POW 11

1. Station	2. Design Change	3. Responsible Department	4.
☐ Surry ☒ North Anna	90-13	System Eng/Rx Eng	Page 1 of 1

5. DCP Title: Steam Generator Replacement

6. Procedure Number	7. Other Referenced Documents (such as VTMs) which must be treated as priorities
1-PT-41.1 (1)	

8. Comments

No System Eng. PT's /surveillance require changes as a result of 90-13.

(1) This PT contains a curve for the Power Correction Factor used to adjust the indicated SG level. The existing curve needs to be evaluated and adjusted if required using measured data from the startup and operation of the plant. (Rx Eng. PT)

9. Information Provided By (Print Name)	9a. Information Provided By (Signature)	9b. Date
Leo T. Meire	Leo T. Meire	8-21-92

Key: DCP-Design Change Package; VTMs-Vendor Technical Manuals

Form No. 721604(Mar 92)

VIRGINIA POWER

Controlled Document Summary (CDS) - Supplement Sheet

POW 1

VPAP-0301

1. Station	2. Design Change Number	3. Responsible Department	4.
☐ Surry ☒ North Anna		CHEMISTRY	Page 1 of 1

5. DCP Title
STEAM GENERATOR REPLACEMENT PROJECT

6. Procedure Number	7. Other Referenced Documents (such as VTMs) which must be treated as priorities.
1-PT-46.3C	CORPORATE CHEMISTRY NUCLEAR PLANT CHEMISTRY MANUAL
1-PT-72.3	
1-PT-72.5	
ADM 19.22	
CAP-28	
CH-93 ~~[illegible]~~ 120	
CH-97.100	
CH-98.701	
CH-87.201	

8. Comments

9. Information Provided By (Print Name)	9a. Information Provided By (Signature)	9b. Date
J R LAMBERSON	JR Lamberson	9-14-92

Key: DCP-Design Change Package; VTMs-Vendor Technical Manuals

Form No. 721804(Mar

VIRGINIA POWER

Controlled Document Summary (CDS) - Supplement Sheet

POW 11

VPAP-0301

1. Station	2. Design Change Number	3. Responsible Department	4.
☐ Surry ☒ North Anna	90-13-1		Page 1 of 1

5. DCP Title

Steam Generator Replacement

6. Procedure Number	7. Other Referenced Documents (such as VTMs) which must be treated as priorities.
	1) SPDS Display Designs, DIS-VEPCO-006
	2) SPDS Algorithm Design, ALG-VEPCO-004
	3) SPDS Software Test Analysis Report
	1) DCP Input/Output List for North Anna Power Station

8. Comments

9. Information Provided By (Print Name)	9a. Information Provided By (Signature)	9b. Date
L. C. Beck	[signature]	9/3/92

Key: DCP-Design Change Package; VTMs-Vendor Technical Manuals

Form No. 7[illegible] (Mar 92)

SAFETY EVALUATION

DC 90-13-1, STEAM GENERATOR REPLACEMENT / NORTH ANNA / UNIT 1

Safety Evaluation Number	2. Applicable Station	3. Applicable Unit
92-SE-MOD-054	[X] North Anna Power Station [ ] Surry Power Station	[X] Unit 1 [ ] Unit 2 [ ] Unit 1 [ ] Unit 2

PART A - Resolution Summary Report

4. List the governing documents for which this safety evaluation was performed.

DC 90-13-1

5. Summarize the change, test, or experiment evaluated.

See Page 1A for response.

6. State the purpose for this change, test, or experiment.

See Page 1B for response.

7. List the limiting conditions and special requirements identified or assumed by this safety analysis.

See Page 1B for response.

8. Will the proposed activity/condition result in or constitute an unreviewed safety question or require a licensing amendment? [ ] Yes [X] No

9. Preparer Name (Print)	10. Preparer Signature	11. Date
M.A. Smith / S.D. Roth / M.S. BARTH / D. NEVERGOLD / S.L. Wilkie	M.A. Smith / S.D. Roth / M.S. Barth / D. Nevergold / S.L. Wilkie	9/18/92 / 9-21-92
12. Cognizant Supervisor Name (Print)	13. Cognizant Supervisor Signature	14. Date
R.K. BAYER / K.L. Basehore	R.K. Bayer / K.L. Basehore	9/21/92 / 9-22-92

15. Disposition [X] Approved [ ] Disapproved [ ] Approved As Modified [ ] Requires Further Evaluation

16. SNSOC Chairman Signature	17. Date
J. Seel	9/30/92

Comments: Supplemented responses to Questions 7, 18, 28, 53, 64B, 65A to acknowledge purge system operation per station procedure, to reflect radiation monitoring available, to identify presence of temporary hatch cover, and to discuss radiological consequences of in-containment releases.

S.D. Roth 9/30/92 / S.L. Wilkie 9-30-92 K.L. Basehore per telecon S.L. Wilkie 9-30-92

1: Attach a Copy of Part A, Resolution Summary Report to the Change/Activity Documentation Package.
Send Copy of Part A to Licensing for Submittal to the NRC in Accordance With VPAP-2802, Reporting Requirements.
Send the Completed Safety Evaluation Original to Records Management.
Use "Safety Evaluation, Supplemental Page" Form No. 730928, if additional space is needed.

5. Summarize the change, test, or experiment evaluated.

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

I. Steam Generator Vessel Repair
II. Piping Removal and Replacement
III. Instrumentation Removal and Replacement
IV. Blowdown System Modification
V. Insulation Removal and Replacement
VI. Rigging Activities
VII. Steam Generator Haul Route
VIII. Temporary Services (impact on permanent facilities only)

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

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

III. The steam generator level instrument piping and tubing will be severed to allow removal of the original steam generator lower assemblies and installation of the new lower assemblies. The condensate pots and instrument root valves will also be removed and replaced. The severed piping and tubing as well as the condensate pots will be reinstalled to their original configurations with material upgrades. In addition, the loose parts monitors will be relocated as part of DC 90-13-1, and the optical templating bracket installed under DC 92-006-1 will be removed.

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

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

VI. Lifting and handling activities required to support removal and installation of steam generator lower assemblies will be performed.

VII. Establish and test the proposed haul route to be used to transport the new steam generators to the equipment hatch and to transport the old steam generators to the Old Steam Generator Storage Facility.

VIII. Temporary modifications to support steam generator replacement will be required. These temporary modifications include attachment of a flexible duct and volume control damper to the purge system, modification to RCP-1B power supply for temporary steam generator replacement power, modification to security door A-9[illegible] an auxiliary crane, a jib crane, and a reactor cavity cover.

6. State the purpose for this change, test, or experiment.

The purpose of the eight major activities associated with steam generator repair is as follows:

I. Steam generator vessel repair will restore the steam generators to their original performance level and improve the steam generator operational characteristics.

II. Piping removal and reinstallation supports the restoration of the steam generators to their original performance level.

III. Instrumentation removal and reinstallation supports the restoration of the steam generators to their original performance level.

IV. Modifications to the blowdown system enhances the piping material properties to reduce the potential for pipe wall thinning in the steam generator blowdown system resulting from erosion/corrosion and provides the capability to increase the blowdown flowrates, if desired.

V. The existing insulation will be replaced with new blanket insulation having insulating properties which are equivalent to or exceed the insulating properties of the original insulation.

VI. Rigging activities are required to support restoration of the steam generators to their original performance level. Safe load handling and transport of heavy loads associated with steam generator replacement are addressed in subsequent sections of this safety evaluation.

VII. Establishing the steam generator haul route is required to support safe transport of the lower assemblies.

VIII. Temporary modifications are being made to provide temporary services, hoists and a cavity cover necessary to to perform the work to restore the steam generators to their original performance level.

7. List the limiting conditions and special requirements identified or assumed by this safety analysis.

All limiting conditions and special requirements identified in DC 90-13-1, Section 2.3, have been assumed in this safety evaluation. The following is a brief summary of those assumptions and special requirements. For a complete listing of the assumptions and special requirements, see DC 90-13-1.

I. The replacement of the steam generator vessel, including replacement of the lower assembly and installation of flow restrictor in the main steam nozzle, will be performed with the reactor vessel defueled. The repaired steam generators were evaluated against the conditions, requirements and assumptions identified in the UFSAR (see Technical Report NE-883 and Westinghouse Safety Evaluation, SECL-90-113). An inspection for loose parts will be performed after installation and startup testing will verify critical SG operating parameters. Removal of the upper lateral restraints will not be performed until the vessel is defueled.

II. The reactor coolant loops A, B, and C will have temporary shielding installed on the hot leg, crossover leg, and bypass line prior to the RCS piping cuts. RC Loop B will be severed first, prior to the completion of fuel offload. This configuration will exist with either the hot or crossover leg severed, or with both lines severed. The acceptability of these configurations is documented in DC 90-13-1, Reference 6.12.21. Cutting of Loop A and C RCS piping will not be performed until the vessel is defueled. The acceptability of the RC loops A, B, and C piping severed from the steam generators after fuel is removed from the reactor is also documented in DC 90-13-1, Reference 6.12.21.

7. (Continued)

Secondary plant piping systems including main steam, feedwater, feedwater drain, wet layup, steam generator blowdown and shell drain, chemical feed, sampling, and steam generator level instrumentation are also planned to be severed from the steam generators while fuel is in the reactor vessel. Prior to severance, an appropriate support configuration will be established that ensures no adverse effects, including gravity missiles and sway interaction concerns, will result to any safety-related components. Potential loads considered include deadweight and seismic effects. This evaluation is documented in DC 90-13-1, Reference 6.12.22.

Following reinstallation of the removed sections of piping, cold gap/cold position measurements for components included in the critical commodities list (DC 90-13-1, Appendix 4-27, Critical Commodities List for Cold Gap Measurements), will be compared to the design values to verify the reinstalled configuration.

Prior to the start of primary cutting operations, the steam generator will be isolated with the loop stop valves. Prior to cutting of the RCS and secondary side piping, piping will be temporarily supported and affected variable springs and constant spring supports in the vicinity of the cut will be pinned (blocked) in the cold position to restrain vertical deflection at these supports, in order to allow for piping removal. Feedwater and ma n steam piping will be severed with the secondary side drained down to just below the transition cone cut l ation to allow for shielding. In this configuration, the RHR system remains available to remove decay heat. The acceptability of operation with the loop stop valves isolated and the means available for backup cooling should RHR be lost in this configuration is documented in Technical Report No. 885, Rev. 1.

RCS decontamination activities will not be performed until all fuel has been removed from the reactor vessel.

III. The work associated with the removal and reinstallation of SG level instrument tubing, piping, condensate pots and loose parts monitors will be performed in accordance with the existing technical specification requirements. Isolation of the instrument tubing and condensate pots will be achieved by tagging and isolating the level transmitters and then closing the root valves. Work done on the piping upstream of the root valves (between the SG nozzles and the valves) will be done after the secondary side of the SGs is drained. All instrumentation affected by the new design of the steam generator will be rescaled/recalibrated, as required, prior to operation.

IV. Replacement of the 1 and 2 inch carbon steel blowdown lines connected to each steam generator with new 1 and 2 1/2 inch chrome-moly lines, respectively, will be performed with the blowdown system not in service. The new installation has been seismically analyzed.

V. The removal and replacement of the steam generator insulation will be performed prior to completion of the steam generator outage. Instrumentation tubing which could be damaged during insulation removal will be isolated. A setpoint change has been prepared and included as an appendix to DC 90-13-1 to revise the RWST transfer to recirculation setpoint as a result of the safety analysis discussed in Technical Report NE-883 (DC 90-13-1, Appendix [illegible]3, Safety Analysis and Evaluations Supporting North Anna 1 Operation Following Steam Generator Replacement).

VI. Installation of the temporarily installed rigging equipment will be performed with the plant defueled with the exception of the temporary jib crane which will be utilized to transfer loads (3 tons or less) during defueling operations. In addition to operation in accordance with the existing heavy loads procedure, the jib crane load path will be restricted by mechanical stops to ensure that a load carried by the jib crane cannot impact spent fuel or the refueling canal and satisfy the requirements of NUREG-0612. All rigging required during cold shutdown or defueling will be conducted in accordance with the existing station heavy loads procedure. During the movement of the steam domes and steam generator lower assemblies, the reactor will be defueled and all fuel stored in the spent fuel pool. To preclude any possible adverse effects on stored spent fuel or systems shared with Unit 2 as

DC 90-13-1, STEAM GENERATOR REPLACEMENT / NORTH ANNA / UNIT 1

7. (Continued)

a result of a postulated steam generator drop, all component cooling water, service water, fuel pit cooling and refueling purification system, and instrument air lines to/from the containment will be isolated during rigging operations for the steam domes and lower assemblies. No adverse impacts on stored spent fuel or spent fuel cooling would result from a postulated heavy load drop inside the containment. Therefore, the heavy loads procedural requirements do not apply. The radiological consequences of a postulated drop of an old steam generator lower assembly inside the containment, adjacent to the equipment hatch during transfer from the equipment hatch platform to the transporter, or within the protected area have been evaluated and determined to be less than the limiting case events of the same type of accidents (i.e., the waste gas decay and volume control tank ruptures) currently evaluated in the UFSAR. The polar crane has been qualified for the maximum load to be lifted.

VII. Prior to removal of the old lower assemblies from the containment, the lower assemblies will be drained, all openings sealed, and a coating of encapsulant applied. The special requirements identified in this safety analysis associated with the transport of the steam generator lower assembly and the haul route load test are that the maximum speed of the transporter will be 5 mph and the transporter bed will be level within 5 degrees during movement. In addition, the transporter must remain within the analyzed route and take all precautions identified in DC 90-13-1, Appendix 4-18, Report for the Haul Route Inspection and Evaluation. Furthermore, the maximum height of the lower assembly centerline during transport shall not exceed 12'-3". Maintaining this height limitation ensures that a drop of the lower assembly off the transporter or during offload at the old steam generator storage facility will not cause a breech in the old lower assembly vessel integrity nor have an adverse impact on buried or adjacent utilities within the haul route boundary. Furthermore, the steam generator transition cone cover plate T-section must be installed prior to movement of the lower assembly outside the protected area.

VIII. Modifications associated with temporary utilities will be conducted with the plant in cold shutdown mode, refueling or defueled. For the temporary attachment to the purge system, the charcoal filters will be protected from smoke and dust damage since the filters will not be in line with the flow stream during purging of the containment. Temporary modification of the existing purge system will not be performed while fuel assemblies are located within containment. No modifications to the logic associated with the purge system is performed. Receipt of an isolation signal will result in purge system isolation. With fuel in the reactor vessel, while loading temporary equipment into containment, any items that penetrate the equipment hatch must be installed such that the hatch can be closed within 4 hours in accordance with station procedures. During core alterations or movement of irradiated fuel assemblies within containment, the equipment hatch will remain closed. Additional temporary modifications covered by this safety evaluation do not require any special plant conditions beyond those normally applicable to cold shutdown or refueling.

In addition to the above requirements, the following additional limitations are assumed in this safety evaluation:

- o While breaching security barriers to support steam generator replacement, a security watch will be posted in accordance with existing station requirements.
- o The steam generator replacement will be performed in accordance with the requirements of ASME Code, Section XI, 1983 Edition and Summer 1983 Addenda.
- o Coincident with equipment hatch removal, containment purge system operation will be in accordance with Operating Procedure 21.2 so as to minimize any air leakage out of the containment. In addition to the vent stack monitors and the containment area radiation monitors, a continuous air monitor will be in use adjacent to the equipment hatch and periodic air sampling will be performed. A temporary cover will also be available to isolate the hatch opening in case of loss of the containment purge exhaust system. For these reasons, there is a negligible potential for any unmonitored leakage out of the equipment hatch.

9/12/96

DC 90-13-1, STEAM GENERATOR REPLACEMENT / NORTH ANNA / UNIT 1

18. Summarize from Part D, Unreviewed Safety Question Determination, the major issues considered; state the reason the change, test, or experiment should be allowed; and state why an unreviewed safety question does or does not exist (a simple conclusion statement is insufficient).

I. The major issues considered for steam generator vessel [illegible] were the following:

- o Could the replacement of the steam generator lower assemblies significantly affect the performance of the steam generators or the reactor coolant system?
- o Could the steam generator replacement affect any of the results of the design basis accident analyses as discussed in the UFSAR?
- o Will the steam generator vessel repair activity affect the steam generator's integrity or lead to a significant increase in radiation dose?
- o Will the steam line flow restrictor affect the performance of the main steam system?

Upon evaluation of these issues, it was concluded that the activities associated with steam generator vessel repair and covered by this safety evaluation can be conducted without undue risk to the health and safety of the public and that this design change does not create an unreviewed safety question as defined by 10 CFR 50.59. These conclusions are based upon the following major points:

- o Although certain design improvements have been made, the design performance of the repaired steam generators meet or exceeds the design performance of the original generators. The repair will result in little change to the original operating parameters and these changes have no adverse effect on the capability of the steam generators to perform their intended function.
- o As described in Technical Report NE-883, Revision 1, results of a transient study have demonstrated that there is an insignificant difference in transient behavior between the Model 51 and 51F designs under postulated accident conditions. In addition, a reanalysis of LOCA mass and energy release and containment response has confirmed the existence of acceptable analysis margins. For each of the accident analyses, it was concluded that all acceptance criteria would continue to be met for operation following steam generator repair and that the UFSAR conclusions would remain valid.
- o The replacement steam generator lower assemblies are nearly identical to the existing lower assemblies such that the reactor coolant system, main steam system, feedwater system, and miscellaneous secondary piping are not adversely affected by the repair. The replacement enhances performance by providing additional tube bundle access, minimizing the potential for secondary side corrosion, and facilitating maintenance and inservice inspections.
- o The steam generator cutting and welding involved in the replacement of the lower assemblies will be in accordance with ASME Code requirements. Post-replacement inspections will also follow applicable Code requirements.
- o The existing steam generator upper supports will be replaced with supports of equivalent design so that design basis loading analyses will not be affected. The upper restraints will not be removed until all fuel has been offloaded.
- o The SG lower assembly was designed and fabricated to ASME Section III 1986 edition, Summer addenda, with reconciliation to the original design code. Therefore, it meets or exceeds the original design.

DC 90-13-1, STEAM GENERATOR REPLACEMENT / NORTH ANNA / UNIT 1

Part A - Resolution Summary Report (Continued)

18. (Continued)

- o A main steam flow restrictor will be installed in the steam nozzle. Following installation, the flow restrictor welds will be subjected to the code required NDE to verify the acceptability of the welds. When installed, the flow restrictor will reduce the pressure drop across the steam generator internal components during a postulated steam line break. The device also reduces the rate of energy released to the containment for this postulated accident. The main steam system flow rates and pressures during normal operation will not be affected. The steam blowdown from a faulted steam generator will not exceed design basis values. The differential pressures experienced by steam generator internal components during blowdown will be less than design basis values.

- o As documented in approved calculations, all applicable design basis seismic stress and support analyses have been evaluated/reperformed, as required, to verify the adequacy of the as installed condition following steam generator vessel repair.

II. The major issues considered for piping removal and replacement are as follows:

- o Will the removal/replacement of piping from the steam generators affect the performance of the steam generators, the reactor coolant system, the main steam system, the feedwater system, the chemical feed system, the wet-layup system, and sample system piping?

- o Will the steam generator piping removal and reinstallation affect any of the results of the design basis accident analyses as discussed in the UFSAR?

- o Will the cutting of the piping and subsequent rewelding adversely affect the steam generator or reactor coolant system integrity or lead to a significant increase in radiation doses?

Upon evaluation of the issues associated with piping removal and replacement, it is concluded that the activities covered by this safety evaluation can be conducted without undue risk to the health and safety of the public and that this design change does not create an unreviewed safety question as defined by 10 CFR 50.59. These conclusions are based on the following major points:

- o All the piping and pipe supports will be reinstalled to their original configuration so the reactor coolant system, main steam system, feedwater system, chemical feed system, wet-layup system, and sample system operation are not affected. As described in approved calculations, all applicable design basis seismic stress and support analyses have been evaluated/performed, as applicable, to verify the capability of the repaired systems to perform their intended functions. Piping removal and replacement will be in accordance with the Special Processes Manual and ASME Section XI Code requirements for piping repairs, welding, and nondestructive examination of pressure retaining components. Hydrostatic pressure testing will be performed in accordance with the ASME Code requirements. As a result, the as installed condition following piping activities associated with steam generator repair will meet or exceed the original design basis.

- o Appropriate measures will be taken to prevent debris from entering the affected systems.

- o Piping material changes have been evaluated and determined to be an acceptable equivalent or better than existing material.

Part A - Resolution Summary Report (Continued)

18. (Continued)

- o The design basis accident analyses are not affected due to the removal/replacement of piping. Prior to cutting the piping, temporary pipe supports will be installed, as necessary, to support the remaining pipe sections once the piping is severed. Prior to cutting RCS piping, the steam generator will be isolated with the loop stop valves.

- o To facilitate reinstallation of the main steam piping, new pipe spool pieces will be available. If a pipe spool is added to aid in fit-up, an additional weld joint may be created. If so, surveillance of the weld joint will be added to the ISI Program as applicable. The new spool pieces will be procured to meet or exceed the design requirements of the original piping and will therefore have no impact on the existing seismic or stress analyses, nor will they create any additional pipe break locations. The additional weld joint created will have no impact on the existing seismic stress analyses.

- o Cutting of all piping will be done by controlled procedures. The reactor coolant loops A, B, and C will have temporary shielding installed on the hot leg, crossover leg, and bypass line prior to the RCS piping cuts. RC Loop B will be severed first, prior to the completion of fuel offload. This configuration will exist with either the hot or crossover leg severed, or with both lines severed. Cutting of the Loop A and C RCS piping will not be performed until all fuel has been removed from the reactor. The acceptability of these configurations is documented in DC 90-13-1, Reference 6.12.21. The acceptability of the RC loops A, B, and C piping severed from the steam generators after fuel is removed from the reactor is also documented in DC 90-13-1, Reference 6.12.21.

  Secondary plant piping systems including main steam, feedwater, feedwater drain, wet layup, steam generator blowdown and shell drain, chemical feed, sampling, and steam generator level instrumentation are also planned to be severed from the steam generators while fuel is in the reactor vessel. Prior to severance, an appropriate support configuration will be established that ensures no adverse effects, including gravity missiles and sway interaction concerns, will result to any safety-related components. Potential loads considered include deadweight and seismic effects. This evaluation is documented in DC 90-13-1, Reference 6.12.22.

  In this configuration, the RHR system remains available to remove decay heat. The acceptability of operation with the loop stop valves isolated and the means available for backup cooling should RHR be lost in this configuration is documented in Technical Report No. 865, Rev. 1. Appropriate means will be employed to prevent debris from entering the piping during the cutting process and loop inspections for debris will be performed following completion of the cuts. A machine cutting method will be used and the installation of debris dams will not be necessary. Debris dams will be installed, if required, during any final weld end preparation. The use of a milling machine minimizes the potential for airborne releases. In addition, decontamination of the reactor coolant piping in the vicinity of the cut locations will be performed to reduce personnel exposure. This decontamination will be performed after fuel has been offloaded in accordance with approved procedures to minimize the base metal removal and to ensure that the piping integrity is not compromised.

III. The major issues considered for removal and reinstallation of the instrumentation associated with steam generator repair are as follows:

- o Whether the reinstalled tubing and condensate pots will function in the same manner as the previously existing piping, instrument tubing, and condensate pots?

18. (Continued)

- Whether the reinstalled piping, instrument tubing, and condensate pots will be as reliable as the previously existing piping, instrument tubing, and condensate pots?

- Whether the reinstalled loose parts monitors will be as reliable as the existing installation?

Upon evaluation of these issues, it is concluded that the activities covered by this safety evaluation can be conducted without undue risk to the health and safety of the public and that this design change does not create an unreviewed safety question as defined by 10 CFR 50.59. These conclusions are based on the following major points:

- Operational and seismic functionality of the level instrumentation will not be affected since the new configuration will be identical to the original configuration and all replacement components will meet or exceed the existing design requirements.

- Relocation of the loose parts monitors will be in accordance with approved procedures. The new loose parts monitors location does not effect the function of the monitors.

- Following modification, all affected instrumentation will be subjected to appropriate testing to include functional checkout, channel calibration and rescaling as appropriate.

- Reliability will not be affected since instrumentation, piping and tubing, including the condensate pots, and instrument root valves, will be reinstalled in accordance with the existing installation specifications with material upgrade (i.e., stainless steel versus the original carbon steel). Following installation, all instrumentation will be recalibrated. In addition, the reinstalled tubing and condensate pots will be inspected and tested per applicable code requirements following installation.

- Removal of the optical templating bracket will have not adverse effect on the steam generator cubicle wall.

IV. The major issues considered for removal and replacement of the blowdown piping with chrome-moly piping is as follows:

- Will the removal/replacement of the blowdown system piping significantly affect the performance or reliability of the steam generators or the blowdown system?

- Will the steam generator blowdown piping modifications affect any of the design basis accident analyses as discussed in the UFSAR?

Upon evaluation of these issues, it is concluded that the activities associated with the blowdown system modifications as addressed in this safety evaluation can be conducted without undue risk to the health and safety of the public and that no unreviewed safety questions, as defined by 10 CFR 50.59, exist. These conclusions are based on the following major points:

- The modification to the blowdown system consists of replacing the existing carbon steel piping with chrome-moly piping. The replaced piping enhances the capability of the blowdown system with respect to erosion/corrosion concerns and makes the system capable of handling higher blowdown rates and velocities than the original design. However, no increase in blowdown flowrates is required as a result of the steam generator lower assembly replacement. The modified blowdown system has been seismically qualified to applicable code requirements. All piping will be reinstalled in accordance with installation specification NAS-

18. (Continued)

1009 to ensure the modified system satisfies the design basis. This modification maintains the safety-related, seismic qualification of the blowdown system.

- The steam generator blowdown system modification increases the reliability of the system and has no effect on the operation or overall design function of the system. Original design requirements have been met or exceeded by the modification. Thus, this modification has no adverse impact on the existing accident analyses presented in the UFSAR.

V. The major issues considered for the removal and replacement of the existing steam generator insulation were the following:

- Will there be any adverse thermal effects of the new insulation on the performance of the steam generators or on containment temperature?

- Will the new insulation be a significant source of post-accident debris causing an adverse impact on either the functioning of the containment sump or the emergency core cooling system performance (IEN 90-07)?

- Will the new blanket insulation have an adverse effect on safety related equipment required to mitigate a design basis accident?

Upon evaluation of these issues, it was concluded that the activities associated with the removal and replacement of thermal insulation as evaluated in this safety evaluation can be conducted without undue risk to the health and safety of the public and that this design change does not create an unreviewed safety question as defined by 10 CFR 50.59. These conclusions are based upon the following major points:

- The replacement insulation has been procured to meet or exceed the insulating properties of the existing insulation. Thus, there is no adverse effect on the performance of the steam generators or on containment temperature.

- The effect of insulation has been evaluated with regard to core power (heat balance) calculations. The net change due to improved insulation on the replacement steam generators is negligible.

- The replacement insulation has been seismically designed to ensure that the insulation remains in place in the event of a seismic occurrence. In addition, the insulation has been evaluated in accordance with the recommendations of Regulatory Guide 1.82. As a result of the evaluation, the RWST recirculation switchover setpoint has been changed (see Technical Report NE-883).

- Removal and re-installation of insulation may be performed during defueling/refueling and will be performed in accordance with the station heavy loads procedure 0-MCM-1303-01. Instrumentation which could be damaged during insulation removal shall be isolated at the instrument root valve prior to removal of the insulation.

- As documented in DCP 90-13-1, Appendix 4-28, Steam Generator Insulation Debris Analyses Letter Report, the impact of the replacement fiberglass blanket insulation with respect to the NPSH requirements for the low head safety injection pumps and the inside and outside containment recirculation spray pumps as well as the potential long term blockage of the sump suction screens has been evaluated and determined to be acceptable.

DC 90-13-1, STEAM GENERATOR REPLACEMENT / NORTH ANNA / UNIT 1

Part A - Resolution Summary Report (Continued)

18. (Continued)

VI. To support repair of the steam generators, numerous lifting and handling activities associated with heavy loads are required. The issues considered in this safety evaluation include:

- o Could a load drop event during the lifting and handling activities associated with steam generator repair result in unacceptable consequences?

Upon evaluation of this issue, it was concluded that the activities associated with heavy load handling which are addressed in this safety evaluation can be conducted without undue risk to the health and safety of the public and that the activities do not create an unreviewed safety question as defined by 10 CFR 50.59. These conclusions rest on the following major points:

- o The polar crane has been qualified for special over rated-load lifts up to 280 tons in accordance with ASME B30.2, Section 2-3.2.1.1 (see DC 90-13-1, Appendix 4-31, Special Overrated-Load Lift Qualification, Reactor Containment Polar Crane Bridge Structure). This capacity is adequate to handle the original/new steam generator lower assemblies or the steam domes, along with the associated rigging equipment. All handling equipment associated with the steam generator removal has been designed in accordance with accepted industry practice (AISC Manual of Steel Construction and ANSI/ASME N45.2.15)to provide reasonable assurance that the equipment will function as intended.

- o The reactor vessel is devoid of fuel for the majority of the heavy load movements associated with steam generator replacement. For those activities which require lifting and handling of heavy loads within containment during defueling/refueling operations, such as handling of insulation, removal of cut piping sections and staging of material, these activities will be performed in accordance with the station heavy loads procedure. Lifting and handling activities associated with the temporary jib crane will be performed prior to complete reactor vessel defueling. All movements of loads utilizing the temporary jib crane will be in accordance with the existing plant heavy loads program. In addition, mechanical stops have been provided on the temporary jib crane to establish a load path boundary which ensures that failure of the crane will not impact the defueling operation or spent fuel. Safe load paths, load handling procedures, and properly trained crane operators will be used for the heavy lifts associated with the rigging operations.

- o The runway beams, the reactor cavity cover, the upper restraint temporary support, the steam dome laydown area and the auxiliary/jib crane support have been evaluated for their effect on permanent plant structures and components, are within design limits, and have no adverse effect on existing structures.

- o Any load drop during rigging operations inside containment will not affect containment integrity or the ability to contain a radioactive release due to a dropped load. Movement of loads with the temporary jib crane will be in accordance with existing plant heavy loads procedure in previously evaluated and approved safe load path boundaries to ensure that existing heavy load analyses bound the jib crane load handling activities. In addition, the jib crane will be provided with mechanical stops to limit the boom travel so that the refueling canal or spent fuel cannot be impacted. Lifting and handling activities performed during defueling will be accomplished in accordance with the existing station heavy loads procedure to ensure safe load handling with the plant in this configuration. To preclude any possible adverse effects on stored spent fuel or systems shared with Unit 2 as a result of a postulated steam generator drop, all component cooling water, service water, fuel pit cooling and refueling purification system, and instrument air lines to/from the containment

18. (Continued)

will be isolated during rigging operations for the steam domes and lower assemblies. Thus, no adverse impacts on stored spent fuel or spent fuel cooling would result from a postulated heavy load drop inside the containment.

Coincident with equipment hatch removal, containment purge system alignment and operation will be in accordance with Operating Procedure 21.2 so as to minimize any air leakage out of the containment. In addition to the vent stack monitors and the containment area radiation monitors, a continuous air monitor will be in use adjacent to the equipment hatch and periodic air sampling will be performed. A temporary cover will also be available to isolate the hatch opening in case of loss of the containment purge exhaust system. For these reasons, there is a negligible potential for any unmonitored leakage out of the equipment hatch. This nothwithstanding, the radiological consequences of a postulated drop of an old steam generator lower assembly inside the containment, during movement from the equipment hatch platform to the transporter, or for a drop within the protected area, have been evaluated against the radiological consequences of other events in the same class of postulated accidents, such as the rupture of a waste gas or volume control tank. Based on this comparison, it is concluded that the drop of a steam generator lower assembly and the potential radiological releases as a result of this drop are within applicable regulatory limits and are bounded by the consequences associated with existing accidents previously evaluated in the UFSAR (Reference DC 90-13-1 Safety Evaluation Question 64B). A postulated drop of a steam generator lower assembly adjacent to the equipment hatch could result in damage to the equipment hatch or locally to the containment structure, but significant structural damage to the containment would not occur. Moreover, containment leaktight integrity is not required at this time. Further, no safety-related equipment could be impacted by a postulated drop at this location. (Reference DC 90-13-1, Appendix 4-18).

VII. During movement of the steam generators lower assemblies on the transporter, the following issues have been addressed:

- o Could the surcharge loads imposed by the loaded transporter or prime mover on buried utilities damage equipment or components within the haul route?
- o Could the drop of the old steam generator lower assembly from the transporter result in unacceptable consequences?
- o Could safety related structures along the haul route be adversely impacted by transport of the old steam generator lower assemblies?

Upon evaluation of these issues, it was concluded that the activities associated with transport of the old steam generators to the old steam generator storage facility as addressed in this safety evaluation can be conducted without undue risk to the health and safety of the public and that the activities do not create an unreviewed safety question as defined by 10 CFR 50.59. These conclusions rest on the following major points:

- o The steam generator haul route has been evaluated and buried utilities within the haul route which can be subjected to the surcharge loadings associated with the loaded transporter, prime mover or the transporter test weight have been identified and protected, if required, to ensure that the equipment or components are unaffected by the movement.
- o Cover plates have been designed and will be installed on the old steam generator lower assembly prior to removal of the old steam generator from the protected area. These cover plates have been designed to remain intact in the inadvertent drop of the steam generator lower assembly from the transporter or during

18. (Continued)

offload of the lower assembly at the old steam generator storage facility. Thus, in the event of a drop of the steam generator from the transporter, no adverse consequences result since the old steam generator integrity is not breached.

o Structures and components buried and adjacent to the haul route have been evaluated for a postulated drop of a lower assembly or the transporter test load (Reference DC 90-13-1, Appendix 4-18, Report for the Haul Route Inspection and Evaluation) and determined to be acceptable. Furthermore, to ensure control and safe transport of the lower assemblies and the load test, the speed of the transporter will be limited to 5 miles per hour, the transporter bed will be level within 5 degrees and the centerline of the steam generator will be limited to a maximum of 12'-3" above grade to ensure the integrity of the steam generator in the unlikely event of a drop from the transporter. Thus, safety related structures buried or adjacent to the haul route will be unaffected in the unlikely event of a generator or transporter load test weight drop off the transporter. No adverse impacts on Unit 2 operation would result.

VIII. The major issues associated with installation of temporary modification associated with the steam generator repair are as follows:

o Could installation of temporary modifications adversely affect plant operations?

o Are adequate separation requirements provided between safety related and non-safety related power supplies?

o Could permanent systems be adversely affected by installation of the temporary modifications associated with the steam generator replacement?

Upon evaluation of the temporary modifications associated with steam generator replacement, it is concluded that the activities covered by this safety evaluation can be conducted without undue risk to the health and safety of the public or the plant and that this activity will not result in an unreviewed safety question as defined in 10 CFR 50.59. These conclusions were based on the following:

o The purge exhaust and supply system will be operated in accordance with approved procedures and within its design flow rates. No modifications to the logic associated with containment purge operation or with containment isolation are performed. Thus, the capability of the purge system will not be affected as a result of this temporary modification. In addition, this temporary modification affects only the purge system ductwork within Unit 1 containment and will have no adverse affect on the capability of the purge system to support Unit 2 operation, if required.

o No modifications are being made to the plant power distribution system. Therefore, there is no impact on the physical separation between non-Class 1E and Class 1E power supplies. Temporary power will be from RCP 1B power feeds during implementation which is a non-Class 1E supply.

o Movement of loads with the temporary jib crane will be in accordance with the existing heavy loads procedure in previously reviewed and approved safe load path boundaries. In addition, the jib crane will be provided with mechanical stops to limit the boom travel so that the refueling canal or spent fuel cannot be impacted.

o Loads on permanent plant structures as a result of installation of the reactor cavity cover, the auxiliary/jib crane, and the temporary support for the steam generator upper lateral support have been evaluated and determined to be acceptable.

DC 90-13-1, STEAM GENERATOR REPLACEMENT / NORTH ANNA / UNIT 1

Part A - Resolution Summary Report (Continued)

18. (Continued)

- o All temporary modifications will be removed and the affected systems returned to normal operating status prior to completion of the steam generator repair outage. This includes removal of the reactor cavity cover and auxiliary/jib cranes.

DC 90-13-1, STEAM GENERATOR REPLACEMENT / NORTH ANNA / UNIT 1

rt B - Applicable References

19. Identify applicable Updated Final Safety Analysis Report (UFSAR) sections.

See Page 3A for response.

20. Identify applicable Technical Specifications sections.

See Page 3B for response.

21. Identify any other references used in this review.

See Page 3C for response.

Part C - Items Considered By This Safety Evaluation

Insert Supplemental Pages if Additional Space if Needed For Explanations. Use "Safety Evaluation, Supplemental Page" Form No. 730928. Double Asterisk (**) Items Answered "Yes" Require Design Authority Approval (s' 72-75 Page 12 of 12).

22. Will the operation of any system or component as described in the Safety Analysis Report be altered? This includes abandonment of equipment or extended periods of equipment out of service. Explain. [ ] Yes ** [X] No

See Page 3C for response.

23. Will the activity alter the performance characteristics of any safety related system or component? Action statements, jumpers, and temporary modifications should be reviewed. Explain. [ ] Yes ** [X] No

See Page 3E for response.

24. Will the ability of operators to control or monitor the plant be reduced in any way? Explain. [ ] Yes [X] No

See Page 3F for response.

25. If a jumper is involved, are testing requirements as stated on the jumper adequate to ensure operability after installation as well as after removal? Explain. [ ] Yes [X] No

See Page 3F for response.

DC 90-13-1, STEAM GENERATOR REPLACEMENT / NORTH ANNA / UNIT 1

rt B - Applicable References (Continued)

19. Identify applicable Updated Final Safety Analysis Report (UFSAR) sections.

I. UFSAR Section 3.7 - Seismic Design
UFSAR Section 5.5.2 - Steam Generator
UFSAR Section 5.5.4 - Residual Heat Removal System
UFSAR Section 5.6 - Reactor Coolant System Instrumentation Application
UFSAR Section 10.3 - Main Steam System
UFSAR Section 15.1 - Normal Operation and Operational Transients
UFSAR Section 15.2 - Faults of Moderate Frequency
UFSAR Section 15.3 - Infrequent Faults
UFSAR Section 15.4 - Limiting Faults
UFSAR Tables 3.7-11 and 3.7-12 - Representative Listing of Seismic Design Margins
UFSAR Tables 5.2-4 through 5.2-13 - Steam Generator Primary-Secondary Boundary Components
UFSAR Table 5.2-17 - Faulted Condition Loads for the Reactor Coolant Pump Foot
UFSAR Table 5.5-18 - Maximum Steam-Generator and Reactor Coolant Pump Foot Loads
UFSAR Table 5.5-19 - Maximum Load, Supports
UFSAR Table 5.5-20 - Maximum Load, Snubbers and Struts
UFSAR Table 5.2-22 - Pressure Boundary Component Material
UFSAR Table 5.5-3 - Steam Generator Design Data
UFSAR Appendix 6A - Mass Energy Release Rates During Reflood

II. UFSAR Sections 5.1.1.4, 5.2.3.1.2, 5.5.3 - RCS Piping
UFSAR Sections 10.3, 10.3.2 - Main Steam System
UFSAR Sections 10.4 - Other Features of Steam and Power Conversion
UFSAR Figure 10.4-7 - Chemical Feed System
UFSAR Figures 10.4-8, 10.4-11 through 10.4-14 - Feedwater Piping

III. UFSAR Section 3 - Design Criteria
UFSAR Section 5.5.2 - Steam Generator
UFSAR Section 7.2.1.1.5 - Steam Generator Trips
UFSAR Section 7.2.2.3.5 - Steam Generator Water Level and Feedwater Flow
UFSAR Section 7.7.1.7 - Steam Generator Water-Level Control

IV. UFSAR Sections 10.4.6.2, 10.4.6.3 and 10.4.6.5
UFSAR Figure 10.4-17
UFSAR Tables 3.2-1 and 5.5-4

V. UFSAR Sections 5.5.2 - Steam Generator
UFSAR Section 6.2 - Containment Systems
UFSAR Section 6.3 - Emergency Core Cooling Systems
UFSAR Section 15.3 - Accidents Analysis, Condition III - Infrequent Faults
UFSAR Section 15.4 - Accident Analysis, Condition IV - Limiting Faults

VI. UFSAR Section 9.1 - Fuel Handling and Storage
UFSAR Section 6.2 - Containment Systems

VII. UFSAR Section 3.5.4, Design For Missiles Postulated Outside of Reactor Containment.

DC 90-13-1, STEAM GENERATOR REPLACEMENT / NORTH ANNA / UNIT 1

't 8 - Applicable References (Continued)

19. (Continued)

VIII. UFSAR Section 1.2.10 -- Auxiliary Systems
UFSAR Section 6.2.1.3.2.1 -- Refueling Cavity
UFSAR Section 8.1.2 -- Onsite Electric System
UFSAR Section 8.3.1 -- Alternating Current Power System
UFSAR Section 9.1.4.3.1 -- Refueling Cavity
UFSAR Section 9.4.9 -- Containment Structures
UFSAR Chapter 15 -- Accident Analyses

20. Identify applicable Technical Specifications sections.

I. Tech. Spec. Section 2.2 - Limiting Safety System Settings
Tech. Spec. Section 3/4.3.1 - Reactor Trip System Instrumentation
Tech. Spec. Section 3/4.3.2 - Engineered Safety Feature Actuation System Instrumentation
Tech. Spec. Section 3/4.4.5 - Steam Generator Tube Inspections
Tech. Spec. Section 4.0.5 - Surveillance Requirements
Tech. Spec. Section 5.4.2 - Reactor Coolant System Volume

II. Tech. Spec. Section 3/4.4 - Reactor Coolant System

Tech. Spec. Section 3/4.3.1 - Reactor Trip System Instrumentation
Tech. Spec. Section 3/4.3.2 - Engineered Safety Feature Actuation System Instrumentation
Tech. Spec. Section 3/4.3.3 - Monitoring Instrumentation
Tech. Spec. Section 3/4.4.1.3 - Reactor Coolant System - Shutdown
Tech. Spec. Section 3/4.9.8.1 - Refueling Operations -- Residual Heat Removal (RHR) and Coolant Circulation Normal Water Levels
Tech. Spec. Section 3/4.9.8.2 - Refueling Operations -- Residual Heat Removal (RHR) and Coolant Circulation Low Water Levels

IV. Tech. Spec. Section 3/4.4.5 - Steam Generator Tube Inspections

V. Tech. Spec. Section 3/4.4.5 - Steam Generator Tube Inspections
Tech. Spec. Section 3/4.4.6 - Containment Systems
Tech. Spec. Section 3/4.5.2 - ECCS Subsystems
Tech. Spec. Section 3/4.5.3 - ECCS Subsystems
Tech. Spec. Section 3/4.6.2 - Depressurization and Cooling Systems
Tech. Spec. Section 3/4.7.2 - Steam Generator Pressure/Temperature Limitations

VI. Tech. Spec. Section 3/4.6.1 - Containment Integrity
Tech. Spec. Section 3/4.9.11 - Spent Fuel Pool Water Level

VII. None

VIII. Tech. Spec. Section 3/4.8.2.2 -- A.C. Distribution (Shutdown)
Tech. Spec. Section 3/4.9.4 -- Containment Building Penetrations
Tech. Spec. Section 3/4.9.9 -- Containment Purge and Exhaust Isolation Valves

DC 90-13-1, STEAM GENERATOR REPLACEMENT / NORTH ANNA / UNIT 1

rt B - Applicable References (Continued)

21. Identify any other references used in this review.

I. Westinghouse letter VP-297 to Virginia Power, dated August 7, 1991.
Technical Report NE-883, Safety Analyses and Evaluations Supporting North Anna 1 Operation Following Steam Generator Replacement

II. Calculation 02072.13-NP(B)-008-XE, "Seismic Evaluation of the Unit 1 Reactor Coolant Loops and Reactor Coolant Loop Components After Cutting the Hot Leg and Crossover Legs at the Steam Generator Nozzles"
Calculation 02072.13-NP(B)-009-XE, "Evaluation of the Unit 1 Secondary Side Reactor Coolant Loop Branch Lines After Their Severance from the Steam Generator"

III. None

IV. NRC Information Notice No. 91-18
NRC Generic Letter 89-08
NRC Bulletin 87-01

V. NRC Information Notice No. 90-07, January 30, 1990.
Stone and Webster Letter NAS-20,879, dated June 29, 1992.

VI. NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants"
Stone and Webster Letter Report - Special Overrated-Load Lift Qualification, Reactor Containment Polar Crane Bridge Structure

Station Drawing 13075-EY-8A-1
Westinghouse drawing 6142E91.
Stone and Webster Report for the Haul Route Inspection and Evaluation

VIII. None

For a complete list of references, see DC 90-13-1, Section 6.0, References.

Part C - Items Considered By This Safety Evaluation (Continued)

22. Will the operation of any system or component as described in the Safety Analysis Report be altered? This includes abandonment of equipment or extended periods of equipment out of service. Explain.

I. The new SG has equivalent design parameters as the original SG and subsequently, there will be no change to its operation as described in the UFSAR. The new SGs will be more efficient and are expected to achieve design steam pressure at a lower operating average reactor coolant temperature. Therefore, the operational parameters predicted in the Westinghouse Thermal/Hydraulic Report (DC 90-13-1, Appendix 4-26) were calculated using the lower T_{avg}. These operational parameters will be incorporated into the Operating Procedures and the I&C Procedures and require a change to the UFSAR. The SG will not be taken out of service and the upper restraint will not be removed when required to be operational per the Technical Specifications.

22. (Continued)

II. The replacement piping will be functionally identical to the current RCS, main steam, feedwater, chemical feed, sampling and wet-layup piping. The operation of systems or components will not be altered. No equipment will be abandoned. However, these systems will be removed from service during unit outage to accomplish the piping removal/replacement in support of SGR. The reactor coolant loops A, B, and C will have temporary shielding installed on the hot leg, crossover leg, and bypass line prior to the RCS piping cuts. RC Loop B will be severed first, prior to the completion of fuel offload. This configuration will exist with either the hot or crossover leg severed, or with both lines severed. Cutting operations for the Loop A and C RCS piping will not be performed until all fuel has been offloaded. The acceptability of these configurations is documented in DC 90-13-1, Reference 6.12.21. The loop isolation valves shall be isolated prior to making any piping cuts. The acceptability of the RC loops A, B, and C piping severed from the steam generators after fuel is removed from the reactor is also documented in DC 90-13-1, Reference 6.12.21.

Secondary plant piping systems including main steam, feedwater, feedwater drain, wet layup, steam generator blowdown and shell drain, chemical feed, sampling, and steam generator level instrumentation are also planned to be severed from the steam generators while fuel is in the reactor vessel. Prior to severance, an appropriate support configuration will be established that ensures no adverse effects, including gravity missiles and sway interaction concerns, will result to any safety-related components. Potential loads considered include deadweight and seismic effects. This evaluation is documented in DC 90-13-1, Reference 6.12.22.

Decontamination of the RCS piping in the vicinity of the cut locations will be performed after fuel offload in accordance with approved procedures to minimize the base metal removal and to ensure that the piping integrity is not compromised.

III. The reinstallation of SG level instrumentation will result in instrument tubing and condensate pots being functionally identical to the original tubing and condensate pots. No equipment will be abandoned. Level instruments will be removed from service to accomplish the piping removal/replacement in support of SGR. Following relocation of the loose parts monitors, a channel check will be performed to verify the operability of the instrument.

IV. The blowdown system modification will require a revision to UFSAR to detail changes made to the steam generator blowdown piping system arrangement. Specifically, the change in pipe size and pipe class for the new 1 and 2 1/2 inch chrome-moly blowdown lines connected to each steam generator needs to be updated. Although the blowdown system modification replaces a portion of the blowdown system with larger diameter piping, the system will be operated at the existing blowdown flow rates. No equipment will be abandoned. The system will be removed from service during unit outage to accomplish the blowdown system modification.

V. As a result of insulation removal and replacement, the operation of systems and components, as described in the UFSAR, will not be altered. A debris analysis (DC 90-13-1, Appendix 4-28, Steam Generator Debris Analyses Letter Report) has been prepared to address the effect of the replacement insulation on the containment sump analysis. Calculations performed in support of this debris analysis indicate a reduction of NPSH margin for the inside and outside recirculation spray pumps. However, adequate NPSH margin is still available for the recirculation spray systems. The calculations have also determined that an increase in NPSH will result for the low head safety injection pumps, assuming the RWST LHSI switchover level setpoint change is implemented.

VI. During rigging activities, the operation of systems and components, as described in the Safety Analysis Report, will not be altered. No equipment will be abandoned. Lifting and handling activities associated with the jib crane will be conducted in accordance with the plant heavy loads procedure. In addition, the jib crane will be provided with mechanical stops to prevent loads from impacting the refueling canal or spent fuel.

DC 90-13-1, STEAM GENERATOR REPLACEMENT / NORTH ANNA / UNIT 1

rt C - Items Considered By This Safety Evaluation (Continued)

22. (Continued)

VII. The steam generator haul route activity will not alter the operation of any systems described in the UFSAR for either Unit 1 or Unit 2. No equipment required to be operational will be removed from service to perform this activity. This activity does not involve abandoning any equipment. Components within the haul route have been evaluated and have been determined to be unaffected or adequate protection will be provided to ensure no damage to buried or adjacent utilities will occur.

VIII. The temporary modifications made to the purge system and the temporary electrical power connections will not alter the operation of any systems as described in the UFSAR. No equipment required to be operational will be removed from service for extended periods of time to accomplish the installation of temporary services. All affected systems will be operational at the completion of the SGR outage. Loading to permanent structures associated with the reactor cavity cover and the auxiliary/jib cranes, runway beams, and laydown of the stem domes have been evaluated and determined to have no adverse impact on the existing structures.

23. Will the activity alter the performance characteristics of any safety related system or component? Action statements, jumpers, and temporary modifications should be reviewed. Explain.

I. Upon completion of the steam generator vessel repair activity, the steam generators will be functionally identical to the current steam generators. Performance characteristics of safety-related systems and components will not be altered by the repair of the steam generators.

II. All the piping will be replaced to its original configuration. The performance characteristics of safety-related systems and components will not be altered by the piping removal and replacement.

III. All instrumentation tubing and the condensate pots will be reinstalled to their original configurations. The performance characteristics of safety-related systems and components will not be altered by the piping, instrument tubing, and condensate pots removal and reinstallation. Recalibration/rescaling, as appropriate, will be performed following the instrumentation modification associated with steam generator level indication. Acceptability of the loose parts monitors following relocation will be established by performance of a channel check. Thus, the instrumentation will be verified to be operable following the modification.

IV. The steam generator blowdown system modification will not alter the system blowdown flow rate, therefore, the performance characteristics of safety-related systems and components will not be altered.

V. The new steam generator insulation will meet or exceed the performance characteristics of the old insulation. The impact of the replacement fiberglass blanket insulation with respect to the NPSH requirements for the low head safety injection pumps and the inside and outside containment recirculation spray pumps as well as the potential long term blockage of the sump suction screens has been evaluated and determined to be acceptable provided a setpoint change to the RWST LHSI switchover setpoint is performed. (DC 90-13-1, Appendix 4-28, Steam Generator Insulation Debris Analyses Letter Report).

VI. The rigging activities will not require the connection to or support from any active safety-related equipment or systems. Performance characteristics of safety-related systems, components and structures will not be altered by rigging operations.

VII. Us of the steam generator haul route will have no adverse impact on any active safety-related equipment or systems, therefore, performance characteristics of safety-related systems and components will not be altered.

DC 90-13-1, STEAM GENERATOR REPLACEMENT / NORTH ANNA / UNIT 1

rt C - Items Considered By This Safety Evaluation (Continued)

23. (Continued)

VIII. Temporary modification of the containment purge system will have no affect on the capability of the purge to perform its function. The modification is performed within containme.., and will not effect the capability to isolate containment or prevent purge operation in support of Unit 2, if required. Following completion of steam generator replacement activities but prior to reload, the purge system will be restored and tested in accordance with approved procedures to ensure that the purge system functions in accordance with the design basis. Other temporary modifications have been evaluated for their effect on permanent plant structures and the loads imposed by these modifications have been determined to be acceptable.

24. Will the ability of operators to control or monitor the plant be reduced in any way? Explain.

III. During the instrumentation removal and replacement activity, all applicable plant technical specification requirements will be followed. Upon completion of the modification, the instrumentation affected will be recalibrated/rescaled, and tested, as required to verify the instruments has been returned to their original state and to verify instrument operability.

VIII. The temporary services activity will not reduce the ability of the operators to control or monitor the plant in any way. The containment radiation monitoring system will remain functional to isolate the purge system upon identification of abnormal containment radiation levels.

None of the following activities involve any modifications to control or monitoring circuits, therefore, they do not impact the ability of the operators to control or monitor the plant:

I. Steam Generator Vessel Repair
II. Piping Removal and Replacement
IV. Blowdown System Modification
V. Insulation Removal and Replacement
VI. Rigging Activities
VII. Steam Generator Haul Route

25. If a jumper is involved, are testing requirements as stated on the jumper adequate to ensure operability after installation as well as after removal? Explain.

VIII. The temporary services activity will require that the RCP-1B feeder breaker 15B3 be temporarily modified to bypass pump motor interlocks. This modification will involve a jumper or temporary procedure, but will not disable any system or function required to be available during the modes of operation planned for the outage. The jumper, if required, will be removed and breaker checkouts will verify operability of the breakers prior to plant start-up. RCP breakers are non-safety related.

No jumpers are required for any of the following activities:

I. Steam Generator Vessel Repair
II. Piping Removal and Replacement
III. Instrumentation Removal and Replacement
IV. Blowdown System Modification
V. Insulation Removal and Replacement
VI. Rigging Activities
VII. Steam Generator Haul Route

DC 90-13-1, STEAM GENERATOR REPLACEMENT / NORTH ANNA / UNIT 1

rt C - Items Considered By This Safety Evaluation (Continued)

26. Could the proposed activity affect reactivity? Explain. The Reactor Engineer must approve the explanation for "Yes" answers. [Commitment 3.2.9] [X] Yes ** [ ] No

See Page 4A for response.

26A. Reactor Engineer Signature	26B. Date
P J McAndrew	9-21-92

27. Will the activity significantly increase the potential for personnel injury or equipment damage? [ ] Yes [X] No

See Page 4A for response.

28. Will the activity create or increase the levels of radiation or airborne radioactivity? [X] Yes [ ] No
Will that change result in a significant unreviewed environmental impact, a significant increase in occupational exposure, or a significant change in the dose to operators performing tasks outside the filtered air boundary during a design basis accident (GDC-19)? Explain. The Superintendent-Radiological Protection must approve the explanation for "Yes" answers. [ ] Yes [X] No

See Page 4B for response.

28A. Superintendent Radiological Protection Signature	28B. Date
[illegible] / [illegible] AHS	9-30-92 9/21/92

29. Could the activity change or decrease the effectiveness of the emergency plan? Explain. The Coordinator-Emergency Preparedness must approve the explanation for "Yes" answers. [ ] Yes [X] No

See Page 4C for response.

29A. Emergency Preparedness Coordinator Signature	29B. Date

30. Will the consequences of failure for this activity affect the ability of systems or components to perform safety functions? Describe the modes and consequences of failure considered during this evaluation. [ ] Yes [X] No

See Page 4D for response.

DC 90-13-1, STEAM GENERATOR REPLACEMENT / NORTH ANNA / UNIT 1

Part C - Items Considered By This Safety Evaluation (Continued)

26. Could the proposed activity affect reactivity? Explain.

I. As discussed in the reactivity management review conducted for DC 90-13-1, Appendix 4-13, the steam generator vessel repair will result in the operational characteristics of the repaired steam generators being essentially identical to those of the old steam generators. However, T_{avg} will be changed from 583 to 580.8 °F. The effects on reactivity due to the change in T_{avg} have been addressed in the Nuclear Core Design and Licensing checklist used as design input for reactivity related analyses for North Anna Unit 1 cycle 10 reload analysis. Reactivity effects as a result of an increase in RCS primary side volume as well as RCS and main steam flowrate have been addressed in Technical Report NE 883 and the necessary instrument scaling and setpoint changes have been provided in DC 90-13-1.

V. The new insulation has performance characteristics that meet or exceed those of the old insulation. Therefore, the impact of the new steam generator insulation material on the heat balance program has been evaluated. The new insulation material has essentially the same insulating capabilities as the existing material and therefore, has no measurable impact on the heat balance program and, consequently, reactivity.

None of the following activities affect reactivity:

II. Piping Removal and Replacement
III. Instrumentation Removal and Replacement
IV. Blowdown System Modification
VI. Rigging Activities
VII. Steam Generator Haul Route
VIII. Temporary Services

27. Will the activity significantly increase the potential for personnel injury or equipment damage?

During the implementation phase of steam generator repair, there is some increase in the potential for personal injury and equipment damage due to the size of the vessels and the amount of associated construction effort. The major risks are those associated with work on plant systems, rigging in and out of heavy components, and industrial safety. All activities associated with lifting and handling will utilize the existing heavy loads procedure or other approved project procedures to ensure that the rigging is performed in accordance with industry requirements and good rigging practices.

The risk associated with this large construction effort will be minimized by adherence to the requirements of station tagging procedures, generation of work procedures, use of mockups, rigging evaluations, and good work practices.

Upon completion of steam generator repair, the original plant configuration will be restored, therefore, there will be no increase in the potential for personal injury or equipment damage during subsequent plant operation.

All portions of the following activities will be accomplished in accordance with existing plant safety precautions:

I. Steam Generator Vessel Repair
II. Piping Removal and Replacement
III. Instrumentation Removal and Replacement
IV. Blowdown System Modification

DC 90-13-1, STEAM GENERATOR REPLACEMENT / NORTH ANNA / UNIT 1

V. Insulation Removal and Replacement
VI. Rigging Activities
VII. Steam Generator Haul Route
VIII. Temporary Services (Impact on permanent facilities only)

DC 90-13-1, STEAM GENERATOR REPLACEMENT / NORTH ANNA / UNIT 1

28. Will the activity create or increase the levels of radiation or airborne radioactivity? Will that change result in a significant unreviewed environmental impact, a significant increase in occupational exposure, or a significant change in the dose to operators performing tasks outside the filtered air boundary during a design basis accident (GDC-19)?

I. During the SGR, there will be increased occupational exposure to radiation. Various measures will be taken to reduce this exposure to ALARA.

The steam generator vessel repair will include a general decontamination of the containment building to ensure that personnel exposures are maintained ALARA. The reactor refueling cavity will be decontaminated to minimize airborne activity. The containment purge system will be operated during the repair and will be balanced so as to maintain a positive airflow into the containment.

The secondary side of the original steam generators will be kept full of water as long as possible prior to cutting and lifting operations to provide shielding from the radioactive material contained inside the primary tubes.

Temporary shielding of residual hot spots will be used if it will result in a significant exposure reduction after decontamination. Protective clothing will be required and respirators will be used as necessary.

Mock-up training and specialized tools will be used in a number of instances to minimize exposures. Automated cutting, milling, and welding machines will be used to the extent practicable to reduce doses in high radiation areas. These machines typically run on tracks attached to the pipes and allow personnel to stand back in a lower radiation area to observe the correct operation of the machines.

Following replacement of the steam generator lower assemblies, the occupational exposure associated with steam generator maintenance during outages is expected to decrease due to the improved design of the replacement assemblies with respect to tube degradation. This improved design reduces the steam generator maintenance and tube plugging required for the old assemblies.

II. The piping removal and replacement activity will result in minimal airborne releases from cutting the main steam, feedwater, RCS, and other miscellaneous piping. Use of a milling machine for making cuts will minimize the potential for airborne releases. Temporary ventilated enclosures with HEPA filters will be employed as needed during pipe cutting operations to minimize the spread of airborne contamination.

Decontamination of the reactor coolant system pipe ends will also be performed. An abrasive-blast decontamination of the internal surfaces of the reactor coolant system hot/crossover leg pipe ends will reduce overall exposure during the machining and welding processes.

In addition to the pipe decontamination, shielding of reactor coolant piping will be provided to reduce dose rates to workers involved in the cutting, pipe end preparation, and welding of the reactor coolant pipe ends.

III. All instrument tubing to be replaced is connected to the secondary side of the steam generator which contains little or no contamination. Due to the small size of the cuts made on the piping and instrument tubing, airborne radiation will not be increased as a result of the cutting of the instrument tubing.

V. The insulation being removed is encapsulated on all sides and will be handled so as not to contribute to airborne radioactivity.

DC 90-13-1, STEAM GENERATOR REPLACEMENT / NORTH ANNA / UNIT 1

art C - Items Considered By This Safety Evaluation (Continued)

28. (Continued)

VI. Prior to removal of the lower assemblies from containment, the SGLAs will be drained and cover plates will be installed on all lower assembly nozzle openings to reduce personnel exposure during rigging as well as to provide a sealed package. In addition, prior to removal from containment, the lower assemblies will be coated with an encapsulant to fix all loose surface contamination to the lower assemblies. Coincident with equipment hatch removal, containment purge system alignment and operation will be in accordance with Operating Procedure 21.2 so as to minimize any air leakage out of the containment. In addition to the vent stack monitors and the containment area radiation monitors, a continuous air monitor will be in use adjacent to the equipment hatch and periodic air sampling will be performed. A temporary cover will also be available to isolate the hatch opening in case of loss of the containment purge exhaust system. For these reasons, there is a negligible potential for any unmonitored leakage out of the equipment hatch.

VII. During transport of the lower assemblies to the old steam generator storage facility, cover plates will be installed to reduce personnel exposure. Loose surface contamination will be encapsulated to the lower assembly. Appropriate radiological controls during transport will be provided to ensure that all personnel maintain a safe distance from the transporter and that activities associated with the transport to the old steam generator storage facility are performed utilizing ALARA principles.

VIII. Following the temporary modifications to the purge system, the system will continue to perform its design function and isolate upon receipt of a containment radiation signal. These temporary modifications do not impact any radioactive system, material, or equipment, therefore, the temporary services will not create or increase levels of radiation or airborne radioactivity.

The following activity does not normally involve a radiation source or radioactive fluids; therefore, they do not create or increase levels of radiation or airborne radioactivity:

IV. Blowdown System Modification

29. Could the activity change or decrease the effectiveness of the emergency plan? Explain.

The following activities will have no impact upon the effectiveness of the emergency plan:

I. Steam Generator Vessel Repair
II. Piping Removal and Replacement
III. Instrumentation Removal and Replacement
IV. Blowdown System Modification
V. Insulation Removal and Replacement
VI. Rigging Activities
VII. Steam Generator Haul Route
VIII. Temporary Services (Impact on permanent facilities only)

30. Will the consequences of failure for this activity affect the ability of systems or components to perform safety functions? Describe the modes and consequences of failure considered during this evaluation.

I & II. Cutting of the secondary system piping and reactor coolant piping may occur prior to fuel offload. During the main steam and feedwater cuts, these systems are not required to be operable and will not be utilized to remove decay heat from the reactor core. Also prior to fuel offload, the B reactor coolant loop may be severed. Prior to severance of RCS piping, the loop stop valves will be isolated to maintain the reactor coolant system fluid boundary. Cutting operations for Loop A and C RCS piping and removal of the upper restraints will not occur until after defueling. In this configuration, the RHR system remains available to remove decay heat. The acceptability of operation with the loop stop valves isolated and the means available for backup cooling should RHR be lost in this configuration is documented in Technical Report No. 865, Rev. 1. Installation flaws will be detected and corrected in accordance with ASME Section XI and verified to meet design criteria and code requirements prior to plant operation. This includes NDE and hydrostatic testing. Critical SG parameters will be verified during performance testing. Technical Report NE-883 evaluates the performance of the replacement steam generators under postulated accident conditions. For each of the accident analyses reviewed, it was concluded that all acceptance criteria would continue to be met for operation following steam generator replacement and that the UFSAR conclusions for each event would remain valid.

III. The instrumentation will be reinstalled to meet or exceed the existing design requirements including all technical specification requirements applicable to the instruments being modified will be met. Prior to being placed in service, the installation will be verified to meet design criteria and code requirements. Calibration and rescaling will be performed on all instrumentation to verify the repair has had no adverse effect on instrument operation. During operation, the system is bound by existing UFSAR accident analysis. Therefore, failure of this activity during construction or operation does not affect the ability of systems or components to perform their safety function.

IV. During construction activities, the blowdown system performs no safety function. The system is being replaced with components that meet or exceed the existing functional requirements. Any flaws in the installation will be detected and corrected in accordance with ASME Section XI and verified to meet the existing design criteria and code requirements. Since the system has been reinstalled to meet or exceed the existing system requirements, the failure of this system during operation is bounded by the existing UFSAR and therefore does not affect the ability of systems or components to perform their safety function.

V. The steam generator insulation does not perform any safety function. Failure of the replacement insulation during postulated accidents as described in the UFSAR have been evaluated. The insulation will be seismically designed and installed to ensure that the insulation remains in place following a seismic occurrence. In addition, the insulation has been procured to meet or exceed the containment post-accident environmental conditions. A debris analysis (DC 90-13-1, Appendix 4-28, Steam Generator Debris Analyses Letter Report) has been prepared to address the effect of the replacement insulation on the containment sump analysis. Calculations performed in support of this debris analysis indicate a reduction of NPSH margin for the inside and outside recirculation spray pumps. However, adequate NPSH margin is still available for the recirculation spray systems. The calculations have also determined that an increase in NPSH will result for the low head safety injection pumps assuming the RWST LHSI switchover level setpoint change is implemented.

VI. Rigging activities will be performed in accordance with the station heavy loads procedure prior to reactor defueling. Subsequent to defueling, there are no in-containment systems or components required to perform a safety function. To preclude any possible adverse result of a postulated steam generator drop, all component cooling water, service water, fuel pit cooling and refueling purification system, and instrument air lines to/from the containment will be isolated during rigging operations for the steam domes and lower assemblies. No adverse impacts on stored spent fuel or spent fuel cooling would result from a postulated heavy load drop inside the containment.

DC 90-13-1, STEAM GENERATOR REPLACEMENT / NORTH ANNA / UNIT 1

Part C - Items Considered By This Safety Evaluation (Continued)

30. (Continued)

VII. The steam generator haul route study evaluated the transport path and potential impact on plant systems and structures. Transporter, prime mover and transporter test weight loadings on underground systems, as well as the impact of a lower assembly drop onto these underground systems from the transporter, have been reviewed and, where necessary, road improvements and protection of buried utilities have been provided. Transport of the steam generators or the haul route load test will not affect systems which are required to perform safety functions.

VIII The temporary modification to the purge system does not affect the ability of the system to perform its safety function. The modification only directs the air intake to the upper dome of containment. A failure of the attachment to the purge system will not affect the ability of the system to isolate the containment, purge the containment or effect the ability of the purge system to support Unit 2, if required. The temporary jib crane utilized during defueling will be operated in accordance with the existing station heavy loads procedure and will have mechanical stops to prevent movement of loads over the refueling canal or spent fuel. These precautions will limit the effects on safety systems in the inadvertent event of a load drop. The remainder of the temporary utilities have been evaluated for their impact on permanent plant structures and have been determined to have no adverse effect on systems that perform safety functions.

DC 90-13-1, STEAM GENERATOR REPLACEMENT / NORTH ANNA / UNIT 1

rt C - Items Considered By This Safety Evaluation (Continued)

31A. Will the activity cause equipment to be exposed (or potentially exposed) to adverse conditions, including those created by temperature, pressure, humidity, radiation meteorological conditions? [Commitment 3.2.11] [ ] Yes ** [X] No

31B. If Yes, could these conditions lead to equipment failure, or a dangerous atmosphere? Explain. [ ] Yes ** [X] No

See Page 5A for response.

32. Could failure of the activity feed back into protective circuitry? Explain. [ ] Yes [X] No

See Page 5A for response.

33. Could failure of the activity feed back into control circuits important to stable plant operation. (e.g. feedwater control, control rods)? [Commitment 3.2.12] [ ] Yes [X] No

See Page 5B for response.

34. Could the activity affect emergency diesel generator sequencing logics (including testing logics), or other logics important to safety. [Commitment 3.2.8] [ ] Yes ** [X] No

See Page 5B for response.

35. Could the activity cause a loss of separation of instrument channels/trains or electrical power supplies? Explain. [ ] Yes ** [X] No

See Page 5C for response.

36. Will the activity involve the addition or deletion of any loads on the Class 1E electrical distribution system? Explain. [ ] Yes ** [X] No

See Page 5C for response.

DC 90-13-1, STEAM GENERATOR REPLACEMENT / NORTH ANNA / UNIT 1

rt C - Items Considered By This Safety Evaluation (Continued)

31A/B. Will the activity cause equipment to be exposed (or potentially exposed) to adverse conditions, including those created by temperature, pressure, humidity, radiation, and/or meteorological conditions? [Commitment 3.2.11] If Yes, could these conditions lead to equipment failure, or a dangerous atmosphere? Explain.

This design change installs equipment which could be potentially exposed to a LOCA, MSLB, and HELB environments. Yet, there is no potential for exposing equipment to a more harsh environment than existed prior to the SGR. Components being installed as replacements as part of this design change will meet or exceed the design and qualification requirements for use in containment and will be replaced in the same configuration as the original equipment. (Reference DC 90-13-1, Appendix 4-23, Technical Report NE-883). None of the following activities will cause the peak and long-term containment conditions of pressure, temperature, humidity, and/or radiation levels to increase:

I. Steam Generator Vessel Repair
II. Piping Removal and Replacement
III. Instrumentation Removal and Replacement
IV. Blowdown System Modification
V. Insulation Removal and Replacement
VI. Rigging Activities
VII. Steam Generator Haul Route
VIII. Temporary Services (Impact on permanent facilities only)

Could failure of the activity feed back into protective circuitry? Explain.

V. A setpoint change will be made that changes the RWST level at which the LHSI transfers to the recirculation mode. This change will not affect the way the circuitry performs, but will affect the point at which the switchover is made. All setpoint changes and rescaling will be in accordance with approved instrumentation and control procedures.

None of the following activities involve modifications to the protective circuitry, or circuits connected to the protective circuitry, and therefore, could not feedback into that system:

I. Steam Generator Vessel Repair
II. Piping Removal and Replacement
III. Instrumentation Removal and Replacement
IV. Blowdown System Modification
VI. Rigging Activities
VII. Steam Generator Haul Route
VIII. Temporary Services (Impact on permanent facilities only)

33. Could failure of the activity feed back into control circuits important to stable plant operation. (e.g. feedwater control, control rods)? [Commitment 3.2.12]

Due to changes to the average coolant temperature (T_{ave}) and delta temperature between the hot and crossover leg of the RCS, the control programs for the steam dump, pressurizer level and rod control systems will require rescaling. These changes are the result of new operational thermal and hydraulic data. This rescaling will be performed in accordance with approved instrumentation and control procedures.

For the following activities, failure of the activity cannot cause feed back into control circuits important to stable plant operation.

I. Steam Generator Vessel Repair
II. Piping Removal and Replacement
III. Instrumentation Removal and Replacement
IV. Blowdown System Modification
V. Insulation Removal and Replacement
VI. Rigging Activities
VII. Steam Generator Haul Route
VIII. Temporary Services (Impact on permanent facilities only)

34. Could the activity affect emergency diesel generator sequencing logics (including testing logics), or other logics important to safety. [Commitment 3.2.8]

The primary power for the SGR is supplied by temporary transformers and switchboards connected to a non-Class 1E source. Other power will be supplied by non-Class 1E sources as well. Hence, none of the following activities will interface with emergency diesel generator sequencing logics or other logics important to safety:

I. Steam Generator Vessel Repair
II. Piping Removal and Replacement
III. Instrumentation Removal and Replacement
IV. Blowdown System Modification
V. Insulation Removal and Replacement
VI. Rigging Activities
VII. Steam Generator Haul Route
VIII. Temporary Services (Impact on permanent facilities only)

DC 90-13-1, STEAM GENERATOR REPLACEMENT / NORTH ANNA / UNIT 1

rt C - Items Considered By This Safety Evaluation (Continued)

35. Could the acti cause a loss of separation of instrument channels/trains or electrical power supplies? Explain.

VIII The temporary facilities and services activity will supply all power requirements for the SGR from a non-Class 1E temporary power source which provides adequate separation and isolation from the existing Class 1E power supplies. Thus, separation of instrument channels/trains and power supplies is not affected.

None of the following activities involve hardware modifications to instrumentation channel/trains or the plant electrical power supplies and therefore, could not cause a loss of separation of these systems:

I. Steam Generator Vessel Repair
II. Piping Removal and Replacement
III. Instrumentation Removal and Replacement
IV. Blowdown System Modification
V. Insulation Removal and Replacement
VI. Rigging Activities
VII. Steam Generator Haul Route

36. Will the activity involve the addition or deletion of any loads on the Class 1E electrical distribution system? Explain.

The use of a temporary non-Class 1E power source for the SGR loads will preclude the addition of any loads onto the Class 1E electrical distribution system. Therefore, none of the following activities will involve the addition or deletion of loads from the Class 1E electrical distribution system:

I. Steam Generator Vessel Repair
II. Piping Removal and Replacement
III. Instrumentation Removal and Replacement
IV. Blowdown System Modification
V. Insulation Removal and Replacement
VI. Rigging Activities
VII. Steam Generator Haul Route
VIII. Temporary Services (Impact on permanent facilities only)

DC 90-13-1, STEAM GENERATOR REPLACEMENT / NORTH ANNA / UNIT 1

rt C - Items Considered By This Safety Evaluation (Continued)

37. Will the activity adversely affect the ability of a system or component to maintain its integrity or code requirements? Explain. [ ] Yes [X] No

See Page 6A for response.

38. Will the activity reconfigure, eliminate, or add components and/or piping to the single or two-phase erosion/corrosion piping inspection program? Explain. [X] Yes [ ] No

See Page 6B for response.

39. Will additional surveillance requirements, as defined in the Technical Specifications, be necessitated by the activity? Explain. [ ] Yes [X] No

See Page 6B for response.

. Will the applicable Technical Specification basis description be altered by the activity? Explain. [ ] Yes [X] No

See Page 6C for response.

41. Will the activity result in a violation of any Limiting Condition for Operation (LCO), as defined in the Technical Specifications? Explain. [ ] Yes [X] No

All applicable Technical Specification LCOs will be maintained.

42. Were any other concerns or items identified during this review? If "Yes", explain. [ ] Yes [X] No

All concerns have been previously addressed in specific sections of the safety evaluation.

rt C - Items Considered By This Safety Evaluation (Continued)

37. Will the activity adversely affect the ability of a system or component to maintain its integrity or code requirements? Explain.

I. The steam generator vessel repair will be performed such that the replacement lower assembly is made up of components that meet or exceed the integrity and/or the code requirements of the original equipment. Code reconciliations have been performed, where appropriate, and addressed in DC 90-13-1, Appendix 4-33, Material Substitution Reconciliation. The repair and inspection will be in accordance with ASME Section XI.

II. The piping removal and reinstallation will not adversely affect the RCS, main steam, feedwater, chemical feed, sampling or wet layup systems, since the reinstallation work will be in accordance with ASME Section XI Code requirements for repairs performed on piping. The welding will also be performed in accordance with ASME Code requirements, as will post-repair piping inspection and testing. The replacement material has been procured to code requirements, stress analyses were verified and the piping and pipe supports were reviewed to ensure that they meet all seismic qualification requirements.

III. The level instrumentation removal and replacement requires that instrument tubing and condensate pots be reconnected in the same way as the original installation. The piping and instrument tubing will be reinstalled in accordance with code requirements utilizing material upgrades which have been analyzed and determined to be acceptable. Therefore, there will be no adverse effects on the integrity of the piping, instrument tubing, and condensate pots. The loose parts monitors are not pressure retaining components.

IV. The blowdown system piping modifications have been seismically qualified and meet all the material and installation code requirements for the affected portions of the steam generator blowdown system.

The new steam generator insulation has been evaluated to ensure that the replacement insulation does not adversely affect systems required to mitigate the effects of a design basis accident. In addition, the insulation has been seismically designed and installed and is a non-pressure boundary component.

VI. Rigging operations will not affect the integrity or code requirements of any system or components. All loading on structures resulting from rigging activities has been evaluated and determined to be within the existing design margins and, hence, are acceptable.

The existing polar crane will be used to lift the new and the original steam generator lower assemblies as well as the steam generator steam domes. Analysis of the polar crane (DC 90-13-1, Appendix 4-31, Letter Report-Special Overrated-Load Lift Qualification Report) has shown that it is capable of handling 280 tons. This capacity is more than enough to handle the original/new steam generator lower assemblies or the steam domes along with the associated rigging equipment and any other heavy loads. The polar crane will not require modification for the steam generator replacement program.

VII. The steam generator haul route study evaluated the transport path and potential impact on plant systems and structures. Transporter, prime mover and transporter test weight loadings on underground systems, as well as the impact of a lower assembly drop from the transporter onto these underground systems have been reviewed and, where necessary, road improvements and protection of buried utilities have been provided. Transport of the steam generators or the haul route load test will not affect systems which are required to perform safety functions.

37. (Continued)

VIII. The temporary facilities and services activity will perform all electrical installations in accordance with NFPA 70 -- National Electric Code. The modification to the purge system will require that the ductwork be breached long enough to install a temporary flexible duct. The blank flange will be reinstalled and leak tested upon completion of the SGR to verify system integrity. At the conclusion of the SGR, all temporary modifications will be removed and the purge system returned to its pre-outage conditions. As previously shown, the modifications will not adversely affect any other structure, system, or component. Thus, no structure, system, or component will lose its integrity or violate its code requirements due to this activity.

38. Will the activity reconfigure, eliminate, or add components and/or piping to the single or two-phase erosion/corrosion piping inspection program? Explain.

II. The piping removal and replacement activity involves secondary system piping/components. The existing feedwater loop piping will be removed and replaced with new chrome-moly piping (piping class 601C) to minimize corrosion/erosion effects. In addition, the main steam system modifications require changes to the erosion/corrosion program if the spool piece is installed. Applicable erosion/corrosion isometric drawings will be updated to reflect the system enhancements being implemented by this design change.

IV. The blowdown system modification will replace portions of the steam generator blowdown piping system, which is part of the erosion/corrosion piping inspection program. Applicable erosion/corrosion isometric drawings will be updated to reflect the system enhancements being implemented by this design change.

The following activities do not involve additions or changes to the erosion/corrosion piping inspection program:

- I. Steam Generator Vessel Repair
- III. Instrumentation Removal and Replacement
- V. Insulation Removal and Replacement
- VI. Rigging Activities
- VII. Steam Generator Haul Route
- VIII. Temporary Services (Impact on permanent facilities only)

39. Will additional surveillance requirements, as defined in the Technical Specifications, be necessitated by the activity? Explain.

All components being repaired or replaced are being returned to their original configuration. Additional items have been added to the ISI program. However, no additional Technical Specification defined surveillance requirements will be necessitated by the following activities:

- I. Steam Generator Vessel Repair
- II. Piping Removal and Replacement
- III. Instrumentation Removal and Replacement
- IV. Blowdown System Modification
- V. Insulation Removal and Replacement
- VI. Rigging Activities
- VII. Steam Generator Haul Route
- VIII. Temporary Services (Impact on permanent facilities only)

Part C - Items Considered By This Safety Evaluation (Continued)

40. Will the applicable Technical Specification basis description be altered by the activity? Explain.

I. The basis for Technical Specification 3/4.1.1.3 currently contains a RCS volume value of 9957 cubic feet. This value is being removed by a proposed administrative Technical Specification change (proposed change #276). The bases assumption that an equivalent RCS volume will be circulated in "approximately" 30 minutes is unchanged by the slight increase in RCS volume associated with the new steam generators.

The steam generator vessel repair activity does not change the description of the steam generators based on a review of Technical Specification Basis 3/4.4.5 - Steam Generators. The new steam generators perform the same function as the old steam generators.

Technical Specification Amendments 153 and 154 were issued to allow operation at reduced power levels until completion of the steam generator lower assembly replacement. These technical specification changes should revert to their pre-reduced power level values following completion of the steam generator replacement.

II. The piping removal and replacement activity does not change the description of the reactor coolant, main steam, feedwater, chemical feed, blowdown, sample or wet layup systems nor does it change any of the components based on a review of Tech Spec Basis description in Section 3/4.4 - Reactor Coolant System and 3/4.6 - Containment Systems. The replacement piping is functionally identical to the current piping and the Technical Specification bases is not affected.

III. The instrumentation removal and replacement requires that instrument tubing and condensate pots be reinstalled to their original configurations. Following level instrumentation activities, calibration of the transmitters will be performed to verify operability. In addition, relocation of the loose parts monitors will be followed by instrument checkout to verify operability. Therefore, applicable Technical Specification basis descriptions will not be altered.

IV. The blowdown system modification does not change the overall design function of the system. Therefore, the basis description of the Technical Specification is not changed by the steam generator blowdown modifications implemented by this design change.

V. The new insulation performs the same function as the old insulation. The bases description of Technical Specifications 3/4.5.2 and 3/4.5.3 ECCS subsystems and 3/4.6.2.2 Recirculation Spray Systems are applicable. The bases descriptions provided in these technical specification section will not be altered as analysis has determined that the safety related functions of the inside/outside recirculation spray and low head safety injection pumps are maintained.

VI. Rigging activities will not require changes to the Technical Specification bases because any releases caused by drops of the steam generators during hauling activities are analyzed to be within the existing bases of the Technical Specification for site releases.

VII. There are no applicable Technical Specification bases pertaining to the haul route activity, therefore, no changes to the Technical Specification bases are required.

VIII. The temporary services activity will not change the Technical Specification bases for the purge system because the system is operated within its design basis and its containment isolation capability remains operable. There are no Technical Specification bases related to the rest of the temporary modifications.

DC 90-13-1, STEAM GENERATOR REPLACEMENT / NORTH ANNA / UNIT 1

art C - Items Considered By This Safety Evaluation (Continued)

Items 42 through 61 consider potential impacts. VPAP-3001 provides engineering review guidelines for these items.

If the answer to any of the questions for these items is "Yes", a detailed engineering review must be performed. The results of the detailed review shall be documented on a supplemental page, identified by this safety evaluation number and Part C item number.

43. Station Security

Will the activity deactivate a security-related system or breach a security barrier? [X] Yes [ ] No

44. Fire Protection/Appendix R

A. Will the activity add or eliminate a significant amount of combustible material from plant areas? [ ] Yes [X] No

B. Will the activity change or affect any plant structure that acts as a fire barrier? [ ] Yes [X] No

C. Will the activity impact the performance of an existing fire protection or detection system? [ ] Yes [X] No

45. Equipment Qualification/Classification

A. Will the activity adversely affect any Class 1E electrical equipment located in a potentially harsh environment (as designated by the Environmental Zone Description)? [ ] Yes ** [X] No

B. Will the activity have the potential to alter any of the environmental parameters identified in the Environmental Zone Description? [ ] Yes ** [X] No

C. Will the activity have the potential to affect any of the Class 1E electrical distribution systems (e.g., 4KV, 480V, 120V(AC))? [ ] Yes ** [X] No

D. Will the activity add, eliminate, or have the potential to affect ASME Section XI equipment adversely? [X] Yes [ ] No

E. Will the activity change a setpoint in the Precautions, Limitations, and Setpoints (PLS) Document? [X] Yes [ ] No

F. Will the activity change or affect equipment on the EQML or Q-List? [X] Yes [ ] No

46. Seismic

Could the activity be adversely affected by a seismic event or could the activity affect surrounding equipment during a seismic event? [X] Yes ** [ ] No

47. Human Factors

A. Will the activity change instrumentation or controls in the Control Room or on the auxiliary shutdown panel? [ ] Yes [X] No

B. Will the activity alter the Control Room or the auxiliary shutdown panel? [ ] Yes [X] No

48. Safety Parameter Display System/Emergency Response Facility

Will the activity change any of the equipment associated with the SPDS/ERF, including SPDS/ERF computer inputs? [X] Yes ** [ ] No

DC 90-13-1, STEAM GENERATOR REPLACEMENT / NORTH ANNA / UNIT 1

Part C - Items Considered By This Safety Evaluation (Continued)

43. Will the activity deactivate a security-related system or breach a security barrier?

VIII. The temporary services activity will require that an auxiliary building security door be replaced with a temporary door that is notched to allow the passage of the temporary services into the containment. This replacement will be performed in accordance with the security plan. During the outage, the temporary door will be closed and secured. Following the outage, the temporary services will be removed and the permanent security door returned to its original configuration.

The following activities do not have any impact on security-related systems:

I. Steam Generator Vessel Repair
II. Piping Removal and Replacement
III. Instrumentation Removal and Replacement
IV. Blowdown System Modification
V. Insulation Removal and Replacement
VI. Rigging Activities
VII. Steam Generator Haul Route

45D. Will the activity add, eliminate, or have the potential to affect ASME Section XI equipment adversely?

I. As a result of the steam generator vessel repair activity, the following ISI joints have been eliminated due to the design of the steam generators:

- lower transition cone to shell barrel
- stub barrel to shell barrel
- transition cone girth weld

II. New piping and valves added to the secondary piping systems as a result of DC 90-13-1 must be included in the subject ISI program. In addition, as a result of the piping removal and replacement activity, the following items are to be added to the NAPS-1 ISI program:

- safe end to RC piping
- channel head nozzle to safe end
- Primary nozzle knuckle radius

IV. The new steam generator blowdown piping and valves must be included in the subject ISI Program. Upon completion of the piping modifications, the affected piping sections will be hydrostatically tested and then returned to service.

V. The replacement insulation is designed to be easily removed in areas that require ISI.

The following activities do not add, eliminate, or have the potential to affect ASME Section XI equipment adversely:

III. Instrumentation Removal and Replacement
VI. Rigging Activities
VII. Steam Generator Haul Route
VIII. Temporary Services (Impact on permanent facilities only)

45E. Will the activity change a setpoint in the Precautions, Limitations, and Setpoints (PLS) Document?

I. As a result of the steam generator replacement activities, operating parameters have changed. This modification will require changes to the PLS and setpoint documents. The details of the setpoint and PLS changes are found in DCP 90-13-1, Appendix 4-7.

V. The insulation removal and replacement activity requires that the RWST level setpoint at which LHSI transfers to recirculation mode be changed. Level Controllers will be revised to reflect this change. This setpoint change will require revision to NAPS 1 Set Point Document, Section 1.A.

The following activities do not change any setpoints in the PLS documents:

II. Piping Removal and Replacement
III. Instrumentation Removal and Replacement
IV. Blowdown System Modification
VI. Rigging Activities
VII. Steam Generator Haul Route
VIII. Temporary Services (Impact on permanent facilities only)

45F. Will the activity change or affect equipment on the EQML or Q-List?

I. The steam generator vessel repair activity replaces the steam generator lower assemblies, which are on the Q-List. The changes therefore affect Q-List component data and will be updated in accordance with DC 90-13-1, Appendix 4-4, Q-List Change Request.

II. The piping removal and replacement activity replaces the feedwater loop seal drain valves, which are on the Q-List. The changes therefore affect Q-List component data and will be updated in accordance with DC 90-13-1, Appendix 4-4, Q-List Change Request.

III. The level instrumentation removal and replacement will replace the level instrument root valves and vent valves, which are on the Q-List. The changes therefore affect Q-List component data and will be updated in accordance with DC 90-13-1 Appendix 4-4, Q-List Change Request.

IV. Equipment for the steam generator blowdown modification portion of the design change will be purchased safety related and include nine manual valves. The new valves will require revision to the existing Q-list component data. Q-list Change Request Forms are included in DC 90-13-1, Appendix 4-4.

The following activities will not change or affect equipment on the EQML or Q-List:

V. Insulation Removal and Replacement
VI. Rigging Activities
VII. Steam Generator Haul Route
VIII. Temporary Services (Impact on permanent facilities only)

DC 90-13-1, STEAM GENERATOR REPLACEMENT / NORTH ANNA / UNIT 1

rt C - Items Considered By This Safety Evaluation (Continued)

46. Could the activity be adversely affected by a seismic event or could the activity affect surrounding equipment during a seismic event?

DURING THE REPAIR:

I. During the steam generator vessel repair activity, the steam generator vessel could be damaged by a seismic event during the repair process, since it will not be seismically supported at all times. However, it is not required to perform a safety function during the repair activity. In addition, in the defueled condition, the surrounding equipment and structures are not required to perform safety functions during the repair activities.

II. During the removal and replacement of piping, damage could result from a seismic event during the cutting process.
III. Prior to cutting RCS and secondary plant piping systems attached to the steam generators including main steam,
IV. feedwater, feedwater drain, blowdown and shell drain, wet layup, steam generator level instrumentation, chemical feed and sampling, an appropriate support configuration will be established for the remaining pipe sections. These support configurations ensure that there are no adverse effects to safety related plant components and preclude the occurrence of any gravity missiles or sway interaction events. These evaluations are documented in DC 90-13-1, Reference 6.12.21 and 6.12.22.

VI. Rigging operations for the steam generator steam domes and lower assemblies will take place with the reactor vessel defueled. A seismic event during rigging operations could result in damage to the original or replacement steam generators. However, the steam generators are not required to perform any safety function during the steam generator replacement outage. A drop of a steam generator steam dome or lower assembly as a result of a seismic event could result in damage to in-containment equipment, the equipment hatch, or locally to the containment structure. However, safety-related systems, structures, or components which could be affected by a drop are not required to perform any safety functions at this time because the reactor is defueled. To preclude potential adverse effects on stored spent fuel or systems shared with Unit 2 as a result of a postulated steam generator drop, all component cooling water, service water, fuel pit cooling and refueling purification system, and instrument air lines to/from the containment will be isolated during rigging operations for the steam domes and lower assemblies. No adverse impacts on stored spent fuel or spent fuel cooling would result from a postulated heavy load drop inside the containment.

Lifting and handling activities required to be accomplished during defueling will be conducted in accordance with the existing station heavy loads procedure and are no different than those activities required during a normal refueling outage. To facilitate lifting activities within containment, a temporary jib crane will be installed and utilized during defueling. This temporary jib crane will be provided with mechanical stops to prevent any load movement over refueling equipment, spent fuel or the refueling canal. The temporary jib crane will also be controlled in accordance with the existing station heavy loads procedure. These requirements are imposed on jib crane operation to limit the affect of a postulated load drop and to ensure that a postulated load drop does not impact spent fuel.

VIII. The temporary services activity includes the auxiliary crane and reactor cavity cover which are not seismically designed. During a postulated seismic event, all temporary modifications will fail in a manner that will not adversely affect any safety-related systems, structures, or components required to be operable in this mode. The reactor cavity cover and auxiliary crane will be installed after the reactor is defueled. Therefore, a seismic event will not affect any system, structure, or component required to maintain the plant in a safe condition.

The following activities would not be adversely affected by a seismic event that occurred during the repair:

V. Insulation Removal and Replacement
VII. Steam Generator Haul Route

46. (Continued)

AFTER COMPLETION OF THE REPAIR:

When all plant systems have been returned to their original configuration, seismic analyses will have been performed to ensure the seismic adequacy of the new systems. Seismic calculations supporting this conclusion are referenced in the DC 90-13-1, Section 6.12 for the following activities:

I. Steam Generator Vessel Repair
II. Piping Removal and Replacement
III. Instrumentation Removal and Replacement
IV. Blowdown System Modification
V. Insulation Removal and Replacement

The following activities will not affect any plant operations or systems following completion of the repair:

VI. Rigging Activities
VII. Steam Generator Haul Route
VIII. Temporary Services (Impact on permanent facilities only)

48. Will the activity change any of the equipment associated with the SPDS/ERF, including SPDS/ERF computer inputs?

I. The steam generator vessel repair activity will require changes to Chapter 5, 6, and 10 of the UFSAR which are included in the ERF Controlled Documents List. See DC 90-13-1, Appendix 4-5, ERF Design Checklist. No changes to the SPDS are required.

V. The insulation removal and replacement activity will require a revision to an alarm setpoint in the ERF computer. The new containment analyses revise the RWST switchover setpoint in order to maintain required NPSH for the LHSI pump while in the recirculation mode. No changes to the SPDS are required.

The following activities do not impact the SPDS/ERF computer systems:

II. Piping Removal and Replacement
III. Instrumentation Removal and Replacement
IV. Blowdown System Modification
VI. Rigging Activities
VII. Steam Generator Haul Route
VIII. Temporary Services (Impact on permanent facilities only)

DC 90-13-1, STEAM GENERATOR REPLACEMENT / NORTH ANNA / UNIT 1

rt C - Items Considered By This Safety Evaluation (Continued)

49.	**Station Computers**		
	Will the activity have a significant potential to modify or add software to station computers?	[X] Yes **	[ ] No
50.	**Environmental Impact/Flooding**		
	A. Will the activity impact more than one-fourth of an acre of land, work in navigable waters, wells, dams, or wetlands, and/or involve any wastes or discharges?	[ ] Yes	[X] No
	B. Will the activity involve changes to site terrain, features, or structures?	[ ] Yes	[X] No
	C. Will the activity have a significant potential to expose safety related equipment to flooding via fluid system equipment/piping malfunction or failure?	[ ] Yes **	[X] No
51.	**Regulatory Guide 1.97**		
	Will the activity have a significant potential to modify equipment and/or instrumentation associated with Regulatory Guide 1.97 variables?	[ ] Yes **	[X] No
52.	**Heating-Ventilation-Air**		
	A. Will the activity have a significant potential to increase the heating or cooling loads in pl[illegible] areas and/or to plant equipment?	[ ] Yes	[X] No
	B. Will the activity change the existing ventilation system in any way?	[X] Yes	[ ] No
	C. Will the activity change any building walls, ceilings, windows, doors, or floors, in a way that may effect existing HVAC systems?	[ ] Yes **	[X] No
	Heavy Loads		
	Will the activity involve heavy loads (including the transfer of heavy loads in areas housing safety related equipment)?	[X] Yes	[ ] No
54.	**Materials**		
	Will detrimental materials be introduced into the containment or other plant areas?	[ ] Yes	[X] No
55.	**As Low As Reasonably Achievable (ALARA)**		
	Have ALARA concepts been included? (Detailed explanation not required.)	[X] Yes	[ ] No
56.	**Cumulative Effects**		
	Will the proposed change adversely impact the current system/component capacities or design performance?	[ ] Yes	[X] No
57.	**Design Basis Document**		
	Will the activity change applicable sections of the station design basis document?	[X] Yes	[ ] No
58.	**Simulator Impact**		
	If a change to the Control Room or Safe Shutdown Panel is considered, will the change need to be replicated in the simulator?	[X] Yes	[ ] No
59.	**Nuclear Materials Control**		
	Will the activity result in the procurement of special nuclear materials or change the handling or storage of special nuclear materials?	[ ] Yes	[X] No

DC 90-13-1, STEAM GENERATOR REPLACEMENT / NORTH ANNA / UNIT 1

Part C - Items Considered By This Safety Evaluation (Continued)

49. Will the activity have a significant potential to modify or add software to station computers?

Impact on SPDS/ERF Computer systems are specifically addressed in question #48 of this safety evaluation. Upgrades to the P250 computer have been addressed in DC 90-13-1, Engineering Review and Design, Section 3.11.

52B. Will the activity change the existing ventilation system in any way?

VIII. The temporary facilities and services activity involves temporary modifications to the purge system, including the attachment of a flexible duct and a volume control damper. The system will be returned to its original configuration following the steam generator repair.

The following activities will not change the ventilation system in any way:

I. Steam Generator Vessel Repair
II. Piping Removal and Replacement
III. Instrumentation Removal and Replacement
IV. Blowdown System Modification
V. Insulation Removal and Replacement
VI. Rigging Activities
VII. Steam Generator Haul Route

53. Will the activity involve heavy loads (including the transfer of heavy loads in areas housing safety related equipment)?

VI. Rigging activities will involve lifting heavy loads. To support repair of the steam generators, numerous lifting and handling activities associated with heavy loads are required. The following areas were evaluated as part of this activity:

- o The polar crane has been qualified for special over rated-load lifts up to 280 tons in accordance with ASME B30.2, Section 2-3.2.1.1 (see DC 90-13-1, Appendix 4-31). This capacity is adequate to handle the original/new steam generator lower assemblies or the steam domes, along with the associated rigging equipment. All handling equipment associated with the steam generator removal has been designed in accordance with accepted industry practice to provide reasonable assurance that the equipment will function as intended. A load test of the polar crane per ASME B30.2, Section 2-3.2.1.1 will be performed.

- o Lifting and handling activities required to be accomplished during defueling/refueling will be conducted in accordance with the existing station heavy loads procedure and are no different than those activities required during a normal refueling outage. To facilitate lifting activities within containment, a temporary jib crane will be installed and utilized during defueling. In accordance with the recommendations of NUREG-0612, this temporary jib crane will be provided with mechanical stops to prevent any load movement over refueling equipment, spent fuel or the refueling canal. The temporary jib crane will also be controlled in accordance with the existing station heavy loads procedure. These requirements are imposed on jib crane operation to limit the affect of a postulated load drop and to ensure that a postulated load drop does not impact spent fuel. Following defueling, all lifting and handling activities will be conducted in accordance with good commercial rigging practices utilizing approved load handling procedures and properly trained crane operators.

DC 90-13-1, STEAM GENERATOR REPLACEMENT / NORTH ANNA / UNIT 1

Part C - Items Considered By This Safety Evaluation (Continued)

53. (Continued)

- o The runway beam, the reactor cavity cover, the upper restraint temporary support, the steam dome laydown area and the auxiliary/jib crane support have been evaluated for their effect on permanent plant structures and components, are within design limits, and have no adverse effect on existing structures.

- o Any load drop during rigging operations inside containment will not affect containment integrity or the ability to contain a radioactive release due to a dropped load. All heavy load movements occurring with fuel in the containment building will be conducted in accordance with the station heavy loads procedure. Movement of loads with the temporary jib crane will be in accordance with existing plant heavy loads procedure in previously evaluated and approved safe load path boundaries to ensure that existing heavy load analyses bound the jib crane load handling activities. To preclude any adverse effects on stored spent fuel or systems shared with Unit 2 as a result of a postulated steam generator drop, all component cooling water, service water, fuel pit cooling and refueling purification system, and instrument air lines to/from the containment will be isolated during rigging operations for the steam domes or lower assemblies. No adverse impacts on stored spent fuel or spent fuel pool cooling would result from a postulated heavy load drop inside the containment.

  For a drop of an old steam generator inside the containment, outside the equipment hatch during movement from the equipment hatch platform to the transporter, or within the protected area, the consequences have been evaluated against the radiological consequences of other events in the same class of postulated accidents, such as the rupture of waste gas decay or volume control tanks. Based on this comparison, it is concluded that the drop of an old steam generator lower assembly inside the containment or during transfer within the protected area are less than applicable regulatory limits and are bounded by the consequences associated with existing accidents previously evaluated in the UFSAR (Reference Safety Evaluation Question 64B). No safety-related equipment would be impacted.

VII. The steam generator haul route activity will involve the movement of heavy loads, namely the lower assemblies. The following areas were evaluated as part of this activity:

- o The steam generator haul route has been evaluated and buried utilities within the haul route which can be subjected to the surcharge loadings associated with the loaded transporter, prime mover or transporter test weight have been identified and protected, if required, to ensure that the equipment or components are unaffected by the movement of the loaded transporter.

- o Cover plates have been designed and will be installed on the old steam generator lower assembly prior to transport to the storage facility. These cover plates have been designed to remain intact in the inadvertent drop of the steam generator lower assembly from the transporter. Thus, in the event of a drop of the steam generator from the transporter, no adverse consequences result since the old steam generator integrity is not breached.

- o The effect on safety related structures as a result of a drop of a lower assembly or test load has been evaluated and determined to be acceptable (Reference DC 90-13-1, Appendix 4-18). Thus, safety related structures buried or adjacent to the haul route will be unaffected in the unlikely event of a generator or test weight drop off the transporter.

53. (Continued)

The following activities will not involve heavy loads:

I. Steam Generator Vessel Repair
II. Piping Removal and Replacement
III. Instrumentation Removal and Replacement
IV. Blowdown System Modification
V. Insulation Removal and Replacement
VIII. Temporary Services (Impact on permanent facilities only)

57. Will the activity change applicable sections of the station design basis document?

V. The insulation removal and replacement activity will require changes to selected design basis documents. The review of the North Anna Power Station System Design Basis Document for the Recirculation Spray System (SDBD-NAPS-RS Revision 0), the Safety Injection System (SDBD-NAPS-SI Revision 0), the Quench Spray System (SDBD-NAPS-QS Revision 0) and the Service Water System (SDBD-NAPS-SW Revision 0) has determined that changes to these System Design Basis Documents will be required.

A Design Basis Document Change Notification Form with proposed revisions has been prepared and is included in DC 90-13-1, Appendix 4-9, Engineering Change Request and Mark-ups for System Design Basis Document Changes.

The following activities will not require changes to station design basis documents:

I. Steam Generator Vessel Repair
II. Piping Removal and Replacement
III. Instrumentation Removal and Replacement
IV. Blowdown System Modification
VI. Rigging Activities
VII. Steam Generator Haul Route
VIII. Temporary Services (Impact on permanent facilities only)

58. If a change to the Control Room or Safe Shutdown Panel is considered, will the change need to be replicated in the simulator?

Changes to the plant simulator as a result of the steam generator lower assembly replacement are addressed in DC 90-13-1, Section 3.17. This section concludes that simulator changes are required due to setpoint changes to initiate switchover from RWST suction to containment sump suction.

DC 90-13-1, STEAM GENERATOR REPLACEMENT / NORTH ANNA / UNIT 1

^rt C - Items Considered By This Safety Evaluation (Continued)

60. Masonry Block Walls

Wil' the activity affect a masonry block wall in any way, either through addition, removal, mounting of equipment, or location of safety related equipment within the vicinity of a block wall?	[ ] Yes	[X] No

61. Hazard/Chemical Release

Will the activity create a potential hazard/chemical release?	[ ] Yes	[X] No

62. Labeling

Will the activity affect station labeling?	[ ] Yes	[X] No

63. Management oversight of infrequent tests or evaluations (as defined by VPAP-1101, Test Control) recommended?	[ ] Yes	[X] No

DC 90-13-1, STEAM GENERATOR REPLACEMENT / NORTH ANNA / UNIT 1

rt D - Unreviewed Safety Question Determination

Part D is based on the results of the items considered in Part C.
Part C must be completed first.

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

See Page 10A for response.

A. Could the activity increase the probability of occurrence for the accidents identified above? State the basis for this conclusion. [ ] Yes [X] No

See Page 10A for response.

B. Could the activity increase the consequences of the accidents identified above? State the basis for this conclusion. [ ] Yes [X] No

See Page 10C for response.

C. Could the activity create the possibility for an accident of a different type than was previously evaluated in the Safety Analysis Report? State the basis for this conclusion. [ ] Yes [X] No

See Page 10F for response.

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

See Page 10G for response.

A. Could the activity increase the probability of occurrence of malfunctions identified above? State the basis for this conclusion. [ ] Yes [X] No

See Page 10G for response.

vt D - Unreviewed S: ety Question Determination (Continued)

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

All UFSAR Chapter 15 and Section 6.2 transient analyses were reviewed except those analyses that are unrelated to SG performance. The analyses not evaluated as part of this review are listed below:

UFSAR Section 15.3.3 - Inadvertent Loading of a Fuel Assembly into an Improper Location
UFSAR Section 15.3.5 - Waste Gas Decay Tank Rupture
UFSAR Section 15.3.6 - Volume Control Tank Rupture
UFSAR Section 15.4.5 - Fuel Handling Accident Outside Containment
UFSAR Appendix 15A - Spent Fuel Shipping Cask Drop Analysis

64A. Could the activity increase the probability of occurrence for the accidents identified above? State the basis for this conclusion.

The probability of occurrence for the accidents previously identified in question 64 have not been increased as discussed below.

I. As identified in the Westinghouse Thermal and Hydraulic Design Data Report (WNEP-9110) and Westinghouse Safety Evaluation SECL-90-113, the replacement steam generators have equivalent or improved physical, thermal, and hydraulic characteristics of the existing steam generators. The replacement generators incorporate component design modifications to provide enhanced flow distribution, minimize the potential for secondary side corrosion, and facilitate maintenance and inservice inspection.

The materials of construction for the new steam generators are compatible with the water chemistries of the primary and secondary systems and their associated components. In addition, the controls on the primary and secondary coolant chemistries, in conjunction with the design improvements incorporated in the replacement steam generators, will minimize the potential for corrosion-related degradation.

The replacement steam generators have been fabricated to ASME Section III, 1986 edition. As such, all pressure boundary materials have been verified to meet the ASME code strength requirements for the expected service conditions. The design cyclic loading and static stress design limits of the replacement steam generators are equal to or more conservative than similar limits for the existing steam generators. The repair will be in accordance with ASME Section XI. Code required stress reports will be updated when the repair is completed.

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

All reactor coolant system and secondary side piping and supports will be restored to their original configuration in accordance with ASME Section XI and ANSI B31.7 code requirements. Replacement materials, including all weld metal utilized, will satisfy the original code requirements and will meet the existing installation specification.

No new weld locations will be added to the reactor coolant piping as the welds will be made at existing weld locations. The reactor coolant piping internal decontamination procedure will utilize a metallic oxide grit to remove the thin oxide layer formed on the inside surface of the reactor coolant piping. This decrease in pipe thickness will not affect the pipes pressure rating as verified by testing. In addition, the reduction in pipe thickness will not affect the corrosion resistance as only the thin oxide layer is removed.

DC 90-13-1, STEAM GENERATOR REPLACEMENT / NORTH ANNA / UNIT 1

Part D - Unreviewed Safety Question Determination (Continued)

64A. (Continued)

The use of chrome-moly piping in the feedwater pipe will not increase the probability of a pipe break since the chrome-moly satisfies the design requirements of the original pipe class pressure and temperature as specified in the original installation specification and the stress analyses have been updated to reflect the new material. The repaired feedwater piping components will be procured and installed to meet or exceed the original pipe design requirements.

The main steam, chemical feed, wet layup and sampling piping will be installed to the original configuration in accordance with the current installation specifications. The new welds on the main steam lines will be performed in accordance with the approved procedures which will satisfy the original design requirements.

All modified piping systems will be subjected to nondestructive examination and hydrostatic testing in accordance with the Section XI, the Special Processes Manual (DC 90-13-1, Reference 6.5) and DC 90-13-1, Appendix 4-21, as applicable. In addition, periodic inspection will continue throughout the remaining life of the plant.

Applicable pipe stress and support analyses were performed for the main steam, feedwater and RCS piping and associated branch piping to document acceptability.

Therefore, with equivalent design, installation, testing and suitable material substitution, the licensing basis for these piping systems remains unchanged and the probability of occurrence of an accident is not increased.

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

IV. The modification to the steam generator blowdown system enhances the reliability of the blowdown system with respect to erosion/corrosion concerns. Physical changes to the blowdown system include the replacement of the existing 1 and 2 inch carbon steel piping with 1 and 2 1/2 inch chrome-moly piping which is an acceptable replacement material in terms of strength and compatibility with system chemistry. All supports associated with the blowdown system modifications have been reviewed to ensure that the blowdown system meets the original seismic design requirements. The improvements made to the blowdown system meet or exceed the current licensing basis requirements and do not increase the probability of occurrence of an accident.

V. The replacement blanket insulation has been procured and installed to meet or exceed the original design requirements for heat transfer and has been seismically qualified. Failure of the blanket insulation cannot initiate any of the relevant accidents addressed by this design change. Therefore, the issue of probability of accident occurrence is not applicable. A debris analysis (DC 90-13-1, Appendix 4-28, Steam Generator Debris Analyses Letter Report) has been prepared to address the effect of the replacement insulation on the containment sump analysis. Calculations performed in support of this debris analysis indicate a reduction of NPSH margin for the inside and outside recirculation spray pumps. However, adequate NPSH margin is still available for the recirculation spray systems. The calculations have also determined that an increase in NPSH will result for the low head safety injection pumps, assuming the RWST LHSI switchover level setpoint change is implemented.

64A. (Continued)

VI. Rigging activities covered by this safety evaluation that may be performed during defueling/refueling will be performed in accordance with the existing station heavy loads procedures and in a manner that will not interfere in any way with defueling or refueling operations that may be in progress. Following completion of defueling, rigging activities will be controlled in accordance with approved procedures. To preclude any adverse effects on stored spent fuel or systems shared with Unit 2 as a result of a postulated steam generator drop, all component cooling water, service water, fuel pit cooling and refueling purification system, and instrument air lines to/from the containment will be isolated during rigging operations for the steam domes or lower assemblies. No adverse impacts on stored spent fuel or spent fuel pool cooling would result from a postulated heavy load drop inside the containment.Therefore, the rigging activities associated with the SG replacement effort cannot initiate any of the relevant accidents being reviewed as part of this design change.

VII. The steam generator haul route has been evaluated and those components within the haul route that could have potential impact on plant operations have been identified and adequate protection provided to withstand the imposed surcharge loads due to the loaded transporter, prime mover and transporter test weights. Therefore, this activity will have no adverse affect on operability of underground systems along the transport route. Therefore, the probability of occurrence of an accident is not increased.

VIII. The temporary modifications do not have the potential to increase the probability of occurrence of relevant accidents while the unit is defueled. The purge exhaust and supply system will be operated in accordance with approved procedures and within its design flow rates. No modifications to the logic associated with containment purge operation or with containment isolation are performed. Thus, the capability of the purge system will not be affected as a result of this temporary modification. In addition, this temporary modification affects only the purge system ductwork within Unit 1 containment and will have no adverse affect on the capability of the purge system to support Unit 2 operation, if required.

No modifications are being made to the plant power distribution system. Therefore, there is no impact on the physical separation between non-Class 1E and Class 1E power supplies. Temporary power will be from RCP 1C power feeds during implementation which is a non-Class 1E supply.

Movement of loads with the temporary jib crane will be in accordance with the existing heavy loads procedure in previously reviewed and approved safe load path boundaries. In addition, the jib crane will be provided with mechanical stops to limit the boom travel so that the refueling canal or spent fuel cannot be impacted.

All temporary modifications will be removed and the affected systems returned to normal operating status prior to completion of the steam generator repair outage. This includes removal of the reactor cavity cover and the auxiliary/jib cranes.

64B. Could the activity increase the consequences of the accidents identified above? State the basis for this conclusion.

The modifications addressed in this design change package do not increase the consequences of the accidents previously identified above for the following reasons:

I. As a result of the steam generator vessel repair, Technical Report NE-883, Revision 1 was prepared to consolidate and summarize the safety analyses and evaluations supporting North Anna 1 operation following the steam generator repair. In this report, each UFSAR Chapter 15 accident analysis applicable to North Anna Unit 1 operation with the repaired steam generator has been evaluated and it has been determined that none of the accident consequences were found to be more limiting than those currently documented in the UFSAR.

64B. (Continued)

The installation of a main steam flow restrictor in the nozzles of each steam generator will decrease the blowdown from a faulted steam generator. The consequences of a main steam line break as evaluated in Technical Report NE-883, Revision 1, did not take credit for the existence of the main steam flow restrictor. As concluded in NE-883, Revision 1, the existing licensing basis main steam line break analysis remains valid for North Anna Unit 1 following steam generator replacement.

II. The reactor coolant, main steam, feedwater, wet layup, sample, and chemical feed systems will be restored to their original configuration after SG replacement. As described in approved calculations, all applicable design basis seismic stress and support analyses have been evaluated/performed, as applicable, to verify the capability of the repaired systems to perform their intended functions. Piping removal and replacement will be in accordance with the Special Process Manual (DC 90-13-1, Reference 6.5), Appendix 4-21 of DC 90-13-1, and ASME Section XI Code requirements for piping repairs, welding, and nondestructive examination to ensure the systems satisfy the original design requirements. Therefore, all design basis evaluations of the consequences of accidents involving these systems remain valid.

III. The piping, instrument tubing, and condensate pots will be reinstalled to their original configurations using upgraded material. Furthermore, the loose parts monitors will be relocated in accordance with approved procedures and will be verified to be operational following relocation by performing channel checks. Therefore, all design basis evaluations of the consequences of accidents for which steam generator level instrumentation is assumed to be operable remain valid.

IV. The function and operation of the steam generator blowdown system will not change. The increase in pipe size does not increase the consequences of an accident since these pipe sizes are bounded by the larger break sizes used in the accident analysis. Therefore, all design basis evaluation of the consequences of accidents involving the blowdown system remain valid.

V. The replacement thermal insulation has been qualified for use within containment and has been procured to meet the post accident environmental conditions within containment. The replacement insulation is seismically installed to ensure the insulation remains attached to the generator in the event of a seismic occurrence. The insulation performs no safety function in the event of a design basis accident. Appropriate evaluations have been performed in accordance with the recommendations of Regulatory Guide 1.82 to ensure that the replacement insulation does not adversely affect emergency core cooling and engineered safeguards systems. Calculations performed in support of the debris analysis (DC 90-13-1, Appendix 4-28, Steam Generator Insulation Debris Analysis Letter), analyze the LOCA with new input resulting from the insulation replacement. The calculations determine that the inside and outside recirculation spray and low head safety injection pumps will perform their safety related functions. This debris analysis has also determined that the insulation debris fragments that pass through both the sump screens and pump suction screens would not affect the operation of the inside and outside recirculation spray and low head safety injection pumps. In addition, the small fragments that pass through the sump and pump suction screens will not cause blockage of the header spray nozzles. Therefore, all design basis evaluations of the consequences of accidents are unaffected by the replacement insulation.

VI. All movements of heavy loads within containment while fuel remains within the reactor containment will be conducted in accordance with the existing station heavy loads procedures to ensure that the loads remain within the established safe load paths and to ensure that, in the inadvertent event of a load drop, the consequences remain within the established acceptance criteria. Once the reactor vessel is defueled and all fuel is safely stored within the spent fuel pool, load handling activities within containment have no effect on the capability to maintain spent fuel pool cooling or reactivity control. To preclude any adverse effects on stored spent fuel or systems shared with Unit 2 as a result of a postulated steam generator drop, all component cooling water, service water, fuel pit cooling and refueling purification system, and instrument air lines to/from the containment will be isolated

64B. (Continued)

during rigging operations for the steam domes or lower assemblies. Not adverse impacts on stored spent fuel or spent fuel pool cooling would result from a postulated heavy load drop inside the containment. Therefore, the rigging activities associated with the SG replacement effort cannot initiate any of the relevant accidents being reviewed as part of this design change. Furthermore, in the defueled condition, all UFSAR accidents associated with reactor operation, reactor criticality, and reactor decay heat removal during shutdown are not credible occurrences.

Prior to removal from containment, the old steam generator lower assemblies will be drained and steel closure plates will be installed over all lower assembly openings. All lifting and handling equipment associated with movement of the lower assemblies has been designed and supplied with sufficient capacity to adequately perform the lift. Nevertheless, in the unlikely event of a steam generator lower assembly drop inside the containment or within the protected area, a breach of the primary side closure plates may occur. No safety-related equipment would be adversely impacted by a drop outside the containment. To assess the radiological consequences associated with this drop, an analysis was performed which conservatively assumed that 10 percent of the solid radioactive corrosion products contained within the primary side of the steam generator are released outside the steam generator boundary following impact. One percent of the released amount was assumed to be of small enough particulate size to become airborne and be transported to the nearest site boundary. All activity was assumed to be Cobalt-60 which results in the maximum organ (lung) dose of isotopes normally found in the steam generator. The accident atmospheric dispersion factor (X/Q) from FSAR Section 2.3.4.1 was used.

Using these conservative assumptions, the maximum calculated offsite dose from a postulated drop was 0.027 mrem whole body. This whole body dose is a small fraction of the 10 CFR Part 100 guidelines values for accidental releases and is within the guideline dose limit for unrestricted areas per 10 CFR Part 20.105(a).

The acceptability of the offsite dose consequences associated with a postulated drop have been evaluated and compared to the consequences of other events in the same class of postulated accidents for waste gas or waste liquid releases. For assessing offsite dose consequences, the drop of a steam generator lower assembly is classified as a rupture of a tank containing radioactive material. The Waste Gas Decay Tank Rupture (Section 15.3.5) and the Volume Control Tank Rupture (Section 15.3.6) are the limiting events currently evaluated in the UFSAR for the accidental releases of waste gas and waste liquid, respectively. These limiting licensing basis events are also of a more permanent nature than the comparatively short-term steam generator rigging operations planned during the repair outage. For the waste gas decay tank rupture, a maximum whole body dose of 1.6 rem is calculated. For the volume control tank rupture, a maximum whole body dose of 0.26 rem is predicted. The evaluated consequences of a steam generator lower assembly drop are within the applicable regulatory guidelines and are less than the limiting, and more permanent, licensing basis accidents currently evaluated in the UFSAR. Thus, the consequences associated with this class of accidents will not be increased.

VII. Modifications to the steam generator haul route, transport of the steam generators and load testing of the haul route have no effect on the design or operation of existing systems and therefore, have no impact on the consequences of accidents previously evaluated. All underground components within the haul routes have been evaluated for the surcharge loads associated with steam generator transport, prime mover and transporter test weights have been determined to either be unaffected by the loads or adequate protection has been provided. Thus, these systems will remain available to perform their function. Furthermore, a drop of a replaced steam generator from the transporter during movement has been evaluated and the structural integrity of the generator is not breached, safety-related structures adjacent to the haul route are not adversely impacted and buried utilities within the haul route are not affected. Thus, a drop of the steam generator lower assembly or test load from the transporter will not result in a release of radioactivity to the environment.

VIII. In the defueled condition, all UFSAR accidents associated with reactor operation, reactor criticality and reactor decay heat removal are not credible occurrences. The only accident which requires consideration regarding temporary modifications is the fuel handling accident inside containment. During all fuel handling, containment integrity is maintained. Planned modification to the containment purge system will not be made until the vessel is defueled. Therefore, the consequences of the fuel handling accident are not increased.

64C. Could the activity create the possibility for an accident of a different type than was previously evaluated in the Safety Analysis Report? State the basis for this conclusion.

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

I. The possibility of an accident that is different from that already evaluated in the UFSAR is not created because, as evaluated in Westinghouse Safety Evaluation SECL-90-113, the replacement steam generators have been designed and fabricated to criteria that are equivalent to or better than the existing steam generators. Design improvements incorporated in the replacement steam generators are similar to the features included on other previous Westinghouse steam generators and no unique operating characteristics have been created. The replacement steam generators have also been determined to have no adverse impact on the function or performance of connected systems, components and structures. The upper restraints are not removed prior to fuel offload.

II. All reactor coolant system and secondary side piping and supports will be restored to their original configuration in accordance with the original code requirements using materials which meet or exceed the original design requirements. No credit has been taken in the accident analyses for the presence of the main steam nozzle flow restrictor. No operational changes occur as a result of the piping activities.

The reactor coolant loops A, B, and C will have temporary shielding installed on the hot leg, cold leg, crossover leg, and bypass line prior to the RCS piping cuts. RC Loop B will be severed first, prior to the completion of fuel offload. This configuration will exist with either the hot or crossover leg severed, or with both lines severed. Cutting operations for the Loop A and C RCS piping will not be performed until after defueling. The acceptability of these configurations is documented in DC 90-13-1, Reference 6.12.21. The loop isolation valves shall be isolated prior to making any piping cuts. The acceptability of the RC loops A, B, and C piping severed from the steam generators after fuel is removed from the reactor is also documented in DC 90-13-1, Reference 6.12.21. In this configuration, the RHR system remains available to remove decay heat. The acceptability of operation with the loop stop valves isolated and the means available for backup cooling should RHR be lost in this configuration is documented in Technical Report No. 865, Rev. 1.

Secondary plant piping systems including main steam, feedwater, feedwater drain, wet layup, steam generator blowdown and shell drain, chemical feed, sampling, and steam generator level instrumentation are also planned to be severed from the steam generators while fuel is in the reactor vessel. Prior to severance, an appropriate support configuration will be established that ensures no adverse effects, including gravity missiles and sway interaction concerns, will result to any safety-related components. Potential loads considered include deadweight and seismic effects. This evaluation is documented in DC 90-13-1, Reference 6.12.22.

Therefore, an accident of a different type than previously evaluated in the UFSAR will not be created.

III. The piping, instrument tubing, and condensate pots are being reinstalled to their original configurations using material which meets or exceeds the existing design requirements. The function and operation of the system following the modification will not change. The loose parts monitors are being relocated and following relocation will be verified to be operable. Therefore, the possibility of an accident of a different type from that evaluated

64C. (Continued)

previously would not be created.

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

V. The replacement insulation performs the same function as the original insulation and is seismically installed. Failure of the SG insulation does not, in itself, initiate any existing type of accident. The change from encapsulated insulation to blanket insulation does not alter this conclusion. Therefore, no new accident is possible as a result of the replacement insulation.

VI. Rigging activities performed during defueling operations will be conducted in accordance with the existing heavy load handling procedures to ensure that load handling occurs only in currently analyzed and approved safe load paths.

The radiological consequences of a postulated drop of an old steam generator lower assembly inside the containment or within the protected area have been evaluated and determined to be within applicable regulatory limits and less than the limiting case events within the same classification of accidents currently evaluated in the UFSAR. Thus, no new accidents are created as a result of the rigging activities.

VII. The steam generator haul route has been evaluated to ensure that any components within the load path subjected to the surcharge loads of the loaded transporter, prime mover and transporter test weight are either capable of sustaining the imposed surcharge loads or are adequately protected using temporary means. Impact associated with drop of a lower assembly from the transporter may result in local damage to the structure, but significant structural damage would not occur. Therefore, no new accident is created as a result of the SG transport or haul route load testing activities.

VIII. Following the completion of the SGR and prior to a return to power, all temporary modifications will be removed. The purge system will be returned to pre-outage conditions and operated in the same fashion as before the SGR project. All other temporary modifications will be removed at the completion of the steam generator outage. This will ensure that no new failure modes are introduced into the plant. Thus, this activity will not create the possibility of an accident of a different type as the operating performance of the plant following SGR is identical to the operating performance before SGR.

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

The assumed single failures which are currently part of Chapter 15 of the UFSAR have been considered.

65A. Could the activity increase the probability of occurrence of malfunctions identified above? State the basis for this conclusion.

The probability of occurrence of malfunction of equipment important to safety previously evaluated in the safety analysis report is not increased by the steam generator replacement (also refer to Answer 64A). Applicable design constraints were analyzed and none were found to be more limiting than those currently documented in the UFSAR, provided North Anna Unit 1 is operated in accordance with the Unit 1 Technical Specifications that will be applicable following steam generator replacement.

DC 90-13-1, STEAM GENERATOR REPLACEMENT / NORTH ANNA / UNIT 1

rt D - Unreviewed Safety Question Determination (Continued)

B. Could the activity increase the consequences of the malfunctions identified above? State the basis for this conclusion. [ ] Yes [X] No

See Page 11A for response.

C. Could the activity create the possibility for a malfunction of equipment of a different type than was previously evaluated in the Safety Analysis Report? State the basis for this conclusion. [ ] Yes [X] No

See Page 11A for response.

66. Has the margin of safety of any part of the Technical Specifications as described in the bases section been reduced? Explain. [ ] Yes [X] No

See Page 11A for response.

67. Does the proposed change, test, or experiment require a change to the Technical Specifications? Explain. [ ] Yes [X] No

See Page 11B for response.

Questions 68 and 69 Apply Only to 10 CFR 72.48 Safety Evaluations for Surry ISFSI

68. Does the proposed change, test, or experiment involve a significant unreviewed environmental impact? Explain. [ ] Yes [ ] No [X] N/A

69. Does the proposed change, test, or experiment involve a significant increase in occupational exposure? State the basis for this conclusion. [ ] Yes [ ] No [X] N/A

DC 90-13-1, STEAM GENERATOR REPLACEMENT / NORTH ANNA / UNIT 1

rt D - Unreviewed Safety Question Determination (Continued)

65B. Could the activity ncrease the consequences of the malfunctions identified above? State the basis for this conclusion.

The consequences of the malfunction of equipment important to safety previously evaluated in the safety analysis report are not increased by the steam generator replacement (also refer to Answer 64B). Applicable design constraints were analyzed and none were found to be more limiting than those currently documented in the UFSAR, provided North Anna Unit 1 is operated in accordance with the Unit 1 Technical Specifications that will be applicable following steam generator replacement.

65C. Could the activity create the possibility for a malfunction of equipment of a different type than was previously evaluated in the Safety Analysis Report? State the basis for this conclusion.

The possibility of a malfunction of equipment of a different type than previously evaluated in the safety analysis report is not created (refer to Answer 64C). Steam generator replacement does not involve any alterations to the physical plant which would introduce any new or unique operational modes or accident precursors. With respect to accident analyses, the Model 51F steam generators may be considered a "replacement" component for the existing Model 51 steam generators. As such, their performance does not create conditions that would lead to an accident or equipment malfunction beyond the scope of those already evaluated in the safety analysis report.

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

The margin of safety as defined in the basis for any Technical Specification is not reduced by the proposed change (Reference question 40 and 67). The replacement steam generators have been demonstrated to insignificantly affect the transient system response during postulated UFSAR Chapter 15 accidents. Accident analyses for all UFSAR Chapter 15 transients have been performed which bound allowable operation in accordance with the North Anna 1 Technical Specifications that will be applicable following steam generator replacement. All accident analyses meet their respective acceptance criteria. It may, therefore, be concluded that steam generator replacement does not decrease the margin of safety as defined in the basis for any Technical Specification.

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

I. Based on a review of Technical Specification Sections 2.2, 3/4.3.1, 3/4.3.2, 3/4.4 and 3/4.6, there are no changes required to the Technical Specifications for steam generator vessel repair activity. An administrative Technical Specification change (proposed change #276) removes a specific RCS volume value from the bases section of section 3/4.1.1.3 and section 5.4.2. Accordingly, no revision is needed as a result of this design change.

II. Based on a review of Technical Specification Sections 3/4.4 and 3/4.6, there are no changes required to the Technical Specifications as a result of the piping removal and replacement activity.

III. Based on a review of Technical Specification Sections 3/4.3.1, 3/4.3.2, and 3/4.3.3, there are no changes required to the Technical Specifications as a result of the instrumentation removal and replacement activity.

IV. The blowdown system modification activity initiated a review of the Technical Specifications that concluded no Technical Specification sections apply, and therefore, no changes are required as a result of this activity.

V. The insulation removal and replacement activity initiated a review of the Technical Specifications that concluded no Technical Specification sections apply, and therefore, no changes are required as a result of this activity.

VI. The movement of the new and old SGs and the steam domes within containment are only performed after the fuel is removed from the reactor core. Therefore, the operability of Technical Specification safety equipment is not required and no change to Tech. Specs. is needed. Based on a review of Technical Specifications, there are no changes required as a result of the rigging activities.

I. The steam generator haul route activity initiated a review of the Technical Specifications that concluded no Technical Specification sections apply, and therefore, no changes are required as a result of this activity.

VIII. The temporary facilities and services activity involves the installation of temporary systems that will be removed after completion of the SGR. Thus, they are not subject to any Technical Specification requirements. In mode 6, the purge system is required to respond to a containment high radiation signal by isolating the containment from the environment per Technical Specification Section 3/4.3.3. This capability will be maintained. There are no Technical Specification requirements related to the "B" RCP power supply or the other non-Class 1E power sources. Therefore, based on a review of applicable sections, this activity will not require a change to the Technical Specifications.

Technical Specification Amendments 153 and 154 were issued to allow operation at reduced power levels until completion of the steam generator lower assembly replacement. These technical specification changes should revert to their pre-reduced power level values following completion of the steam generator replacement.

DC 90-13-1, STEAM GENERATOR REPLACEMENT / NORTH ANNA / UNIT 1

rt D - Unreviewed Safety Question Determination (Continued)

If all responses are "No" to Questions 64 through 69, the proposed activity may be implemented following SNSOC approval. All related documentation must be retained.

If a response is "Yes" to any part of Questions 64 through 67, an operating license amendment must be approved by the NRC before the change, test, or experiment may be implemented.

If a response is "Yes" to Question 68 or 69, an application for an ISFSI license amendment must be approved before the change, test, or experiment may be implemented.

70. Reviewer Name (Print) Donald L Reid /	71. Reviewer Title Shift Technical Advisor	
72. Reviewer Signature		73. Date 9-21-92
Concurrence Documentation for ** Items Answered "Yes" in Part C		
74. Design authority Reviewer Name (Print) G.T. BISCHOF	75. Design Authority Reviewer Title ASST. SUPT. OF ENGINEERING	
76. Design Authority Reviewer Signature / K.L. Basehore		77. Date 9-2

Table 5.2-10A
(Unit 1)

STEAM GENERATOR PRIMARY/SECONDARY BOUNDARY COMPONENTS

Condition: Normal, Upset and Test Conditions

Location (Figure 5.2-1A)	Description	Maximum Fatigue Usage
1	Tubesheet Center	0.16

UFSAR CHANGE REQUEST

VIRGINIA POWER

UFSAR Change Request

GOV 02-54

LINS-2802

1. Affected UFSAR Facility: [X] North Anna [ ] Surry [ ] Surry ISFSI	2. Change Number

3. UFSAR Chapter/Sections Affected

See attached supplement sheet

4. Description of Change

See attached supplement sheet

5. Basis of Change

See attached supplement sheet

6. Supporting Safety Evaluation Number (Reference DC 90-13-1)

7. Other Supporting Documents (List and Attach)

DC 90-13-1 STEAM GENERATOR REPLACEMENT (Not attached)

	Date
8. Change Originator (Signature) D.W. Lippard — Daniel W. Lippard	9/14/92
9. Responsible Supervisor Approval (Signature) R.K. ~~Bayer~~ / K.J. Basehore	9/15/92 9/22/92

Signatures in Blocks 10 - 13 not required if approval evident in attached supporting documentation.

	Date
10. Design Authority Approval (Signature) K.J. Basehore	9-22-92
11. Other Approval - Optional (Signature)	
12. SNSOC Approval (Signature)	9/30/92
13. MSRC Review (Signature)	
14. Implementation Date / 15. Supervisor Licensing Corporate Approval (Signature)	

Key UFSAR-Updated Final Safety Analysis Report; SNSOC-Station Nuclear Safety Operating Committee; MSRC-Management Safety Review Committee; ISFSI-Independent Spent Fuel Storage Installation

Form No. 722806(Apr 92)

3. UFSAR Chapter/Sections Affected:

Sections; 3.7, 5.2, 5.5, 6.2, 6.3, 10.3, 15.2, 15.3, 15.4

Tables; 3.7-11, 3.7-12, 5.2-4 thru 12, 5.2-14, 5.2-17, 5.2-22, 5.5-3, 5.5-18 thru 20, 6.2-2, 6.2-5, 6.2-13 thru 17, 6.2-47 thru 53, 6.2-77, 15.4-1 thru 6

Figures; 5.2-1 thru 4, 5.5-3, 5.5-17, 6.2-11 thru 15, 6.2-85 thru 95, 6.3-11 thru 17, 10.4-7, 8, 11, 13, 14, 17, 15.4-1 thru 17

Add Figure; 6.2-116

Delete Figures; 5.5-24, 5.5-25, 6.2-84

4. Description of Change: This change package is necessary to incorporate the details of the **STEAM GENERATOR REPLACEMENT** and associated design and component modification. Chapter 3 updates design methodology, Chapter 5 updates the SG design, Chapter 6 addresses the effects of the new insulation on the containment, Chapter 10 includes material upgrades and changes to the main steam, feedwater, blowdown, and chemical feed systems, and Chapter 15 provides detailed updates to plant transient analysis resulting from SG insulation replacement and, where applicable, the reduction of tube plugging levels to original design values.

5. Basis of Change: The Steam Generator Replacement effort replaces the current Model 51 SGs with functionally equivalent Model 51F SGs. The 51F SGs have several design enhancements that provide improved durability and a reduction in corrective maintenance. Due to minor changes in dimensions and weights, the new SGs have been reevaluated for stresses and loads to demonstrate continued design and code compliances. The UFSAR requires updates to document these design enhancements and revised structural analysis. Changes to materials in the feedwater and blowdown systems for improved resistance to erosion/corrosion effects are documented by this update.

The replacement of the old SG insulation required a reanalysis of NPSH at the containment sump per RG 1.82. Additionally, a containment analysis was performed to verify that required safety parameters were met. The mass and energy release values were based on the new SG design. This update documents the satisfactory results of this evaluation.

(2)

3.7.2.6 Validation of Computer Programs

3.7.2.6.1 Programs within Stone & Webster Scope

The following computer programs were used in dynamic and static analyses for Seismic Class I Stone & Webster designed equipment and components:

1. STRUDL II - multipurpose mechanics program.
2. STARDYNE - dynamic analysis program.
3. ST-176 - seismic spectra response calculations.
4. SHELL 1 - shell analysis program.
5. Stress Analysis of Shells of Revolution.
6. MARC - nonlinear finite element program, static.
7. LIMITA II - nonlinear transient dynamic analysis.
8. MAT 5 - foundation mat analysis.
9. Time-History Program - seismic response spectra.
10. PRATO - mixed finite element with curved surfaces.
11. NUPIPE-SW - performs a linear elastic analysis of three dimensional piping systems subjected to thermal, static and dynamic loads.

ADD

3.7.2.6.1.1 STRUDL II. STRUDL II has been designed as a modified subsystem of the Integrated Civil Engineering System (ICES)[4] which was designed and formulated at the Massachusetts Institute of Technology, Department of Civil Engineering.

The finite element method[5] provides for the solution of a wide range of solid mechanics problems. Its use within the context of the STRUDL analysis facilities expands these for the treatment of plane stress, plane strain, plate bending, shallow shell, and three-dimensional stress analysis problems.

STRUDL II also provides a dynamic analysis capability for linear elastic structures undergoing small displacements. Either free or forced vibrational response may be obtained; in the latter case, the forcing function may be in the form of time histories or response spectra.

The three-dimensional finite element capability of STRUDL is used to analyze the containment at the regions of the personnel and equipment hatches and other specific regions of interest.

12. STRUDL-SW - multipurpose static and/or dynamic analysis program
13. STEHAM - determines flow induced forcing functions on piping systems during a steamhammer event

ADD

(3)

A 5010.65

CALCULATION IDENTIFICATION NUMBER				
J.O. OR W.O. NO.	DIVISION & GROUP	CALCULATION NO.	OPTIONAL TASK CODE	PAGE ____

3.7.2.6.1.11 NUPIPE - SW

The NUPIPE - SW piping program performs a linear elastic analysis of three dimensional piping systems subjected to thermal, static and dynamic loads. NUPIPE - SW utilizes the finite element method of analysis with special features incorporated to accommodate specific requirements in piping analysis. These features include simplified input for piping system description, use of special curved elements to represent piping elbows, and analytical conformance to the ASME Section III Nuclear Power Plant Components Code.

NUPIPE - SW will handle all loading conditions required for complete nuclear piping analyses. A given piping configuration may be analyzed successively for a number of static and dynamic load conditions in a single computer run. Separate load cases, such as thermal expansion and anchor displacements, may be combined to form additional analysis cases. The piping deadload analysis considers both distributed weight properties of the piping and any added concentrated weights.

The NUPIPE - SW program is designed to perform analysis in accordance with the ASME Boiler and Pressure Vessel Code Section III, Nuclear Power Plant Components (Code). Features insuring code conformance include use of accepted analysis methods, incorporation of specified stress indices and flexibility factors, proper combination of moment resultants, and provision to (automatically) generate results of combined loading cases. A program option is available to specify ~~among~~ Class 1 analysis per Article NB-3600 of the code, Class 2 analysis per Article NC-3600 of the Code, analysis per ANSI B31.1.0 power piping code, analysis per ANSI B31.3 petrochemical code and combined Class 1 and Class 2 analysis per Articles NB-3600 and NC-3600 of the code.

A 5010 65

(4)

CALCULATION IDENTIFICATION NUMBER				
J.O. OR W.O. NO.	DIVISION & GROUP	CALCULATION NO.	OPTIONAL TASK CODE	PAGE ____

3.7.2.6.1.12 STRUDL-SW

STRUDL-SW

~~TH~~ performs a static and/or dynamic analysis of a structure composed of members. The capability also exists (for a static analysis) to check or design structural members based on various code requirements. This program is a completely documented and qualified subset of STRUDL-II (ST-015).

STRUDL-SW may be applied to a wide range of structural problems using the same basic input. It handles two-dimensional trusses, frames, and grids, as well as three-dimensional trusses and frames. Only elastic, small displacement analysis is available.

The solution method used is the displacement method for structural analysis. This procedure requires the specification of member properties in some acceptable form and treats the joint displacements as unknowns. Stiffness and mass matrices (for dynamics) of the structure are assembled or input and the static and/or dynamic problem is solved.

A 5010 65

(5)

CALCULATION IDENTIFICATION NUMBER				
J.O. OR W.O. NO.	DIVISION & GROUP	CALCULATION NO.	OPTIONAL TASK CODE	PAGE ____

3.7.2.6 1.13 STEHAM

The STEHAM program determines the flow induced forcing functions on piping systems during a steamhammer event for the use of subsequent piping dynamic analysis.

The analysis is based upon the method of characteristics with finite difference approximations for solutions of unsteady one-dimensional homogeneous adiabatic, compressible fluid flows.

The required program input consists of
Numerical codes representing the flow network of the piping system, pipe dimensions, valve flow characteristics, valve operation characteristics, initial steam flow conditions in the piping system, and flow frictional coefficients.

The program output will generate the following:
Time values of flow pressure, density, velocity, nodal forces for all nodes, and segment forces for all segments of the flow network at each time increment.

Using either the NUPIPE-SW computer code (Ref 55) or a combination of SHOCK 3, PIPESTRESS, NKCODE and STRESSCOMBINER computer programs. **

(6)

Seismic analyses of Class I piping, which include all ASME Code Classes 1, 2, and 3 piping systems, are performed by the modal analysis response spectra method* Each piping system is idealized mathematically as an elastically coupled dynamic structural model in three-dimensional space. Inertial characteristics of the piping system are simulated by discrete masses of piping components, including all eccentric masses such as valves and valve operators, lumped at selecte nodes. The stiffness matrix of the piping system is calculated by Sto & Webster's computer program, PIPESTRESS.[43] Modal seismic responses at each node of the piping system, due to amplified response spectra excitation applied at its support points, are calculated by Stone & Webster's computer program, SHOCK3. The modal analysis technique used in SHOCK3 computes the peak response quantities for each mode. These quantities are then combined in Equation 3.7-32 of Section 3.7.3.1.2.4. Normal mode, linear elastic, and small displacement theory are incorporated in SHOCK3 and PIPESTRESS. For piping analyses which utilize the NUPIPE-SW program, the complete qualification/analysis is performed by NUPIPE-SW computer program. For other analyses, the following programs are used

Structural response spectra, consisting of peak responses of a family of seismic loadings for the piping systems, are the amplified response spectra, obtained for discrete locations in the structure where the piping system is supported. (See Section 3.7.2 for the development of the amplified response spectra.) Damping factors used for critical piping and components are 0.5% for the operating-basis earthquake and 1% for the design-basis earthquake. As an alternative, the following damping values given in the ASME Code Case N-411 may be used for both the operating-basis earthquake and the design-basis earthquaks. These values specifically are: five percent below frequency of 10 Hz; linear reduction from five percent to two percent between 10 Hz and 20 Hz and two percent above 20 Hz. These damping values are used in the following situations and the following additional considerations:

a. For seismic analyses in cases where new piping is added, existing systems are modified, existing systems are re-evaluated and for support optimization.

b. For seismic analyses using response spectrum methods and not for seismic analyses using time-history analyses methods.

** INSERT (A)

STONE & WEBSTER ENGINEERING CORPORATION

CALCULATION SHEET

5010 65

CALCULATION IDENTIFICATION NUMBER				
J.O. OR W.O. NO	DIVISION & GROUP	CALCULATION NO.	OPTIONAL TASK CODE	PAGE ____

Original plant pipe stress analyses utilized a combination of SHOCK 3, PIPESTRESS, NCCODE and STRESSCOMBINER computer programs as required. Certain subsequent pipe stress analyses have utilized the NUPIPE-SW computer program which can perform a complete piping analysis/qualification. A majority of the discussion in section 3.7.3.1.2 (Basic Steps and Equations Used in the Analytical Procedure) contains specific reference to analytical techniques used by the SHOCK 3, PIPESTRESS, NCCODE and STRESSCOMBINER computer programs. However, reference to code equations and damping values within these sections is applicable to all analyses regardless of which computer program is being utilized. A description of the NUPIPE-SW computer program is contained in section 3.7.3.1.2.5

section 3.7.3.1.1 (General Analytical Procedure) and

UFSAR INSERT A

c. When these alternate damping values are used, they are used in a given analysis in their entirety.

d. When these damping values are used together with changes in the support arrangement that increases the flexibility of piping systems, the predicted maximum displacements are reviewed to ensure that such displacements do not cause adverse interaction with adjacent structures, components or equipments.

e. When these damping values are used, the ±15% peak broadening criteria of Regulatory Guide 1.122, "Development of Floor Design Response Spectra for Seismic Design of Floor-Supported Equipment or Components," are used.

The uncertainties in the calculated values of fundamental structural frequencies due to expected variations in subgrade and structural material properties are taken into account. The peak resonant period value(s) in the amplified response spectra developed in Section 3.7.2 are subject to variations of 15% for Units 1 and 2 and its site. Accordingly, piping systems designed using amplified response spectra having modal periods within ±15% of the peak resonant period(s) are assigned the peak response value(s). Outside this range, the amplified response spectra are used exactly as stated.

Where a piping system is subjected to more than one amplified response spectrum, such as support points located in different parts of the structure, the amplified response spectrum closest to and higher in elevation than the center of mass of the piping system is applied to the system.

For a single-degree-of-freedom linear system oriented arbitrarily with respect to horizontal directions, the vectorial sum of the responses by considering both horizontal components of the spectra is an upper bound of the response of the system.[45] Since the seismic analysis of the piping system is based on eigensolutions of the system's dynamic structural model, this method is conservative in computing both piping seismic responses and seismic reactions on equipment and supports.

The SRSS method is an acceptable procedure if certain approximations in random vibration analysis for earthquake effects in the amplified response spectra are made.[46] Justifications of the applicable amplified response spectra in comparison to the time-history analysis of primary systems are presented in Appendix 3B.

The approach used in the SRSS method - both in Reference 46 and in the Stone & Webster analyses - is based on the assumption that an earthquake is a stationary random process, with no need for any special consideration of the spacing of the modes of the secondary systems.

Section 3.7.3.1.2.2 shows a comparison of responses obtained by the time-history method and the response spectrum method for structures. Examination of mode frequencies reveals that closely spaced frequencies occur at higher modes that have an insignificant contribution to the total response of the structure.

INSERT (B)

3.7.3.1.3 Analytical Procedure and Design Criteria

NUPIPE-SW or

3.7.3.1.3.1 ASME Code Class 1 Piping. The ASME Code Class 1 piping systems are analyzed using NCCODE, based on formulations and criteria specified in Subarticle NB-3650. Subarticle NB-3112.3(b) requires a number of earthquake cycles and seismic events used in the analyses of the ASME Code Class 1 components to be specified as part of the piping design criteria. The specifications are as follows:

1. A total of five operational-basis earthquake (OBE) (one-half safe-shutdown earthquake) and one design-basis earthquake (DBE)

CALCULATION IDENTIFICATION NUMBER				
J.O. OR W.O. NO.	DIVISION & GROUP	CALCULATION NO.	OPTIONAL TASK CODE	PAGE ____

UFSAR INSERT (B) page 1 of 4

3.7.3.1.2.5 Description of NUPIPE-SW Computer Program

The following is a brief description of the analytical procedure used in the NUPIPE-SW computer program. A detailed description of the subject computer program can be obtained in the NUPIPE-SW users manual which is on file at Stone and Webster Engineering Corporation offices.

3.7.3.1.2.5.1 General Description

The basic method of analysis used is the finite element stiffness method. The continuous piping is mathematically idealized as an assembly of elastic structural members (simple beam elements) connecting discrete nodal points. System loads and displacements such as deadweight, equivalent thermal forces, earthquake inertia forces, and anchor displacement are applied at the nodal points. Piping system restraints are represented by stiffness values (table 7.1).

Analysis for each type of load is performed individually, either statically or dynamically, and results superimposed on the results of other load analyses as required to meet stress requirements of the appropriate codes. Loadings such as pressure, thermal expansion, deadweight, and building and support point motions are typically evaluated by static analysis. Response of a piping system to seismic excitation and other occasional loads, such as flow-induced transients, is commonly determined utilizing dynamic methods of analysis.

(11)

5010 65

CALCULATION IDENTIFICATION NUMBER				
J.O. OR W.O. NO.	DIVISION & GROUP	CALCULATION NO.	OPTIONAL TASK CODE	PAGE ____

UFSAR INSERT (B) pg 2 of 4

3.7.3.1.2.5.2 Static Analysis

The static events to be considered in the design and analysis of a nuclear piping system include the loads resulting from deadweight, applied forces, thermal expansion, uniform acceleration and anchor movement conditions. For static loadings representing these conditions, the following equation is used:

F = Ku A-1

where F = The applied nodal force

K = The global stiffness matrix

u = The unknown displacement

The global stiffness matrix is formulated by adding the contributions of the element and support stiffnesses. Depending on the load case, NUPIPE may form appropriate types of static stiffness matrices.

The unknown nodal displacements are obtained in NUPIPE by solving the simultaneous equations resulting from equation A-1, using the Gauss method. These nodal displacements are then applied to the individual members, and the member stiffness used to determine the internal forces. The nodal displacements at support locations can be used along with the stiffnesses to determine support reactions.

(12)

A 5010 65

CALCULATION IDENTIFICATION NUMBER				
J.O. OR W.O. NO.	DIVISION & GROUP	CALCULATION NO.	OPTIONAL TASK CODE	PAGE ____

UFSAR INSERT (B) page 3 of 4

3.7.3.1.2.5.3 Dynamic Analysis

3.7.3.1.2.5.3.1 Response Spectra Method

The mathematical model utilized for static analysis is supplemented through addition of concentrated mass points at suitable locations (nodal points) to provide response to the particular type of dynamic loading being considered.

The eigenvalues (natural frequencies) and the eigenvectors (mode shapes) for each of the natural modes are calculated by solving the frequency equation. The natural mode shapes are then used to effect an orthogonal transformation of equation of equilibrium. This yields a series of independent equations of motion uncoupled in the system modes. The uncoupled equations are solved by either the step-by-step integration or the response spectrum method to obtain system response within each mode, and the individual modal results are combined to determine the total system dynamic response.

System response to seismic disturbance is obtained using the method of modal superposition. The inertia forces are calculated for each of the system natural modes and applied as static forces in the same manner as the weight or equivalent thermal forces in order to find internal forces and moments in each mode. system response is then obtained.

3.7.3.1.2.5.3.2 Time History with Modal Superposition

The stress analysis for dynamic forces resulting from safety or relief valve blowdown or steam and waterhammer analysis is generally performed by direct time history

13

A 5010.65

CALCULATION IDENTIFICATION NUMBER				
J.O. OR W.O. NO.	DIVISION & GROUP	CALCULATION NO.	OPTIONAL TASK CODE	PAGE ___

UFSAR INSERT B pg 4 of 4

integration using the same discrete beam element model described above. The equation of motion is written for each normal mode. It is solved directly in finite time increments for the generalized displacement. The total generalized displacement is transformed using the mode shape vector to form the system total displacement vector. This transformation is performed for each mode for each time step. These modal displacements are then applied to the system to determine internal forces, moments, and reactions for each mode for each time step. Finally, the modal responses at each time step are combined directly to form total response for each time step.

(14)

(safe-shutdown earthquake) seismic events will occur during the life of plant.

2. One hundred seismic stress cycles are imposed on the piping system during each operational-basis earthquake.

3.7.3.1.3.2 <u>ASME Code Class 2 and Class 3 Piping</u>. The ASME Code Class 2 and Class 3 piping systems are analyzed using NUPIPE-SW or STRESSCOMBINER, based on formulations specified in Subarticles NC-3600 and ND-3600. The seismic stresses are governed by the following allowables:

Pressure stress (S_{1p}) + dead load stress $(S_{dl}) \leq S_L$ (3.7-33)

Pressure stress + dead load stress + OBE stress $\leq 1.2\ S_h$ (3.7-34)

Pressure stress (S_{1p}) + dead load stress (S_{dl}) + DBE stress $\leq 1.8\ S_h$ (3.7-35)

Thermal stress $\leq (1.25\ S_c + 0.25\ S_h)f + (S_h - |S_{1p} + S_{dl}|)$ (3.7-36)

where

S_h = allowable stress of material at hot temperature (Tables I-7.1, I-7.2, I-8.1, and I-8.2 of ASME Code Section III)

S_c = allowable stress of material at cold temperature (Tables I-7.1, I-7.2, I-8.1, I-8.2 of ASME Code Section III)

f = stress range reduction factor for cyclic condition (Table NC-3611.1(b)(3)-1 of ASME Code Section III)

Equation 3.7-33 is based on Subarticle NC-3611.(c) for normal condition.

Equation 3.7-34 is based on Subarticles NC-3611.(c) and NC-3612.3, which state that seismic events of operational-basis earthquake for normal and upset conditions occur for less than 1% of the operating period. The stress limit is increased 20%.

Equation 3.7-35 is based on Subarticles NB-3652 and NB-3655, which state that a stress limit of 2.25 S_m for emergency condition (during DBE occurrence) is 1.5 times greater than the stress limit of 1.5 S_m for normal and upset conditions (during OBE occurrence). Based on the stress limit of 1.2 S_h for ASME Code Class 2 and Class 3 piping, in normal and upset conditions, the stress limit for emergency/faulted condition is thus derived to be 1.5 times 1.2 S_h, or 1.8 S_h.

Equation 3.7-36 is based on Subarticles NC-3611.(b)(3) and (b)(4).

All stress calculations for ASME Code Class 2 and Class 3 piping are based on equations given in Subarticle NC-3672.9, including bending and torsional effects. All inertial effects of eccentric mass, such as valve and valve operators connected to the piping system, are included in the dynamic structural model for the stress analysis.

Dynamic force loadings, resulting from sudden closure of an isolation valve or a turbine throttle valve on the piping system (for example, transient loading on steam line due to turbine trip), are to be included as occasional mechanical loads in piping analysis. Constraints or hydraulic snubbers are used as required to control excessive displacements or moments due to these transient loadings.

Field located seismic supports and constraints for Seismic Class I piping systems, including snubbers and dampers, are installed in accordance with seismically designed piping shown on approved construction drawings. Inspections are conducted to verify that these seismic restraints are fabricated and located in accordance with the construction plan and other applicable documents.

3.7.3.1.3.3 Buried Seismic Class I Piping. Responses of buried Seismic Class I piping to differential ground motion, due to particle motions caused

For the steam generator replacement effort at North Anna Unit 1, the STEHAM computer program was utilized to develop dynamic force loads resulting from a main steam line break.

Seismic analyses of typical Westinghouse-supplied Seismic Class I mechanical equipment, including heat exchangers, pumps, tanks, and valves were performed using a multi-degree-of-freedom modal analysis. Appendages, such as motors attached to motor-operated valves, are included in the models. The natural frequencies and normal modes are obtained using analytical techniques developed to solve eigenvalue-eigenvector problems. A response spectrum analysis is then performed based on the simultaneous occurrence of horizontal and vertical input motions. The response spectra are combined with the modal participation factors and the mode shapes to give the structural response for each mode from which the modal stresses are determined. The combined total seismic response is obtained by adding the individual modal responses using the square root of the sum of the squares method. Combined total response for closely spaced modal frequencies whose eigenvectors are perpendicular are handled in the above-described manner. In the rare event that two significantly closely-spaced in-phase modes occur, the combined total response is obtained by adding the square root of the sum of the squares of all other modes to the absolute value of one of the closely spaced modes.

Hydrodynamic analysis of tanks is performed using the methods described in Chapter 6 of the U. S. Atomic Energy Commission - TID-7024. Bridge and trolley structures are designed so that restraints prevent derailing due to the design-basis earthquake. The manipulator crane is designed to prevent disengagement of a fuel assembly from the gripper under the design-basis earthquake.

Components and supports of the reactor coolant system are designed for the loading combinations given in Section 5.2. These components are designed in complete accordance with the ASME Code, Section III, Nuclear Vessels, and the USAS B31.7 Code for Nuclear Power Piping. The allowable stress limits for these components and supports are also given in Section 5.2.

The loading combinations and stress limits for other components and supports are given in Section 3.9.

For the steam generator replacement effort at North Anna Unit 1, the STRUDL-SW computer program was utilized to qualify the steam generator lower support structure.

3.7 REFERENCES (continued)

52. Letter, NRC to Vepco, Serial #85-908, dated December 20, 1985.

53. ASME Boiler and Pressure Vessel Code, Section III, Division 1 Code Case N-411. Alternative Damping Values for Seismic Analysis of Classes 1, 2, and 3 Piping Sections, American Society of Mechanical Engineers, 345 E. 47th Street, New York, NY 10017, dated September 17, 1984.

54. NRC Regulatory Guide 1.122, Development of Floor Design Response Spectra for Seismic Design of Floor-Supported Equipment or Components.

55. NUPIPE-SW "Computer Code for Stress Analysis of Nuclear Piping", ME-110.

56. STRUDL-SW "Structural Design Language", ST-346.

57. STEHAM "Steamhammer Analysis for Piping Systems", ME-167U.

Unit 1 values represent margins associated with the ~~repaired~~ SGs repaired after cycle 9. Unit 2 values represent margins corresponding to the initial plant design.

Table 3.7-11

REPRESENTATIVE LISTING OF SEISMIC DESIGN MARGINS[a]
NORTH ANNA 1 AND 2
STONE & WEBSTER SCOPE OF SUPPLY
(MECHANICAL ITEMS)

System/Component	Total Loading Combination	Max Seismic Stress, psi OBE	Max Seismic Stress, psi DBE	Total Max Combined Stress, psi OBE	Total Max Combined Stress, psi DBE	Allowable Stress, psi OBE	Allowable Stress, psi DBE	Design Margin[a b] OBE	Design Margin[a b] DBE
Reactor Coolant system									
Steam-generator supports Unit 1	Normal[b c] +SRSS(EQ& pipe rupture)	---	~~15,900~~ 5,790	---	~~27,830~~ 16,880	---	~~32,550~~ 17,000	---	~~1.17~~ 1.01
Steam-generator supports Unit 2	DL+EQ+Faulted (pipe rupture)	8,600	8,600	25,400	25,400	64,810	32,400	2.55	1.28
Reactor coolant pump supports Unit 1	Normal[b c] +SRSS(EQ& pipe rupture)	---	~~15,640~~ 53,160	---	~~16,910~~ 57,720	---	~~30,700~~ 70,200	---	~~1.8[illegible]~~ 1.21
Reactor coolant pump supports Unit 2	DL+EQ+Faulted (pipe rupture)	13,000	13,000	33,700	33,700	64,480	34,240	2.03	1.02
Recirculation spray system									
Recirculation spray heat exchanger	DL+EQ+Oper	12,400	15,467	25,33[illegible]	28,150	25,380	36,720	1.01	1.30
Recirculation spray heat exchanger seismic restraint (main support structure)	DL+EQ+Oper	2,190	3,586	10,520	11,950	35,000	35,000	3.33	2.93
Component cooling system									
Component cooling pumps	DL+EQ+Oper	2,540	3,390	21,750	20,380	32,400	32,400	1.49	1.59
Compressed air system									
Main instrument air receivers	DL+EQ	485	833	485	833	5,047	9,270	10.40	11.10

a

b ~~a~~ Design Margin = $\frac{\text{allowable stress}}{\text{total max. combined stress}}$

c ~~b~~ Normal = deadweight + thermal + internal pressure

THESE DATA TO BE ADDED TO EXISTING TABLE 3.7-11

(5)

Table 3.7-12

REPRESENTATIVE LISTING OF SEISMIC DESIGN MARGINS
FOR NORTH ANNA UNITS 1 AND 2
STONE & WEBSTER SCOPE OF SUPPLY
(STRUCTURAL ITEMS)

Component	Failure Mode Controlling Design Margin at Interface of Component/Structure	Criteria for Allowable Capacity (FA)	Criteria Reference	Maximum Reaction, F	Design Margin, FA/F	Seismic Event	Estimated % of Reaction Due to Seismic Event
Steam-generator supports	Punching shear in concrete	FA = 4 f'c b_od	ACI 318-63, Section 1707 (c)	2,173*	1.37	DBE	49
Reactor coolant pump supports	punching shear in concrete	FA = 4 f'c b_od	ACI 318-63, Section 1707 (c)	1,895*	1.11	DBE	39
Safety injection accumulator supports	Combined shear and tension on ASTM A307, Gr. A bolts	FA = 1.33 (28.0-1.6 fv) 27.0	AISC 69 Spec., Section 1.6.3 plus 1/3 increase	20.64 ksi tens. 3.59 ksi shear	1.31	DBE	46
Refueling water storage tank	Tank sliding on concrete foundation	FA = 1.1 (sliding force)	SRP 3.8.5 Section II,3,c	701 kips	1.54	DBE	100
Quench spray pump anchor	Combined shear and tension on ASTM A307, Gr. A bolts	FA = 1.33 (28.0-1.6 fv) 27.0	AISC 69 Spec., Section 1.6.3 plus 1/3 increase	1.83 ksi tens. .91 ksi shear	14.75	DBE	100
Residual heat exchanger supports							
Main	Tension on ASTM A307, Gr. A bolts	FA = 0.9 F_y	Section 3.8.1.3	25.44 K	1.14	DBE	75

Note: The table addresses the concerns of the USNRC as they pertain to the anchorage of safety related components to the building structure. The computed seismic design margins are based on the allowable capacity for a local failure at the component/structure interface. the controlling failure mode is identified for that interface as well as the criteria for defining allowable capacities. Maximum reactions are given in terms of force or stress, depending on what appeared to be more meaningful for the stated failure mode. Loads from the operational-basis earthquake generally do not control and, thus, were omitted from the presentation. the estimated percentage of the reaction which is due to the design-basis earthquake is provided.

*The Unit 2 loads are presented, the Unit 1 loads are less.

THESE DATA TO BE ADDED TO EXISTING TABLE 3.7-12

7

The criterion presented in the ASME Code is used for the fatigue failure analysis. The cumulative usage factor is less than 1.0 and hence the fatigue design is adequate.

The reactor vessel vendor's stress report is reviewed by Westinghouse. The stress report includes a summary of the stress analysis for regions of discontinuity analyzed in the vessel, a discussion of the results including a comparison with the corresponding code limits, a statement of the assumptions used in the analyses, descriptions of the methods of analysis and computer programs used, a presentation of the actual calculations used, a listing of the input and output of the computer programs used, and a tabulation of the references cited in the report. The content of the stress report is in accordance with the requirements of the ASME Code.

The Westinghouse analysis of the steam-generator tube-tubesheet complex is included as part of the stress report requirement for ASME Code Class A nuclear pressure vessels. The evaluation is based on the stress and fatigue limitations outlined in ASME Code, Section III.

The stress analysis techniques used include all factors considered appropriate to conservative determination of the stress levels used in evaluation of the tube-tubesheet complex. The analysis of the tube-tubesheet complex includes the effect of all appurtenances attached to the perforated region of the tubesheet that are considered appropriate for conservative analysis of the stresses for evaluation on the basis of the ASME Code, Section III, stress limitations. The evaluation involves the heat conduction and stress analysis of the tubesheet, channel head, and secondary-shell structure for particular steady design conditions for which code stress limitations are to be satisfied and for discrete points during transient operation for which the temperature/pressure conditions must be known to evaluate stress maximum and minimum for fatigue life. In addition, ~~limit~~ analyses are performed to determine tubesheet capability to sustain faulted conditions. ~~for which elastic analysis does not suffice~~. The analytic techniques used are computerized. ~~and significant stress problems are verified experimentally to justify the techniques where possible.~~

The major concern in fatigue evaluation of the tube weld is the fatigue strength reduction factor to be assigned to the weld root notch. For this reason, Westinghouse has conducted low-cycle fatigue tests of tube material samples to determine the fatigue strength reduction factor and applied them to the analytic interaction analysis results in accordance with the accepted techniques in the ASME Code for experimental stress analysis. For the Unit 1 Replacement Steam Generators, a fatigue strength reduction factor of 4.0 is assigned to the weld root notch in the fatigue evaluation of the tube weld.

The steam-generator tube-tubesheet complex integrity is verified by analysis for most adverse conditions resulting from a rupture of either primary or secondary piping.

It has been established that for such accident conditions, where a primary-to-secondary-side differential pressure exists, the primary membrane stresses in the tubesheet ligaments, averaged across the ligament and through the tubesheet thickness, satisfy the conditions given in Table ~~5.2-4~~ 5.2-8 and 5.2-11 for Unit 1 and Table 5.2-4 for Unit 2 for this faulted event. Also, for such accident conditions, the primary membrane stress plus primary bending stress in the tubesheet ligaments, averaged across the ligament width at the tubesheet surface location giving maximum stress, must not exceed the faulted condition criteria. In the case of a primary pressure-loss accident, the secondary-primary pressure differential is somewhat higher than the primary-secondary design pressure differential. However, rigorous analysis shows that no stresses in excess of those covered by the ASME Code for faulted conditions are experienced by the tubesheet for this accident.

Tables 5.2-8 and 5.2-11 for Unit 1 and Table 5.2-4 for Unit 2 summarizes the tubesheet stress results for a pressure differential of 2485 psig. Tabulations of other primary/secondary pressures significant results of the tubesheet complex are in Tables 5.2-5 through 5.2-12 and Figures 5.2-1A, B through 5.2-4. Figures 5.2-2, 3 and 4 apply to the Unit 2 results only.

INSERT 18A →

The tubes for Unit 2 have been designed to the requirements of the ASME Code, assuming 2485 psig as the design pressure differential. Hence, neither a primary nor a secondary pressure-loss accident imposes stresses beyond that normally expected and considered as normal operation by the Code. ASME, Section VIII, design curves for iron-chromium-nickel steel cylinders under external pressure indicate a collapse pressure of 2310 psi for tubes having

Insert 18A

The tubes in the Unit 1 replacement steam generators have been designed to the requirements of the ASME code, assuming 1600 psi as the design pressure differential. None of the normal and upset conditions impose stresses beyond that normally expected and considered as normal operation by the code.

A tube analysis for Unit 1 for the external pressures showed the collapse pressure of 2389 psi in the straight leg of the tube and 1944 psi in the U-bend region of the tube considering the thinnest tube and including the wear/corrosion allowances for a faulted condition considering the minimum strength properties required by the ASME code, Section III. This provides a calculated minimum factor of safety against collapse of 1.88.

in the calculation of the equivalent plate stresses, the use of the stress concentration factors around the pertinent holes, the calculate of the various alternating stresses, and the use of the ASME Code, Section III, fatigue curves.

In the limit stress analysis performed for the tube-tubesheet complex, the deformations and displacements induced at the location of the steam-generator supports are negligible. Channel head deflections due to the limit analysis of the tubesheet occur only in the region of the tubesheet and the tubesheet to channel head junction and are not appreciable at the location of the supports. The support feet are approximately 3 ft away from the tubesheet. Since support deflections due to the tubesheet analysis are negligible, the system analysis will not be affected.

The vessels of the reactor coolant pressure boundary are designated ASME Code, Section III, 1968, including addenda up to Winter 1968, Class A. Piping, pumps, and valves are designated to USAS B31.7, 1969, except as noted in Section 5.2.

Loading combination and allowable stresses for ASME Code, Section III, 1968, Winter 1968, Class A components and piping are given in Tables 5.2-13 through 5.2-16. Design criteria for supports are given in Section 5.5.9.

When the components and systems for the North Anna units were being designed, only general design requirements existed for faulted conditions. There were no specific stress limits or associated methods of analysis established for faulted conditions. To provide a conservative basis for the analysis of Class 1 components, the collapse curves given in the PSAR were developed. The criteria represented by the collapse curves have evolved into the criteria of Table 5.2-16. The methods and criteria in Table 5.2-16 should thus be reviewed with respect to the criterion agreed to in the PSAR, rather than with the more recently derived methods and limits established in the nonmandatory Appendix F of the ASME Code, Section III. These methods of analysis in conjunction with the faulted condition stress limits ensure that the general design requirements of the ~~AEC~~ NRC for faulted conditions will be met and the plant can thus be safely shut down under accident conditions.

After the material was heat treated, the material was not heated above 800°F during subsequent fabrication except as described in Section 5.2.3.1.9 and the paragraphs below.

Methods and material techniques that were used to avoid partial or local severe sensitization are as follows:

1. Nozzle Safe Ends

   a. For the pressurizer, weld deposit with Inconel (Ni-Cr-Fe weld metal F No. 43) was used, followed by the final post-weld heat treatment.

   (b. For the Unit 1 steam generator, a SA336-CLF 316LN forging welded to Inconel buildup on the nozzle was used.)

   c. For the Unit 2 steam generator and the reactor vessel, a stainless steel weld metal analysis A-7 containing more than 5% ferrite was used.

2. All welding was conducted using those procedures that have been approved by the ASME Code, Sections III and IX.

3. All welding procedures were qualified by nondestructive and destructive testing according to the ASME Code, Sections III and IX.

   When these welding procedure tests were performed on test welds that were made from base metal and weld metal materials that were from the same lot(s) of materials used in the fabrication of components, additional testing was frequently required to determine the metallurgical, chemical, physical, corrosion, etc., characteristics of the weldment. The additional tests that were conducted on a technical case basis were as follows: light and electron microscopy, elevated temperature mechanical properties, chemical check analysis, fatigue tests, intergranular corrosion tests, and static and dynamic corrosion tests within reactor water chemistry limitations.

4. The following welding methods were tested individually and in multiprocess combinations as outlined above using these prudent

Table 5.2-4
(Unit 2 Only)
STRESSES DUE TO MAXIMUM STEAM-GENERATOR TUBESHEET PRESSURE DIFFERENTIAL (2485 psig)[a]

Stress	(600°F) Computed Value	Allowable Value
Primary membrane stress	24,356 psi	37,000 psi ($0.9\ S_y$)
Primary membrane plus primary bending stress	54,946 psi	55,600 psi ($1.35\ S_y$)

[a] In addition to the foregoing evaluation, elasto-plastic limit analysis of the tubesheet-head-shell combination indicates a limit pressure of 3000 psi at operating temperature.

Table 5.2-5 B
(Unit 1)

STEAM GENERATOR PRIMARY/SECONDARY BOUNDARY COMPONENTS

Condition: Design Condition

Primary/Secondary Pressures = 2485/885 psig(1)
Primary Chamber Design Temperature 650°F
Secondary Chamber Design Temperature 600°F

Location (Figure 5.2-1B)	Description	Stress Category	Maximum Stress Intensity (ksi)	Allowable Stress Limit (ksi)
2	Channel Head to Tubesheet	PL	17.76	40.05
	Junction, in the Channel Head	PL+Pb	43.13(2)	40.05
3	Channel Head to Tubesheet	PL	16.82	45.00
	Junction, in the Tubesheet	PL+Pb	48.10(2)	45.00
4	Tubesheet to Stub Barrel	PL	15.06	40.05
	Junction	PL+Pb	15.41	40.05

Notes: (1) Based on 1600 psig Primary to Secondary Design Pressure Differential

(2) Exceeds the allowable stress limit. A limit analysis was performed per N-417.6 (b) of the ASME Code Section III.

Specified Primary Pressure = 2485 psig
2/3 Lower Bound Collapse Load (Primary Pressure) = 3390 psig

Secondary Pressure = 885 psi

Table 5.2-5 B
(Part 2)
STEAM-GENERATOR PRIMARY-SECONDARY BOUNDARY COMPONENTS

Condition: 100% load operation, 2485/885 psig[a] 650/600°F
normal operation stress limits.

Location (Figure 5.2-1B)	Description	Inside Limit Center Limit Outer Limit	Stress Limit Center Limit Stress Limit	Inside Surface Stress Center Surface Stress Outer Surface Stress
7	Jct. of short	$3S_m$	80,100	-10,063 psi
	cyl. with	S_m	26,700	- 8,597 psi
	tubesheet	$3S_m$	80,100	+27,247 psi
8	1/2 through	$3S_m$	80,100	+ 8,914 psi
	short cyl.	S_m	26,700	+ 8,597 psi
	discontinuity	$3S_m$	80,100	+ 7,670 psi
9	Jct. of short	$3S_m$	80,100	+10,740 psi
	cyl. with	S_m	26,700	+ 8,597 psi
	shell	$3S_m$	80,100	6,443 psi
10	On shell	$3S_m$	80,100	+10,269 psi
		S_m	26,700	8,597 psi
		$3S_m$	80,100	+ 6,912 psi

[a]Based on 1600 psig design pressure differential.

Table 5.2-5B (continued)

STEAM-GENERATOR PRIMARY-SECONDARY BOUNDARY COMPONENTS

Condition: 100% load operation, 2485/885 psig[a] 650/600°F
normal operation stress limits.

Location (Figure 5.2-1B)	Description	Inside Limit Center Limit Outer Limit	Stress Limit Center Limit Stress Limit	Inside Surface Stress Center Surface Stress Outer Surface Stress
11	On shell	$3S_m$	80,100	+ 9,746 psi
		S_m	26,700	+ 8,597 psi
		$3S_m$	80,100	+ 7,435 psi
12	Jct. of primary	$3S_m$	80,100	+58,701 psi
	short cyl. with	S_m	26,700	+14,528 psi
	tube plate	$3S_m$	80,100	-29,646 psi
13	1/2 through	$3S_m$	80,100	+50,836 psi
	primary short	S_m	26,700	+14,528 psi
	cyl.	$3S_m$	80,100	-21,781 psi
	discontinuity			
14	Jct. of primary	$3S_m$	52,200	42,286 psi
	short cyl.	S_m	19,400	+14,528 psi
	with head	$3S_m$	52,200	-13,231 psi

[a]Based on 1600 psig design pressure differential.

Table 5.2-6 A
(Unit 1)

STEAM GENERATOR PRIMARY/SECONDARY BOUNDARY COMPONENTS

Condition: Primary Hydrotest

Primary Chamber Hydrotest Pressure 3107 psig
Secondary Chamber Hydrotest Pressure 0 psig
Test Temperature 70-250°F

Location (Figure 5.2-1A)	Description	Stress Category	Maximum Stress Intensity (ksi)	Allowable Stress Limit (ksi)
2	Channel Head to Tubesheet	PL	27.41	62.47
	Junction, in the Channel Head	PL+Pb	67.21(1)	62.47
3	Channel Head to Tubesheet	PL	28.03	82.42
	Junction, in the Tubesheet	PL+Pb	75.82	82.42
4	Tubesheet to Stub Barrel	PL	22.49	62.47
	Junction	PL+Pb	42.50	62.47

Notes: (1) Exceeds the allowable stress limit. A limit analysis was performed per N-417.6 (b) of the ASME Code Section III.

Specified Primary Pressure = 3107 psig
2/3 Lower Bound Collapse Load (Primary Pressure) = 3400 psig

Secondary Pressure = 0 psi

Table 5.2-6 B
(Unit 2)

STEAM-GENERATOR PRIMARY-SECONDARY BOUNDARY COMPONENTS

Condition: Primary hydrotest, 3107/0 psig.

Location (Figure 5.2-1B)	Description	Code Limit	Primary Membrane Stress Limit	Axial Primary Membrane Stress Intensity
7	Jct. of short cyl. with tubesheet	$0.9\ S_y$	45,[illegible]	0 psi
8	1/2 through short cyl. discontinuity	$0.9\ S_y$	45,000	0 psi
9	Jct. of short cyl. with shell	$0.9\ S_y$	45,000	0 psi
10	On shell	$0.9\ S_y$	45,000	0 psi
11	On shell	$0.9\ S_y$	45,000	0 psi
12	Jct. of primary short cyl. with tube plate	$0.9\ S_y$		18,158 [illegible]
13	1/2 through primary short cyl. discontinuity	$0.9\ S_y$	45,000	18,158 psi
14	Jct. of primary short cyl. with head	$0.9\ S_y$	36,000	18,158 psi

Table 5.2-7 A
(Unit 1)

STEAM GENERATOR PRIMARY/SECONDARY BOUNDARY COMPONENTS

Condition: Secondary Hydrotest

Secondary Chamber Hydrotest Pressure 1357 psig
Primary Chamber Hydrotest Pressure 0 psig
Test Temperature 70-180°F

Location (Figure 5.2-1A)	Description	Stress Category	Maximum Stress Intensity (ksi)	Allowable Stress Limit (ksi)
2	Channel Head to Tubesheet	PL	9.20	62.47
	Junction, in the Channel Head	PL+Pb	18.66	62.47
3	Channel Head to Tubesheet	PL	10.00	82.42
	Junction, in the Tubesheet	PL+Pb	21.63	82.42
4	Tubesheet to Stub Barrel	PL	16.53	62.47
	Junction	PL+Pb	49.12	62.47

Table 5.2-7 B
(Unit 2)
STEAM-GENERATOR PRIMARY-SECONDARY BOUNDARY COMPONENTS

Condition: Secondary chamber hydrotest, 0/1356 psig.

Location (Figure 5.2-1B)	Description	Code Limit	Primary Membrane Stress Limit	Axial Primary Membrane Stress Intensity
7	Jct. of short cyl. with tubesheet	0.9 S_y	45,000	13,169 psi
8	1/2 through short cyl. discontinuity	0.9 S_y	45,000	13,169 psi
9	Jct. of short cyl. with shell	0.9 S_y	45,000	13,169 psi
10	On shell	0.9 S_y	45,000	13,169 psi
11	On shell	0.9 S_y	45,000	13,169 psi
12	Jct. of primary short cyl. with tube plate	0.9 S_y	45,000	0 psi
13	1/2 through primary short cyl. discontinuity	0.9 S_y	45,000	0 psi
14	Jct. of primary short cyl. with head	0.9 S_y	36,000	0 psi

Table 5.2-8A
(Unit 1)

STEAM GENERATOR PRIMARY/SECONDARY BOUNDARY COMPONENTS

Condition: Faulted Condition, Loss of Secondary Side Pressure

Primary Chamber Pressure 2485 psig
Secondary Chamber Pressure 0 psig
Temperature 668°F

Location (Figure 5.2-1A)	Description	Stress Category	Maximum Stress Intensity (ksi)	Allowable Stress Limit (ksi)
2	Channel Head to Tubesheet	PL	21.92	36.94
	Junction, in the Channel Head	PL+Pb	53.76	55.41
3	Channel Head to Tubesheet	PL	22.42	50.72
	Junction, in the Tubesheet	PL+Pb	60.64	76.08
4	Tubesheet to Stub Barrel	PL	17.99	36.94
	Junction	PL+Pb	33.99	55.41

Table 5.2-8B
(Unit 2)
STEAM-GENERATOR PRIMARY-SECONDARY BOUNDARY COMPONENTS

Condition: Loss of secondary pressure (steam-line break), faulted condition 2485/0 psig, 660°F.

Location (Figure 5.2-1B)	Description	Primary Membrane Stress Emergency Condition Limits Code Limit	Primary Membrane Stress Emergency Condition Limits Stress	Primary Membrane Stress
7	Jct. of short cyl. with tubesheet[a]	S_y	41,112	0 psi
8	1/2 through short cyl. discontinuity	S_y	41,112	0 psi
9	Jct. of short cyl. with shell	S_y	41,112	0 psi
10	On shell	S_y	41,112	0 psi
11	On shell	S_y	41,112	0 psi
12	Jct. of primary short cyl. with tube plate	S_y	41,112	14,528 psi
13	1/2 through primary short cyl. discontinuity	S_y	41,112	14,528 psi
14	Jct. of primary short cyl. with head	S_y	29,000	14,528 psi

[a] Complete tubesheet structure complex also evaluated on limit analysis basis.

Table 5.2-9 A
(Unit 1)

STEAM GENERATOR PRIMARY/SECONDARY BOUNDARY COMPONENTS

Condition: Normal, Upset and Test Conditions

Location (Figure 5.2-1A)	Description	Maximum Fatigue Usage
2	Channel Head to Tubesheet Junction, in the Channel Head	0.03
3	Channel Head to Tubesheet Junction, in the Tubesheet	0.13
4	Tubesheet to Stub Barrel Junction	0.06

Table 5.2-10A
(Unit 1)

STEAM GENERATOR PRIMARY/SECONDARY BOUNDARY COMPONENTS

Condition: Normal, Upset and Test Conditions

Location (Figure 5.2-1A)	Description	Maximum Fatigue Usage
1	Tubesheet Center	0.16

Table 5.2-9B
(Part 2)

51,500-FT^2 STEAM GENERATOR USAGE FACTORS, INDIVIDUAL TRANSIENTS, PRIMARY AND SECONDARY BOUNDARY COMPONENTS

Transient	Number of Cycles	In Tubesheet Inlet SALR*	In Tubesheet Inlet SALR	In Tubesheet Inlet SASR	In Tubesheet Outlet SALR	In Tubesheet Outlet SASR	In Tubesheet Outlet SRSR	On Tubesheet Inlet SALR	On Tubesheet Inlet SALR	On Tubesheet Inlet SASR	On Tubesheet Outlet SALR	On Tubesheet Outlet SASR	On Tubesheet Outlet SRSR	Inhead Inlet SALR	Inhead Inlet SALR	Inhead Inlet SASR	Inhead Outlet SALR	Inhead Outlet SASR	Inhead Outlet SRSR	Unperforated Outer Ring Primary Inlet	Unperforated Outer Ring Primary Outlet	Unperforated Outer Ring Secondary Inlet	Unperforated Outer Ring Secondary Outlet	Junction of Shell to Tubesheet MAPR	Junction of Shell to Tubesheet SANR	Junction of Shell to Tubesheet MRSR
Heatup-cooldown	200	0.008	0.01	0	0.008	0.01	0	0	0	0	0	0	0	0.009	0.003	0	0.009	0.003	0	0	0	0	0	0	0	0
Loading-unloading	18,300	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0.009	0.007	0.016	0
Small step increase	2,000	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
Small step decrease	2,000	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
Large step decrease	200	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0.007	0.001	0
Loss of load	80	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0.001	0.004	0	0	0	0.001	0.002	0.006	0.003	0
Loss of power	40	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
Loss of flow	80	0.008	0.009	0	0.008	0.016	0	0	0	0	0	0	0	0.013	0.003	0.001	0.012	0.006	0	0	0	0.002	0.006	0.003	0.002	0
Reactor trip	400	0	0	0	0	0.003	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0.001	0.005	0.012	0.008	0
Reactor coolant pipe break	1	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
Steam line break	1	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0.001	0
Primary hydrotest	5	0.004	0.007	0	0.004	0.007	0	0	0	0	0	0	0	0.005	0.002	0	0.005	0.002	0	0	[illegible]	0.001	0.001	0.001	0.001	0
Secondary hydrotest	5	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0.001	0.002	0
Turbine roll test	10	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0

*Principal stress difference codes.

Table 5.2-10 B
(Cont 2)
51,500-FT^2 STEAM GENERATOR USAGE FACTORS, INDIVIDUAL TRANSIENTS, CENTER OF TUBESHEET

Transient	Number of Cycles	Primary Inlet Angle[a] (degrees)							Primary Outlet Angle[a] (degrees)							Secondary Inlet Angle[a] (degrees)							Secondary Outlet Angle[a] (degrees)						
		0°	15°	30°	45°	60°	75°	90°	0°	15°	30°	45°	60°	75°	90°	0°	15°	30°	45°	60°	75°	90°	0°	15°	30°	45°	60°	75°	90°
Heatup-cooldown	200	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
Loading-unloading	18,300	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
Small step increase	2,000	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
Small step decrease	1,000	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
Large step decrease	200	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
Loss of load	80	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
Loss of power	40	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
Loss of flow	80	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
Reactor trip	400	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
Reactor coolant pipe break	1	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
Steam-line break	1	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
Primary hydrotest	5	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
Secondary hydrotest	5	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
Turbine roll test	10	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0

[a] Angular location around perforation.

TABLE 5.2-11 A
(Unit 1)
STEAM GENERATOR PRIMARY/SECONDARY BOUNDARY COMPONENTS

Location: Tubesheet Center

Condition Description	Stress Category	Maximum Stress Intensity (ksi)	Allowable Stress Limit (ksi)
Design Condition, 2485/885 psig, 650/600°F	Pm	9.92	30.00
	P_L+P_b	46.44(1)	45.00
Primary Hydrotest, 3107/0 psig, 70-250°F	P_m	10.28	54.95
	P_L+P_b	81.04	82.42
Secondary Hydrotest, 0/1357 psig, 70-180°F	P_m	2.64	54.95
	P_L+P_b	33.44	82.42
Faulted Condition (Loss of Secondary Pressure)	P_m	8.22	50.72
	P_L+P_b	64.82	76.08

Notes: (1) Exceeds the allowable stress limit. A limit analysis was performed per N-417.6 (b) of the ASME Code Section III.

Specified Primary Pressure = 2485 psig
2/3 Lower Bound Collapse Load (Primary Pressure) = 3390 psig
Secondary Pressure = 885 psi

Table 5.2-11 B
(Unit 2)

TUBESHEET STRESS ANALYSIS RESULTS

FOR 51,500-FT^2 STEAM GENERATORS

Conditions	Maximum Primary Membrane Plus Primary Bending Average Ligament Stress (psi)	Maximum Effective Ligament Membrane Stress (psi)
100% normal operation, 2485/885 psi, 650/600°F	33,979 (40,050)[a]	15,653 (26,700)[b]
Primary hydrotest, 3107/0 psi, 100°F	67,300 (67,500)[c]	30,355 (45,000)[d]
Secondary hydrotest, 0/1356 psi, 100°F	29,811 (67,500)[c]	13,159 (45,000)[d]
Steam-line break (fault condition), 2485/0 psi, 560°F	56,785 (limit)[e]	24,356 (limit)[e]

Parentheses indicate code allowable stress:

[a] 1.5 S_m

[b] 1.0 S_m

[c] 1.35 S_y

[d] 0.9 S_y

[e] Limit analysis results apply.

TABLE 5.2-12A
(Unit 1)
LIMIT ANALYSIS CALCULATIONS RESULTS

Location	Case	Limit Pressure (psig)
Channel Head to Tubesheet Junction, in the Channel Head	Design Condition Primary/Secondary Pressure 2485/885 psig	3390
	Primary Hydrotest Primary/Secondary Pressure 3107/0 psig	3400
Channel Head to Tubesheet Junction, in the Tubesheet	Design Condition Primary/Secondary Pressure 2485/885 psig	3390
Tubesheet Center	Design Condition Primary/Secondary Pressure 2485/885 psig	3390

Table 5.2-12 B
(Unit 2)

LIMIT ANALYSIS CALCULATION RESULTS: STRAINS, LIMIT PRESSURES, AND FATIGUE EVALUATIONS FOR 51,500-FT^2 STEAM GENERATORS

Case	Location	Meridional Strain (in./in.)	Circumferential Strain (in./in.)	Peak Stress Intensity (psi)	Allowable Number of Cycles, N_1	Number of Cycles, N_2	Usage Factor, N_2/N_1	Limit Pressure (psi)
Hot 2500/0 psi, 650°F	Channel/primary shell	0.0188	-0.000559	508,000	46	10	0.22	
	Tubesheet/ secondary shell	-0.00193	0.00602	83,700	5,000	10	0.0020	3158
	Tubesheet center	0.00159	0.00159	77,400	6,600	10	0.0015	
Cold hydro 3105/0 psi, 70°F	Tubesheet/ primary shell	0.0145	-0.000537	434,000	80	5	0.053	
	Tubesheet/ secondary shell	-0.00220	0.000684	106,000	3,500	5	0.0014	3887
	Tubesheet center	0.00177	0.00177	95,400	5,000	5	0.0010	
Cold hydro with secondary pressure 3105/700 psi, 70°F	Tubesheet/ primary shell	0.00730	-0.000348	218,000	500	5	0.010	
	Tubesheet/ secondary shell	-0.000962	0.000560	50,700	40,000	5	0.0001	4401
	Tubesheet center	0.00147	0.00147	79,000	8,000	5	0.0005	
Hot hydro 2485/0 psi, 400°F	Tubesheet/ primary shell	0.00777	-0.000407	222,000	400	50	0.13	
	Tubesheet/ secondary shell	-0.00176	0.000551	80,900	7,000	50	0.0071	3354
	Tubesheet center	0.00148	0.00148	76,300	8,500	50	0.0059	

Table 5.2-14

LOADING CONDITIONS AND STRESS LIMITS: CLASS A COMPONENTS

Loading Conditions	Stress Intensity Limits	Note
Design ~~Normal~~ condition	(a) $P_m \leq S_m$	
	(b) $P_L \leq 1.5S_m$	
	(c) P_m (or P_L) + $P_B \leq 1.5S_m$	1
	~~(d) P_m (or P_L) + P_B + $Q \leq 3.0S_m$~~	~~2~~
Normal and Upset condition	~~(a) $P_m \leq S_m$~~	
	~~(b) $P_L \leq 1.5S_m$~~	
	~~(c) P_m (or P_L) + $P_B \leq 1.5S_m$~~	
	(a) ~~(d)~~ P_m (or P_L) + P_B + $Q \leq 3.0S_m$	2
	(b) Cummulative Fatigue Usage < 1.0	
Faulted condition	(a) $P_m \leq 1.2S_m$ or S_y, whichever is larger, $P_L \leq 1.5(1.2)\ S_m$ or $1.5S_y$, whichever is larger, and P_m (or P_L) + $P_B \leq 1.5(1.2)\ S_m$ or $1.5S_y$, whichever is larger, or	3
	(b) Faulted condition limits in Table 5.2-16	

Key: P_m = primary general membrane stress intensity

P_L = primary local membrane stress intensity

P_B = primary bending stress intensity

Q = secondary stress intensity

S_m = stress intensity value from ASME Code, Section III

S_y = minimum specified material yield from ASME Code, Section III, Table N-421 or equivalent

Notes: 1. The limits on local membrane stress intensity ($P_L \leq 1.5S_m$) and primary membrane plus primary bending stress intensity (P_m (or P_L) + $P_B \leq 1.5S_m$) need not be satisfied at a specific location if it can be shown by means of limit analysis or by tests that the specified loadings do not exceed two-thirds of the lower bound collapse load per paragraph N-417.6(b) of the ASME Code, Section III.

Table 5.2-17A
Unit 1

FAULTED CONDITION LOADS FOR THE REACTOR COOLANT PUMP FEET

	F_x	F_y	F_z	M_x[a]	M_y[a]	M_z[a]
W Umbrella	±3305	±3400	±2605	±7059	±4010	±7083
Case 1[b]	-	390	294	-	-	-
Case 2[b]	-	521	330	-	-	-
Case 3[b]	-	959	376	-	-	-
Case 4[b]	-	527	326	-	-	-

[a]No moments are transmitted to the pump feet due to the specific design of the supports.

[b]These four cases represent the largest loading conditions on pump feet.

Case 1: Deadweight + Thermal + Internal Pressure + SRSS (SSE & main steam line break)
Case 2: Deadweight + Thermal + Internal Pressure + SRSS (SSE & Residual heat removal line break)
Case 3: Deadweight + Thermal + Internal Pressure + SRSS (SSE & accumulator/safety injection line break)
Case 4: Deadweight + Thermal + Internal Pressure + SRSS (SSE & pressurizer surge line break)

REPLACE EXISTING TABLE 5.2-17 with TABLES 5.2-17 A & B

Table 5.2-17B
Unit 2

FAULTED CONDITION LOADS FOR THE REACTOR COOLANT PUMP FEET

	F_x	F_y	F_z	M_x[a]	M_y[a]	M_z[a]
W Umbrella	±3305	±3400	±2605	±7059	±4010	±7083
Case 1[b]	331	1633	701	-	-	-
Case 2[b]	708	1507	2029	-	-	-
Case 3[b]	506	2919	1326	-	-	-
Case 4[b]	608	3055	1445	-	-	-

[a]No moments are transmitted to the pump feet due to the specific design of the supports.

[b]These four cases represent the largest loading conditions on pump feet.

Case 1: RCPINB + SSE + Deadweight + Pressure
Case 2: RCVINB + SSE + Deadweight + Pressure
Case 3: LCWB + SSE + Deadweight + Pressure
Case 4: SGONB + SSE + Deadweight + Pressure

REPLACE EXISTING
TABLE 5.2-17 WITH
TABLES 5.2-17A & B

Table 5.2-22
REACTOR COOLANT PRESSURE BOUNDARY MATERIALS

Component	Type
Reactor vessel components:	
Shell and head plates (other than core region)	SA 533 Grade A, B, or C; Class 1 or 2 (vacuum treated)
Shell, flange and nozzle forgings nozzle safe ends	SA 508 Class 2 or 3 SA 182 Type F304 or F316 weld buildup
CRDM appurtenances - upper head	SB 166 or 167 and SA 182 Type F304
Instrumentation tube appurtenances - lower head	SB 166 or 167 and SA 182 Type F304, F304L, or F316
Closure studs, nuts, and washers	SA 540 Class 3 Grade B23 or B24
Core support pads	SB 166 with carbon less than 0.10%
Monitor tubes and vent pipe	SA 312 or 376 Type 304 or 316 or SB 167
Vessel supports, seal ledge	SA 516 Grade 70 quenched and tempered or SA 533 Grade A, B, or C; Class 1 or 2 (vessel supports may be of weld metal buildup of equivalent strength)
Cladding	Stainless steel weld metal analysis A-7 and Ni-Cr-Fe weld metal F-Number 43
Heat lifting lugs	SA 212 Grade B
95A Steam-generator components Unit 2	
Pressure plates	SA 533 Grade A, B, or C; Class 1 or 2
Pressure forgings	SA 508 Class 2 or 3
Nozzle safe ends	Stainless steel weld metal analysis A-7
Channel heads	SA 216 Grade WCC or SA 533 Grade A, B, or C; Class 1 or 2
Tubes	SB 163 Ni-Cr-Fe, annealed
Cladding	Stainless steel weld metal analysis A-7 and Ni-Cr-Fe weld metal F-Number 43
Closure bolting	<u>Primary Side</u>: Studs: ASME SA-193 Gr. B7 Nuts: ASME SA-194 Gr. 7 (16 each per manway - specially designed for use with hydraulic stud tensioner) <u>Secondary Side</u>: Heavy Hex Bolts: ASME SA-193 Gr. B7 or ASTM A-193 Gr. B7 (20 each per manway)

INSERT 95A

UNIT 1 (Replacement)

Lower Assembly Shell	ASME SA-508, Class 3 Forging
Transition Cone (Below Girth Weld)	ASME SA-508, Class 3 Forging
Transition Cone (Above Girth Weld)	ASME SA-533, Grade A Class 1 Plate
Tubesheet	ASME SA-508, Class 3a Forging
Channel Head	ASME SA-508, Class 3 Forging
Support Plates	ASME SA-240, Type 405
Channel Head Cladding	ASME SFA-5.9, Class ER 309L
Tubesheet Cladding	ASME SFA-5.14, Class ERNiCr-3
Tubes	ASME SB-163, Alloy 690 TT
Primary Nozzle Safe Ends	ASME SA-336, Class F316LN Forgings

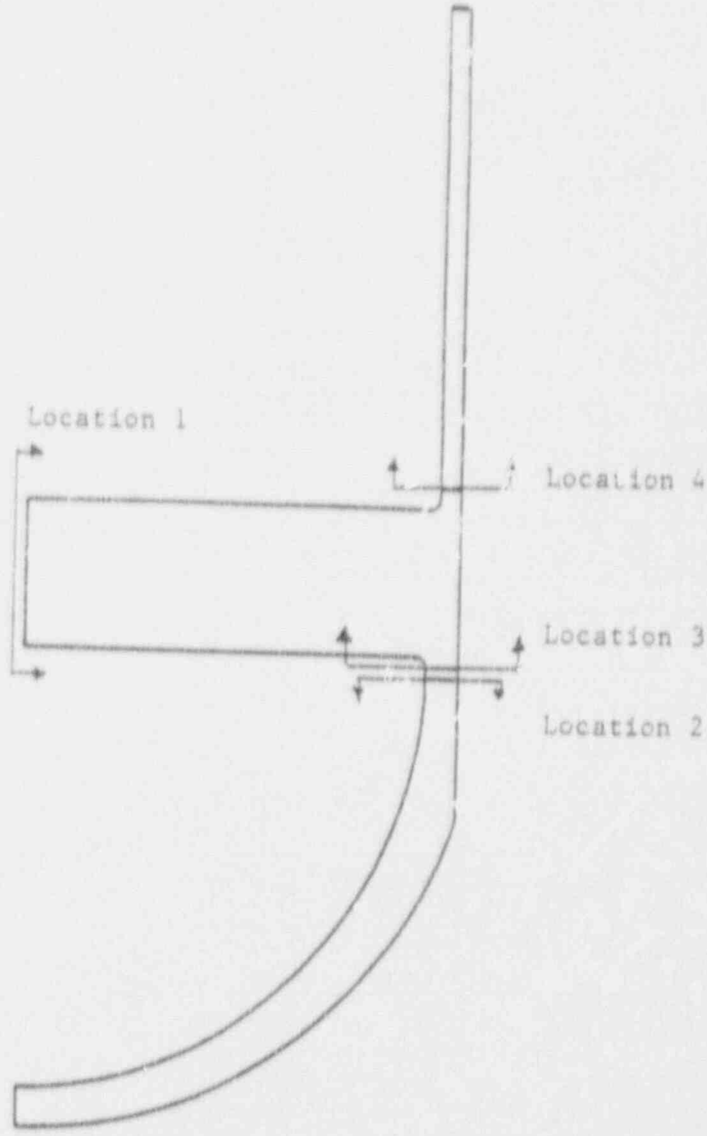
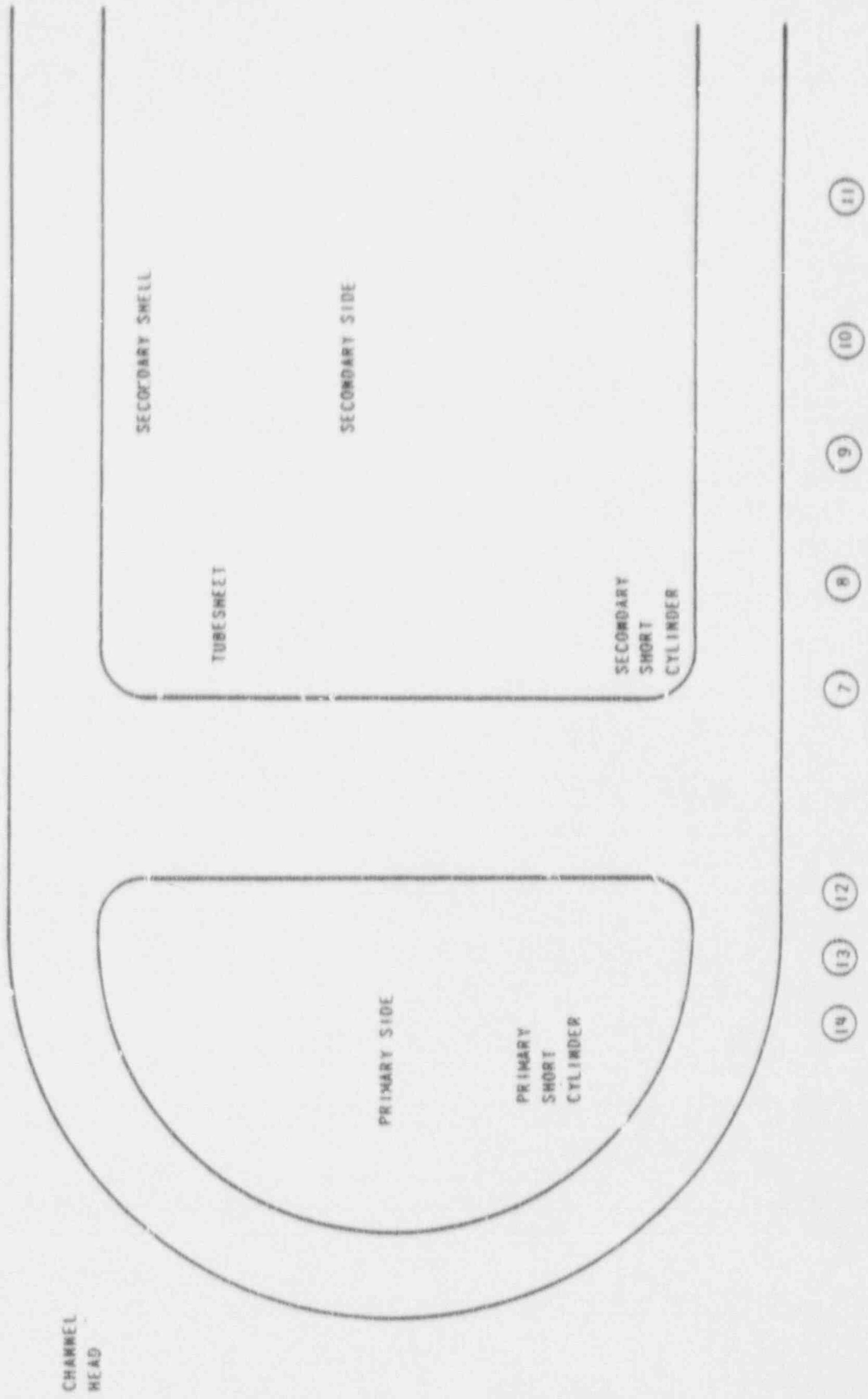


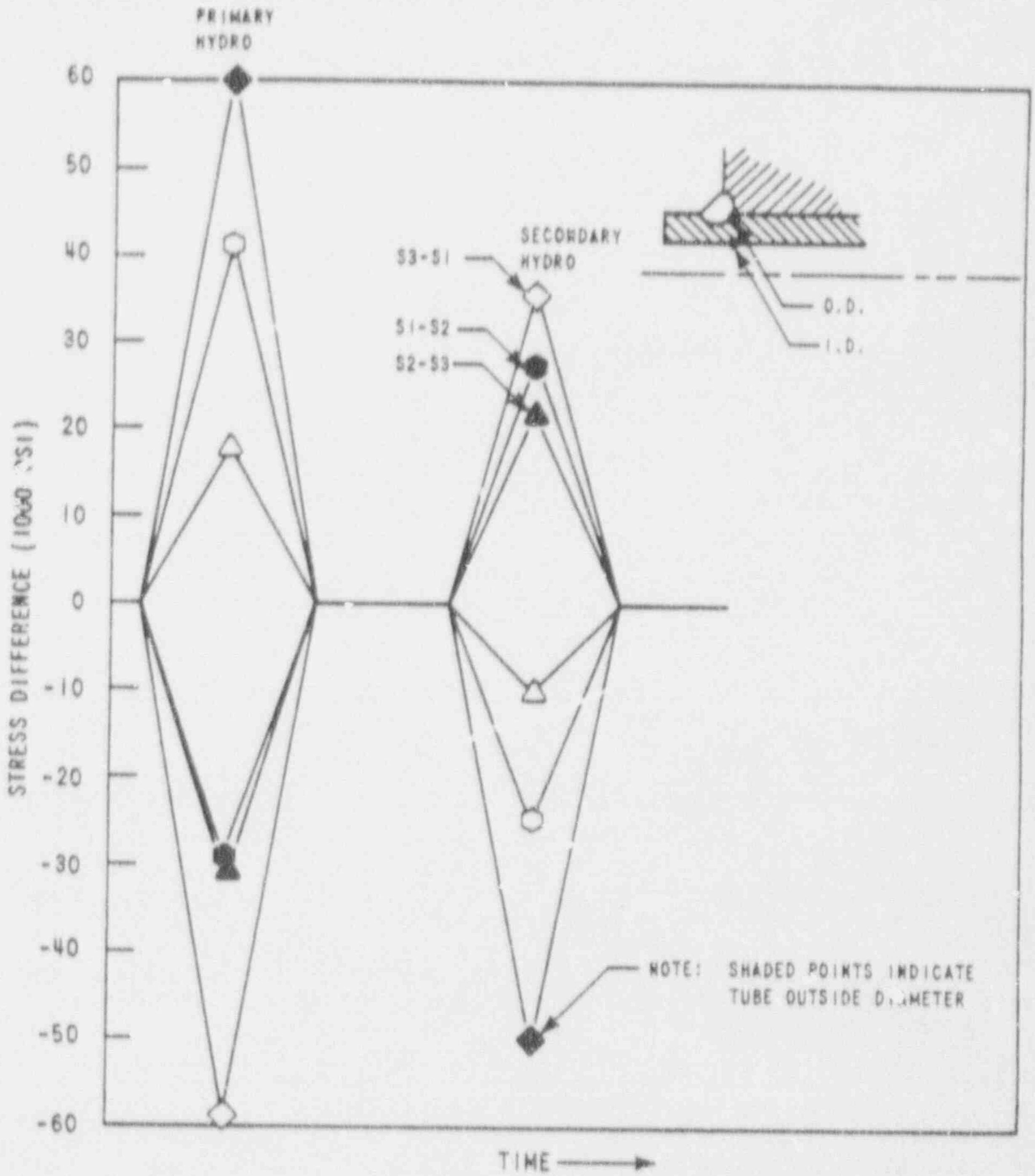
Figure 5.2-1A
(Unit 1)

Primary-Secondary Boundary Components

Shell Locations of Stress Investigation

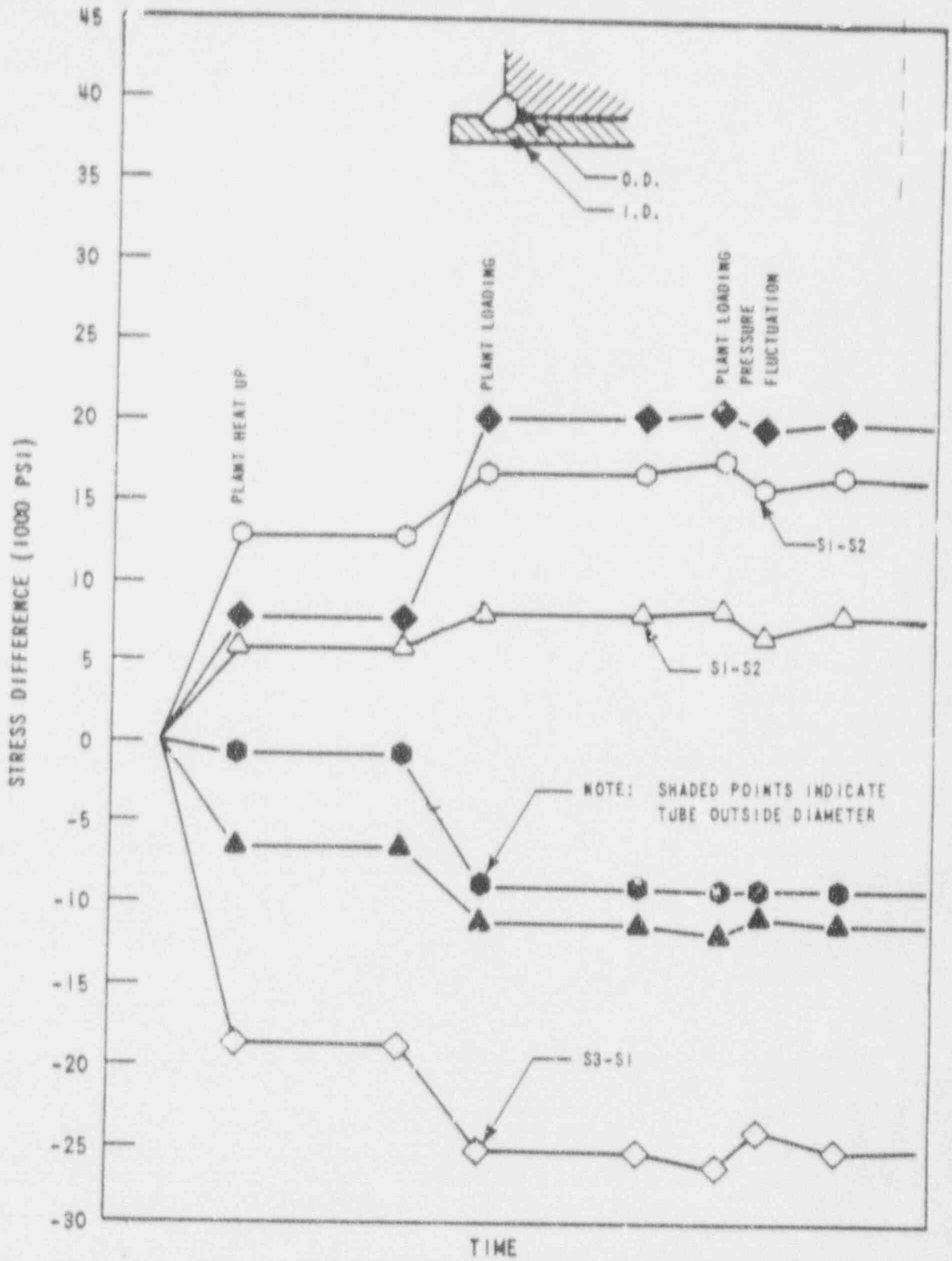


Primary-Secondary Boundary Components Shell Locations of Stress Investigations



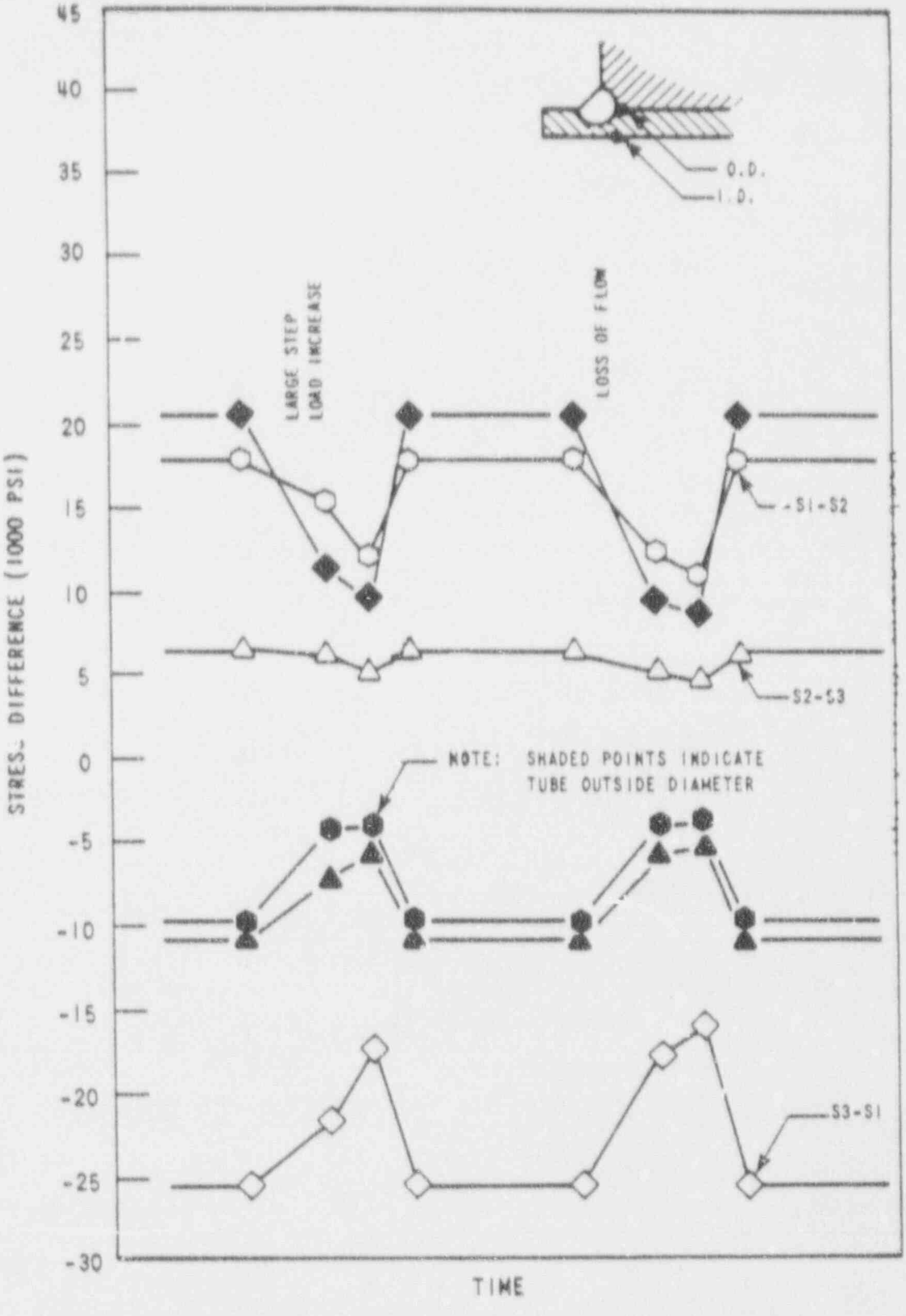
Primary and Secondary Hydrostatic Test Stress History for the Center Hole Location

(Unit 2 Only)



Plant Heat up and Loading Operational Transients (with Steady-State Plateau)
Stress History for the Hot Side Center Hole Location

(Unit 2 Only)



Large Step Load Decrease and Loss of Flow Stress History for the Hot Side Center Hole Location

(Unit 2 Only)

3. A step-load change of 10% of full power in the range from 15% to 100% full-load steam flow.

The steam-generator tubesheet complex meets the stress limitations and fatigue criteria specified in the ASME Code, Section III, as well as emergency condition limitations specified in Section 5.1. Codes and materials requirements of the steam-generator are given in Section 5.2.

The steam generator design maximizes integrity against hydrodynamic excitation and failure of the tubes for plant life.

The water chemistry in the reactor side is selected to provide the necessary boron content for reactivity control and to minimize the corrosion of reactor coolant system surfaces. The water chemistry of the steam side is given in Table 5.5-4.

5.5.2.2 Design Description

15A

Unit 2

The steam generator shown in Figure 5.5-3 is a vertical shell and U-tube evaporator with integral moisture separating equipment. The reactor coolant flows through the inverted U-tubes, entering and leaving through the nozzles located in the hemispherical bottom head of the steam generator. The head is divided into inlet and outlet chambers by a vertical partition plate extending from the head to the tubesheet. Manways are provided for access to both sides of the divided head. Steam is generated on the shell side and flows upward through the moisture separators to the outlet nozzle at the top of the vessel. The unit is primarily carbon steel. The heat-transfer tubes and the divider plate are Inconel and the interior surfaces of the reactor coolant channel heads and nozzles are clad with austenitic stainless steel. The primary side of the tubesheet is weld clad with Inconel.

Feedwater flows from a feedring into the annulus formed by the shell and tube bundle wrapper before entering the boiler section of the steam generator. Subsequently, a water-steam mixture flows upward through the tube bundle and into the steam drum section. A set of centrifugal moisture separators, located above the tube bundle, removes most of the entrained water from the

INSERT 15A

UNIT 1

The steam generator shown in Figure 5.5-3A is a vertical shell and U-tube evaporator with integral moisture separating equipment. The reactor coolant flows through the inverted U-tubes, entering and leaving through the nozzles located in the hemispherical bottom head of the steam generator. The head is divided into inlet and outlet chambers by a vertical partition plate extending from the head to the tubesheet. Manways are provided for access to both sides of the divided head. Steam is generated on the shell side and flows upward through the moisture separators to the outlet nozzle at the top of the vessel. The unit is primarily carbon steel. The heat-transfer tubes are NiCrFe Alloy 690 and the channel head divider plate is NiCrFe Alloy 600. The interior surfaces of the reactant coolant channel head are weld clad with austenitic stainless steel. The primary side interior surfaces of the tubesheet are weld clad with NiCrFe alloy 600.

Feedwater flows from a feedring into the annulus formed by the shell and tube bundle wrapper before entering the boiler section of the steam generator. Subsequently, a water-steam mixture flows upward through the tube bundle and into the steam drum section. A set of centrifugal moisture separators, located above the tube bundle, removes most of the entrained water from the steam. Steam dryers increase the steam quality to a minimum of 99.90%(0.10% moisture). The moisture separators recirculate water that mixes with feedwater as it passes through the annulus formed by the shell and tube bundle wrapper.

The steam drum has two bolted and gasketed access openings for inspection and maintenance of the dryer vane assembly, which can be disassembled and the vanes removed through the opening.

A 4.0-in. hole has been machined through the steam generator shell and wrapper between the sixth and seventh support plates to provide an inspection port for visual inspection of the tube region.

5.5.2.3.4 Corrosion

(17 A)

Unit 2

No significant general corrosion of the Inconel tubing is expected during the life of the unit. Corrosion tests show a worst-case rate of 15.0 mg/dm^2 in the 2000-hr test under simulated reactor coolant chemistry conditions. The conversion of this rate to a 40-year plant life gives a corrosion loss of 1.3×10^{-3} in., which is insignificant compared to the minimum wall thickness.

Comparable tests with Inconel-600 exposed to simulated steam-generator water chemistry have shown equally low general corrosion rates. Testing to investigate the susceptibility of heat exchanger construction materials to stress corrosion in caustic and chloride aqueous solutions has indicated that Inconel-600 has excellent resistance to general and pitting-type corrosion in severe operating water conditions, hence its selection for use in the steam generator.

Chemistry control of steam-generator water to minimize corrosive attack on the steam-generator components is discussed in Section 10.4.3.2.

5.5.2.3.5 Flow-Induced Vibration

(17 B)

Unit 2

In the design of Westinghouse steam generators, consideration has been given to the possibility of vibratory failure of tubes due to mechanical or flow-induced excitation. This consideration includes a detailed analysis of the tube supporting system as well as an extensive research program with tube vibration model tests at the Westinghouse Research and Development Laboratories.

The major cause of tube vibratory failure in heat exchanger components is that resulting from hydrodynamic excitation by the fluid outside the tube. Consideration is given by Westinghouse to the following three regions where the possibility of flow-induced vibration may exist:

1. At the entrance of downcomer feed to the tube bundle (cross flow).
2. Along the straight sections of the tube (parallel flow).
3. In the curved tube section of the U-bend (cross flow).

INSERT 17A

UNIT 1

In sizing components, design formulas take into account allowances for general corrosion. These allowances are based on corrosion tests on representative materials in simulated primary and secondary side environments. Conservative design basis allowances for ferritic and stainless steels and NiCrFe Alloy 600 are summarized in WNEP-8661, Rev. 1. Corrosion allowances established for NiCrFe Alloy 600 provide conservative upper limits for corrosion losses in NiCrFe Alloy 690. Corrosion tests (EPRI Report NP-6997-SD) on Alloy 690TT tubes exposed to simulated primary water coolant at 330°C for 1500 hours showed an average metal loss to the stream of 0.0113 mg/dm^2/day, and an average total descaled metal loss 0.0425 mg/dm^2/day. The latter descaled metal loss corresponds to 0.0072 mils per year, or 0.29 mils in 40 years, which is insignificant compared to the minimum tube wall thickness. The general corrosion loss in secondary side AVT water chemistry is expected to be equally low.

Extensive corrosion testing of thermally treated Alloy 690 tubing (EPRI Report NP-6997-SD) has shown that Alloy 690 is (i) virtually immune to primary water stress corrosion cracking, (ii) extremely resistant to stress corrosion cracking in secondary side AVT water chemistry, and (iii) resistant to secondary side faulted off-chemistry conditions.

Chemistry control of steam-generator water to minimize corrosive attack on the steam-generator components is discussed in Section 10.4.3.2.

INSERT 17B

UNIT 1

In the design of Westinghouse Model 51F steam generators used in Unit 1 of the North Anna Power Station, the possibility of tube degradation due to either mechanical or flow-induced excitation was considered. This evaluation included detailed analysis of the tube support systems as well as an extensive research program with tube vibration model tests.

Consideration was given to potential sources of tube excitation including primary fluid flow within the U-tubes, mechanically-induced vibration, and secondary fluid flow on the outside of the tubes. The effects of primary fluid flow and mechanically-induced vibration are considered to be negligible during normal operation. The primary source of potential tube degradation due to vibration is the hydrodynamic excitation by the secondary fluid on the outside of the tubes, and this area has been emphasized in both analyses and tests, including evaluation of steam generator operating experience.

Three potential tube vibration mechanisms due to hydrodynamic excitation by the secondary fluid on the outside of the tubes have been identified and evaluated. These include potential flow-induced vibrations resulting from vortex shedding, turbulance, and fluidelastic vibration mechanisms.

Vortex shedding is possible, at most, only for the outer few rows in the wrapper inlet region of steam generators such as the Model 51F for which nonuniform, two-phase turbulent flow exists throughout most of the tube bundle. Moderate tube response caused by vortex shedding is observed in some carefully controlled laboratory tests on idealized tube arrays. However, no evidence of tube response caused by vortex shedding is observed in steam generator scale model tests simulating the wrapper inlet region. Bounding calculations consistent with laboratory test parameters confirmed that vibration amplitudes would be less than .001 inches, even if the carefully controlled laboratory conditions were unexpectedly reproduced in the steam generator.

Flow-induced vibrations due to flow turbulance are also small: root mean square amplitudes are less than allowances used in tube sizing, and these vibrations cause stresses which are two orders of magnitude below fatigue limits for the tubing material. Therefore, neither unacceptable tube wear nor fatigue degradation is anticipated due to secondary flow turbulance in the Model 51F design configuration.

Fluidelastic tube vibration is potentially more severe than either vortex shedding or turbulence because it is a self-excited

INSERT 17B (Cont.)

mechanism: relatively large tube amplitudes can feedback proportional driving forces if an instability threshold is exceeded. Tube support spacing incorporated into design of both the tube support plates and the anti-vibration bars in the U-bend region provides tube response frequencies such that the instability threshold is not exceeded for secondary fluid flow conditions. This approach provides large margins against initiation of fluidelastic vibration for tubes which are effectively supported by the Model 51F tube support configuration.

Small clearances between the tubes and supporting structure are required for steam generator fabrication. These clearances introduce the potential that any given tube support location may not be totally effective in restraining tube motion if there is a finite gap around the tube at that location. Fluidelastic tube response within available support clearances is therefore theoretically possible if secondary flow conditions exceed the instability threshold assuming no support at the location with a gap around the tube.

This potential has been investigated both with tests and analyses for the U-bend region where secondary flow conditions have the potential to exceed the instability threshold if a tube does not contact two or more sequential supports as a result of fabrication tolerances. Tube vibration response is shown to have wear potential within available design margins even for limiting tube fitup conditions which are not expected. Corresponding tube bending stresses remain more than an order of magnitude below fatigue limits as a consequence of vibration amplitudes constrained by available clearances. These analyses and tests for limiting postulated fitup conditions include simultaneous contributions from flow turbulence.

Potential tube fatigue subject to postulated conservative tube support, material, and environmentalconditions has also been evaluated to demonstrate added margin against rapidly propagating fatigue. Reduced damping, due to postulated clamped conditions at the top tube supports and at anti-vibration bars (AVB'S) in the U-bend region, does not result in fluidelastic instability for small radius tubes as a result of the consistently controlled depth of AVB insertion which is deeper than that of previous operating steam generators. Postulated combinations of tube clamping and adjacent support with clearances for larger radius tubes do not lead to tube stresses above fatigue limits which have been reduced below ASME Code limits to address postulated material and environmental degradation.

Analysis and tests therefore demonstrate that unacceptable tube degradation resulting from tube vibration is not expected for the 51F steam generators at North Anna Power Station Unit 1. Operating experience with similar steam generators supports this conclusion.

consistent with accepted standards of heat exchanger design used throughout the industry (spacing, clearance, etc.). Furthermore, the design techniques are supplemented with a continuing research and development program to understand the complex mechanism of concern.

Service experience of Westinghouse PWRs steam generators shows that flow-induced vibration and cavitation effects do not cause tube thinning. Preliminary estimates of tube degradation from erosion/corrosion mechanisms indicate that approximately 2.5-mils wall thinning (2-mils primary, 0.5-mil secondary side) will result over the 40-year lifetime.

The effects of vibration, erosion, and cavitation have been given consideration, and the stress limitations for each category have been met. The analysis of LOCA blowdown forces on as-fabricated U-tubes has shown that the maximum bending load elastic stress intensity is well below the faulted condition limit. The maximum bending load elastic stress intensity (based on the nominal tube wall thickness) would still be below the faulted condition limit. Therefore, as a minimum, at least 2.5-mils (per wall) thinning can be tolerated without exceeding the allowable stress limits. Vibration effects are eliminated during normal operation by the supporting system. Under LOCA conditions, vibration is of a short duration and there is no endurance problem.

It has been determined that some tubes which experience denting at the seventh support plate, are susceptible to fatigue failure induced by fluid elastic excitation. Through extensive eddy current testing and computer analysis, these tubes have been identified and plugged with sentinel plugs.

5.5.2.4 <u>Tests and Inspections</u>

The steam generator quality assurance program is given in Table 5.5-5.

Radiographic inspection and acceptance standard are in accordance with the requirements of Section III of the ASME Code, ~~1968~~ 1986 (for Unit 1) and 1968 (for Unit 2).

Liquid penetrant inspection is performed on weld-deposited tubesheet cladding, channel head cladding, tube-to-tubesheet weldments, and weld-deposited cladding. Liquid-penetrant inspection and acceptance standards are in accordance with the requirements of Section III of the ASME Code, ~~1968~~ 1986 (for Unit 1) and 1968 (for Unit 2)

Magnetic particle inspection is performed on the tubesheet forging, channel head casting, nozzle forgings, and the following weldments:

1. Nozzle to shell.
2. Support brackets.
3. Instrument connections (primary and secondary).
4. Temporary attachments after removal.
5. All accessible pressure containing welds after hydrostatic test.

Magnetic particle inspection and acceptance standards are in accordance with the requirements of Section III of the ASME Code, ~~1968~~ 1986 (for Unit 1) and 1968 (for Unit 2).

An ultrasonic test is performed on the tubesheet forging, tubesheet cladding, secondary shell, and head plate and nozzle forgings.

The heat-transfer tubing is subjected to eddy current test.

Hydrostatic tests were performed in accordance with Section III of the ASME Code, ~~1968~~ 1986 (for Unit 1) and 1968 (for Unit 2). In addition, the heat-transfer tubes are subjected to a hydrostatic test pressure, before installation into the vessel, which is not less than 1.25 times the primary-side design pressure multiplied by the ratio of the material allowable stress at the testing and design temperatures.

Manways are to provide access to both the primary and secondary sides.

A specific plan for inservice inspection of steam generator tubes is not available. Because of the activity in the channel head and the large number of tubes involved, tube testing is done on a per plant basis. The extent of tube testing planned in any particular plant will depend on tube performance to date, the channel head activity, and the results of tube sample testing. Inservice inspection of the steam-generator tubes is not planned at this time, but the eddy current testing method is available.

no case did the hydrostatic pressure exceed 35 psig anywhere in the tank. The tank was also leak tested with dry air at 20 psig by applying soapsuds to all welds accessible from the outside of the tank.

5.5.9.2.2 Steam-Generator and Reactor Coolant Pump Supports

The steam-generator and reactor coolant pump supports are shown in Figures 5.5-11 and 5.5-12, and the load paths into the reinforced concrete are shown in Figure 5.5-13. The materials used were for the most part commercially available structural shapes of A36 steel. High-strength quenched and tempered alloy steels were used for local attachments at the steam-generator support pads, in the hydraulic snubbing assemblies, and in the pump support columns.

The steam-generator support system consists of an upper support ring and a lower support frame. The upper support ring was shimmed in the cold condition to the steam generator with a 0.090- to 0.120-in. radial gap to permit full-pressure expansion of the steam generator and insulated so that it expands thermally as the steam generator is brought up to temperature. The upper support ring transmits horizontal forces from the steam generator through four tangential load trains to the reinforced-concrete charging floor. The charging floor in turn transmits these horizontal forces to the reactor shield wall, the crane wall, and the cubicle walls, where, through shearing actions, it is further transmitted downward to the mat. A 4-ft 8-in. octagonal concrete column between the cubicle floor and the mat beneath the steam generator provides an additional load path that transmits some of the vertical forces directly from the cubicle floor to the mat. The two tangential load trains from the upper support ring to the charging floor parallel to the hot leg of the reactor coolant loop are equipped with hydraulic snubbing cylinders that permit limited slow motion of the steam generator to allow for thermal expansion of the reactor coolant piping from the reactor to the steam generator. However, the cylinders react to resist suddenly applied forces that occur from earthquake and/or pipe rupture conditions. The other two tangential loads trains from the upper support ring to the charging floor act in a direction perpendicular to the direction of the reactor coolant loop hot leg. Since the movement in that direction is not significant, two strut members are designed to resist applied forces primarily from earthquake and/or postulated pipe rupture conditions.

0.050 to 0.060 in.

2

floor close to the center of gravity of the vessel. These brackets permit the pressure vessel to expand vertically but restrain horizontal displacements.

The ring girder is fabricated from ASTM A516, Grade 70 steel and the striker plate assemblies are fabricated from ASTM A543, Grade B, Class II steel. The hanger columns are fabricated from ASTM A106, Grade B pipe. The majority of the fasteners and shear pins used in the support are fabricated from either ASTM A193, Grade B7, or ASTM A540, Grade B23, depending on the stress level.

5.5.9.3 Evaluation

5.5.9.3.1 Steam-Generator and Reactor Coolant Pump Supports

Dynamic analyses were performed to determine loads on the support structures and components resulting from rupture of the branch lines to the reactor coolant piping (RHR, Pressurizer Surge, Accumulator lines), main steam lines and main feedwater lines and design-basis horizontal and vertical earthquakes. The combined loadings were obtained by first alegbraically summing the loads due to pressure, deadweight and thermal and then by directly adding that to SRSS to SSE and the worst effect of Postulated Main Steam Line Break, Main Feedwater Line Break or Reactor Coolant Loop branch line (RHR, Pressurizer Surge, Accumulator) break.

The dynamic model for a loop as indicated in Figure 5.5-17 includes the steam generator, reactor coolant pump, associated piping, and supports as a coupled system. To complete the model, the inlet and outlet nozzles of the reactor vessel were assumed to be rigidly attached to the vessel. The mass and stiffness characteristics of each of the major subsystems were accurately transformed to a lumped parameter system. Approximately 80 nodes (450 static degrees of freedom) were employed in the dynamic model representation.

Natural frequencies, characteristic mode shapes, and modal participation factors were calculated for the undamped multidegree of freedom combined structural system using the "Stardyne" program developed by Mechanics Research. The dynamic loading conditions were specified as spatial load vectors and associated time histories. The "step-by-step direct integration method" was employed to obtain a time history of forces and displacements for

"NUPIPE-SW" computer program for Unit 1 and the "STARDYNE" computer program (developed by Mechanics Research)

Table 5.5-3

STEAM GENERATOR DESIGN DATA

Parameter	Un. #1 Value	Un. #2 Value
Design pressure, reactor coolant side, psig	2485	2485
Design pressure, steam side, psig	1085	1085
Design temperature, reactor coolant side, °F	650	650
Design temperature, steam side, °F	600	600
Total heat transfer surface area, ft^2	54,500	51,500
Maximum moisture carryover, wt %	0.10	0.25
Overall height, ft-in.	67-8	67-8
Number of U-tubes	3592	3388
U-tube o.d., in.	0.875	0.875
Tube wall thickness, nominal, in.	0.050	0.050
Number of manways	4	[illegible]
I.d. of manways, in.	16	16
Number of handholes	6	2
I.d. of handholes. in.	6	6
Number of inspection ports	2	1
I.d. of inspection port, in.	4	2.5

Table 5.5-4

STEAM GENERATOR AVT CHEMISTRY GUIDELINES

Parameter	In the Absence of Locatable Condenser Leakage: Without Boric Acid	In the Absence of Locatable Condenser Leakage: With Boric Acid	In the Presence of Locatable Condenser Leakage: Without Boric Acid[1]	In the Presence of Locatable Condenser Leakage: With Boric Acid[1]
pH @ 25°C	>8.5	>7.0	>8.5	>7.0
Cation Conductivity µmhos/cm @ 25°C	<2.0	<2.0	2.0	2.0
Boron, ppm	N/A	5 - 10	N/A	5 - 10
Sodium, ppm	<0.04	<0.04	0.1	0.1
Chloride, ppm	<0.05	<0.05	0.15	0.15
Oxygen, ppb	<5	<5	<5	<5
Hydrazine, ppb	$[O_2] + \geq 20$	$[O_2] + \geq 20$	$[O_2] + \geq 20$	$[O_2] + \geq 20$
Ammonia, ppm	>0.06	>0.06	>0.06	>0.06
Silica, ppm	<0.05	<0.05	0.05	0.05
Blowdown, gpm	Continuous[2]	Continuous[2]	Continuous[3]	Continuous[3]

2

1 Continued operation with locatable contaminant ingress is not recommended.
2 Operate at the minimum continuous blowdown rate required to maintain continuous monitoring capability, approximately 5 gpm/SG.
3 Blowdown continuously at a rate required to maintain chemistry parameters.

Table 5.5-18A
Unit 1

MAXIMUM STEAM-GENERATOR AND REACTOR COOLANT PUMP
FOOT LOADS (kips)

Component	Direction	Total[a]
Steam generator	Tangential	~~[illegible]~~ 520
	Vertical compression	960
	Vertical tension	71[illegible]
Reactor coolant pump	Tangential	381
	Vertical compression	1197
	Vertical tension	946

[a]Faulted condition, thermal + operating pressure + deadweight + SRSS of seismic (DBE) and pipe rupture.

REPLACE EXISTING
TABLE 5.5-18 WITH
TABLES 5.5-18A & B

(23)

Table 5.5-18B
Unit 2

MAXIMUM STEAM-GENERATOR AND REACTOR COOLANT PUMP FOOT LOADS (kips)

Component	Direction	Total[a]
Steam generator	Tangential	795
	Vertical compression	773
	Vertical tension	557
Reactor coolant pump	Tangential	441
	Vertical compression	1590
	Vertical tension	1142

[a]Faulted condition, thermal + operating pressure + deadweight + SRSS of seismic (DBE) and pipe rupture.

REPLACE EXISTING
TABLE 5.5-18 WITH
TABLES 5.5-18 A & B

Table 5.5-19A
Unit 1

MAXIMUM LOAD (KIPS), SUPPORTS

Concrete Reactions From	Node[a]	Direction	Loads: Pipe Rupture	Seismic (DBE)	Normal[d]	Total[b]
Steam-Generator Support	52	Z	±685[c]	±688	±268	1239
	61	Z	±529[c]	±424	±387	1065
	71	Y	±461[c]	±695	±62	896
	82	Y	±236[c]	±322	±99	499
	88	Z	±824[c]	±707	±148	1234
	97	Z	±615[c]	±501	±178	971
Pump Columns	9	Z	±362[c]	±430	±461	1023
	13	Z	±631[c]	±239	±227	901
	16	Z	±529[c]	±264	±188	779

[a]Refer to Figure 5.5-13

[b]Obtained by adding normal to the SRSS of pipe rupture and seismic.

[c]Loads are conservatively obtained by the absolute sum of the reactions resulting from each component of force moment applied to the support frames.

[d]Deadweight + thermal + operating pressure.

REPLACE EXISTING
TABLE 5.5-19 WITH
TABLES 5.5-19 A & B

Table 5.5-19B
Unit 2

MAXIMUM LOAD (KIPS), SUPPORTS

Concrete Reactions From	Node[a]	Direction	Loads: Pipe Rupture	Seismic (DBE)	Normal[d]	Total[b]
Steam-Generator Support	52	Z	±473[c]	±491	+234	916
	61	Z	±469[c]	±342	+303	883
	71	Y	±273[c]	±393	-19	498
	82	Y	±299[c]	±221	-83	401
	88	Z	±503[c]	±545	+77	818
	97	Z	±463[c]	±386	+80	687
Pump Columns	9	Z	±1278[c]	±255	+478	1781
	13	Z	±1404[c]	±255	+217	1639
	16	Z	±971[c]	±329	-204	1220

[a]Refer to Figure 5.5-13

[b]Obtained by adding normal to the SRSS of pipe rupture and seismic.

[c]Loads are conservatively obtained by the absolute sum of the reactions resulting from each component of force moment applied to the support frames.

[d]Deadweight + thermal + operating pressure.

REPLACE EXISTING
TABLE 5.5-19 WITH
TABLES 5.5-19 A & B

Table 5.5-20A
Unit 1

MAXIMUM LOAD (KIPS), SNUBBERS AND STRUTS

Concrete Reactions From	Node[a]	Direction	Loads: Normal[c]	Pipe Rupture	Seismic (DBE)	Total[b]
Steam-Generator Upper Supports	110	Y	±52	±644	±282	755
	112	X	0	±561	±355	664
	115	Y	±52	±644	±282	755
	118	X	0	±561	±355	664

[a] Refer to Figure 5.5-13

[b] Normal plus SRSS of pipe rupture and seismic.

[c] Deadweight + thermal + operating pressure.

REPLACE EXISTING
TABLE 5.5-20 WITH
TABLES 5.5-20 A & B

Table 5.5-20B
Unit 2

MAXIMUM LOAD (KIPS), SNUBBERS AND STRUTS

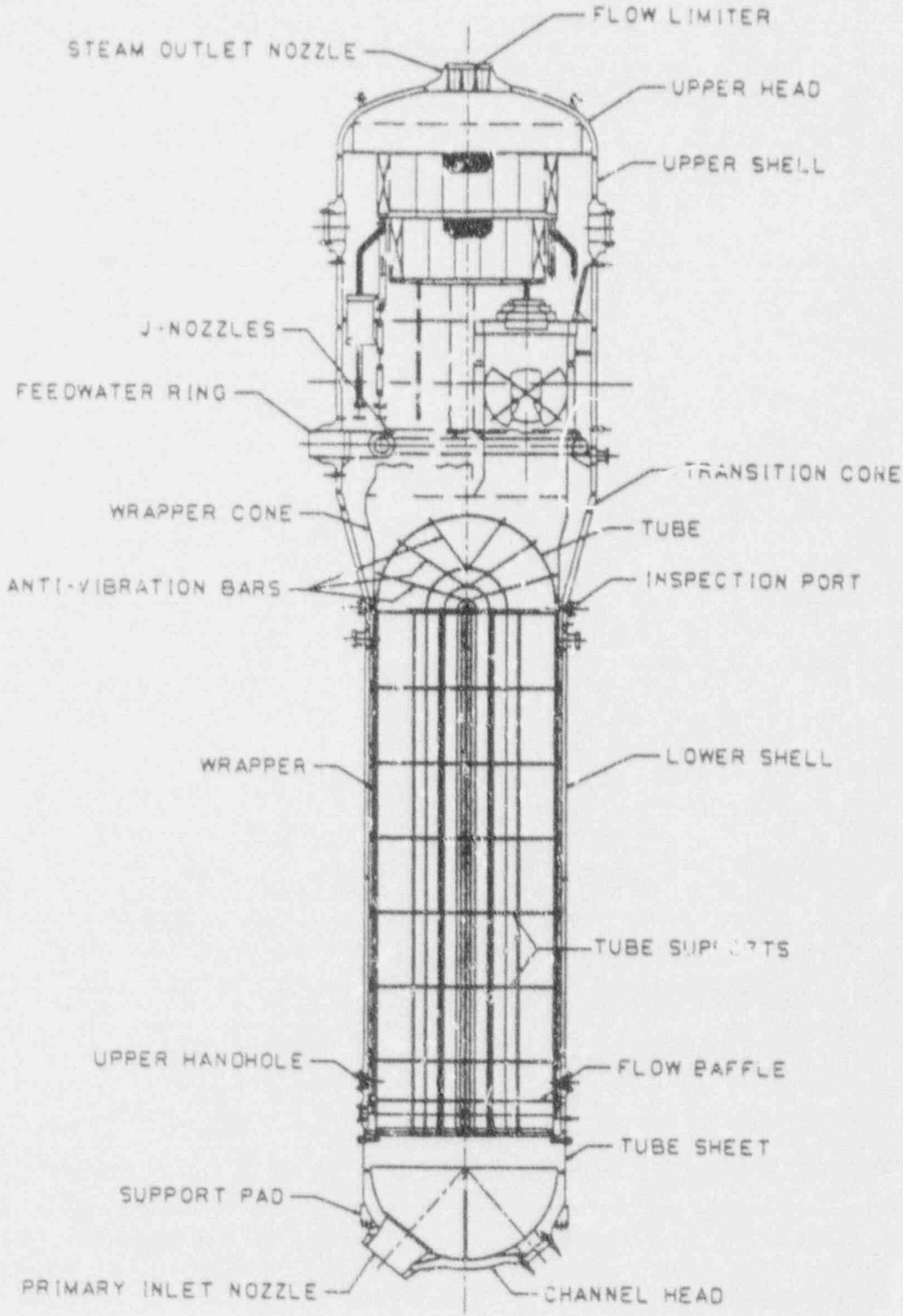
Concrete Reactions From	Node[a]	Direction	Loads: Normal[c]	Pipe Rupture	Seismic (DBE)	Total[b]
Steam-Generator Upper Supports	110	Y	±59	±798	±282	915
	112	X	0	±1534	±359	1575
	115	Y	±59	±798	±282	915
	118	X	C	±1534	±359	1575

[a]Refer to Figure 5.5-13

[b]Normal plus SRSS of pipe rupture and seismic.

[c]Deadweight + thermal + operating pressure.

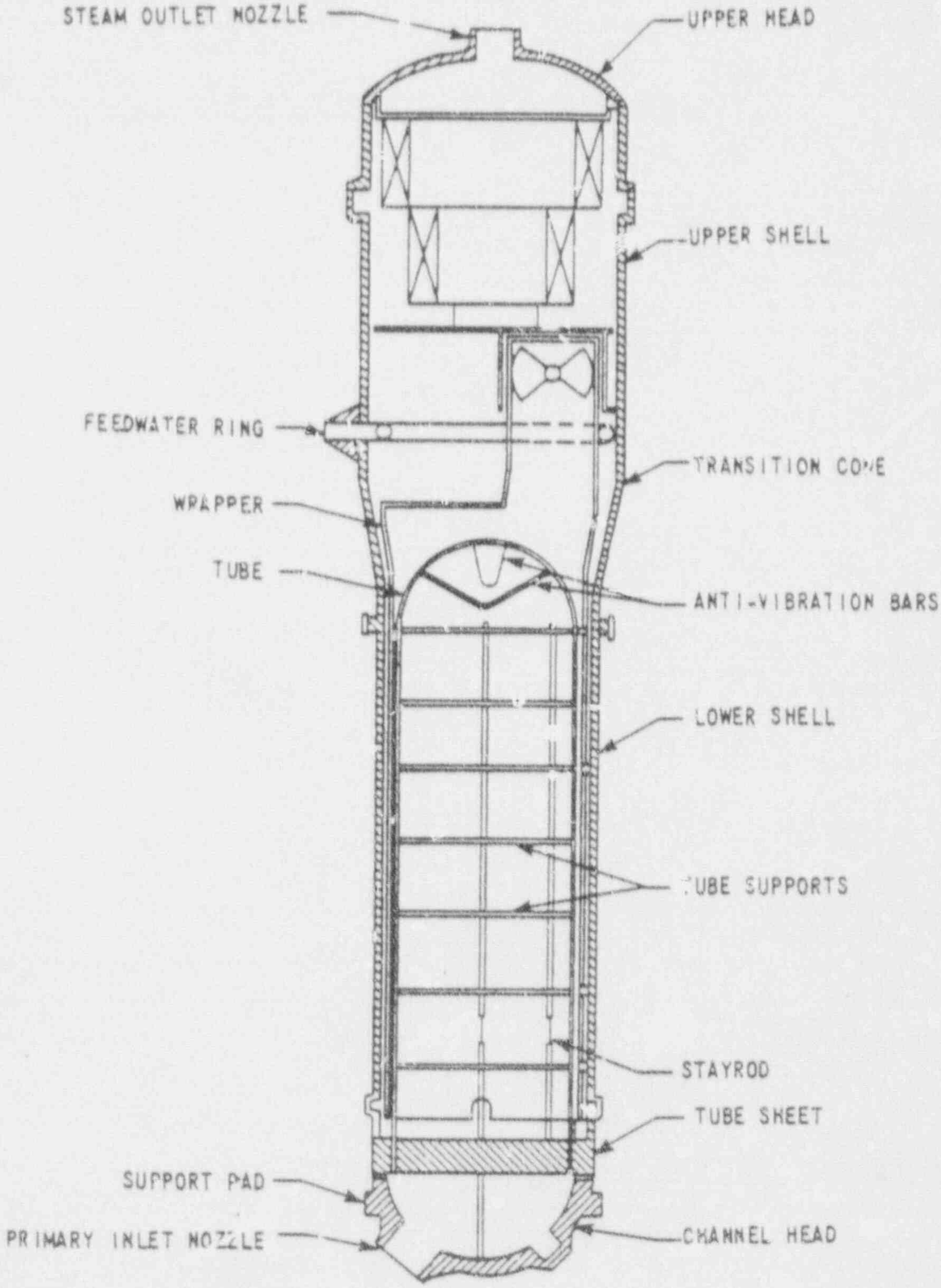
REPLACE EXISTING
TABLE 5.5-20 WITH
TABLES 5.5-20 A & B



~~54F~~ 51F SERIES STEAM GENERATOR

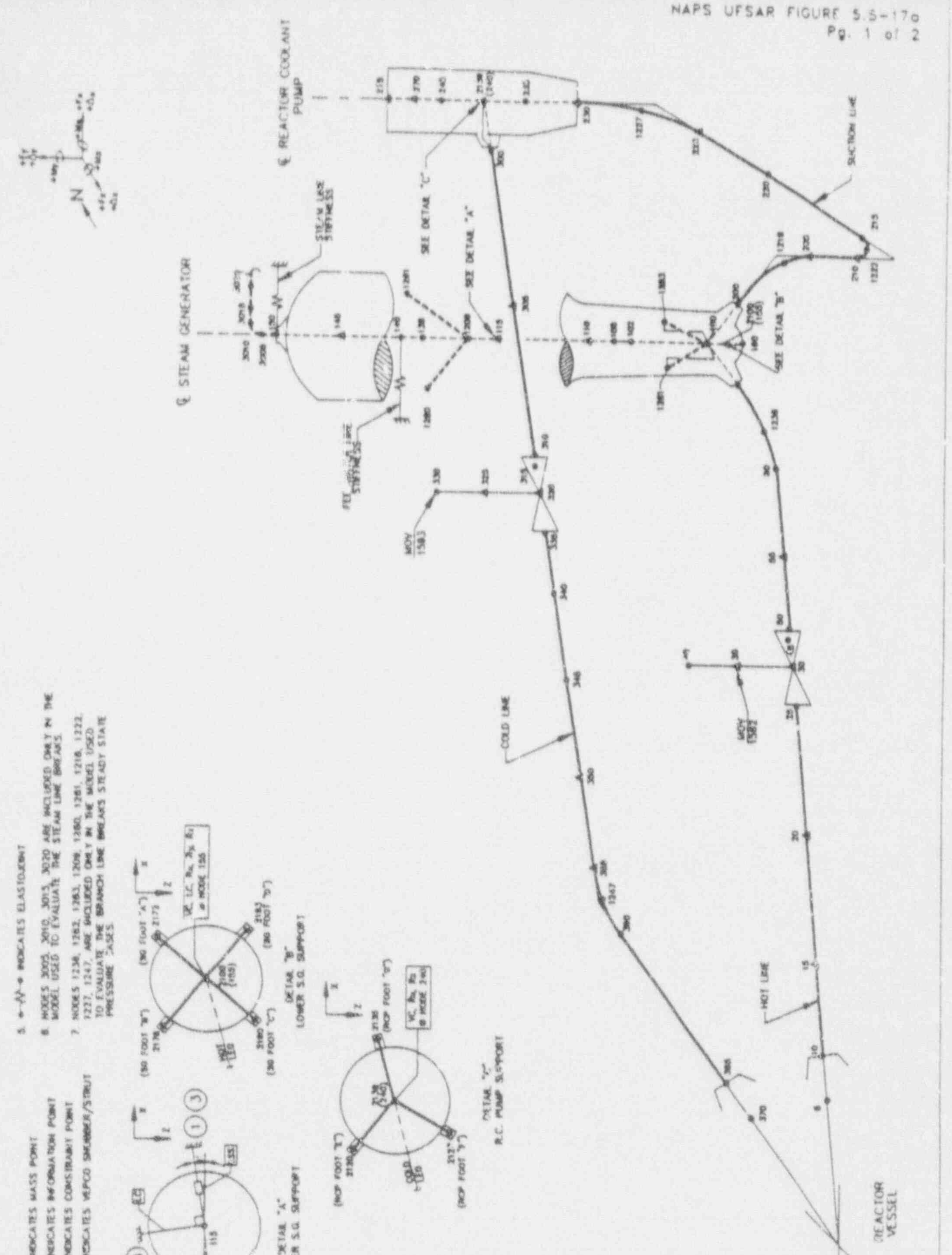
(Unit 1)

~~FIGURE 5.5-3~~



51 SERIES STEAM GENERATOR
(Unit 2)

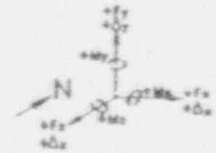
REPLACE EXISTING
FIGURE 5.5-17 WITH
FIGURE 5.5-17a (2 sheets)(UNIT 1)
and FIGURE 5.5-17b (UNIT 2)



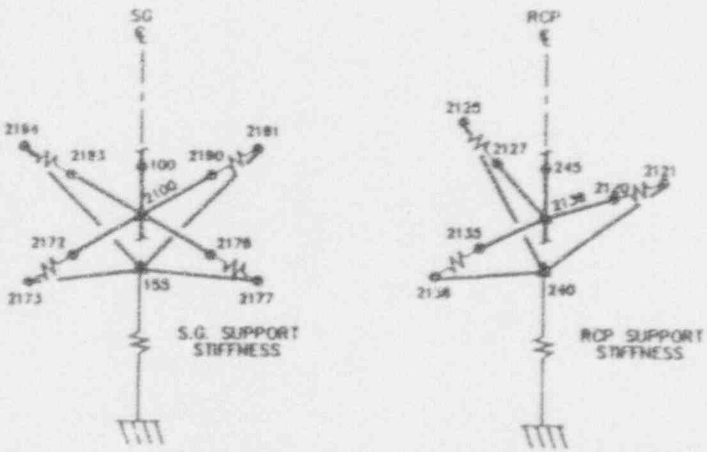
DYNAMIC MODEL OF
STEAM GENERATOR AND PUMP SUPPORTS
NORTH ANNA POWER STATION
UNIT 1

NOTES:

1. Δ INDICATES MASS POINT
2. • INDICATES INFORMATION POINT
3. ⊕ INDICATES CONSTRAINT POINT
4. ○ INDICATES VEPCO SNUBBER/STRUT
5. INDICATES ELASTOJOINT

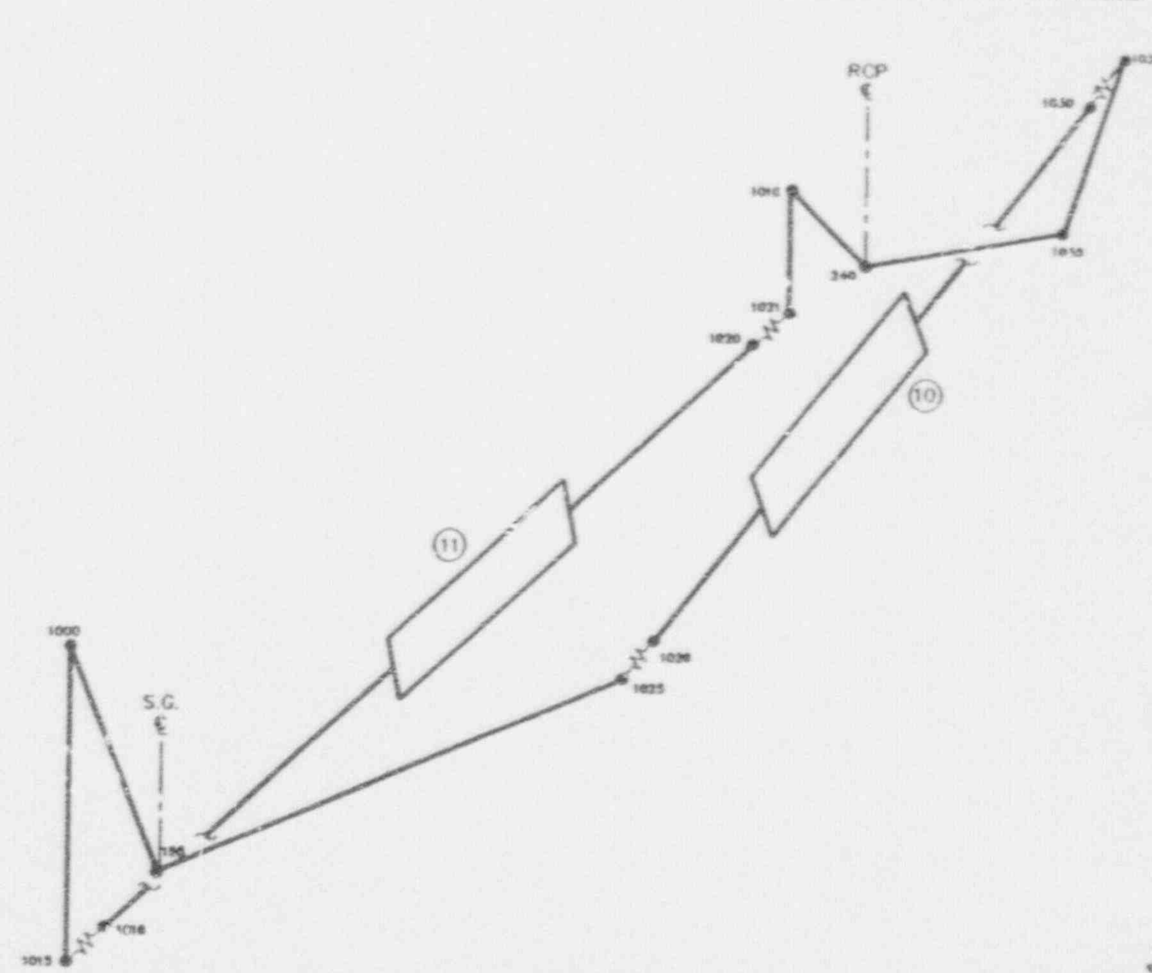


SG AND RCP SUPPORT FEET REPRESENTATIONS



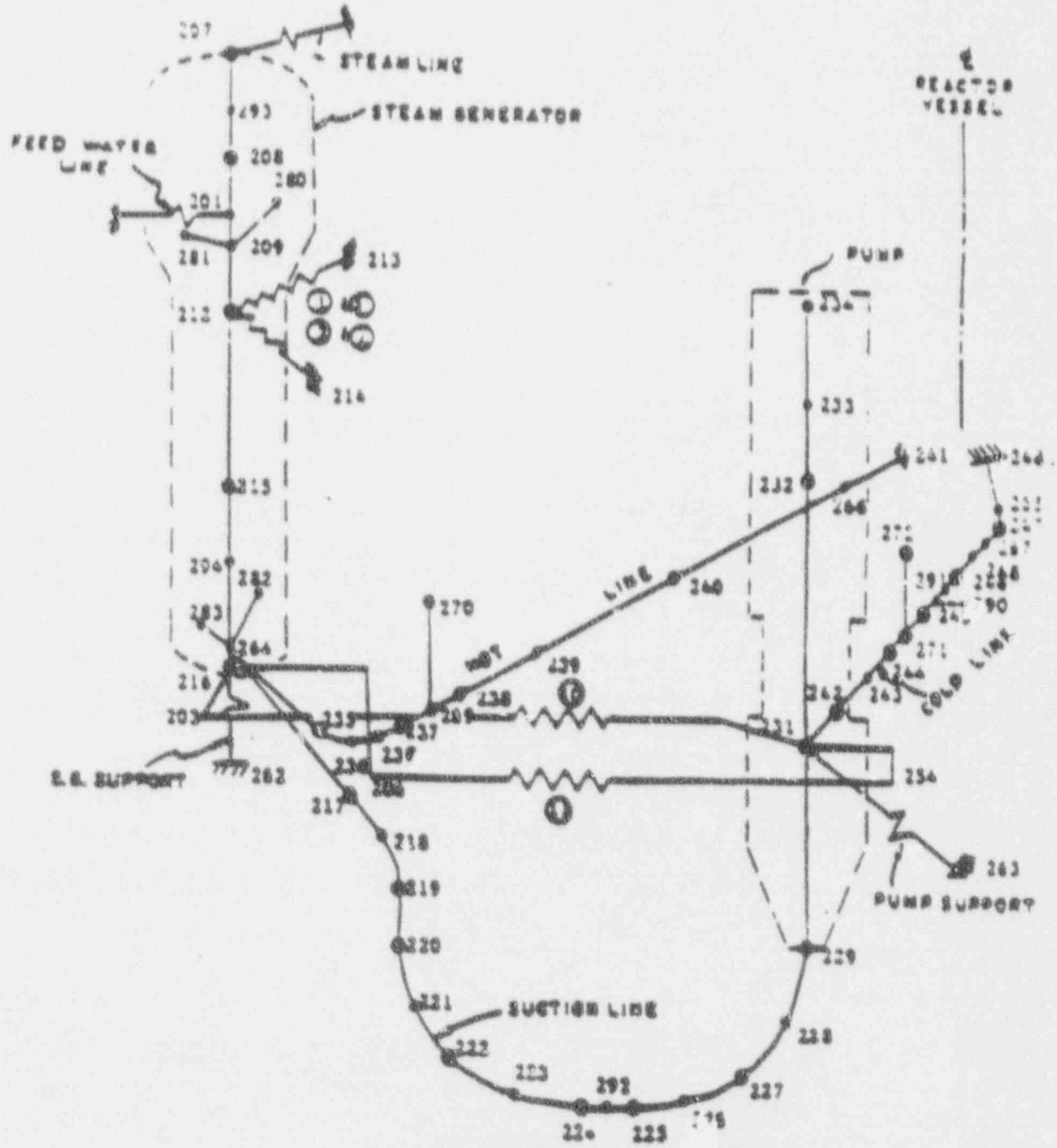
ALL SUPPORT MEMBERS ARE RIGID. ALL FEET (PTS. 2130, 2127, 2135, 2172, 2180 & 2183) ARE AXIALLY AND ROTATIONALLY FREE. PT. 155 IS COINCIDENT WITH PT. 2100 AND PT. 240 IS COINCIDENT WITH PT. 2130.

SG AND RCP SUPPORT FRAME REPRESENTATION

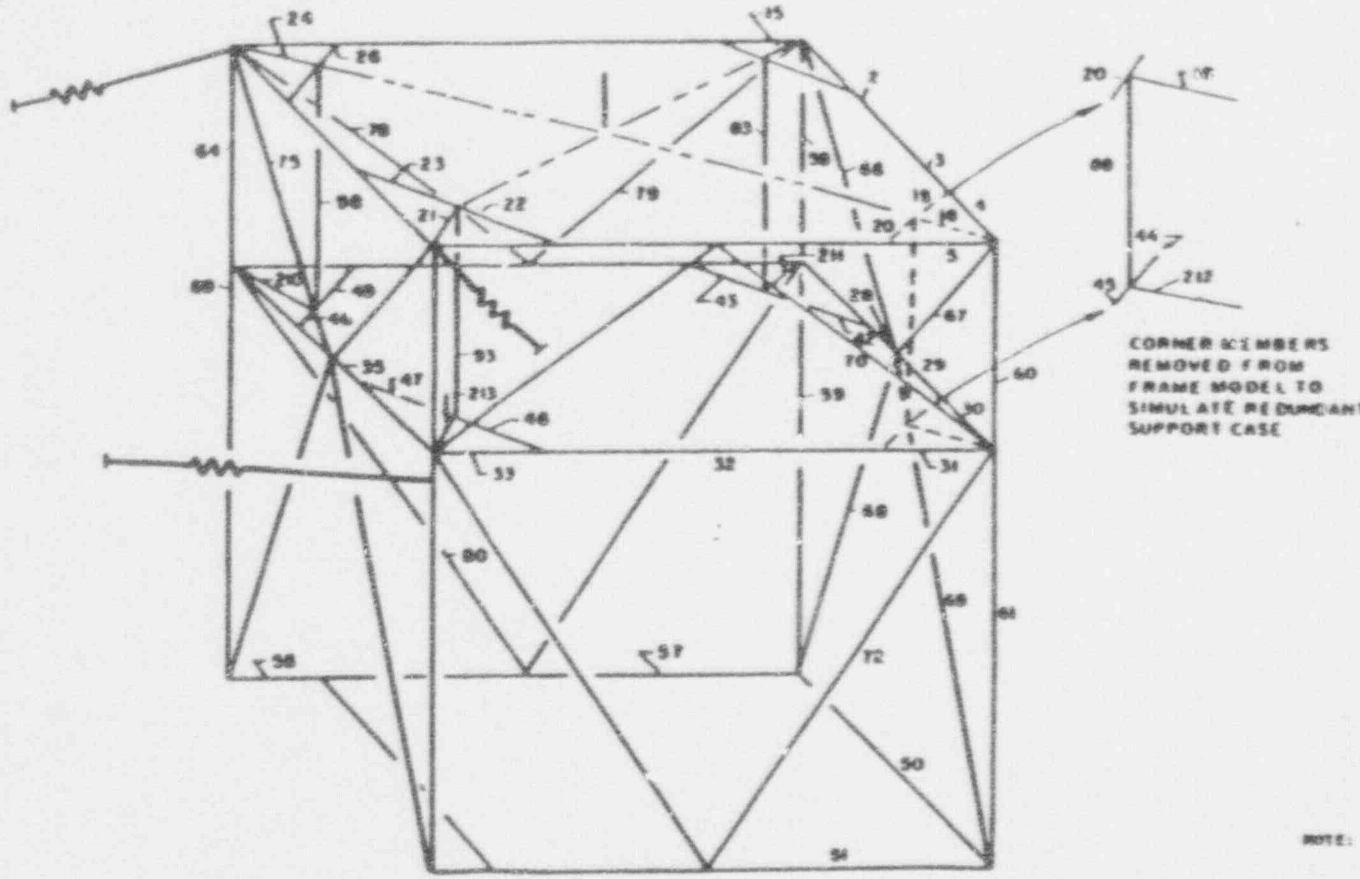


DYNAMIC MODEL OF
STEAM GENERATOR AND PUMP SUPPORTS
NORTH ANNA POWER STATION
UNIT 1

NAPS UFSAR FIGURE 5.5-17a
Pg. 2 of 2



DYNAMIC MODEL OF STEAM GENERATOR AND PUMP SUPPORTS
NORTH ANNA POWER STATION
UNITS 1 AND 2 UNIT 2

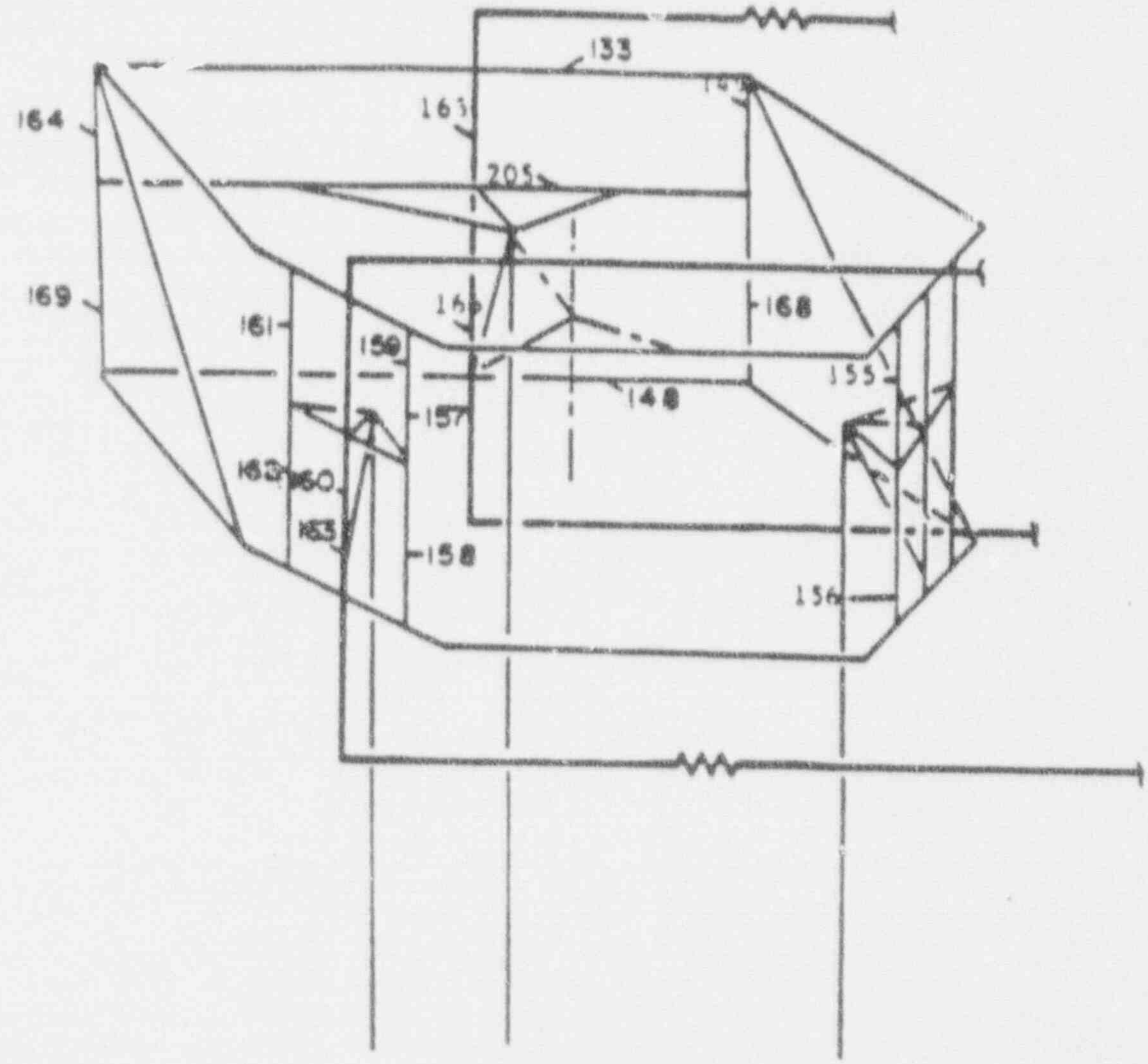


NOTE: INCLUDED IN TABLES 5.5-21 AND 5.5-22

IDENTIFICATION OF
STEAM GENERATOR
SUPPORT MEMBERS

(31)

DELETE



NOTE: INCLUDED IN TABLES 5.5-21 AND 5.5-22

IDENTIFICATION OF R. C. PUMP SUPPORT MEMBERS

6 DELETE

6.2 CONTAINMENT SYSTEMS

6.2.1 CONTAINMENT FUNCTIONAL DESIGN

6.2.1.1 Design Basis

6.2.1.1.1 Containment

Insert A

6.2.1.1.1.1 Design Criteria. The design of the subatmospheric containment structure is based on the following criteria:

1. The peak calculated containment atmosphere pressure shall not exceed the design pressure of 45 psig.

2. The containment shall be depressurized following a design basis accident to below 1 atm absolute pressure in less than 60 min.

3. Once depressurized, the containment shall be maintained at a pressure less than 1 atm absolute for the duration of the accident.

e

The peak containment pressure due to a postulated loss-of-coolant accident (LOCA) occurs after a double-ended rupture (DER) of a reactor coolant pump suction line and is a function of the initial total pressure and average temperature of the containment atmosphere, the containment free volume, the passive heat sink in the containment, the quench spray (QS) system design, and the rates of mass and energy released to the containment. The passive heat sinks in the containment are considered to be at the same initial temperature as the initial average containment atmosphere temperature. Maximizing the initial containment total pressure and average atmospheric temperature maximizes the calculated peak pressure.

The time required to depressurize the containment and the capability to maintain it depressurized below 1 atm of pressure after a pump suction double-ended rupture (PSDER) depends on the mass of air in the containment, the design of the containment depressurization system (both QS and recirculation spray (RS) subsystems, see Section 6.2.2.1), and on the service

INSERT A

A new analysis of containment peak pressure and depressurization following a LOCA was performed as a result of replacement of the steam generators in Unit 1. The mass and energy releases for this analysis are based on the models described in Section 6.2.1.1.1.3. Where methodology or results differ from the previous anslysis, the new Unit 1 information is provided and the previous information is relabeled as applicable to Unit 2 only. The new Unit 1 analysis incorporates the new model 51F steam generator parameters.

Power Coastdown Heat

After receipt of a reactor trip signal caused by the LOCA, the reactor power decreases to fission product decay levels over a finite period of time, depending upon the time it takes for the control rods to drop, the rate of boron injection, the half-life of the longest-lived delayed neutron precursor, moderator and fuel temperatures, and moderator levels in the reactor vessel during blowdown. ¶The power coastdown heat curves for the pipe breaks being analyzed are shown on Figure 6.2-2.

Insert B

The heat from fission product decay (Figure 6.2-1) and power coastdown (Figure 6.2-2) during the time interval under consideration is computed and added as sensible heat to the reactor core.

The stored energy is evaluated using a detailed temperature model of the pellet, clad, and gap. The temperature distribution within the fuel pellet is predominantly a function of the local power density and the UO_2 thermal conductivity. However, the computation of radial fuel temperature distribution combines crud, oxide, clad, gap, and pellet conductances. The factors that influence these conductances, such as a gap size (or contact pressure), internal gas pressure, gas composition, pellet density, and radial power distribution within the pellet, etc., have been combined into a semi-empirical thermal model. This thermal model has been incorporated into a computer code to determine these factors and their net effects on temperature profiles. The temperature predictions of the code have been compared to inpile fuel temperature measurements and melt radius data with good results. Table 6.2-4 presents the results of a sensitivity study on core stored energy, in full power seconds above average coolant temperature, varying the following parameters:

1. Average power level.
2. Number of nodes assumed in the pellet.
3. Effect of fuel densification.

INSERT B

Unit 1

Power coastdown heat and decay heat generation during the blowdown, reflood and post-reflood phases of the accident (until intact loop steam generator equilibration) are included in the calculation of mass and energy releases as described in Section 6.2.1.1.1.3. Subsequent to intact loop steam generator equilibration, LOCTIC utilizies the aforementioned decay heat model to calculate the mass and energy released to containment.

Unit 2

16.83

The reactor thermal power is 2900 MW, and the heat in the core is 22.08 x 10^6 Btu above the average coolant temperature. These values have been used in the LOCTIC code to evaluate containment pressure. The core stored heat of 16.83 22.05 x 10^6 Btu is equivalent to 8 6 full-power sec at 2900 MW(t). The conservatism in this value of stored heat is demonstrated below.

A sensitivity study for the effect of

A conservative value of 8.0 full-power sec includes fuel densification and additional margins. It should be noted that Table 6.2-4 shows a minimum of 17% margin. is shown in

insert C →

During the post-reflood froth phase, the mass and energy release rates provided in WCAP-8312A[3] are modified by LOCTIC as described below.

The steam generator energy is released to the containment atmosphere in two stages referred to as the "equilibration stage" and the "depressurization stage." In the former, the energy sources above the reference pressure used in calculating the mass and energy releases are brought into equilibrium with the containment pressure. The rate for this phase is set by the Westinghouse froth calculation model. In the latter stage, the sources give up additional energy as the containment pressure decreases. The rate for this stage is set by the containment depressurization rate.

The intact loop steam generators and metal energies are lumped together for this calculation.

After the post-reflood froth period, LOCTIC computes the mass and energy release (boil-off) to the containment atmosphere, which is governed by decay heat.

Broken Loop Steam Generator-Equilibration Stage

The mass and energy release rates into the containment during the time prior to broken loop steam generator equilibration ($T^*_{eq(bl)}$) are presented in Table 6.2-16.

At the time ($t^*_{eq(bl)}$) when the broken loop equilibrates with the reference pressure (P*), the calculated containment pressure may be less than the reference pressure. If so, an extension of the broken loop equilibration

INSERT C

Post-Reflood Energy Release From Steam Generators

The mass and energy release models during this phase of the accident are discussed in Section 6.2.1.1.1.3.

$\Delta E_{dep(bl)}$ = The energy transferred from the steam generator secondary during a time increment.

and ΔP = The change in containment pressure during the previous time increment.

The additional mass increment is then calculated by:

$$\Delta M = \frac{\Delta E_{dep(bl)}}{h_{fg}}$$

where h_{fg} is the latent heat of vaporization at the current containment conditions.

Intact Loop Steam Generators - Equilibration Stage

The same procedure is used as for the broken loop. However, metal and core energy are lumped with the steam generator energy for this calculation.

The equations are the same as those for the broken loop except that the subscript "(il)" replaces the subscript "(bl)".

Intact Loop Steam Generators - Depressurization Stage

Insert D

Again the procedure sued is the same as for the broken loop case except that decay heat, q_{dcy}, is added to the heat addition rate.

The additional mass rate to be added is then

$$M_{dcy} = \frac{q_{dcy}}{h_{fg}}$$

Core Sensible Heat

Insert E

Since the core contains considerable heat at a temperature above the [illegible] computes the heat transfer from the core to the reactor coolant as a function

INSERT D

During this period, the steam generator secondary is brought to the ambient conditions with the containment and reactor coolant is assumed saturated for the containment pressure. The secondary energy which remains after equilibration is:

$$E_{dep(II)} = E_{av(II)} - \Delta E_{eq(II)}$$

where

$E_{dep(II)}$ = Energy which must be transferred from the intact loop steam generator secondary during the depressurization stage.

$\Delta E_{eq(II)}$ = Energy transferred from the intact loop steam generator secondary during the extension of the equilibration stage.

The energy release rates during this period are calculated as:

$$\Delta E_{dep(II)} = E_{av(II)} \left(\frac{\Delta p}{p^{*} - PLO} \right) + \Delta E_{decay} + \Delta E_{RC}$$

where

$\Delta E_{dep(II)}$ = The total energy removed from primary and secondary systems through boiloff during the time increment

ΔE_{decay} = The core heat added during the time increment

ΔE_{RC} = The energy transferred from the reactor coolant during the time increment (Note: The reactor coolant is assumed saturated.)

Thus, the mass required to remove the energy ($\Delta E_{dep(II)}$) by boiling at total containment pressure is calculated as:

$$\Delta M = \frac{\Delta E_{dep(II)}}{h_g - h_{inj}}$$

Insert D (cont'd)
where

h_g = saturated water vapor enthalpy at total containment pressure

h_{inj} = injected water enthalpy

ECCS Spillage

Spillage during the post reflood is adjusted by LOCTIC as the difference between the pumped injection flow rate and the break effluent flow rate. This adjustment is necessary to account for the discrepancy between the actual pumped injection flow rate and the pumped injection flow rate assumed in the calculation of mass and energy release. LOCTIC increases spillage mass and energy by surplus injection if actual injection exceeds the assumed injection and decreases spillage mass but leaves spillage energy unchanged if actual injection does not meet the assumed injection. In this manner, LOCTIC conserves the mass conservatively.

INSERT E

Unit 1

The release of the core sensible heat is included in the mass and energy release model as described in section 6.2.1.1.1.3.

Unit 2

of the amount of water in the core and determines whether boiling or film heat transfer is occurring. This relationship is conservative because more heat is transferred to the reactor coolant during blowdown and, hence, to the containment than would be expected to occur. The program adds fission product decay heat and power coastdown heat to the core sensible heat, which is transferred to the coolant for transport to the containment through the break. Heat from the zirconium-water reaction is negligible and not included in containment analysis.

Reactor Coolant System Hot Metal

Insert F

Sensible heat is transferred from the RCS hot metal to the reactor coolant. This is a transient heat transfer calculation, and the Dusinberre numerical method,[4] discussed below, is used for solution of the problem. The RCS hot metal is divided into the following cases for analysis:

1. Ruptured loop piping.
2. Ruptured loop metal, two cases.
3. Other loop piping.
4. Other loop metal, two cases.
5. Reactor vessel head.
6. Reactor vessel shell and thermal shield.
7. Reactor vessel bottom.
8. Pressurizer and surge line.

The metal of the valves, pumps, and steam generator heads is included in the two thick metal cases for the ruptured loop and other loops. Table 6.2-5 presents the equivalent thicknesses and amounts of metal to be treated by the Dusinberre method for the RCS hot metal. When a metal, as listed above, is covered with water, the surface temperature is conservatively set equal to the water temperature. When the metal is uncovered, heat outflow is controlled by an input heat transfer coefficient.

Accumulators

Insert G

The water flow from the accumulators to the RCS is calculated on the basis of differential pressure and an input flow coefficient. A differential

INSERT F

Unit 1

Heat released from the RCS hot metal is included in the mass and energy release model as discuseed in Section 6.2.1.1.1.3.

Unit 2

INSERT G

Unit 1

Water flow from the accumulators is included in the mass and energy release model as described in Section 6.2.1.1.1.3.

Unit 2

mass transfer is calculated and added to the reactor coolant inventory. T' loss of the accumulator water results in an increase in the accumulator gas volume. The pressurized nitrogen gas is assumed to be a perfect gas and is expanded to the new volume, with a resulting change in the accumulator pressure. When the accumulator water is completely discharged, the pressurized gas exhausts to the containment.

High-Head Safety Injection

Insert H →

Safety injection becomes effective 30 sec after a LOCA, at which time water is transferred to the RCS as a function of RCS pressure. Differential mass and heat flows are calculated and added to the coolant inventory. High-head safety injection is accomplished by the charging pumps. Appropriate delay times for recc)t of safeguards signals, valve operation, and pump start are inputs to the LOCTIC program.

Low-Head Safety Injection

The treatment of LHSI is similar to high-head safety injection, with a system discharge curve used to calculated differential mass flow to the RCS. This mass and associated heat are added to the reactor coolant inventory.

Reactor Coolant Blowdown

Insert I →

A summation is made at each time interval of all heat and mass inputs to, and outputs from, the RCS, and new system conditions are established. A differential discharge of mass and heat from the RCS to the containment is calculated. It is assumed that this discharge reaches equilibrium with the containment steam-air atmosphere, and that any water remaining as liquid after flashing takes place falls to the containment floor where it mixes with water on the containment floor.

Mass and energy release rates during reflood are discussed in Appendix 6A. Sections 6A.1 and 6A.2 analyze LOCTIC-calculated mass and energy release rates for a cold-leg DER an hot-leg DER, respectively. Section 6A.3 [illegible] described in WCAP-8312A.[3]

INSERT H

Unit 1

The effect of high head safety injection flow is accounted for in the mass and energy release model as discussed in Section 6.2.1.1.1.3.

Unit 2

INSERT I

Unit 1

The mass and energy release model for the blowdown and reflood phases of the accident are described in Section 6.2.1.1.1.3.

Unit 2

equilibrium with the containment atmosphere, whereas the large water droplets approach 99% of equilibrium with the containment atmosphere. The LOCTIC computer program conservatively assumes that all of the spray droplets reach 90% of equilibrium with the containment steam-air atmosphere for containment analysis and 100% for net positive suction head (NPSH) analysis. Heat transfer between the quench and recirculation sprays, and the containment atmosphere is computed in each time interval.

The casing cooling pumps inject chilled water from the casing cooling tank to the suction of the outside recirculation spray (ORS) pumps. The primary function of casing cooling is to provide adequate NPSH for the ORS pumps. The chilled water added to the ORS pump suction decreases the recirculation spray temperature, which reduces containment depressurization time.

The capability for ~~the~~ postaccident pressure reduction in the containment structure is a function of the design of the containment depressurization system (Section 6.2.2).

Insert (1)

6.2.1.1.2 Subcompartments

6.2.1.1.2.1 Design Criteria. The containment structure subcompartment walls are designed for the maximum differential pressure developed across the walls should there be a break of a high-energy pipe inside the compartment.

The computer programs THREED and RELAP are used for determining the design pressures for the interior compartments (or rooms) such as the reactor cavity and steam-generator and pressurizer cubicles. In order to calculate the pressure transient within the compartment, THREED and RELAP numerically solve equations defining heat and mass flows into and out of the interior of the compartment. The programs are a mathematical description of the compartment and calculate the pressure effects of reactor coolant discharging into the compartment and the heat and mass flows from the compartment to the main volume of the containment atmosphere. The mass and energy flow rates from a DER of the primary coolant pipe into the compartment are obtained from SATAN V results.

~~1.0 INTRODUCTION~~ Insert (1) (12 pages)

This section presents the long term LOCA mass and energy releases that were generated in support of the steam generator replacement program for North Anna Unit 1. ~~These mass and energy releases are then subsequently used by Stone and Webster Engineering Corporation in the containment integrity analysis.~~

6.2.1.1.1.3 ~~2.0~~ LOCA MASS AND ENERGY RELEASE - ~~NORTH~~ UNIT 1 REPLACEMENT STEAM GENERATORS

~~2.1 LOCA Mass and Energy Release Phases~~

The containment system receives mass and energy releases following a postulated rupture of the RCS. These releases continue through blowdown and post-blowdown.

The LOCA transient is typically divided into four phases:

1. Blowdown - which includes the period from accident initiation (when the reactor is at steady state operation) to the time that the RCS pressure reaches initial equilibrium with containment.

2. Refill - the period of time when the lower plenum is being filled by accumulator and safety injection water. At the end of blowdown, a large amount of water remains in the cold legs, downcomer, and lower plenum. To conservatively consider the refill period for the purpose of containment mass and energy releases, this water is instantaneously transferred to the lower plenum along with sufficient accumulator water to completely fill the lower plenum. This allows an uninterrupted release of mass and energy to containment. Thus, the refill period is conservatively neglected in the mass and energy release calculation.

3. Reflood - begins when the water from the lower plenum enters the core and ends when the core is completely quenched.

4. Post-Reflood (Froth) - describes the period following the reflood transient. For the pump suction break, a two-phase mixture exits the core, passes through the hot

legs, and is superheated in the steam generators. After the broken loop steam generator cools, the break flow becomes two phase.

~~2.2~~ LOCA Mass and Energy Release Analysis

The evaluation model for the long term LOCA double-ended pump suction and double ended hot leg (blowdown phase) mass and energy release calculations was the March 1979 model described in Reference ~~2~~ ~~49~~ 51 This evaluation model has been reviewed and approved by the NRC, and has been used in the analysis of other dry containment plants.

~~2.2.1~~ Break Size and Location

Generic studies have been performed with respect to the effect on the LOCA mass and energy releases relative to postulated break size. The double ended guillotine break has been found to be limiting due to larger mass flow rates during the blowdown phase of the transient. During the reflood and froth phases, the break size has little effect on the releases.

Three distinct locations in the reactor coolant system loop can be postulated for pipe rupture:

1. Hot leg (between vessel and steam generator)
2. Cold leg (between pump and vessel)
3. Pump suction (between steam generator and pump)

The breaks analyzed in support of the double-ended rupture, HLDER steam generator replacement are the ~~double-ended~~ hot leg ~~guillotine break~~ (9.17 ft^2), and the ~~double ended~~ pump suction ~~guillotine break~~ (10.48 ft^2). Break releases have been calculated for the blowdown, reflood, and post-reflood phases of the LOCA for each case analyzed.
double-ended rupture, PSDER

~~The following information provides a discussion on each break location~~. The double-ended hot leg guillotine has been shown in previous studies to result in the highest blowdown mass and energy release rates. Although the core flooding rate would be highest for this break location, the amount of energy released from the steam generator secondary side is minimal because the majority of the fluid which exits the core bypasses the steam generators ~~in~~ by venting directly to containment. As a result, the reflood mass and energy releases are reduced significantly as compared to either the pump suction or cold leg break locations where the core exit mixture must pass through the steam generators before venting through the break. For the hot leg break, generic studies have confirmed that there is no reflood peak ~~as determined by generic studies~~ (i.e., from the end of the blowdown period the containment pressure ~~releases~~ would continually decrease). The double-ended hot leg reflood and post-reflood phase calculations are not required to determine peak containment pressure, but were calculated for use in ~~to support~~ the NPSH analysis of the containment spray system and low head safety injection pumps using the 1975 mass and energy evaluation model described in reference[5A]. ~~The mass and energy releases for the hot leg break blowdown phase have been included in the scope of this containment integrity analysis.~~

The cold leg break location has also been found in previous studies to be much less limiting in terms of the overall containment energy releases. The cold leg blowdown is faster than that of the pump suction break, and more mass is released into the containment. However, the core heat transfer is greatly reduced, and this results in a considerably lower energy release into containment. Studies have determined that the blowdown transient for the cold leg is, in general, less limiting than that for the pump suction break. During reflood, the flooding rate is greatly reduced and the energy release rate into the containment is reduced. Therefore, the containment peak pressure for a cold leg break occurs at the end of blowdown. The cold leg break is not usually performed since the hot leg break is expected to result in the highest blowdown peak pressure, and the pump suction break results in the highest post blowdown energy releases into containment.

The pump suction break combines the effects of the relatively high core flooding rate, as in the hot leg break, and the addition of the stored energy in the steam generators. As a result, the pump suction break yields the highest energy flow rates during the post-blowdown period.

~~2.2.2~~ Application of Single Failure Criteria

An analysis of the effects of the single failure criteria has been performed on the mass and energy release rates for ~~the double-ended pump suction (DEPS) and hot leg (DEHLG) breaks. For the DEPS~~ each break analyzed.

~~results presented in this report~~. An inherent assumption in the generation of the mass and energy release is that offsite power is lost. This results in the actuation of the emergency diesel generators, required to power the safety injection system. ~~This is not an issue for the blowdown period which is limited by the DEHL break.~~

Two cases have been analyzed for the effects of a single failure. In the case of minimum safeguards, the single failure postulated to occur is the loss of an emergency diesel generator. This results in the loss of one pumped safety injection train, thereby minimizing the safety injection flow. For the case of maximum safeguards, no failure is postulated to occur. The analysis of both maximum and minimum safeguards cases ensures that the effect of all credible single failures is bounded.

~~2.3~~ Significant Modeling Assumptions

The following items ensure that the mass and energy releases are conservatively calculated to maximize energy release to containment:

1. Maximum expected operating temperature of the reactor coolant system

2. Allowance in temperature for instrument error and dead band (+4.0°F)

3. Margin in volume of 3% (which is composed of 1.6% allowance for thermal expansion, and 1.4% for uncertainty)

4. Power level of 2910 MWt (1898 MWt, core + 12 MWt, ~~for~~ pump heat)

5. Allowance for calorimetric error (+2 percent of power)

6. Conservative coefficients of heat transfer (i.e., steam generator primary/secondary heat transfer and reactor coolant system metal heat transfer)

7. Allowance in core stored energy for effect of fuel densification

8. Margin in core stored energy (+15 percent)

9. Allowance for RCS pressure uncertainty (+36 psi)

10. 0% Steam Generator tube plugging level

    - Maximizes reactor coolant volume
    - Maximizes heat transfer area across the SG tubes
    - Lower resistance in loop, therefore increased break flow, lower delta P up stream of break

With respect to transient behavior of the limiting breaks, analyzed, 0% Steam generator tube plugging (SGTP) assumptions bound asymmetric SGTP because of the following:

- For the Double-ended hot leg guillotine, which is the limiting break location for peak pressure, the amount of energy released from the steam generator secondary side is minimal because the majority of the fluid which exits the core bypasses the steam generator ~~in~~ by venting to containment.

- The effects of asymmetric tube plugging on the DEPS case has been assessed and determined to be bounded by the assumption of no tube plugging. This is due to the effects described above as well as the insensitivity of total energy released to tube plugging levels.

- The double-ended pump suction break, confirmed to be the limiting break location for the post blowdown phase of a LOCA, is not impacted by the distribution of tube plugging by virtue of the loop flow split assumption prescribed in the model. The model considers a 95/5 flow split through the loops, where 95% of the flow will go through the broken loop. A sensitivity study documented in Reference 1 confirms that this flow split maximizes total energy release.

11. A constant backpressure has been assumed in the mass and energy model equal to the containment design pressure (45.0 psig).

12. A containment depressurization time of 3600 seconds is assumed consistent with the NRC, Containment Systems Branch, General Design Criterion 38. To satisfy the requirement of GDC 38 to rapidly reduce the containment pressure, the containment pressure for subatmospheric containment should be reduced to below atmospheric pressure within one hour after the postulated accident.

~~2.4~~ Mass and Energy Release Data

~~2.4.1~~ Blowdown Mass and Energy Release Data

The SATAN-VI code is used for computing the blowdown transient, and is the same as that used for the ECCS calculation in Reference 52 ~~2~~. The methodology for the use of this model is described in Reference ~~2~~ 51.

No 4 Tables 6.2-16 ~~2-1~~ and 6.2-77 ~~2-14~~ present the calculated mass and energy releases for the blowdown phase of the break analyzed for the DEPS and DEHL breaks, respectively. ~~The mass and energy release for the double-ended pump suction break and the double-ended hot leg break, given in Table 2-1 and 2-14 terminate 20.8 and 24.0 seconds respectively after the initiation of the postulated accident.~~

~~2.4.2~~ Reflood Mass and Energy Release Data

The WREFLOOD code ~~is~~ used for computing the 198; reflood transient~~, and is~~ a modified version of that used in the ECCS ~~calculation in~~ evaluation model (Reference ~~2~~ 52). The methodology for the use of this model is described in Reference ~~2.~~51

The model assumes a complete mixing condition (i.e., thermal equilibrium) for the steam/water interaction. The complete mixing process, however, is made up of two distinct physical processes. The first is a two phase interaction with condensation of steam by cold injection water. The second is a single phase mixing of condensate and injection water. ~~Since the mass and energy of the steam released has the most important influence on the containment~~

← Insert 2

~~2-6~~

Insert 2

A complete thermal equilibrium mixing condition for the steam and emergency core cooling injection water during the reflood phase has been assumed for each loop receiving injection water. This is consistent with the usage and application of the Reference 1 mass and energy release evaluation model, in recent analyses, e.g. D. C. Cook Docket (Reference 6). Even though the Reference 1 model credits steam/mixing only in the intact loop and not in the broken loop, justification, applicability, and NRC approval for using the mixing model in the broken loop has been documented (Reference 5). This assumption is justified and supported by test data, and is summarized as follows:

51 51 56 56

~~transient, the steam condensation part of the mixing process is theonly part that need be considered.~~

The most applicable steam/water mixing test data has been reviewed for validation of the containment integrity reflood steam/water mixing model. This data is that generated in 1/3 scale tests (Reference 3) 53, which are the largest scale data available and thus most closely simulates the flow regimes and gravitational effects that would occur in a PWR. These tests were designed specifically to study the steam/water interaction for PWR reflood conditions.

From the entire series of 1/3 scale tests, a group corresponds almost directly to containment integrity reflood conditions. The injection flow rates for this group cover all phases and mixing conditions calculated during the reflood transient. The data from these tests were reviewed and discussed in detail in Reference 1. 51 For all of these tests, the data clearly indicates the occurrence of very effective mixing with rapid steam condensation. The mixing model used in the containment integrity reflood calculation is therefore wholly supported by the 1/3 scale steam/water mixing data.

Additionally, the following justification is also noted. The limiting break for the containment integrity peak pressure analysis during the post-blowdown phase is the double ended pump suction break. For this break, there are two flow paths available in the RCS by which mass and energy may be released to containment. One is through the outlet of the steam generator, the other via reverse flow through the reactor coolant pump. Steam which is not condensed by ECC injection in the intact RCS loops passes around the downcomer and through the broken loop cold leg and pump in venting to containment. This steam also encounters ECC injection water as it passes through the broken loop cold leg, complete mixing occurs and a portion of it is condensed. It is this portion of steam which is condensed that is taken credit for in this analysis. This assumption is justified based upon the postulated break location, and the actual physical presence of the ECC injection nozzle. A description of the test and test results is contained in References 1 51 and 3. 53

The methodology previously discussed and described in Reference 1 51 has been utilized and approved on the Dockets for Catawba Units 1 and 2, McGuire Units 1 and 2, Sequoyah Units 1 and 2, Watts Bar Units 1 and 2, Millstone Unit 3, and Beaver Valley Unit 2.

6.2-16
Table~~s~~ 2-2 ~~and 2-8~~ present~~s~~ the calculated mass and energy release for the reflood phase of the Double-Ended Pump Suction break, with minimum ~~and maximum~~ safety injection ~~respectively.~~ A significantly higher discharge occurs during the period the accumulators are injecting (from 26.9 to 52.0 seconds) ~~for the minimum safety injection case and 26.9 to 52.0 seconds for maximum safety injection~~ as illustrated in Table 2-2 ~~and 2-8~~.

6.2-16a 6.2-16
The transient~~s~~ of the principal parameters during reflood are given in Table~~s~~ 2-4 ~~and 2-10 for the minimum and maximum safety injection double ended pump suction break cases. Tables 2-18 and 2-19 provide the minimum and maximum safety injection flow data.~~

~~2.4.3~~ Post-Reflood Mass and Energy Release Data

The FROTH code (Reference ~~4~~ 54) is used for computing the post-reflood transient. The methodology for the use of this model is described in Reference ~~1~~ 51 The mass and energy release rates calculated by FROTH are used in the containment analysis ~~in conjunction with calculations performed by the Stone and Webster~~ LOCTIC code. ~~(i.e. containment backpressure profile, Table 2-17) until the time of containment depressurization.~~ The interface with LOCTIC is described in Section 6.2.1.1.1.2.

Tables 2-7 and 2-13 present the interface data required in the LOCTIC calculations for the cases analyzed.

After depressurization, the mass and energy release from decay heat is based on the 1979 ANSI/ANS Standard, shown in Reference ~~5~~ 55 and the following input:

1. Decay heat sources considered are fission product decay and heavy element decay of U-239 and Np-239.

2. Decay heat power from fissioning isotopes other than U-235 is assumed to be identical to that of U-235.

3. Fission rate is constant over the operating history of maximum power level.

4. The factor accounting for neutron capture in fission products has been taken from Table 10 of ANS (1979).

5. Operation time before shutdown is 3 years.

6. The total recoverable energy associated with one fission has been assumed to be 200 MeV/fission.

7. Two sigma uncertainty (2 times the standard deviation) has been applied to the fission product decay.

Table ~~s 2-3 and 2-9~~ 6.2-16 presents the two phase (froth) mass and energy release data for the double-ended pump suction break ~~, minimum and maximum safety injection cases~~.

~~2.4.4~~ Steam Generator Equilibration

Steam generator equilibration is the process by which reactor coolant energy is removed from the steam generators in stages. The system is initially assumed to be at design pressure at the beginning of the FROTH transient. The energy available to be released for the interval is determined by calculating the energy difference between the design pressure and the (lower) specified pressure by assuming saturated conditions. This energy is then divided by the energy removal rate, resulting in a specific time. The energy is extracted from the steam generators until the available energy content of all steam generators has been exhausted. The rate of release of steam generator energy drops substantially when the broken and intact loop steam generators have equilibrated. All the available energy is conservatively assumed to be released by imposing a mandatory cooldown and subsequent depressurization down to atmospheric pressure at 3600 seconds, i.e., 14.7 psia and 212°F.

~~2.5~~ Sources of Mass and Energy

The sources of mass considered in the LOCA mass and energy release analysis are ~~given in Tables 2-5, 2-11, and 2-15. These sources are~~ the reactor coolant system, accumulators, and pumped safety injection.

11

The energy inventories considered in the LOCA mass and energy release analysis ~~are given in Tables 2-5, 2-12, and 2-16. The energy sources include:~~

1. Reactor Coolant System Water
2. Accumulator Water
3. Pumped Injection Water
4. Decay Heat
5. Core Stored Energy
6. Reactor coolant System Metal
7. Steam Generator Metal
8. Steam Generator Secondary Energy
9. Secondary Transfer of Energy (feedwater into and steam out of the steam generator secondary)

In the mass and energy release data presented, no Zirc-water reaction heat was considered because the clad temperature did not rise high enough for the rate of the Zirc-water reaction heat to be of any significance.

~~System parameters needed to perform confirmatory analyses are provided in Table 2-17.~~ e

The consideration of the various energy sources in the mass and energy release analysis provides assurance that all available sources of energy have been included in this analysis. Thus the review guidelines presented in Standard review Plan Section 6.2.1.3 have been satisfied.

The mass and energy inventories are presented at the following times, as appropriate:

1. Time zero (initial conditions)

/2

2. End of blowdown time

3. End of refill time

4. End of reflood time

5. Time of full depressurizations

6. End of analysis

The methods and assumptions used to release the various energy sources are given in Reference 1, except as noted in section 2.4.1.3, which has been approved as a valid evaluation model by the Nuclear Regulatory Commission.

For cold-leg ruptures, accumulator and ECCS flow necessary to provide sufficient injection water to match the Westinghouse downcomer level and core level transients during core reflood is assumed to flow into the reactor vessel. Any excess accumulator and ECCs flow spills directly out the break.

Mass and energy release rates are calculated by LOCTIC, as discussed in Section 6.2.1.1.1.2.

Pump-suction breaks yield the highest energy flow to the containment during the post-blowdown period. For the pump-discharge break, all the fluid leaving the top of the core passes through the steam generators and may become superheated. However, the core flooding rate (and therefore the rate of fluid leaving the core) is limited to a relatively low value by the resistance of the pump in the broken loop. For a hot-leg break, the core flooding rate is not so restricted and the majority of the fluid leaving the top of the core bypasses the steam generators and is not superheated. Thus, the steam generators add little or no energy to the containment for a pump-discharge or hot-leg break. The pump-suction break, on the other hand, has a relatively high core flooding rate combined with all the fluid passing through the primary side of the steam generators. Pump-suction breaks are therefore characterized by a second peak containment pressure at or after the end of the core reflooding period.

As shown n Table 6.2-13, the highest containment pressure for the pump-suction breaks for Unit 2 is the second peak pressure, which occurs at the end of core reflooding. The highest containment pressure for the hot-leg break and pump-discharge break occurs during the blowdown period. The pump suction peak pressure for Unit 1 also occurs during the blowdown period.

The containment pressure transients for a DER at the three different break locations are shown in Figure 6.2-11. Failure of one quench spray pump is

if safety injection water mixing with the steam is not assumed (Unit 2). When water/steam mixing occurs there is no second peak pressure (Unit 1)

(Unit 2 only) Insert J

assumed for this analysis. (Other failures are discussed below.) Plotted transients for the pump suction, hot-leg and pump-discharge breaks are calculated with an initial containment total pressure and temperature of 13.69 psia and 120°F. These conditions maximize the containment peak pressure.

The peak containment pressure does not depend directly on the service water temperature since the peak pressure occurs prior to or just at the time the recirculation spray subsystem becomes effective. However, the peak containment pressure is maximized by a high initial containment pressure, which is permissible only at low service water temperatures.

Pressure transients for three different size pump-suction ruptures are plotted in Figure 6.2-12. (Unit 2 only)

The break location and size giving the highest containment pressure is a full pump suction double-ended rupture (PSDER) for Unit 2 and a full hot leg double-ended rupture (HLDER) for Unit 1.

For Unit 2, A single-failure analysis for the PSDER is summarized in Table 6.2-14. A diesel-generator failure results in minimum engineered safety features, as described in Section 6.2.1.1. This failure is shown to produce the highest containment pressure of 44.1 psig. Containment pressue transients for the failures analyzed are shown in Figure 6.2-13. The containment pressure for the diesel-generator failure continues to rise after the end of core reflood until the recirculation sprays become effective at 302 sec. A chronology of events for this accident is presented in Table 6.2-15.

Mass and energy release data for the PSDER with minimum ESF are given in Table 6.2-16. The energy distribution in the containment for this accident is presented in Table 6.2-17. The steam condensing coefficient (discussed in Section 6.2.1.1) is shown in Figure 6.2-14. The containment atmosphere and sump temperature transients are shown in Figure 6.2-15.

Insert J

For Unit 1, the containment peak pressure is independent of single failure since the peak occurs before containment depressurization systems start.

The containment sump screens are designed to prevent particles, larger in size than the smallest restriction within the RS system spray nozzles, from entering the RS and the LHSI systems.

As described in Section 3.8.2.7.6, all protective coatings (paints) on exposed concrete and carbon steel surfaces remain intact if subjected to the environment associated with a postulated LOCA. It is therefore unlikely that paint chips will reach the containment floor. However, should paint chips or other debris reach the containment floor near the sump, all particles larger than the smallest restriction within the systems taking suction from the sump will be prevented from entering the sump by the second- and third-stage fine mesh sump screens. Smaller particles and silt that do not settle out on the containment floor away from the sump will be recirculated and will have no adverse effects upon the system components. The turbulent flow characteristics within the piping systems and components are sufficient to maintain the particles in suspension.

Insulation Debris and Transport

X

The following sources are considered to determine the potential origin of debris due to a pipe or equipment failure that results in a LOCA:

1. The containment structure contains no loose insulation. All insulation is (a) metal jacketed with type 304 austenitic stainless steel; and (b) completely encapsulated in a silicone rubber-coated cloth; (c) completely encapsulated in a silicone rubber-coated cloth with additional encapsulation of stainless steel knitted mesh.

2. Insulation in the path of the high energy coolant jet and/or a whipping pipe from the following areas:

   a. Steam-generator cubicles.

   b. Pressurizer cubicle.

   c. Reactor cavity.

   d. Adjacent to the portion of the pressurizer spray piping under the steam-generator and pressurizer cubicles.

3. Supplementary reactor shield material saddles (see Chapter 12 for description) located in the reactor cavity.

4. Particle debris of the type that is uniformly distributed throughout the containment.[23]

5. Failure of non-safety-related equipment within the containment. The debris that this failure could generate would be a small quantity of relatively large and heavy pieces. These items, if they were to reach the containment floor, would sink rapidly and would not be expected to contribute to sump screen blockage.

No sand plugs, sand bags, or loose insulation are located inside the containment.

As discussed above, the steam-generator cubicles contain the largest quantity of insulation that could be exposed to a high-energy coolant jet and/or whipping pipe. No other mechanism for insulation dislodgement has been identified. The area of influence of a high-energy coolant jet is also the largest in the steam-generator cubicles due to the large pipe diameters present. Break areas inside the steam-generator cubicles are discussed in Section 6.2.1.1.2.

The design of the steam-generator cubicles is such that it is very difficult for insulation or debris to exit the cubicles. The personnel entrances at elevations 241 ft 0 in. and 262 ft 10 in. are arranged such that debris would have to transverse a labyrinth-type path with several direction changes and pass through a locked wire mesh door to get outside the cubicles. Once outside the cubicles, the annulus floor grating and concrete would prevent significantly large pieces of insulation or debris from reaching the containment floor or sump.

Any insulation or debris of significant size that would be ejected upward and out of cubicle openings at the operating floor (elevation 291 ft 10 in.) would not likely be transported from the operating floor to the containment sump due to the complex and tortuous path through grating or down stairwells.

→ Insert D

→ Insert A

Insert A, page 1 of 2

Unit 1

In accordance with the requirements of Regulatory Guideline 1.82, reference 41, an insulation debris inventory and transport analysis was performed on Unit 1 containment emergency sump to evaluate the insulation installed on the replacement steam generators. The results of the analysis are shown on Figure 6.2-116 which gives the transient head loss at the containment emergency sump fine mesh screens due to the accumulation of insulation debris.

The replacement steam generators employ a removable encapsulated fiberglass insulation system. The insulation system consists of a light density fiberglass insulating material encapsulated in a tough woven fiberglass cloth to form a blanket or pillow. The pillows are attached together with Velcro® and are covered with a protective and removable stainless steel sheathing. Encapsulation of the fiberglass results in a significant increase in insulation system strength and resistance to the impinging jet forces which emanate from a postulated pipe rupture.

Some insulation fragments or small pieces are carried immediately to the floor by the pressure difference resulting from the rupture. Some fragments are directed away from the containment floor and emergency sump by the pressure difference and some remain in the subcompartment on gratings and horizontal surfaces. After the blowdown terminates, the spray systems begin to wash down the small fragments of insulation. The wash-down process is inefficient and, therefore, time consuming since most of the fragments, due to their location, are not in the direct path of the spray droplets or the flow of water. For conservatism, however, the analysis assumes all transportable debris, including large fragments and "as fabricated" sections, reaches the containment sump screens in 1800 seconds.

The volume of insulation contributing to the total debris inventory is determined by the intersection of the jet emerging from the rupture with the insulated surfaces of equipment. The insulation is assumed to be severely damaged and removed from the equipment up to a distance of seven pipe diameters from the jet as recommended by NUREG/CR-0897, reference 42.

The insulation debris is classified in three categories or damage states according to NUREG/CR-2791, reference 43, and NUREG/CR-0897, reference 42. Thirty percent is small fine fragments that can exist suspended in water. Small fragments readily transport to the screens at any water velocity. Forty percent is fluffy fragments that eventually sink to the floor and transport to the screens provided the water velocity in the vicinity of these fragments, is greater than 0.2 ft/sec. Thirty percent is large fragments and "as fabricated" sections that float and do not interact with or block the sump screens.

Figure 6.2-116 depicts two head loss curves, each employing different assumptions relative to the number of Engineered Safety Features (ESF) trains operating. The maximum case assumes both ESF trains are operating. Since this case exhibits the highest water withdrawal rate from the sump, it also exhibits the greatest head loss across the debris. The minimum case assumes one ESF train is operating. The additional head loss due to insulation debris results in a reduction in the net positive suction head (NPSH) available to all pumps drawing water from the containment emergency sump. The available NPSH for the recirculation spray and low head safety injection pumps is discussed in Section 6.2.2 and 6.3.2, respectively.

→ Unit 2

Metallic reflective insulation consists of multiple layers of 0.002-in. thick type 304 stainless steel foil suitably supported by stainless steel type 304 clips and spacers, enclosed by a casing of 24-gauge type 304 stainless steel. The type 304 stainless steel foil does not break down when wet, but shredding and/or tearing may occur if it is directly impinged upon by a high-energy coolant jet. All material employed in this insulation has a specific gravity greater than 1 and thus would sink and is not likely to reach the screens. If it did reach the sump screens, it would not pass through the screens.

Insulation matting is a flexible lightweight fibrous glass insulation and is composed of 100% selected-grade type E glass fibers fabricated in mat fo[illegible] Insulation matting is not subject to deterioration wh[illegible]t, theref[illegible] direct impingement would cause a breakdown of the mat. [illegible] total s[illegible] the mat [illegible]red, the fibers [illegible]ld sink to the containment floor an[illegible] not be expec[illegible] to block the sump screens as discussed in Reference 24.

Mirror (metallic reflective) insulation houses only multiple layers of 304 stainless steel foil.

Table C of Reference 24 has been revised, and included herein as Table 6.2-41, to show the total weight of debris in each particle size class.

The most plausible path for debris to exit the steam-generator cubicles is through the grating or blowout panel openings at elevation 241 ft 0 in. adjacent to the primary shield wall. Debris passing through these openings would fall to the containment floor no closer than 32 ft from the sump screen. Any debris created by a LOCA in the pressurizer cubicle that could conceivably exit the cubicle would descend to the containment floor at a greater distance than 32 ft from the containment sump. A particle settling analysis[25] indicated that only a small percentage of debris with a specific gravity greater than 1 could be drawn into the sump at this distance.

A significant quantity of debris created by a LOCA in the reactor cavity would not be expected to reach the containment sump since the reactor cavity is completely enclosed on the bottom and sides and any debris blown upward to the operating floor at elevation 291 ft 10 in. would have to migrate a very tortuous and complex path through grating and down stairwells before it could reach the containment sump.

An insignificant amount of debris generated by a LOCA could migrate to the annulus grating at elevation 241 ft 0 in. over the containment sumps. Any debris of significant size that may reach this grating would be restrained by the grating. Any small size debris that might pass through the grating would have minimal effect on sump performance.

Therefore, flow paths do not exist that direct significant quantities of debris into the lower containment in the vicinity of the sump.

Move to Insert 1

The resolution of the concerns associated with the provisions of adequate NPSH under non-debris conditions, and adequate housekeeping practices, are expected to reduce the likelihood of problems during recirculation. However, in the event that LHSI recirculation system problems such as pump cavitation or air entrainment do occur, the operator should have the capability to recognize and contend with these problems. Instrumentation available to monitor recirculation is summarized in Table 6.2-42.

The water entering the containment sump first passes through inclined 1-in. x 1/8-in. grating bars on 1-3/16-in. centers that remove large debris. Downstream of the grating, there are three stages of screening. The first stage is a coarse mesh with 0.615-in. openings for Unit 2 and 0.558-in. openings for Unit 1. The second and third stages are fine meshes with square openings of 0.12-in. The third stage consists of cylindrical screens mounted over the intake to the LHSI and RS subsystem pumps. One cylindrical screen surrounds the intake of each pump. From the cylindrical screens, the flow paths to each of the systems are as follows:

1. Inside Recirculation Spray (IRS) System. From the cylindrical screen, the water travels into the inside recirculation spray pump suction well, through the pump to a 10-in. discharge line, through the recirculation spray cooler shell side, up to the 8-in. diameter,

The assumptions made for NPSHA analyses for the recirculation spray and low head safety injection pumps minimize the energy release to the containment atmosphere and maximize the energy release to the containment floor. Thus, the containment pressure is underestimated, and the containment floor water vapor pressure is overestimated. Since containment pressure is a positive term in the NPSHA equation and the floor water vapor pressure is a negative term, a conservative calculation of NPSH results. Table 6.2-47 identifies the different assumptions. Table 6.2-48 lists the inputs to the LOCTIC code for both cases.

The coldest possible containment sump temperature following a cold-leg DER is presented as a function of time in Figure 6.2-84. A cold-leg DER was selected for this analysis since this location results in the coldest sump water temperature of all possible break locations.

All NPSH values are referenced to the centerline of the first stage of the pump impeller. The flow rates and NPSH for the RS pumps are as follows:

Pump	Flow Rate (gpm)	NPSHR (ft)	Minimum NPSHA (ft) Unit 1	Minimum NPSHA (ft) Unit 2
1. Outside RS	3040	11.0	15.5	16.8
2. Inside RS	3300	9.4	10.4	11.9

6.2.2.3.2.2 NPSH Sensitivity Studies. Tables 6.2-48A through 6.2-48I list the input parameters used in numerous studies to determine the limiting case for NPSH. These studies considered various break locations, service water and RWST temperatures, break sizes, and initial conditions.

6.2.2.3.2.3 Single Failure and Break Locations. Table 6.2-49 shows the minimum NPSHA calculated for both the pump suction double-ended rupture (PSDER) and the hot-leg double-ended rupture (HLDER) with various single failures.

The results indicate the following:

1. HLDERs are more limiting than PSDERs for the recirculation spray pumps.

2. The inside recirculation spray pumps are insensitive to the single-failure assumption.

3. Normal engineered safety features (ESF) and failure of one outside is recirculation spray pump are the most limiting single failures for both the inside and outside recirculation spray pumps.

6.2.2.3.2.4 Service Water Temperature and RWST Temperature. NPSHA is calculated for both the PSLER with minimum ESF and the HLDER with normal ESF for the following temperature conditions to show the effect on NPSH:

1. 50°F RWST and 35°F service water temperature.
2. 50°F RWST and 93°F service water temperature.
3. 40°F RWST and 35°F service water temperature.
4. 40°F RWST and 95°F service water temperature.

Table 6.2-48 lists the remaining input parameters. The results shown in Table 6.2-50 indicate the following:

1. For the recirculation spray pumps, cold service water is more limiting because cold water injection into the pump suction is less effective in increasing NPSHA when the sump water is colder, which it is for the case of colder service water. containment pressure is reduced rapidly, allowing the sump water to become saturated.
2. Warm RWST temperature is more limiting for the IRS pumps since it results in less effective cooling of the water at the pump suction.

6.2.2.3.2.5 Break Size. Since minimum ESF is the most limiting single failure for NPSH to the LHSI pump, further analyses assuming [illegible] for various break locations and break sizes for both recirculation spray and LHSI pumps. Table 6.2-51 shows the minimum NPSHA to the pumps for this analysis. It is concluded that break size has a negligible effect.

6.2 REFERENCES (continued)

10. E. Quandt, "Analysis of Gas-Liquid Flow Patterns," A.I.Ch.E. Preprint No. 47, presented at Sixth National Heat Transfer Conference, Boston, Mass., August 11-13, 1963.

11. K. V. Moore and W. H. Rettig, RELAP4 - A Computer Program for Transient Thermal-Hydraulic Analysis, Aerojet Nuclear Company, ANCR-1127, December 1973.

12. F. J. Moody, "Maximum Two-Phase Vessel Blowdown From Pipes," Journal of Heat Transfer, Trans. of ASME, August 1968.

13. F. J. Moody, Prediction of Blowdown Thrust and Jet Forces, ASME Paper No. 69-HT-31.

14. F. J. Moody, Time-Dependent Pipe Forces Caused by Blowdown and Flow Stoppage, ASME Paper No. 73-FE-23.

15. H. K. Fauske, The Discharge of Saturated Water Through Tubes, Chemical Engineering Progress-Symposium Series, Vol. 61, No. 59, pp. 210-216.

16. Uchida, H., Oyama, A., and Togo, Y., "Evaluation of Post-Incident Cooling Systems of Light-Water Power Reactors," Proceedings of the Third International Conference on the Peaceful Uses of Atomic Energy held in Genevia, August 31 - September 9, 1964, Vol. 13, New York: United Nations 65-104 (A/CONF 28/P/436).

17. Rohsenow and Choi, Heat, Mass, and Momentum Transfer, Prentice-Hall, Englewood Cliffs, NJ, 1961.

18. Deleted.

19. I. E. Idel'chik, Handbook of Hydraulics Resistance, AEC-TR-6630, 1960.

20. Flow of Fluids Through Valves, Fittings, and Pipe, Crane Technical Paper No. 410, 1969.

6.2 REFERENCES (continued)

41. L. F. Parsly, Jr., Design Considerations of Reactor Containment Spray Systems - Part IV, Calculation of Iodine-Water Partition Coefficients, ORNL-TM-2412 Part IV, January 1970.

42. V. Griffiths, "Use of Sprays as a Safeguard in Reactor Containment Structures," Nuclear Safety, Vol. 6, No. 2, pp. 191-194.

43. R. K. Hilliard et al., Removal of Iodine and Particles from Containment Atmospheres by Sprays, BNWL-1244, UC-41, February 1970, p. 159.

44. L. F. Parsly, Jr., Calculation of Iodine-Water Partition Coefficients, ORNL-TM-2412, Part IV, January 1970.

45. Safety Evaluation by the Division of Reactor Licensing, United States Atomic Energy Commission in the Matter of Commonwealth Edison Co., Zion Station, Units 1 and 2, Docket Nos. 50-295/304, August 21, 1968.

46. J. C. Griess and Baccarella, Design Considerations of Reactor Containment Spray Systems, Part III, The Corrosion of Material in Spray Solutions USAEC Report, ORNL-TM-2412, Part III, Oak Ridge National Laboratory, December 1969.

47. J. C. Griess, and G. E. Creek, Design Considerations of Reactor Containment Spray Systems, Part X, Corrosion Tests with Low pH Spray Solution, USAEC Report ORNL-TM-2412, Part X, Oak Ridge National Laboratory.

48. R. H. Leyse, PWR FLECHT Group II Test Report, WCAP-7544, September 1970, Figure 4.3.

Insert B

Insert R

49. USNRC, Water Sources For Long-term Recirculation Cooling Following a Loss-of-Coolant-Accident, USNRC Regulatory Guideline 1.82, November 1985.

50. A.W. Serkiz, Containment Emergency Sump Performance, USNRC, NUREG-0897, October 1985.

~~51~~ 57. J. Wysocki and R. Kolbe, Methodology for Evaluation of Insulation Debris Effects, Burns and Roe, Inc. and Sandia National Laboratories, NUREG/CR-2791 and SAND82-7067, September 1982.

~~4.0~~ ~~REFERENCES~~

51. WCAP-10325-A, "Westinghouse LOCA Mass and Energy Release Model For Containment Design - March 1979 Version", April 1979. (Proprietary)

52. "Westinghouse ECCS Evaluation Model - 1981 Version", WCAP-9220-P-A, Rev. 1, February 1982 (Proprietary), WCAP-9221-A, Rev. 1.

53 EPRI 294-2, Mixing of Emergency Core Cooling Water with Steam: 1/3 Scale Test and Summary, (WCAP-8423), Final Report June 1975.

54. WCAP-8264-P-A, Rev. 1, "Topical Report Westinghouse Mass and Energy Release Data for Containment Design", August 1975. (Proprietary)

55. ANSI/ANS-5.1-1979, American National Standard for Decay Heat Power in Light Water Reactors", August 1979.

56. Docket No. 50-315, "Amendment No, 126 Facility Operating License No. DPR-58 (TAC No. 7106), for D. C. Cook Nuclear Plant Unit 1", June 9, 1989.

Ta .2-2

INPUT DATA TO THE LOCTIC PROGRAM NORTH ANNA POWER STATION - UNIT UNITS ~~NO.~~ 1 OR and 2

Reactor and Reactor Coolant System	UNIT 1	UNIT 2
Maximum rated core power including 2% uncertainty (MWt)	2958	2958
Internal energy of reactor coolant water (includes pressurizer) (Btu)	[a] 245.1×10^6	245.1×10^6
Total water in system (lb)	[a] 4.107×10^5	4.092×10^5
Temperature (mass average excluding pressurizer) (°F)	[a] 586.6	586.8
System pressure (psia)	[a] 2.25×10^3	2.28×10^3
Reactor coolant system volume (excluding pressurizer) (ft^3) (a) e	8526	8394
Pressurizer volume total (ft^3)	Not Used	1479.9
1. Water volume (ft^3)	Not Used	932.9
2. Steam volume (ft^3)	Not Used	547
Heat Transfer Data		
Core heat transfer area (ft^2)	Not Used	48,439

b (a) The fluid volume contained in the primary system, calculated at room temperature, reflects the system volume. This volume is calculated from component dimensions, plus a factor of 1.016 for thermal expansion and 1.014 to account for uncertainties. This gives a total factor of 1.03.

a This data is only used for initialization.

Table 6.2-2 (continued)

INPUT DATA TO THE LOCTIC PROGRAM NORTH ANNA POWER STATION - UNIT NO. 1 OR 2

	UNIT 1	UNIT 2
Heat Transfer Data (continued)		
Core average heat transfer coefficients		
1. Full power (Btu/hr-ft^2-°F)	Not Used	6000
2. Pool boiling (Btu/hr-ft^3-°F)	Not Used	300
3. UA - One steam generator (Btu/hr-°F)	Not Used	68.1×10^6
Safety Injection System		
Three gas accumulators - total volume (ft^3)	[c] 4350	4350
1. Water volume per accumulator (ft^3)	[c] 994-1056	994-1056
2. Pressure (psia)	[c] 600-696	600-696
3. Temperature (°F)	[c] 80-120	80-120

Safety injection - charging pumps and low safety injection pumps - curves of flow vs reactor pressure are input for conditions of normal and minimum safeguards.

One or two pumps for minimum or normal safeguards, respectively, for both high and low head safety injection systems.

[c] Used in the calculation of nitrogen released to containment only.

Table 6.2-5

REACTOR COOLANT SYSTEM HOT METAL CASES - INPUT DATA FOR LOCTIC

UNIT2

Cases	Thickness (in.)	Weight (lb)	Heat Transfer Coefficient (Btu/hr-ft^2-°F)
1 Ruptured loop metal	4.5	38,000	40.0
2 Intact loop metal	4.5	76,000	10.0
3 Other ruptured loop metal	5.13	33,000	40.0
4 Other loops thick metal	5.13	66,000	10.0
5 Ruptured loop piping	2.5	67,000	40.0
6 Other loops piping	2.5	134,000	10.0
7 Reactor vessel head	8.0	309,000	40.0
8 Reactor vessel shell and thermal shield	6.97	556,100	100.0
9 Reactor vessel bottom	6.57	173,000	2000.0
10 Pressurizer and surge line	5.63	230,540	5.0

Table [illegible]-13

SUMMARY OF RESULTS OF CONTAINMENT ANALYSIS
SPECTRUM OF BREAK SIZES AND LOCATIONS

Break Location	Break Area (ft^2)	Initial Containment Pressure/Temperature (psia/°F)	Peak Containment Pressure (psig) Unit 1	Peak Containment Pressure (psig) Unit 2	Time of Peak Pressure (sec) Unit 1	Time of Peak Pressure (sec) Unit 2	Peak Containment Temperature (°F) Unit 1	Peak Containment Temperature (°F) Unit 2
Pump suction	10.48	13.69/120	42.4	43.0	18.8	192.0	270.2	270.1
Pump suction	6.28	13.69/120	—	43.0	—	192.0	—	270.3
Pump suction	3.00	13.69/120	—	42.7	—	198.0	—	269.8
Pump discharge	8.25	13.69/120	—	41.3	—	11.1	—	268.5
Hot leg	9.17	13.69/120	43.9	41.2	18.6	11.9	272.3	268.5

Notes:

1. All cases for Unit 2 assume failure of one quench spray pump.
2. All breaks are full double-ended ruptures (DERs) except the 6.28-ft^2 and 3.00-ft^2 pump suction breaks, which are limited displacement ruptures.
3. Peak containment pressure and peak containment temperature occur concurrently.
4. Results are based on power uprate parameters listed in Tables 6.2-1 and 6.2-2.

Table 6.2-14

SUMMARY OF RESULTS OF CONTAINMENT INTEGRITY
SINGLE-FAILURE ANALYSIS FOR PUMP SUCTION DER
UNIT 2

Single Failure	Peak Containment Pressure (psig)	Time of Peak Pressure (sec)	Peak Containment Temperature (°F)
None	41.6	192.0	268.1
Quench spray pump	43.0	192.0	270.1
Diesel generator	44.1	302.0	271.7

Notes: 1. Initial containment pressure and temperature are 13.69 psia and 120°F, respectively.

2. Peak containment pressure and peak containment temperature occur concurrently.

Table 5.2-15A

Accident Chronology

Pump Suction DER with Failure of Diesel Generator (Limiting Case for Containment Depressurization)
Unit 1

Time (sec)	Event
0.0	Accident occurs
2.2	Containment depressurization actuation signal
18.4	First containment peak pressure occurs
20.8	End of blowdown; core reflooding begins; safety injection pumps become effective
42.0	Accumulators empty
62.5	Quench spray subsystem & casing cooling become effective
253.4	Core reflooding ends; post-reflood frothing begins
304.0	Recirculation spray system becomes effective
1595.8	Post-reflood frothing ends
3370.0	Containment pressure becomes subatmospheric
3420.0	Safety injection pumps switch to recirc. mode
5350.0	Quench spray pumps stop; RWST empty
6340.0	Subatmospheric peak containment pressure occurs

Table 6.2-15 B

ACCIDENT CHRONOLOGY

PUMP SUCTION DER WITH FAILURE OF A DIESEL GENERATOR
(LIMITING CASE FOR MAXIMUM POST-LOCA CONTAINMENT PRESSURE)
UNIT 2

Time (sec)	Event
0.0	Accident occurs.
2.4*	Containment depressurization actuation signal
15.1	First containment peak pressure occurs.
24.4	End of blowdown; core reflooding begins.
30.0	Safety injection pumps become effective.
37.0	Accumulators empty.
62.5	Quench spray subsystem and casing cooling become effective.
204	Core reflooding ends; post-reflood frothing begins.
302	Second peak containment pressure occurs; recirculation spray system becomes effective.
1277	Post-reflood frothing ends.
2830	Containment pressure becomes subatmospheric.
3460	Safety injection pumps switch to recirculation mode.
5350	Quench spray flow stops, RWST is empty.
5710	Subatmospheric third peak containment pressure occurs.

*For a large break LOCA, the time of quench spray pump start is independent of the time the CDA signal occurs. See page 6.2-89.

Table 6.2-16

BLOWDOWN MASS AND ENERGY RELEASE
DOUBLE-ENDED PUMP SUCTION - MIN SI
UNIT 1

TIME	BREAK PATH NO.1 FLOW		BREAK PATH NO.2 FLOW	
SECONDS	LBM/SEC	THOUSAND BTU/SEC	LBM/SEC	THOUSAND BTU/SEC
0.000	0.0	0.0	0.0	0.0
0.100	41759.4	23055.4	21093.4	11584.6
0.200	45782.3	25483.4	23575.4	12962.4
0.400	47589.0	27079.8	23234.7	12809.8
0.500	46036.5	26528.1	22364.8	12339.3
0.700	45400.4	26751.1	20526.5	11334.3
1.00	42478.4	2571[illegible].7	19208.3	10617.0
1.50	38676.7	24[illegible]	18414.4	10177.7
1.90	35152.2	2288[illegible].2	17926.4	9902.2
2.20	30333.9	206[illegible]1.4	17446.3	9634.8
2.50	21280.3	1[illegible]063.0	16717.5	9232.6
2.70	17951.9	12942.9	16230.2	8965.9
2.90	15958.2	11609.0	15592.8	8617.8
3.60	12651.1	9402.1	14046.6	7780.0
4.20	11137.7	8355.5	13081.8	7257.7
4.60	10585.9	7903.9	13644.8	7578.8
5.40	10145.8	7454.0	12856.0	7150.2
6.00	10151.4	7329.1	12542.9	6984.4
6.40	10614.5	7759.0	12374.9	6859.9
6.80	8702.3	7082.5	12005.8	6679.6
7.00	8498.7	6895.2	11854.3	6594.2
8.20	8713.5	6478.9	10845.0	6025.3
9.00	8143.4	6029.7	10195.1	5658.1
10.8	6366.8	5015.7	8779.7	4865.6
12.2	5373.6	4318.5	7761.9	4301.6
13.0	4840.6	3871.3	6890.0	3819.6
14.0	4243.6	3353.0	6511.4	3303.2
16.0	3277.4	2881.4	5227.1	2305.3
16.2	3138.1	2838.4	5654.2	2486.1
16.4	2995.7	2815.0	4870.9	2172.2
17.0	2424.9	2717.9	4672.9	2120.5
17.4	1835.2	2276.9	3310.5	1471.3
18.4	951.7	1214.9	1562.7	897.2
19.6	337.5	434.7	614.7	416.7
20.8	0.0	0.0	0.0	0.0

REFLOOD

Table 6.2-16 (Cont'd)

BLOWDOWN MASS AND ENERGY RELEASE
DOUBLE-ENDED PUMP SUCTION - MIN SI
UNIT 1

TIME	BREAK PATH NO.1 FLOW		BREAK PATH NO.2 FLOW	
SECONDS	LBM/SEC	THOUSAND BTU/SEC	LBM/SEC	THOUSAND BTU/SEC
20.8	0.0	0.0	0.0	0.0
21.6	0.0	0.0	0.0	0.0
22.5	61.0	71.8	0.0	0.0
24.8	127.0	149.7	0.0	0.0
25.8	147.0	173.[illegible]	0.0	0.0
26.8	349.4	413.1	3296.6	433.6
27.4	443.0	524.7	4272.7	579.9
27.9	452.9	536.6	4357.6	598.5
28.9	444.4	526.4	4275.9	591.6
30.9	422.4	500.2	4069.1	568.4
32.9	402.3	476.2	3874.6	546.4
33.9	393.0	465.0	3783.2	536.0
35.9	375.8	444.5	3611.7	516.5
37.9	360.2	426.0	3454.0	498.6
38.8	392.6	464.7	3823.4	497.1
39.9	385.1	455.6	3748.1	488.4
41.9	372.2	440.3	3618.5	473.5
43.9	360.3	426.1	3497.7	459.6
45.9	349.3	413.1	3384.5	446.6
46.1	348.3	411.8	3373.5	445.3
49.9	329.7	389.7	3177.7	422.9
51.9	320.8	379.2	3082.7	412.0
52.9	186.4	219.8	353.0	99.6
53.9	177.4	209.1	365.4	100.3
64.9	161.0	189.7	387.9	102.2
83.9	153.2	180.6	401.2	99.9
85.9	152.7	179.9	402.3	99.6
149.9	135.8	160.1	436.5	91.1
151.9	135.3	159.4	437.5	90.8
167.9	131.0	154.3	445.8	88.7
169.9	130.4	153.7	446.8	88.5
201.9	121.8	143.5	463.3	84.5
207.9	120.1	141.[illegible]	466.4	83.8
239.9	111.7	131.6	482.9	80.4
253.4	108.4	127.7	490.1	79.2

Table 6.2-16 (Cont'd)

POST-REFLOOD ~~BLOWDOWN~~ MASS AND ENERGY RELEASE
DOUBLE-ENDED PUMP SUCTION - MIN SI
UNIT 1

TIME	BREAK PATH NO.1 FLOW		BREAK PATH NO.2 FLOW	
SECONDS	LBM/SEC	THOUSAND BTU/SEC	LBM/SEC	THOUSAND BTU/SEC
253.4	125.0	157.1	508.9	84.0
258.4	125.7	157.9	508.3	83.6
273.4	124.6	156.6	509.4	83.4
278.4	125.3	157.4	508.[illegible]	83.1
293.4	124.2	156.0	509.8	82.9
313.4	124.7	156.6	509.3	82.1
328.4	123.6	155.2	510.4	81.9
333.4	124.2	156.0	509.8	81.5
358.4	123.2	154.8	510.8	80.9
378.4	123.6	155.3	510.4	80.1
393.4	122.4	153.8	511.6	79.9
413.4	123.0	154.6	511.0	79.1
428.4	122.2	153.5	511.8	78.8
433.4	122.8	154.3	511.2	78.4
453.4	122.6	154.0	511.4	77.8
488.4	121.4	152.5	512.6	76.9
523.4	121.8	153.0	512.2	75.6
548.4	121.0	152.0	513.0	74.9
568.4	121.3	152.4	512.7	76.8
593.4	120.3	151.1	513.7	76.1
608.4	120.9	151.8	513.1	75.3
633.4	119.9	150.6	514.1	74.6
648.4	120.4	151.2	513.6	73.9
738.4	119.1	149.6	514.9	72.9
753.4	119.5	150.2	514.5	72.1
818.4	118.5	148.8	515.5	71.7
1233.4[1]	118.4	148.8	515.6	71.5
1233.5	67.1	83.7	566.9	81.1
1465.0[2]	67.0	83.6	566.9	81.8

Note: [1] S.G. Broken loop equilibration time @ 30.2 psia
[2] S.G. Intact loop equilibration time @ 30.2 psia

Table 6.2-16 (Cont'd)

POST-REFLOOD

BLOWDOWN MASS AND ENERGY RELEASE
DOUBLE-ENDED PUMP SUCTION - MIN SI
UNIT 1

RATE DATA					
TIME INTERVAL		BLOWDOWN		SPILLAGE	
START (SEC)	END (SEC)	MASS (LBM/SEC)	ENERGY (BTU/SEC)	MASS (LBM/SEC)	ENERGY (BTU/SEC)
1400.0	1650.0	6.7486E+01	8.2736E+04	5.6889E+02	6.4787e+04
1650.0	1900.0	6.5364E+01	7.5710E+04	5.7399E+02	1.0997E+04
1900.0	2150.0	5.9914E+01	6.9276E+04	5.8124E+02	1.1135E+04
2150.0	2400.0	5.5413E+01	6.3976E+04	5.8[illegible]E+02	1.1235E+04
2400.0	2650.0	5.1564E+01	5.9457E+04	5.9020E+02	1.1306E+04
2650.0	2900.0	4.8273E+01	[illegible].5606E+04	5.9261E+02	1.1353E+04
2900.0	3150.0	4.5509E+01	5.2386E+04	5.9394E+02	1.1380E+04
3150.0	3400.0	4.3223E+01	4.9724E+04	5.9438E+02	1.1391E+04
3400.0	3650.0	4.4042E+01	5.0648E+04	5.0867E+02	5.9163E+04
3650.0	3900.0	4.2575E+01	4.8947E+04	5.0698E+02	5.9629E+04
3900.0	4150.0	4.1255E+01	4.7419E+04	5.0844E+02	5.9413E+04
4150.0	4400.0	4.0318E+01	4.6337E+04	5.0950E+02	5.9128E+04
4400.0	4650.0	3.9530E+01	4.5424E+04	5.1043E+02	5.8814E+04
4650.0	4900.0	3.8796E+01	4.4576E+04	5.1130E+02	5.8[illegible]98E+04
4900.0	5150.0	3.8086E+01	4.3752E+04	5.1215E+02	5.81[illegible]9E+04
5150.0	5400.0	3.6966E+01	4.2465E+04	5.1341E+02	5.7880E+04
5400.0	5650.0	3.5226E+01	4.0489E+04	5.1524E+02	5.7804E+04
5650.0	5900.0	3.5186E+01	4.0457E+04	5.1530E+02	5.7713E+04
5900.0	6150.0	3.4838E+01	4.0065E+04	5.1567E+02	5.7721E+04
6150.0	6400.0	3.4537E+01	3.9722E+04	5.1597E+02	5.7730E+04
6400.0	6650.0	3.4217E+01	3.9353E+04	5.1629E+02	5.7755E+04
6650.0	6900.0	3.3883E+01	3.8967E+04	5.1664E+02	5.7779E+04
6900.0	7150.0	3.3539E+01	3.8571E+04	5.1698E+02	5.7802E+04
7150.0	7800.0	3.2913E+01	3.7854E+04	5.1766E+02	5.7834E+04

Note: LOCTIC calculates post reflood mass and energy release rates after the intact loop S.G. equilibration time (1465.0 sec) as described in Section 6.2.1.1.2

TABLE ~~5~~ 6.2-16~~a~~ A

NORTH ANNA UNIT 1
PRINCIPAL PARAMETERS DURING REFLOOD
DOUBLE-ENDED PUMP SUCTION - MIN SI / LHSI - 4150 GPM

TIME	FLOODING TEMP	FLOODING RATE	CARRYOVER FRACTION	CORE HEIGHT	DOWNCOMER HEIGHT	FLOW FRACTION	INJECTION TOTAL	INJECTION ACCUMULATOR	INJECTION SPILL	ENTHALPY
SECONDS	DEGREE F	IN/SEC		FT	FT		(POUNDS MASS PER SECOND)			BTU/LBM
20.8	252.1	0.000	0.000	0.00	0.00	0.333	0.0	0.0	0.0	0.00
21.5	248.1	27.148	0.000	0.67	1.54	0.000	7453.1	7453.1	0.0	94.48
21.6	246.2	30.866	0.000	1.04	1.62	0.000	7380.0	7380.0	0.0	94.48
22.8	244.4	2.654	0.319	1.50	[illegible].95	0.410	6944.3	6944.3	0.0	94.48
23.8	244.0	2.506	0.446	1.63	[illegible]	0.442	6654.9	6654.9	0.0	94.48
27.4	242.5	4.748	0.658	2.00	15.59	0.681	5290.0	5290.0	0.0	94.48
28.9	241.3	4.463	0.700	2.18	15.61	0.679	5013.8	5013.8	0.0	94.48
32.3	239.2	3.949	0.758	2.51	15.61	0.669	4565.3	4565.3	0.0	94.48
37.9	237.2	3.485	0.758	2.94	15.61	0.652	3997.9	3997.9	0.0	94.48
38.8	237.0	3.698	0.761	3.00	15.61	0.[illegible]58	4411.2	3789.3	0.0	84.43
46.1	235.7	3.332	0.768	3.50	15.61	0.632	3891.7	3267.2	0.0	83.04
52.9	235.5	2.235	0.760	3.91	15.61	0.518	632.6	0.0	0.0	23.23
54.9	235.8	2.182	0.760	4.00	15.61	0.508	633.2	0.0	0.0	23.23
66.9	238.6	2.037	0.761	4.51	15.61	0.489	633.7	0.0	0.0	23.23
79.3	243.2	1.982	0.763	5.00	15.61	0.489	633.7	0.0	0.0	23.23
93.9	249.8	1.920	0.767	5.56	15.61	0.490	633.7	0.0	0.0	23.23
106.0	255.7	1.868	0.770	6.00	15.61	0.491	633.7	0.0	0.0	23.23
121.9	262.[illegible]	1.802	0.775	6.55	15.61	0.492	633.7	0.0	0.0	23.23
1[illegible].5	267.2	1.746	0.779	7.00	15.61	0.4[illegible]	633.7	0.0	0.0	23.23
151.9	272.1	1.679	0.784	7.51	15.61	0.493	633.7	0.0	0.0	23.23
168.6	276.4	1.611	0.789	8.00	15.61	0.493	633.7	0.0	0.0	23.23
187.9	280.5	1.534	0.795	8.53	15.61	0.493	633.7	0.[illegible]	0.0	23.23
20[illegible].7	283.8	1.459	0.802	9.00	15.61	0.493	633.[illegible]	0.0	0.0	23.23
229.9	287.2	1.368	0.812	9.53	15.61	0.493	633.8	0.0	0.0	23.23
253.4	290.1	1.277	0.825	10.00	15.61	0.493	633.8	0.0	0.0	23.23

Table 6.2-16B

MASS AND ENERGY RELEA S TO CONTAINMENT
PUMP SUCTION DER WITH FAILURE OF A DIESEL GENERATOR
(LIMITING CASE FOR MAXIMUM POST-LOCA CONTAINMENT PRESSURE)[a]

UNIT 2

RATE DATA						INTEGRATED DATA				
TIME INTERVAL		BLOWDOWN		SPILLAGE		TIME	BLOWDOWN		SPILLAGE	
START (SEC)	END (SEC)	MASS (LBM/SEC)	ENERGY (BTU/SEC)	MASS (LBM/SEC)	ENERGY (BTU/SEC)	(SEC)	MASS (LBM)	ENERGY (BTU)	MASS (LBM)	ENERGY (BTU)
0.0	0.1	7.7972E+04	4.6428E+07	0.0	0.0	0.1	7.7972E+03	4.6428E+06	0.0	0.0
0.1	1.0	3.7845E+04	2.2500E+07	0.0	0.0	1.0	4.1858E+04	2.4893E+07	0.0	0.0
1.0	2.0	3.6870E+04	2.2118E+07	0.0	0.0	2.0	7.8728E+04	4.7011E+07	0.0	0.0
2.0	3.0	3.6425E+04	2.1996E+07	0.0	0.0	3.0	1.1515E+05	6.9006E+07	0.0	0.0
3.0	4.0	3.5741E+04	2.1489E+07	0.0	0.0	4.0	1.5893E+05	9.0595E+07	0.0	0.0
4.0	5.0	3.4983E+04	2.1225E+07	0.0	0.0	5.0	1.8592E+05	1.1192E+08	0.0	0.0
5.0	6.0	3.4072E+04	2.0670E+07	0.0	0.0	6.0	2.1999E+05	1.3259E+08	0.0	0.0
6.0	7.0	3.2835E+04	1.9905E+07	0.0	0.0	7.0	2.5282E+05	1.5250E+08	0.0	0.0
7.0	8.0	3.1390E+04	1.8930E+07	0.0	0.0	8.0	2.8422E+05	1.7143E+08	0.0	0.0
8.0	9.0	3.0261E+04	1.8104E+07	0.0	0.0	9.0	3.1448E+05	1.8953E+08	0.0	0.0
9.0	10.0	2.7751E+04	1.6726E+07	0.0	0.0	10.0	3.4223E+05	2.0426E+08	0.0	0.0
10.0	11.0	8.5459E+03	8.2463E+06	0.0	0.0	11.0	3.5079E+05	2.1452E+08	0.0	0.0
11.0	12.0	5.1972E+03	6.0748E+06	0.0	0.0	12.0	3.5599E+05	2.2040E+08	0.0	0.0
12.0	13.0	3.9502E+03	4.7508E+06	0.0	0.0	13.0	3.5994E+05	2.2535E+08	0.0	0.0
13.0	14.0	2.7432E+03	3.3100E+06	0.0	0.0	14.0	3.6268E+05	2.2866E+08	0.0	0.0
14.0	15.0	1.7804E+03	2.1488E+06	0.0	0.0	15.0	3.6446E+05	2.3081E+08	0.0	0.0
15.0	16.0	1.1929E+03	1.4384E+06	0.0	0.0	16.0	3.6566E+05	2.3225E+08	0.0	0.0
16.0	17.0	8.0350E+02	9.6822E+05	0.0	0.0	17.0	3.6646E+05	2.3321E+08	0.0	0.0
17.0	18.0	5.9919E+02	7.2162E+05	0.0	0.0	18.0	3.6706E+05	2.3394E+08	0.0	0.0
18.0	19.0	4.6872E+02	5.6389E+05	0.0	0.0	19.0	3.6753E+05	2.3450E+08	0.0	0.0
19.0	20.0	3.7881E+02	4.5429E+05	0.0	0.0	20.0	3.6791E+05	2.3496E+08	0.0	0.0
20.0	21.0	3.1812E+02	3.8340E+05	0.0	0.0	21.0	3.6822E+05	2.3534E+08	0.0	0.0
21.0	22.0	2.7619E+02	3.2997E+05	0.0	0.0	22.0	3.6850E+05	2.3567E+08	0.0	0.0
22.0	23.0	2.5100E+02	2.9603E+05	0.0	0.0	23.0	3.6875E+05	[illegible]596E+08	0.0	0.0
23.0	24.0	2.0656E+02	2.4334E+05	0.0	0.0	24.0	3.6896E+05	2.3621E+08	0.0	0.0
24.0	25.0	2.3456E+02	3.0211E+05	0.0	0.0	25.0	3.6919E+05	2.3651E+08	0.0	0.0
25.0	26.0	7.3669E+02	9.5773E+05	7.5321E+02	4.5062E+04	26.0	3.6993E+05	2.3747E+08	7.5321E+02	4.5062E+04
26.0	27.0	3.4081E+02	4.4493E+05	2.2734E+03	1.3481E+05	27.0	3.7027E+05	2.3791E+08	3.0268E+03	1.7987E+05
27.0	28.0	3.2525E+02	4.2442E+05	1.2957E+03	7.4314E+04	28.0	3.7060E+05	2.3834E+08	4.3225E+03	2.5618E+05
28.0	29.0	3.7719E+02	4.8840E+05	4.6343E+02	2.7078E+04	29.0	3.7097E+05	2.3883E+08	4.7859E+03	2.8326E+05
29.0	30.0	4.5106E+02	5.8816E+05	2.4876E+02	1.4212E+04	30.0	3.7142E+05	2.3941E+08	5.0347E+03	2.9747E+05
30.0	35.0	4.9467E+02	6.4296E+05	6.1963E+03	3.3700E+05	35.0	3.7389E+05	2.4263E+08	3.6013E+04	1.9829E+06
35.0	40.0	4.9550E+02	6.4202E+05	2.3479E+03	1.2855E+05	40.0	3.7637E+05	2.4584E+08	4.7856E+04	2.6254E+06
40.0	45.0	4.9315E+02	6.3801E+05	2.1797E-01	2.1000E+01	45.0	3.7884E+05	2.4903E+08	4.7857E+04	2.6258E+06
45.0	50.0	4.9177E+02	6.3524E+05	8.5448E+01	8.0312E+03	50.0	3.8130E+05	2.5221E+08	4.8284E+04	2.6659E+06

[a] Rate data are constant over the time interval and are equal t integrated data over the time interval divided by the duration of t

~~Unit 2 only~~

Table 6.2-18 (continued)

MASS AND ENERGY RELEASES TO CONTAINMENT
PUMP SUCTION DER WITH FAILURE OF A DIESEL GENERATOR
(LIMITING CASE FOR MAXIMUM POST-LOCA CONTAINMENT PRESSURE)[a]
UNIT 2

RATE DATA: TIME INTERVAL START (SEC)	END (SEC)	BLOWDOWN MASS (LBM/SEC)	BLOWDOWN ENERGY (BTU/SEC)	SPILLAGE MASS (LBM/SEC)	SPILLAGE ENERGY (BTU/SEC)	INTEGRATED DATA: TIME (SEC)	BLOWDOWN MASS (LBM)	BLOWDOWN ENERGY (BTU)	SPILLAGE MASS (LBM)	SPILLAGE ENERGY (BTU)
[illegible]	55.0	4.8115E+02	6.2084E+05	1.3314E+02	1.1924E+04	55.0	3.8370E+05	2.5531E+08	4.8950E+04	2.7255E+06
[illegible]	60.0	4.6922E+02	6.0481E+05	1.5439E+02	1.3169E+04	60.0	3.8605E+05	2.5833E+08	4.9722E+04	2.7914E+06
[illegible]	72.5	4.4767E+02	5.7584E+05	1.3394E+02	1.0713E+04	72.5	3.9164E+05	2.6553E+08	5.1396E+04	2.9253E+06
[illegible]	85.0	4.2324E+02	5.5249E+05	1.3150E+02	9.6082E+03	85.0	3.9693E+05	2.7232E+08	5.3040E+04	3.0454E+06
[illegible]	105.0	4.0154E+02	5.1282E+05	1.1119E+02	7.5381E+03	105.0	4.0497E+05	2.8257E+08	5.5264E+04	3.1962E+06
[illegible]	140.0	3.7650E+02	4.7762E+05	1.0269E+02	6.2203E+03	140.0	4.1815E+05	2.9929E+08	5.8857E+04	3.4139E+06
[illegible]	190.0	3.4043E+02	4.2720E+05	1.7090E+02	9.1087E+03	190.0	4.3517E+05	3.2065E+08	6.7404E+04	3.8693E+06
[illegible]	240.0	2.3686E+02	2.9677E+05	3.7147E+02	5.1603E+04	240.0	4.4701E+05	3.3549E+08	8.5900E+04	6.4534E+06
[illegible]	290.0	2.0599E+02	2.5272E+05	4.7[illegible]1E+02	6.7116E+04	290.0	4.5731E+05	3.4812E+08	1.0743E+05	9.8092E+06
[illegible]	340.0	2.0061E+02	2.4840E+05	4.3435E+02	6.6320E+04	340.0	4.6734E+05	3.6014E+08	1.2915E+05	1.3125E+07
[illegible]	390.0	1.4491E+02	1.7408E+05	4.8806E+02	8.1559E+04	390.0	4.7468E+05	3.6885E+08	1.5355E+05	1.7203E+07
[illegible]	500.0	1.0738E+02	1.2745E+05	5.2762E+02	9.1049E+04	500.0	4.8649E+05	3.8287E+08	2.1159E+05	2.7307E+07
[illegible]	650.0	9.9198E+01	1.1752E+05	5.3584E+02	9.1921E+04	650.0	5.0137E+05	4.0049E+08	2.9197E+05	4.1092E+07
[illegible]	900.0	9.2013E+01	1.0775E+05	5.4303E+02	9.0274E+04	900.0	5.2438E+05	4.2743E+08	4.2773E+05	6.3460E+07
[illegible]	1450.0	8.1572E+01	9.6149E+04	5.5357E+02	6.2102E+04	1450.0	5.6924E+05	4.8031E+08	7.3220E+05	9.7816E+07
[illegible]	2200.0	6.9816E+01	7.5025E+04	5.7616E+02	1.1037E+04	2200.0	6.1785E+05	5.3458E+08	1.1643E+06	1.0609E+08
[illegible]	2950.0	4.8661E+01	5.1033E+04	5.9609E+02	1.1415E+04	2950.0	6.5435E+05	5.7841E+08	1.4114E+06	1.1466E+08
[illegible]	3730.0	4.1489E+01	4.7699E+04	5.6499E+02	2.6762E+04	3730.0	6.8471E+05	6.1581E+08	2.0521E+06	1.3553E+08
[illegible]	5230.0	3.8420E+01	4.4151E+04	5.1492E+02	4.8221E+04	5230.0	7.4434E+05	6.8204E+08	2.8245E+06	2.0784E+08
[illegible]	6730.0	3.4298E+01	3.9420E+04	5.2013E+02	4.4612E+04	6730.0	7.9579E+05	7.4117E+08	3.6047E+06	2.7470E+08
[illegible]	8230.0	3.2031E+01	3.6808E+04	5.2294E+02	4.2626E+04	8230.0	8.4383E+05	7.9638E+08	4.3691E+06	3.3872E+08
[illegible]	9730.0	3.0276E+01	3.4785E+04	5.2520E+02	4.0755E+04	9730.0	8.8925E+05	8.4856E+08	5.1769E+06	3.9985E+08
[illegible]	11230.0	2.8993E+01	3.3306E+04	5.2690E+02	3.9042E+04	11230.0	9.3274E+05	8.9852E+08	5.9673E+06	4.5844E+08
[illegible]	12730.0	2.7954E+01	3.2106E+04	5.2832E+02	3.7442E+04	12730.0	9.7467E+05	9.4660E+08	6.7597E+06	5.1961E+08

a. Rate data are constant over the time interval and are equal to the change in the integrated data over the time interval divided by the duration of the interval.

Table 6.2-17 A

ENERGY BALANCE
DOUBLE-ENDED PUMP SUCTION - MIN SI / LHSI - 4150 GPM
UNIT 1

	TIME (SECONDS)	0.00	20.60	20.80	253.37	1238.40	1465.00
		ENERGY (MILLION BTU)					
INITIAL ENERGY	IN RCS,ACC, S GEN	678.94	678.94	678.94	678.94	678.94	678.94
ADDED ENERGY	PUMPED INJECTION	0.00	0.00	0.00	3.17	17.67	37.06
	DECAY HEAT	0.00	5.42	5.42	28.20	95.63	108.70
	HEAT FROM SECONDARY	0.00	-2.41	-2.41	-2.41	11.85	11.92
	TOTAL ADDED	0.00	3.01	3.01	28.95	125.15	157.68
	*** TOTAL AVAILABLE ***	678.94	681.95	681.95	707.89	804.09	836.62
DISTRIBUTION	REACTOR COOLANT	245.15	12.11	13.18	30.34	30.34	30.34
	ACCUMULATOR	17.94	14.24	13.16	0.00	0.00	0.00
	CORE STORED	22.59	11.12	11.12	3.92	3.30	3.24
	PRIMARY METAL	126.98	123.65	123.65	96.40	55.30	50.96
	SECONDARY METAL	76.48	75.30	75.30	69.18	41.59	37.10
	STEAM GENERATOR	189.79	191.22	191.22	172.38	111.67	100.99
	TOTAL CONTENTS	678.94	427.64	427.64	372.22	242.20	222.64
EFFLUENT	BREAK FLOW	0.00	254.32	254.32	329.42	555.64	607.73
	ECCS FLOW	0.00	0.00	0.00	0.00	0.00	0.00
	TOTAL EFFLUENT	0.00	254.32	254.32	329.42	555.64	607.73
	*** TOTAL ACCOUNTABLE ***	678.94	681.95	681.95	701.64	797.84	830.37

Table 6.2-17A (Cont'd)

ENERGY BALANCE
DOUBLE-ENDED PUMP SUCTION - MIN SI / LHSI - 4150 GPM
UNIT 1

LOCTIC ENERGY DISTRIBUTION (MILLIONS OF BTU'S)

TIME=	0.0SEC	20.8SEC	1470.0SEC	3600.0SEC	7800.0SEC
H E A T S O U R C E S					
PRIMARY COOLANT	0.0	0.0	0.0	0.0	0.0
PRIMARY HOT MEATL					
PIPING,PUMPS,VALVES	0.0	0.0	0.0	0.0	0.0
REACTOR VESSEL + INTERNALS	0.0	0.0	0.0	0.0	0.0
PRESSURIZER METAL + LINES	0.0	0.0	0.0	0.0	0.0
STEAM GENERATOR WATER	0.0	0.0	0.0	0.0	0.0
PRESSURIZER WATER	0.0	0.0	0.0	0.0	0.0
CORE SENSIBLE HEAT	0.0	0.0	0.0	0.0	0.0
EXTERNAL WATER STORAGE TANK(S)	87.81	87.81	78.09	62.97	45.83
ACCUMULATOR CONTENTS	0.04	0.04	0.00	0.00	0.00
H E A T S I N K S					
CONTAINMENT ATMOSPHERE WATER	3.49	187.48	67.96	17.38	17.12
CONTAINMENT ATMOSPHERE AIR	0.75	3.32	2.62	1.68	1.68
CONTAINMENT FLOOR WATER	0.0	44.62	322.59	470.77	551.00
CONCRETE SINKS	0.0	5.71	79.32	92.74	105.30
CONTAINMENT LINER + METAL SINKS	0.0	15.87	59.18	43.30	29.41
H E A T I N P U T S					
INPUT BLOWDOWN (MERC)	0.0	252.82	468.67	600.20	776.44
INPUT SPILLAGE (MERC)	0.0	0.0	119.48	161.46	405.99
DELAYED FISSIONS	0.0	0.0	0.0	0.0	0.0
DECAY HEAT	0.0	0.0	0.0	0.0	0.0
ZIRCONIUM-WATER REACTION	0.0	0.0	0.0	0.0	0.0
FEEDWATER	0.0	0.0	0.0	0.0	0.0
PUMP AND FAN HEAT	0.0	0.0	0.65	1.80	3.78
TURBINE PLANT STEAM INVENTORY	0.0	0.0	0.0	0.0	0.0
H E A T O U T P U T S					
LHSI & HHSI SUMP SUCTION (MERC)	0.0	0.0	0.0	12.74	274.00
RECIRCULATION COOLERS	0.0	0.0	70.77	153.52	253.45
CONTAINMENT AIR COOLERS	0.0	0.0	0.0	0.0	0.0
HEAT LOST TO ATMOSPHERE	0.0	0.0	0.0	0.0	0.0
STEAM GENERATOR DUMP	0.0	0.0	0.0	0.0	0.0
STEAM GENERATOR RELIEF VALVE	0.0	0.0	0.0	0.0	0.0

1

IMAGE EVALUATION TEST TARGET (MT-3)

150mm

6"

1

IMAGE EVALUATION
TEST TARGET (MT-3)

150mm

6"

1

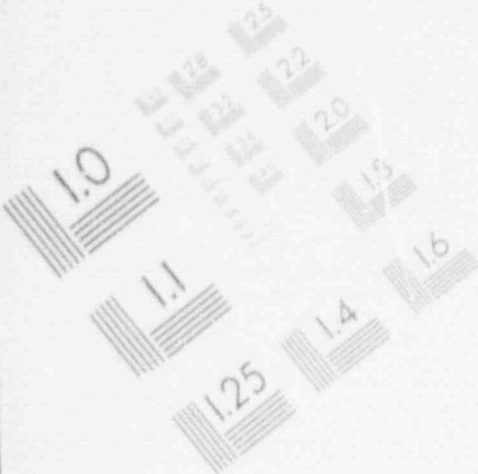
IMAGE EVALUATION TEST TARGET (MT-3)

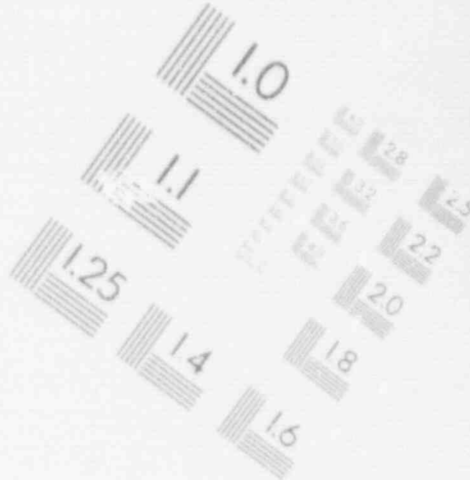
150mm

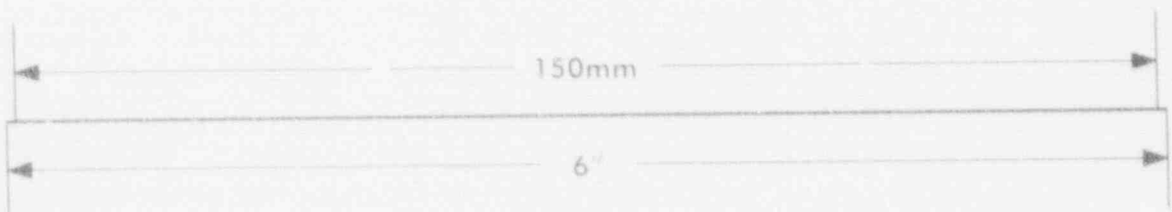
6"

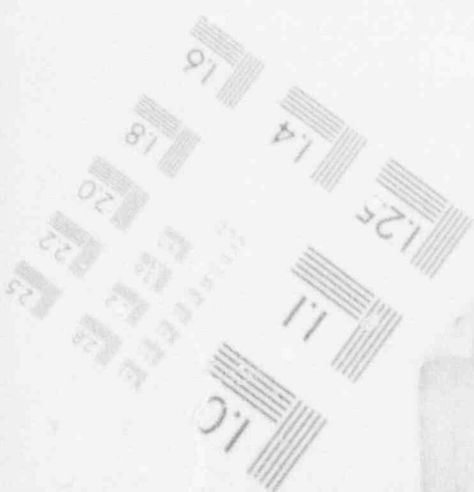
1

IMAGE EVALUATION TEST TARGET (MT-3)









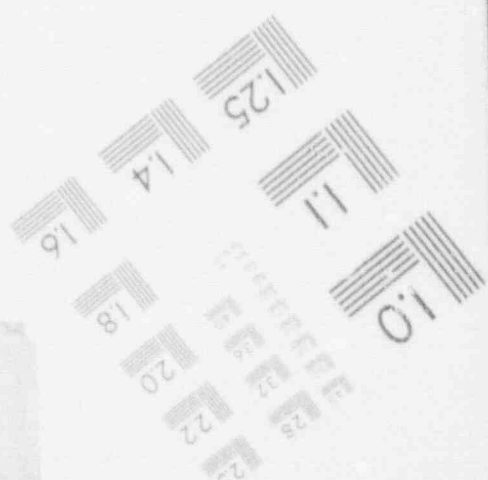


Table 6.2-17 a3

ENERGY DISTRIBUTION PUMP SUCTION DER WITH FAILURE OF A DIESEL GENERATOR
(LIMITING CASE FOR MAXIMUM POST-LOCA CONTAINMENT PRESSURE IN UNITS OF MILLIONS OF BTUS)

UNIT 2

TIME -	0.0SEC	24.4SEC	204.0SEC	1300.0SEC	3610.0SEC
HEAT SOURCES					
PRIMARY COOLANT	217.63	35.41	13.29	13.29	13.29
PRIMARY HOT METAL					
PIPING, PUMPS, VALVES	25.27	24.53	23.91	23.91	23.91
REACTOR VESSEL + INTERNALS	63.35	61.39	55.15	55.[illegible]	55.15
PRESSURIZER METAL + LINES	14.07	13.28	13.21	13.21	13.21
STEAM GENERATOR METAL	116.01	115.38	102.46	102.46	102.46
STEAM GENERATOR SECONDARY WATER	149.82	157.80	111.21	111.21	111.21
PRESSURIZER WATER	27.47	0.0	0.0	0.0	0.0
CORE SENSIBLE HEAT	26.65	7.94	4.73	4.73	4.73
EXTERNAL WATER STORAGE TANK(S)	87.81	87.81	84.92	77.67	61.23
ACCUMULATOR CONTENTS	8.89	4.27	0.01	0.01	0.00
HEAT SINKS					
CONTAINMENT ATMOSPHERE WATER	9.41	180.77	197.94	71.09	7.00
CONTAINMENT ATMOSPHERE AIR	1.51	4.05	4.30	3.06	1.41
CONTAINMENT FLOOR WATER	0.0	40.29	85.51	33[illegible].91	400.69
CONCRETE SINKS	0.0	5.43	23.51	62.94	63.02
CONTAINMENT LINER + METAL SINKS	0.0	16.53	52.53	52.37	28.90
HEAT INPUTS					
INPUT BLOWDOWN	0.0	0.0	0.0	140.94	284.18
INPUT SPILLAGE	0.0	0.0	0.0	92.18	[illegible]25.38
DELAYED FISSIONS	0.0	3.50	3.50	3.50	[illegible].50
DECAY HEAT	0.0	3.53	21.46	21.46	21.46
ZIRCONIUM-WATER REACTION	0.0	0.0	0.0	0.0	0.0
FEEDWATER	0.0	0.0	0.0	0.0	0.0
PUMP AND FAN HEAT	0.0	0.0	0.14	0.69	[illegible].93
TURBINE PLANT STEAM INVENTORY	0.0	0.0	0.0	0.0	0.0
HEAT OUTPUTS					
LHSI & HHSI SUMP SUCTION	0.0	0.0	0.0	0.0	3.98
RECIRCULATION COOLERS	0.0	0.0	0.0	109.33	287.79
CONTAINMENT AIR COOLERS	0.0	0.0	0.0	0.0	0.0
HEAT LOST TO ATMOSPHERE	0.0	0.0	0.0	0.0	0.0
STEAM GENERATOR DUMP	0.0	0.0	0.0	0.0	0.0
STEAM GENERATOR RELIEF VALVE	0.0	0.0	0.0	0.0	0.0

THE BALANCE AT 204.0 SEC IS FOR A HYBRID RUN JUST AS 1 [illegible] B DATA BEGINS.

Table 6.2-47

DEPRESSURIZATION

BREAK EFFLUENT ASSUMPTIONS FOR CONTAINMENT DECOMPRESSION
AND NPSH ANALYSIS

Assumptions	Depressurization Decompression Analysis	NPSH Analysis
1. ‑eak effluent flashing ‑del	Temperature flash - the break effluent is added in its entirety t‑ the containment atmosphere. This maximizes the energy input to the containment atmosphere. The water that cannot be supported as saturated vapor by the available energy in the containment atmosphere falls to the floor at the dewpoint temperature.	Pressure flash - the break effluent expands at constant enthalpy to the containment total pressure. The saturated vapor component goes to the containment atmosphere, and the saturated liquid component goes to the sump, unmixed with the containment atmosphere. This assumption conservatively neglects the evaporative cooling effect that the liquid component most certainly will realize.
2. ‑ing of steam with ‑fety injection in the ‑ld legs (cold leg breaks)	During the reflood period, no credit[a] is taken for mixing between the steam and ECCS injection water. During the post-reflood period, credit is taken for complete mixing beween the steam from the intact loops only (5% of core boiloff) and ECCS injection water. Thus, the energy addition to the containment atmosphere is maximized.	Complete mixing at the break between the break effluent (either steam or two-phase) and the ECCS injection spillage (liquid). Thus, some or all of the steam is quenched, resulting in more mass and energy release to the floor and less to the atmosphere.

Table 6.2-47 (continued)

BREAK EFFLUENT ASSUMPTIONS FOR CONTAINMENT DECOMPRESSION DEPRESSURIZATION
AND NPSH ANALYSIS

Assumptions	Decompression Analysis (Depressurization)	NPSH Analysis
3. Sensible energy transfer to spillage	None [a]	If the spillage temperature is less than the dewpoint temperature, it is heated up to the dewpoint temperature before it is added to the floor. The energy in the containment atmosphere is decremented by the amount of heat transferred to the spillage. Thus, more energy is added to the floor and less to the atmosphere.

[a] Break effluent assumptions for Unit 1 are the same as for Unit 2 except complete mixing occurs between the steam and ECCS injection water during the reflood period.

Table 6.2-48

SENSITIVITY STUDY INPUT PARAMETERS

UNIT 2

Depressurization	NPSH	Depressurization
1. Reactor thermal power (MWt)	2900	2900
2. Initial pressure (psia)		
a. Total	10.0	9.87
b. Air partial	8.9	9.25
3. Initial temperature (°F)		
a. Dry bulb	105	86
b. Dewpoint	105	86
4. Initial service water temperature (°F)	35-95	95
5. RWST		
a. Temperature (°F)	40-50	40
b. Inventory (gal)	475,000	475,000
c. Gal expended at end of QS operation	456,000	456,000
6. Containment free volume (10^6 ft^3)	1.916	1.825
7. Spray thermal effectiveness (%)	100	90
8. Condensing heat transfer correlation	Tagami	Tagami
9. Effective spray start time (sec after CIB signal)		
a. Quench spray	60	60
b. Recirculation spray		
Inside	300	300
Outside	300	300
10. LHSI switchover to sump (gal expended from RWST)	326,000	386,000

Table 6.2-48 (continued)

SENSITIVITY STUDY INPUT PARAMETERS
UNIT 2.

	NPSH	Depressurization
11. System assumptions	150 gpm of QS injection into inside RS pump suction. Inside RS pump flow is 3300 gpm. Outside RS pump is 3640 gpm. LHSI recirculation mode is 4030 gpm for minimum ESF. 50°F (max) 800 gpm (from external source) injected into each outside RS pump suction line.	
12. Containment effective diameter (ft)	121	121
13. Difference between average floor elevation and center-line of first-stage impeller		
a. Inside RS (ft)	6.9	6.9
b. Outside RS (ft)	6.8	6.8
c. LHSI (ft)	6.2	6.2
14. Recirculation phase suction friction loss		
a. Inside RS	1.1 ft at 3300 gpm	
b. Outside RS	4.9 ft at 3640 gpm[a]	
c. LHSI	9.2 ft at 4030 gpm/ pump (minimum ESF)	
15. Mass and energy release rates are presented in Tables 6.2-48A through 6.2-48I.		

---

[a] 2840 gpm from sump and 800 gpm from casing cooling system.

UNIT

Ta 6.2-48A

NORTH ANNA 1 AND 2 NPSH ANALYSIS - HOT LEG DER - NORMAL ESF*

Mass and Energy Release Rates, Integrated Releases, and Integrated Spillage

Time Interval (Sec) Start	End	Mass Release Rate (lbm/Sec)	Energy Release Rate (Btu/Sec)	Integrated Mass Release (lbm)	Integrated Energy Release (Btu)	Integrated Spillage Mass Release (lbm)	Integrated Spillage Energy Release (Btu)
0.0	0.1	110434.62	63097320.	0.110E+05	0.639E+07	0.0	0.0
0.1	1.0	49342.09	28514448.	0.535E+05	0.321E+08	0.0	0.0
1.0	2.0	47918.21	27966912.	0.103E+06	0.600E+08	0.0	0.0
2.0	3.0	46973.06	27816144.	0.150E+06	0.876E+08	0.0	0.0
3.0	4.0	45818.69	27042256.	0.196E+06	0.115E+09	0.0	0.0
4.0	5.0	44549.06	26294832.	0.241E+06	0.141E+09	0.0	0.0
5.0	6.0	43009.37	25379104.	0.284E+06	0.166E+09	0.0	0.0
6.0	7.0	41106.12	24302496.	0.325E+06	0.191E+09	0.0	0.0
7.0	8.0	31890.31	20339952.	0.357E+06	0.211E+09	0.0	0.0
8.0	9.0	9734.69	10390944.	0.367E+06	0.221E+09	0.0	0.0
9.0	10.0	5745.50	6791584.	0.372E+06	0.228E+09	0.0	0.0
10.0	11.0	3262.06	3927136.	0.376E+06	0.232E+09	0.0	0.0
11.0	12.0	1977.50	2371040.	0.377E+06	0.234E+09	0.0	0.0
12.0	13.0	1380.00	1615168.	0.379E+06	0.236E+09	0.0	0.0
13.0	14.0	1049.75	1194400.	0.380E+06	0.237E+09	0.0	0.0
14.0	15.0	844.50	927440.	0.381E+06	0.238E+09	0.0	0.0
15.0	16.0	697.12	738064.	0.382E+06	0.238E+09	0.0	0.0
16.0	17.0	482.56	506368.	0.382E+06	0.239E+09	0.0	0.0
17.0	18.0	89.25	104944.	0.382E+06	0.240E+09	0.0	0.0
18.0	19.0	0.0	0.	0.382E+06	0.240E+09	0.0	0.0
19.0	20.0	0.0	0.	0.382E+06	0.240E+09	0.0	0.0
20.0	22.5	2400.10	528843.	0.388E+06	0.241E+09	0.0	0.0
22.5	25.0	7413.90	1607795.	0.407E+06	0.245E+09	0.0	0.0
25.0	27.5	7028.55	1463657.	0.424E+06	0.249E+09	0.0	0.0
27.5	30.0	6750.45	1356684.	0.441E+06	0.252E+09	0.0	0.0
30.0	35.0	3059.81	741066.	0.460E+06	0.256E+09	0.0	0.0
35.0	40.0	904.77	177965.	0.465E+06	0.257E+09	0.0	0.0
40.0	45.0	874.90	176570.	0.469E+06	0.257E+09	0.0	0.0
45.0	50.0	866.90	178102.	0.474E+06	0.258E+09	0.0	0.0
50.0	55.0	861.27	179433.	0.478E+06	0.259E+09	0.0	0.0
55.0	60.0	852.61	179783.	0.482E+06	0.260E+09	0.0	0.0
60.0	72.5	850.10	181964.	0.493E+06	0.262E+09	0.0	0.0
72.5	85.0	845.28	183960.	0.503E+06	0.265E+09	0.0	0.0
85.0	105.0	840.12	185145.	0.520E+06	0.268E+09	0.0	0.0
105.0	140.0	837.23	186211.	0.550E+06	0.275E+09	0.0	0.0
140.0	190.0	832.37	184479.	0.591E+06	0.284E+09	0.0	0.0
190.0	240.0	831.68	180721.	0.633E+06	0.293E+09	0.0	0.0
240.0	290.0	832.91	175595.	0.674E+06	0.302E+09	0.0	0.0
290.0	360.0	835.03	171130.	0.714E+06	0.311E+09	0.0	0.0
360.0	435.0	840.68	163562.	0.759E+06	0.324E+09	0.0	0.0

Note: Rate data are constant over the time interval and are equal to the change in the integrated data over the interval divided by the duration of the interval.

*The data correspond to a power level of 2900 MWt.

Table 6.2-48A (continued)

UNIT

NORTH ANNA 1 AND 2 NPSH ANALYSIS - HOT LEG DER - NORMAL ESF

Mass [illegible] Energy Release Rates, Integrated Releases, and Integrated Spillage

Time Interval (Sec) Start	End	Mass Release Rate (Lbm/Sec)	Energy Release Rate (Btu/Sec)	Integrated Mass Release (Lbm)	Integrated Energy Release (Btu)	Integrated Spillage Mass Release (Lbm)	Integrated Spillage Energy Release (Btu)
435.0	560.0	854.51	133371.	0.905E+06	0.345E+09	0.0	0.0
560.0	770.0	869.60	137624.	0.109E+07	0.373E+09	0.0	0.0
770.0	1020.0	867.00	124073.	0.130E+07	0.404E+09	0.0	0.0
1020.0	1270.0	854.60	111592.	0.152E+07	0.434E+09	0.0	0.0
1270.0	1520.0	853.72	101875.	0.173E+07	0.459E+09	0.0	0.0
1520.0	1770.0	849.12	93824.	0.194E+07	0.482E+09	0.0	0.0
1770.0	2020.0	828.56	90334.	0.215E+07	0.505E+09	0.0	0.0
2020.0	2270.0	793.78	104601.	0.235E+07	0.532E+09	0.0	0.0
2270.0	2520.0	793.60	112116.	0.255E+07	0.560E+09	0.0	0.0
2520.0	2770.0	793.70	111640.	0.275E+07	0.588E+09	0.0	0.0
2770.0	3020.0	793.36	109660.	0.294E+07	0.615E+09	0.0	0.0
3020.0	3270.0	793.40	107143.	0.314E+07	0.642E+09	0.0	0.0
3270.0	3520.0	793.76	104358.	0.334E+07	0.668E+09	0.0	0.0
3520.0	3770.0	794.07	101964.	0.354E+07	0.693E+09	0.0	0.0
3770.0	4020.0	794.18	99412.	0.374E+07	0.718E+09	0.0	0.0
4020.0	4270.0	794.29	97901.	0.394E+07	0.743E+09	0.0	0.0
4270.0	4520.0	794.69	96409.	0.413E+07	0.767E+09	0.0	0.0
4520.0	4770.0	794.53	94937.	0.433E+07	0.791E[illegible]	0.0	0.0
4770.0	5020.0	794.43	93590.	0.453E+07	0.81[illegible]	0.0	0.0
5020.0	5270.0	794.76	92446.	0.473E+07	0.83[illegible]	0.0	0.0
5270.0	5520.0	794.74	91339.	0.493E+07	0.860E+09	0.0	0.0
5520.0	5770.0	794.77	90316.	0.513E+07	0.883E+09	0.0	0.0
5770.0	6020.0	795.26	89343.	0.533E+07	0.905E+09	0.0	0.0
6020.0	6270.0	794.75	88316.	0.553E+07	0.927E+09	0.0	0.0
6270.0	6520.0	795.13	87455.	0.572E+07	0.949E+09	0.0	0.0
6520.0	6770.0	795.22	86643.	0.592E+07	0.971E+09	0.0	0.0
6770.0	7020.0	795.22	85830.	0.612E+07	0.993E+09	0.0	0.0
7020.0	7270.0	795.28	85074.	0.632E+07	0.101E+10	0.0	0.0
7270.0	7520.0	795.23	84339.	0.652E+07	0.103E+10	0.0	0.0
7520.0	7770.0	795.91	83793.	0.672E+07	0.106E+10	0.0	0.0
7770.0	8020.0	795.24	82971.	0.692E+07	0.108E+10	0.0	0.0
8020.0	8270.0	795.28	82417.	0.712E+07	0.110E+10	0.0	0.0
8270.0	8520.0	796.00	81909.	0.732E+07	0.112E+10	0.0	0.0
8520.0	8770.0	795.27	81169.	0.751E+07	0.114E+10	0.0	0.0
8770.0	9020.0	796.11	80803.	0.771E+07	0.116E+10	0.0	0.0

[illegible]e: Rate data are constant over the time interval and are equal to the change in the integrated data over th[illegible]erval divided by the duration of the interval.

Table 6.2-48B

NORTH ANNA UNIT 1 AND 2 NPSH ANALYSIS - HOT LEG DER - MIN ESF*

[illegible] Energy Release Rates, Integrated Releases, and Integrated Spillage

Time Interval (Sec) Start	End	Mass Release Rated (Lbm/Sec)	Energy Release Rate (Btu/Sec)	Integrated Mass Release (Lbm)	Integrated Energy Release (Btu)	Integrated Spillage Mass Release (Lbm)	Integrated Spillage Energy Release (Btu)
[illegible]	0.1	110634.62	63877520.	0.110E+05	0.639E+07	0.0	0.0
[illegible]	1.0	49942.09	20514448.	0.555E+05	0.321E+08	0.0	0.0
[illegible]	2.0	47918.21	27966912.	0.103E+06	0.600E+08	0.0	0.0
[illegible]	3.0	46973.06	27616144.	0.150E+06	0.876E+08	0.0	0.0
[illegible]	4.0	45818.69	27142256.	0.196E+06	0.115E+09	0.0	0.0
[illegible]	5.0	44549.06	26296832.	0.241E+06	0.141E+09	0.0	0.0
[illegible]	6.0	43008.37	25389104.	0.284E+06	0.166E+09	0.0	0.0
[illegible]	7.0	41186.12	24302496.	0.325E+06	0.191E+09	0.0	0.0
[illegible]	8.0	31880.31	20139952.	0.357E+06	0.211E+09	0.0	0.0
[illegible]	9.0	9754.69	10396944.	0.367E+06	0.221E+09	0.0	0.0
[illegible]	10.0	5745.50	6751504.	0.372E+06	0.228E+09	0.0	0.0
[illegible]	11.0	3262.06	3927136.	0.376E+06	0.232E+09	0.0	0.0
[illegible]	12.0	1907.50	2371040.	0.378E+06	0.234E+09	0.0	0.0
[illegible]	13.0	1350.00	1615168.	0.379E+06	0.236E+09	0.0	0.0
[illegible]	14.0	1049.75	1194400.	0.380E+06	0.237E+09	0.0	0.0
[illegible]	15.0	844.50	927440.	0.381E+06	0.238E+09	0.0	0.0
[illegible]	16.0	697.12	730064.	0.382E+06	0.239E+09	0.0	0.0
[illegible]	17.0	482.56	506368.	0.382E+06	0.239E+09	0.0	0.0
[illegible]	18.0	89.25	104944.	0.382E+06	0.240E+09	0.0	0.0
[illegible]	19.0	0.0	0.	0.382E+06	0.240E+09	0.0	0.0
19.0	20.0	0.0	0.	0.382E+06	0.240E+09	0.0	0.0
[illegible]	22.5	2400.10	520043.	0.388E+06	0.241E+09	0.0	0.0
[illegible]	25.0	7413.90	1607795.	0.407E+06	0.245E+09	0.0	0.0
[illegible]	27.5	7028.55	1469657.	0.424E+06	0.249E+09	0.0	0.0
[illegible]	30.0	6741.37	1356899.	0.441E+06	0.252E+09	0.0	0.0
30.0	35.0	3679.70	707066.	0.460E+06	0.255E+09	0.0	0.0
[illegible]	40.0	721.17	142964.	0.463E+06	0.256E+09	0.0	0.0
[illegible]	45.0	697.31	142477.	0.467E+06	0.257E+09	0.0	0.0
[illegible]	50.0	684.77	143091.	0.470E+06	0.258E+09	0.0	0.0
[illegible]	55.0	678.59	144381.	0.473E+06	0.258E+09	0.0	0.0
55.0	60.0	671.89	145162.	0.477E+06	0.259E+09	0.0	0.0
[illegible]	72.5	665.36	147035.	0.485E+06	0.261E+09	0.0	0.0
[illegible]	85.0	658.24	149172.	0.493E+06	0.263E+09	0.0	0.0
[illegible]	105.0	643.74	165560.	0.506E+06	0.266E+09	0.0	0.0
105.0	140.0	632.96	187541.	0.528E+06	0.273E+09	0.0	0.0
[illegible]	193.0	627.10	174254.	0.560E+06	0.281E+09	0.0	0.0
[illegible]	240.0	632.78	170685.	0.591E+06	0.290E+09	0.0	0.0
[illegible]	290.0	633.18	156672.	0.623E+06	0.298E+09	0.0	0.0
[illegible]	340.0	633.58	154803.	0.655E+06	0.305E+09	0.0	0.0
[illegible]	435.0	[illegible]	149564.	0.716E+06	0.320E+09	0.0	0.0

Note: Rate data are constant over the time interval and are equal to the change in the integrated data over the interval divided by the duration of the interval.

*The data correspond to a power level of 2900 MWt.

UNIT Table 6.2-48B (continued)

NORTH ANNA 1 AND 2 NPSH ANALYSIS - HOT LEG DER - MIN ESF

... and Energy Release Rates, Integrated Releases, and Integrated Spillage

Time Interval (Sec) Start	End	Mass Release Rate (Lbm/Sec)	Energy Release Rate (Btu/Sec)	Integrated Mass Release (Lbm)	Integrated Energy Release (Btu)	Integrated Spillage Mass Release (Lbm)	Integrated Spillage Energy Release (Btu)
[illegible]	360.0	646.33	139293.	0.796E+06	0.337E+09	0.0	0.0
[illegible]	685.0	652.25	132663.	0.877E+06	0.354E+09	0.0	0.0
[illegible]	810.0	654.34	124318.	0.960E+06	0.369E+09	0.0	0.0
[illegible]	970.0	655.01	117952.	0.106E+07	0.388E+09	0.0	0.0
[illegible]	1220.0	656.62	107408.	0.123E+07	0.413E+09	0.0	0.0
[illegible]	1470.0	656.14	99527.	0.139E+07	0.440E+09	0.0	0.0
[illegible]	1720.0	656.67	92418.	0.156E+07	0.463E+09	0.0	0.0
[illegible]	1970.0	656.30	86448.	0.172E+07	0.485E+09	0.0	0.0
[illegible]	2220.0	656.01	81813.	0.188E+07	0.505E+09	0.0	0.0
[illegible]	2470.0	655.05	73868.	0.205E+07	0.524E+09	0.0	0.0
[illegible]	2720.0	654.30	74197.	0.221E+07	0.543E+09	0.0	0.0
[illegible]	2970.0	627.60	70401.	0.237E+07	0.561E+09	0.0	0.0
[illegible]	3220.0	559.17	78690.	0.251E+07	0.580E+09	0.0	0.0
[illegible]	3470.0	555.92	88036.	0.265E+07	0.602E+09	0.0	0.0
[illegible]	3720.0	554.65	92103.	0.279E+07	0.625E+09	0.0	0.0

...: Rate data are constant over the time interval and are equal to the change in the integrated data over the ...erval divided by the duration of the interval.

Table 6.2-48C

NORTH ANNA UNIT 1 AND 2 NPSH ANALYSIS - HOT LEG DER - NORM ESF MIN ECCS*

Mass and Energy Release Rates, Integrated Releases, and Integrated Spillage

Time Interval (Sec) From	To	Mass Release Rate (Lbm/Sec)	Energy Release Rate (Btu/Sec)	Integrated Mass Release (Lbm)	Integrated Energy Release (Btu)	Integrated Spillage Mass Release (Lbm)	Integrated Spillage Energy Release (Btu)
[illegible]	0.1	110434.62	63897570.	0.110E+05	0.639E+07	0.0	0.0
[illegible]	1.0	49342.09	28314448.	0.355E+05	0.321E+08	0.0	0.0
[illegible]	2.0	47918.21	27966912.	0.103E+06	0.600E+08	0.0	0.0
[illegible]	3.0	46973.06	27616144.	0.150E+06	0.876E+08	0.0	0.0
[illegible]	4.0	45818.69	27042256.	0.196E+06	0.115E+09	0.0	0.0
[illegible]	5.0	44349.06	26296837.	0.241E+06	0.141E+09	0.0	0.0
[illegible]	6.0	43000.37	25384106.	0.284E+06	0.166E+09	0.0	0.0
[illegible]	7.0	41106.12	24302494.	0.325E+06	0.191E+09	0.0	0.0
[illegible]	8.0	31000.31	20339752.	0.357E+06	0.211E+09	0.0	0.0
[illegible]	9.0	9754.69	10136344.	0.367E+06	0.221E+09	0.0	0.0
[illegible]	10.0	5745.50	6751584.	0.372E+05	0.228E+09	0.0	0.0
10.	11.0	3262.06	3927136.	0.376E+06	0.232E+09	0.0	0.0
11.	12.0	1981.50	2371940.	0.378E+06	0.234E+09	0.0	0.0
12.	13.0	1380.00	1619178.	0.379E+06	0.236E+09	0.0	0.0
13.	14.0	1049.75	1194400.	0.380E+06	0.237E+09	0.0	0.0
[illegible]	15.0	844.50	927440.	0.381E+06	0.238E+09	0.0	0.0
[illegible]	16.0	697.12	739864.	0.382E+06	0.239E+09	0.0	0.0
16.	17.0	482.36	506418.	0.382E+06	0.239E+09	0.0	0.0
[illegible]	18.0	89.25	104944.	0.382E+06	0.240E+09	0.0	0.0
[illegible]	19.0	0.0	0.	0.382E+06	0.240E+09	0.0	0.0
19	20.0	0.0	0.	0.382E+06	0.240E+09	0.0	0.0
20.	22.5	2400.10	320045.	0.388E+06	0.241E+09	0.0	0.0
[illegible]	25.0	7413.90	1607495.	0.407E+06	0.245E+09	0.0	0.0
[illegible]	27.5	7020.55	1469657.	0.424E+06	0.249E+09	0.0	0.0
[illegible]	30.0	6741.37	1356099.	0.441E+06	0.252E+09	0.0	0.0
[illegible]	35.0	3679.70	709184.	0.470E+06	0.255E+09	0.0	0.0
35.	40.0	721.17	167644.	0.463E+06	0.256E+09	0.0	0.0
40.	45.0	697.31	162477.	0.467E+06	0.257E+09	0.0	0.0
45.	50.0	684.77	143091.	0.470E+06	0.258E+09	0.0	0.0
[illegible]	55.0	678.59	144381.	0.473E+06	0.259E+09	0.0	0.0

Note: Rate data are constant over the time interval and are equal to the change in the integrated data over the interval divided by the duration of the interval.

*The data correspond to a power level of 2900 MWt.

Table 6.2-48C (continued)

UNIT

NORTH ANNA 1 AND 2 NPSH ANALYSIS - HOT LEG DER - NORM ESF MIN ECCS

Mass and Energy Release Rates, Integrated Releases, and Integrated Spillage

Time Interval (Sec) Start	End	Mass Release Rate (Lbm/Sec)	Energy Release Rate (Btu/Sec)	Integrated Mass Release (Lbm)	Integrated Energy Release (Btu)	Integrated Spillage Mass Release (Lbm)	Integrated Spillage Energy Release (Btu)
33.0	60.0	671.89	143162.	0.477E+06	0.259E+09	0.0	0.0
60.0	72.5	663.34	147026.	0.485E+06	0.261E+09	0.0	0.0
72.5	85.0	658.05	149119.	0.493E+06	0.263E+09	0.0	0.0
85.0	103.0	641.29	170041.	0.506E+06	0.266E+09	0.0	0.0
103.0	140.0	634.72	185448.	0.529E+06	0.273E+09	0.0	0.0
140.0	190.0	624.50	185290.	0.560E+06	0.282E+09	0.0	0.0
190.0	240.0	636.57	172400.	0.591E+06	0.291E+09	0.0	0.0
240.0	290.0	637.29	158177.	0.623E+06	0.298E+09	0.0	0.0
290.0	340.0	634.38	162459.	0.655E+06	0.307E+09	0.0	0.0
340.0	433.0	648.43	160704.	0.716E+06	0.322E+09	0.0	0.0
433.0	540.0	646.05	142639.	0.797E+06	0.340E+09	0.0	0.0
540.0	685.0	652.23	120467.	0.870E+06	0.356E+09	0.0	0.0
685.0	810.0	658.52	127850.	0.951E+06	0.372E+09	0.0	0.0
810.0	970.0	656.29	112373.	0.107E+07	0.390E+09	0.0	0.0
970.0	1220.0	651.63	103959.	0.123E+07	0.416E+09	0.0	0.0
1220.0	1470.0	656.35	97807.	0.139E+07	0.440E+09	0.0	0.0
1470.0	1720.0	655.65	91745.	0.154E+07	0.463E+09	0.0	0.0
1720.0	1970.0	654.89	86191.	0.172E+07	0.485E+09	0.0	0.0
1970.0	2220.0	652.38	81421.	0.189E+07	0.505E+09	0.0	0.0
2220.0	2470.0	565.00	78683.	0.203E+07	0.525E+09	0.0	0.0
2470.0	2720.0	557.73	86452.	0.217E+07	0.546E+09	0.0	0.0
2720.0	2970.0	556.75	85169.	0.231E+07	0.569E+09	0.0	0.0
2970.0	3220.0	556.43	87086.	0.244E+07	0.591E+09	0.0	0.0
3220.0	3470.0	556.22	86794.	0.258E+07	0.612E+09	0.0	0.0
3470.0	3720.0	556.40	85239.	0.272E+07	0.634E+09	0.0	0.0

Note: Rate data are constant over the time interval and are equal to the change in the integrated data over the interval divided by the duration of the interval.

Table 6.2-48D

UNIT

NORTH ANNA 1 AND 2 NPSH ANALYSIS - 6 IN. HOT LEG DER - MIN ESF*

Mass and Energy Release Rates, Integrated Releases, and Integrated Spillage

Time Interval (Sec) Start	End	Mass Release Rate (Lbm/Sec)	Energy Release Rate (Btu/Sec)	Integrated Mass Release (Lbm)	Integrated Energy Release (Btu)	Integrated Spillage Mass Release (Lbm)	Integrated Spillage Energy Release (Btu)
[illegible]	0.1	4417.48	[illegible]	0.442E+03	0.256E+06	0.0	0.0
[illegible]	1.0	4260.85	[illegible]	[illegible]	0.246E+07	0.0	0.0
[illegible]	2.0	4173.18	[illegible]	[illegible]	0.491E+07	0.0	0.0
[illegible]	3.0	[illegible]	[illegible]	[illegible]	[illegible]	0.0	0.0
[illegible]	4.0	[illegible]	[illegible]	[illegible]	[illegible]	0.0	0.0
[illegible]	5.0	3693.36	[illegible]	[illegible]	0.117E+08	0.0	0.0
[illegible]	6.0	3537.61	[illegible]	[illegible]	[illegible]	0.0	0.0
[illegible]	7.0	3414.66	[illegible]	[illegible]	[illegible]	0.0	0.0
[illegible]	8.0	3213.33	[illegible]	[illegible]	[illegible]	0.0	0.0
[illegible]	9.0	3130.69	[illegible]	[illegible]	[illegible]	0.0	0.0
[illegible]	10.0	[illegible]	1300340.	[illegible]	0.215E+08	0.0	0.0
[illegible]	11.0	2041.69	[illegible]	[illegible]	0.237E+08	0.0	0.0
[illegible]	12.5	2719.50	[illegible]	[illegible]	[illegible]	0.0	0.0
[illegible]	13.0	2511.02	[illegible]	[illegible]	[illegible]	0.0	0.0
[illegible]	14.0	[illegible]	[illegible]	[illegible]	[illegible]	0.0	0.0
[illegible]	15.0	1957.76	[illegible]	[illegible]	[illegible]	0.0	0.0
[illegible]	16.0	1892.52	[illegible]	[illegible]	[illegible]	0.0	0.0
[illegible]	17.0	1810.08	1115344.	[illegible]	0.311E+08	0.0	0.0
[illegible]	18.0	1853.35	[illegible]	[illegible]	0.322E+08	0.0	0.0
[illegible]	19.0	1842.57	[illegible]	[illegible]	[illegible]	0.0	0.0
[illegible]	20.0	1830.09	[illegible]	[illegible]	[illegible]	0.0	0.0
[illegible]	22.5	[illegible]	[illegible]	[illegible]	0.371E+08	0.0	0.0
[illegible]	25.0	[illegible]	[illegible]	[illegible]	[illegible]	0.0	0.0
[illegible]	27.5	1701.55	[illegible]	0.711E+05	[illegible]	0.0	0.0
[illegible]	30.0	[illegible]	[illegible]	[illegible]	[illegible]	0.0	0.0
[illegible]	35.0	[illegible]	[illegible]	0.837E+05	[illegible]	0.0	0.0
[illegible]	40.0	[illegible]	[illegible]	[illegible]	[illegible]	0.0	0.0
[illegible]	45.0	[illegible]	[illegible]	[illegible]	0.591E+08	0.0	0.0
[illegible]	50.0	[illegible]	[illegible]	[illegible]	[illegible]	0.0	0.0
[illegible]	55.0	[illegible]	[illegible]	[illegible]	[illegible]	0.0	0.0
[illegible]	60.0	[illegible]	[illegible]	[illegible]	[illegible]	0.0	0.0
[illegible]	72.5	[illegible]	[illegible]	[illegible]	[illegible]	0.0	0.0
[illegible]	85.0	[illegible]	[illegible]	[illegible]	0.931E+08	0.0	0.0
[illegible]	105.0	1557.01	[illegible]	[illegible]	0.112E+09	0.0	0.0
[illegible]	140.0	[illegible]	[illegible]	[illegible]	[illegible]	0.0	0.0
[illegible]	190.0	1315.05	[illegible]	[illegible]	[illegible]	0.0	0.0
[illegible]	240.0	[illegible]	[illegible]	[illegible]	[illegible]	0.0	0.0
[illegible]	290.0	[illegible]	[illegible]	[illegible]	[illegible]	0.0	0.0
[illegible]	340.0	[illegible]	[illegible]	[illegible]	[illegible]	0.0	[illegible]
[illegible]	435.0	[illegible]	[illegible]	[illegible]	[illegible]	0.0	[illegible]

Note: Rate data are constant over the time interval and are equal to the change in the integrated data over the interval divided by the duration of the interval.

*The data correspond to a power level of 2900 MWt.

Table 6.2-48D (continued)

NORTH ANNA UNIT 1 AND 2 NPSH ANALYSIS - 6 IN. HOT LEG DER - MIN ESF

Mass and Energy Release Rates, Integrated Releases, and Integrated Spillage

Time Interval (Sec) Start	End	Mass Release Rate (Lbm/Sec)	Energy Release Rate (Btu/Sec)	Integrated Mass Release (Lbm)	Integrated Energy Release (Btu)	Integrated Spillage Mass Release (Lbm)	Integrated Spillage Energy Release (Btu)
[illegible]	560.0	601.65	153047.	0.521E+06	0.255E+09	0.0	0.0
[illegible]	770.0	590.82	139952.	0.645E+06	0.284E+09	0.0	0.0
[illegible]	1070.0	585.62	126317.	[illegible]	0.311E+09	0.0	0.0
[illegible]	1270.0	595.32	115729.	0.941E+06	0.345E+09	0.0	0.0
[illegible]	1570.0	1288.45	227404	0.127E+07	0.402E+09	0.0	0.0
[illegible]	1770.0	655.24	99649.	0.143E+07	0.426E+09	0.0	0.0
[illegible]	2020.0	656.14	91074	0.159E+07	0.449E+09	0.0	0.0
[illegible]	2270.0	656.31	84977.	0.175E+07	0.470E+09	0.0	0.0
[illegible]	2520.0	655.50	78884.	0.192E+07	0.490E+09	0.0	0.0
[illegible]	2770.0	655.40	75913.	0.208E+07	0.509E+09	0.0	0.0
[illegible]	3020.0	653.66	72410.	0.225E+07	0.527E+09	0.0	0.0
[illegible]	3270.0	590.45	[illegible]	0.239E+07	0.545E+09	0.0	0.0
[illegible]	3520.0	557.95	[illegible]	0.253E+07	0.565E+09	0.0	0.0

Rate data are constant over the time interval and are equal to the change in the integrated data over the interval divided by the duration of the interval.

Table 6.2-48E

NORTH ANNA UNIT 1 AND 2 NPSH ANALYSIS - 0.6 HOT LEG DER - MIN ESF*

Mass and Energy Release Rates, Integrated Releases, and Integrated Spillage

Time Interval (Sec) Initial	Time Interval (Sec) Final	Mass Release Rate (Lbm/Sec)	Energy Release Rate (Btu/Sec)	Integrated Mass Release (Lbm)	Integrated Energy Release (Btu)	Integrated Spillage Mass Release (Lbm)	Integrated Spillage Energy Release (Btu)
0.0	0.1	81430.12	47101968.	0.814E+04	0.471E+07	0.0	0.0
0.1	1.0	35292.27	20399900.	0.399E+05	0.231E+08	0.0	0.0
1.0	2.0	34722.81	19771400.	0.745E+05	0.431E+08	0.0	0.0
2.0	3.0	34119.06	20096304.	0.108E+06	0.632E+08	0.0	0.0
3.0	4.0	33930.62	20061800.	0.142E+06	0.832E+08	0.0	0.0
4.0	5.0	33437.06	19901312.	0.176E+06	0.103E+09	0.0	0.0
5.0	6.0	32973.25	19407000.	0.209E+06	0.123E+09	0.0	0.0
6.0	7.0	32067.81	19132872.	0.241E+06	0.142E+09	0.0	0.0
7.0	8.0	30900.31	18667056.	0.273E+06	0.160E+09	0.0	0.0
8.0	9.0	29341.81	17704640.	0.301E+06	0.178E+09	0.0	0.0
9.0	10.0	28767.69	17011600.	0.330E+06	0.195E+09	0.0	0.0
10.0	11.0	23370.62	14807312.	0.353E+06	0.210E+09	0.0	0.0
11.0	12.0	8805.94	8707800.	0.362E+06	0.219E+09	0.0	0.0
12.0	13.0	5744.62	6175216.	0.368E+06	0.225E+09	0.0	0.0
13.0	14.0	3570.44	4299936.	0.371E+06	0.229E+09	0.0	0.0
14.0	15.0	2356.00	2870120.	0.373E+06	0.232E+09	0.0	0.0
15.0	16.0	1620.81	1902560.	0.375E+06	0.234E+09	0.0	0.0
16.0	17.0	1169.50	1368496.	0.376E+06	0.235E+09	0.0	0.0
17.0	18.0	887.69	974400.	0.377E+06	0.236E+09	0.0	0.0
18.0	19.0	699.62	759304.	0.378E+06	0.237E+09	0.0	0.0
19.0	20.0	577.50	601840.	0.378E+06	0.237E+09	0.0	0.0
20.0	22.5	331.33	345513.	0.379E+06	0.238E+09	0.0	0.0
22.5	25.0	0.0	0.	0.379E+06	0.238E+09	0.0	0.0
25.0	27.5	678 .10	1509921.	0.39[illegible]E+06	0.242E+09	0.0	0.0
27.5	30.0	7240.50	1551056.	0.414E+06	0.246E+09	0.0	0.0
30.0	35.0	7368.25	1464711.	0.451E+06	0.253E+09	0.0	0.0
35.0	40.0	2312.81	4[illegible]61.	0.463E+06	0.255E+09	0.0	0.0
40.0	45.0	708.07	139938.	0.466E+06	0.256E+09	0.0	0.0
45.0	50.0	690.12	139403.	0.470E+06	0.257E+09	0.0	0.0
50.0	55.0	186.69	141546.	0.471E+06	0.257E+09	0.0	0.0

Note: Rate data are constant over the time interval and are equal to the change in the integrated data over the interval divided by the duration of the interval.

*The data correspond to a power level of 2900 MWt.

Table 6.2-48E (continued)

NORTH ANNA UNIT 1 AND 2 NPSH ANALYSIS - 0.6 HOT LEG DER - MIN ESF

[illegible] Energy Release Rates, Integrated Releases, and Integrated Spillage

Time Interval (Sec) Start	End	Mass Release Rate (lbm/Sec)	Energy Release Rate (Btu/Sec)	Integrated Mass Release (lbm)	Integrated Energy Release (Btu)	Integrated Spillage Mass Release (lbm)	Integrated Spillage Energy Release (Btu)
[illegible]	60.0	675.86	141040.	0.476E+06	0.258E+09	0.0	0.0
[illegible]	72.5	667.62	143715.	0.485E+06	0.260E+09	0.0	0.0
[illegible]	85.0	661.71	146458.	0.493E+06	0.262E+09	0.0	0.0
[illegible]	105.0	656.34	150070.	0.506E+06	0.265E+09	0.0	0.0
[illegible]	140.0	629.39	169720.	0.528E+06	0.271E+09	0.0	0.0
[illegible]	190.0	631.72	178371.	0.560E+06	[illegible]	0.0	0.0
[illegible]	240.0	632.06	176608.	0.591E+06	0.288E+09	0.0	0.0
[illegible]	290.0	632.27	156380.	0.623E+06	0.296E+09	0.0	0.0
[illegible]	340.0	634.90	155404.	0.655E+06	0.304E+09	0.0	0.0
[illegible]	435.0	639.26	148372.	0.715E+06	0.318E+09	0.0	0.0
[illegible]	540.0	647.74	140886.	0.796E+06	0.336E+09	0.0	0.0
[illegible]	770.0	653.28	120534.	0.894E+06	0.363E+09	0.0	0.0
[illegible]	1020.0	655.69	117796.	0.110E+07	0.392E+09	0.0	0.0
[illegible]	1270.0	656.14	105440.	0.126E+07	0.418E+09	0.0	0.0
[illegible]	1520.0	656.26	98138.	0.143E+07	0.443E+09	0.0	0.0
[illegible]	1770.0	656.59	91291.	0.159E+07	0.466E+09	0.0	0.0
[illegible]	2020.0	656.53	85769.	0.175E+07	0.487E+09	0.0	0.0
[illegible]	2270.0	655.88	81117.	0.192E+07	0.507E+09	0.0	0.0
[illegible]	2520.0	655.35	73075.	0.208E+07	0.527E+09	0.0	0.0
[illegible]	2770.0	654.30	73605.	0.225E+07	0.545E+09	0.0	0.0
[illegible]	3020.0	609.19	70513.	0.240E+07	0.563E+09	0.0	0.0
[illegible]	3270.0	558.10	81035.	0.254E+07	0.583E+09	0.0	0.0
[illegible]	3520.0	555.68	84750.	0.268E+07	0.605E+09	0.0	0.0

[illegible]: Rate data are constant over the time interval and are equal to the change in the integrated data over [illegible]erval divided by the duration of the interval.

Table 6.2-48F

NORTH ANNA UNIT 1 AND 2 NPSH ANALYSIS - 0.326 HOT LEG DER - MIN ESF*

Energy Release Rates, Integrated Releases, and Integrated Releases, and Integrated Spillage

Time Interval (Sec) Start	End	Mass Release Rate (Lbm/Sec)	Energy Release Rate (Btu/Sec)	Integrated Mass Release (Lbm)	Integrated Energy Release (Btu)	Integrated Spillage Mass Release (Lbm)	Integrated Spillage Energy Release (Btu)
0.0	0.1	53607.64	31017376.	0.536E+04	0.310E+07	0.0	0.0
0.1	1.0	24039.43	13596912.	0.270E+05	0.156E+08	0.0	0.0
1.0	2.0	23570.00	13373503.	0.506E+05	0.294E+08	0.0	0.0
2.0	3.0	23093.50	13479016.	0.733E+05	0.429E+08	0.0	0.0
3.0	4.0	22961.94	13631600.	0.964E+05	0.565E+08	0.0	0.0
4.0	5.0	22743.69	13703936.	0.119E+06	0.702E+08	0.0	0.0
5.0	6.0	22963.37	13680960.	0.142E+06	0.839E+08	0.0	0.0
6.0	7.0	22540.50	13543200.	0.165E+06	0.974E+08	0.0	0.0
7.0	8.0	22126.50	13296144.	0.187E+06	0.111E+09	0.0	0.0
8.0	9.0	21603.75	13024320.	0.209E+06	0.124E+09	0.0	0.0
9.0	10.0	21221.54	12729104.	0.230E+06	0.136E+09	0.0	0.0
10.0	11.0	20750.37	12416624.	0.251E+06	0.149E+09	0.0	0.0
11.0	12.0	20259.62	12082064.	0.271E+06	0.161E+09	0.0	0.0
12.0	13.0	19751.62	11729320.	0.291E+06	0.173E+09	0.0	0.0
13.0	14.0	19206.87	11410512.	0.310E+06	0.184E+09	0.0	0.0
14.0	15.0	18841.19	11101840.	0.329E+06	0.195E+09	0.0	0.0
15.0	16.0	17666.94	10520784.	0.346E+06	0.206E+09	0.0	0.0
16.0	17.0	7830.12	6906752.	0.354E+06	0.213E+09	0.0	0.0
17.0	18.0	4621.62	5131696.	0.359E+06	0.218E+09	0.0	0.0
18.0	19.0	3662.31	4156400.	0.362E+06	0.222E+09	0.0	0.0
19.0	20.0	2031.00	3605648.	0.365E+06	0.225E+09	0.0	0.0
20.0	22.5	1753.62	2062361.	0.369E+06	0.230E+09	0.0	0.0
22.5	25.0	895.03	965002.	0.372E+06	0.233E+09	0.0	0.0
25.0	27.5	564.07	520234.	0.373E+06	0.234E+09	0.0	0.0
27.5	30.0	409.10	314700.	0.374E+06	0.235E+09	0.0	0.0
30.0	35.0	6135.26	1419740.	0.405E+06	0.242E+09	0.0	0.0
35.0	40.0	7556.50	1560801.	0.443E+06	0.250E+09	0.0	0.0
40.0	45.0	4576.64	[illegible] 3773.	0.466E+06	0.254E+09	0.0	0.0
45.0	50.0	316.64	1301791.	0.469E+06	0.255E+09	0.0	0.0
50.0	55.0	600.59	1137012.	0.473E+06	0.256E+09	0.0	0.0
55.0	60.0	604.66	1175411.	0.476E+06	0.256E+09	0.0	0.0
60.0	72.5	673.95	1641054.	0.484E+06	0.258E+09	0.0	0.0
72.5	85.0	665.02	1644800.	0.493E+06	0.260E+09	0.0	0.0
85.0	105.0	658.19	1575791.	0.506E+06	0.263E+09	0.0	0.0
105.0	140.0	646.19	1568333.	0.529E+06	0.269E+09	0.0	0.0
140.0	190.0	630.01	1603528.	0.560E+06	0.277E+09	0.0	0.0
190.0	240.0	632.40	1526657.	0.591E+06	0.285E+09	0.0	0.0
240.0	290.0	635.02	1577571.	0.623E+06	0.292E+09	0.0	0.0
290.0	340.0	637.56	1535961.	0.655E+06	0.300E+09	0.0	0.0
340.0	435.0	643.90	1600195.	0.716E+06	0.316E+09	0.0	0.0

Note: Rate data are constant over the time interval and are equal to the change in the integrated data over the interval divided by the duration of the interval.

*These data correspond to a power level of 2900 MWt.

Table 6.2-48F (continued)

NORTH ANNA UNIT 1 AND 2 NPSH ANALYSIS - 0.326 HOT LEG DER - MIN ESF

Mass and Energy Release Rates, Integrated Releases, and Integrated Spillage

Time Interval (Sec) Start	End	Mass Release Rate (Lbm/Sec)	Energy Release Rate (Btu/Sec)	Integrated Mass Release (Lbm)	Integrated Energy Release (Btu)	Integrated Spillage Mass Release (Lbm)	Integrated Spillage Energy Release (Btu)
[illegible]	560.0	645.31	141961.	0.797E+06	0.332E+09	0.0	0.0
[illegible]	770.0	654.49	132183.	0.934E+06	0.340E+09	0.0	0.0
[illegible]	1020.0	654.80	118245.	0.110E+07	0.389E+09	0.0	0.0
[illegible]	1270.0	656.06	105773.	0.126E+07	0.416E+09	0.0	0.0
[illegible]	1520.0	656.50	95585.	0.143E+07	0.440E+09	0.0	0.0
[illegible]	1770.0	656.42	91568.	0.159E+07	0.463E+09	0.0	0.0
[illegible]	2020.0	656.64	84446.	0.175E+07	0.484E+09	0.0	0.0
[illegible]	2270.0	656.00	81333.	0.192E+07	0.505E+09	0.0	0.0
[illegible]	2520.0	655.37	77279.	0.208E+07	0.524E+09	0.0	0.0
[illegible]	2770.0	653.80	73300.	0.225E+07	0.543E+09	0.0	0.0
[illegible]	3020.0	609.32	70619.	0.240E+07	0.561E+09	0.0	0.0
[illegible]	3270.0	558.38	81137.	0.254E+07	0.578E+09	0.0	0.0
[illegible]	3520.0	555.33	89794.	0.268E+07	0.603E+09	0.0	0.0

Note: Rate data are constant over the time interval and are equal to the change in the integrated data over the interval divided by the duration of the interval.

Table 6.2-48G

UNIT

NORTH ANNA 1 AND 2 NPSH ANALYSIS - PUMP SUCTION DER - NORM ESF*

Mass and Energy Release Rates and Integrated Releases

Time Interval (Sec) Start	End	Mass Release Rate (lbm/Sec)	Energy Release Rate (Btu/Sec)	Integrated Mass Release (Lbm)	Integrated Energy Release (Btu)
0.0	0.1	39614.01	46063104.	0.396E+04	0.461E+07
0.1	1.0	37067.37	20834752.	0.404E+05	0.234E+08
1.0	2.0	34140.37	20336256.	0.732E+05	0.437E+08
2.0	3.0	34785.50	20472304.	0.110E+06	0.642E+08
3.0	4.0	34404.69	20402816.	0.145E+06	0.846E+08
4.0	5.0	33943.81	20137168.	0.178E+06	0.105E+09
5.0	6.0	33777.50	19777152.	0.212E+06	0.124E+09
6.0	7.0	32335.44	19044896.	0.244E+06	0.144E+09
7.0	8.0	31224.87	18547776.	0.275E+06	0.162E+09
8.0	9.0	30130.00	17899712.	0.306E+06	0.180E+09
9.0	10.0	29451.94	17304320.	0.335E+06	0.197E+09
10.0	11.0	20028.50	13174416.	0.355E+06	0.211E+09
11.0	12.0	6589.19	6805456.	0.362E+06	0.217E+09
12.0	13.0	4312.50	5140048.	0.366E+06	0.223E+09
13.0	14.0	3137.56	3775904.	0.369E+06	0.226E+09
14.0	15.0	2082.69	2500912.	0.37[illegible]E+06	0.229E+09
15.0	16.0	1470.00	1736960.	0.373E+06	0.231E+09
16.0	17.0	1090.56	1278960.	0.374E+06	0.232E+09
17.0	18.0	835.87	930368.	0.375E+06	0.233E+09
18.0	19.0	744.85	762096.	0.375E+06	0.233E+09
19.0	20.0	676.19	642224.	0.376E+06	0.234E+09
20.0	22.5	558.97	462944.	0.378E+06	0.235E+09
22.5	25.0	799.10	837919.	0.378E+06	0.236E+09
25.0	27.5	1955.72	381507.	0.383E+06	0.237E+09
27.5	30.0	1546.40	599482.	0.387E+06	0.238E+09
30.0	35.0	5909.20	972264.	0.417E+06	0.243E+09
35.0	40.0	6117.22	960136.	0.447E+06	0.248E+09
40.0	45.0	5654.75	929449.	0.476E+06	0.253E+09
45.0	50.0	2418.89	710125.	0.489E+06	0.256E+09
50.0	55.0	683.72	644797.	0.492E+06	0.260E+09

Note: Rate data are constant over the time interval and are equal to the change in the integrated data over the interval divided by the duration of the interval.

*These data correspond to a power level of 2900 MWt.

Table 6.2-48C (continued)

NORTH ANNA UNIT 1 AND 2 NPSH ANALYSIS - PUMP SUCTION DER - NORM ESF

Mass and Energy Release Rates and Integrated Releases

Time Interval (Sec) Start	End	Mass Release Rate (Lbm/Sec)	Energy Release Rate (Btu/Sec)	Integrated Mass Release (Lbm)	Integrated Energy Release (Btu)
55.0	60.0	689.06	633715.	[illegible]	[illegible]
60.0	72.5	692.48	619164.	0.504E+06	0.271E+09
72.5	85.0	696.48	597413.	0.513E+06	0.278E+09
85.0	105.0	703.97	582334.	0.527E+06	0.289E+09
105.0	140.0	708.34	513423.	0.552E+06	0.301E+09
140.0	190.0	721.19	[illegible]	0.588E+06	0.318E+09
190.0	240.0	827.90	[illegible]	0.629E+06	0.344E+09
240.0	290.0	830.87	287102.	0.671E+06	0.355E+09
290.0	340.0	832.34	281446.	0.712E+06	0.373E+09
340.0	390.0	834.97	278410.	0.754E+06	0.387E+09
390.0	500.0	839.45	276424.	0.846E+06	0.417E+09
500.0	650.0	845.33	257591.	0.973E+06	0.455E+09
650.0	900.0	854.44	201975.	0.119E+07	0.505E+09
900.0	1450.0	857.60	183302.	0.166E+07	0.606E+09
1450.0	2200.0	830.76	101255.	0.228E+07	0.682E+09
2200.0	2950.0	790.34	121790.	[illegible]	0.773E+09
2950.0	3730.0	792.33	108926.	0.348E+07	0.857E+09
3730.0	5230.0	793.72	98019.	0.466E+07	0.101E+10
5230.0	6730.0	794.55	90630.	0.587E+07	0.114E+10
6730.0	8230.0	795.13	85247.	0.707E+07	0.127E+10
8230.0	9730.0	795.55	81246.	0.826E+07	0.139E+10
9730.0	11230.0	795.87	77912.	0.945E+07	0.151E+10
11230.0	12730.0	796.14	73309.	0.1064E+08	0.162E+10

Rate data are constant over the time interval and are equal to the change in the integrated data over [illegible]al divided by the duration of the interval.

Table 6.2-48H

UNIT

NORTH ANNA 1 AND 2 NPSH ANALYSIS - PUMP SUCTION DER - MIN ESF*

Mass and Energy Release Rates, Integrated Releases, and Integrated Spillage

Time Interval (Sec) Start	End	Mass Release Rate (Lbm/Sec)	Energy Release Rate (Btu/Sec)	Integrated Mass Release (Lbm)	Integrated Energy Release (Btu)	Integrated Spillage Mass Release (Lbm)	Integrated Spillage Energy Release (Btu)
0.0	0.1	79614.81	46063104.	0.796E+04	0.461E+07	0.0	0.0
0.1	1.0	36047.37	20834752.	0.404E+05	0.234E+08	0.0	0.0
1.0	2.0	34830.37	20336256.	0.752E+05	0.437E+08	0.0	0.0
2.0	3.0	34785.50	20437304.	0.110E+06	0.647E+08	0.0	0.0
3.0	4.0	34484.64	20402816.	0.145E+06	0.846E+08	0.0	0.0
4.0	5.0	33943.81	20137168.	0.178E+06	0.105E+09	0.0	0.0
5.0	6.0	33277.50	19737152.	0.212E+06	0.124E+09	0.0	0.0
6.0	7.0	32335.44	19244896.	0.244E+06	0.144E+09	0.0	0.0
7.0	8.0	31274.87	18547776.	0.275E+06	0.162E+09	0.0	0.0
8.0	9.0	30330.00	17899712.	0.306E+06	0.180E+09	0.0	0.0
9.0	10.0	29451.94	17304520.	0.335E+06	0.197E+09	0.0	0.0
10.0	11.0	20028.56	13174416.	0.355E+06	0.211E+09	0.0	0.0
11.0	12.0	6586.19	6805456.	0.362E+06	0.217E+09	0.0	0.0
12.0	13.0	4312.50	5140048.	0.366E+06	0.223E+09	0.0	0.0
13.0	14.0	3137.56	3725920.	0.369E+06	0.226E+09	0.0	0.0
14.0	15.0	2082.69	2500912.	0.371E+06	0.229E+09	0.0	0.0
15.0	16.0	1478.00	1736960.	0.373E+06	0.231E+09	0.0	0.0
16.0	17.0	1080.56	1270080.	0.374E+06	0.232E+09	0.0	0.0
17.0	18.0	855.87	931368.	0.375E+06	0.233E+09	0.0	0.0
18.0	19.0	744.25	762096.	0.375E+06	0.233E+09	0.0	0.0
19.0	20.0	676.19	642224.	0.376E+06	0.234E+09	0.0	0.0
20.0	22.5	559.12	463021.	0.378E+06	0.235E+09	0.0	0.0
22.5	25.0	501.87	835606.	0.378E+06	0.236E+09	0.0	0.0
25.0	27.5	2252.95	585574.	0.384E+06	0.237E+09	0.0	0.0
27.5	30.0	1243.10	538490.	0.387E+06	0.239E+09	0.0	0.0
30.0	35.0	6740.50	949786.	0.418E+06	0.243E+09	0.0	0.0
35.0	40.0	3133.45	708493.	0.434E+06	0.247E+09	0.0	0.0
40.0	45.0	492.44	633734.	0.436E+06	0.253E+09	0.0	0.0
45.0	50.0	566.10	643347.	0.439E+06	0.256E+09	0.0	0.0
50.0	55.0	613.32	633987.	0.442E+06	0.257E+09	0.0	0.0

Note: Rate data are constant over the time interval and are equal to the change in the integrated data over the interval divided by the duration of the interval.

*These data correspond to a power level of 2900 MWt.

Table 6.2-48H (continued)

UNIT

NORTH ANNA 1 AND 2 NPSH ANALYSIS - PUMP SUCTION DER - MIN ESF

Mass and Energy Release Rates, Integrated Releases, and Integrated Spillage

Time Interval (Sec) Start	End	Mass Release Rate (lbm/Sec)	Energy Release Rate (Btu/Sec)	Integrated Mass Release (lbm)	Integrated Energy Release (Btu)	Integrated Spillage Mass Release (lbm)	Integrated Spillage Energy Release (Btu)
55.0	60.0	622.55	619472.	0.445E+06	0.260E+09	0.0	0.0
60.0	72.5	594.56	590294.	0.453E+06	0.267E+09	0.0	0.0
72.5	85.0	556.33	553036.	0.460E+06	0.274E+09	0.0	0.0
85.0	105.0	513.35	523562.	0.470E+06	0.285E+09	0.0	0.0
105.0	140.0	479.56	485325.	0.487E+06	0.307E+09	0.0	0.0
140.0	190.0	510.79	439651.	0.512E+06	0.324E+09	0.0	0.0
190.0	240.0	390.07	354050.	0.542E+06	0.341E+09	0.0	0.0
240.0	290.0	634.44	319821.	0.573E+06	0.357E+09	0.0	0.0
290.0	340.0	634.44	306714.	0.605E+06	0.373E+09	0.0	0.0
340.0	390.0	634.46	299464.	0.637E+06	0.388E+09	0.0	0.0
390.0	500.0	634.48	228094.	0.707E+06	0.413E+09	0.0	0.0
500.0	650.0	634.51	207712.	0.802E+06	0.444E+09	0.0	0.0
650.0	900.0	634.59	195434.	0.960E+06	0.493E+09	0.0	0.0
900.0	1150.0	639.52	185158.	0.117E+07	0.539E+09	0.0	0.0
1150.0	1400.0	645.17	172713.	0.128E+07	0.582E+09	0.0	0.0
1400.0	1650.0	650.27	100427.	0.144E+07	0.607E+09	0.0	0.0
1650.0	1900.0	654.93	91124.	0.161E+07	0.630E+09	0.0	0.0
1900.0	2150.0	656.09	70067.	0.177E+07	0.650E+09	0.0	0.0
2150.0	2400.0	655.64	68914.	0.194E+07	0.667E+09	0.0	0.0
2400.0	2650.0	653.48	64624.	0.210E+07	0.683E+09	0.0	0.0
2650.0	2900.0	630.74	62478.	0.226E+07	0.699E+09	0.0	0.0
2900.0	3150.0	575.06	104673.	0.241E+07	0.725E+09	0.0	0.0
3150.0	3400.0	547.42	117400.	0.254E+07	0.754E+09	0.0	0.0
3400.0	3650.0	547.73	115582.	0.268E+07	0.783E+09	0.0	0.0
3650.0	4380.0	548.28	112295.	0.308E+07	0.865E+09	0.0	0.0
4380.0	5130.0	548.98	107666.	0.349E+07	0.946E+09	0.0	0.0
5130.0	5880.0	549.61	103831.	0.390E+07	0.102E+10	0.0	0.0
5880.0	6630.0	550.15	100551.	0.432E+07	0.110E+10	0.0	0.0
6630.0	7380.0	550.51	97830.	0.473E+07	0.117E+10	0.0	0.0
7380.0	8130.0	550.49	97276.	0.514E+07	0.125E+10	0.0	0.0
8130.0	8880.0	550.45	96487.	0.555E+07	0.132E+10	0.0	0.0
8880.0	9630.0	550.41	95951.	0.597E+07	0.139E+10	0.0	0.0
9630.0	10380.0	550.37	95443.	0.638E+07	0.146E+10	0.0	0.0
10380.0	11130.0	550.37	94901.	0.679E+07	0.153E+10	0.0	0.0
11130.0	11880.0	550.37	94607.	0.721E+07	0.160E+10	0.0	0.0

Note: Rate data are constant over the time interval and are equal to the change in the integrated data over the interval divided by the duration of the interval.

Tab. 6.2-48I

NORTH ANNA UNIT 1 AND 2 NPSH ANALYSIS - PUMP SUCTION DER - MIN ECCS*

Mass and Energy Release Rates, Integrated Releases, and Integrated Spillage

Time Interval (Sec) Start	End	Mass Release Rate (Lbm/Sec)	Energy Release Rate (Btu/Sec)	Integrated Mass Release (Lbm)	Integrated Energy Release (Btu)	Integrated Spillage Mass Release (Lbm)	Integrated Spillage Energy Release (Btu)
0.0	0.1	79634.81	46065184.	0.796E+04	0.461E+07	0.0	0.0
0.1	1.0	36041.37	20034752.	0.404E+05	0.234E+08	0.0	0.0
1.0	2.0	34839.37	20356256.	0.752E+05	0.437E+08	0.0	0.0
2.0	3.0	34785.50	20472304.	0.110E+06	0.642E+08	0.0	0.0
3.0	4.0	34456.69	20402816.	0.145E+06	0.846E+08	0.0	0.0
4.0	5.0	33943.81	20137168.	0.178E+06	0.105E+09	0.0	0.0
5.0	6.0	33277.50	19777152.	0.212E+06	0.124E+09	0.0	0.0
6.0	7.0	32335.44	19346096.	0.244E+06	0.144E+09	0.0	0.0
7.0	8.0	31274.87	18547776.	0.275E+06	0.162E+09	0.0	0.0
8.0	9.0	30393.00	17099712.	0.306E+06	0.180E+09	0.0	0.0
9.0	10.0	29451.94	17304320.	0.335E+06	0.197E+09	0.0	0.0
10.0	11.0	20029.50	13124416.	0.355E+06	0.211E+09	0.0	0.0
11.0	12.0	6589.19	6845456.	0.362E+06	0.217E+09	0.0	0.0
12.0	13.0	4312.50	5146464.	0.366E+06	0.223E+09	0.0	0.0
13.0	14.0	3137.56	3779520.	0.369E+06	0.226E+09	0.0	0.0
14.0	15.0	2002.69	2500912.	0.371E+06	0.229E+09	0.0	0.0
15.0	16.0	1478.00	1736520.	0.373E+06	0.231E+09	0.0	0.0
16.0	17.0	1050.56	1249680.	0.374E+06	0.232E+09	0.0	0.0
17.0	18.0	855.87	990328.	0.375E+06	0.233E+09	0.0	0.0
18.0	19.0	744.25	[illegible]	0.375E+06	0.233E+09	0.0	0.0
19.0	20.0	676.19	642224.	0.376E+06	0.234E+09	0.0	0.0
20.0	22.5	559.12	463021.	0.378E+06	0.235E+09	0.0	0.0
22.5	25.0	301.87	215286.	0.378E+06	0.236E+09	0.0	0.0
25.0	27.5	2752.95	509774.	0.384E+06	0.237E+09	0.0	0.0
27.5	30.0	1243.10	530690.	0.387E+06	0.239E+09	0.0	0.0
30.0	35.0	6740.50	949706.	0.418E+06	0.243E+09	0.0	0.0
35.0	40.0	3133.45	704653.	0.434E+06	0.247E+09	0.0	0.0
40.0	45.0	497.44	635754.	0.436E+06	0.251E+09	0.0	0.0
45.0	50.0	566.10	643347.	0.439E+06	0.254E+09	0.0	0.0
50.0	55.0	611.37	633407.	0.442E+06	0.257E+09	0.0	0.0

Note: Rate data are constant over the time interval and are equal to the change in the integrated data over the interval divided by the duration of the interval.

*These data correspond to a power level of 2900 MWt.

Table 6.2-48I (continued)

NORTH ANNA UNIT 1 AND 2 NPSH ANALYSIS - PUMP SUCTION DER - MIN ECCS

Mass and Energy Release Rates, Integrated Releases, and Integrated Spillage

Time Interval (Sec) Start	End	Mass Release Rate (Lbm/Sec)	Energy Release Rate (Btu/Sec)	Integrated Mass Release (Lbm)	Integrated Energy Release (Btu)	Integrated Spillage Mass Release (Lbm)	Integrated Spillage Energy Release (Btu)
55.0	60.0	622.55	619472.	0.445E+06	0.260E+09	0.0	0.0
60.0	72.5	584.56	588305.	0.453E+06	0.267E+09	0.0	0.0
72.5	85.0	556.58	553907.	0.460E+06	0.274E+09	0.0	0.0
85.0	105.0	513.41	521587.	0.470E+06	0.283E+09	0.0	0.0
105.0	140.0	479.61	485391.	0.487E+06	0.302E+09	0.0	0.0
140.0	190.0	510.88	430704.	0.512E+06	0.324E+09	0.0	0.0
190.0	240.0	590.13	356116.	0.542E+06	0.341E+09	0.0	0.0
240.0	290.0	634.47	319071.	0.573E+06	0.357E+09	0.0	0.0
290.0	340.0	634.47	302714.	0.605E+06	0.373E+09	0.0	0.0
340.0	390.0	634.49	299464.	0.637E+06	0.388E+09	0.0	0.0
390.0	500.0	634.59	259033.	0.701E+06	0.416E+09	0.0	0.0
500.0	650.0	645.76	218033.	0.803E+06	0.449E+09	0.0	0.0
650.0	900.0	655.91	198882.	0.967E+06	0.496E+09	0.0	0.0
900.0	1650.0	658.04	178383.	0.1?3E+07	0.597E+09	0.0	0.0
1650.0	2700.0	656.03	76572.	0.187E+07	0.654E+09	0.0	0.0
2700.0	2950.0	562.01	90491.	0.224E+07	0.726E+09	0.0	0.0
2950.0	3730.0	555.48	89704.	0.268E+07	0.796E+09	0.0	0.0
3730.0	5230.0	556.50	80000.	0.351E+07	0.916E+09	0.0	0.0
5230.0	6730.0	557.03	73973.	0.435E+07	0.103E+10	0.0	0.0
6730.0	8230.0	557.39	69603.	0.518E+07	0.113E+10	0.0	0.0
8230.0	9730.0	557.64	66540.	0.602E+07	0.123E+10	0.0	0.0
9730.0	11230.0	557.86	64106.	0.686E+07	0.133E+10	0.0	0.0
11230.0	12730.0	558.02	62056.	0.769E+07	0.147E+10	0.0	0.0

Note: Rate data are constant over the time interval and are equal to the change in the integrated data over the interval divided by the duration of the interval.

Table 6.2-49

SENSITIVITY OF NPSH TO SINGLE FAILURES AND BREAK LOCATIONS ASSUMING TEMPERATURES OF 50°F RWST AND 35°F SERVICE WATER[a]

UNIT 2

Single Failure	Minimum NPSHA (ft) at time t (sec)		
	IRS	ORS	LHSI
PSDER			
Minimum ESF	12.6 (1770)	18.1 (1860)	16.6 (2950)
Normal ESF	12.4 (835)	18.5 (925)	23.8 (1960)
Minimum ECCS	12.7 (855)	19.5 (975)	20.2 (2260)
Normal sprays			
QS pump	12.8 (1120)	19.7 (1255)	23.5 (2440)
RS pump (outside pump)	12.5 (1030)	18.7 (1105)	22.0 (1990)
HLDER			
Minimum ESF	11.8 (923)	17.3 (1088)	19.3 (2908)
Normal ESF	11.5 (560)[b]	16.8 (635)	25.3 (1920)
Minimum ECCS	12.2 (580)	17.6 (665)	21.2 (2230)
QS pump	11.6 (700)	17.7 (815)	25.0 (2400)
RS pump (outside pump)	11.5 (638)	16.8 (718)	24.2 (1928)

[a]These data correspond to a power level of 2900 MWt.

[b]Worst case for RS pump NPSHA.

Table 6.2-50

SENSITIVITY OF NPSH TO RWST AND SERVICE WATER TEMPERATURE[a]

UNIT 2

Break	ESF	Initial Water Temp. (°F) RWST	Service	Minimum NPSHA (ft) at Time (sec) IRS	ORS	LHSI
PSDER	Min	50	35	12.6 (1770)	18.1 (1860)	16.8 (2950)
PSDER	Min	50	93	13.7 (1820)	21.3 (2040)	15.5 (2970)[b]
PSDER	Min	40	35	12.8 (1725)	18.0 (1825)	17.3 (2950)
PSDER	Min	40	95	14.0 (1785)	21.1 (1995)	16.3 (2960)
HLDER	Norm	50	35	11.5 (560)[c]	16.8 (635)	25.3 (1920)
HLDER	Norm	50	93	14.5 (593)	20.1 (710)	25.7 (1930)
HLDER	Norm	40	35	11.7 (550)	16.6 (615)[c]	25.4 (1920)
HLDER	Norm	40	95	15.3 (593)	20.3 (710)	26.0 (1930)

[a]These data correspond to a power level of 2900 MWt.

[b]Worst case for LHSI pump NPSHA.

[c]Worst case for RS pump NPSHA.

Table 6.2-51

SENSITIVITY OF NPSH TO BREAK SIZE, ASSUMING MINIMUM ESF AND TEMPERATURES OF 50°F RWST AND 35°F SERVICE WATER[a]

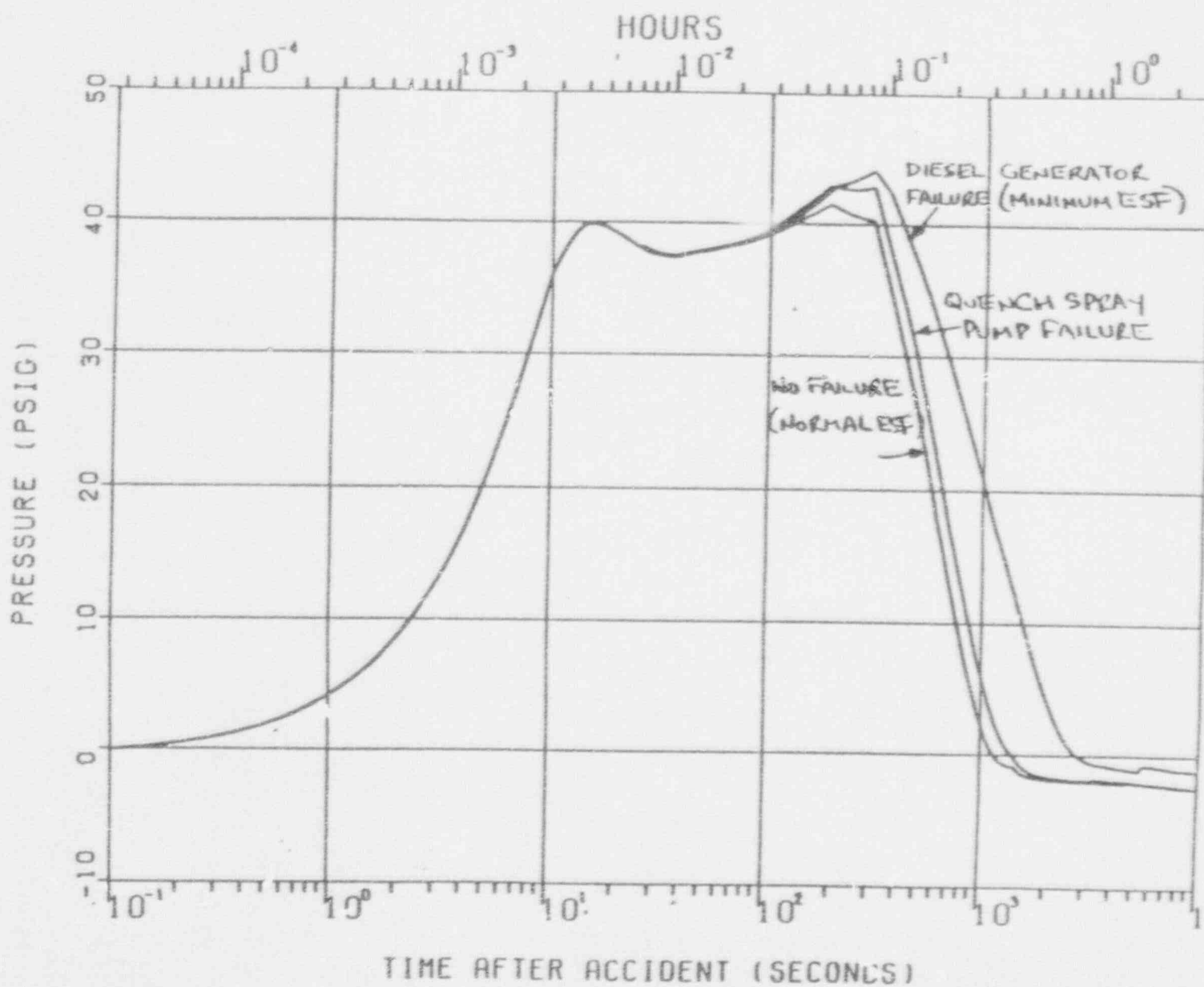
UNIT 2

Break Location	Break Size	Minimum NPSHA (ft) at time t (sec)		
		IRS	ORS	LHSI
PS	DER	12.6 (1770)	18.1 (1860)	16.8 (2950)
HL[b]	DER	11.8 (923)	17.3 (1088)	19.3 (2908)
HL[b]	0.6 DER	11.7 (928)	17.3 (1083)	19.3 (2908)
HL[b]	0.326 DER	11.7 (903)	17.4 (1068)	19.3 (2908)
HL	6-in. equiv. SER[b]	16.2 (1005)	19.7 (1210)	19.7 (3110)
HL	6-in. equiv. SER[c]	17.9 (1695)	21.2 (1960)	18.6 (3700)

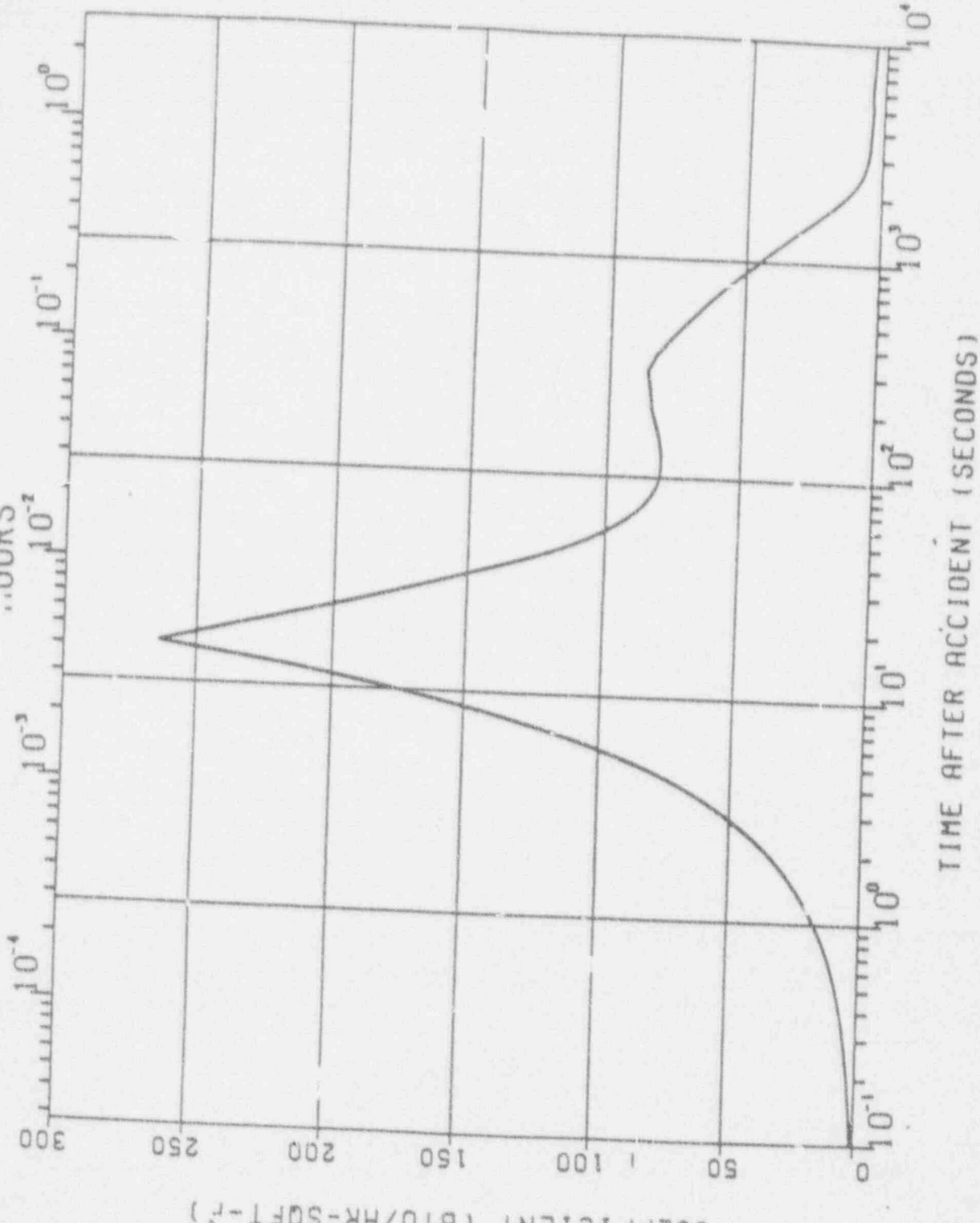
[a] These data correspond to a power level of 2900 MWt.

[b] The LOCTIC program used in these analyses included heat transfer from the primary system to the steam generator secondary side. The program as run for these analyses did not return this energy from the secondary side to the primary system when blowdown was completed.

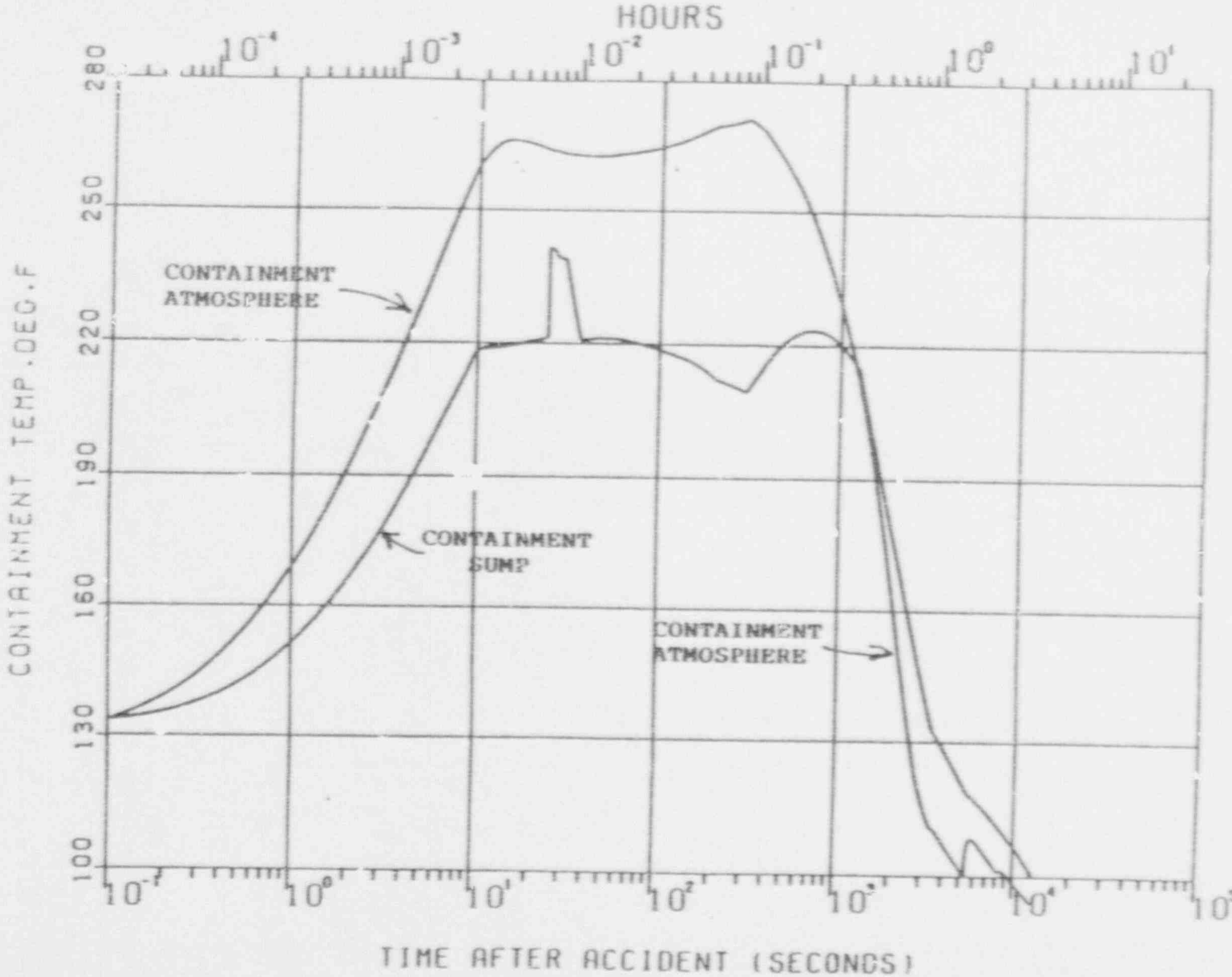
[c] The LOCTIC program used in these analyses maintained thermal equilibriumbetween the primary system and the steam generator secondary side. Thus, the energy in the steam generator is added to the primary system as it cools down.



CONTAINMENT PRESSURE VS TIME
PSDER-SINGLE FAILURE ANALYSIS
NORTH ANNA UNIT 2



STEAM CONDENSING COEFFICIENT VS TIME
PUMP SUCTION - MINIMUM ESF
NORTH ANNA UNIT 2



CONTAINMENT TEMF=RATURE VS TIME
PUMP SUCTION DER - MINIMUM ESF

NORTH ANNA UNIT2

NORMAL ESF EXCEPT MINIMUM QUENCH SPRAYS
WINTER CONDITIONS

This study was performed for a power level of 2900 MWt.

CONTAINMENT SUMP TEMPERATURE VS TIME
COLD LEG DER
NORTH ANNA POWER STATION
UNITS 1 & 2

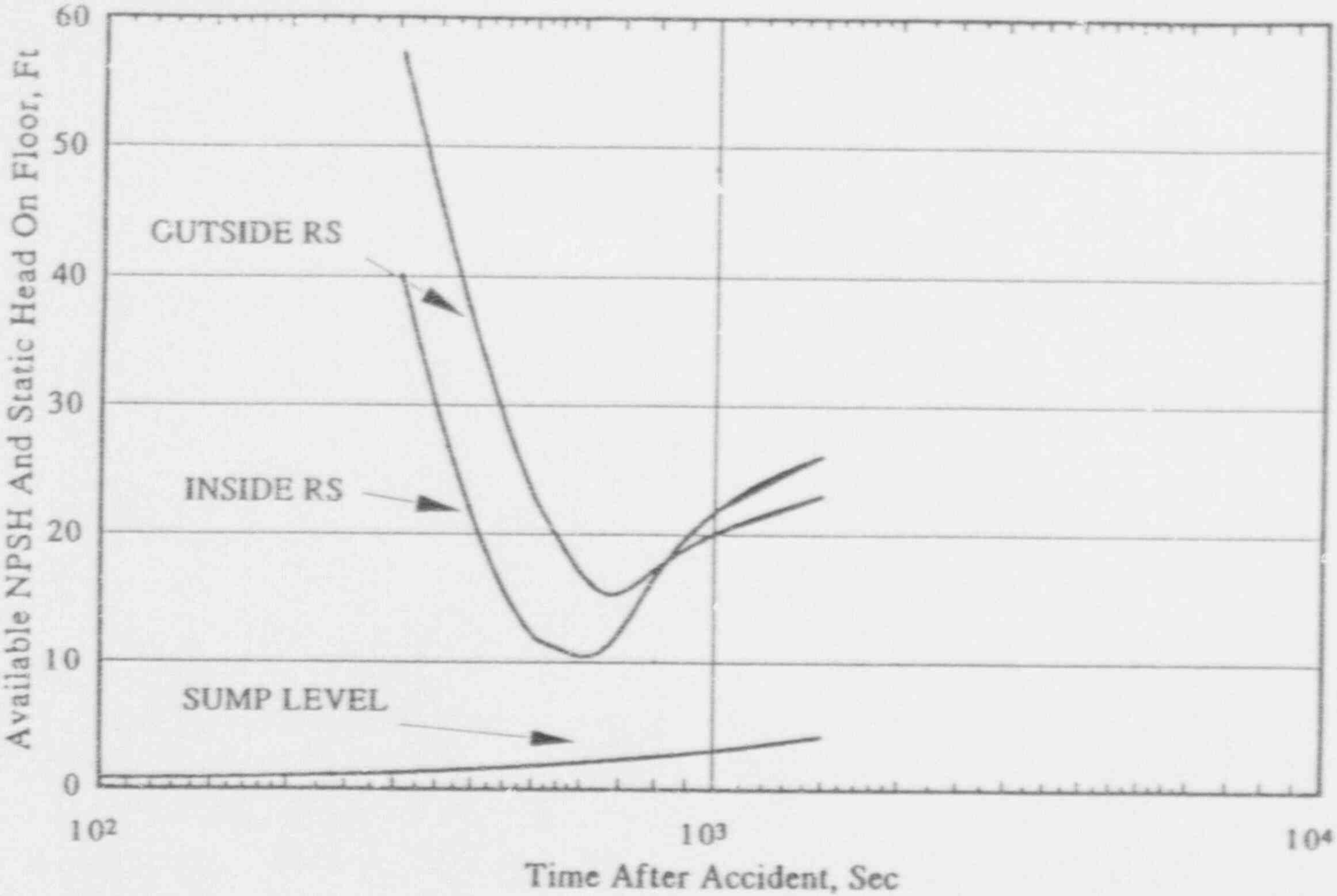
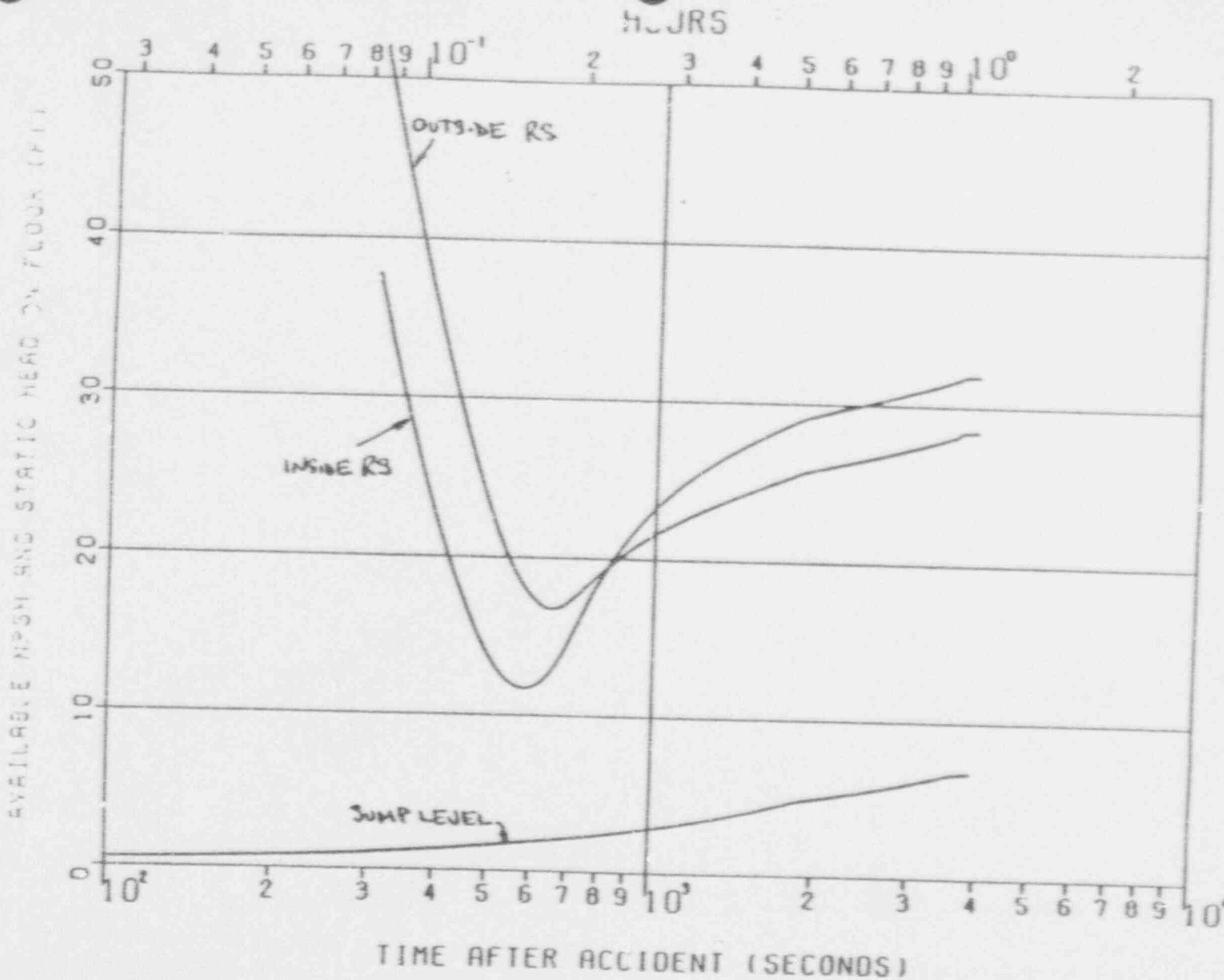


Figure 6.2-85 A
RS Pump Available NPSH
RS Pump NPSH Analysis
HLDER - Normal ESF
North Anna Unit 1



NPSHA TRANSIENTS
WORST CASE RS PUMP NPSH
NORTH ANNA UNIT 2

Figure 6.2-85 D

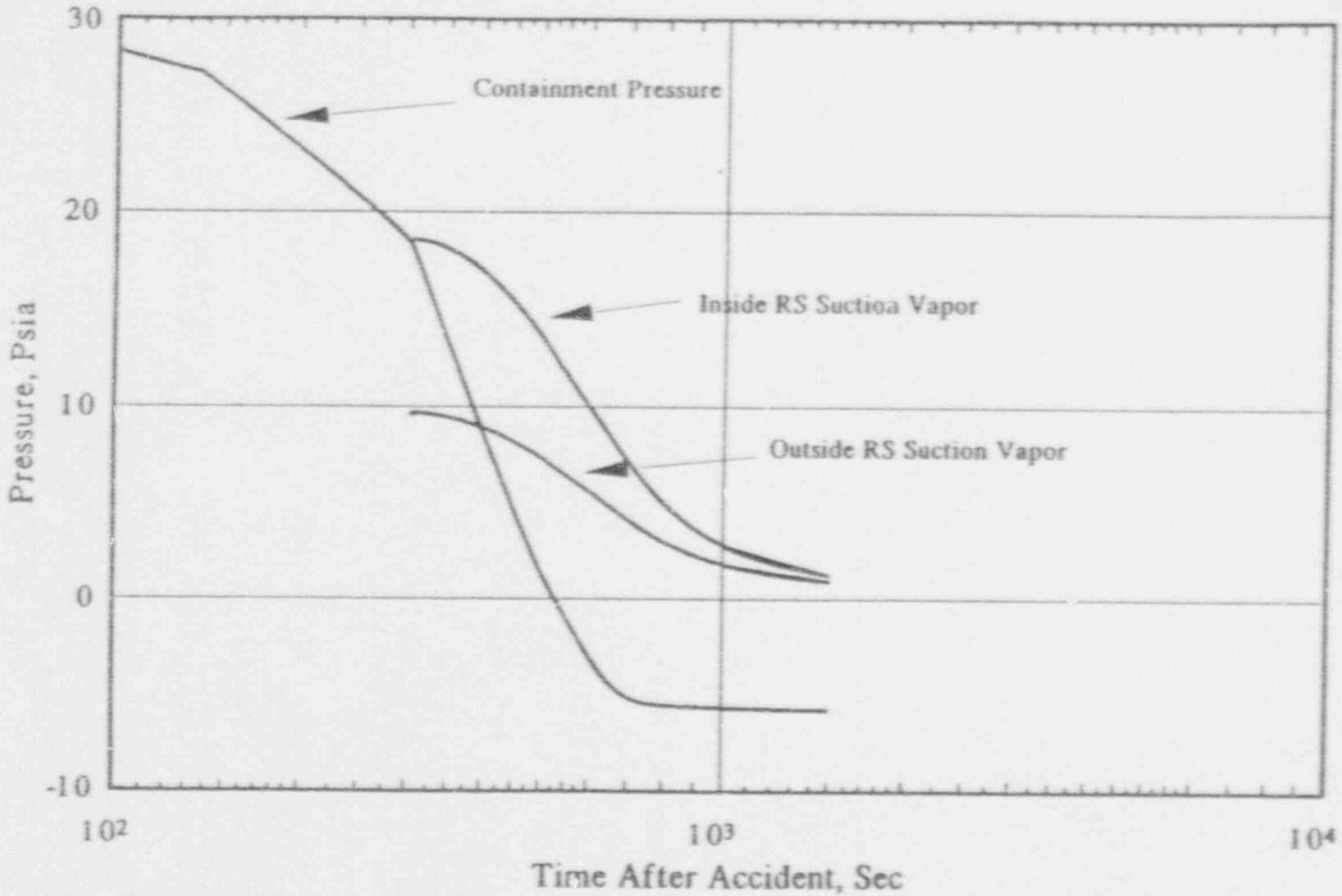
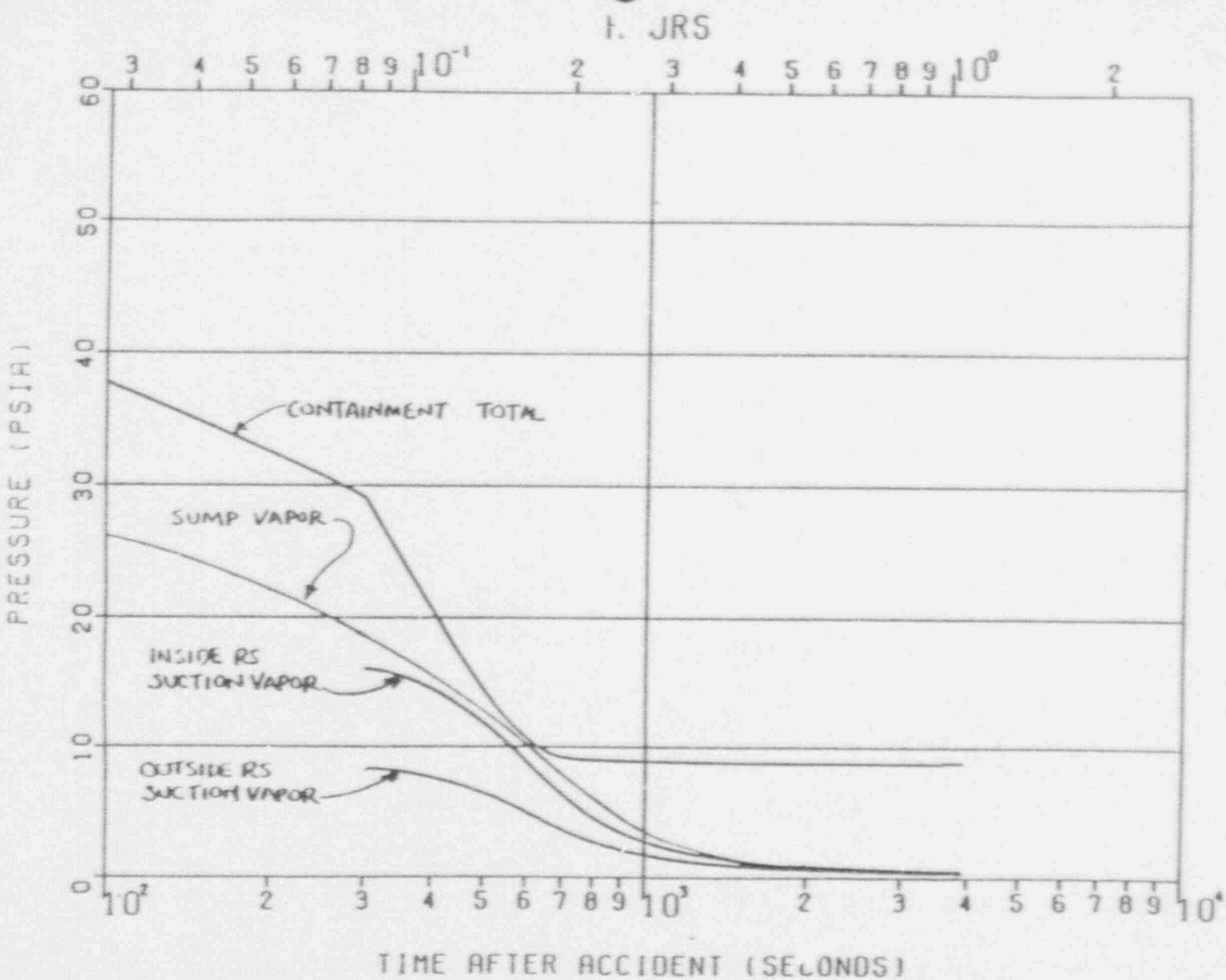


Figure 6.2-86 A

Containment and Suction Vapor Pressure
RS Pump NPSH Analysis
HLDER - Normal ESF
North Anna Unit 1



PRESSURE TRANSIENTS
WORST CASE RS PUMP NPSH
UNDER NORM ESF
NORTH ANNA UNIT 2

Figure 6.2-86B

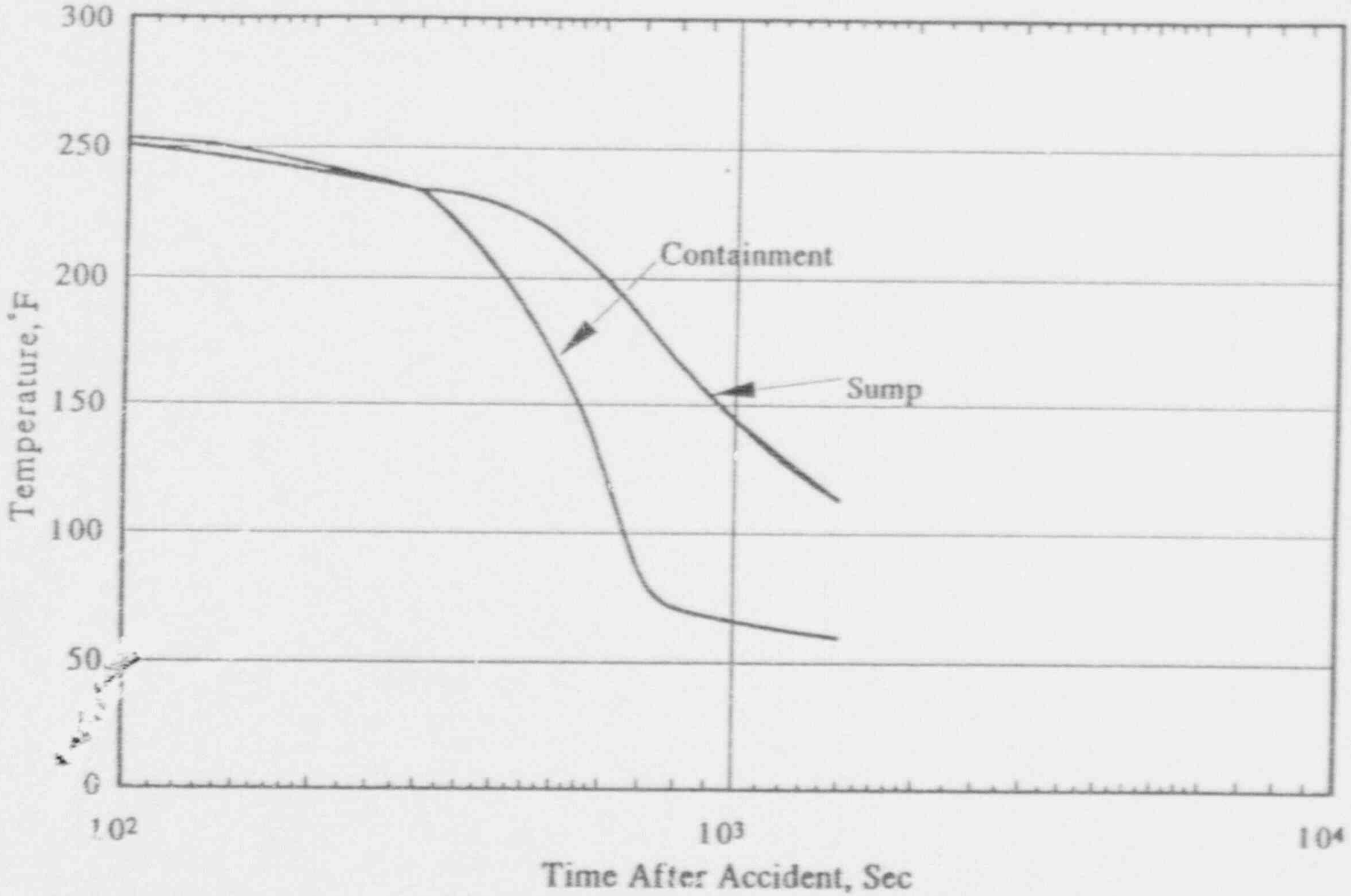
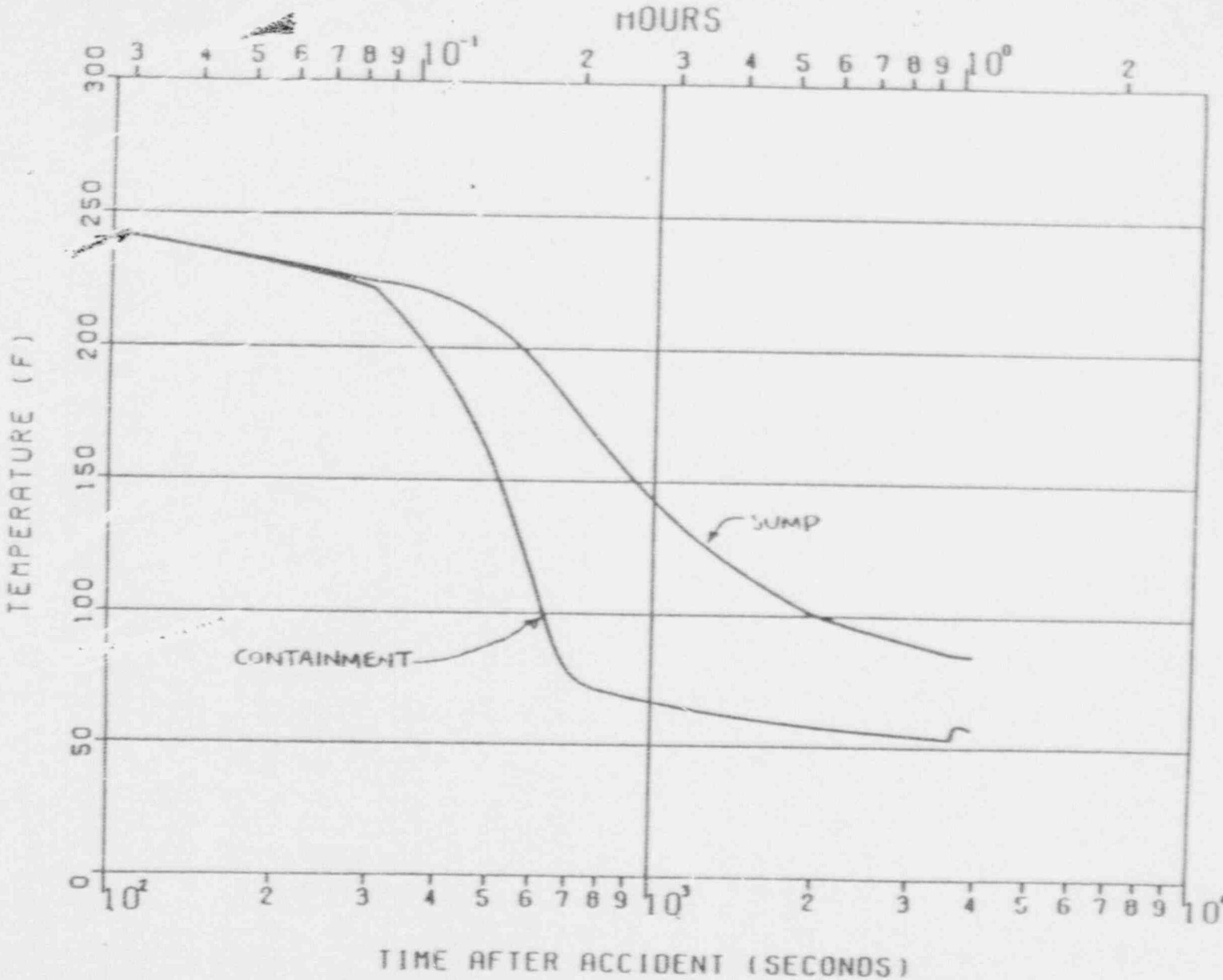


Figure 6.2-87A
Containment and Sump Temperature
RS Pump NPSH An is
HLDER - Normal ES
North Anna Unit 1



TEMPERATURE TRANSIENTS
WORST CASE RS PUMP NPSH
UNDER NORM ESF

NORTH ANNA UNIT 2

Figure 6.2-87B

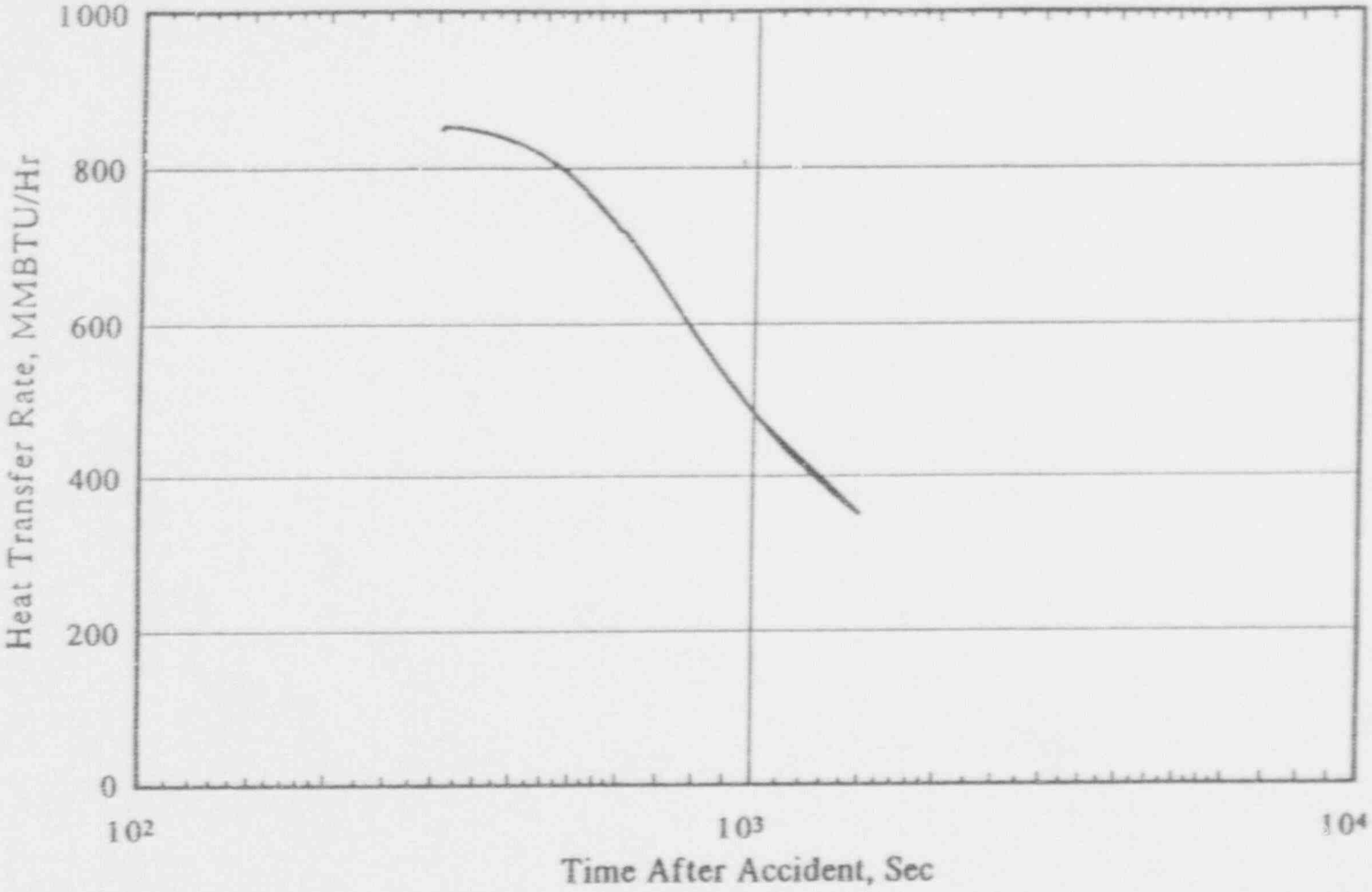
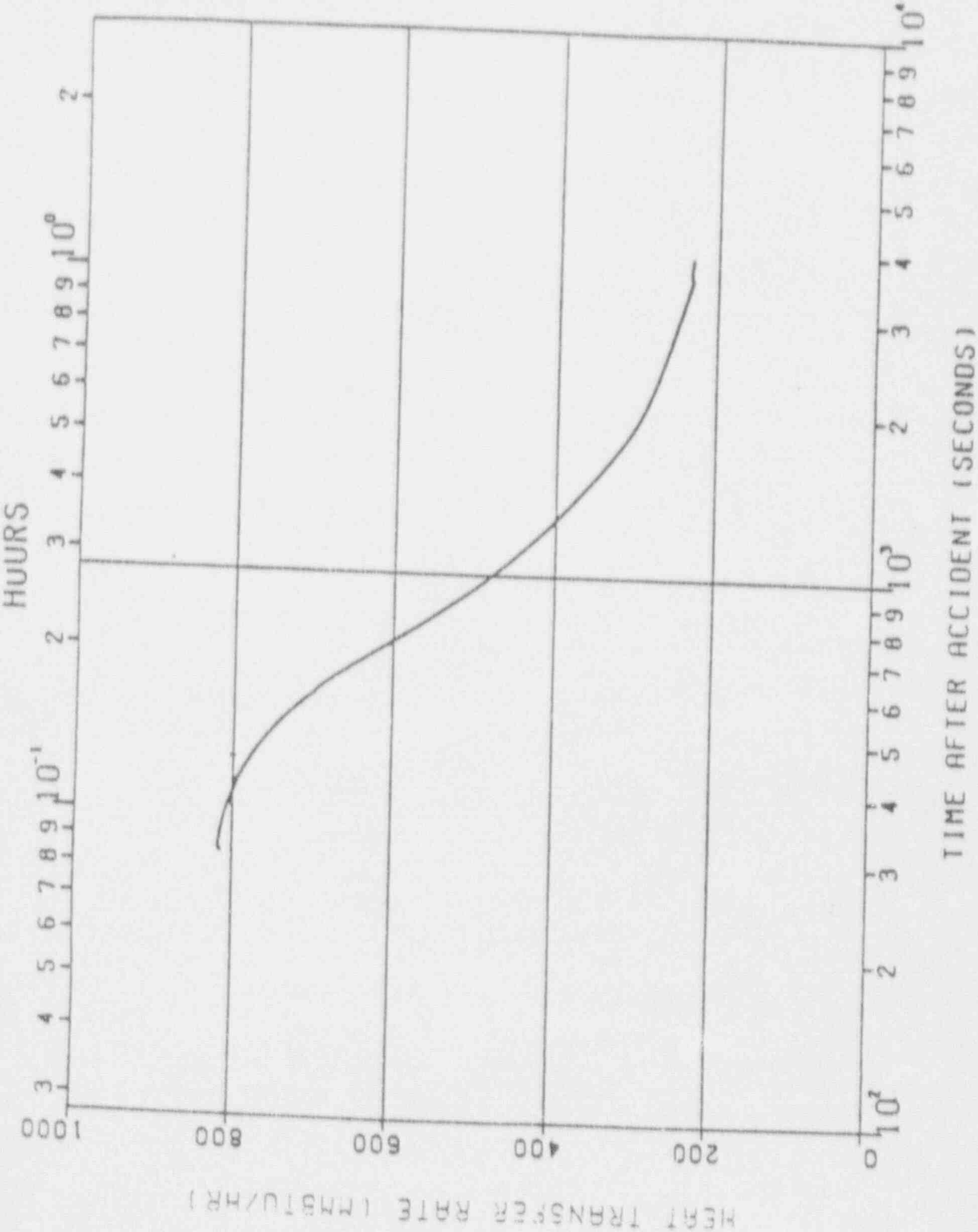


Figure 6.2-88 A
Recirculation Spray Cooler Duty
RS Pump NPSH Analysis
HLDER - Normal ESF
North Anna Unit I



RECIRC. SPRAY COOLER DUTY
WORST CASE RS PUMP NPSH
NORTH ANNA UNIT 2

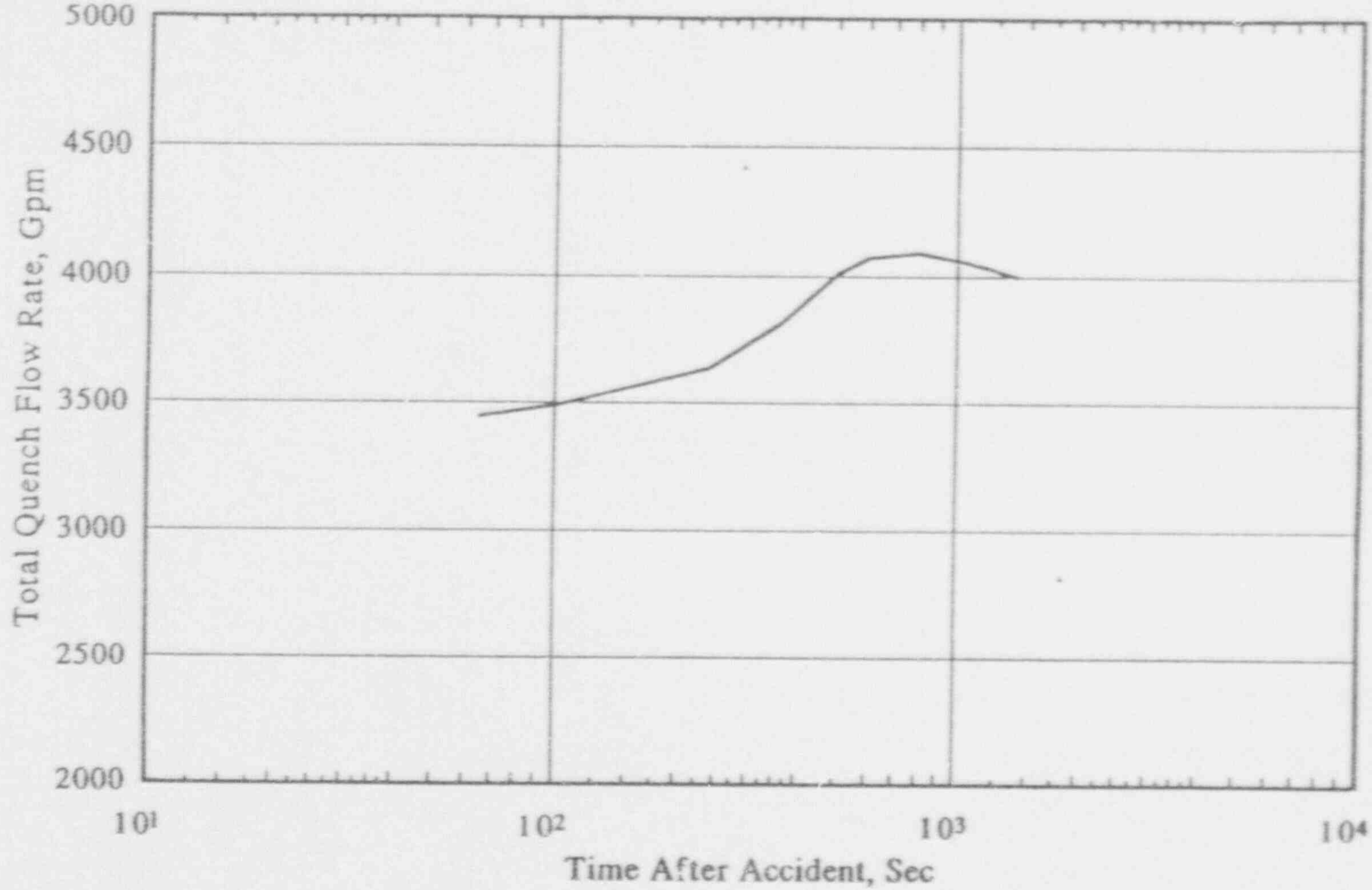
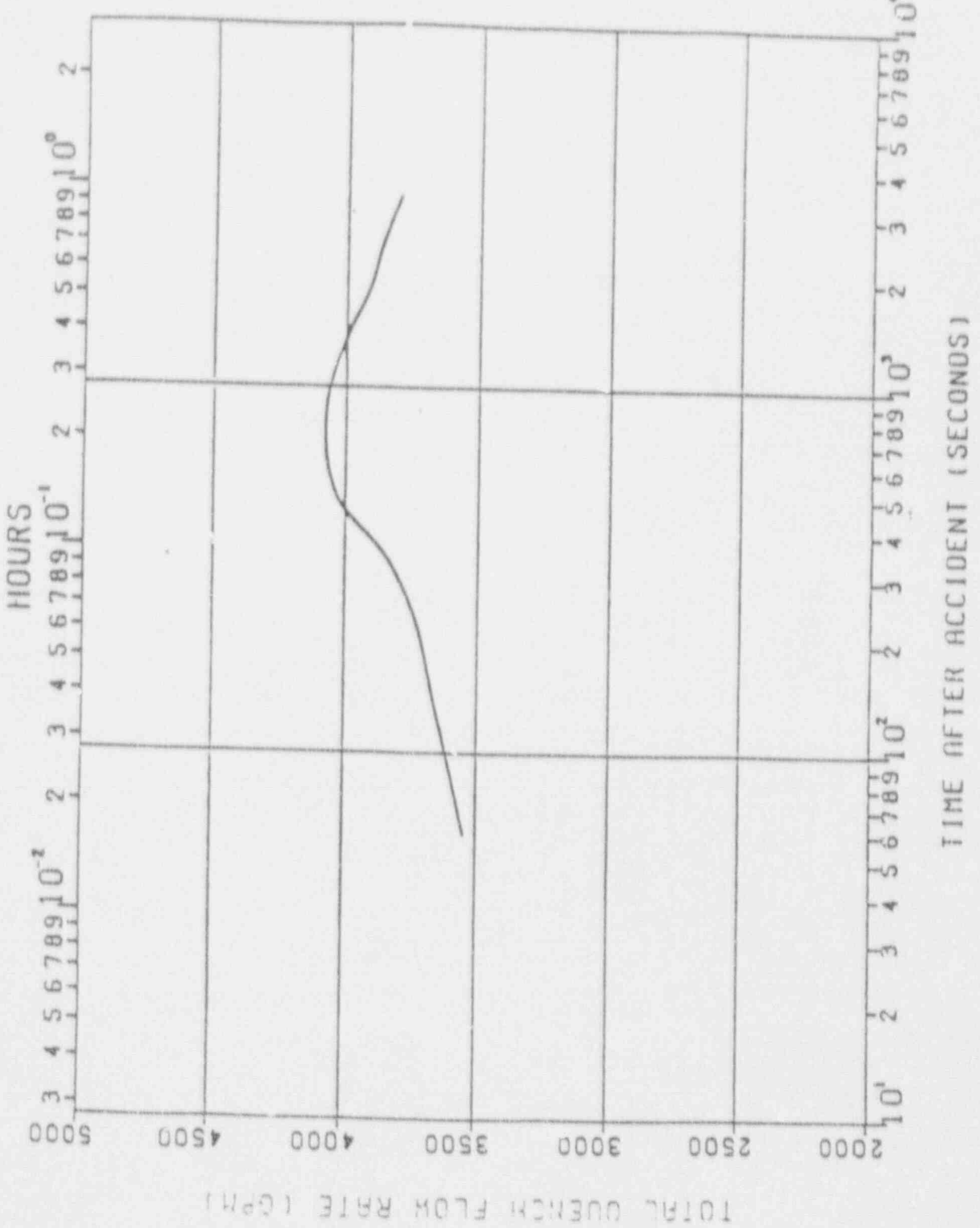


Figure 6.2-89 A
Quench Spray Flow Rate
RS Pump NPSH Analysis
HLDER - Normal ESF
North Anna Unit 1



QUENCH SPRAY FLOW RATE
WORST CASE RS PUMP NPSH

NoRTH ANNA UNIT 2

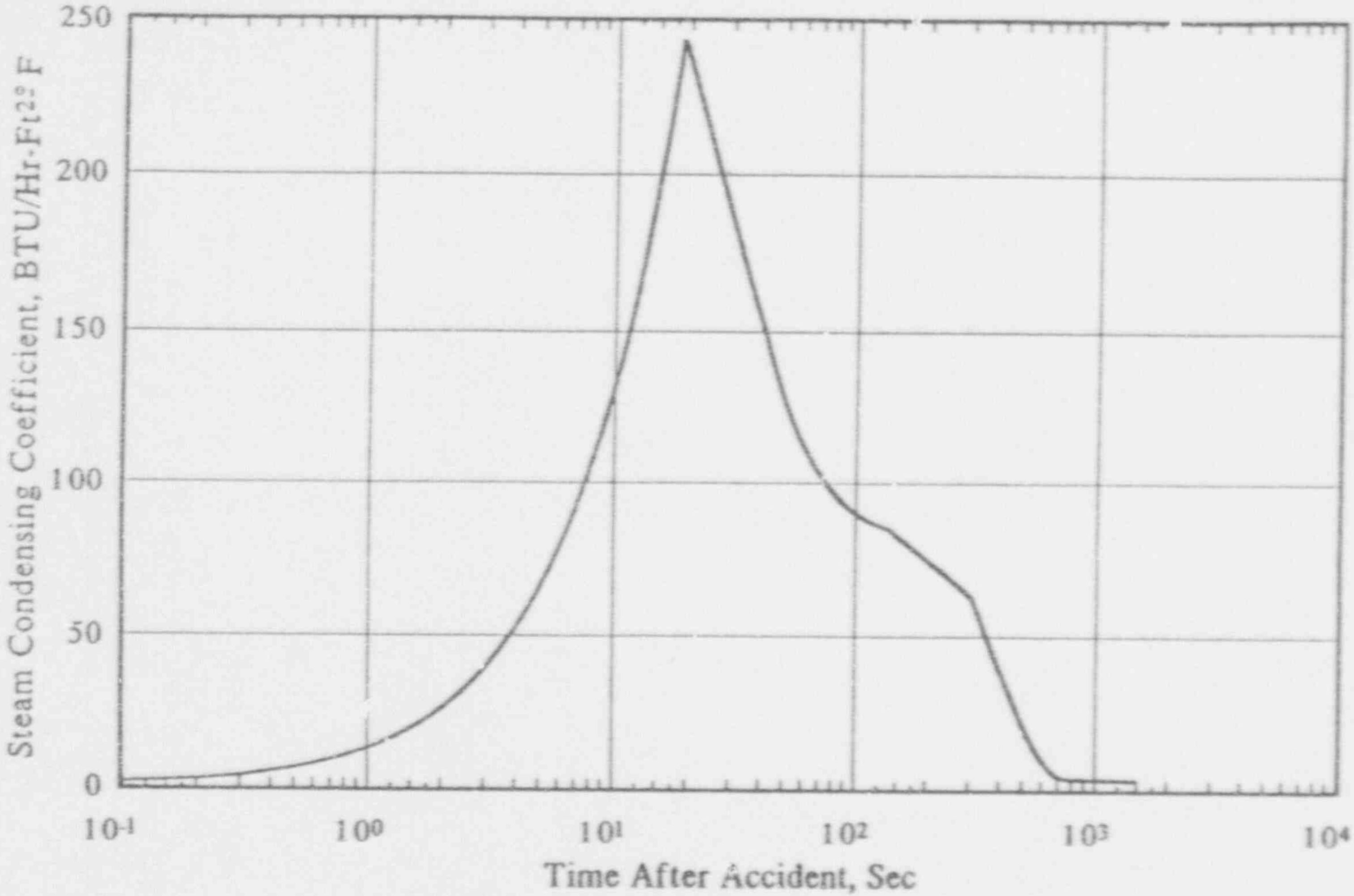
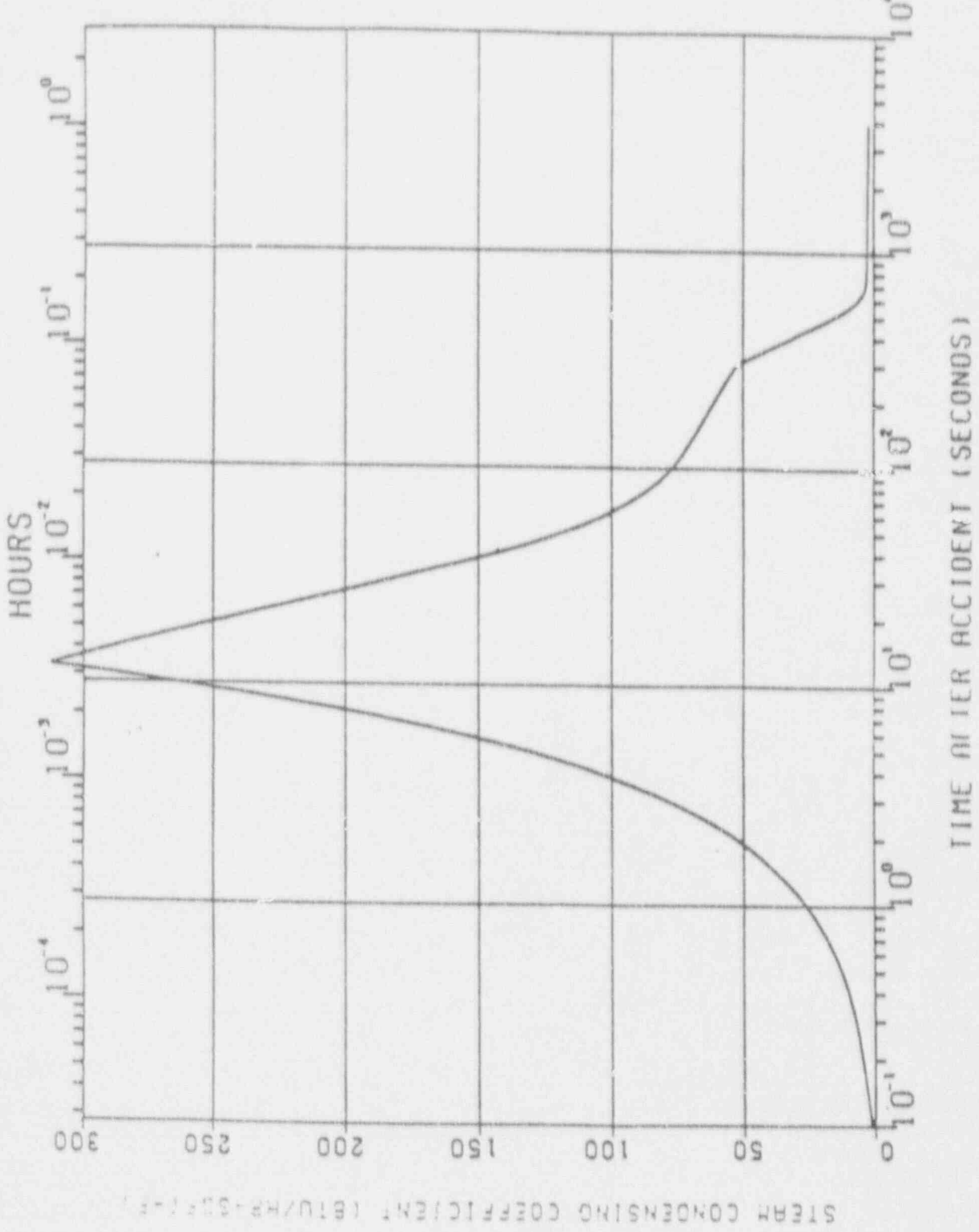


Figure 6.2-90 A
Steam Condensing Coefficient
RS Pump NPSH Analysis
HLDER - Normal ESF
North Anna Unit 1



STEAM CONDENSING COEFFICIENT
WORST CASE RS PUMP NPSH
NORTH ANNA UNIT 2

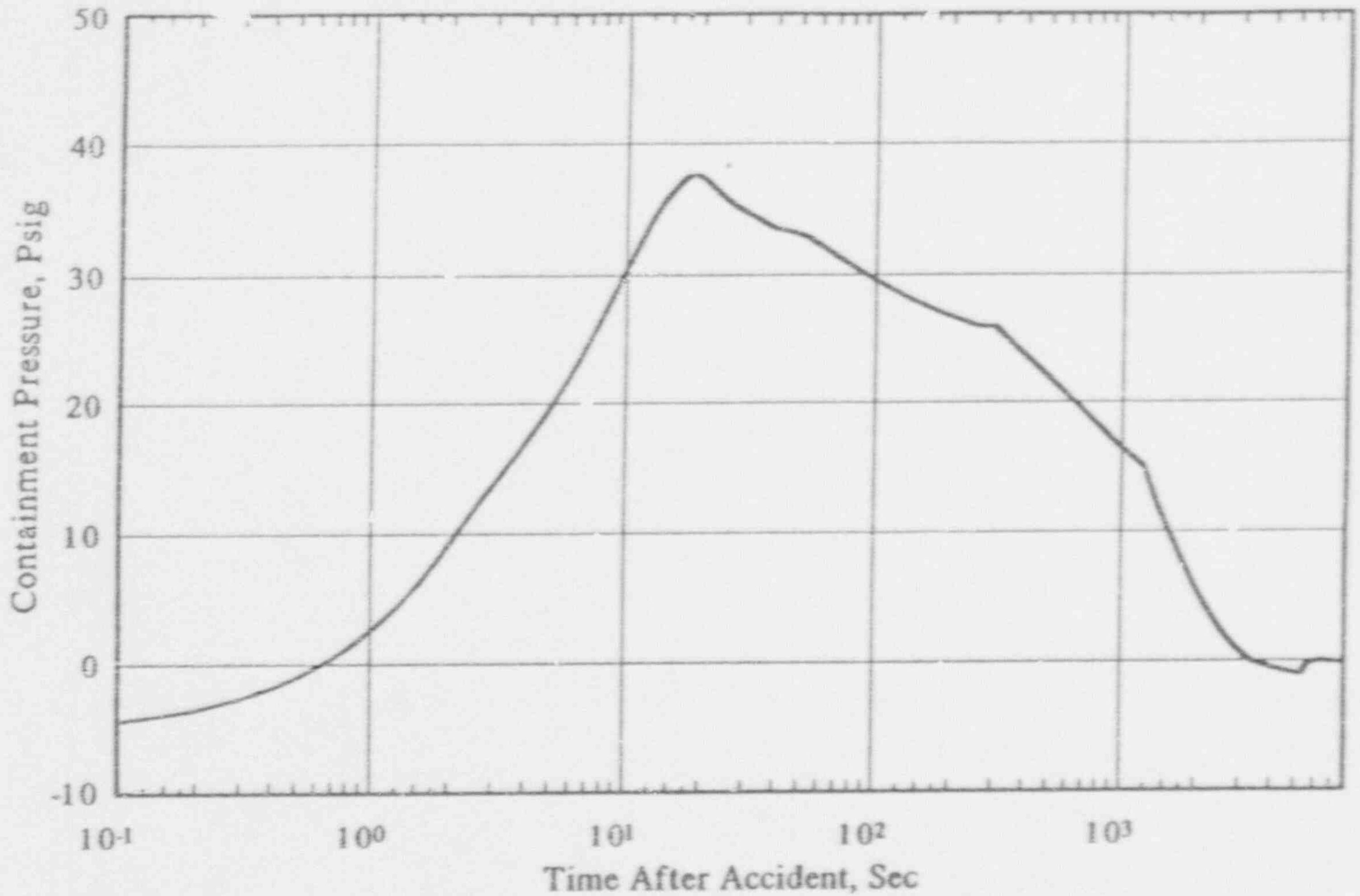
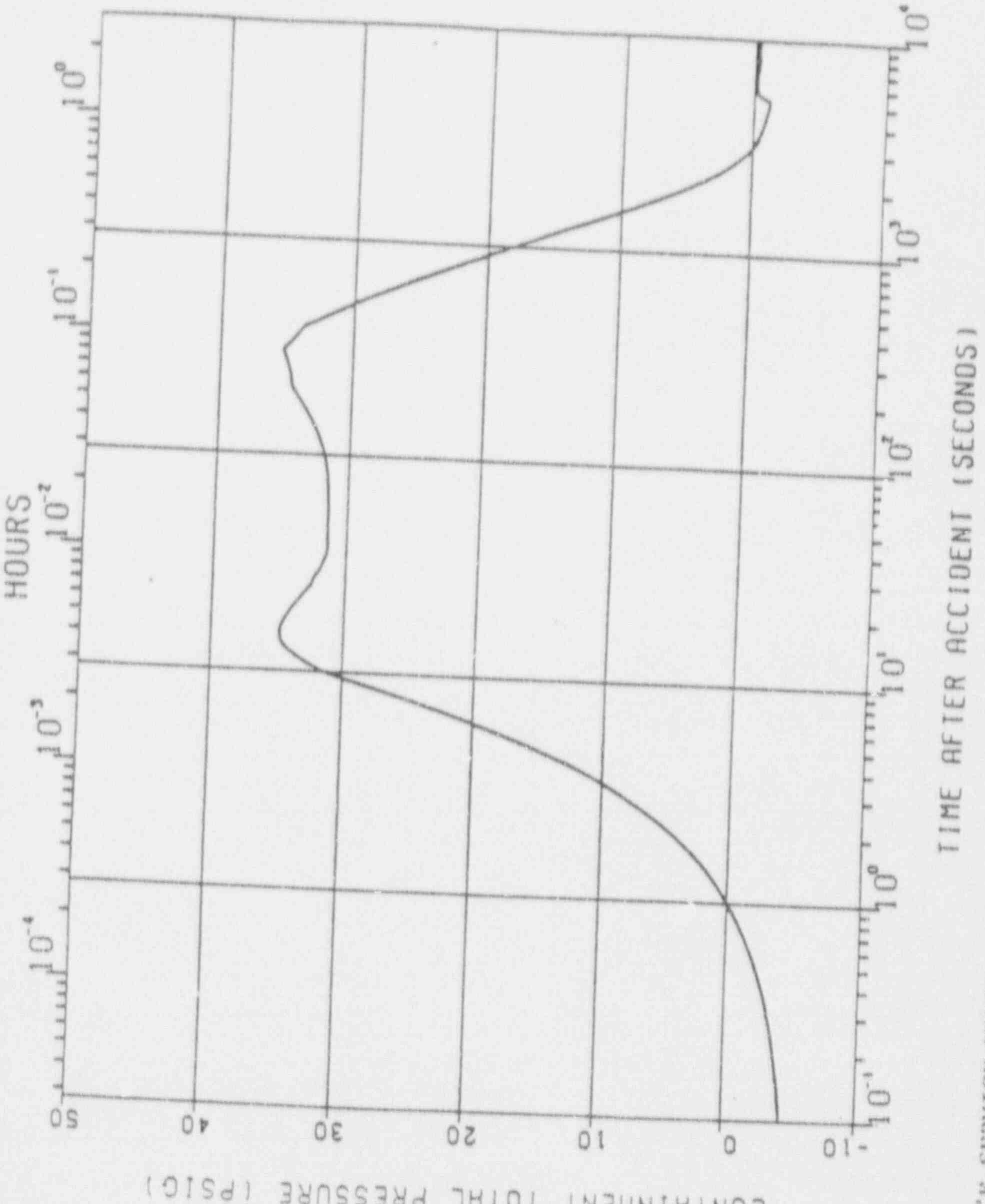


Figure 6.2-91A
Containment Pressure
Depressurization Analysis
PSDER - Minimum ESF
North Anna Unit 1



CONTAINMENT PRESSURE
WORST CASE DEPRESSURIZATION
PADER, MINIMUM ESF
NORTH ANNA UNIT 2

°F RWST, °F SERVICE WATER

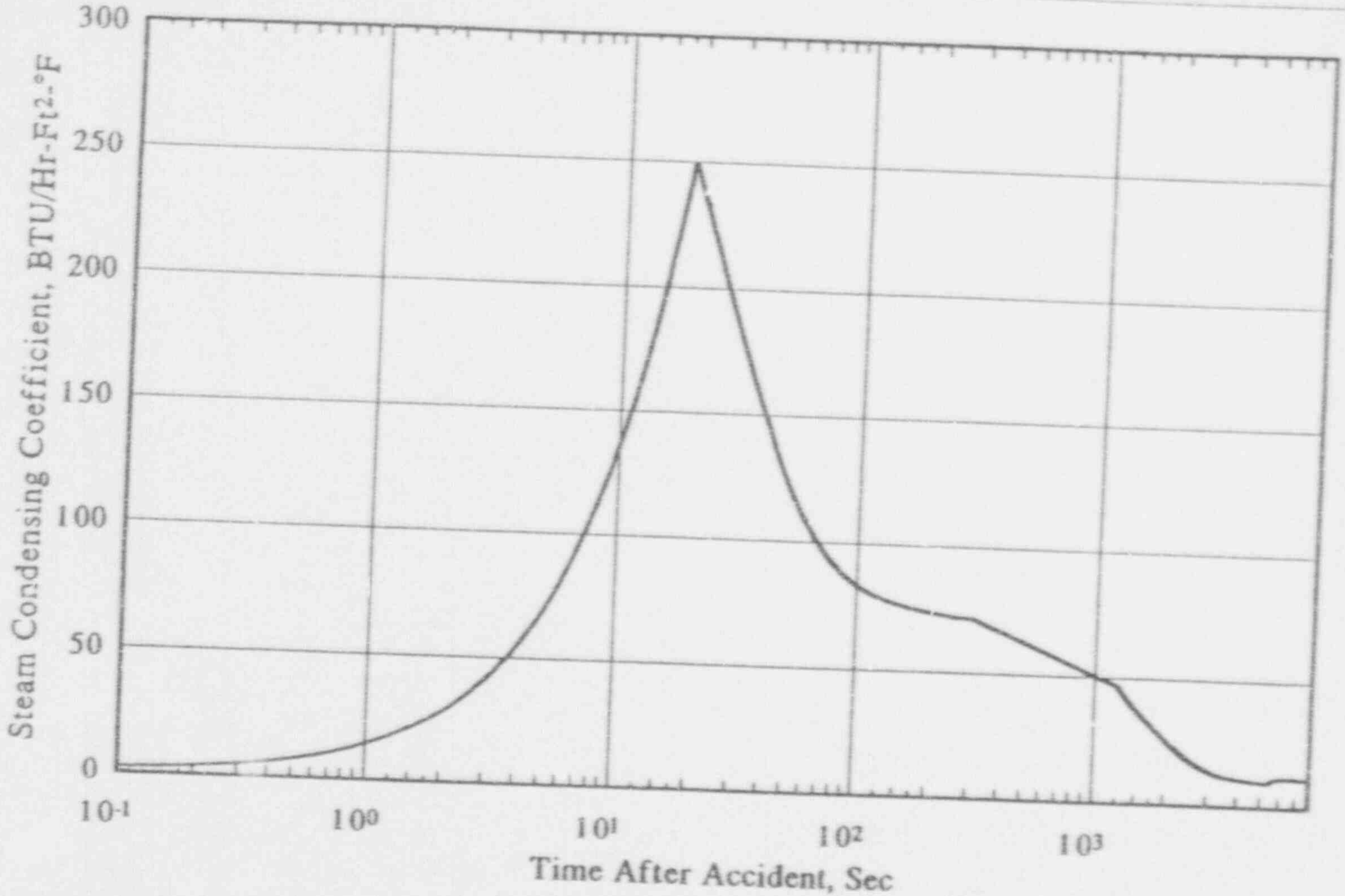
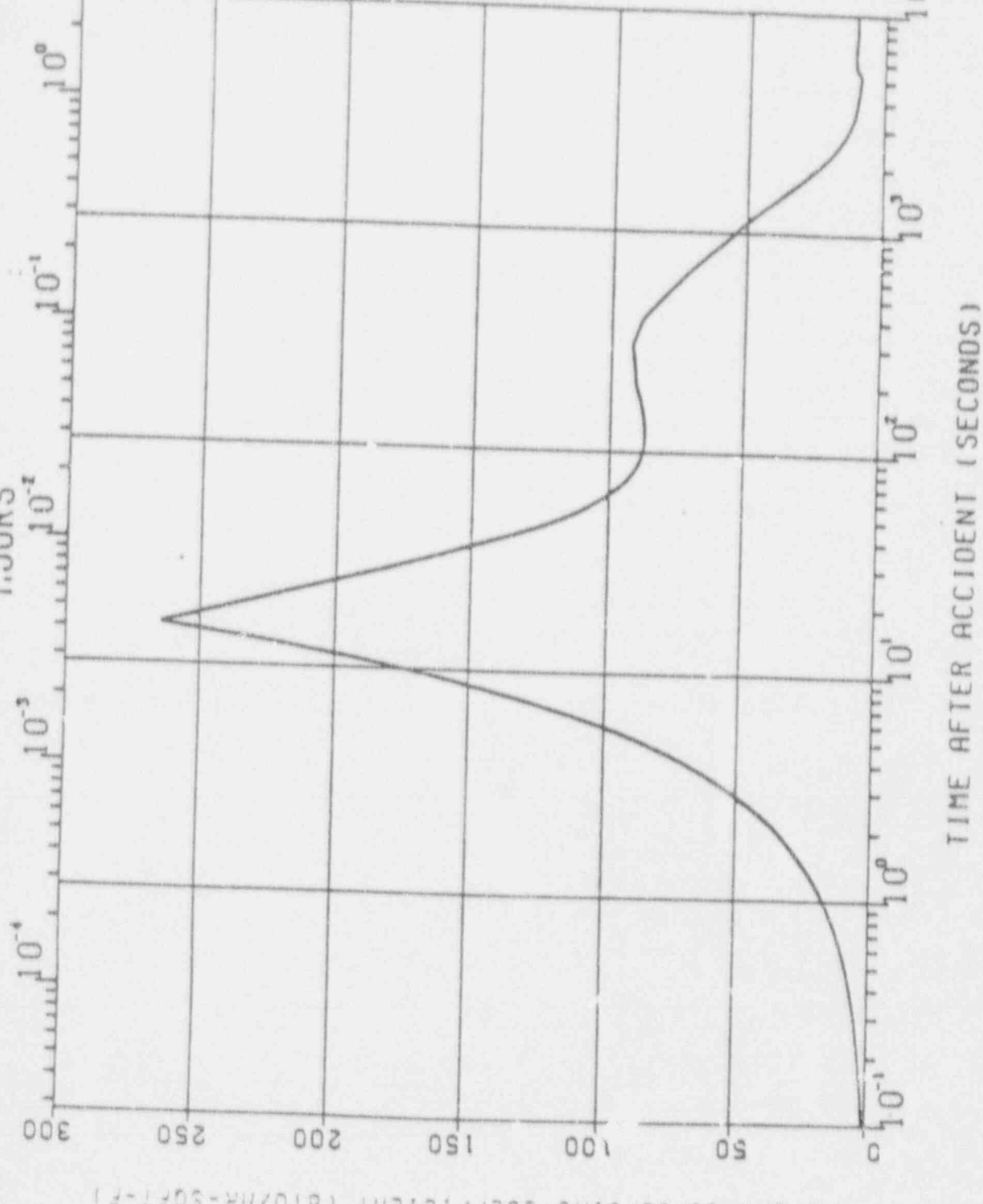


Figure 6.2-92
Steam Condensing Coefficient
Depressurization Analysis
PSDER - Minimum ESF
North Anna Unit 1



CONDENSING COEFFICIENT
WORST CASE DEPRESSURIZATION
FADER MINIMUM CAP
North Anna Unit 2

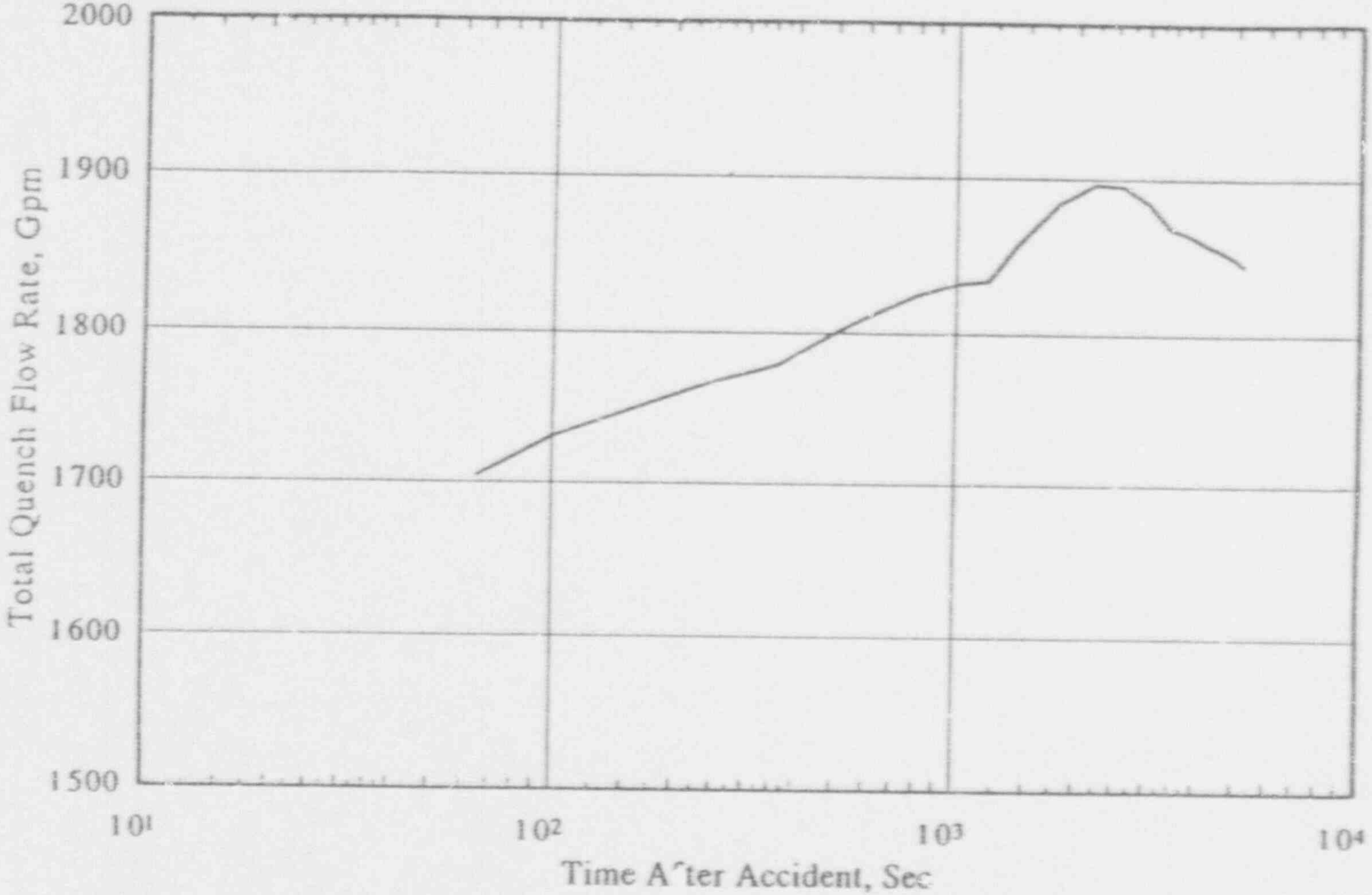
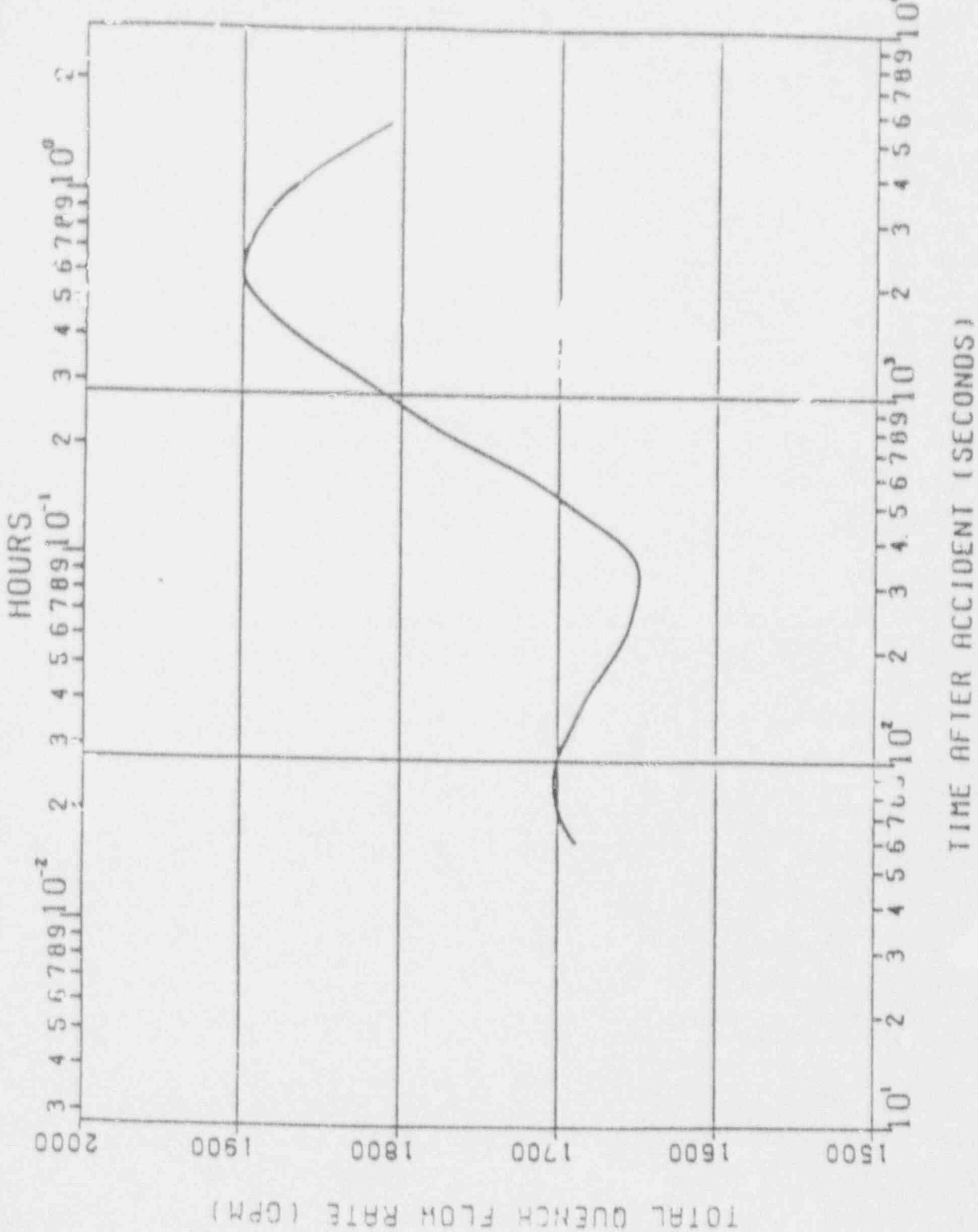


Figure 6.2-93A
Quench Spray Flow Rate
Depressurization Analysis
PSDER - Minimum ESF
North Anna Unit 1



QUENCH SPRAY FLOW RATE
WORST CASE DEPRESSURIZATION
[illegible] MINIMUM ESF
NORTH ANNA UNIT 2

Table 6.2-52

SENSITIVITY OF NPSH TO INITIAL CONDITIONS*

UNIT 2

Initial Temperature (°F)		Initial Containment			Minimum NPSHA (ft)		
RWST	Service Water	Temp (°F)	Dew Point (°F)	Ptot (psia)	IRS	ORS	LHSI
			PSDER, Minimum ESF				
50	93	105	105	10.0	13.7	21.3	15.4
50	93	105	71	9.28	13.7	21.1	15.7
50	93	86	86	9.52	13.7	21.6	16.9
50	93	86	59	9.15	13.7	21.6	16.9
			HLDER, Normal ESF				
50	35	105	105	10.0	11.5	16.8	25.3
50	35	105	71	9.28	11.5	16.8	25.3
50	35	86	86	9.52	12.3	17.6	26.0
50	35	86	59	9.15	12.4	17.6	26.0

*These data correspond to a power level of 2900 MWt.

Table 6.2-53

SENSITIVITY OF DEPRESSURIZATION TO INITIAL CONDITIONS ASSUMING PSDER, MINIMUM ESF

UNIT 2

Initial Temperature (°F)		Initial Containment					
RWST	Service Water	Temp (°F)	Dew Point (°F)	P-air (psia)	P-total (psia)	Depressurization Time (sec)	Third Peak (psig)
50	95	120	96	9.25	10.10	3,280	-0.15
50	95	85	64	9.25	9.55	3,280	-0.04
50	35	120	120	12.0	13.69	2,830	-0.66
[illegible]	35	120	96	12.0	12.95	2,930	[illegible]
[illegible]	[illegible]	[illegible]	64	9.15	9.45	[illegible]	[illegible]

TABLE ~~2-1~~ 6.2-77

NORTH ANNA UNIT 1
BLOWDOWN MASS AND ENERGY RELEASE
DOUBLE-ENDED HOT LEG GUILLOTINE
[NOTE: BLOWDOWN Applicable ~~to all DEHLG Cases, i.e., all SI Cases~~ for Minimum and Maximum ESF]

TIME	BREAK PATH	NO.1 FLOW	BREAK PATH	NO.2 FLOW
		THOUSAND		THOUSAND
SECONDS	LBM/SEC	BTU/SEC	LBM/SEC	BTU/SEC
0.000	0.0	0.0	0.0	0.0
0.100	39760.5	26113.4	28011.9	18127.0
0.200	36656.6	24029.9	23800.5	15324.4
0.401	34532.8	22601.0	20612.5	12907.7
0.600	34123.2	22367.0	18899.5	11478.0
0.900	32260.4	21421.6	17486.7	10237.1
1.10	31478.7	21173.5	16930.4	9732.2
1.30	30372.8	20665.4	16745.4	9486.1
2.00	26058.4	18256.0	16540.9	9066.1
2.50	23500.6	16604.1	15916.1	8647.8
3.10	21310.5	15007.2	14647.1	7967.2
3.40	20599.8	14396.8	13925.7	7601.3
4.00	19787.8	13527.6	12366.4	6808.9
4.40	19804.3	13347.6	11260.0	6231.4
5.00	20927.5	13597.0	9798.9	5458.0
5.20	16357.4	11542.8	9347.7	5217.9
5.60	16477.5	11328.8	8451.9	4741.4
6.20	16851.0	11339.6	7509.9	4254.4
6.80	17022.6	11290.7	6767.3	3877.3
7.60	17036.0	11027.9	5969.7	3478.5
8.00	16911.6	10828.2	5621.3	3305.3
8.60	16177.3	10268.3	5130.8	3062.9
10.60	12737.0	8075.9	3716.3	2388.5
12.00	10333.1	6661.6	2909.2	2026.0
14.20	6527.6	4691.4	1790.8	1534.0
14.80	5426.1	4224.7	1486.4	1390.0
16.00	3564.9	3436.6	1197.6	1229.6
16.40	2254.4	2541.1	1134.5	1201.1
16.80	1561.9	1927.6	1066.6	1167.7
17.80	974.0	1242.7	819.3	984.8
19.20	504.9	649.7	280.9	359.2
19.80	330.3	426.9	135.4	175.2
22.20	327.7	425.6	84.4	110.0
23.20	498.1	619.7	151.6	196.7
24.00	0.0	0.0	0.0	0.0

Table 6.2-77 (continued)

NORTH ANNA UNIT 1
REFLOOD AND POST REFLOOD MASS AND ENERGY RELEASE
DOUBLE-ENDED HOT LEG GUILLOTINE - MAX SI / LHSI - 6000 GPM

TIME	STEAM RELEASE		WATER RELEASE	
SECONDS	LBM/SEC	1000 BTU/SEC	LBM/SEC	1000 BTU/SEC
24.0	0.0	0.0	0.0	0.0
24.3	0.0	0.0	0.0	0.0
24.5	544.0	263.4	0.0	0.0
24.6	349.1	291.1	0.0	0.0
27.5	1125.1	477.9	0.0	0.0
31.3	1740.8	605.3	0.0	0.0
41.0	1631.5	582.1	0.0	0.0
46.4	1574.3	566.2	2774.1	222.6
50.0	1533.2	554.6	2384.3	187.6
54.0	1490.5	542.8	2164.9	[illegible]
55.3	1477.3	461.2	1939.5	148.1
70.8	965.2	385.9	0.0	0.0
100.0	707.1	343.3	0.0	0.0
102.6	682.1	339.3	0.0	0.0
128.0	647.2	326.5	0.0	0.0
136.5	642.8	323.7	0.0	0.0
136.6	124.0	147.4	0.0	0.0
200.0	111.3	132.3	0.0	0.0
500.0	80.6	95.8	825.1	19.1
1000.0	62.5	74.3	855.8	19.9
1499.9	55.5	65.9	873.9	20.3
1500.0	69.8	83.0	880.9	20.4
2000.0	64.7	76.9	866.6	226.9
5000.0	49.5	58.9	871.7	228.3
10000.0	40.6	48.2	886.9	232.2
20000.0	33.3	39.6	895.8	234.6
50000.0	25.8	30.6	903.1	236.5
100000.0	21.1	25.1	910.6	238.5
1000000.0	10.1	12.0	915.3	239.7
			926.3	242.6

ENTRAINMENT ENDS AT 136.50 SECONDS

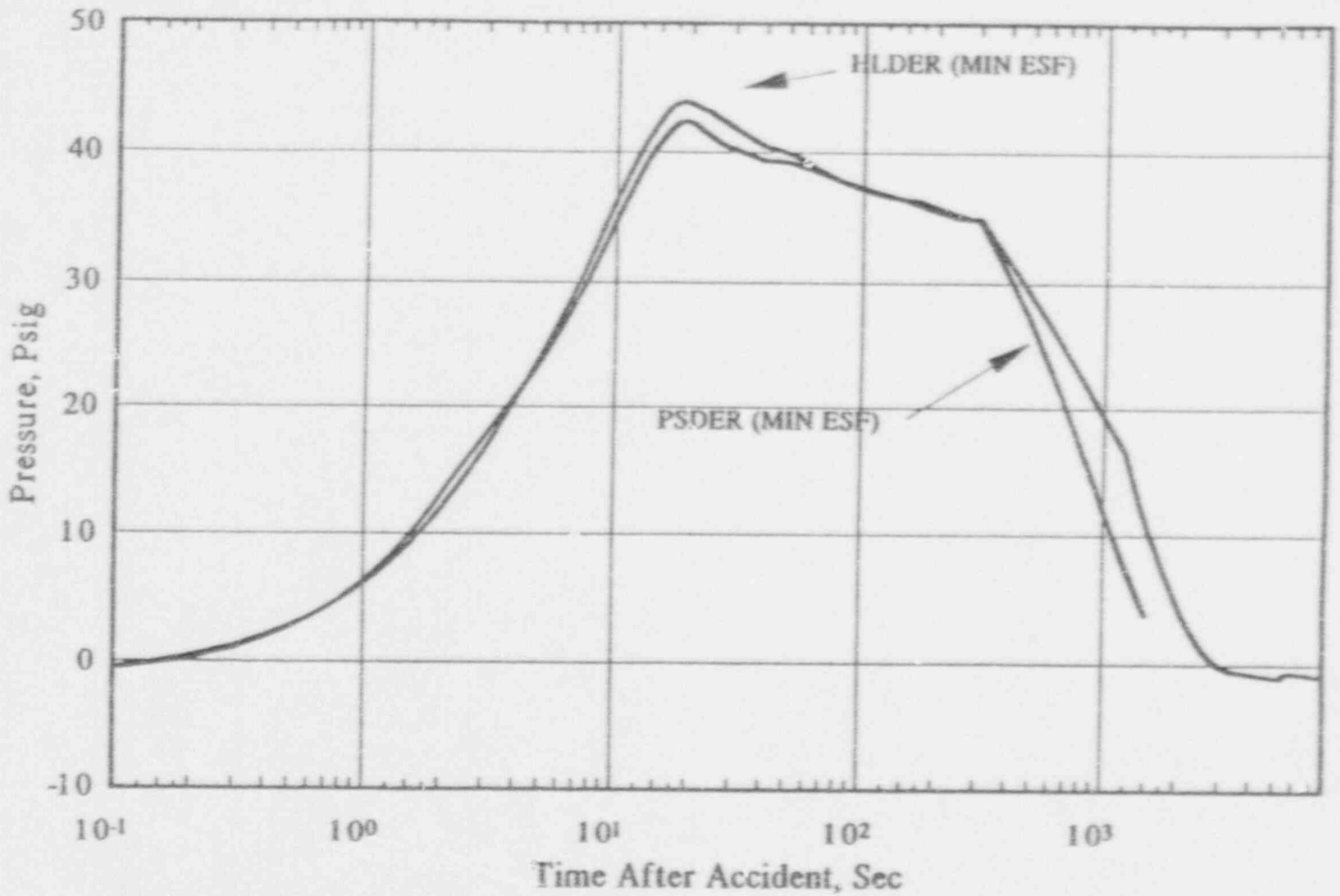
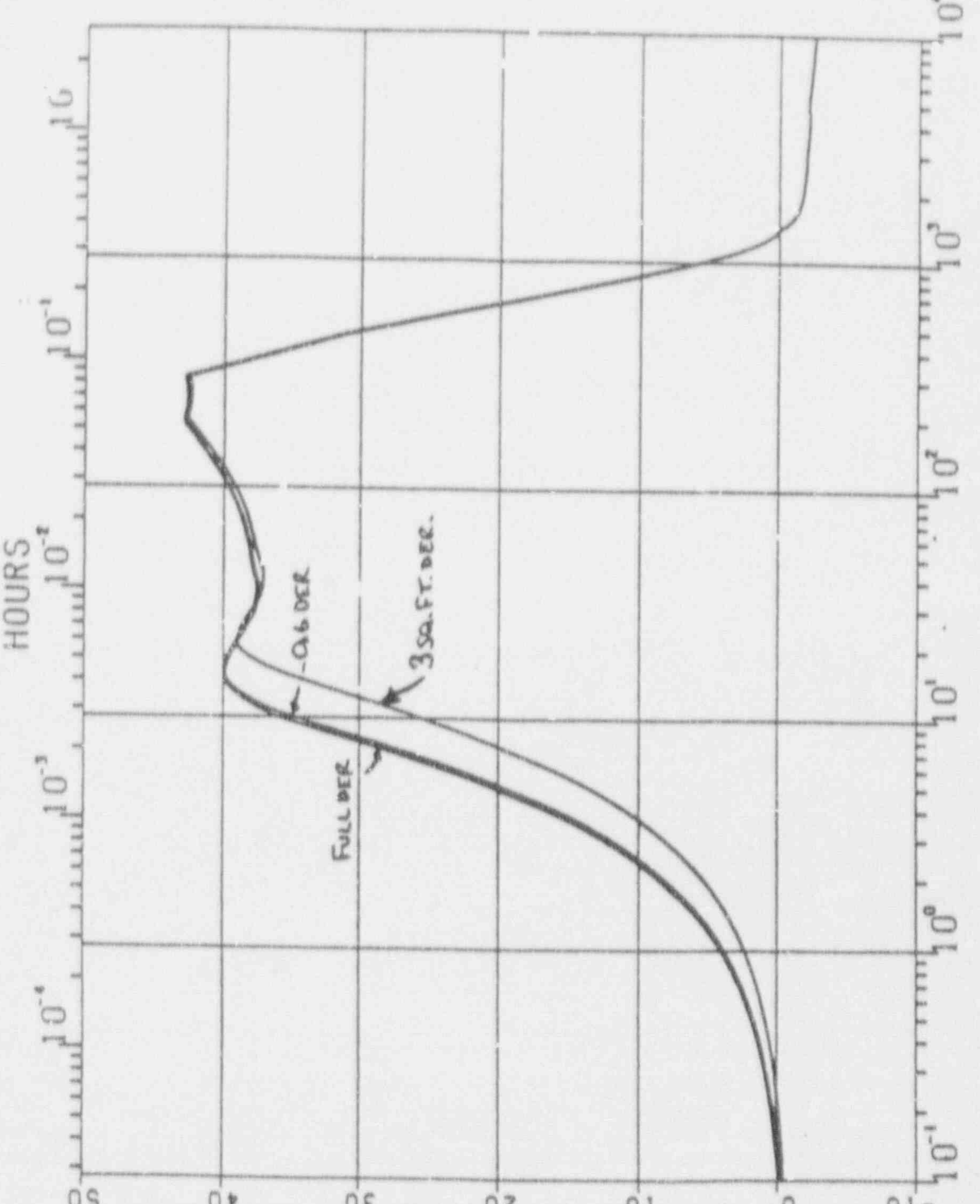


Figure 6.2-11A
Containment Pressure
Containment Peak Pressure Analysis
Spectrum of Break Locations
North Anna Unit I

Figure 6.2-11A

CONTAINMENT PRESSURE VS TIME
SPECTRUM OF BREAK LOCATIONS
NORTH ANNA UNIT 2

FAILURE OF ONE QUENCH SPRAY PUMP



CONTAINMENT PRESSURE VS TIME
SPECTRUM OF PUMP SUCTION BREAKS
NORTH ANNA UNIT 2

FAILURE OF ONE QUENCH SPRAY PUMP

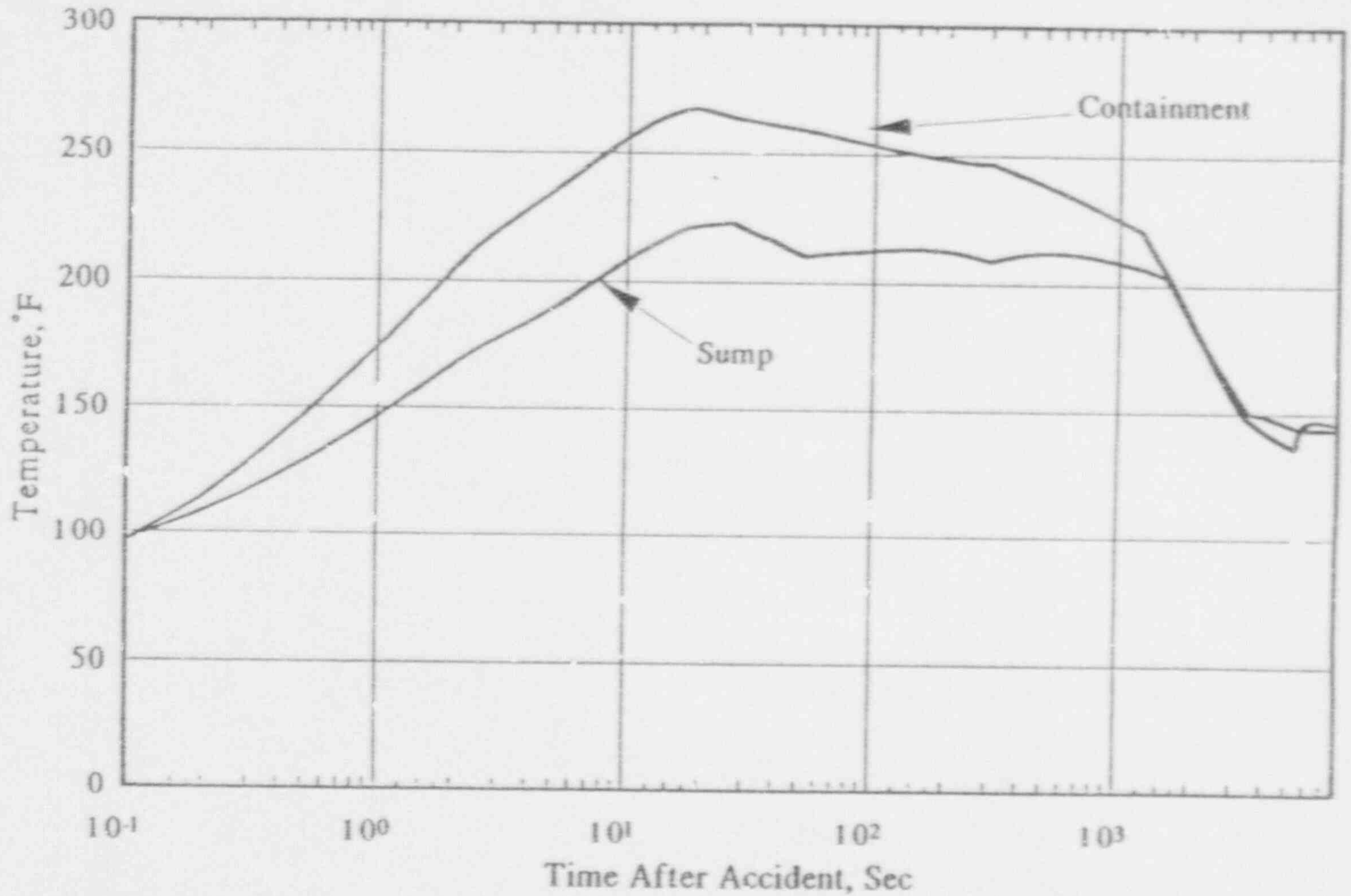
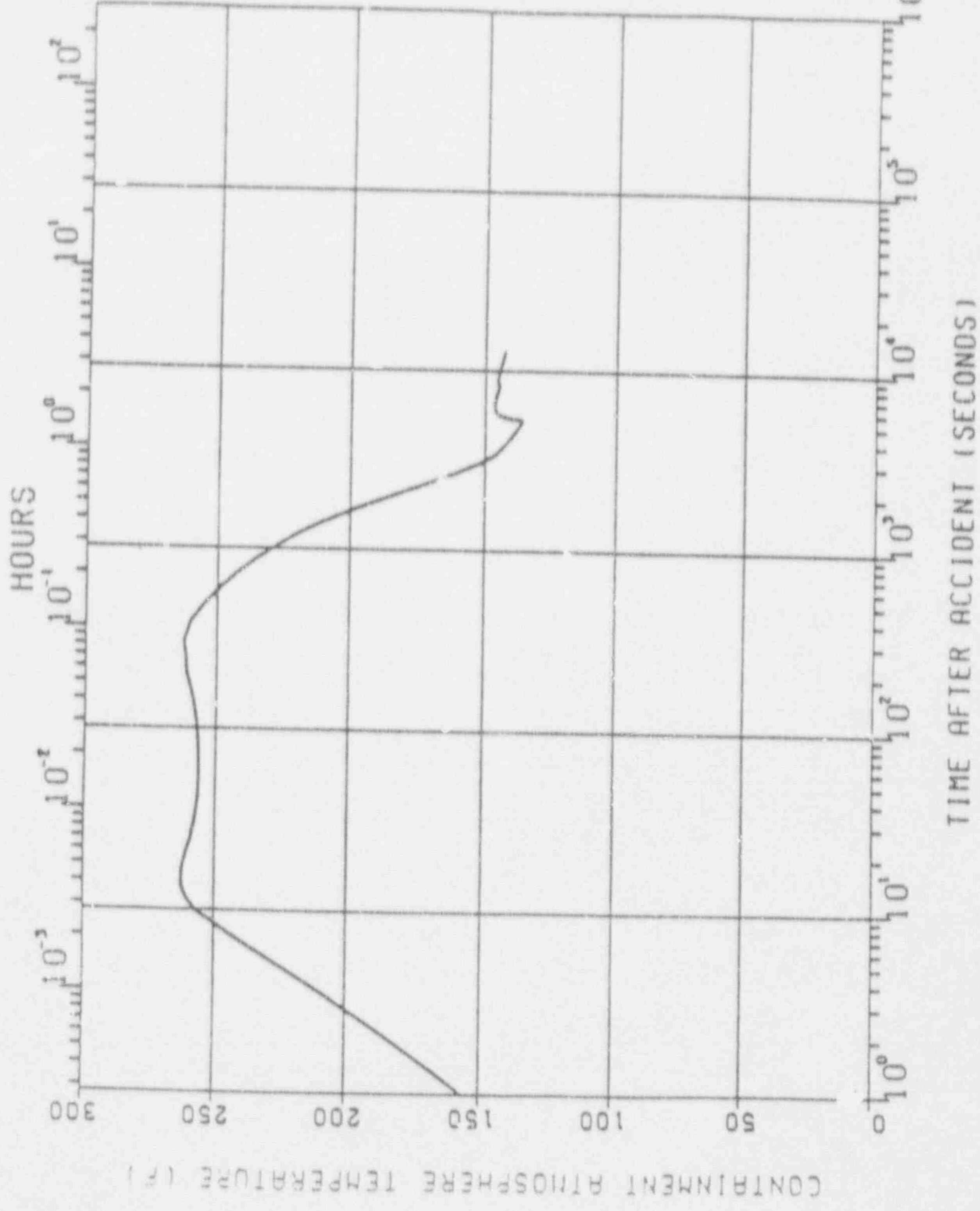


Figure 6.2-94 A
Containment and Sump Temperature
Depressurization Analysis
PSDER - Minimum ESF
North Anna Unit 1



TEMPERATURE TRANSIENT
WORST CASE DEPRESSURIZATION
NORTH ANNA UNIT 2

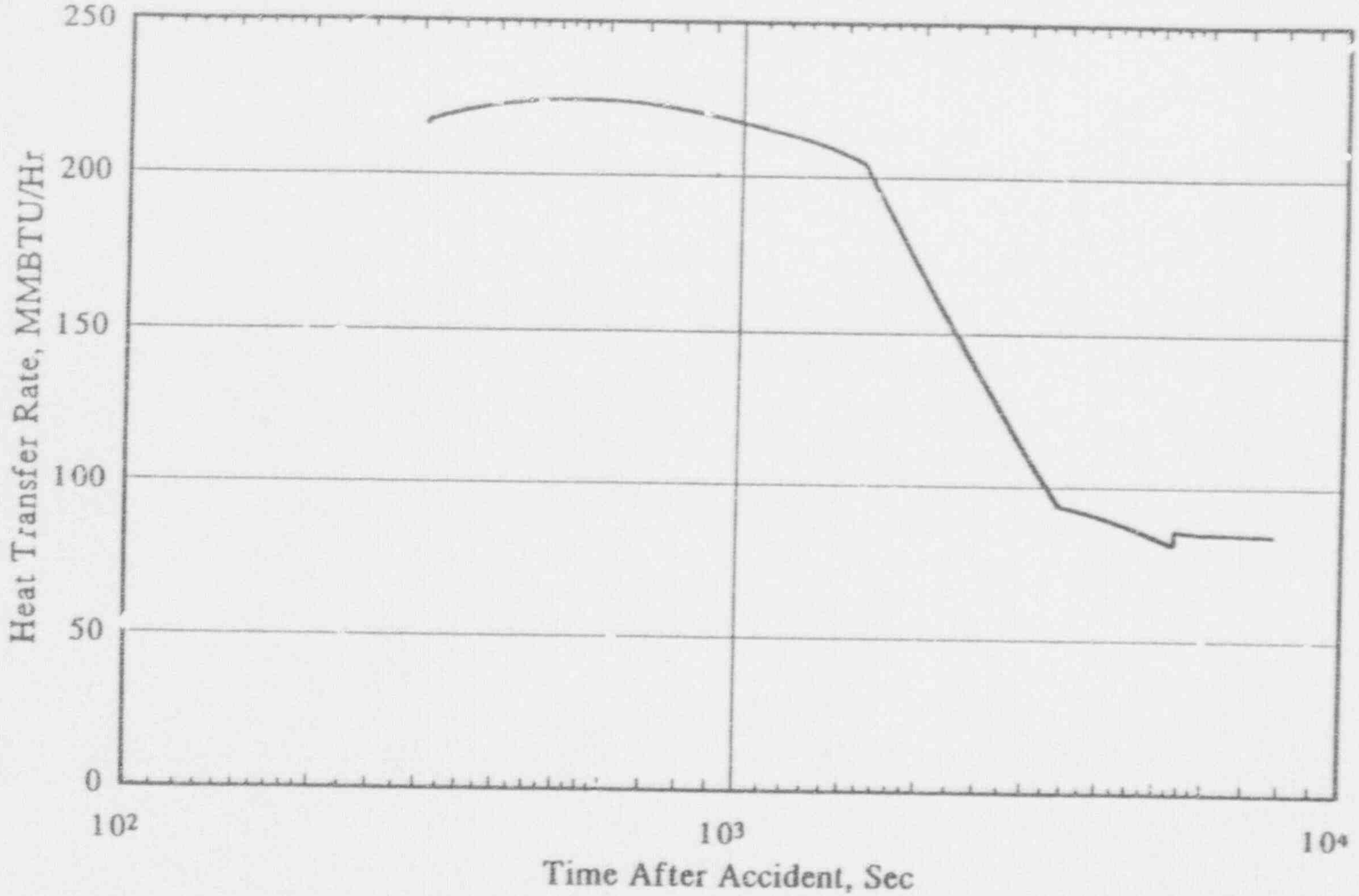
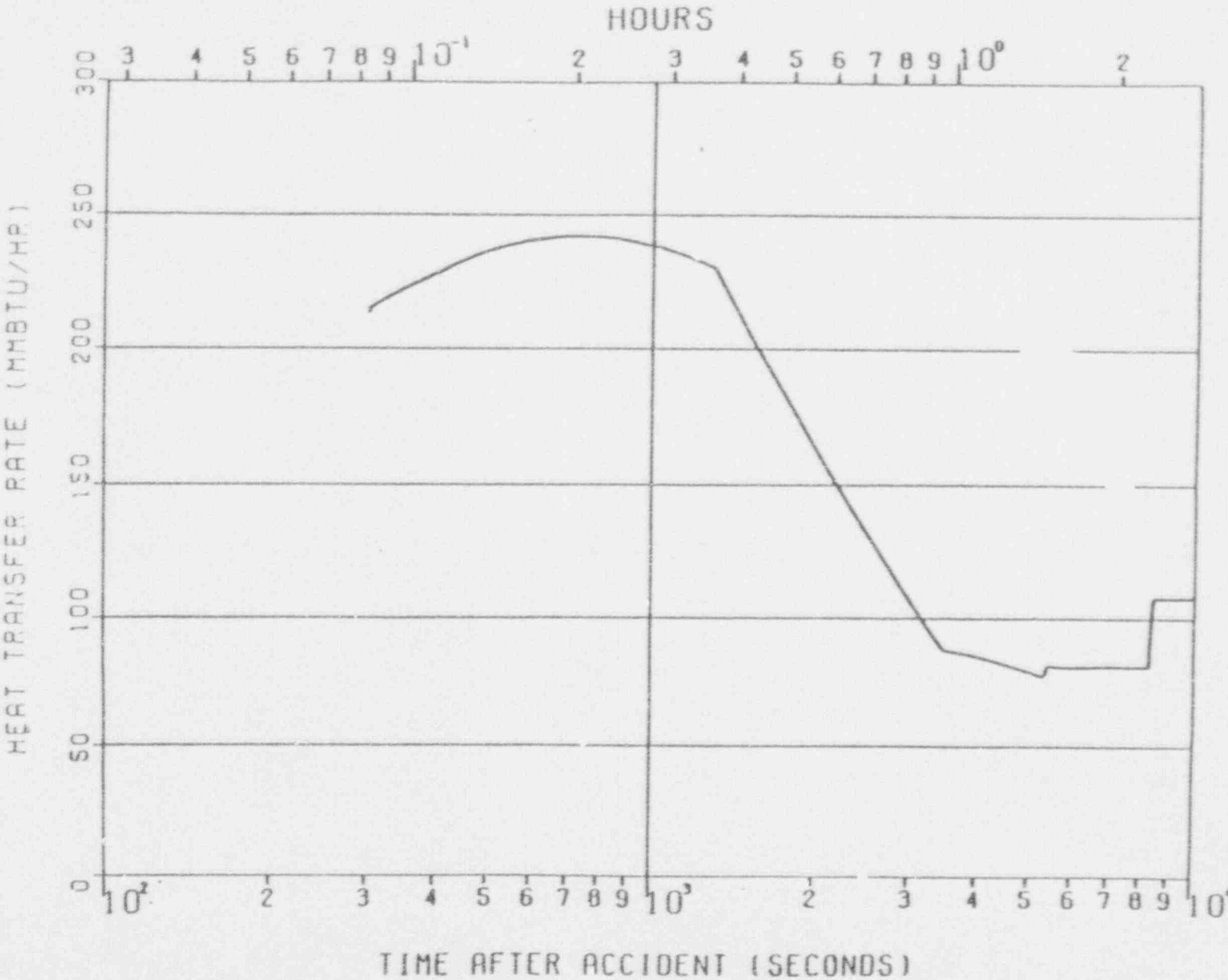


Figure 6.2-95A
Recirculation Spray Cooler Duty
Depressurization Analysis
PSDER - Minimum ESF
North Anna Unit 1



RECIRC. SPRAY COOLER DUTY
WORST CASE DEPRESSURIZATION
F60ER MINIMUM ESF

NORTH ANNA UNIT 2

Figure 6.2-95B

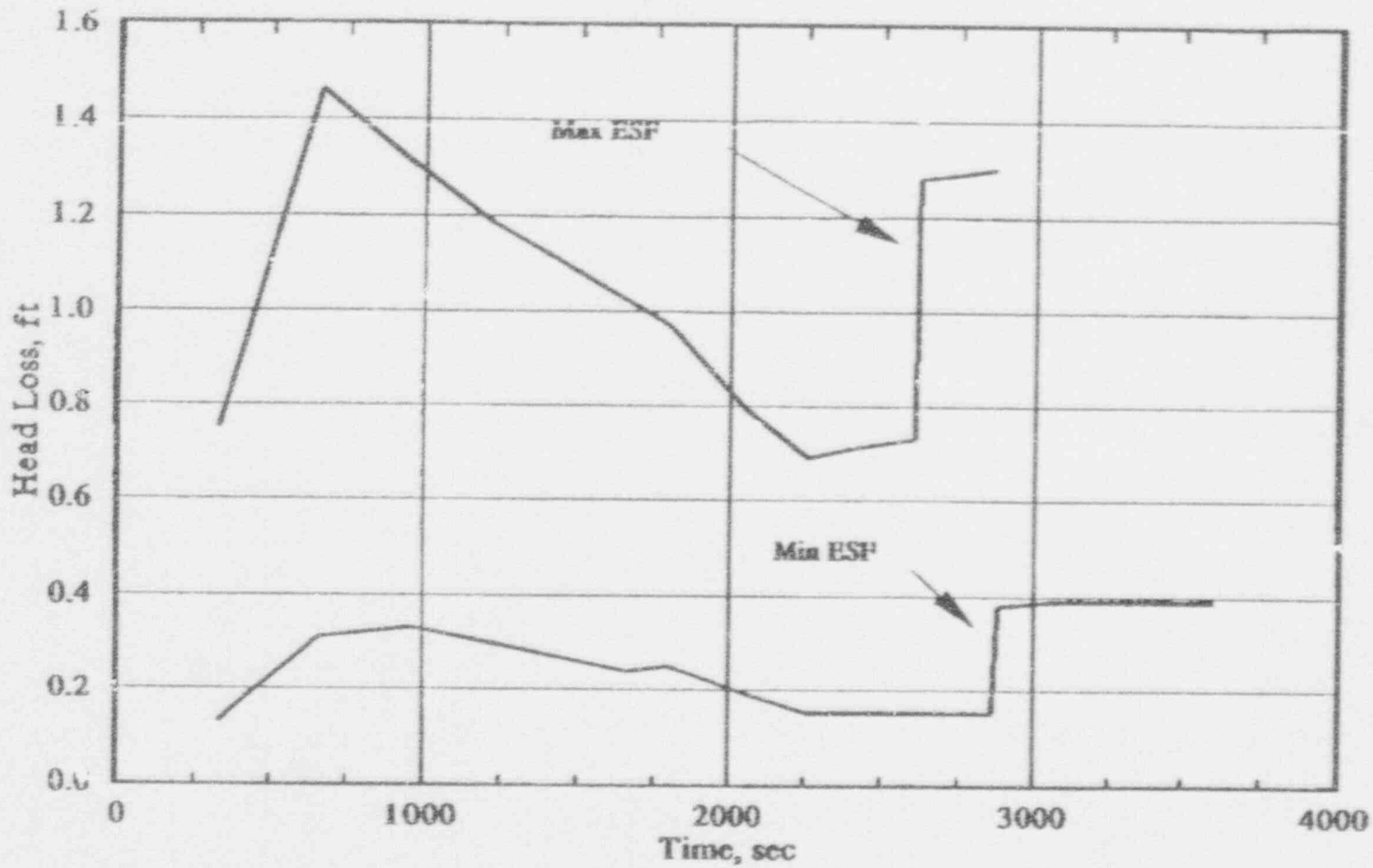


Figure 6.2-116
Insulation Debris Head Loss Following a LOCA
Unit 1
Updated Final Safety Analysis Report

Adequate NPSH is shown to be available for all pumps as follows:

1. Low Head Safety Injection - The NPSH of the LHSI pumps is evaluated for both the injection and recirculation modes of operation for various LOCAs. Recirculation mode operation gives the limiting NPSH requirement.

   As can be seen in Table 6.2-50, the pump suction double-ended rupture (PSDER) with minimum ESF is the worst case. Maximizing the sump water temperature minimizes the NPSH available. This is done by minimizing containment heat removed and maximizing the sump and RWST water temperatures. This yields an NPSHA of 13.5 ft Insert K at a flow rate of 4030 gpm.

   Tables 6.2-51 and 6.2-52 give the results of sensitivity studies performed by varying the break size and the initial conditions.

   The sensitivity of the NPSH to the condensing heat transfer coefficient (Tagami) is shown by comparing a pre-uprate calculation that uses a condensing heat transfer coefficient with a peak value equal to four times Tagami. The calculated NPSHA for the LHSI pump (PSDER with minimum ESF) was the same for both four times Tagami, and Tagami. This demonstrates that NPSH is insensitive to this parameter.

   For the 4030-gpm maximum flow calculated for minimum ESF, NPSHA analyses for the LHSI pumps show that NPSH is always above the required NPSH (NPSHR) of 13.4 ft.
   13.1

   Transient conditions encountered during pump start or during a switchover of suction expose the suction impeller, as well as other parts of the pump, to short-term mechanical and hydraulic changes that will not alter long-term capability or reliability. Extensive shop testing of vertical can-type pumps has shown that when exposed to suction conditions well below steady-state requirements for [illegible] impact on performance.

Insert K

for Unit 2 and 14.1 ft for Unit 1. The Unit 1 analysis includes the effect of sump screen pressure drop caused by insulation debris. The debris analysis is discussed in Section 6.2.2.2.

following a LOCA is terminated before the RWST is completely emptied, all pipes are kept filled with water before recirculation is initiated.

Manual switchover from injection to recirculation can be accomplished by the operator. The sequence (as delineated in Table 6.3-6) is followed, regardless of which power supply is available (offsite or emergency onsite). The time required to complete the operation is the time for the switchgear to function. Controls for ECCS components are grouped together on the main control board. The component position lights verify when the function of a given switch has been completed.

If the operator fails to manually accomplish the switchover prior to a minimum level in the RWST, automatic switchover from injection to recirculation mode will occur.

The Train A and Train B automatic switchovers are initiated when actuation signals are generated by the two-of-four RWST low-low level protection logic and the safeguards protection logic (safety injection signal). The automatic switchover sequence is shown below for Train A valves with Train B valves in parentheses.

1. Valve 1863A (1863B) opens.
2. Valves 1885 A and C (1885B and D) close.
3. Valve 1860A (1860B) opens.
4. Valve 1862A (1862B) closes.

341,290 gal (unit 1) / 315,170 gal (unit 2) The nominal low level alarm for initiation of manual switchover is set at gal expended from the RWST (assuming that the RWST level is initially at the "tank not full" alarm level). Assuming the error in the level instrumentation to be giving maximum high reading and a minimum RWST drawdown rate of 6700 gpm (minimum ESF) with the quickest transfer completion of 125 sec, the earliest that the transfer could be completed is when have been expended from the RWST. This value is used for the LHSI-NPSH calculations since early transfer results in minimum NPSH for the LHSI pumps.

355248 gal (unit 1)
329128 gal (unit 2)

The latest transfer completion is based on the following:

1. Operator ignores manual transfer alarm and automatic switchover starts at ⚠ gal expended from the RWST (from "tank not full" level).
{357,470 gal (unit 1)
{331,350 gal (unit 2)

2. The level instrument error is giving its minimum level reading.

3. Maximum time to complete switchover equals 240 Sec (unit 1) 480 Sec (unit 2)

384270 gal (unit 1)
384950 gal (unit 2)

The above assumptions result in a maximum of ⚠ gal expended before switchover is completed. This value is used for all depressurization calculations in Section 6.2.2 since later switchover results in less RWST water available for the quench sprays in maintaining the containment pressure subatmospheric (refer to Figure 6.3-11A Unit 1) and Figure 6.3-11B (Unit 2)).

Provisions are included in the system to permit online testing of the switchover sequence without affecting normal plant operation.

6.3.2.2.3 Recirculation Mode After Loss of Primary Coolant

After the injection operation, water collected in the containment sump is returned to the reactor coolant system by the low head or low head/high head recirculation flow paths. Cooling of sump water is provided by the RS subsystems (Section 6.2).

Water enters the sump and passes through the cylindrical screen surrounding the suction intake to the LHSI pumps. The water then enters a 12-in. pipe and flows to the LHSI pumps through a containment isolation valve, a check valve, and, in the case of Unit 1, a gate valve. The discharge from 4006 the LHSI pumps takes one of the following two paths: through containment isolation valves to 6-in. lines into the individual reactor coolant loops, or through 8-in. headers to the suction of the high head safety injection/charging pumps and then from the charging pumps through containment isolation valves to 2-in. lines into individual reactor coolant loops. From the loops, in both cases, the injected fluid flows through the RCS.

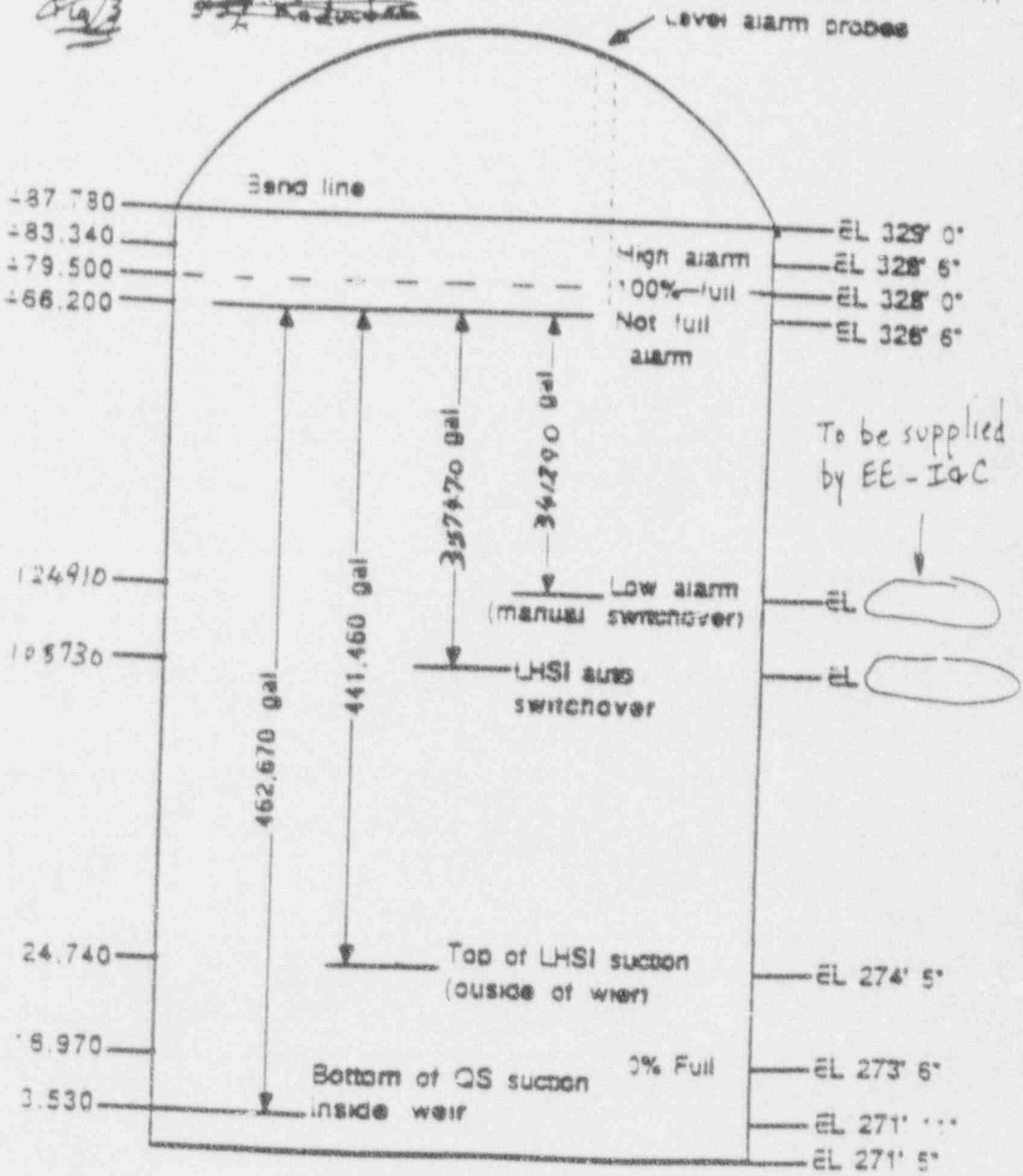
ENGINEERING WORK SHEET SHEET NO. 14 OF 31
Document Type- SM Calc Note Doc. No. SM-471 Rev. 0 Add. B
VIRGINIA POWER Project- RWST Level for Containment Temperature Increase at NA
System- North Anna Units 1 and 2 (VRA/VGB)
Prepared By GMB Date 6-11-92 Reviewed By JCS Date 6/12/92

~~Figure 3~~

REVISION 11

NAPS UFSAR Figure 6.2-11 A

Fig 3 ~~Reduction~~



Nominal water levels shown.
Tank radius= 19' 0".

RWST LEVEL AND ALARM SET POINTS
NORTH ANNA POWER STATION
UNIT 1

~~SM Calc Note SM-471 Rev. 0 Add. B~~

DC 90-13-1, Appendix 4-2, Page 200

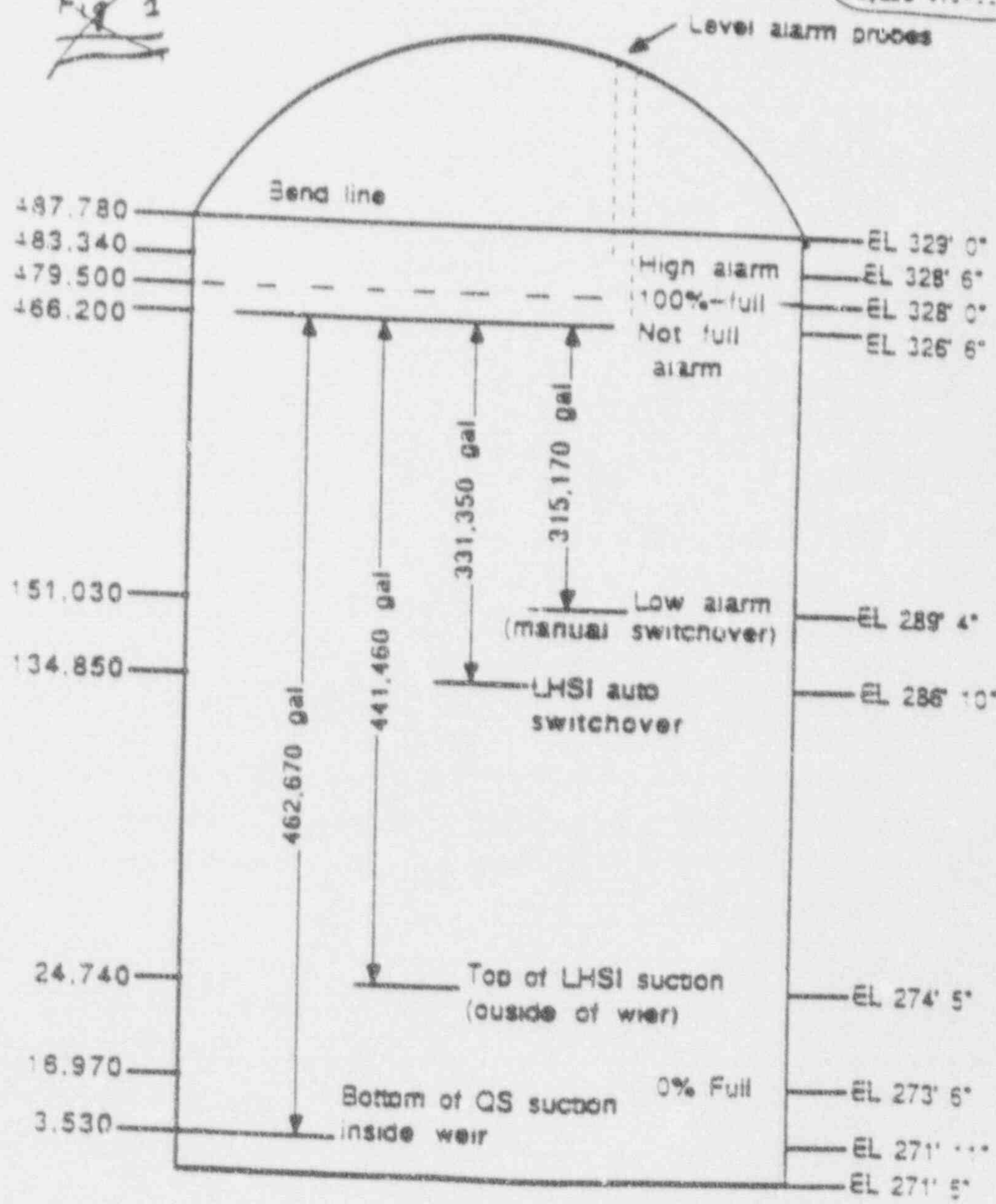
~~Figure 1~~

REVISION 15

~~Fig 1~~

Figure 6.3-11 B

NAPS UFSAR Figure 6.3-11



Nominal water levels shown.
Tank radius = 19' 0".

RWST LEVEL AND ALARM SET POINTS
NORTH ANNA POWER STATION
UNIT 2

~~SM Calc Note SM-471 Rev. 0 Add. B~~

UNIT 2

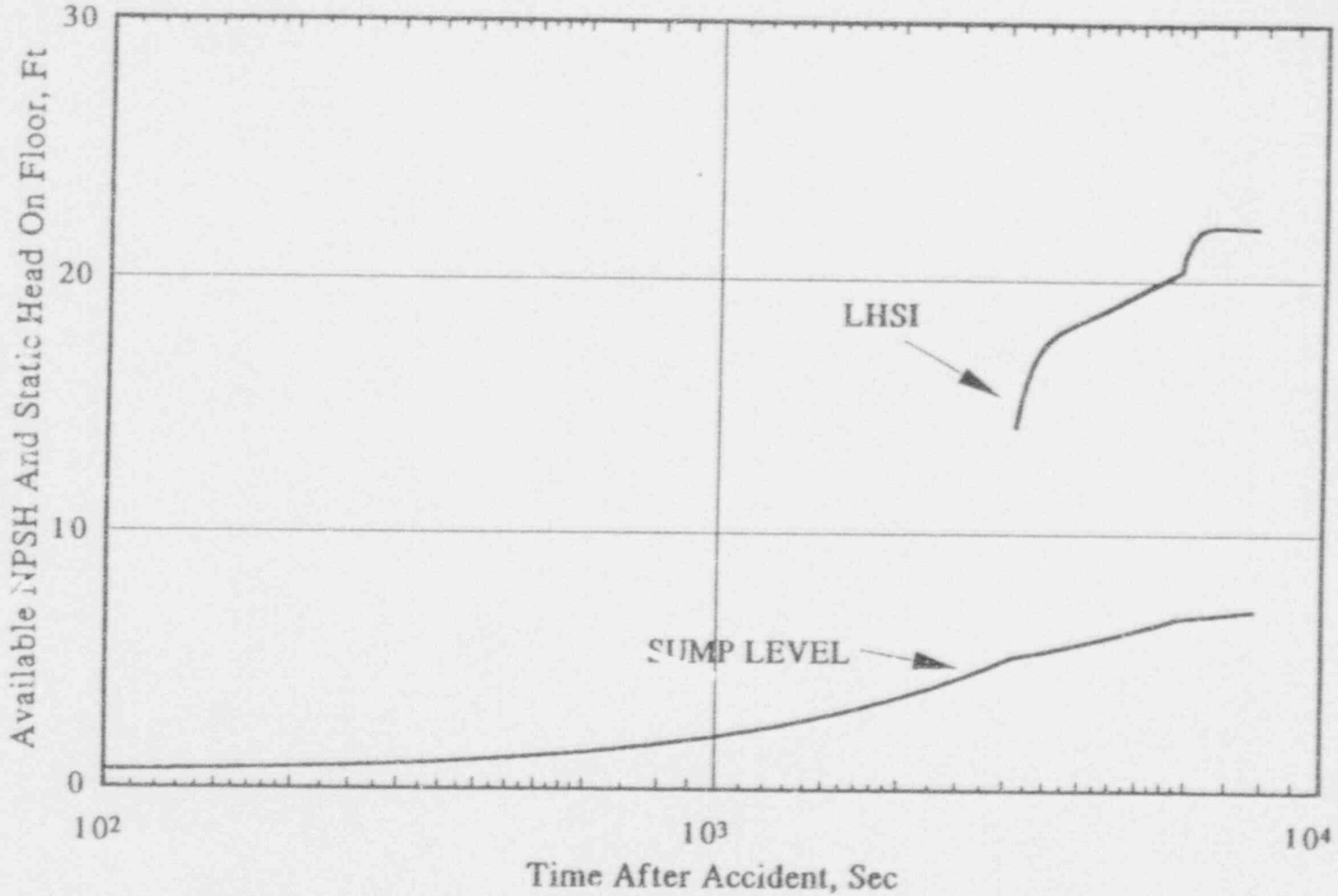
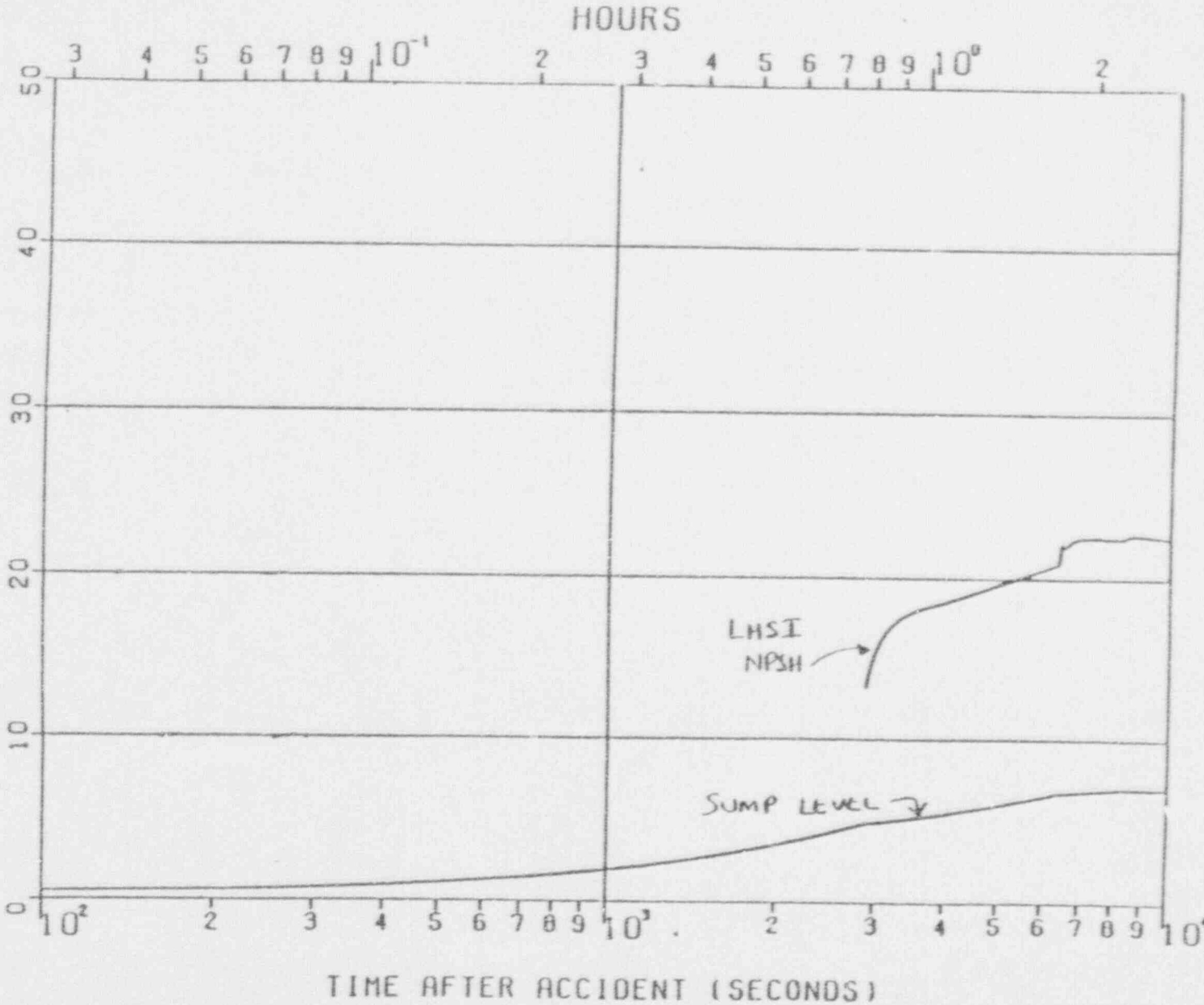


Figure 6.3-12A
LHSI Pump Available NPSH
PSDER - Minimum ESF
North Anna Unit 1



NPSHA TRANSIENTS
WORST CASE LHSI PUMP NPSH
POWER MIN ESF
North Anna Unit 2

Figure 6.3-12B

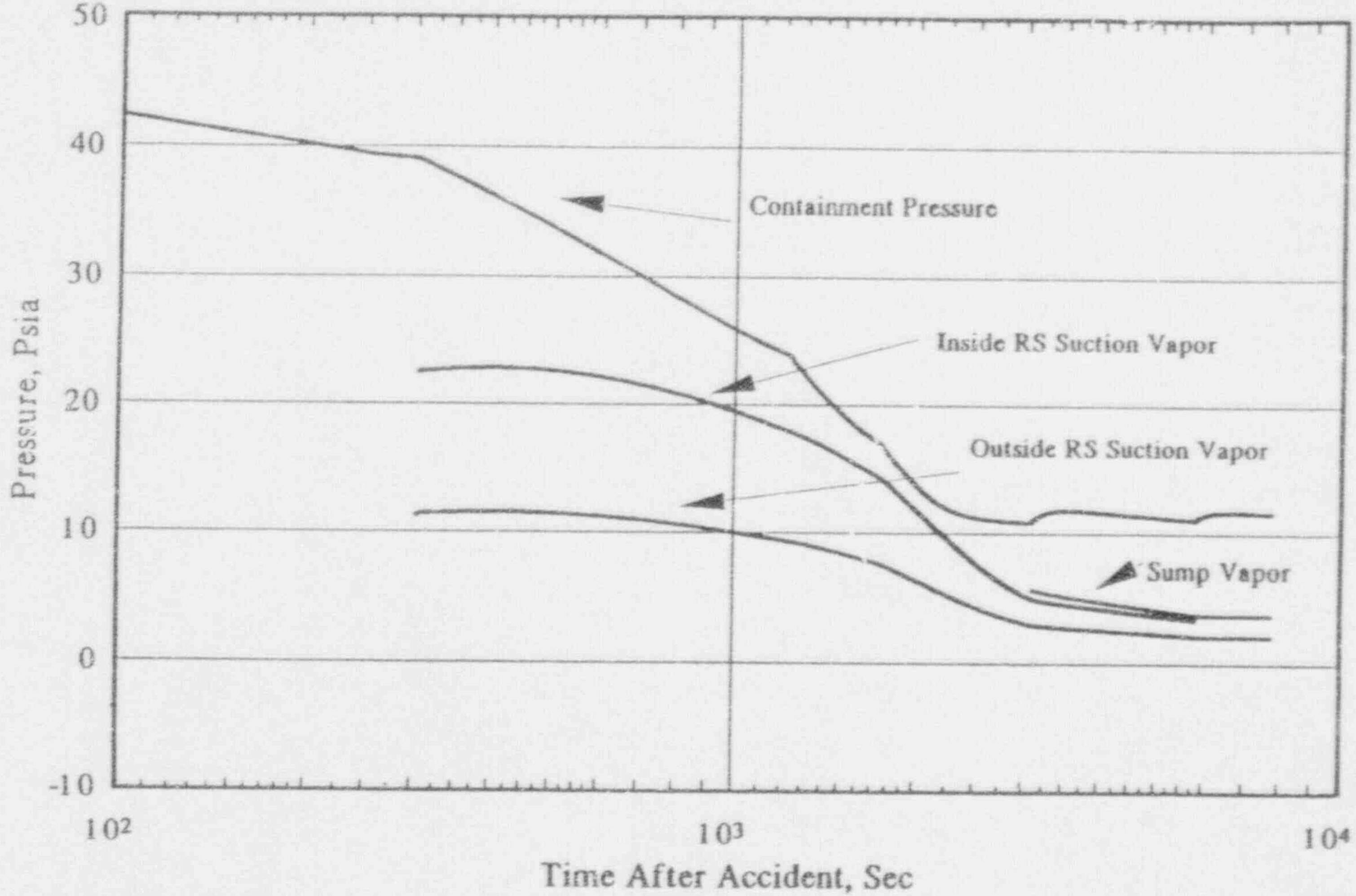
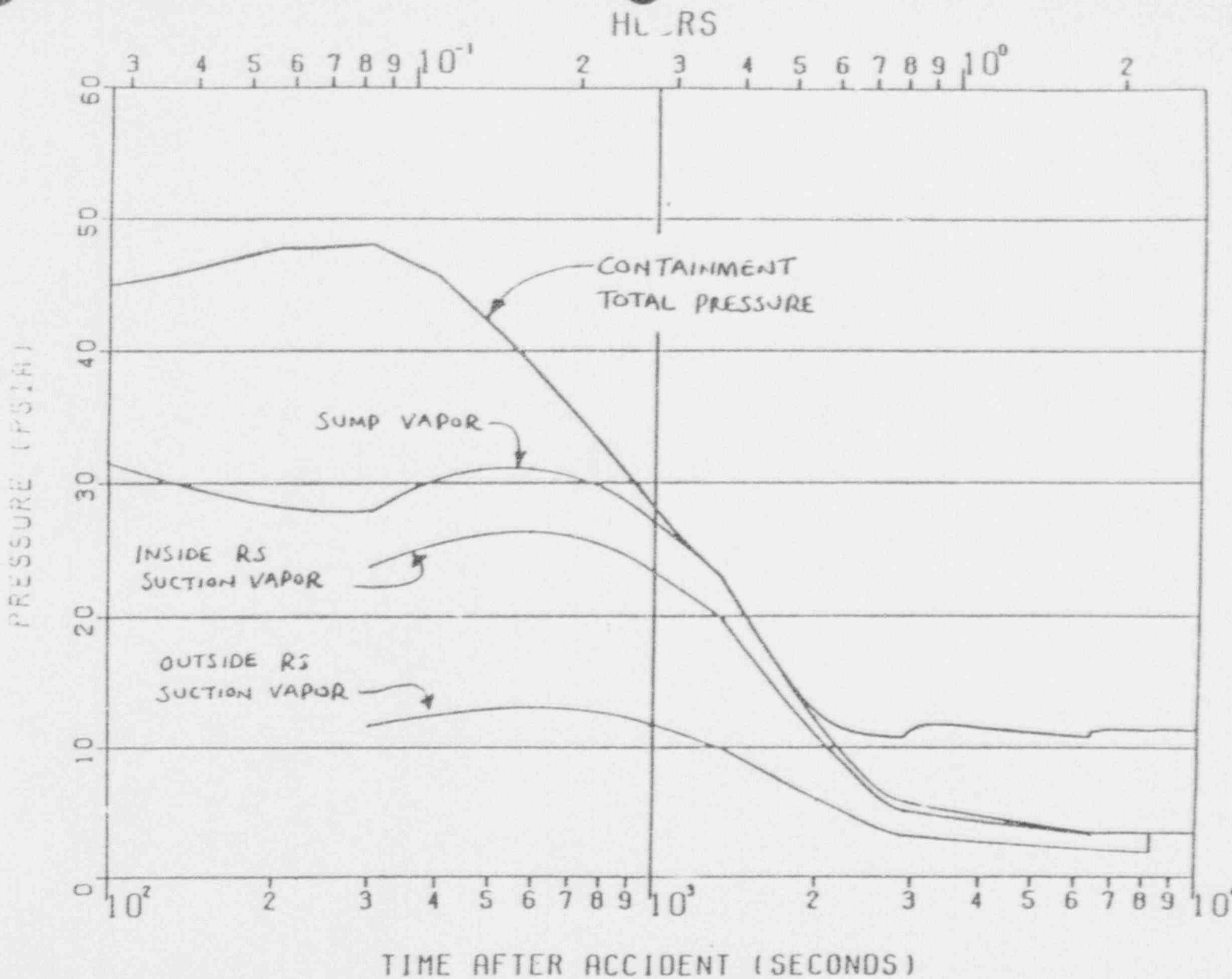


Figure 6.3-13A

Containment and Suction Vapor Pressure
LHSI Pump NPSH Analysis
PSDER - Minimum ESF
North Anna Unit 1



PRESSURE TRANSIENTS
WORST CASE LHSI PUMP NPSH
FOUER MINIMUM ESF

North Anna Unit 2

Figure 6.3-13B

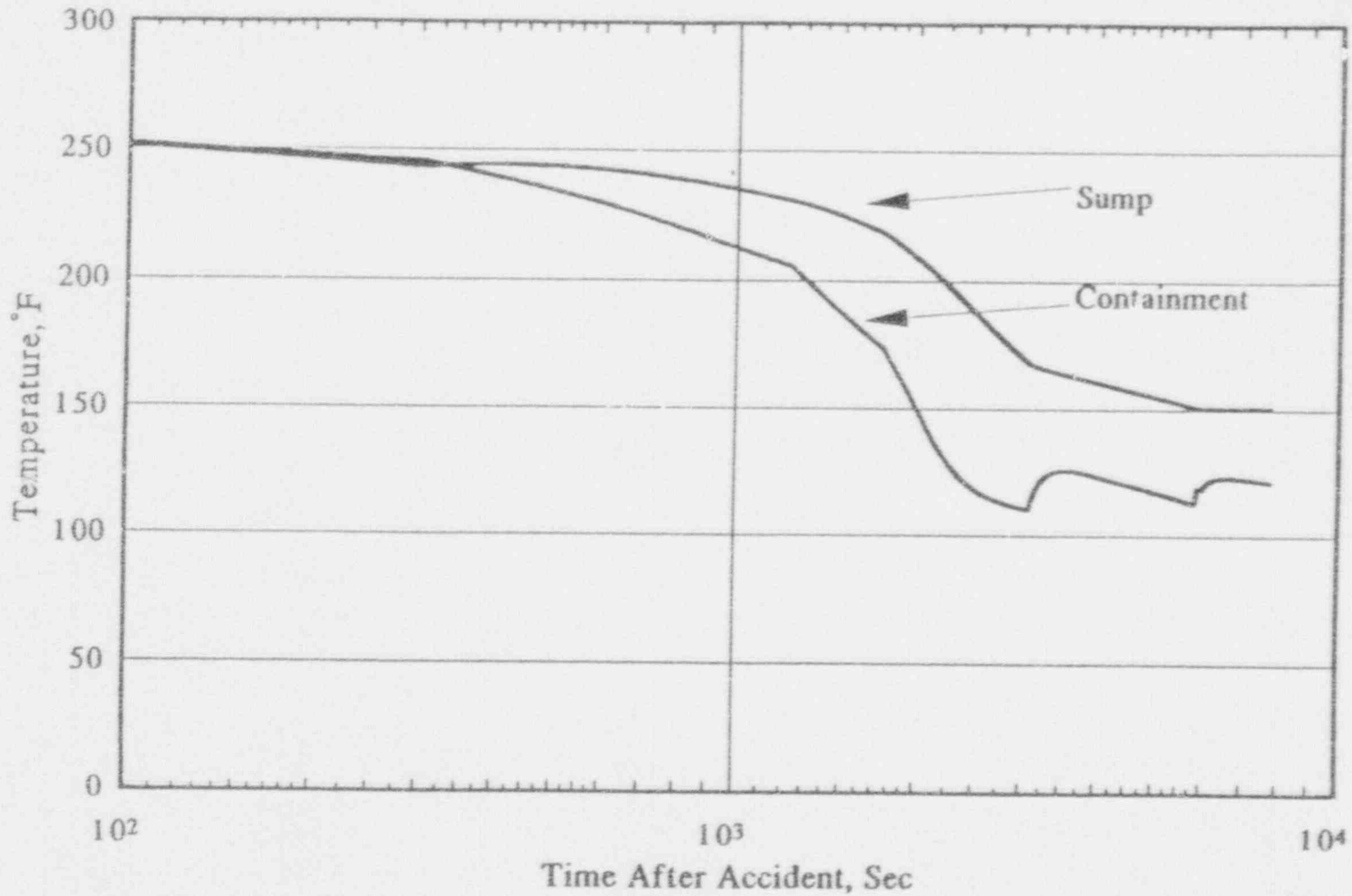
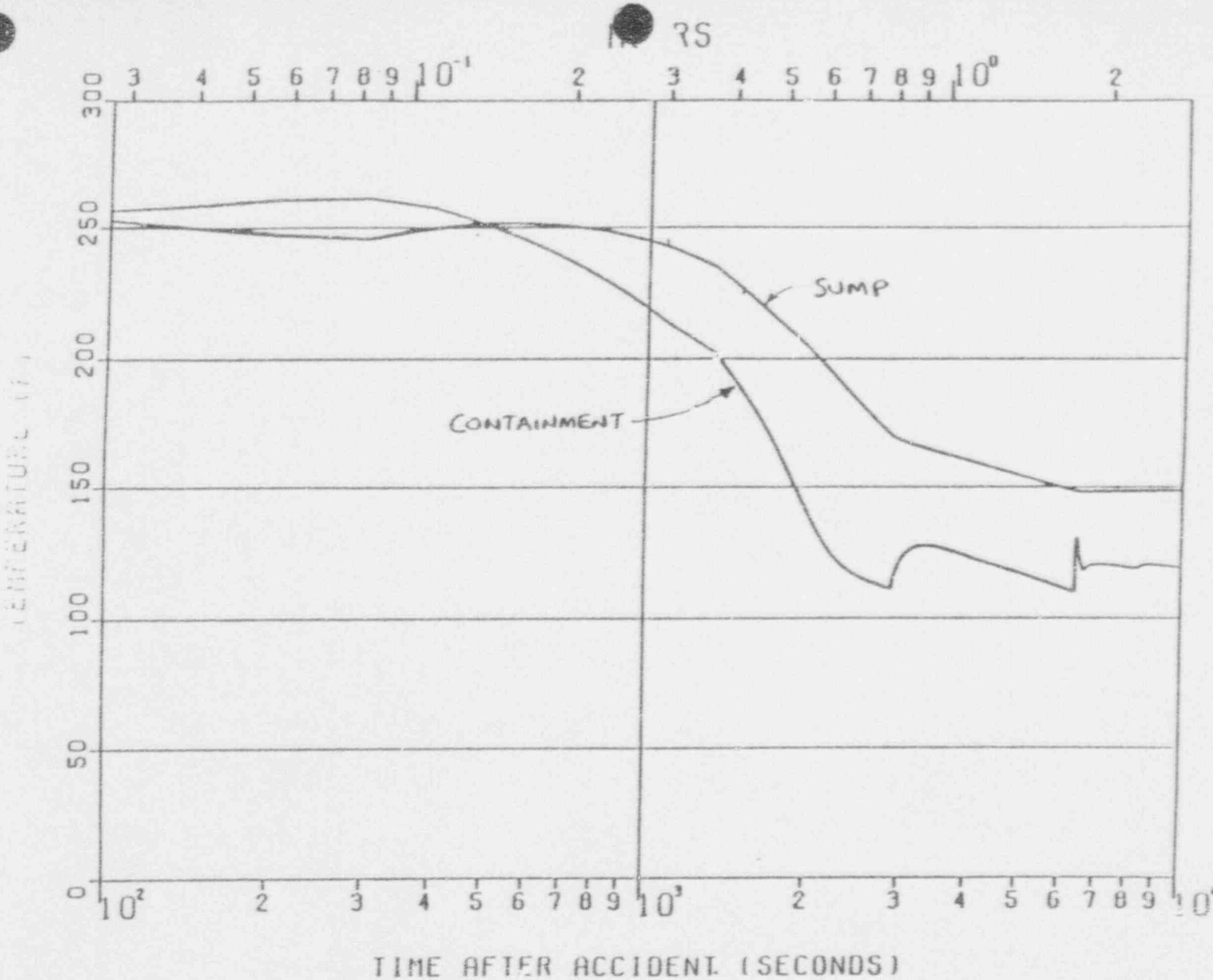


Figure 6.3-14A
Containment and Sump Temperature
LHSI Pump NPSH Analysis
PSDER - Minimum ESF
North Anna Unit 1



TEMPERATURE TRANSIENTS
WORST CASE LHSI PUMP NPSH
UNDER MINIMUM ESF

North Anna Unit 2

Figure 6.3-14B

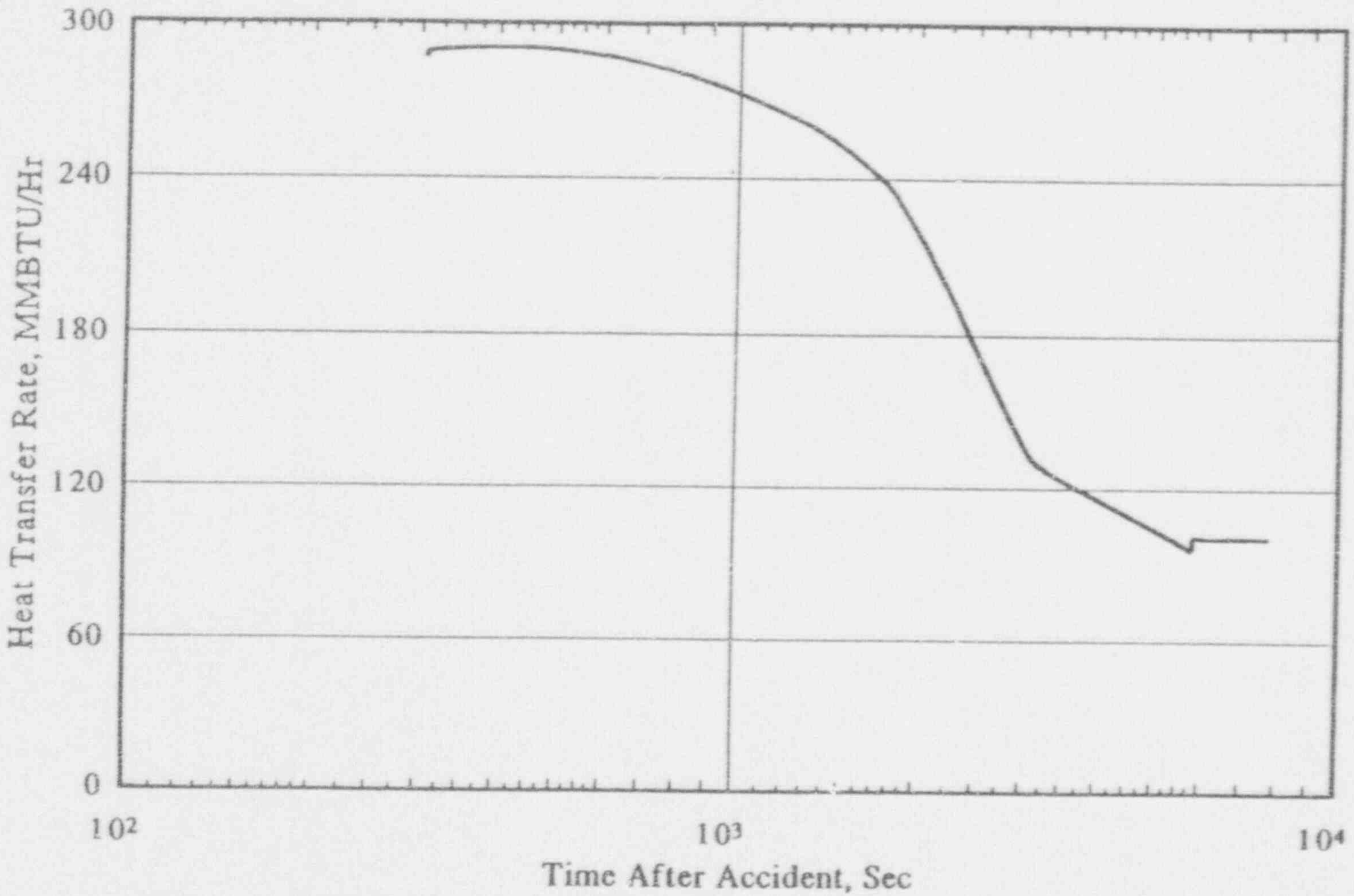
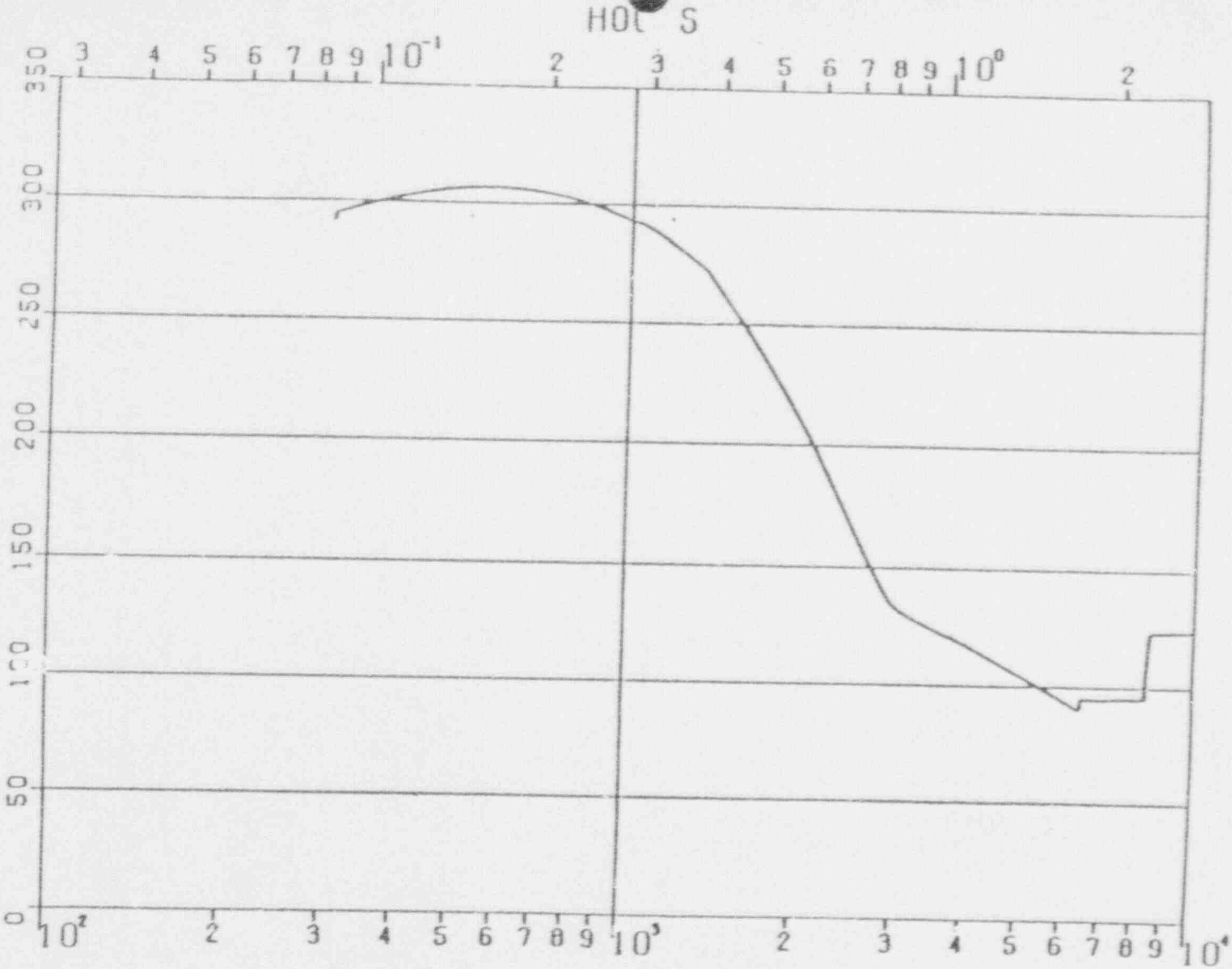


Figure 6.3-15 A
Recirculation Spray Cooler Duty
LHSI Pump NPSH Analysis
PSDER - Minimum ESF
North Anna Unit 1



RECIRC. SPRAY COOLER DUTY
WORST CASE LHSI PUMP NPSH
FOUR MINIMUM ESF

North Anna Unit 2

Figure 6.3-15B

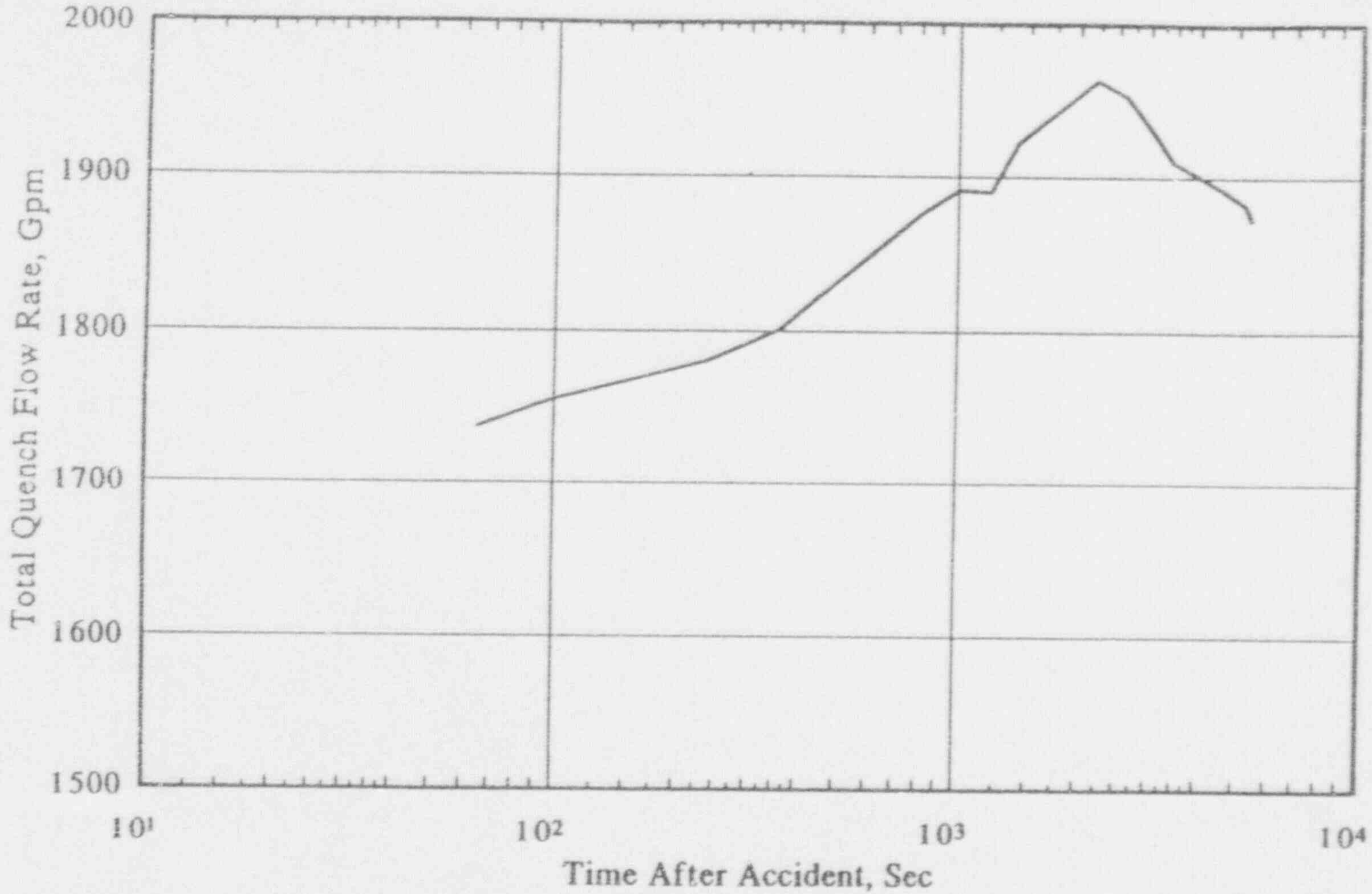
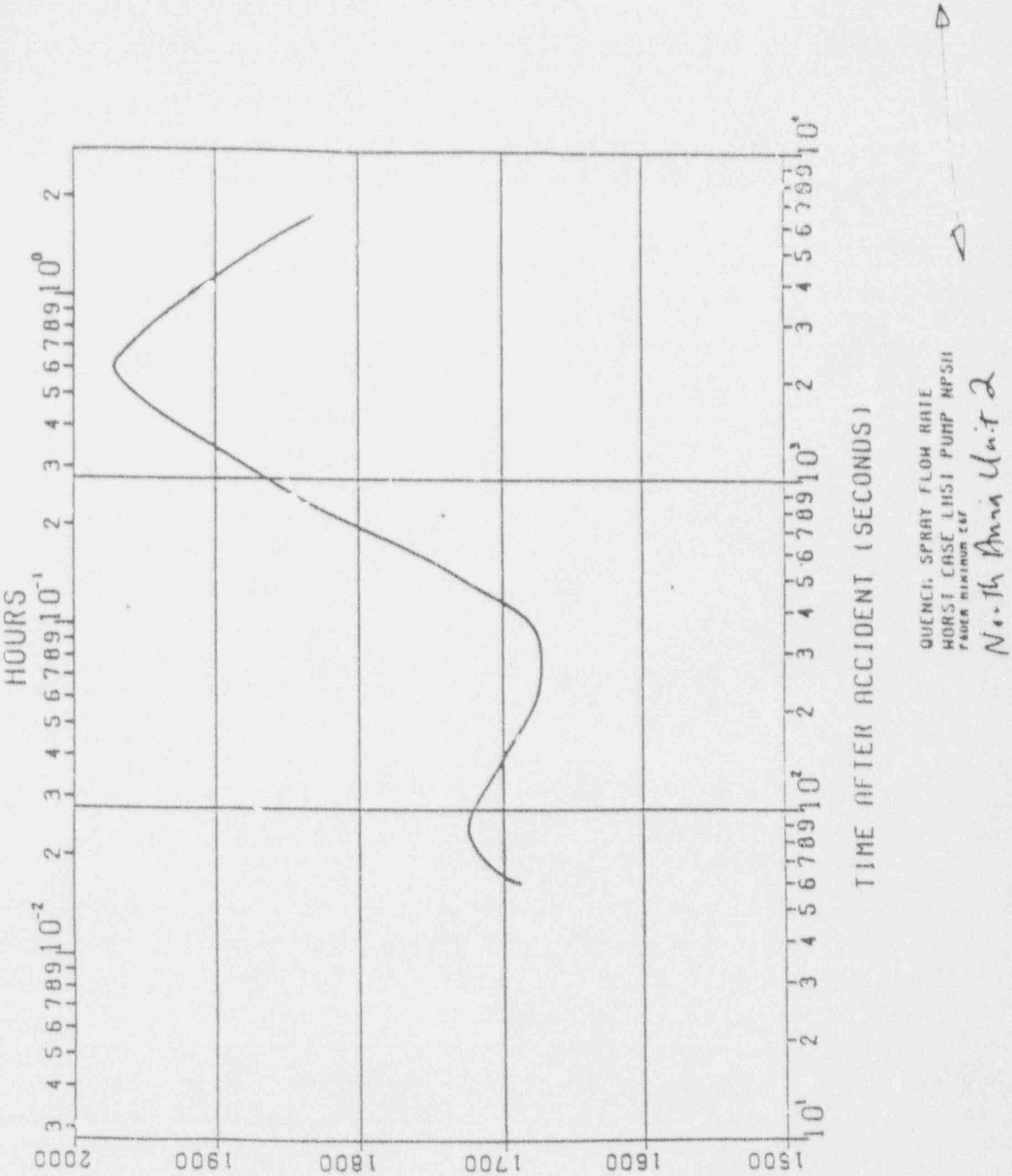


Figure 6.3-16 A
Quench Spray Flow Rate
LHSI Pump NPSH Analysis
PSDER - Minimum ESF
North Anna Unit 1


HOURS
TIME AFTER ACCIDENT (SECONDS)
2000
1900
1800
1700
1600
1500
QUENCH SPRAY FLOW RATE
WORST CASE LHSI PUMP NPSH
North Anna Unit 2

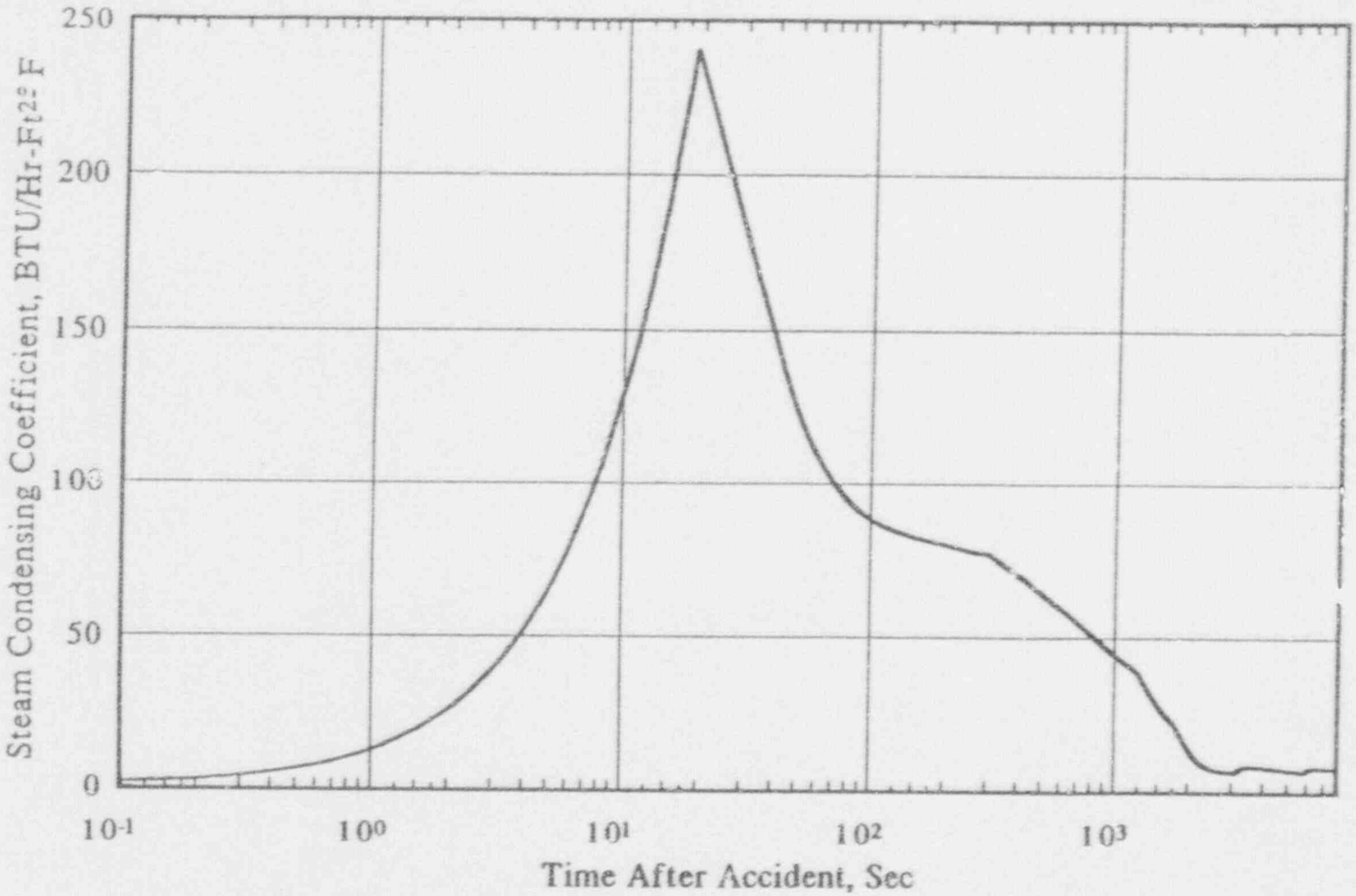
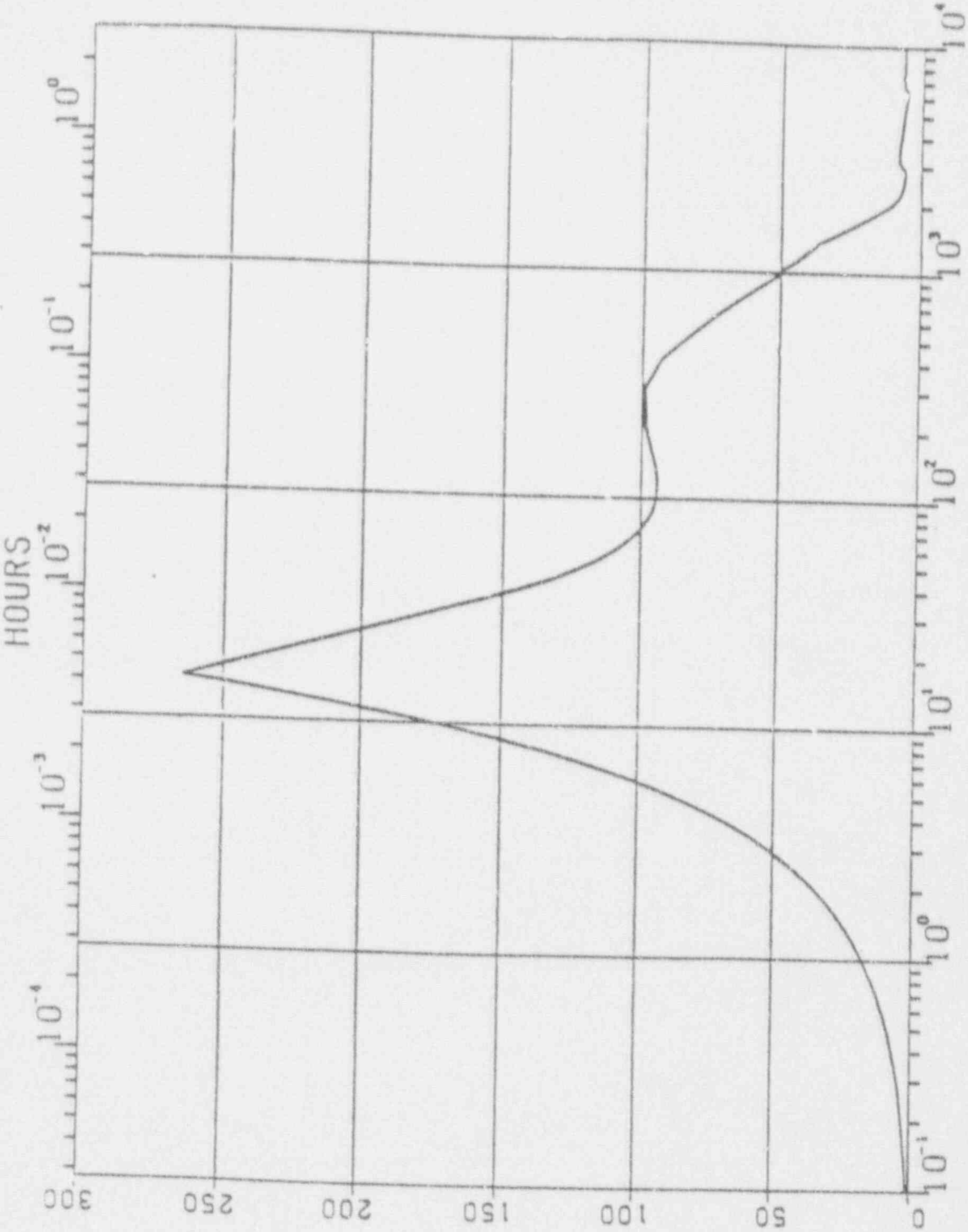


Figure 6.3-17A
Steam Condensing Coefficient
LHSI Pump NPSH Analysis
PSDER - Minimum ESF
North Anna Unit 1



CONDENSING COEFFICIENT
WORST CASE LHSI SUMP NPSH
North Anna Unit 2

5. To withstand the number of pressure and thermal cycles experienced in the life of the plant.

6. To maintain flow restrictor integrity in the event of a double-ended severance of a main steam line immediately downstream of the flow restrictor.

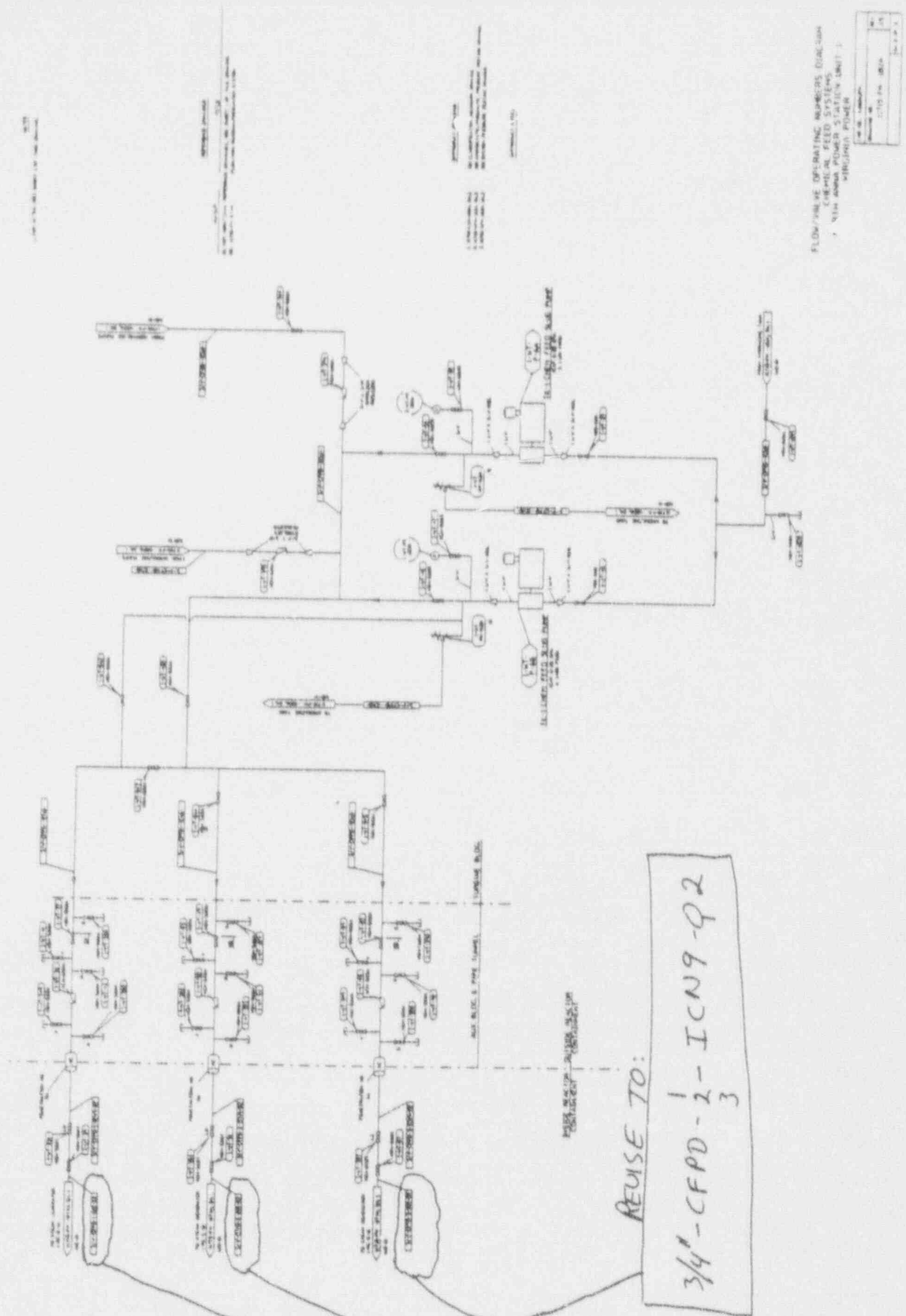
One flow restrictor is located in each main steam line. The restrictor is located as close as possible to the steam-generator outlet nozzle to minimize the length of piping subject to a break that would result in unrestricted flow. Each restrictor is comprised of a minimum equivalent 16-in.-diameter throat venturi nozzle section and a carbon steel discharge cone. The restrictor is permanently welded inside a length of steam piping by a circumferential weld at the discharge end of the restrictor.

In the event of a main steam line break, steam-flow rate is restricted in the throat of the venturi nozzle by a critical flow phenomenon wherein the maximum flow is limited to a value corresponding to sonic velocity at the throat. This limits the rate of blowdown of steam from the steam generators. ~~Information on the steam-generator blowdown system is contained in Section 10.4.8.~~

The swing-disk-type trip valves in series with the nonreturn valves contain swinging disks that are normally held up out of the main steam flow path by air cylinder operators. If a steam line pipe rupture occurs, as discussed in Section 15.4.2, downstream of the trip valves, an excess flow signal from the steam flow meter, combined with low T average or low steam line pressure in two out of three matrices, will release the air pressure on the air cylinders and spring action will cause these valves to trip closed, thus stopping the flow of steam through the steam lines. Valve closure checks the sudden and large release of energy that is in the form of main steam, thereby preventing rapid cooling of the reactor coolant system and ensuing reactivity insertion. Trip valve closure also ensures a supply of steam to the turbine drive of the auxiliary steam-generator feed pump described in Section 10.4.3.

Additionally, a steam flow limiting device having a flow area of 1.4 ft^2 is installed in the main steam outlet nozzle of each Steam Generator dome (Unit 1 steam generators only).

Figure 10.4-7
NAPS UFSAR FIGURE
REVISION 12, 12/91



NOTE:
NEW FEEDWATER LOOP SEAL PIPING IS CLASS 601C

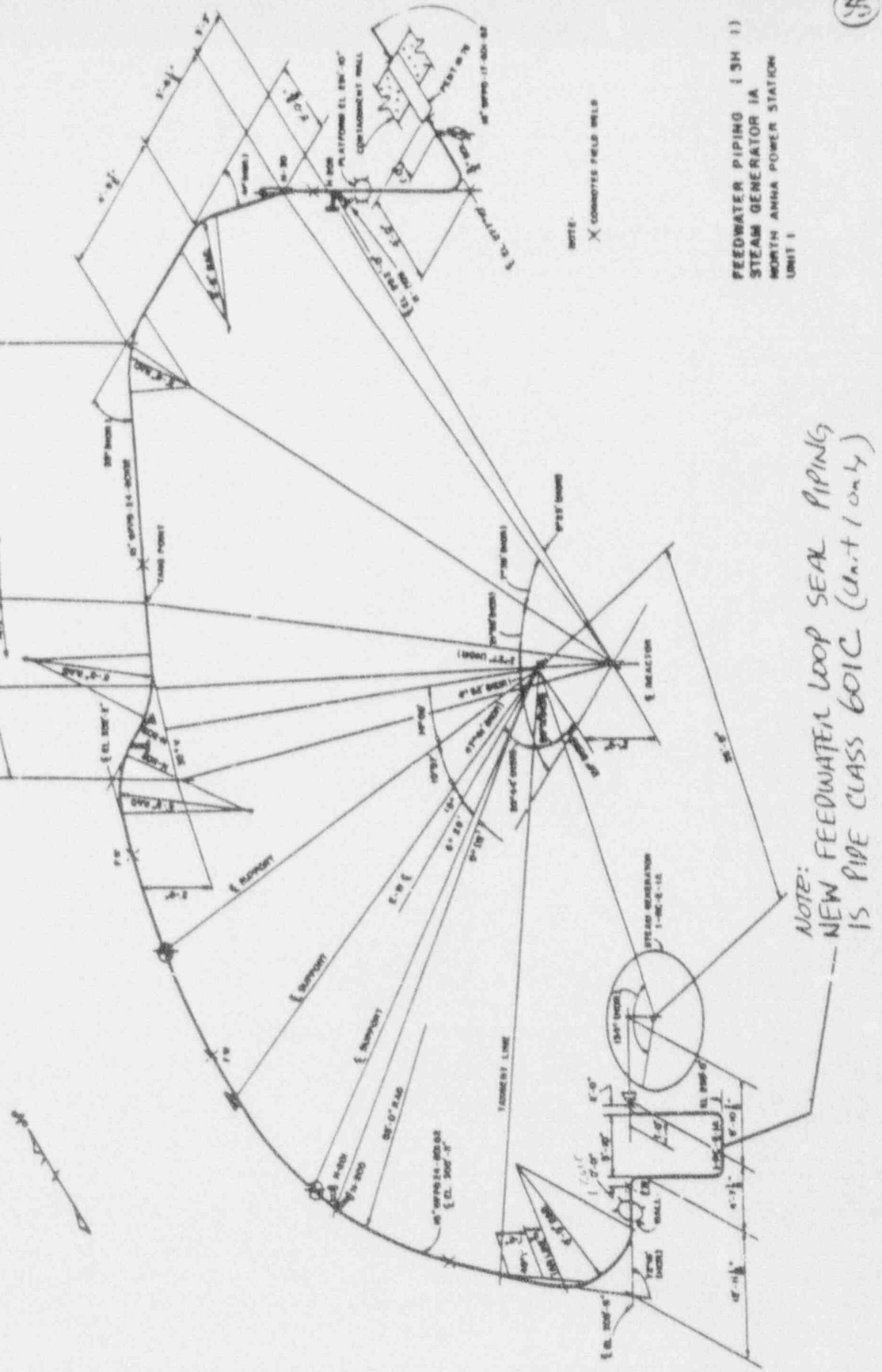
34

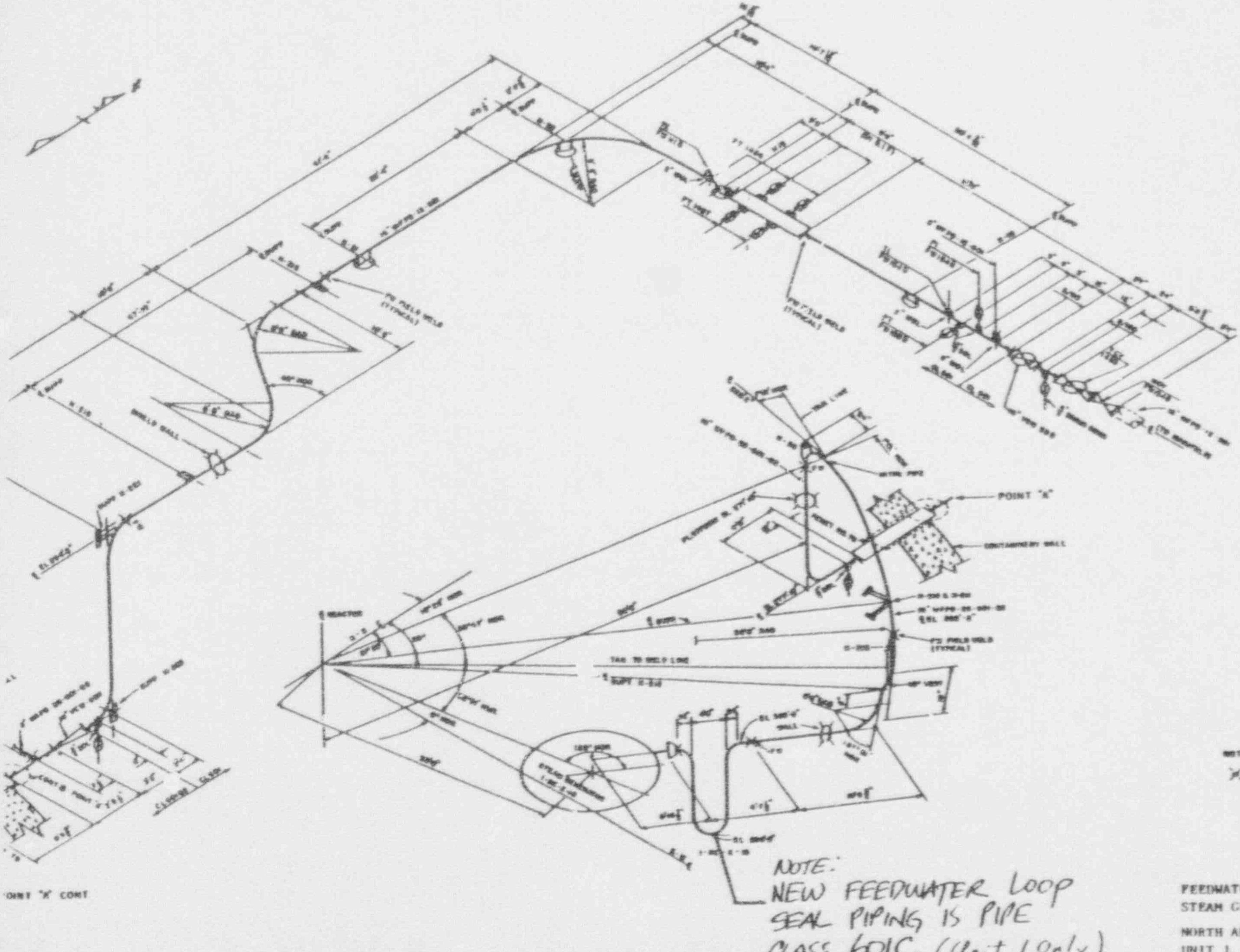
THIS CHEM FEED LINE IS NO LONGER CLASS 601 PIPING.
THE INSTALLED CHEM FEED LINE IS 3/4" TUBING (CLASS ICN9).

NAPS UFSAR FIGURE 10.4-8
SHEET 1

FSR

Figure 10.4-11



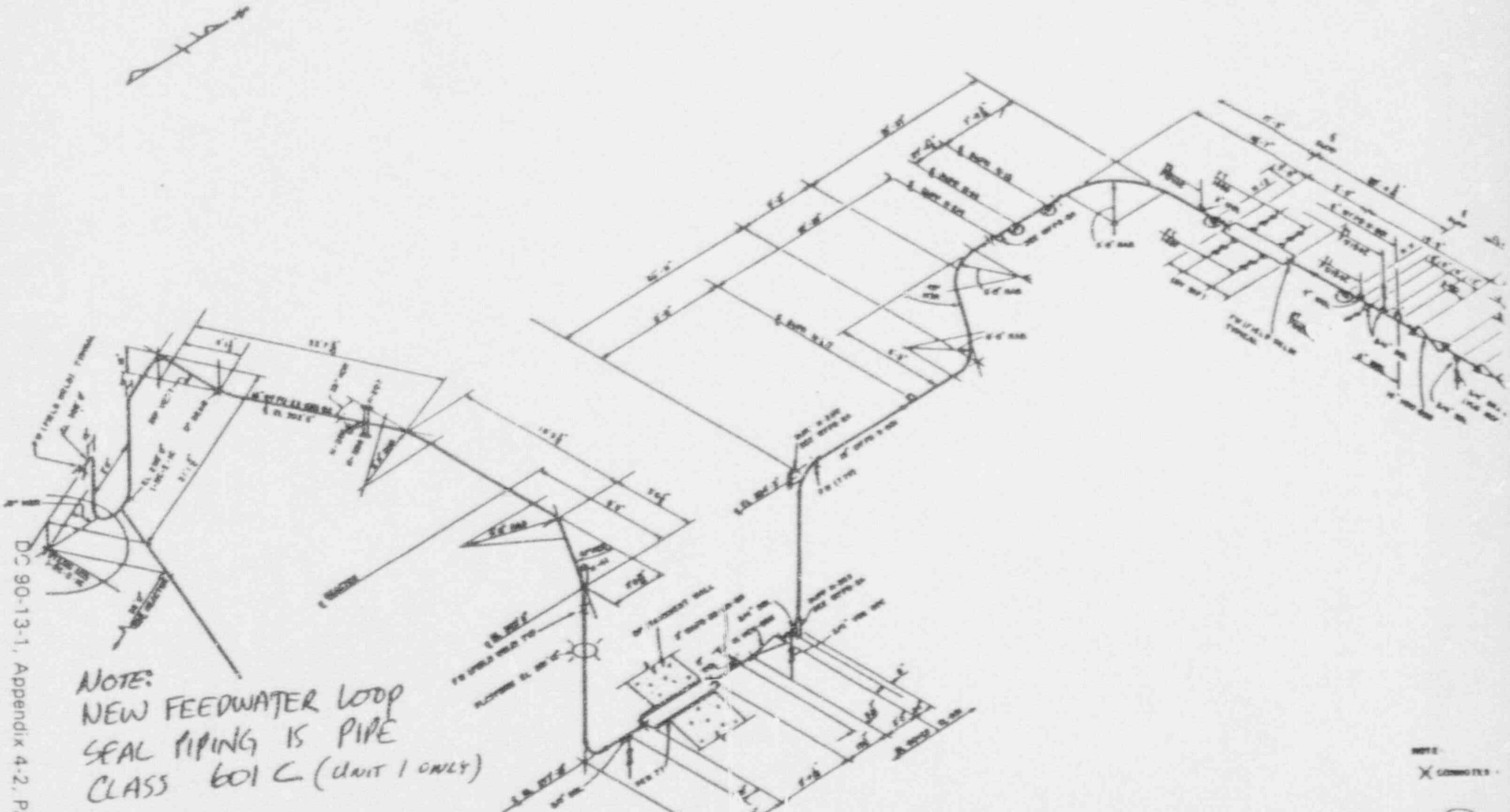


NOTE:
X CONNOTES FIELD WELD

FEEDWATER PIPING
STEAM GENERATOR 1B

NORTH ANNA POWER STATION
UNIT 1

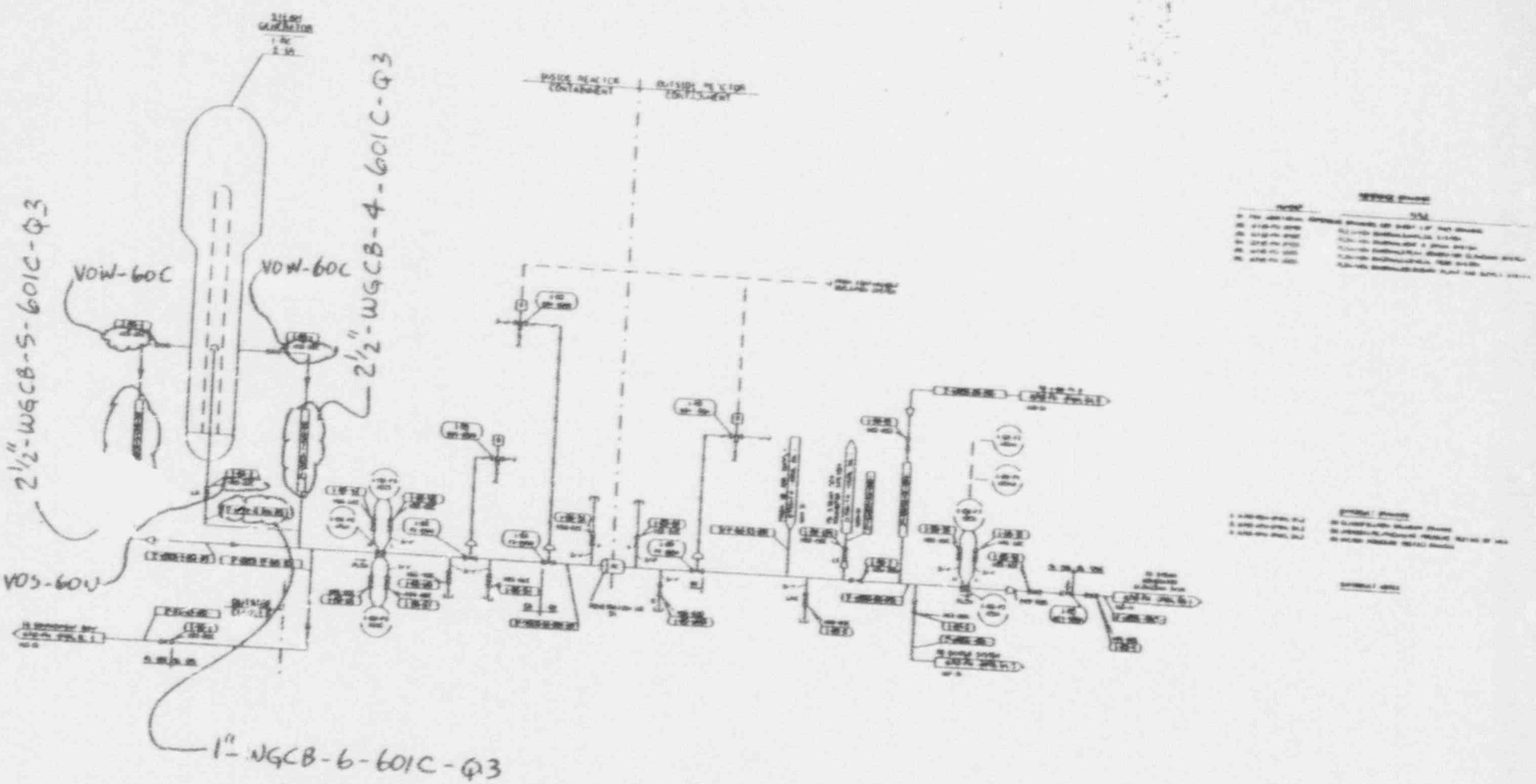
NAPS UFSAR
Figure 10.4-14



(37)

Feedwater Piping
Steam Generator 1C
N.A. Unit 1

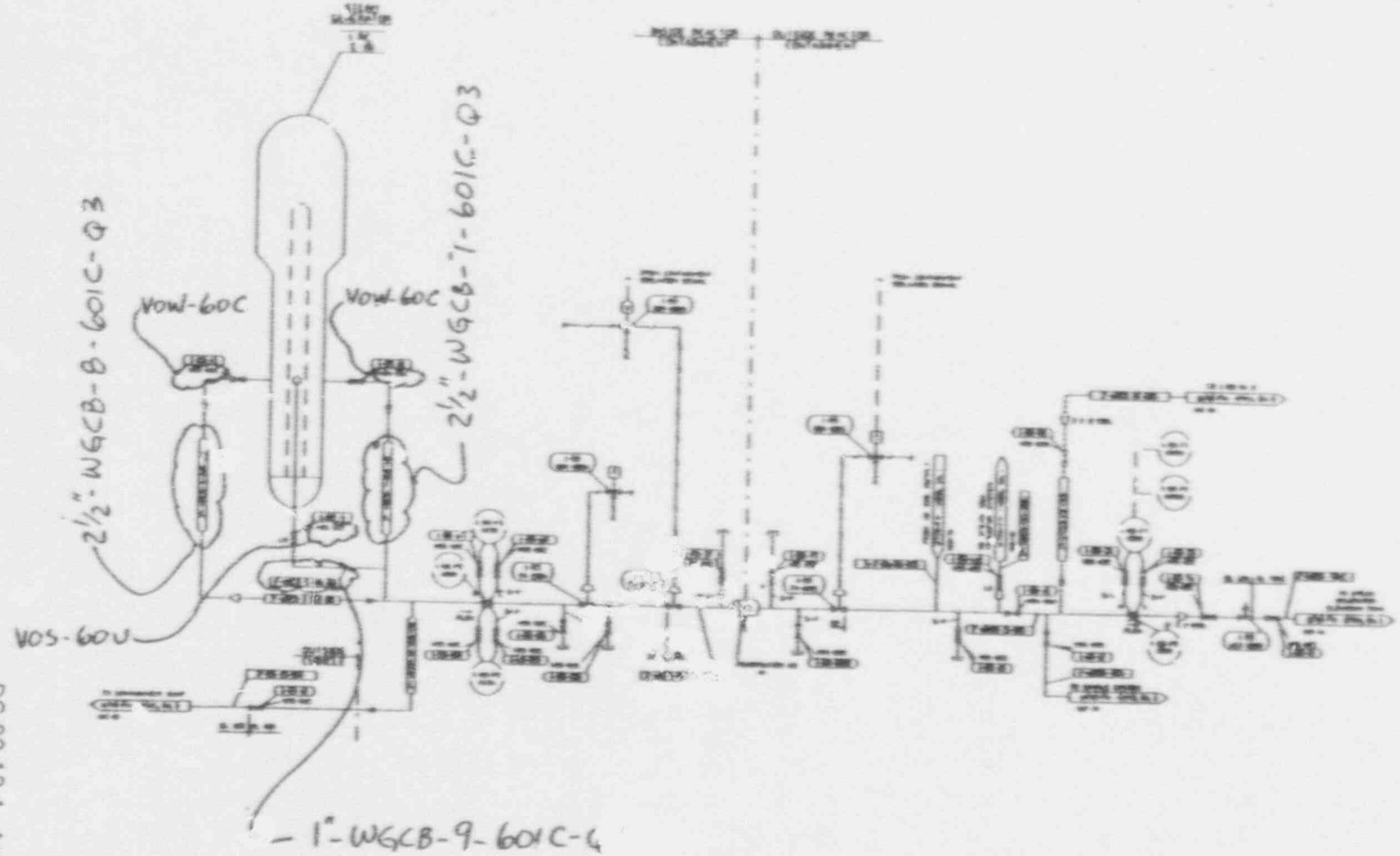
FEEDWATER P
STEAM GENER
NORTH ANNA
UNIT 1



NAPS UFSAR FIGURE 10.4-17 (SHEET 2)

FLOW/VALVE OPERATING NUMBERS DIAGRAM
STEAM GENERATOR BLOWDOWN SYSTEM
N TH ANNA POWER STATION UNIT 1
VIRGINIA POWER

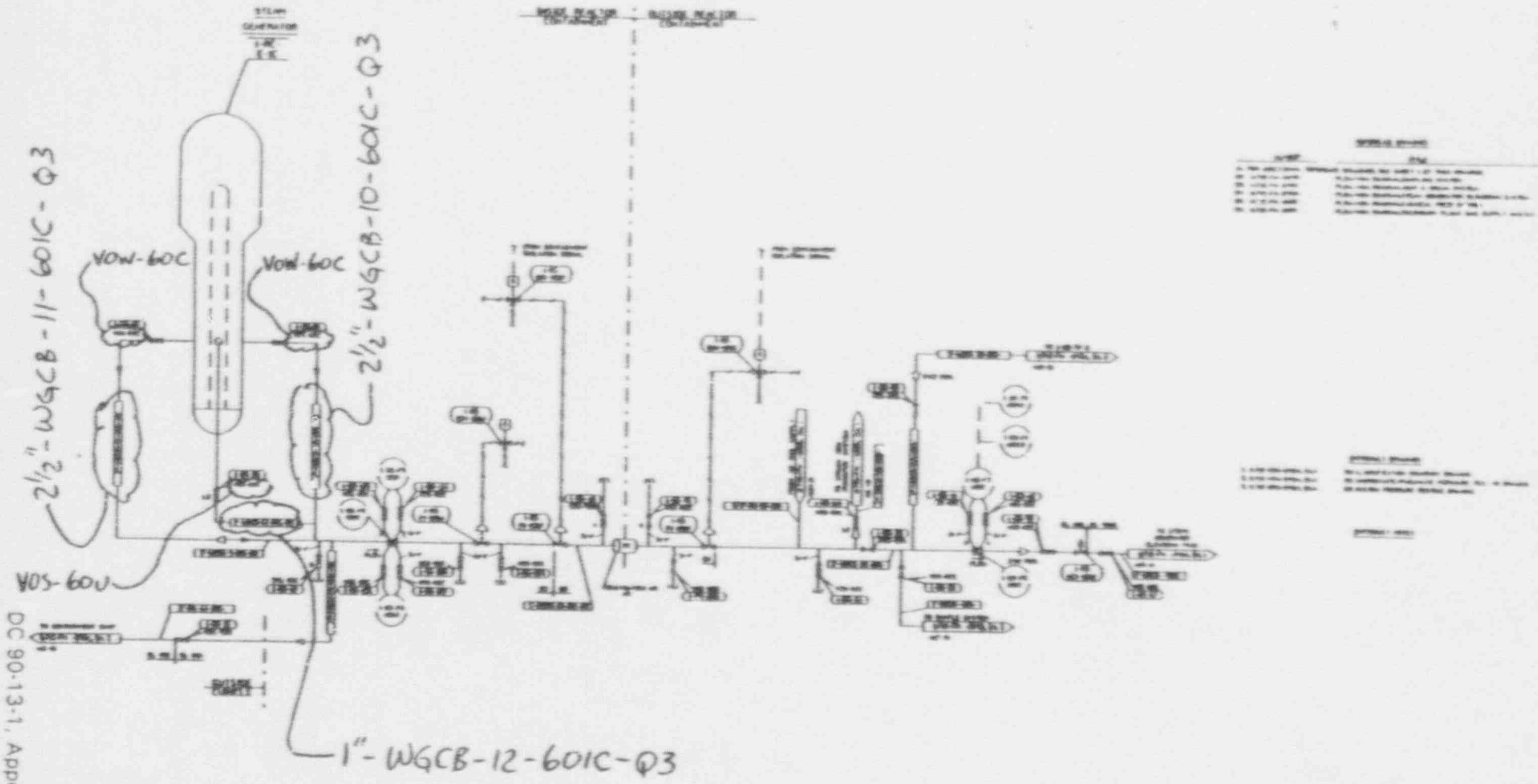
35



NAPS UFSAR FIGURE 10.4-17 (SHEET 3)

FLOW/VALVE OPERATING NUMBERS DIAGRAM
STEAM GENERATOR BLOWDOWN SYSTEM
NORTH ANNA POWER STATION UNIT 1
VIRGINIA POWER

39



NAPS UFSAR FIGURE 10.4-17
(SHEET 4)

FLOW/VALVE OPERATING NUMBERS DIAGRAM
STEAM GENERATOR BLOWDOWN SYSTEM
NORTH ANNA POWER STATION UNIT 1
VIRGINIA POWER

(4)

THE FOLLOWING PAGES
REFLECT ALL REQUIRED
CHANGES TO CHAPTER 15
OF THE NAPS UFSAR.

THESE PAGES COME FROM
APPENDIX C OF THE
NE TECH REPORT, NE-883, Rev. 1
"STEAM GENERATOR REPLACEMENT SAFETY ANALYSIS."

UFSAR updates have been prepared to support North Anna 1 operation following steam generator replacement. These updates supplement those which were previously provided for the extended SGTP analysis effort under UFSAR Change Number FN-92-14. The following sections of the UFSAR require an update:

1. Complete Loss of Flow (UFSAR Section 15.3.4)

2. Loss of External Electrical Load (UFSAR Section 15.2.7)

3. Loss of Normal Feedwater (UFSAR Section 15.2.8)

4. Rod Bank Withdrawal at Power (UFSAR Section 15.2.2)

5. Large Break Loss of Coolant Accident (UFSAR Section 15.4.1):
   Replace entire large break LOCA Section 15.4.1 and associated figures with attached Section 15.4.1 and figures. (Attached LBLOCA writeup will replace two separate unit-specific LBLOCA analysis descriptions in Sections 15.4.1.1 and 15.4.1.2; see SN-92-14.)

6. Containment ~~Analysis (UFSAR Section 6.2 and new Appendix 6E).~~

Note that the UFSAR Sections for 1 through 4 above will be revised when low flow changes have been submitted and approved for North Anna Unit 2.

Rod Bank Withdrawal at Power (UFSAR Section 15.2.2)

The following paragraph was suggested for inclusion in the UFSAR to support N1C9 operation with extended SGTP. (See UFSAR Change Number SN-92-14.)

INSERT I

A reanalysis of the rod withdrawal at power event was performed to support North Anna 1 Cycle 9 (N1C9) restart operation with average steam generator tube plugging plugging (SGTP) levels as high as 40%. The analysis demonstrated acceptable DNBR results for the rod withdrawal at power event under conditions of extended SGTP. All analysis inputs were conservative with respect to allowable N1C9 restart operating conditions.

The following sentence should be appended to the above paragraph:

The extended SGTP rod withdrawal at power reanalysis will continue to be the licensing basis analysis supporting North Anna Unit 1 operation following Unit 1 steam generator replacement.

reactivity feedback. It can be seen that two reactor trip channels provide protection over the whole range of reactivity insertion rates. These are the high-neutron-flux and overtemperature delta T trip channels. The minimum DNBR was determined as a function of reactivity insertion rate using the COBRA[2] code. The limiting case for DNB margin is a reactivity insertion rate of 0.80 x 10^{-5} delta k/k/sec. The minimum DNBR is never less than the limit value. These results demonstrate that the conclusions presented are still valid. That is, the core and reactor coolant system are not adversely affected since the nuclear flux and overtemperature delta T trips prevent the core minimum DNB ratio from falling below the limit value.

Figure 15.2-8a demonstrates the effect of initial RCS average temperature on minimum DNBR as a function of reactivity insertion rate. The minimum feedback, full power cases are presented. The figure demonstrates that the protection system provides DNB protection for the range of operating temperatures considered.

Figures 15.2-9 and 15.2-10 show the minimum DNBR as a function of reactivity insertion rate for rod cluster control assembly withdrawal incidents starting at 60 and 10% power, respectively. The results are similar to the 100% power case, except that as the initial power is decreased, the range over which the overtemperature delta T trip is effective is increased. In neither case does the DNBR fall below the limit value.

INSERT I

15.2.2.3 Uncontrolled Rod Withdrawal Transient for Two-Loop Operation With and Without the Loop Stop Valves Closed

The preceding analyses were for normal three-loop operation. In addition, an analysis of an uncontrolled rod withdrawal transient for two-loop operation with loop stop valves open and closed was performed with the following initial conditions using the LOFTRAN[4] code and the W-3 correlation.

1. Initial Power Levels (based on nominal NSSS power of 2893 MWT)

   a. 65 and 10% of nominal three-loop power for two-loop operation with loop stop valves closed.

Loss of External Electrical Load (UFSAR Section 15.2.7)

The following paragraph was suggested for inclusion in the UFSAR to support N1C9 operation with extended SGTP. (See UFSAR Change Number SN-92-14.)

INSERT G

A reanalysis of the loss of external electrical load event was performed to support North Anna 1 Cycle 9 (N1C9) restart operation with average steam generator tube plugging plugging (SGTP) levels as high as 40%. The analysis demonstrated acceptable DNBR and overpressurization analysis results for the loss of load event under conditions of extended SGTP. All analysis inputs were conservative with respect to allowable N1C9 restart operating conditions.

The following sentence should be appended to the above paragraph:

The extended SGTP loss of load reanalysis will continue to be the licensing basis analysis supporting North Anna Unit 1 operation following Unit 1 steam generator replacement.

Figures 15.2-29 and 15.2-30 show the response for the total loss of load at end of life, assuming a large (absolute value) negative moderator temperature coefficient. All other plant parameters are the same as the case above. The reactor power decreases due to the negative reactivity effect of the increase in core water temperature. No trip setpoint is reached. The pressure initially increases, and then decreases after about 10 sec. as a result of the reduction in neutron flux. The DNBR increases throughout the transient and never drops below its initial value.

The pressurizer safety valves are not actuated in the transients shown in Figures 15.2-29 and 15.2-30; however, the pressurizer safety valves are actuated for the beginning-of-life case.

The total-loss-of-load accident was also studied, assuming the plant to be initially operating at 100% of full power with no credit taken for the pressurizer spray, pressurizer power-operated relief valves, or steam dump. The reactor is tripped on the high pressurizer pressure signal. Figures 15.2-31 and 15.2-32 show the beginning-of-life transient with a positive moderator coefficient. The neutron flux increases to 104% of full power until the reactor is tripped. The minimum DNBR remains well above the limit value. In this case the pressurizer safety valves are actuated.

Figures 15.2-33 and 15.2-34 show the transient at the end of life with the other assumptions the same as in Figures 15.2-31 and 15.2-32. Again, the DNBR increases throughout the transient and the pressurizer safety valves are actuated.

Table 15.2-1 presents the sequence of events for the Loss of External Electrical Load Transient.

Section 15.2.8 presents additional results of analysis for a complete loss of heat sink including loss of main feedwater. This report shows the overpressure protection that is afforded by the pressurizer and steam-generator safety valves.

⇒ INSERT G

Loss of Normal Feedwater (UFSAR Section 15.2.8)

The following paragraph was suggested for inclusion in the UFSAR to support N1C9 operation with extended SGTP. (See UFSAR Change Number SN-92-14.)

INSERT H

A reanalysis of the loss of normal feedwater event was performed to support North Anna 1 Cycle 9 (N1C9) restart operation with average steam generator tube plugging plugging (SGTP) levels as high as 40%. Two loss of normal feedwater cases (with and without offsite power) were considered. The analysis demonstrated in both cases that the auxiliary feedwater system will remove the stored and residual heat, thus preventing overpressurization and relief of RCS inventory through the pressurizer power operated relief valves. All analysis inputs were conservative with respect to allowable N1C9 restart operating conditions.

The following sentence should be appended to the above paragraph:

The extended SGTP loss normal feedwater reanalysis will continue to be the licensing basis analysis supporting North Anna Unit 1 operation following Unit 1 steam generator replacement.

The capacity of the auxiliary feedwater pumps is such that the water level in the steam generators being supplied does not recede below the lowest level at which sufficient heat transfer area is available to dissipate core residual heat without water relief from the reactor coolant system relief or safety valves.

⇒ INSERT H

15.2.8.4 Conclusions

Results of the analysis show that a loss of normal feedwater does not adversely affect the core, the reactor coolant system, or the steam system, since the auxiliary feedwater capacity is such that the reactor coolant water is not relieved from the pressurizer relief or safety valves.

On loss of power to the reactor coolant pumps, coolant flow necessary for core cooling and the removal of residual heat is maintained by natural circulation in the reactor coolant loops.

15.2.9.2 Analysis of Effects and Consequences

15.2.9.2.1 Method of Analysis

A detailed analysis using the RETRAN[11] code is done to obtain the plant transient following a station blackout. The simulation describes the plant thermal kinetics, the reactor coolant system including natural circulation, pressurizer, steam generators, and feedwater system. The digital program computes pertinent variables, including the steam-generator level, pressurizer water level, and reactor coolant average temperature.

The first few seconds of the transient will closely resemble a simulation of the complete loss-of-flow incident (see Section 15.3.4), i.e., core damage due to rapidly increasing core temperatures is prevented by promptly tripping the reactor. After the reactor trip, stored and residual heat must be removed to prevent damage to either the reactor coolant system or the core. The assumptions used in the analyses are similar to the loss of normal feedwater flow incident, except that power is assumed to be lost to the reactor coolant pumps at the time of reactor trip and a conservative total auxiliary feedwater flow of 340 gpm has been used

15.2.9.2.2 Results

Figure 15.2-35b shows plant parameters following a station blackout.

The RETRAN results show that natural circulation flow available is sufficient to provide adequate core decay heat removal following a reactor trip and RCP coastdown.

INSERT H

15.2.9.3 Conclusions

Analysis of the natural circulation capability of the RCS has demonstrated that sufficient heat removal capability exists following reactor coolant pump coastdown to prevent fuel or clad damage. The DNBR is maintained above the limit value. The Reactor Coolant System is not overpressurized and no water relief will occur through the pressurizer relief or safety valves. Thus there will be no cladding damage and no release of fission products to the Reactor Coolant System.

Complete Loss of Flow (UFSAR Section 15.3.4)

The following paragraph was suggested for inclusion in the UFSAR to support N1C9 operation with extended SGTP. (See UFSAR Change Number SN-92-14.)

INSERT F

A reanalysis of the complete loss of flow event was performed to support North Anna 1 Cycle 9 (N1C9) restart operation with average steam generator tube plugging plugging (SGTP) levels as high as 40%. The analysis demonstrated acceptable DNBR results for the complete loss of flow event under conditions of extended SGTP. All analysis inputs were conservative with respect to allowable N1C9 restart operating conditions.

The following sentence should be appended to the above paragraph:

The extended SGTP Complete Loss of Flow reanalysis will continue to be the licensing basis analysis supporting North Anna Unit 1 operation following Unit 1 steam generator replacement.

For Case 1, this transient was analyzed by two digital computer codes using the Statistical DNB methodology. First, the RETRAN[12] code was used to calculate the loop and core flow during the transient, the time of reactor trip, and the nuclear power and heat flux transients following reactor trip. The COBRA[13] code was then used to calculate the minimum DNBR during the transient based upon the heat flux and flow from RETRAN. The WRB-1 correlation was used.

In the original analysis for cases 2 and 3, this transient was analyzed by four digital-computer codes. First, the PHOENIX code was used to calculate the loop and core flow during the transient. The LOFTRAN[10] code was then used to calculate the time of reactor trip and the nuclear power transient following reactor trip. The FACTRAN[11] code was then used to calculate the heat flux transient based on the nuclear power from LOFTRAN and flow from PHOENIX. Finally, the THINC cod was used to calculate the minimum DNBR during the transient based on the heat flux from FACTRAN and flow from PHOENIX. The R-grid spacer factor was applied to the W-3 correlation. For all cases, the transients presented represent the minimum of the typical or thimble cell.

This method of analysis and the assumptions made regarding initial operating conditions and reactivity coefficients are identical to those discussed in Section 15.2.5, except that, following the loss of supply to all pumps at power, a reactor trip is actuated by either bus undervoltage or bus underfrequency.

15.3.4.2.2 Results

The calculated sequence of events is shown in Table 15.3-3 for the three cases analyzed. Figures 15.3-21 through 15.3-29 show the loop and reactor vessel flow coastdowns, the nuclear power changes, the average and hot-channel heat flux changes and the DNBR changes for each of the three cases. The bus-underfrequency reactor trip yielded the minimum DNBR results for the first case. The reactor is assumed to trip on thw undervoltage signal. The DNBR curve for each of the cases is not less than the limit value.

⇒ INSERT F

15.4 CONDITION IV - LIMITING FAULTS

15.4.1 LOSS OF REACTOR COOLANT FROM RUPTURED PIPES OR FROM CRACKS IN LARGE PIPES INCLUDING DOUBLE ENDED RUPTURE THAT ACTUATES THE EMERGENCY CORE COOLING SYSTEM (LARGE BREAK LOSS OF COOLANT ACCIDENT)

This discussion presents the results of the 15% steam generator tube plugging a reanalysis of the large break LOCA transient for North Anna Power Station. Analysis assumptions have been made which reflect operation with 15% steam generator tube plugging (SGTP) in addition to changes in other key analysis inputs. The input changes and the analysis are described in the following sections.

15.4.1.1 GENERAL

A reanalysis of the emergency core cooling system (ECCS) performance for the postulated large break loss of coolant accident (LOCA) has been performed in compliance with Appendix K to 10 CFR 50 (Reference 1). The results of this reanalysis are presented here, and are in compliance with 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Reactors." This analysis was performed with the NRC-approved version of the Westinghouse LOCA-ECCS evaluation model denoted as the 1981 model with BART (References 2, and 3). The analytical techniques are in full compliance with 10 CFR 50, Appendix K.

As required by Appendix K to 10 CFR 50, certain conservative assumptions were made for the LOCA-ECCS analysis. The assumptions pertain to the conditions of the reactor and associated safety system equipment at the time that the LOCA is assumed to occur, and include such items as the core peaking factors, the containment pressure, and the performance of the emergency core cooling system. The details of the large break LOCA analysis are documented in (Reference 1). Selection of input parameters for Appendix K analyses is made to represent an appropriately conservative configuration of the plant initial conditions. This was accomplished by assuming bounding input values for key parameters such as core power, FΔh, FQ, steam generator tube plugging and RCS flow. In general, the remaining key assumptions included in the current analysis are consistent with previous large break analyses performed by Virginia Power. Additional discussion of these assumptions is provided in Section 15.4.1.4.

15.4.1.2 IDENTIFICATION OF CAUSES AND ACCIDENT DESCRIPTION

A LOCA is the result of a rupture of the reactor coolant system (RCS) piping or of any line connected to the system. The system boundaries considered in the LOCA analysis are defined in Section 3.6 of the UFSAR. Sensitivity studies (Reference 4) have indicated that a double-ended cold-leg guillotine (DECLG) pipe break is limiting. Should a DECLG occur, rapid depressurization of the RCS occurs. The reactor trip signal subsequently occurs when the pressurizer low-pressure trip setpoint is reached. A safety injection system (SIS) signal is actuated when the appropriate setpoint is reached, activating the high-head safety injection pumps. The actuation and subsequent activation of the Emergency Core Cooling System, which occurs with the SIS signal, assumes the most limiting single-failure event. These countermeasures will limit the consequences of the accident in two ways:

1. Reactor trip and borated water injection complement void formation in causing rapid reduction of power to a residual level corresponding to fission product decay heat. No credit is taken in the analysis for the insertion of control rods to shut down the reactor.

2. Injection of borated water provides heat transfer from the core and limits the clad temperature increase.

Before the break occurs, the unit is in an equilibrium condition, i.e., the heat generated in the core is being removed via the secondary system. During blowdown, heat from fission product decay, hot internals and the vessel continue to be transferred to the reactor coolant system. At the beginning of the blowdown phase, the entire reactor coolant system contains subcooled liquid that transfers heat from the core by forced convection with some fully developed nucleate boiling. After the break develops, the time to DNB is calculated, consistent with Appendix K of 10 CFR 50. Thereafter, the core heat transfer is based on local conditions, with transition boiling and forced convection to steam as the major heat transfer mechanisms.

During the refill period, it is assumed that rod-to-rod radiation is the only core heat transfer mechanism. The heat transfer between the reactor coolant system and the secondary system may be in either direction, depending on the relative temperatures. For the case of continued heat addition to the secondary side, secondary-side pressure increases and the main safety valves may actuate to reduce the pressure. Makeup to the secondary side is automatically provided by the auxiliary feedwater system. Coincident with the safety injection signal, normal feedwater flow is stopped by closing the main feedwater control valves and tripping the main feedwater pumps. Emergency feedwater flow is initiated by starting the auxiliary feedwater pumps. The secondary-side flow aids in the reduction of RCS pressure. When the reactor coolant system depressurizes to 600 psia, the accumulators begin to inject borated water into the reactor coolant loops. The conservative assumption is then made that injected accumulator water bypasses the core and goes out through the break until the termination of bypass. This conservatism is again consistent with Appendix K of 10 CFR 50. In addition, the reactor coolant pumps are assumed to be tripped at the initiation of the accident, and effects of pump coastdown are included in the blowdown analysis.

The water injected by the accumulators cools the core, and subsequent operation of the low-head safety injection pumps supplies water for long-term cooling. When the refueling water storage tank (RWST) is nearly empty, the long-term cooling of the core is accomplished by switching to the recirculation mode of core cooling, in which the spilled borated water is drawn from the containment sump by the low-head safety injection pumps and returned to the reactor vessel. The containment spray system and the recirculation spray system operate to return the containment environment to subatmospheric pressure.

15.4.1.3 ANALYSIS ASSUMPTIONS

As required by Appendix K of 10 CFR 50, certain conservative assumptions were made for the Large Break LOCA-ECCS analysis. The assumptions pertain to the condition of the reactor and associated safety system equipment at the time that the LOCA is assumed to occur, and include such items as the core peaking factors, core decay heat and the performance of the Emergency Core Cooling System. Tables 15.4-1 and 15.4-2 present the values assumed for several key parameters in this analysis. Assumptions and initial operating conditions which reflect the requirements of Appendix K to 10 CFR 50 have been used in this analysis. These assumptions include:

* The break is located in the cold leg between the pump discharge and the vessel inlet.

* The safety injection flow spills to containment back pressure in the broken loop. Safety injection occurs only in the intact loops cold legs.

* The accumulator in the broken loop also spills to containment.

* 120 percent of 1971 ANS decay heat is assumed following reactor trip.

* Initial power is 102% of the full core power, to account for the calorimetric uncertainty.

As required by Technical Specification 6.9.1.7, the Core Operating Limits Report (COLR) documents the applicable limit values of key core-related parameters for each reload core. The COLR specifies the appropriate limits which account for all design considerations, particularly large and small break LOCA effects.

The current analysis models a full core of 17 x 17 Standard fuel. Analysis results have shown that for the BART or BASH evaluation models, analyzing a full core of 17 x 17 Vantage 5H would result in a 30°F to 100°F PCT benefit when compared to a full core of 17 x 17 Standard fuel. The only mechanism available to cause a greater calculated PCT for a transition core than for a full core of either fuel design is the possibility of flow redistribution due to a mismatch in fuel assembly hydraulic resistances. Full assembly testing of the Vantage 5H fuel has demonstrated that the 17 x17 Vantage 5H and 17x17 Standard fuel designs are hydraulically equivalent and flow distribution does not occur. It has thus been concluded that the North Anna transition from 17x17 Standard fuel to a mixed core of 17x17 Standard and Vantage 5H requires no transition core PCT penalty (Reference 5).

The large break LOCA analysis for the North Anna Power Station bounds a full core of North Anna Improved Fuel (NAIF), which is similar and compatible to Westinghouse Vantage 5 Hybrid (V5H) fuel. The most significant NAIF design change from the 17x17 standard fuel assembly design is the use of Zircaloy grids to replace the six intermediate LOPAR Inconel grids. The guide thimble and instrumentation tube diameters are reduced to accommodate this change. The NAIF fuel assembly has the same cross-sectional envelope as the LOPAR assembly, however, it is 0.210 inches longer overall. NAIF also includes a slight increase in overall rod length, thimble plug removal, reconstitutable top nozzle, debris filter bottom nozzle and core bypass flow of 6%. For the North Anna transition from 17x17 Standard fuel to a mixed core of Standard and V5H fuel, it is not necessary to apply a LOCA analysis transition core penalty (Reference 5).

Using these assumptions in the BART ECCS evaluation model, it has been demonstrated that operation at the rated thermal power of 2893 MWt with

SGTP up to 15% in any SG will comply with the 2200°F acceptance limit of 10 CFR 50.46.

15.4.1.4 ANALYSIS OF EFFECTS AND CONSEQUENCES

15.4.1.4.1 METHOD OF ANALYSIS

The large break LOCA is divided, for analytical purposes, into three phases: blowdown, refill and reflood. There are three distinct transients analyzed in each phase, including the thermal-hydraulic transient in the reactor coolant system, the pressure and temperature transient within the containment and the fuel clad temperature transient of the hottest fuel rod in the core. Based on these considerations, a system of interrelated computer codes has been developed for the analysis.

The description of the various aspects of the LOCA analysis methodology is given in WCAP-8339 (Reference 6). This document describes the major phenomena modelled, the interfaces among the computer codes and the features of the codes that ensure compliance with 10 CFR 50, Appendix K. The SATAN-VI, COCO, WREFLOOD, BART and LOCTA-IV codes, which are used in the LOCA analysis, are described in detail in WCAP-8306 (Reference 7), WCAP-8326 (Reference 8), WCAP-8171 (Reference 9), WCAP-9695 (Reference 2), WCAP-9561 (Reference 3), and WCAP-8305 (Reference 10), respectively. These codes assess whether sufficient heat transfer geometry and core amenability to cooling are preserved during the time spans applicable to the blowdown, refill and reflood phases of the LOCA. The SATAN-VI computer code analyzes the thermal-hydraulic transient of the reactor coolant system during blowdown, and the COCO computer code calculates the containment pressure transient during all three phases of the LOCA analysis. The thermal-hydraulic response of the reactor coolant system during refill and reflood is calculated by the WREFLOOD code; WREFLOOD also calculates the mass and energy flow rates that are assumed to be vented to the containment. Since the mass flow rate depends upon the core pressure, which is a function of the containment backpressure, the WREFLOOD and COCO codes are interactively linked. With the input and boundary conditions from WREFLOOD, the mechanistic core heat transfer model in BART calculates the fluid and heat transfer conditions in the core during reflood.

SATAN-VI is used to determine the RCS pressure, enthalpy and density, as well as the mass and energy flow rates in the reactor coolant system and steam generator secondary, as a function of time during the blowdown phase of the LOCA. SATAN-VI also calculates the accumulator mass and pressure and the pipe break mass and energy flow rates that are assumed to be vented to the containment during blowdown. At the end of the blowdown, the mass and energy release rates during blowdown are transfered to the COCO code for use in the determination of the containment pressure response during this first phase of the LOCA. Additional SATAN-VI output data from the end of blowdown, including the core inlet flow rate and enthalpy, the core pressure and the core power decay transient are input to the LOCTA code.

With input from the SATAN-VI code at the end of blowdown, WREFLOOD is used to determine the vessel flooding rate, the coolant pressure and temperature and the quench of vessel metal mass during the refill phase of the LOCA (time period from end of blowdown to that time when flow enters the bottom of the core). WREFLOOD is also used to calculate the mass and energy flowrates assumed to be vented to the containment for refill and reflood phases. Since the mass flowrate to the containment depends on core pressure, which is a function of the containment backpressure, the WREFLOOD and COCO codes are interactively linked.

The COCO code, which is used throughout all three phases of the LOCA analysis, calculates the containment pressure. Input to COCO is obtained from the mass and energy flow rates assumed to be vented to the containment, as calculated by the SATAN-VI and WREFLOOD codes. In addition, conservatively chosen initial containment conditions and an assumed mode of operation for the containment cooling system are input to COCO. These initial containment conditions and assumed modes of operation are provided in Table 15.4-2.

LOCTA-IV is used throughout the analysis of the LOCA transient to calculate the fuel and clad temperature of the hottest rod in the core. The input to LOCTA-IV consists of appropriate thermal-hydraulic outputs from SATAN-VI, WREFLOOD, and BART and conservatively selected initial RCS operating conditions. These initial conditions are summarized in Table 15.4-1 and Figure 15.4-1. The axial power shape of Figure 15.4-1 assumed for LOCTA-IV is a cosine curve that has been previously verified to be the shape that produces the maximum peak clad temperature.

15.4.1.4.2 RESULTS

Tables 15.4-1 and 15.4-2 and Figure 15.4-1, present the initial conditions and the modes of operation that were assumed in the analysis.

The results of this analysis are tabulated in Tables 15.4-3 and 15.4-4 for a double ended guillotine break with a CD=0.4 discharge coefficient at Tavg of 576.8°F. Prior Virginia Power and Westinghouse analyses employing the approved large break LOCA evaluation models have demonstrated that limiting PCT is obtained for this case. Results for other typical cases (CD=0.6, CD=0.8) have PCT results which are consistently 150 - 200°F less than that for the CD=0.4 case. The double ended guillotine break has been determined to be the limiting break size and location based on the sensitivity studies reported in Reference 4. The mass and energy release for limiting cases are given in Tables 15.4-5 and 15.4-6. The attached figures show the following:

- Axial Power Shape - Figure 15.4-1 shows the cosine power shape used in this analysis.

- Mass Velocity - Figure 15.4-2 shows the mass velocity at the clad burst and hot-spot locations on the hottest fuel rod.

- Core Pressure - Figure 15.4-3 shows the calculated pressure in the core.

- Accumulator Mass Flow - Figure 15.4-4 shows the calculated accumulator flow. The accumulator delivery during blowdown is discarded until the end of bypass is calculated. Accumulator flow, however, is established in the refill-reflood calculations. The accumulator flow assumed is the sum of that injected in the intact cold legs.

- Core Pressure Drop - Figure 15.4-5 shows the calculated core pressure drop. The core pressure drop is interpreted as the pressure immediately before entering the core inlet to the pressure just outside the core outlet.

- Break Mass Flow - Figure 15.4-6 shows the calculated flowrate out of the break. The flowrate out of the break is plotted as the sum of flow at both the pressure vessel end and the reactor coolant pump end of the guillotine break.

- Core Power - Figure 15.4-7 shows the core power transient calculated by the SATAN-VI code.

- Containment Wall Heat Transfer Coefficient - Figure 15.4-8 shows the containment wall heat transfer coefficient.

- Containment Pressure - Figure 15.4-9 shows the calculated pressure transient. The analysis of this pressure transient is based on the containment data, reflood mass and energy release, and accumulator flow to containment.

- Pumped ECCS Flow (Reflood) - Figure 15.4-10 shows the calculated flow of the emergency core cooling system.

- Core and Downcomer Water Levels - Figure 15.4-11 shows the reactor vessel downcomer and core water levels.

- Core Flow - Figure 15.4-12 shows the calculated core flow, both top and bottom, for the discharge coefficient used.

- Core Inlet Velocity - Figure 15.4-13 shows the core inlet velocity.

- Hot Rod Clad Average Temperature - Figure 15.4-14 shows the calculated hot-spot clad temperature transient and the clad temperature transient at the burst location. The peak clad temperature for the limiting discharge coefficient of 0.4 is 2116.5°F at 8.00 ft elevation in the core.

- Fluid Temperature - Figure 15.4-15 shows the calculated vapor temperature for the hot spot and burst locations.

- Hot Rod Heat Transfer Coefficient - Figure 15.4-16 shows the heat transfer coefficient at the clad burst and hot spot locations on the hottest rod.

- Break Energy - Figure 15.4-17 shows the break energy released to the containment.

15.4.1.5 CONCLUSIONS

This large break LOCA analysis was performed for a double ended rupture of a reactor coolant pipe with CD=0.4, and at rated thermal power of 2893 MWt, assuming the operating conditions specified in Tables 15.4-1 and 15.4-2. Based upon these results, the emergency core cooling system meets the acceptance criteria as presented in 10 CFR 50.46 as follows:

1. The calculated peak fuel rod clad temperature is below the requirement of 2200°F.
2. The amount of fuel element cladding that reacts chemically with water or steam does not exceed 1% of the total amount of Zircaloy in the reactor.
3. The clad temperature transient is terminated at a time when the core is still amenable to cooling. The localized cladding oxidation limits of 17% are not exceeded during or after quenching.
4. The core remains amenable to cooling during and after the break.
5. The core temperature is reduced and the long-term heat is removed for an extended period of time.

15.4.1.6 REFERENCES

1. North Anna Power Station, Safety Evaluation No. 92-SE-OT-042, for Unit 2 Cycle 9 Operation, April 13, 1992.

2. Young, M. Y. et al, "BART-A1: A Computer Code for the Best Estimate Analysis of Reflood Transients", WCAP-9695, January 1980.

3. Young, M. Y., "Addendum to BART-A1: A Computer Code for the Best Estimate Analysis of Reflood Transients," (Special Report: Thimble modeling in Westinghouse ECCS Evaluation Model): WCAP-9561-P, Addendum 3, Revision 1, July, 1986.

4. WCAP-8356, "Westinghouse ECCS Plant Sensitivity Studies," July 1974.

5. Letter from W. L. Stewart (Va. Power) to NRC, "North Anna Power Station Units 1 and 2 - Proposed Technical Specification change, North Anna Fuel Assembly Design Change," Serial No. 89-795, January 15, 1990.

6. WCAP-8339, "Westinghouse ECCS Evaluation Model-Summary," July 1974.

7. WCAP-8306, "SATAN-VI Program: Comprehensive Space-Time Dependent Analysis of Loss-of-Coolant," June, 1974.

8. WCAP-8326, "Containment Pressure Analysis Code (COCO)," June 1974.

9. WCAP-8171, "Calculational Model for Core Reflooding After a Loss-of-Coolant Accident (WREFLOOD Code)," June, 1974.

10. Bordelon, et al, "LOCTA-IV Program: Loss-of-Coolant Transient Analysis," WCAP-8305, June, 1974.

Table 15.4-1

INITIAL CORE CONDITIONS ASSUMED FOR THE
DOUBLE ENDED COLD LEG GUILLOTINE BREAK (DECLG)
15% STEAM GENERATOR TUBE PLUGGING CASE

Calculational Input	
Core Power (MWt), 102% of	2893
Peak Linear Power (Kw/ft), 102% of	12.43
Peak Heat Flux Hot Channel Factor, FQ(z)	2.19
Peak Nuclear Enthalpy Hot Channel Factor, FNdh	1.55
Accumulator Water Volume (cu. ft./accumulator)	1025
Reactor Vessel Upper Head Temperature	(T_{hot})

Limiting Fuel Region and Cycle	Cycle	Region
Unit 2	All	All Regions

Table 15.4-2
15% STEAM GENERATOR TUBE PLUGGING CASE

CONTAINMENT DATA

Net Free Volume (ft^3)	1.916×10^6
Initial Conditions[a]	
Pressure (psia)	9.608
Temperature (°F)	86.0
RWST Temperature (°F)	40.0
Outside Temperature (°F)	-10.0
Spray System[a]	
Number of Pumps Operating	2
Runout Flow Rate (per pump)	2000 gpm
Time in Which Spray is Effective	59 sec
Structural Heat Sinks[a]	
Thickness (in.)	Area (ft^2), with allowance for uncertainties
6 concrete	8,393
12 concrete	62,271
18 concrete	55,365
24 concrete	11,591
27 concrete	9,404
36 concrete	3,636
.375 steel, 54 concrete	22,039
.375 steel, 54 concrete	28,393
.500 steel, 30 concrete	25,673
26.4 concrete, .25 steel, 120concrete	12,110
.407 stainless steel	10,527
.371 steel	160,328
.882 steel	9,894
.059 steel	60,875

a See UFSAR Section 6.3.3.12 for a detailed breakdown of the containment heat sinks and for justification of the other input parameters use to calculate containment pressure.

Table 15.4-3

TIME SEQUENCE OF EVENTS
15% STEAM GENERATOR TUBE PLUGGING CASE

Description of Parameters	DECLG (0.4) Tavg = .8°F (seconds)
End of Bypass/ End of Blowdown (sec	.262
Safety System Actions	
Reactor Trip (Sec)	0.467
Accumulator Injection (Sec)	14.9
SI Signal Generated (Sec)	3.6
Pump SI Starts (Sec)	28.6
Bottom of Core Recovery (sec)	45.87
Accumulator Empty (sec)	56.27

Table 15.4-4

RESULTS FOR DECLG
15% STEAM GENERATOR TUBE PLUGGING CASE

Description of Parameters	DECLG (Cd=0.4) Tavg = 576.8°F (seconds)
Peak Clad Temperature (°F)	2116.5
Peak Clad Location (ft)	8.00
Hot Rod Burst Data	
Location (ft)	6.00
Time (Sec)	44.80
Zr/H_2O Results Data	
Local Maximum Reaction (%)	4.99
Location of Maximum (ft)	8.0
Total Reaction (%)	< 1.0

Table 15.4-5

REFLOOD MASS AND ENERGY RELFASES DECLG (CD=0.4, TAVG=576.8°F)
15% STEAM GENERATOR TUBE PLUGGING CASE

Time (Sec)	Total Mass Flow Rate (lbm/sec)	Total Energy Flow Rate (10^8 Btu/sec)
45.87	0.0	0.0
46.49	0.664	.00859
46.59	0.6619	.00857
46.79	0.6525	.00845
46.895	0.6556	.00849
47.09	0.6463	.00837
.11	84.63	1.0492
72.86	130.13	1.1990
91.81	243.84	1.4556
112.61	262.06	1.4393
135.01	268.15	1.3914
172.84	311.61	1.4229

Table 15.4-6

BROKEN LOOP ACCUMULATOR FLOW TO CONTAINMENT DECLG

(CD=0.4,TAVG=576.8)

15% STEAM GENERATOR TUBE PLUGGING CASE

TIME (sec)	MASS FLOW RATE (lb/sec)
0.0	4096.76
1.01	3693.44
3.01	3158.17
5.01	2805.15
7.01	2546.97
10.01	2257.28
15.01	1924.58
20.01	1696.31
25.01	1534.08
29.01	1590.03

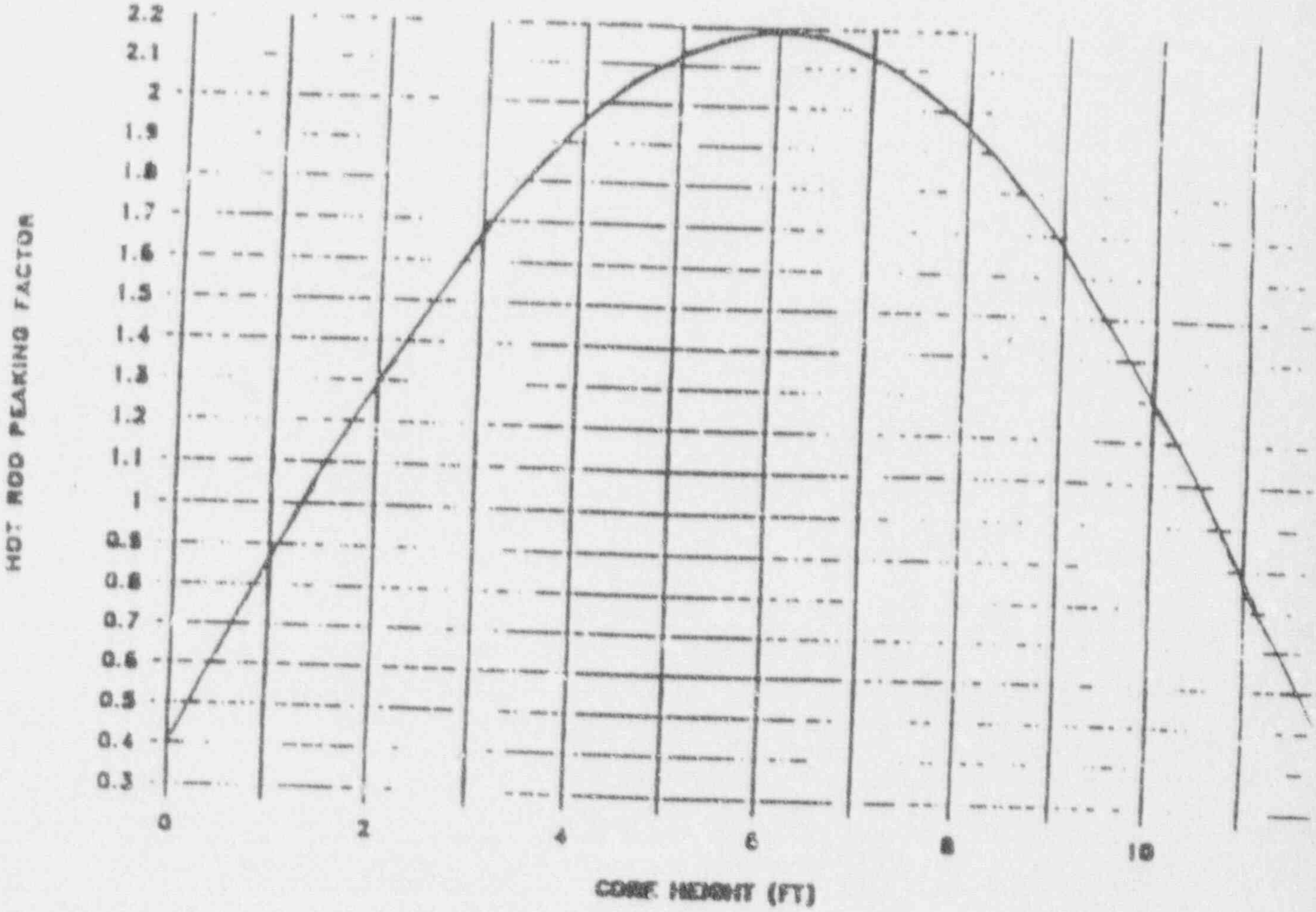


Figure 15.4-1 Axial Power Shape
15% STEAM GENERATOR TUBE PLUGGING CASE

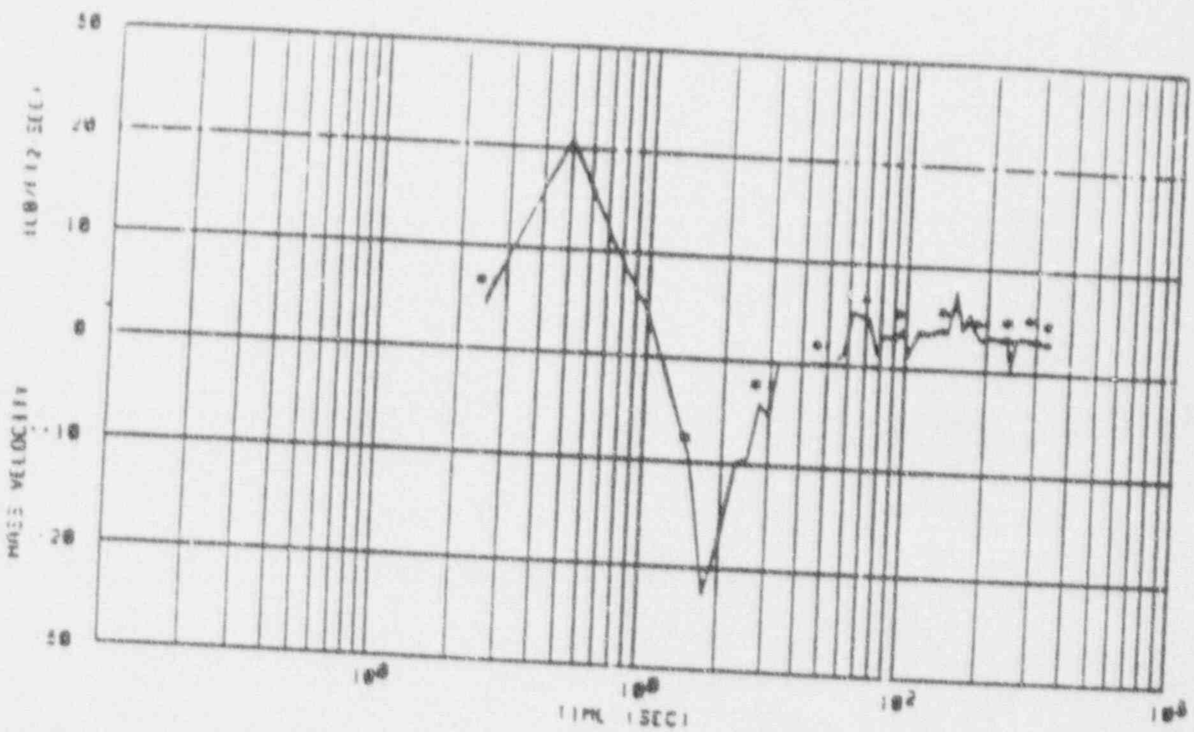
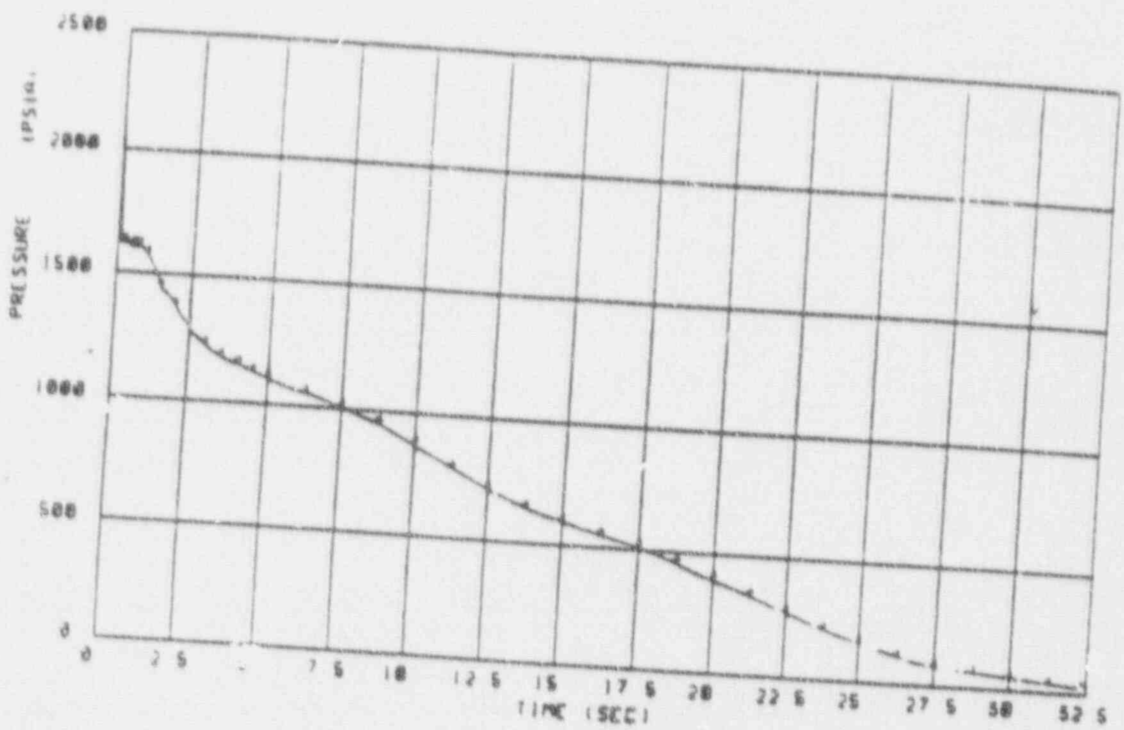
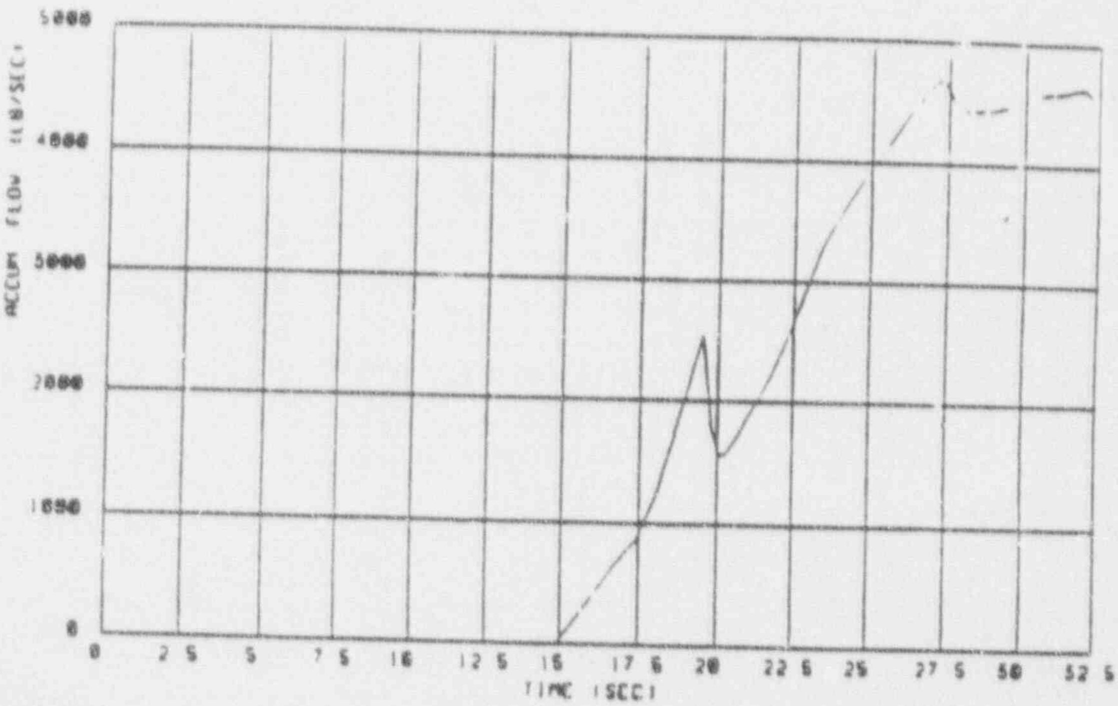


Figure 15.4-2 Mass Velocity (DECLG, CD=0.4)
15% STEAM GENERATOR TUBE PLUGGING CASE



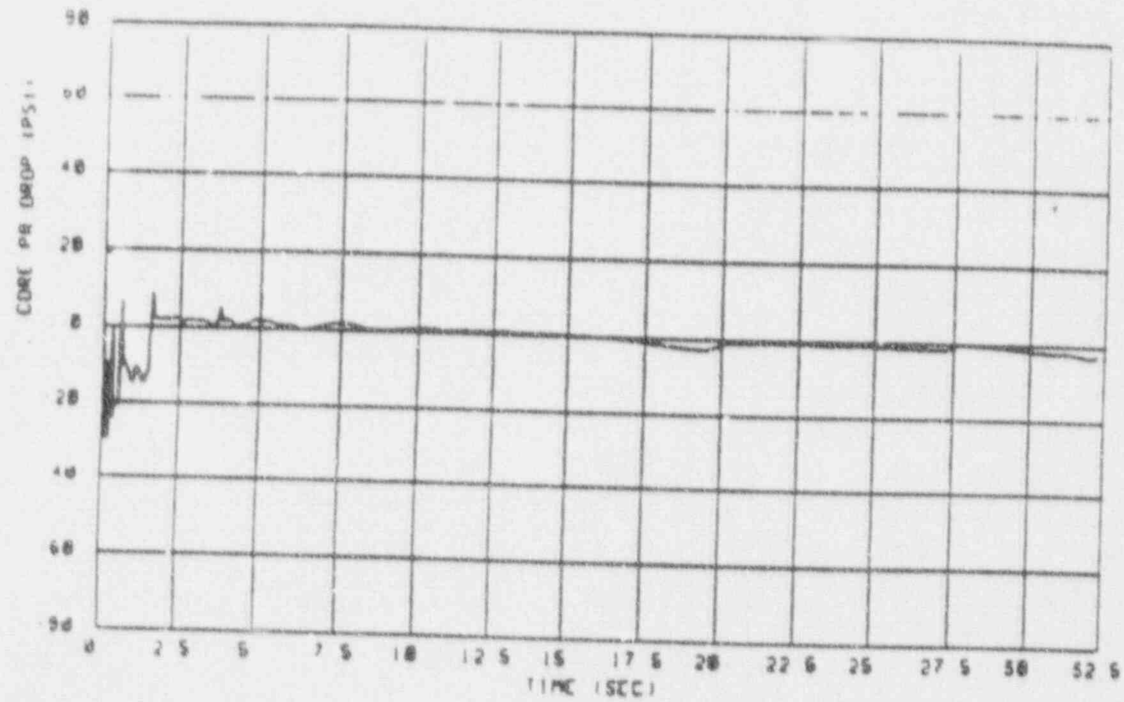
09/18/89

Figure 15.4-3 Core Pressure (DECLG, CD=0.4)
15% STEAM GENERATOR TUBE PLUGGING CASE



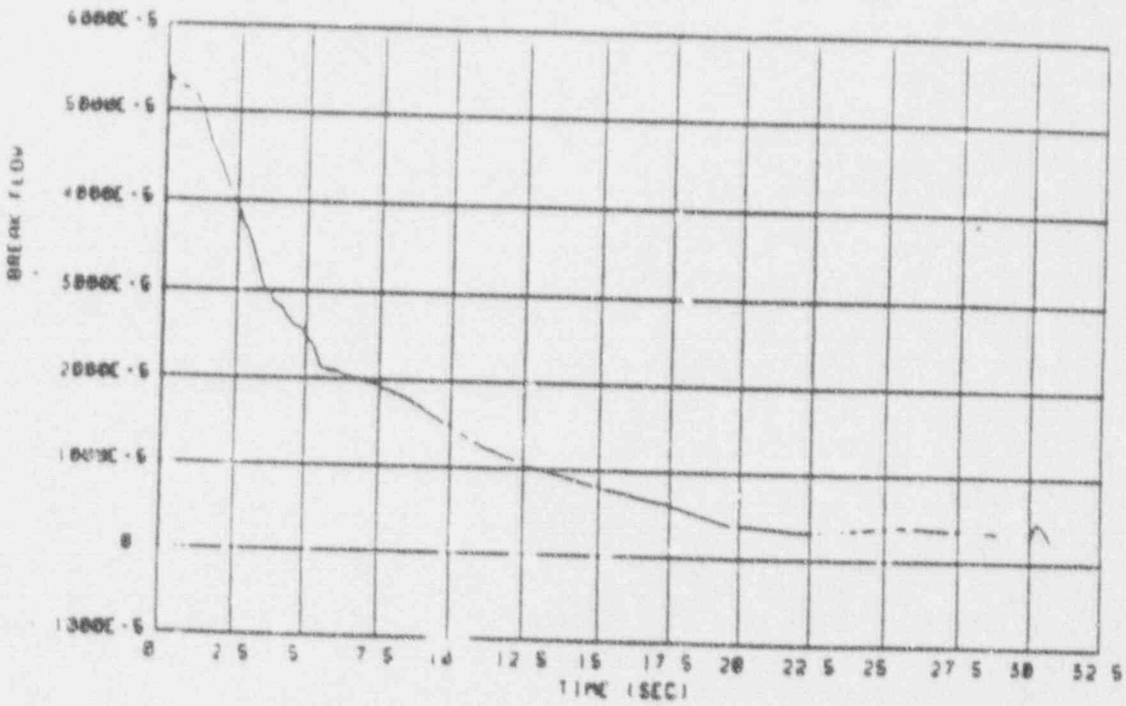
09/18/89

Figure 15.4-4 Accumulator Mass Flow (DECLG, CD=0.4)
15% STEAM GENERATOR TUBE PLUGGING CASE



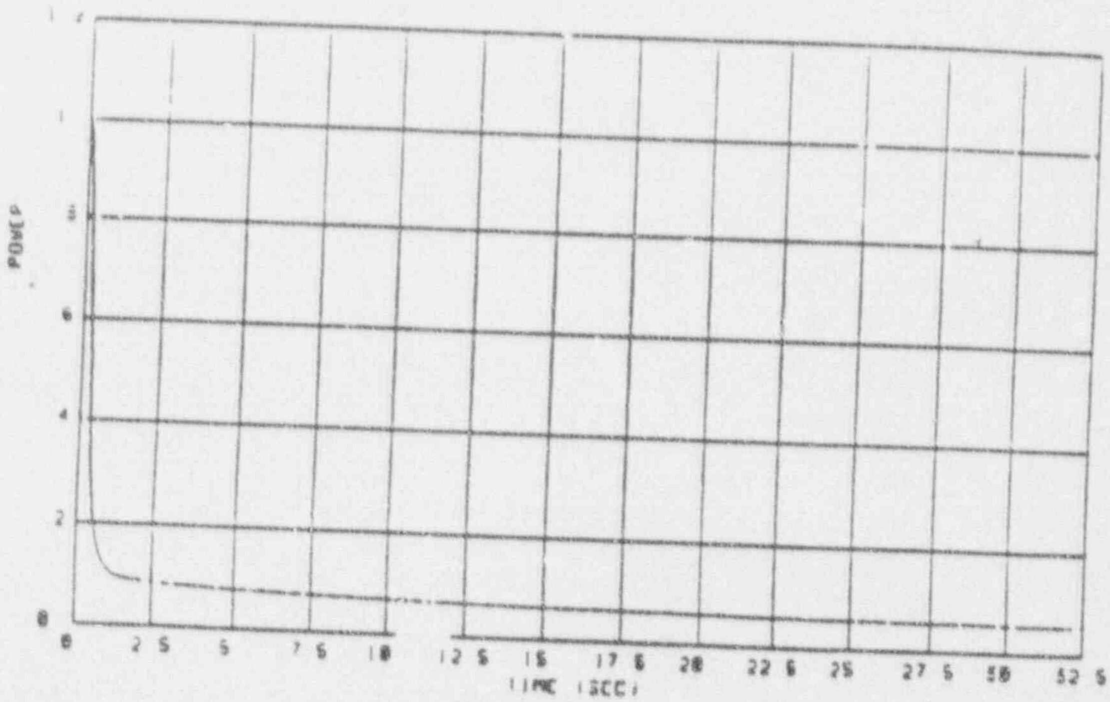
09/18/89

Figure 15.4-5 Core Pressure Drop (DECLG, CD=0.4)
15% STEAM GENERATOR TUBE PLUGGING CASE



09/18/89

Figure 15.4-6 Break Mass Flow (DECLG, CD=0.4)
15% STEAM GENERATOR TUBE PLUGGING CASE



09/18/89

Figure 15.4-7 Core Power (DECLG, CD=0.4)
15% STEAM GENERATOR TUBE PLUGGING CASE

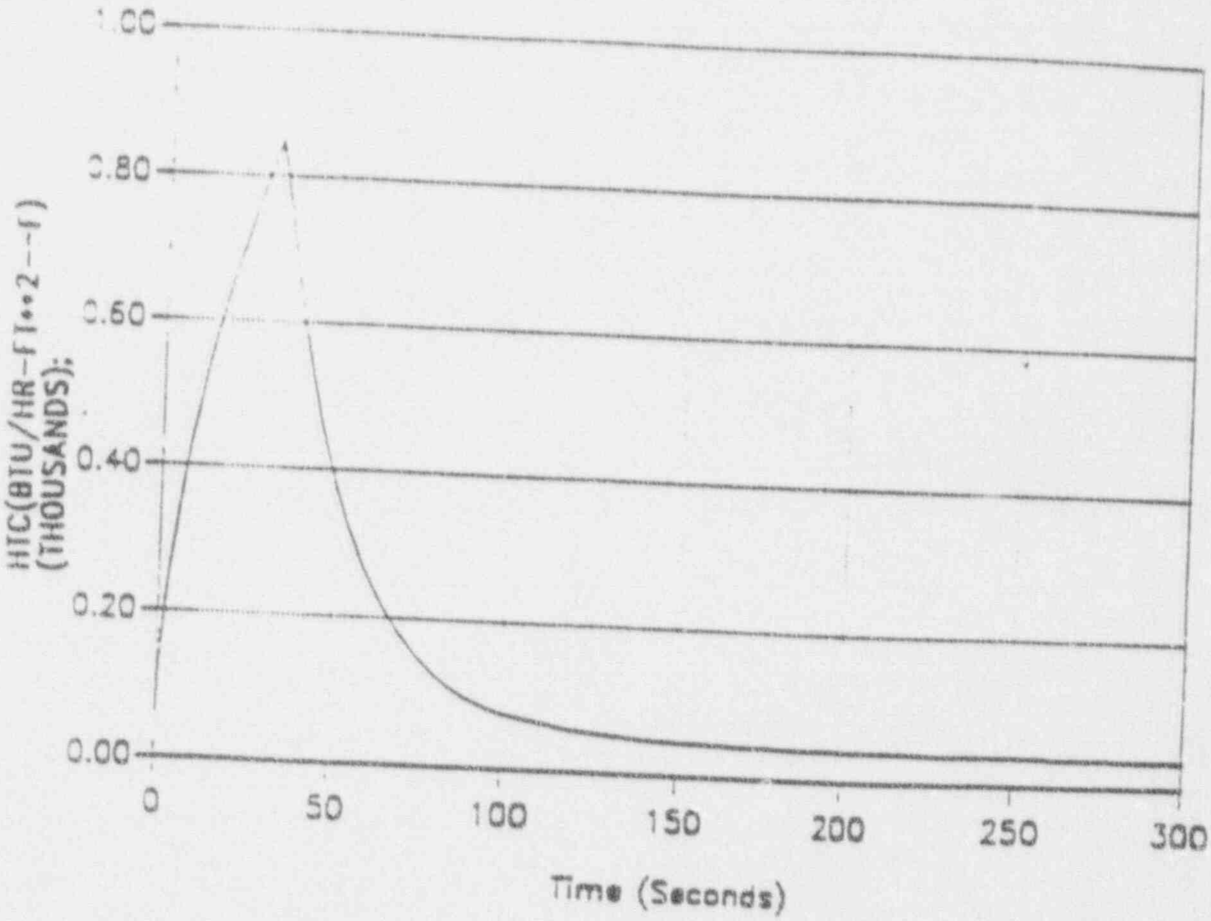


Figure 15.4-8 Containment Wall Heat Transfer Coefficient (DECLG, CD=0.4)
15% STEAM GENERATOR TUBE PLUGGING CASE

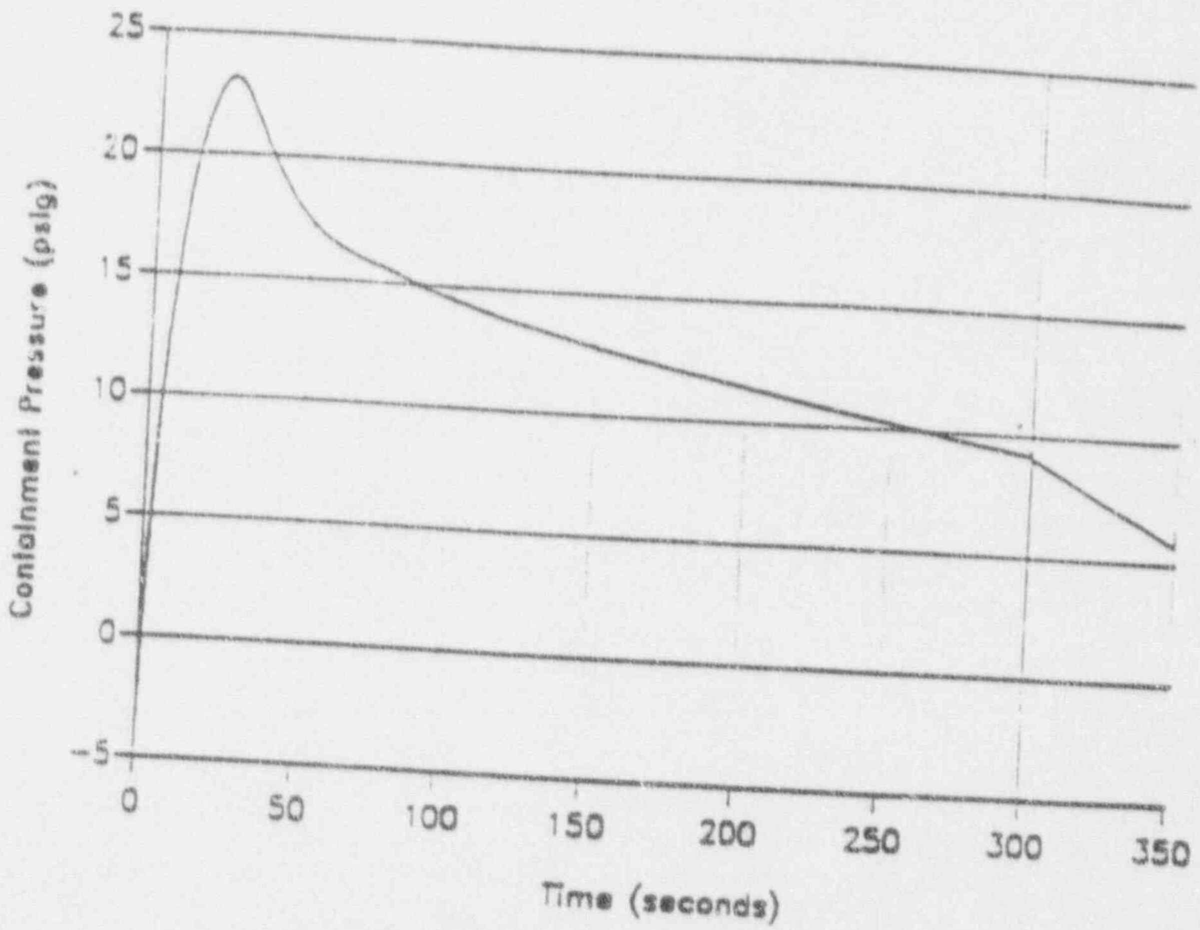


Figure 15.4-9 Containment Pressure (DECLG, CD=0.4)
15% STEAM GENERATOR TUBE PLUGGING CASE

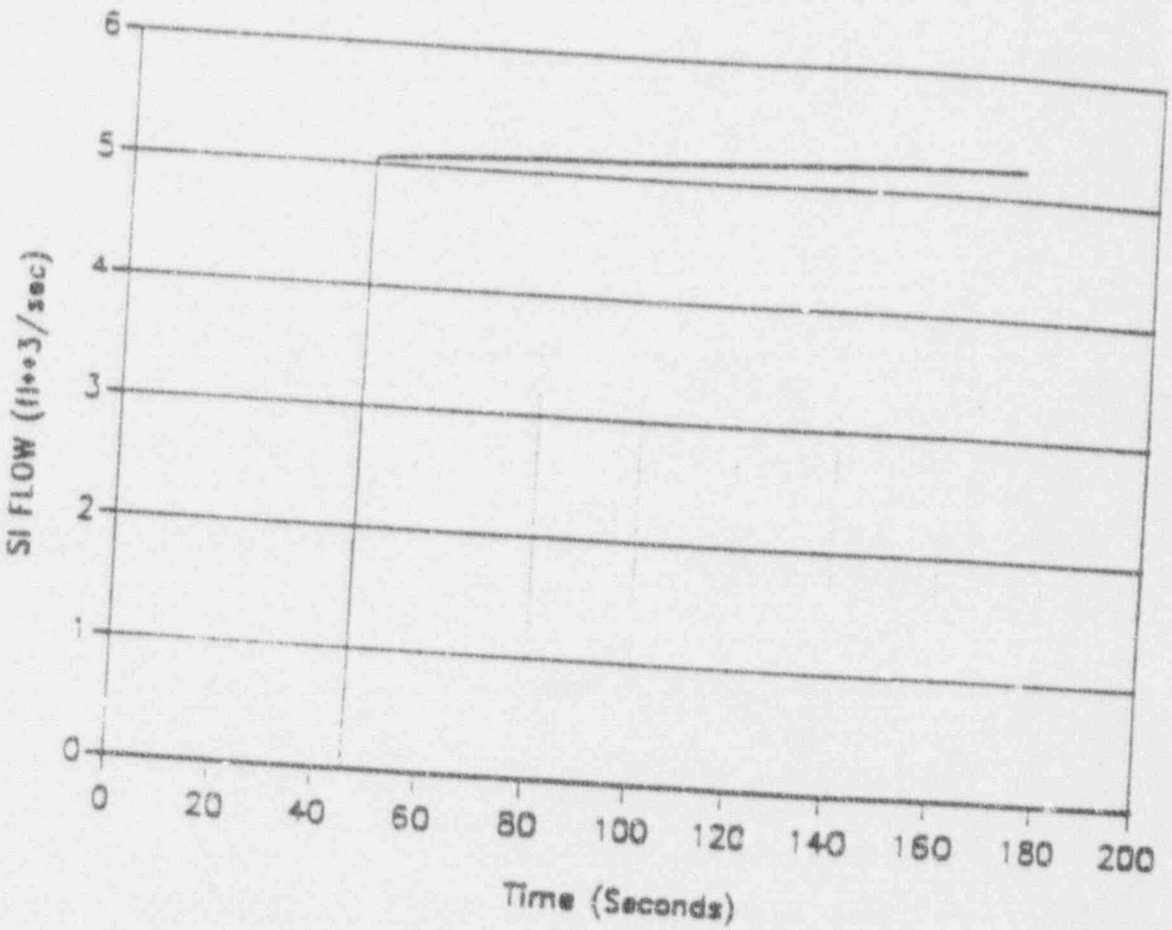
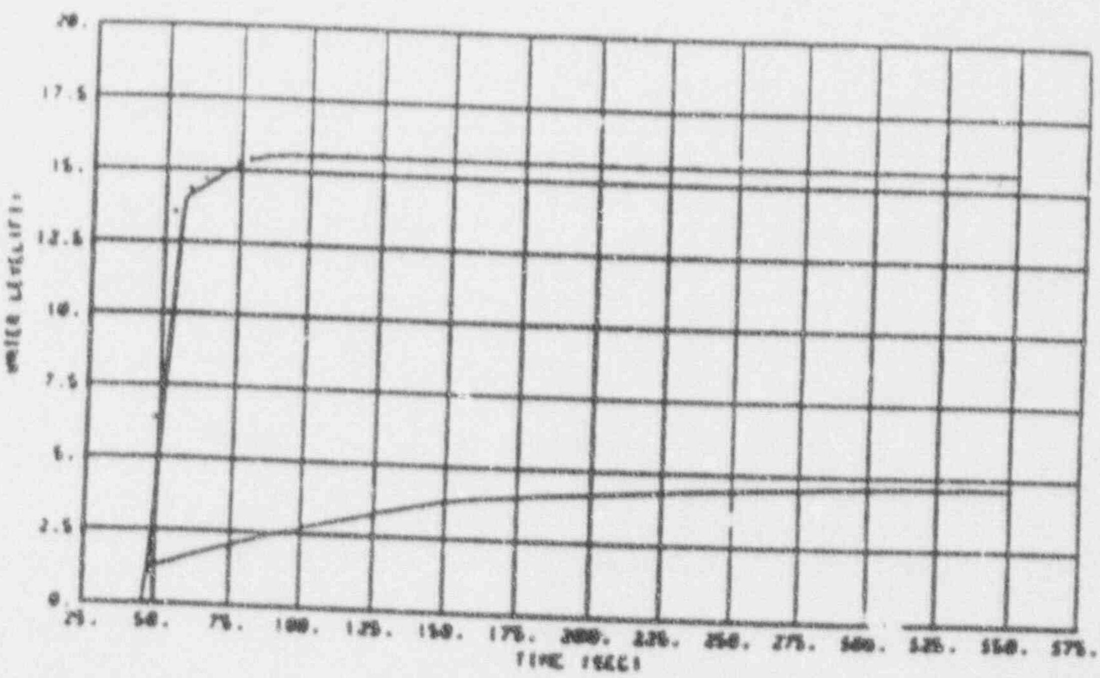
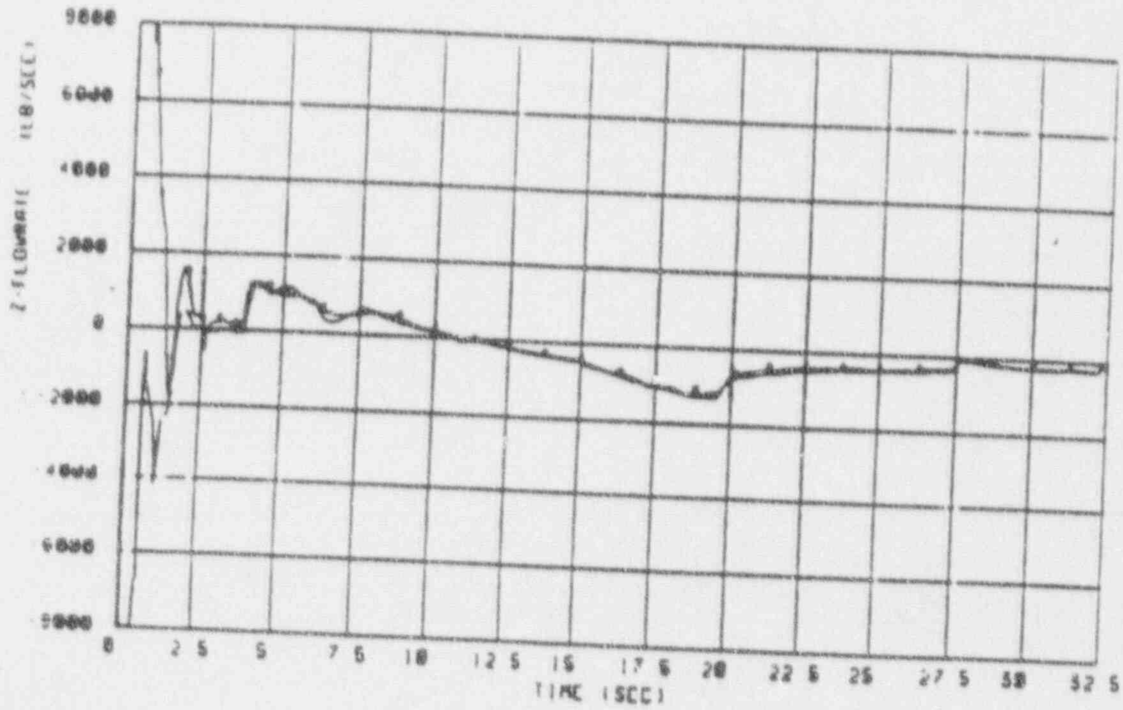


Figure 15.4-10 Pumped ECCS Flow (DECLG, CD=0.4)
15% STEAM GENERATOR TUBE PLUGGING CASE



09/18/89

Figure 15.4-11 Core and Downcomer Water Levels (DECLG, CD=0.4)
15% STEAM GENERATOR TUBE PLUGGING CASE



09/18/89

Figure 15.4-12 Core Flow (DECLG, CD=0.4)
15% STEAM GENERATOR TUBE PLUGGING CASE

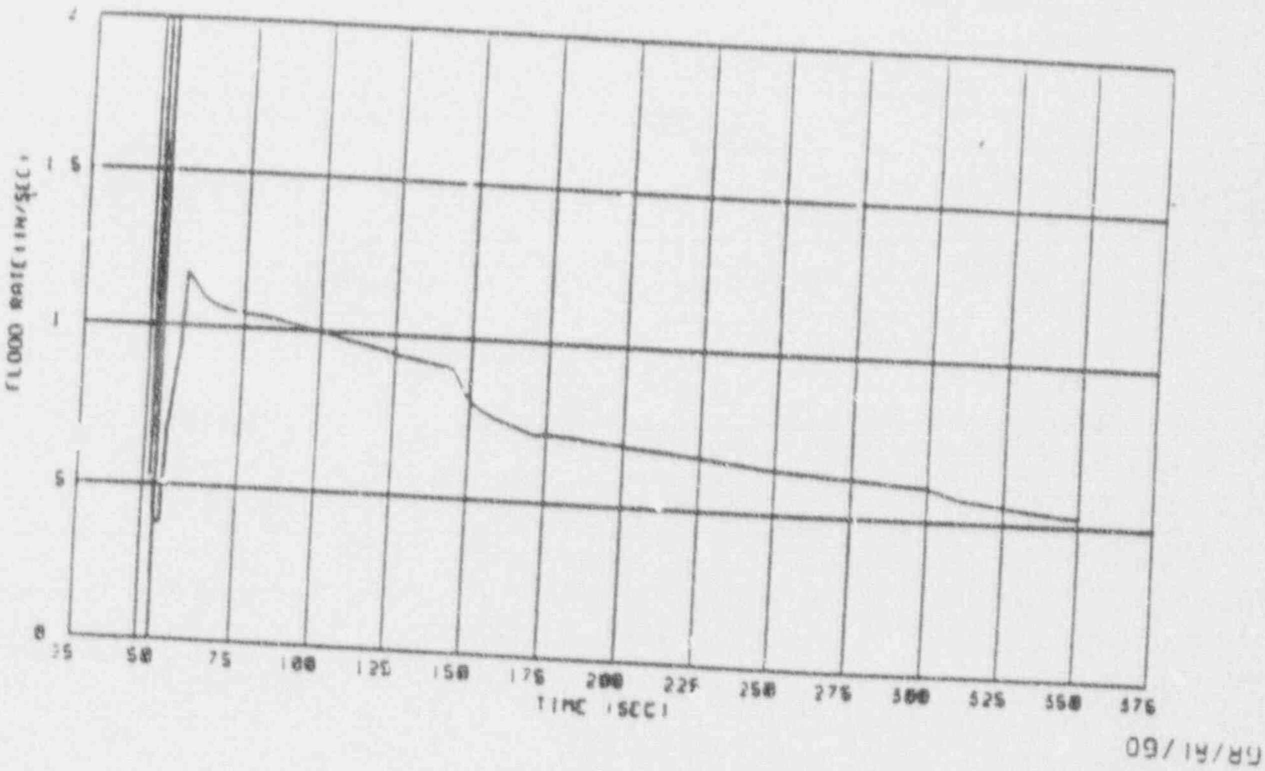


Figure 15.4-13 Core Inlet Velocity (DECLG, CD=0.4)
15% STEAM GENERATOR TUBE PLUGGING CASE

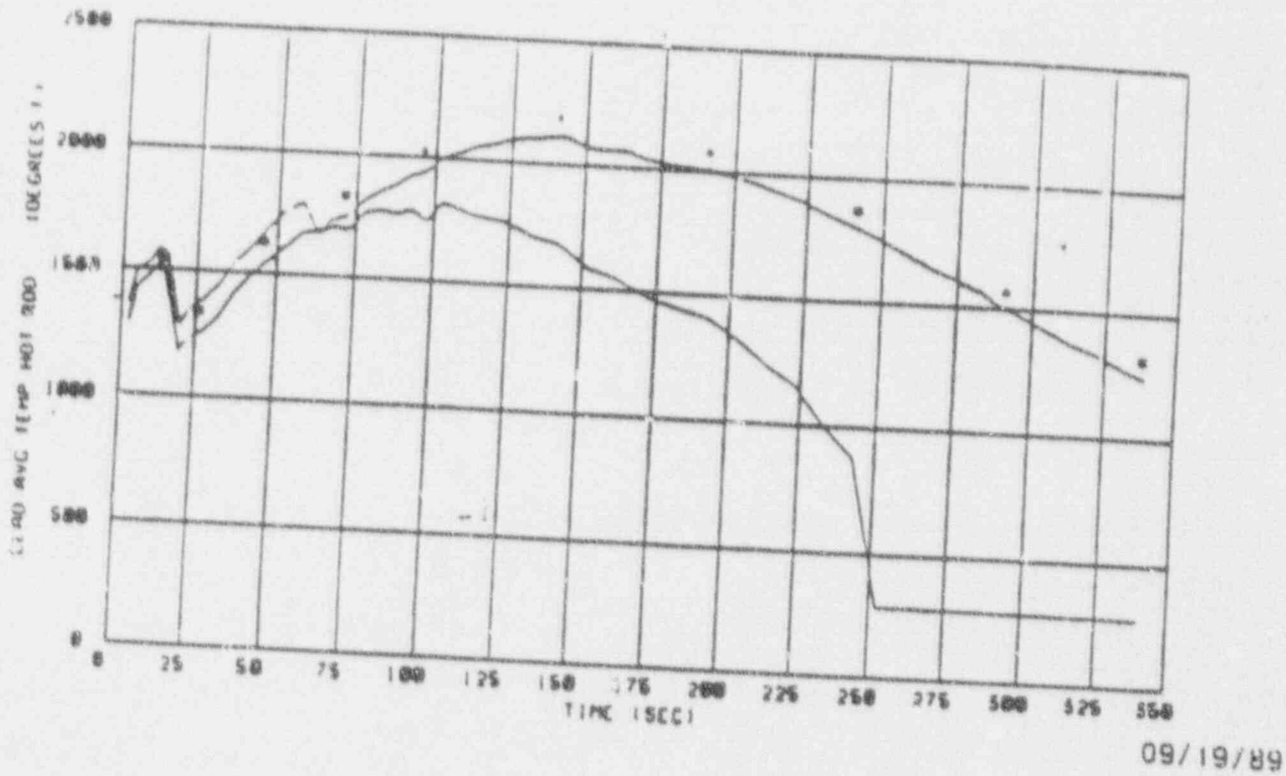


Figure 15.4-14 Hot Rod Clad Average Temperature (DECLG, CD=0.4)
15% STEAM GENERATOR TUBE PLUGGING CASE

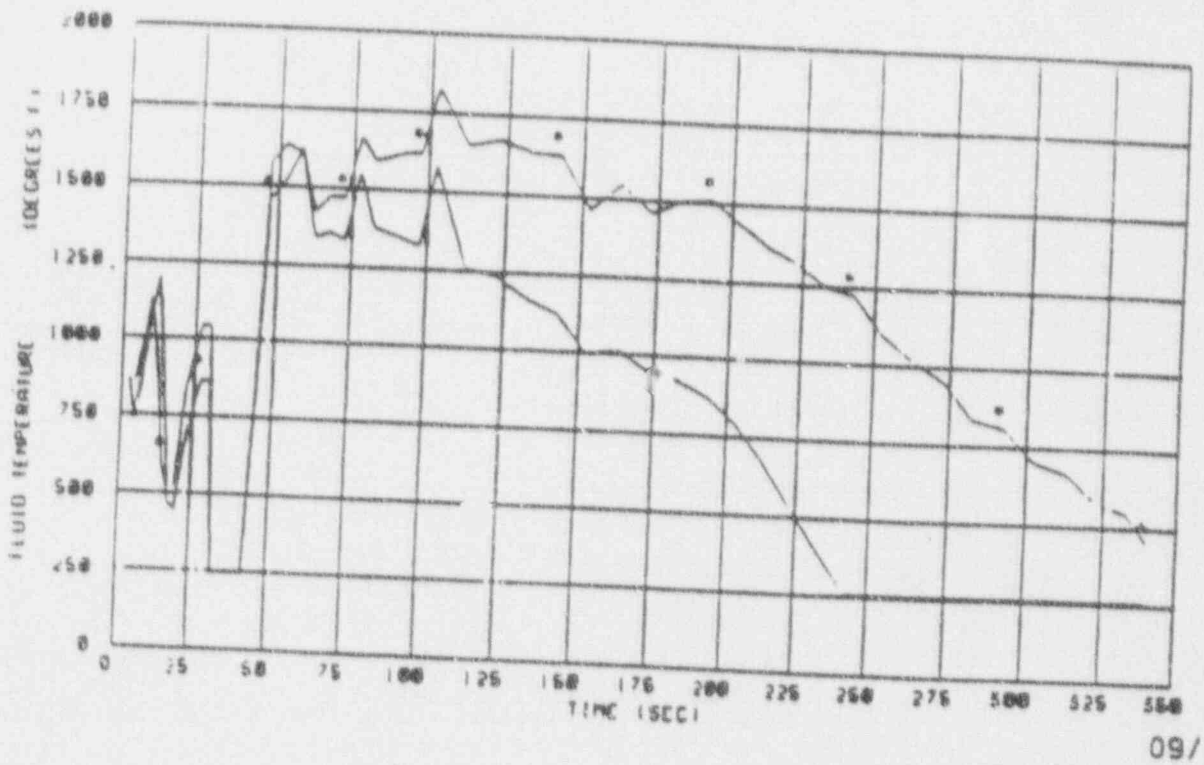


Figure 15.4-15 Fluid Temperature in Hot Assembly (DECLG, CD=0.4)
15% STEAM GENERATOR TUBE PLUGGING CASE

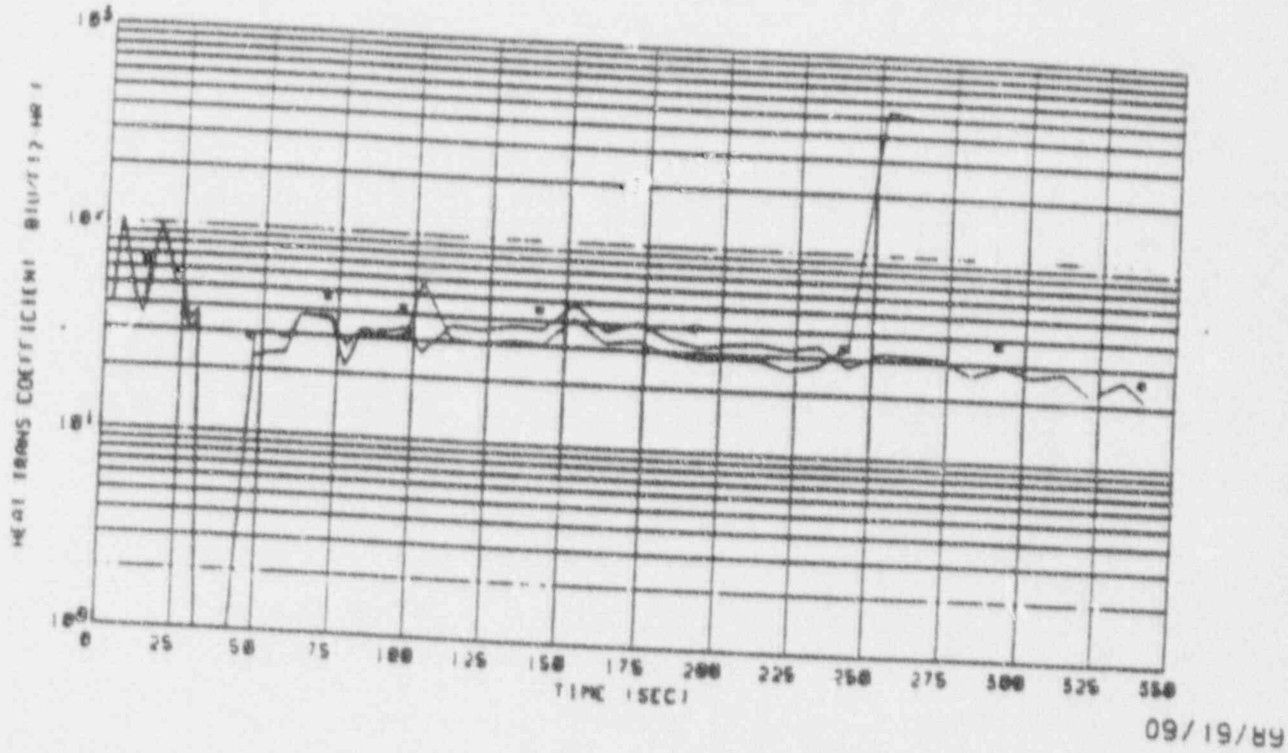
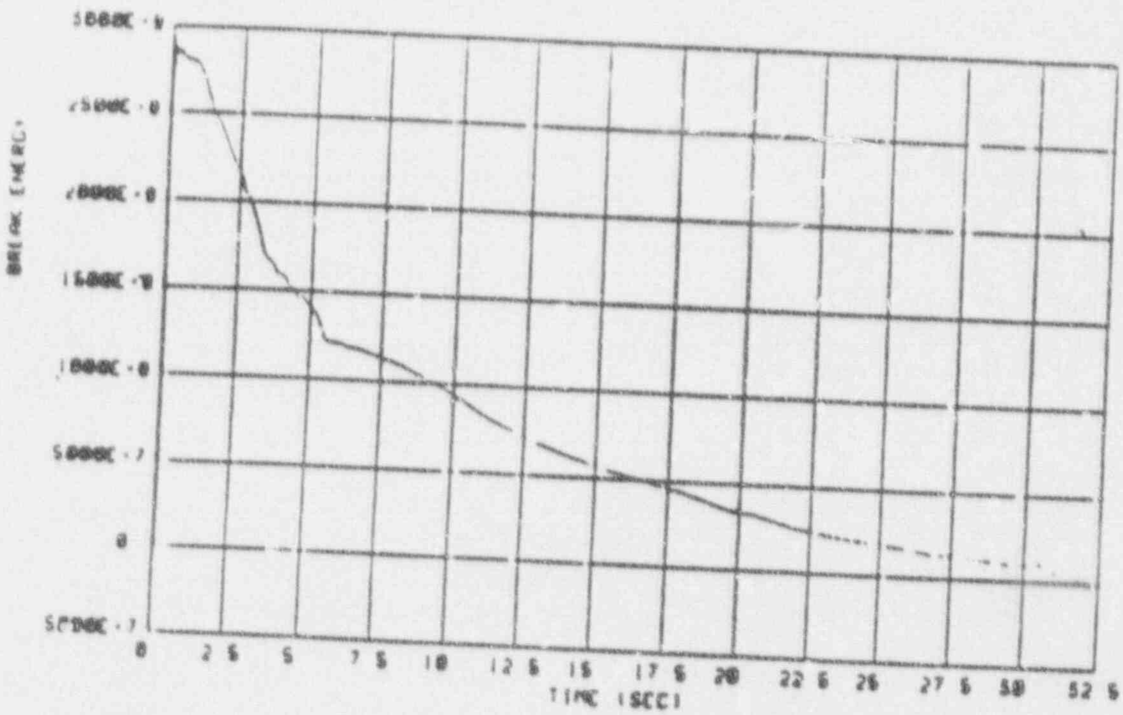


Figure 15.4-16 Hot Rod Heat Transfer Coefficient (DECLG, CD=0.4)
15% STEAM GENERATOR TUBE PLUGGING CASE



09/18/89

Figure 15.4-17 Break Energy Release (DECLG, CD=0.4)
15% STEAM GENERATOR TUBE PLUGGING CASE

APPENDIX "R" DESIGN SUMMARY SHEET AND CHECKLIST

APPENDIX "R" DESIGN SUMMARY SHEET

TITLE/STATION/UNIT: STEAM GENERATOR REPAIR, NORTH ANNA, UNIT 1 (Reference: Programs Review Checklist)	DCP NO. DC 90-13-1
PREPARING ENGINEER/ORGANIZATION/LOCATION: Mark S. Barth/Bechtel/Gaithersburg, MD	DATE: 6/15/92

INSTRUCTIONS:

This design summary is to be completed to ensure that Appendix "R" requirements are maintained with the implementation of this Design Change. Both Part I and II below must be completed.

Pages 2 through 4 must be attached unless the scope of the DCP is clearly minor and does not affect any Appendix R commitments (e.g., part replacements or identical one-for-one component replacements).

PART I: Summary of Compliance

a. Could the DCP impact any Appendix "R" commitments or documentation by modifying any component required for Appendix "R"? [ ] YES [X] NO

b. Does this modification have an impact on the Appendix "R" Report or Drawings such that an "Appendix R Report Change Notification" is included? [ ] YES [X] NO

Regardless of answers above, provide comment or explanation:

The steam generator repair effort will involve equivalent one-for-one component replacements and will not impact any Appendix "R" commitments or documentation.

Signature of Preparing Engineer: MSBarth

Date: 6/15/92

PART II: Summary of Appendix R Coordinator review:

This DCP has been reviewed. Adequate measures are being taken to ensure compliance with Appendix R.

Signature of Appendix R Coordinator: [illegible]

Date: 9/22/92

DELETED

Q-LIST CHANGE REQUEST

Q-LIST CHANGE REQUEST (QLCR)

STATION: NORTH ANNA UNIT(S): 1 QLCR NO. ______

PART 1 - Reason For Change

✓ DCP No. 90-13-1 ___ EWR No. ______ ___ Other (Explain)

If Other/Safety Analysis & 50.59 required and attached? ___ Yes ___ No ✓ N/A

Comments ______

PART II - Required Changes

Components To Be ___ Added ✓ Revised ___ Deleted

(See Page 3, List of Affected Components)

Description of Change: Q-LIST MUST BE UPDATED TO REFLECT THE MANUAL VALVE REPLACEMENTS RELATED TO THE STEAM GENERATOR BLOWDOWN PIPING SYSTEM MODIFICATIONS

MS Barth	9/18/92	M.A. Smith	9/18/92
Preparer	Date	Independent Revwr.	Date

PART III - Site Engineering Review (For "other" reasons only)

A. Regulatory Review

___ RG 1.97 ___ App. "R" ___ No Regulatory Applicability

___ E.Q. ___ Other ___ Regul. Applicability & Acceptance

Comments (Notes Regulatory Issues): ______

Supervisor ISI/NDE & Engr. Programs	Date

B. Engineering Review - This QLCR is ___ Acceptable ___ Not Acceptable

Comments: ______

Systems Engineer	Date	SQL Coordinator	Date

QLCR No. ______________

PART IV - SNSOC Review

____ SNSOC Reviewed & Approved

____ SNSOC Review Not Required

______________________ Reviewed by ______________ Date

PART V - Site Engineering Review (DCPs & EWRs only)

____ QLCR properly reflects (as annotated) implementation through the Final Technical Review.

____ QLCR has been annotated to denote those changes required by the Partial Technical Review completed on ______________.

Comments: ____

______________________ Design Engineer ______________ Date

PART VI - Computerized Q-List Update

A. The computerized Q-List has been revised to reflect this QLCR. (See Attached Memorandum)

______________________ SQL Coordinator ______________ Date

PART VII - Q-List Backup Document (QLBD) Update

The QLCR has been received for incorporation into the QLBD at a later date. (See Section 4.3 of STD-GN-003)

______________________ Records Management ______________ Date

QLCR No. ____________

PART II Continued - List of Affected Components

Components To be Added To The QL: Mark Number	Q.A. Cat.	Components For Which QL Data Needs Revised: Mark Number	Q.A. Cat.	Components To Be Deleted From The QL: Mark Number	Q.A. Cat.
		1-BD-1	SR		
		1-BD-2	SR		
		1-BD-4	SR		
		1-BD-10	SR		
		1-BD-11	SR		
		1-BD-13	SR		
		1-BD-19	SR		
		1-BD-20	SR		
		1-BD-22	SR		

QLCR No.________

QUALITY CLASSIFICATION ANALYSIS (QCA)

STATION: ✓ North Anna __ Surry Unit(s) 1

Component ID No.: 1-BD-1 System: BD

WPTS Mark No: 01-BD- -1 E.Q. Applicable: __ Yes ✓ No
E.Q. Host: __ Yes ✓ No

Component Nomenclature: GLOBE VALVE

Associated Component ID. No.: 1-RC-E-1A

Drawing Reference: 11715-FM-98A Sheet: 2 Revision 16

Other References: 1. DC 90-13-1 2.____
3.____ 4.____
5.____ 6.____

Application:

Describe Functions of the Component:

	Component Function(s)	Function Codes	QA. CAT DEF
1.	SYSTEM PRESSURE BOUNDARY	SYPB	5.1.3
2.	ISOLATION	ISO	5.1.3
3.			
4.			
5.			

Quality Classification ✓ Safety Related (SR) __ Non Safety Related with Special Regulatory Significance (NSQ) __ Non Safety Related (NS)

Comment(s)

Organization Performing Analysis BECHTEL

Analysis Performed By MS Barth Date 9/18/92

Analysis Reviewed By M.A. Smd Date 9/18/92

SUPPLEMENTARY DATA COMPILATION FORM

STATION 59 UNIT 1 SYSTEM BD
COMPONENT ID 1-BD-1
WPTS MARK # 01 - BD - ____ - 1 VERIFIED YES (Y/N)
NOMENCLATURE GLOBE VALVE
APPLICATION ________

REF DRAWING 11715-FM-98A SHEET 2 REVISION 16
MFR # CONVAL (C419) MODEL OR CATALOG 2.50-10G4J-F228K
SERIAL # ________ PO# CNT 390737
PURCH SPEC # NAP-0017 ITEM# ________
AUTH. DCP/EWR DC 90-13-1

PURCH REQUIREMENTS ________

QA CATEGORY SR
COMPONENT FUNCTION SYPB QA CATEGORY DEFINITION 5.1.3

NOMINAL SIZE 2 1/2" NOMINAL CAPACITY ________ UNITS ________
VALVE I.D. ________

COMPONENT DESIGN PRESSURE ________ TEMPERATURE ________
SYSTEM DESIGN PRESSURE ________ TEMPERATURE ________
SYSTEM WORKING PRESSURE ________ TEMPERATURE ________
POWER SOURCE N/A ROTATION N/A
ELECTRICAL POWER VOLTAGE N/A PHASE N/A CURRENT N/A
NEMA CLASS N/A FRAME SIZE N/A INSULATION CLASS N/A
THERMAL/OVERLOAD PROTECTION N/A
CODES AND STANDARDS:

REGULATORY REQUIREMENTS:

BUILDING CONT ELEV 263 COLUMN ________
ENVIRONMENTAL ZONE RC-262A
ASSOCIATED COMPONENT ID 1 - RC - E - 1A
LOCATION DRAWING COORD ________ UFSAR REFERENCE SECTION 10.4.6
ELEC DRAWING REFERENCE N/A TEST LOOP DIAGRAM N/A
PIPE LINE # 2 1/2"-WGCB-4-601C-Q3
DATA BASE COMMENTS ________
ADDITIONAL REFERENCE ________

NPRDS APPLICABLE ____ (Y/N) APPENDIX R APPLICABLE NO (Y/N)
EQ APPLICABLE NO (Y/N) EQ HOST NO (Y/N)
CONFINED ENTRY PERMIT ____ (Y/N) CLASS 1-E NO (Y/N)
REG GUIDE 1.97 NO (Y/N) ISI APPLICABLE YES (Y/N)

STD-GN-0003
Attachment 3
QLCR No.________

QUALITY CLASSIFICATION ANALYSIS (QCA)

STATION: ✓ North Anna __ Surry Unit(s) 1

Component ID No.: 1-BD-2 System: BD

WPTS Mark No: 01-BD- -2 E.Q. Applicable: __ Yes ✓ No
E.Q. Host: __ Yes ✓ No

Component Nomenclature: GLOBE VALVE

Associated Component ID. No.: 1-RC-E-1A

Drawing Reference: 11715-FM-98A Sheet: 2 Revision 16

Other References: 1. DC 90-13-1 2. ____
3. ____ 4. ____
5. ____ 6. ____

Application: ____

Describe Functions of the Component:

	Component Function(s)	Function Codes	QA CAT DEF
1.	SYSTEM PRESSURE BOUNDARY	SYPB	5.1.3
2.	ISOLATION	ISO	5.1.3
3.			
4.			
5.			

Quality Classification: ✓ Safety Related (SR) __ Non Safety Related with Special Regulatory Significance (NSQ) __ Non Safety Related (NS)

Comment(s)____

Organization Performing Analysis BECHTEL

Analysis Performed By M S Barth Date 9/18/92

Analysis Reviewed By M. A. [illegible] Date 9/18/92

SUPPLEMENTARY DATA COMPILATION FORM

STATION 59 UNIT 1 SYSTEM BD
COMPONENT ID 1-BD-2
WPTS MARK # 01 - BD - ____ - 1 VERIFIED YES (Y/N)
NOMENCLATURE GLOBE VALVE
APPLICATION ________

REF DRAWING ________ SHEET ____ REVISION ____
MFR # CONVAL (C419) MODEL OR CATALOG 1.00-10G2J-F225F
SERIAL # ________ PO# ANT 392293
PURCH SPEC # NAP-0023 R# ____
AUTH. DCP/EWR DC 90-13-1

PURCH REQUIREMENTS ________

QA CATEGORY SR
COMPONENT FUNCTION SYPB QA CATEGORY DEFINITION 5.1.3

NOMINAL SIZE 1" NOMINAL CAPACITY ____ UNITS ____
VALVE I.D. ____

COMPONENT DESIGN PRESSURE ____ TEMPERATURE ____
SYSTEM DESIGN PRESSURE ____ TEMPERATURE ____
SYSTEM WORKING PRESSURE ____ TEMPERATURE ____

POWER SOURCE ____ ROTATION ____
ELECTRICAL POWER VOLTAGE ____ PHASE ____ CURRENT ____
NEMA CLASS ____ FRAME SIZE ____ INSULATION CLASS ____
THERMAL/OVERLOAD PROTECTION ____
CODES AND STANDARDS:

REGULATORY REQUIREMENTS:

BUILDING CONT ELEV 263 COLUMN ____
ENVIRONMENTAL ZONE RC-262A
ASSOCIATED COMPONENT ID 1 - RC - E - 1A
LOCATION DRAWING COORD ____ UFSAR REFERENCE SECT. 10.4.6
ELEC DRAWING REFERENCE ____ TEST LOOP DIAGRAM ____
PIPE LINE # 1"-WGCB-6-601C-Q3
DATA BASE COMMENTS ____
ADDITIONAL REFERENCE ____

NPRDS APPLICABLE	____	(Y/N)	APPENDIX R APPLICABLE	NO	(Y/N)
EQ APPLICABLE	NO	(Y/N)	EQ HOST	NO	(Y/N)
CONFINED ENTRY PERMIT	____	(Y/N)	CLASS 1-E	NO	(Y/N)
REG GUIDE 1.97	NO	(Y/N)	ISI APPLICABLE	YES	(Y/N)

QUALITY CLASSIFICATION ANALYSIS (QCA)

STATION: ✓ North Anna ___ Surry Unit(s) 1

Component ID No.: 1-BD-4 System: BD

WPTS Mark No: 01-BD- -4 E.Q. Applicable: ___ Yes ✓ No
E.Q. Host: ___ Yes ✓ No

Component Nomenclature: GLOBE VALVE

Associated Component ID. No.: 1-RC-E-1A

Drawing Reference: 11715-FM-98A Sheet: 2 Revision 16

Other References: 1. DC 90-13-1 2.______
3.______ 4.______
5.______ 6.______

Application: ______

Describe Functions of the Component:

	Component Function(s)	Function Codes	QA. CAT DEF
1.	SYSTEM PRESSURE BOUNDARY	SYPB	5.1.3
2.	ISOLATION	ISO	5.1.3
3.			
4.			
5.			

Quality Classification: ✓ Safety Related (SR) ___ Non Safety Related with Special Regulatory Significance (NSQ) ___ Non Safety Related (NS)

Comment(s)______

Organization Performing Analysis BECHTEL

Analysis Performed By MSBarth Date 9/18/92

Analysis Reviewed By M D [illegible] Date 9/18/92

GN-0003.AT3 02/15/91 9 1

SUPPLEMENTARY DATA COMPILATION FORM

STATION 59 UNIT 1 SYSTEM BD
COMPONENT ID 1-BD-4
WPTS MARK # 01 . BD . ____ . 4 VERIFIED YES (Y/N)
NOMENCLATURE GLOBE VALVE
APPLICATION ________

REF DRAWING ________ SHEET ____ REVISION ____
MFR # C419 MODEL OR CATALOG 2.50-10G4J-F228K
SERIAL # ________ PO# CNT 390737
PURCH SPEC # NAP-0017 ITEM# ________
AUTH. DCP/EWR DC 90-13-1

PURCH REQUIREMENTS ________

QA CATEGORY SR
COMPONENT FUNCTION SYPB QA CATEGORY DEFINITION 5.1.3

NOMINAL SIZE ________ NOMINAL CAPACITY ________ UNITS ________
VALVE I.D. ________

COMPONENT DESIGN PRESSURE ________ TEMPERATURE ________
SYSTEM DESIGN PRESSURE ________ TEMPERATURE ________
SYSTEM WORKING PRESSURE ________ TEMPERATURE ________
POWER SOURCE ________ ROTATION ________
ELECTRICAL POWER VOLTAGE ________ PHASE ________ CURRENT ________
NEMA CLASS ________ FRAME SIZE ________ INSULATION CLASS ________
THERMAL/OVERLOAD PROTECTION ________
CODES AND STANDARDS:

REGULATORY REQUIREMENTS:

BUILDING CONT ELEV 263 COLUMN ________
ENVIRONMENTAL ZONE RC-262A
ASSOCIATED COMPONENT ID 1 - RC - E - 1A
LOCATION DRAWING COORD ________ UFSAR REFERENCE SECTION 10.4.6
ELEC DRAWING REFERENCE ________ TEST LOOP DIAGRAM ________
PIPE LINE # 2½"-WGCB-5-601C-Q3
DATA BASE COMMENTS ________
ADDITIONAL REFERENCE ________

NPRDS APPLICABLE ____ (Y/N) APPENDIX R APPLICABLE NO (Y/N)
EQ APPLICABLE NO (Y/N) EQ HOST NO (Y/N)
CONFINED ENTRY PERMIT ____ (Y/N) CLASS 1-E NO (Y/N)
REG GUIDE 1.97 NO (Y/N) ISI APPLICABLE YES (Y/N)

GN-0005.AT5 02/16/91 9:16am

QUALITY CLASSIFICATION ANALYSIS (QCA)

STATION: ✓ North Anna __ Surry   Unit(s) 1

Component ID No.: 1-BD-10   System: BD

WPTS Mark No: 01-BD- -10   E.Q. Applicable: __ Yes ✓ No
E.Q. Host: __ Yes ✓ No

Component Nomenclature: GLOBE VALVE

Associated Component ID. No.: 1-RC-E-1B

Drawing Reference: 11715-FM-98A   Sheet: 3   Revision 15

Other References: 1. DC90-13-1   2.
3.   4.
5.   6.

Application:

Describe Functions of the Component:

	Component Function(s)	Function Codes	QA CAT DEF
1.	SYSTEM PRESSURE BOUNDARY	STPB	5.1.3
2.	ISOLATION	ISO	5.1.3
3.			
4.			
5.			

Quality Classification: ✓ Safety Related (SR)   __ Non Safety Related with Special Regulatory Significance (NSQ)   __ Non Safety Related (NS)

Comment(s)

Organization Performing Analysis BECHTEL

Analysis Performed By MSBarth   Date 9/18/92

Analysis Reviewed By M.A [illegible]   Date 9/18/92

GN-0003.AT3   02/15/91 9:16am

QLCR NO. ________

SUPPLEMENTARY DATA COMPILATION FORM

STATION 59 UNIT 1 SYSTEM BD
COMPONENT ID 1-BD-10
WPTS MARK # 01 . BD . ____ . 10 VERIFIED YES (Y/N)
NOMENCLATURE GLOBE VALVE
APPLICATION ________

REF DRAWING ________ SHEET ____ REVISION ____
MFR # C419 MODEL OR CATALOG 2.50-10G4J-F22BK
SERIAL # ________ PO# CNT 390737
PURCH SPEC # NAP-0017 ITEM# ____
AUTH. DCP/EWR DC 90-13-1

PURCH REQUIREMENTS ________

QA CATEGORY SR
COMPONENT FUNCTION SYPB QA CATEGORY DEFINITION 5.1.3

NOMINAL SIZE $2\frac{1}{2}$" NOMINAL CAPACITY ____ UNITS ____
VALVE I.D. ____
COMPONENT DESIGN PRESSURE ____ TEMPERATURE ____
SYSTEM DESIGN PRESSURE ____ TEMPERATURE ____
SYSTEM WORKING PRESSURE ____ TEMPERATURE ____
POWER SOURCE ____ ROTATION ____
ELECTRICAL POWER VOLTAGE ____ PHASE ____ CURRENT ____
NEMA CLASS ____ FRAME SIZE ____ INSULATION CLASS ____
THERMAL/OVERLOAD PROTECTION ____
CODES AND STANDARDS:

REGULATORY REQUIREMENTS:

BUILDING CONT ELEV 263 COLUMN ____
ENVIRONMENTAL ZONE RC-262A
ASSOCIATED COMPONENT ID 1 - RC - E - 1B
LOCATION DRAWING COORD ____ UFSAR REFERENCE SECTION 10.4.6
ELEC DRAWING REFERENCE ____ TEST LOOP DIAGRAM ____
PIPE LINE # $2\frac{1}{2}$"-WGCB-7-601C-Q3
DATA BASE COMMENTS ____
ADDITIONAL REFERENCE ____

NPRDS APPLICABLE		(Y/N)	APPENDIX R APPLICABLE	NO	(Y/N)
EQ APPLICABLE	NO	(Y/N)	EQ HOST	NO	(Y/N)
CONFINED ENTRY PERMIT		(Y/N)	CLASS 1-E	NO	(Y/N)
REG GUIDE 1.97	NO	(Y/N)	ISI APPLICABLE	YES	(Y/N)

STD-GN-0003
Attachment 3
QLCR No. ________

QUALITY CLASSIFICATION ANALYSIS (QCA)

STATION: ✓ North Anna ___ Surry   Unit(s) 1

Component ID No.: 1-BD-11   System: BD

WPTS Mark No: 01-BD- -11   E.Q. Applicable: ___ Yes ✓ No
E.Q. Host: ___ Yes ✓ No

Component Nomenclature: GLOBE VALVE

Associated Component ID. No.: 1-RC-E-1B

Drawing Reference: 11715-FM-98A   Sheet: 3   Revision 15

Other References: 1. DC 90-13-1   2. ________
3. ________   4. ________
5. ________   6. ________

Application: ________

Describe Functions of the Component:

	Component Function(s)	Function Codes	QA. CAT DEF
1.	SYSTEM PRESSURE BOUNDARY	SYPB	5.1.3
2.	ISOLATION	ISO	5.1.3
3.			
4.			
5.			

Quality Classification: ✓ Safety Related (SR)   ___ Non Safety Related with Special Regulatory Significance (NSQ)   ___ Non Safety Related (NS)

Comment(s) ________

Organization Performing Analysis BECHTEL

Analysis Performed By MS Barth   Date 9/18/92

Analysis Reviewed By M. A. Pound   Date 9/18/92

SUPPLEMENTARY DATA COMPILATION FORM

STATION 59 UNIT 1 SYSTEM BD
COMPONENT ID 1-BD-11
WPTS MARK # 01 . BD . ____ . 11 VERIFIED YES (Y/N)
NOMENCLATURE GLOBE VALVE
APPLICATION ________

REF DRAWING ________ SHEET ____ REVISION ____
MFR # C419 MODEL OR CATALOG 1.00-10G2J-F225F
SERIAL # ________ PO# CNT 392.293
PURCH SPEC # NAP-0023 ITEM# ________
AUTH. DCP/EWR DC90-13-1
PURCH REQUIREMENTS ________

QA CATEGORY SR
COMPONENT FUNCTION SYPB QA CATEGORY DEFINITION 5.1.3

NOMINAL SIZE 1" NOMINAL CAPACITY ____ UNITS ____
VALVE I.D. ____

COMPONENT DESIGN PRESSURE ____ TEMPERATURE ____
SYSTEM DESIGN PRESSURE ____ TEMPERATURE ____
SYSTEM WORKING PRESSURE ____ TEMPERATURE ____
POWER SOURCE ____ ROTATION ____
ELECTRICAL POWER VOLTAGE ____ PHASE ____ CURRENT ____
NEMA CLASS ____ FRAME SIZE ____ INSULATION CLASS ____
THERMAL/OVERLOAD PROTECTION ____
CODES AND STANDARDS:

REGULATORY REQUIREMENTS:

BUILDING CONT ELEV 263 COLUMN ____
ENVIRONMENTAL ZONE RC-262A
ASSOCIATED COMPONENT ID 1 - RC - E - 1B
LOCATION DRAWING COORD ____ UFSAR REFERENCE SECTION 10.4.6
ELEC DRAWING REFERENCE ____ TEST LOOP DIAGRAM ____
PIPE LINE # 1"-WGCB-9-601C-Q3
DATA BASE COMMENTS ____
ADDITIONAL REFERENCE ____

NPRDS APPLICABLE	____	(Y/N)	APPENDIX R APPLICABLE	NO	(Y/N)
EQ APPLICABLE	NO	(Y/N)	EQ HOST	NO	(Y/N)
CONFINED ENTRY PERMIT	____	(Y/N)	CLASS 1-E	NO	(Y/N)
REG GUIDE 1.97	NO	(Y/N)	ISI APPLICABLE	YES	(Y/N)

STD-GN-0003
Attachment 3
QLCR No.________

QUALITY CLASSIFICATION ANALYSIS (QCA)

STATION: ✓ North Anna __ Surry   Unit(s) 1

Component ID No.: 1-BD-13   System: BD

WPTS Mark No: 01-BD- -13   E.Q. Applicable: __ Yes ✓ No
E.Q. Host: __ Yes ✓ No

Component Nomenclature: GLOBE VALVE

Associated Component ID. No.: 1-RC-E-1B

Drawing Reference: 11715-FM-98A   Sheet: 3   Revision 15

Other References: 1. DC 90-13-1   2. ____
3. ____   4. ____
5. ____   6. ____

Application: ________

Describe Functions of the Component:

	Component Function(s)	Function Codes	QA. CAT DEF
1.	SYSTEM PRESSURE BOUNDARY	SYPB	5.1.3
2.	ISOLATION	ISO	5.1.3
3.			
4.			
5.			

Quality Classification: ✓ Safety Related (SR)   __ Non Safety Related with Special Regulatory Significance (NSQ)   __ Non Safety Related (NS)

Comment(s)________

Organization Performing Analysis BECHTEL

Analysis Performed By M.S. Beith   Date 9/18/92

Analysis Reviewed By M.A. [illegible]   Date 9/18/92

SUPPLEMENTARY DATA COMPILATION FORM

STATION 59 UNIT 1 SYSTEM BD
COMPONENT ID 1-BD-13
WPTS MARK # 01 . BD . ____ . 13 VERIFIED YES (Y/N)
NOMENCLATURE GLOBE VALVE
APPLICATION ______

REF DRAWING ______ SHEET ______ REVISION ______
MFR # C419 MODEL OR CATALOG 2.50-10G4J-F22BK
SERIAL # ______ PO# CNT 390737
PURCH SPEC # NAP-0017 ITEM# ______
AUTH. DCP/EWR DC 90-13-1

PURCH REQUIREMENTS ______

QA CATEGORY SR
COMPONENT FUNCTION SYPB QA CATEGORY DEFINITION 5.1.3

NOMINAL SIZE 2½" NOMINAL CAPACITY ______ UNITS ______
VALVE I.D. ______

COMPONENT DESIGN PRESSURE ______ TEMPERATURE ______
SYSTEM DESIGN PRESSURE ______ TEMPERATURE ______
SYSTEM WORKING PRESSURE ______ TEMPERATURE ______
POWER SOURCE ______ ROTATION ______
ELECTRICAL POWER VOLTAGE ______ PHASE ______ CURRENT ______
NEMA CLASS ______ FRAME SIZE ______ INSULATION CLASS ______
THERMAL/OVERLOAD PROTECTION ______
CODES AND STANDARDS:

REGULATORY REQUIREMENTS:

BUILDING CONT ELEV ______ COLUMN ______
ENVIRONMENTAL ZONE RC-262A
ASSOCIATED COMPONENT ID 1 - RC - E - 1B
LOCATION DRAWING COORD ______ UFSAR REFERENCE SECT. 10.4.6
ELEC DRAWING REFERENCE ______ TEST LOOP DIAGRAM ______
PIPE LINE # 2½"-WGCP-B-601C-Q3
DATA BASE COMMENTS ______
ADDITIONAL REFERENCE ______

NPRDS APPLICABLE ______ (Y/N) APPENDIX R APPLICABLE NO (Y/N)
EQ APPLICABLE NO (Y/N) EQ HOST NO (Y/N)
CONFINED ENTRY PERMIT ______ (Y/N) CLASS 1-E NO (Y/N)
REG GUIDE 1.97 NO (Y/N) ISI APPLICABLE YES (Y/N)

QUALITY CLASSIFICATION ANALYSIS (QCA)

STATION: ✓ North Anna __ Surry   Unit(s) ______

Component ID No.: 1-BD-19   System: BD

WPTS Mark No: 01-BD- -19   E.Q. Applicable: __ Yes ✓ No
E.Q. Host: __ Yes ✓ No

Component Nomenclature: GLOBE VALVE

Associated Component ID. No.: 1-RC-E-1C

Drawing Reference: 11715-FM-98A   Sheet: 4   Revision 13

Other References: 1. DC 90-13-1   2. ______
3. ______   4. ______
5. ______   6. ______

Application: ______

Describe Functions of the Component:

	Component Function(s)	Function Codes	QA CAT DEF
1.	SYSTEM PRESSURE BOUNDARY	SYPB	5.1.3
2.	ISOLATION	ISO	5.1.3
3.			
4.			
5.			

Quality Classification: ✓ Safety Related (SR)   __ Non Safety Related with Special Regulatory Significance (NSQ)   __ Non Safety Related (NS)

Comment(s) ______

Organization Performing Analysis: BECHTEL

Analysis Performed By: M S Barth   Date 9/18/92

Analysis Reviewed By: M D Smith   Date 9/18/92

QLCR NO. ________

SUPPLEMENTARY DATA COMPILATION FORM

STATION 59 UNIT 1 SYSTEM BD
COMPONENT ID 1-BD-19
WPTS MARK # 01 . BD . ____ . 19 ____ VERIFIED YES (Y/N)
NOMENCLATURE GLOBE VALVE
APPLICATION ________

REF DRAWING ________ SHEET ____ REVISION ____
MFR # C419 MODEL OR CATALOG 2.50-1064J-F22K
SERIAL # ________ PO# CNF 390737
PURCH SPEC # NAP-0017 ITEM# ________
AUTH. DCP/EWR DC 90-13-1

PURCH REQUIREMENTS ________

QA CATEGORY SR
COMPONENT FUNCTION SIPB QA CATEGORY DEFINITION 5.1.3

NOMINAL SIZE 2½" NOMINAL CAPACITY ____ UNITS ____
VALVE I.D. ____
COMPONENT DESIGN PRESSURE ____ TEMPERATURE ____
SYSTEM DESIGN PRESSURE ____ TEMPERATURE ____
SYSTEM WORKING PRESSURE ____ TEMPERATURE ____
POWER SOURCE ____ ROTATION ____
ELECTRICAL POWER VOLTAGE ____ PHASE ____ CURRENT ____
NEMA CLASS ____ FRAME SIZE ____ INSULATION CLASS ____
THERMAL/OVERLOAD PROTECTION ____
CODES AND STANDARDS: ________

REGULATORY REQUIREMENTS: ________

BUILDING CONT ELEV 263 COLUMN ____
ENVIRONMENTAL ZONE RC-262A
ASSOCIATED COMPONENT ID 1 - RC - E - 1C
LOCATION DRAWING COORD ____ UFSAR REFERENCE SECTION 10.4.6
ELEC DRAWING REFERENCE ____ TEST LOOP DIAGRAM ____
PIPE LINE # 2½" W6CB-10-601C-Q3
DATA BASE COMMENTS ________
ADDITIONAL REFERENCE ________

NPRDS APPLICABLE ____ (Y/N) APPENDIX R APPLICABLE NO (Y/N)
EQ APPLICABLE NO (Y/N) EQ HOST NO (Y/N)
CONFINED ENTRY PERMIT ____ (Y/N) CLASS 1-E NO (Y/N)
REG GUIDE 1.97 NO (Y/N) ISI APPLICABLE YES (Y/N)

GN-0005.ATB 02/16/91 9:16am

QUALITY CLASSIFICATION ANALYSIS (QCA)

STATION: ✓ North Anna ___ Surry   Unit(s) 1

Component ID No.: 1-BD-20   System: BD

WPTS Mark No: 1-BD- -20   E.Q. Applicable: ___ Yes ✓ No
E.Q. Host: ___ Yes ✓ No

Component Nomenclature: GLOBE VALVE

Associated Component ID. No.: 1-RC-E-1C

Drawing Reference: 11715-FM-98A   Sheet: 4   Revision 13

Other References: 1. DC 90-13-1   2. ____
3. ____   4. ____
5. ____   6. ____

Application: ____

Describe Functions of the Component:

	Component Function(s)	Function Codes	QA. CAT DEF
1.	SYSTEM PRESSURE BOUNDARY	SYPB	5.1.3
2.	ISOLATION	ISO	5.1.3
3.			
4.			
5.			

Quality Classification: ✓ Safety Related (SR)   ___ Non Safety Related with Special Regulatory Significance (NSQ)   ___ Non Safety Related (NS)

Comment(s)____

Organization Performing Analysis BECHTEL

Analysis Performed By R S Brith   Date 9/18/92

Analysis Reviewed By M.A Smith   Date 9/18/92

SUPPLEMENTARY DATA COMPILATION FORM

STATION 59 UNIT 1 SYSTEM BD
COMPONENT ID 1-BD-20
WPTS MARK # 01 - BD - ____ - 20 VERIFIED YES (Y/N)
NOMENCLATURE GLOBE VALVE
APPLICATION ______

REF DRAWING ______ SHEET ______ REVISION ______
MFR # C419 MODEL OR CATALOG 1.00-10G2J-F225F
SERIAL # ______ PO# CNT 392 293
PURCH SPEC # NAP-0023 ITEM# ______
AUTH. DCP/EWR DC 90-13-1

PURCH REQUIREMENTS ______

QA CATEGORY SR
COMPONENT FUNCTION SYPB QA CATEGORY DEFINITION 5.1.3

NOMINAL SIZE 1" NOMINAL CAPACITY ______ UNITS ______
VALVE I.D. ______
COMPONENT DESIGN PRESSURE ______ TEMPERATURE ______
SYSTEM DESIGN PRESSURE ______ TEMPERATURE ______
SYSTEM WORKING PRESSURE ______ TEMPERATURE ______
POWER SOURCE ______ ROTATION ______
ELECTRICAL POWER VOLTAGE ______ PHASE ______ CURRENT ______
NEMA CLASS ______ FRAME SIZE ______ INSULATION CLASS ______
THERMAL/OVERLOAD PROTECTION ______
CODES AND STANDARDS:

REGULATORY REQUIREMENTS:

BUILDING CONT ELEV 262 COLUMN ______
ENVIRONMENTAL ZONE RC-262A
ASSOCIATED COMPONENT ID 1 - RC - E - 1C
LOCATION DRAWING COORD ______ UFSAR REFERENCE SECTION 10.4.6
ELEC DRAWING REFERENCE ______ TEST LOOP DIAGRAM ______
PIPE LINE # 1" WGCB-12-601C-Q3
DATA BASE COMMENTS ______
ADDITIONAL REFERENCE ______

NPRDS APPLICABLE ______ (Y/N) APPENDIX R APPLICABLE NO (Y/N)
EQ APPLICABLE NO (Y/N) EQ HOST NO (Y/N)
CONFINED ENTRY PERMIT ______ (Y/N) CLASS 1-E NO (Y/N)
REG GUIDE 1.97 NO (Y/N) ISI APPLICABLE YES (Y/N)

QUALITY CLASSIFICATION ANALYSIS (QCA)

STATION: ✓ North Anna ___ Surry   Unit(s) 1

Component ID No.: 1-BD-22   System: BD

WPTS Mark No: 01-BD- -22   E.Q. Applicable: ___ Yes ✓ No
E.Q. Host: ___ Yes ✓ No

Component Nomenclature: GLOBE VALVE

Associated Component ID. No.: 1-RC-E-1C

Drawing Reference: 11715-FM-98A   Sheet: 4   Revision 13

Other References: 1. DC 90-13-1   2. ______
3. ______   4. ______
5. ______   6. ______

Application: ______

Describe Functions of the Component:

	Component Function(s)	Function Codes	QA CAT DEF
1.	SYSTEM PRESSURE BOUNDARY	SYPB	5.1.3
2.	ISOLATION	ISC	5.1.3
3.			
4.			
5.			

Quality Classification: ✓ Safety Related (SR)   ___ Non Safety Related with Special Regulatory Significance (NSQ)   ___ Non Safety Related (NS)

Comment(s) ______

Organization Performing Analysis BECHTEL

Analysis Performed By MSBarth   Date 9/18/92

Analysis Reviewed By M. D Smiel   Date 9/18/92

SUPPLEMENTARY DATA COMPILATION FORM

STATION 59 UNIT 1 SYSTEM BD
COMPONENT ID 1-BD-22
WPTS MARK # 01 . BD . ___ . 22 VERIFIED YES (Y/N)
NOMENCLATURE GLOBE VALVE
APPLICATION ___

REF DRAWING ___ SHEET ___ REVISION ___
MFR # C419 MODEL OR CATALOG 2.50-10G4J-F22K
SERIAL # ___ PO# CNT 390737
PURCH SPEC # NAP-0017 ITEM# ___
AUTH. DCP/EWR DC 90-13-1

PURCH REQUIREMENTS ___

QA CATEGORY SR
COMPONENT FUNCTION SYPB QA CATEGORY DEFINITION 5.1.3

NOMINAL SIZE 2½" NOMINAL CAPACITY ___ UNITS ___
VALVE I.D. ___

COMPONENT DESIGN PRESSURE ___ TEMPERATURE ___
SYSTEM DESIGN PRESSURE ___ TEMPERATURE ___
SYSTEM WORKING PRESSURE ___ TEMPERATURE ___
POWER SOURCE ___ ROTATION ___
ELECTRICAL POWER VOLTAGE ___ PHASE ___ CURRENT ___
NEMA CLASS ___ FRAME SIZE ___ INSULATION CLASS ___
THERMAL/OVERLOAD PROTECTION ___
CODES AND STANDARDS: ___

REGULATORY REQUIREMENTS: ___

BUILDING CONT ELEV ___ COLUMN ___
ENVIRONMENTAL ZONE RC-262A
ASSOCIATED COMPONENT ID 1 - RC - E - 1C
LOCATION DRAWING COORD ___ UFSAR REFERENCE SECTION 10.4.6
ELEC DRAWING REFERENCE ___ TEST LOOP DIAGRAM ___
PIPE LINE # 2½"-WGCB-11-601C-Q3
DATA BASE COMMENTS ___
ADDITIONAL REFERENCE ___

NPRDS APPLICABLE ___ (Y/N) APPENDIX R APPLICABLE NO (Y/N)
EQ APPLICABLE NO (Y/N) EQ HOST NO (Y/N)
CONFINED ENTRY PERMIT ___ (Y/N) CLASS 1-5 NO (Y/N)
REG GUIDE 1.97 NO (Y/N) ISI APPLICABLE YES (Y/N)

Q-LIST CHANGE REQUEST (QLCR)

STATION: North Anna UNIT(S): 1 QLCR NO. ______

PART I - Reason For Change

✓ DCP No. DC 90-13-1 ___ EWR No. ______ ___ Other(Explain)

If Other/safety Analysis & 50.59 required and attached? __Yes __No ✓N/A

Comments ______

PART II - Required Changes

Components To Be ___Added ✓ Revised ___Deleted

(See Page 3, List of Affected Components)

Description of Change: Steam Generators - Replace lower assembly and add main steam nozzle flow limiter.
Piping systems - replace existing valves.

MS Barth	9/18/92	M.A. [illegible]	9/18/92
Preparer	Date	Independent Revwr.	Date

PART III - Site Engineering Review (For "other" reasons only)

A. Regulatory Review

__ RG 1.97 __ App. "R" __ No Regulatory Applicability

__ E.Q. __ Other __ Regul. Applicability & Acceptance

Comments (Notes Regulatory Issues): ______

Supervisor ISI/NDE & Engr. Programs ______ Date ______

B. Engineering Review - This QLCR is __ Acceptable __ Not Acceptable

Comments: ______

Systems Engineer	Date	SQL Coordinator	Date

QLCR No.______ ______

PART IV - SNSOC Review

___ SNSOC Reviewed & Approved

___ SNSOC Review Not Required

_______________ _______
Reviewed by Date

PART V - Site Engineering Review (DCPs & EWRs only)

___ QLCR properly reflects (as annotated) implementation through the Final Technical Review.

___ QLCR has been annotated to denote those changes required by the Partial Technical Review completed on ____________________.

Comments:____

____

____

____

____

_______________ _______
Design Engineer Date

PART VI - Computerized Q-List Update

A. The computerized Q-List has been revised to reflect this QLCR (See Attached Memorandum)

_______________ _______
SQL Coordinator Date

PART VII - Q-List Backup Document (QLBD) Update

The QLCR has been received for incorporation into the QLBD at a later date. (See Section 4.3 of STD-GN-0003).

_______________ _______
Records Management Date

QLCR No. ____________

PART II continued - List of Affected Components

Components To Be Added To The OL		Component For Which OL Data Needs Revised		Components To Be Deleted From The OL	
Mark Number	Cat.	Mark Number	Cat.	Mark Number	Cat.
		1-FW-70	SR		
		1-FW-71	SR		
		1-FW-72	SR		
		1-FW-73	SR		
		1-FW-74	SR		
		1-FW-75	SR		
		1-FW-76	SR		
		1-FW-77	SR		
		1-FW-102	SR		
		1-FW-103	SR		
		1-FW-104	SR		
		1-FW-105	SR		
		1-FW-106	SR		
		1-FW-[illegible]07	SR		
		1-FW-108	SR		
		1-FW-[illegible]09	SR		
		1-FW-13[illegible]	SR		
		1-FW-135	SR		
		1-FW-136	SR		
		1-FW-137	SR		
		1-FW-138	SR		

QLCR No. ____________

PART II continued - List of Affected Components

Components To Be Added To The OL: Mark Number	Cat.	Component For Which OL Data Needs Revised: Mark Number	Cat.	Components To Be Deleted From The OL: Mark Number	Cat.
		1-FW-139	SR		
		1-FW-140	SR		
		1-FW-141	SR		
		1-FW-256	SR		
		1-FW-258	SR		
		1-FW-260	SR		
		1-RC-E-1A	SR		
		1-RC-E-1B	SR		
		1-RC-E-1C	SR		

COMPONENT ID: 1-FW-70
QA CATEGORY: SR

* * * * * * * * * * * * * * * * * COMPONENT IDENTIFICATION DATA * * * * * * * * * * * * * * *

ST. 19 UNIT: 01 SYSTEM: FW COMPONENT CODE: VALVE
01-FW - -70 EQ HOST: EQML: N
R APPLICABLE: N VERIFIED: Y REFERENCE DWG: 11715-FM-74A
NOMENCLATURE: VALVE, ~~GATE~~ GLOBE REVISION: 16
APPLICATION: STEAM GENERATOR 1-RC-E-1A LT-1476 I ELEC DRAWING:
SOLATION 01 TEST LOOP DIAGRAM:

* * * * * * * * * * * * * * * * * QUALITY CLASSIFICATION DATA * * * * * * * * * * * * * * * *

ASSOC. COMPONENT ID: 01-FW -LT -1476 COMPONENT CODE: IXMITR

COMPONENT FUNCTION(S)	QA CATEGORY DEFINITION(S)
SYPB SYSTEM PRESSURE BOUNDARY	5.1.2.K
IISO INSTRUMENT ISOLATION	5.4

* EQML DATA *

EQML SOURCE DOC:
FILE: EQML FE DRAWING: REG. GUIDE 1.97:
INTERFACE: EQML LOOP DOC:
FUNCTION: EQML STATUS:
IN DATE: MAINT. CATEGORY: SOUND REASONS TO CONTRARY (SRC):

* * * * * * * * * * * * * * * * COMPONENT PURCHASING DATA * * * * * * * * * * * * * * * *

MANUFACTURER: C419 Conval, Inc. MFGR SOURCE:
MODEL OR CATALOG#: - 0.25-11635-SK-3D MODEL SOURCE: -
P.O. NO.: - P.O. SOURCE: -
SERIAL NO.: - CNT 396432 SERIAL NO. SOURCE: -
STOCK NO.: STOCK NO. SOURCE:
PURCHASE SPEC.: - NAP-0023 PURCHASE SPEC. SOURCE: -
PURCHASE REQ.: - PURCHASE REQ. SOURCE: -

CODES AND STANDARDS:
-

REG DOCUMENT: -

REG. REQUIREMENTS: -

REF DOCUMENT: -

* * * * * * * * * * * * * * * * MECHANICAL EQUIPMENT DATA * * * * * * * * * * * * * * * *

FLOW DIAGRAM VALVE DESIGNATOR: ~~405-008~~ NOMINAL SIZE: .75 NOMINAL SIZE SOURCE: 01
COMPONENT DESIGN PRESSURE: 600 TEMPERATURE: - COMP. DESIGN SOURCE: 46
SYSTEM DESIGN PRESSURE: 1100 TEMPERATURE: 440 SYSTEM DESIGN SOURCE: 08
SYSTEM WORKING PRESSURE: 850 TEMPERATURE: 440 SYSTEM WORKING SOURCE: 08

* * * * * * * * * * * * * * * * ELECTRICAL EQUIPMENT DATA * * * * * * * * * * * * * * * *

POWER SOURCE: NA POWER PHASE: NA POWER SOURCE SOURCE: NA
ELECTRICAL POWER VOLTAGE: NA POWER CURRENT: NA CURRENT SOURCE: NA
ELECTRICAL CONTROL VOLTAGE: NA CONTROL CURRENT: NA CONTROL SOURCE: NA
NEMA CLASS: NA NEMA CLASS SOURCE: NA
INSULATION CLASS: NA INSULATION CLASS SOURCE: NA
FRAME SIZE: NA FRAME SIZE SOURCE: NA
THERMAL/OVERLOAD PROTECTION: NA THERMAL/OVERLOAD SOURCE: NA

* * * * * * * * * * * * * * * * * OTHER COMPONENT DATA * * * * * * * * * * * * * * * * *

NORMAL CAPACITY: NA UNITS: NA CAPACITY SOURCE: N
ROTATION: NA ROTATION SOURCE: NA
BUILDING: RC ELEVATION: 291 BLDG/ELEV. SOURCE:
COLUMN/LOCATION: - COL/LOC SOURCE:
ENVIRONMENTAL ZONE: - ENV. ZONE SOURCE: -
ADD REFERENCE: NAPS UFSAR SECTION 10.4.3 NA
NA NA
CO :

COMPONENT ID: 1-FW-11
QA CATEGORY: SR

* * * * * * * * * * * * * * * * COMPONENT IDENTIFICATION DATA * * * * * * * * * * * * * * *

STATION: 59 UNIT: 01 SYSTEM: FW COMPONENT CODE: VALVE
WPT 01-FW - -71 EQ HOST: EQML: N
R APPLICABLE: N VERIFIED: Y REFERENCE DWG: 11715-FM-74A
ATURE: VALVE, ~~GATE~~ GLOBE REVISION: 16
APPLICATION: STEAM GENERATOR 1-RC-E-1A LT-1476 I ELEC DRAWING:
SOLATION 01 TEST LOOP DIAGRAM:

* * * * * * * * * * * * * * * QUALITY CLASSIFICATION DATA * * * * * * * * * * * * * * * *

ASSOC. COMPONENT ID: 01-FW -LT -1476 COMPONENT CODE: IXMITR

COMPONENT FUNCTION(S)		QA CATEGORY DEFINITION(S)
SYPB	SYSTEM PRESSURE BOUNDARY	5.1.2.K
IISO	INSTRUMENT ISOLATION	5.4

* * * * * * * * * * * * * * * * * * EQML DATA * * * * * * * * * * * * * * * * * *

EQML SOURCE DOC:
FILE: EQML FE DRAWING: REG. GUIDE 1.97:
INTERFACE: EQML LOOP DOC:
FUNCTION: EQML STATUS:
IN DATE: MAINT. CATEGORY: SOUND REASONS TO CONTRARY (SRC):

* * * * * * * * * * * * * * * * COMPONENT PURCHASING DATA * * * * * * * * * * * * * * * *

MANUFACTURER: C419 Conval, Inc. MFGR SOURCE:
MODEL OR CATALOG#: - 0.75-11G35-S16-3D MODEL SOURCE: -
P.O. NO.: - CNT 396432 P.O. SOURCE: -
SERIAL NO.: - SERIAL NO. SOURCE: -
STOCK NO.: STOCK NO. SOURCE:
PURCHASE SPEC.: - NAP-0023 PURCHASE SPEC. SOURCE: -
PURCHASE REQ.: - PURCHASE REQ. SOURCE: -

CODES AND STANDARDS:
-

JMENT: -

REG. REQUIREMENTS: -

REF DOCUMENT: -

* * * * * * * * * * * * * * * * MECHANICAL EQUIPMENT DATA * * * * * * * * * * * * * * * *

FLOW DIAGRAM VALVE DESIGNATOR: ~~HCS-608~~ NOMINAL SIZE: .75 NOMINAL SIZE SOURCE: 01
COMPONENT DESIGN PRESSURE: 1538 TEMPERATURE: - COMP. DESIGN SOURCE: 46
SYSTEM DESIGN PRESSURE: 1100 TEMPERATURE: 440 SYSTEM DESIGN SOURCE: 08
SYSTEM WORKING PRESSURE: 550 TEMPERATURE: 440 SYSTEM WORKING SOURCE: 08

* * * * * * * * * * * * * * * * ELECTRICAL EQUIPMENT DATA * * * * * * * * * * * * * * * *

POWER SOURCE: NA POWER PHASE: NA POWER SOURCE SOURCE: NA
ELECTRICAL POWER VOLTAGE: NA POWER CURRENT: NA CURRENT SOURCE: NA
ELE AL CONTROL VOLTAGE: NA CONTROL CURRENT: NA CONTROL SOURCE: NA
NEM ASS: NA NEMA CLASS SOURCE: NA
INSULATION CLASS: NA INSULATION CLASS SOURCE: NA
FRAME SIZE: NA FRAME SIZE SOURCE: NA
THERMAL/OVERLOAD PROTECTION: NA THERMAL/OVERLOAD SOURCE: NA

* * * * * * * * * * * * * * * * * OTHER COMPONENT DATA * * * * * * * * * * * * * * * * * *

NORMAL CAPACITY: NA UNITS: NA CAPACITY SOURCE: N
ROTATION: NA ROTATION SOURCE: NA
BUILDING: RC ELEVATION: 291 BLDG/ELEV. SOURCE:
COLUMN/LOCATION: - COL/LOC SOURCE:
ENVIRONMENTAL ZONE: - ENV. ZONE SOURCE: -
ADD' REFERENCE: NAPS UFSAR SECTION 10.4.3 NA
NA NA
C(:

COMPONENT ID: 1-FW-72
QA CATEGORY: SR

* * * * * * * * * * * * * * * * * COMPONENT IDENTIFICATION DATA * * * * * * * * * * * * * * *

STAT[illegible]: 59 UNIT: 01 SYSTEM: FW COMPONENT CODE: VALVE
WPT 01-FW - -72 EQ HOST: EQML: N
[illegible] R APPLICABLE: N VERIFIED: Y REFERENCE DWG: 11715-FM-74A
[illegible]ATURE: VALVE, GATE GLOBE REVISION: 16
APPLICATION: STEAM GENERATOR 1-[illegible]-E-1A LT-1475 I ELEC DRAWING:
SOLATION 01 TEST LOOP DIAGRAM:

* * * * * * * * * * * * * * * * QUALITY CLASSIFICATION DATA * * * * * * * * * * * * * * * *

ASSOC. COMPONENT ID: 01-FW -LT -1475 COMPONENT CODE: IXMITR

COMPONENT FUNCTION(S)		QA CATEGORY DEFINITION(S)
SYPB	SYSTEM PRESSURE BOUNDARY	5.1.2.K
IISO	INSTRUMENT ISOLATION	5.4

* EQML DATA *

EQML SOURCE [illegible]:
FILE: EQML FE DRAWING: REG. GUIDE 1.97:
INTERFACE: EQML LOOP DOC:
FUNCTION: EQML STATUS:
IN DATE: MAINT. CATEGORY: SOUND REASONS TO CONTRARY (SRC):

* * * * * * * * * * * * * * * * COMPONENT PURCHASING DATA * * * * * * * * * * * * * * * *

MANUFACTURER: C419 Conval, Inc. MFGR SOURCE:
MODEL OR CATALOG#: - 0.75-1163J-S16-3D MODEL SOURCE: -
P.O. NO.: - CNT 396432 P.O. SOURCE: -
SERIAL NO.: - SERIAL NO. SOURCE: -
STOCK NO.: STOCK NO. SOURCE:
PURCHASE SPEC.: - NAP-0023 PURCHASE SPEC. SOURCE: -
PURCHASE REQ.: - PURCHASE REQ. SOURCE: -

CODES AND STANDARDS:
-

[illegible]UMENT: -

REG. REQUIREMENTS: -

REF DOCUMENT: -

* * * * * * * * * * * * * * * * MECHANICAL EQUIPMENT DATA * * * * * * * * * * * * * * * *

FLOW DIAGRAM VALVE DESIGNATOR: VDS-60B NOMINAL SIZE: .75 NOMINAL SIZE SOURCE: 01
COMPONENT DESIGN PRESSURE: 600 TEMPERATURE: - COMP. DESIGN SOURCE: 46
SYSTEM DESIGN PRESSURE: 1100 TEMPERATURE: 440 SYSTEM DESIGN SOURCE: 08
SYSTEM WORKING PRESSURE: 850 TEMPERATURE: 440 SYSTEM WORKING SOURCE: 08

* * * * * * * * * * * * * * * * ELECTRICAL EQUIPMENT DATA * * * * * * * * * * * * * * * *

POWER SOURCE: NA POWER PHASE: NA POWER SOURCE SOURCE: NA
ELECTRICAL POWER VOLTAGE: NA POWER CURRENT: NA CURRENT SOURCE: NA
ELECTRICAL CONTROL VOLTAGE: NA CONTROL CURRENT: NA CONTROL SOURCE: NA
NEMA CLASS: NA NEMA CLASS SOURCE: NA
INSULATION CLASS: NA INSULATION CLASS SOURCE: NA
FRAME SIZE: NA FRAME SIZE SOURCE: NA
THERMAL/OVERLOAD PROTECTION: NA THERMAL/OVERLOAD SOURCE: NA

* * * * * * * * * * * * * * * * * OTHER COMPONENT DATA * * * * * * * * * * * * * * * * *

NORMAL CAPACITY: NA UNITS: NA CAPACITY SOURCE: N
ROTATION: NA ROTATION SOURCE: NA
BUILDING: RC ELEVATION: [illegible]1 BLDG/ELEV. SOURCE:
COLUMN/LOCATION: - COL/LOC SOURCE:
ENVIRONMENTAL ZONE: - ENV. ZONE SOURCE: -
ADD' REFERENCE: NAPS UFSAR SECTION 10.4.5 NA
NA NA
CO[illegible]:

COMPONENT ID: 1-FW-73
QA CATEGORY: SR

* * * * * * * * * * * * * * * * * COMPONENT IDENTIFICATION DATA * * * * * * * * * * *

STAT[illegible]: 59 UNIT: 01 SYSTEM: FW COMPONENT CODE: VALVE
WP[illegible] 01-FW - -73 EQ HOST: EQML: N
R APPLICABLE: N VERIFIED: Y REFERENCE DWG: 11715-FM-74A
[illegible]ATURE: VALVE, ~~GATE~~ GLOBE REVISION: 16
APPLICATION: STEAM GENERATOR 1-RC-E-1A LT-1475 I ELEC DRAWING:
SOLATION 01 TEST LOOP DIAGRAM:

* * * * * * * * * * * * * * * * QUALITY CLASSIFICATION DATA * * * * * * * * * * * * * * * * *

ASSOC. COMPONENT ID: 01-FW -LT -1475 COMPONENT CODE: IXMITR

COMPONENT FUNCTION(S)		QA CATEGORY DEFINITION(S)
SYPB	SYSTEM PRESSURE BOUNDARY	5.1.2.K
IISO	INSTRUMENT ISOLATION	5.4

* * * * * * * * * * * * * * * * * * * EQML DATA *

EQML SOURCE DOC:
FILE: EQML FE DRAWING: REG. GUIDE 1.97:
INTERFACE: EQML LOOP DOC:
FUNCTION: EQML STATUS:
[illegible] DATE: MAINT. CATEGORY: SOUND REASONS TO CONTRARY (SRC):

* * * * * * * * * * * * * * * * COMPONENT PURCHASING DATA * * * * * * * * * * * * * * * *

MANUFACTURER: C419 Conval, Inc. MFGR SOURCE:
MODEL OR CATALOG#: - 0.75"-1163J-S16-3D MODEL SOURCE: -
P.O. NO.: - CNT 346432 P.O. SOURCE: -
SERIAL NO.: - SERIAL NO. SOURCE: -
STOCK NO.: STOCK NO. SOURCE:
PURCHASE SPEC.: - NAP-0023 PURCHASE SPEC. SOURCE: -
PURCHASE REQ.: - PURCHASE REQ. SOURCE: -

CODES AND STANDARDS:
-

[illegible]MENT: -

REG. REQUIREMENTS: -

REF DOCUMENT: -

* * * * * * * * * * * * * * * * MECHANICAL EQUIPMENT DATA * * * * * * * * * * * * * * * *

FLOW DIAGRAM VALVE DESIGNATOR: ~~VGS-60R~~ NOMINAL SIZE: .75 NOMINAL SIZE SOURCE: 61
COMPONENT DESIGN PRESSURE: 600 TEMPERATURE: - COMP. DESIGN SOURCE: 46
SYSTEM DESIGN PRESSURE: 1100 TEMPERATURE: 440 SYSTEM DESIGN SOURCE: 08
SYSTEM WORKING PRESSURE: 850 TEMPERATURE: 440 SYSTEM WORKING SOURCE: 08

* * * * * * * * * * * * * * * * ELECTRICAL EQUIPMENT DATA * * * * * * * * * * * * * * * *

POWER SOURCE: NA POWER PHASE: NA POWER SOURCE SOURCE: NA
ELECTRICAL POWER VOLTAGE: NA POWER CURRENT: NA CURRENT SOURCE: NA
ELECTRICAL CONTROL VOLTAGE: NA CONTROL CURRENT: NA CONTROL SOURCE: NA
NEMA CLASS: NA NEMA CLASS SOURCE: NA
INSULATION CLASS: NA INSULATION CLASS SOURCE: NA
FRAME SIZE: NA FRAME SIZE SOURCE: NA
THERMAL/OVERLOAD PROTECTION: NA THERMAL/OVERLOAD SOURCE: NA

* * * * * * * * * * * * * * * * * * OTHER COMPONENT DATA * * * * * * * * * * * * * * * * * *

NORMAL CAPACITY: NA UNITS: NA CAPACITY SOURCE: N
ROTATION: NA ROTATION SOURCE: NA
BUILDING: RC ELEVATION: 291 BLDG/ELEV. SOURCE:
COLUMN/LOCATION: - COL/LOC SOURCE:
ENVIRONMENTAL ZONE: - ENV. ZONE SOURCE: -
ADD[illegible] REFERENCE: NAPS UFSAR SECTION 10.4.3 NA
NA NA
CO[illegible]:

E[illegible]F REPORT

COMPONENT ID: 1-FW-74
QA CATEGORY: 52

* * * * * * * * * * * * * * * * COMPONENT IDENTIFICATION DATA * * * * * * * * * * * * * *

ST/ 59 UNIT: 01 SYSTEM: FW COMPONENT CODE: VALVE
01-FW - -74 EQ HOST: N EQML: N
R APPLICABLE: N VERIFIED: Y REFERENCE DWG: 11715-FM-74A
NOMENCLATURE: VALVE, GLOBE REVISION: 28
APPLICATION: STEAM GENERATOR 1-RC-E-1A LT-1474 I ELEC DRAWING:
SOLATION, ROOT VALVE 01 TEST LOOP DIAGRAM:

* * * * * * * * * * * * * * * * QUALITY CLASSIFICATION DATA * * * * * * * * * * * * * * * *

ASSOC. COMPONENT ID: 01-FW -LT -1474 COMPONENT CODE: IXMITR

COMPONENT FUNCTION(S)		QA CATEGORY DEFINITION(S)
SYPB	SYSTEM PRESSURE BOUNDARY	5.1.2.K
IISO	INST- 'ENT ISOLATION	5.4

* EQML DATA *

EQML SOURCE DOC:
FILE: EQML FE DRAWING: REG. GUIDE 1.97:
INTERFACE: EQML LOOP DOC:
FUNCTION: EQML STATUS:
IN DATE: MAINT. CATEGORY: SOUND REASONS TO CONTRARY (SRC):

* * * * * * * * * * * * * * * * COMPONENT PURCHASING DATA * * * * * * * * * * * * * * * *

MANUFACTURER: C419 CONVAL, INC. MFGR SOURCE: -
MODEL OR CATALOG#: ~~[illegible]~~ 0.75-1163J-S16-3D MODEL SOURCE: -
P.O. NO.: ~~NS-[illegible]~~ CNT 396432 P.O. SOURCE: -
SERIAL NO.: - SERIAL NO. SOURCE: -
STOCK NO.: STOCK NO. SOURCE:
PURCHASE SPEC.: - NAP-0023 PURCHASE SPEC. SOURCE: -
PURCHASE REQ.: 3 1 2T PURCHASE REQ. SOURCE: -

COD' ND STANDARDS:
A A4 K3 A2 A12

DOCUMENT: -

REG. REQUIREMENTS: 04 02

REF DOCUMENT: -

* * * * * * * * * * * * * * * * MECHANICAL EQUIPMENT DATA * * * * * * * * * * * * * * * *

FLOW DIAGRAM VALVE DESIGNATOR:	~~01-08F~~	NOMINAL SIZE:	3/4	NOMINAL SIZE SOURCE:	01
COMPONENT DESIGN PRESSURE:	3925	TEMPERATURE:	600	COMP. DESIGN SOURCE:	46
SYSTEM DESIGN PRESSURE:	1085	TEMPERATURE:	560	SYSTEM DESIGN SOURCE:	08
SYSTEM WORKING PRESSURE:	1005	TEMPERATURE:	545	SYSTEM WORKING SOURCE:	08

* * * * * * * * * * * * * * * * ELECTRICAL EQUIPMENT DATA * * * * * * * * * * * * * * * *

POWER SOURCE: NA POWER PHASE: NA POWER SOURCE SOURCE: NA
ELECTRICAL POWER VOLTAGE: NA POWER CURRENT: NA CURRENT SOURCE: NA
ELECTRICAL CONTROL VOLTAGE: NA CONTROL CURRENT: NA CONTROL SOURCE: NA
NEMA CLASS: NA NEMA CLASS SOURCE: NA
INSULATION CLASS: NA INSULATION CLASS SOURCE: NA
FRAME SIZE: NA FRAME SIZE SOURCE: NA
THERMAL/OVERLOAD PROTECTION: NA THERMAL/OVERLOAD SOURCE: NA

* * * * * * * * * * * * * * * * OTHER COMPONENT DATA * * * * * * * * * * * * * * * * * *

NORMAL CAPACITY: NA UNITS: NA CAPACITY SOURCE: N
ROTATION: NA ROTATION SOURCE: NA
BUILDING: RC ELEVATION: 291 BLDG/ELEV. SOURCE: -
COLUMN/LOCATION: - COL/LOC SOURCE: -
ENVIRONMENTAL ZONE: - ENV. ZONE SOURCE: -
AD' EFERENCE: NAPS UFSAR SECTION 10.4.3 NA
NA NA

CO J:

COMPONENT ID: 1-FW-75
QA CATEGORY: SR

* * * * * * * * * * * * * * * * * COMPONENT IDENTIFICATION DATA * * * * * * * * * * * * * * *

STAT'''N: 59 UNIT: 01 SYSTEM: FW COMPONENT CODE: VALVE
WP' 01-FW - -75 EQ HOST: EQML: N
R APPLICABLE: N VERIFIED: Y REFERENCE DWG: 11715-FM-74A
ATURE: VALVE, ~~GATE~~ GLOBE REVISION: 16
APPLICATION: STEAM GENERATOR 1-RC-E-1A LT-1474 I ELEC DRAWING:
SOLATION 01 TEST LOOP DIAGRAM:

* * * * * * * * * * * * * * * * * QUALITY CLASSIFICATION DATA * * * * * * * * * * * * * * * * *

ASSOC. COMPONENT ID: 01-FW -LT -1474 COMPONENT CODE: IXMITR

COMPONENT FUNCTION(S)	QA CATEGORY DEFINITION(S)
SYPB SYSTEM PRESSURE BOUNDARY	5.1.2.K
IISO INSTRUMENT ISOLATION	5.4

* EQML DATA * * * * * * * * * * * * * * * * * * *

EQML SOURCE DOC:
FILE: EQML FE DRAWING: REG. GUIDE 1.97.
INTERFACE: EQML LOOP DOC:
FUNCTION: EQML STATUS:
IN DATE: MAINT. CATEGORY: SOUND REASONS TO CONTRARY (SRC):

* * * * * * * * * * * * * * * * * COMPONENT PURCHASING DATA * * * * * * * * * * * * * * * * *

MANUFACTURER: C419 Conval, Inc. MFGR SOURCE:
MODEL OR CATALOG#: 0.75"-116.35-S16-3D MODEL SOURCE: -
P.O. NO.: - CNT 396432 P.O. SOURCE: -
SERIAL NO.: - SERIAL NO. SOURCE: -
STOCK NO.: STOCK NO. SOURCE:
PURCHASE SPEC.: - NAP-0023 PURCHASE SPEC. SOURCE: -
PURCHASE REQ.: - PURCHASE REQ. SOURCE: -

CODES AND STANDARDS:
-

MENT: -

REG. REQUIREMENTS: -

REF DOCUMENT: -

* * * * * * * * * * * * * * * * * MECHANICAL EQUIPMENT DATA * * * * * * * * * * * * * * * * *

FLOW DIAGRAM VALVE DESIGNATOR: ~~VGS-600~~ NOMINAL SIZE: .75 NOMINAL SIZE SOURCE: 01
COMPONENT DESIGN PRESSURE: 600 TEMPERATURE: - COMP. DESIGN SOURCE: 46
SYSTEM DESIGN PRESSURE: 1100 TEMPERATURE: 440 SYSTEM DESIGN SOURCE: 08
SYSTEM WORKING PRESSURE: 850 TEMPERATURE: 440 SYSTEM WORKING SOURCE: 08

* * * * * * * * * * * * * * * * * ELECTRICAL EQUIPMENT DATA * * * * * * * * * * * * * * * * *

POWER SOURCE: NA POWER P SE: NA POWER SOURCE SOURCE: NA
ELECTRICAL POWER VOLTAGE: NA POWER CURRENT: NA CURRENT SOURCE: NA
ELECTRICAL CONTROL VOLTAGE: NA CONTROL CURRENT: NA CONTROL SOURCE: NA
NEMA CLASS: NA NEMA CLASS SOURCE: NA
INSULATION CLASS: NA INSULATION CLASS SOURCE: NA
FRAME SIZE: NA FRAME SIZE SOURCE: NA
THERMAL/OVERLOAD PROTECT? NA THERMAL/OVERLOAD SOURCE: NA

* * * * * * * * * * * * * * * * * OTHER COMPONENT DATA * * * * * * * * * * * * * * * * * * *

NORMAL CAPACITY: NA UNITS: NA CAPACITY SOURCE: N
ROTATION: NA ROTATION SOURCE: NA
BUILDING: RC ELEVATION: 302 BLDG/ELEV. SOURCE:
COLUMN/LOCATION: - COL/LOC SOURCE:
ENVIRONMENTAL ZONE: - ENV. ZONE SOURCE: -
ADD' REFERENCE: NAPS UFSAR SECTION 10.4.3 NA
NA NA
CO :

COMPONENT ID: 1-FW-76
QA CATEGORY: SR

* * * * * * * * * * * * * * * * COMPONENT IDENTIFICATION DATA * * * * * * * * * * * * * *

STATION: 59 UNIT: 01 SYSTEM: FW COMPONENT CODE: VALVE
WPT 01-FW - -76 EQ HOST: N EQML: N
APPLICABLE: N VERIFIED: Y REFERENCE DWG: 11715-FM-74A
NOMENCLATURE: VALVE, ~~GATE~~ GLOBE REVISION: 26
APPLICATION: STEAM GENERATOR 1-RC-E-1A LT-1477 ; ELEC DRAWING:
SOLATION 01 TEST LOOP DIAGRAM:

* * * * * * * * * * * * * * * * QUALITY CLASSIFICATION DATA * * * * * * * * * * * * * * * *

ASSOC. COMPONENT ID: 01-FW -LT -1477 COMPONENT CODE: IXMITR

COMPONENT FUNCTION(S)	QA CATEGORY DEFINITION(S)
SYPB SYSTEM PRESSURE BOUNDARY	5.1.2.K
IISO INSTRUMENT ISOLATION	5.4

* EQML DATA *

EQML SOURCE DOC:
FILE: EQML FE DRAWING: REG. GUIDE 1.97:
INTERFACE: EQML LOOP DOC:
FUNCTION: EQML STATUS:
IN DATE: MAINT. CATEGORY: SOUND REASONS TO CONTRARY (SRC):

* * * * * * * * * * * * * * * * COMPONENT PURCHASING DATA * * * * * * * * * * * * * * * *

MANUFACTURER: ~~V010 VOGT, HENRY, MACH. CO., INC~~ C419 Conval, Inc MFGR SOURCE: -
MODEL OR CATALOG#: ~~SW 0000~~ 0.75-1163J-516-2D MODEL SOURCE: -
P.O. NO.: ~~0000~~ - CNT396432 P.O. SOURCE: -
SERIAL NO.: - SERIAL NO. SOURCE: -
STOCK NO.: STOCK NO. SOURCE:
PURCHASE SPEC.: ~~NAS 3016~~ NAP-0023 PURCHASE SPEC. SOURCE: -
PURCHASE REQ.: 1 2A3 6 PURCHASE REQ. SOURCE: -

CODES AND STANDARDS:
A2 A30 B21 C4 C12 K1 K3

ATTACHMENT: -

REG. REQUIREMENTS: 02

REF DOCUMENT: 04

* * * * * * * * * * * * * * * * MECHANICAL EQUIPMENT DATA * * * * * * * * * * * * * * * *

FLOW DIAGRAM VALVE DESIGNATOR: ~~KGS-66C~~ NOMINAL SIZE: .75 NOMINAL SIZE SOURCE: 01
COMPONENT DESIGN PRESSURE: 1575 TEMPERATURE: 500 COMP. DESIGN SOURCE: 46
SYSTEM DESIGN PRESSURE: 1100 TEMPERATURE: 440 SYSTEM DESIGN SOURCE: 08
SYSTEM WORKING PRESSURE: 850 TEMPERATURE: 440 SYSTEM WORKING SOURCE: 08

* * * * * * * * * * * * * * * * ELECTRICAL EQUIPMENT DATA * * * * * * * * * * * * * * * *

POWER SOURCE: NA POWER PHASE: NA POWER SOURCE SOURCE: NA
ELECTRICAL POWER VOLTAGE: NA POWER CURRENT: NA CURRENT SOURCE: NA
ELECTRICAL CONTROL VOLTAGE: NA CONTROL CURRENT: NA CONTROL SOURCE: NA
NEMA CLASS: NA NEMA CLASS SOURCE: NA
INSULATION CLASS: NA INSULATION CLASS SOURCE: NA
FRAME SIZE: NA FRAME SIZE SOURCE: NA
THERMAL OVERLOAD PROTECTION: NA THERMAL/OVERLOAD SOURCE: NA

* * * * * * * * * * * * * * * * * OTHER COMPONENT DATA * * * * * * * * * * * * * * * * * *

NORMAL CAPACITY: NA UNITS: NA CAPACITY SOURCE: N
ROTATION: NA ROTATION SOURCE: NA
BUILDING: RC ELEVATION: 291 BLDG/ELEV. SOURCE: -
COLUMN/LOCATION: - COL/LOC SOURCE: -
ENVIRONMENTAL ZONE: - ENV. ZONE SOURCE: -
ADDL REFERENCE: NAPS UFSAR SECTION 10.4.3 NA
NA NA
COMMENTS:

COMPONENT ID: 1-FW-77
QA CATEGORY: SR

* * * COMPONENT IDENTIFICATION DATA * * *

STAT : 59 UNIT: 01 SYSTEM: FW COMPONENT CODE: VALVE
01-FW - -77 EQ HOST: EQML: N
R APPLICABLE: N VERIFIED: Y REFERENCE DWG: 11715-FM-74A
ATURE: VALVE, ~~GATE~~ GLO E REVISION: 16
APPLICATION: STEAM GENERATOR 1-RC-E-1A LT-1477 I ELEC DRAWING:
SOLATION 01 TEST LOOP DIAGRAM:

* * * QUALITY CLASSIFICATION DATA * * *

ASSOC. COMPONENT ID: 01-FW -LT -1477 COMPONENT C XMITR

COMPONENT FUNCTION(S) QA CATEGORY D (S)
SYPB SYSTEM PRESSURE BOUNDARY 5.1.2.K
IISO INSTRUMENT ISOLATION 5.4

* * * EQML DATA * * *

EQML SOURCE DOC:
FILE: EQML FE DRAWING: REG. GUIDE 1.97:
INTERFACE: EQML LOOP DOC:
FUNCTION: EQML STATUS:
IN DATE: MAINT. CATEGORY: SOUND REASONS TO CONTRARY (SRC):

* * * COMPONENT PURCHASING DATA * * *

MANUFACTURER: C419 Conval Inc. MFGR SOURCE:
MODEL OR CATALOG#: - 0.75-1163J-16-3D MODEL SOURCE: -
P.O. NO.: - CNT 396432 P.O. SOURCE: -
SERIAL NO.: - SERIAL NO. SOURCE: -
STOCK NO.: STOCK NO. SOURCE:
PURCHASE SPEC.: - NAP-0023 PURCHASE SPEC. SOURCE: -
PURCHASE REQ.: - PURCHASE REQ. SOURCE: -

CODES AND STANDARDS:
-

UMENT: -

REG. REQUIREMENTS: -

REF DOCUMENT: -

* * * MECHANICAL EQUIPMENT DATA * * *

FLOW DIAGRAM VALVE DESIGNATOR: ~~VGS-600~~ NOMINAL SIZE: .75 NOMINAL SIZE SOURCE: 01
COMPONENT DESIGN PRESSURE: 600 TEMPERATURE: - COMP. DESIGN SOURCE: 46
SYSTEM DESIGN PRESSURE: 1100 TEMPERATURE: 440 SYSTEM DESIGN SOURCE: 08
SYSTEM WORKING PRESSURE: 850 TEMPERATURE: 440 SYSTEM WORKING SOURCE: 08

* * * ELECTRICAL EQUIPMENT DATA * * *

POWER SOURCE: NA POWER PHASE: NA POWER SOURCE SOURCE: NA
ELECTRICAL POWER VOLTAGE: NA POWER CURRENT: NA CURRENT SOURCE: NA
ELECTRICAL CONTROL VOLTAGE: NA CONTROL CURRENT: NA CONTROL SOURCE: NA
NEMA CLASS: NA NEMA CLASS SOURCE: NA
INSULATION CLASS: NA INSULATION CLASS SOURCE: NA
FRAME SIZE: NA FRAME SIZE SOURCE: NA
THERMAL/OVERLOAD PROTECTION: NA THERMAL/OVERLOAD SOURCE: NA

* * * OTHER COMPONENT DATA * * *

NORMAL CAPACITY: NA UNITS: NA CAPACITY SOURCE: N
ROTATION: NA ROTATION SOURCE: NA
BUILDING: RC ELEVATION: 262 BLDG/ELEV. SOURCE:
COLUMN/LOCATION: - COL/LOC SOURCE:
ENVIRONMENTAL ZONE: - ENV. ZONE SOURCE: -
ADD REFERENCE: NAPS UFSAR SECTION 10.4.3 NA
NA NA
CO S:

OF REPORT

COMPONENT ID: 1-FW-102
QA CATEGORY: SR

* * * * * * * * * * * * * * * * * * COMPONENT IDENTIFICATION DATA * * * * * * * * * * * * * * * *

STATION: 69 UNIT: 01 SYSTEM: FW COMPONENT CODE: VALVE
WP' - 01-FW - -102 EQ HOST: EQML: N
R APPLICABLE: N VERIFIED: Y REFERENCE DWG: 11715-FM-74A
ATURE: VALVE, ~~GATE~~ GLOBE REVISION: 16
APPLICATION: STEAM GENERATOR 1-RC-E-1B LT-1486 I ELEC DRAWING:
SOLATION 01 TEST LOOP DIAGRAM:

* * * * * * * * * * * * * * * * * QUALITY CLASSIFICATION DATA * * * * * * * * * * * * * * * * *

ASSOC. COMPONENT ID: 01-FW -LT -1486 COMPONENT CODE: IXMITR

COMPONENT FUNCTION(S)		QA CATEGORY DEFINITION(S)
SYPB	SYSTEM PRESSURE BOUNDARY	5.1.2.K
IISO	INSTRUMENT ISOLATION	5.4

* EQML DATA *

EQML SOURCE DOC:
FILE: EQML DRAWING: REG. GUIDE 1.97:
INTERFACE: EQML LOOP DOC:
FUNCTION: EQML STATUS:
IN DATE: MAINT. CATEGORY: SOUND REASONS TO CONTRARY (SRC):

* * * * * * * * * * * * * * * * * COMPONENT PURCHASING DATA * * * * * * * * * * * * * * * * *

MANUFACTURER: C419 Conval, Inc. MFGR SOURCE:
MODEL OR CATALOG#: - 0.75-11G35-516-3D MODEL SOURCE: -
P.O. NO.: - CNT 396432 P.O. SOURCE: -
SERIAL NO.: - SERIAL NO. SOURCE: -
STOCK NO.: STOCK NO. SOURC.
PURCHASE SPEC.: - NAP-0023 PURCHASE SPEC. SOURCE:
PURCHASE REQ.: - PURCHASE REQ. SOURCE: -

CODES AND STANDARDS:
-

MENT: -

REG. REQUIREMENTS: -

REF DOCUMENT: -

* * * * * * * * * * * * * * * * * MECHANICAL EQUIPMENT DATA * * * * * * * * * * * * * * * * *

FLOW DIAGRAM VALVE DESIGNATOR: ~~VBS 60B~~ NOMINAL SIZE: .75 NOMINAL SIZE SOURCE: 01
COMPONENT DESIGN PRESSURE: 600 TEMPERATURE: - COMP. DESIGN SOURCE: 46
SYSTEM DESIGN PRESSURE: 1100 TEMPERATURE: 440 SYSTEM DESIGN SOURCE: 08
SYSTEM WORKING PRESSURE: 850 TEMPERATURE: 440 SYSTEM WORKING SOURCE: 08

* * * * * * * * * * * * * * * * * ELECTRICAL EQUIPMENT DATA * * * * * * * * * * * * * * * * *

POWER SOURCE: NA POWER PHASE: NA POWER SOURCE SOURCE: NA
ELECTRICAL POWER VOLTAGE: NA POWER CURRENT: NA CURRENT SOURCE: NA
ELECTRICAL CONTROL VOLTAGE: NA CONTROL CURRENT: NA CONTROL SOURCE: NA
NEMA CLASS: NA NEMA CLASS SOURCE: NA
INSULATION CLASS: NA INSULATION CLASS SOURCE: NA
FRAME SIZE: NA FRAME SIZE SOURCE: NA
THERMAL/OVERLOAD PROTECTION: NA THERMAL/OVERLOAD SOURCE: NA

* * * * * * * * * * * * * * * * * * OTHER COMPONENT DATA * * * * * * * * * * * * * * * * * *

NORMAL CAPACITY: NA UNITS: NA CAPACITY SOURCE: N
ROTATION: NA ROTATION SOURCE: NA
BUILDING: RC ELEVATION: 291 BLDG/ELEV. SOURCE:
COLUMN/LOCATION: - COL/LOC SOURCE:
ENVIRONMENTAL ZONE: - ENV. ZONE SOURCE: -
ADDL REFERENCE: NAPS UFSAR SECTION 10.4.3 NA
NA NA
CC :

COMPONENT ID: 1-FW-103
QA CATEGORY: SR

* * * * * * * * * * * * * * * * * * COMPONENT IDENTIFICATION DATA * * * * * * * * * * * * * * *

STATION: 59 UNIT: 01 SYSTEM: FW COMPONENT CODE: VALVE
WPT 01-FW - -103 EQ HOST: EQML: N
R APPLICABLE: N VERIFIED: Y REFERENCE DWG: 11715-FM-74A
TURE: VALVE, GATE GLOBE REVISION: 16
APPLICATION: STEAM GENERATOR 1-RC-E-1B LT-1486 I ELEC DRAWING:
SOLATION 01 TEST LOOP DIAGRAM:

* * * * * * * * * * * * * * * * * QUALITY CLASSIFICATION DATA * * * * * * * * * * * * * * * *

ASSOC. COMPONENT ID: 01-FW -LT -1486 COMPONENT CODE: IXMITR

COMPONENT FUNCTION(S)	QA CATEGORY DEFINITION(S)
SYPB SYSTEM PRESSURE BOUNDARY	5.1.2.K
IISO INSTRUMENT ISOLATION	5.4

* EQML DATA *

EQML SOURCE DOC:
FILE: EQML FE DRAWING: REG. GUIDE 1.97:
INTERFACE: EQML LOOP DOC:
FUNCTION: EQML STATUS:
IN DATE: MAINT. CATEGORY: SOUND REASONS TO CONTRARY (SRC):

* * * * * * * * * * * * * * * * COMPONENT PURCHASING DATA * * * * * * * * * * * * * * * *

MANUFACTURER: C419 Conval, Inc. MFGR SOURCE:
MODEL OR CATALOG#: - 0.75-11G35-516-3D MODEL SOURCE: -
P.O. NO.: - CNT 396432 P.O. SOURCE: -
SERIAL NO.: - SERIAL NO. SOURCE: -
STOCK NO.: STOCK NO. SOURCE:
PURCHASE SPEC.: - NAP-0023 PURCHASE SPEC. SOURCE: -
PURCHASE REQ.: - PURCHASE REQ. SOURCE: -

CODES AND STANDARDS:
-

MENT: -

REG. REQUIREMENTS: -

REF DOCUMENT: -

* * * * * * * * * * * * * * * * MECHANICAL EQUIPMENT DATA * * * * * * * * * * * * * * * *

FLOW DIAGRAM VALVE DESIGNATOR: ~~HOS-648~~ NOMINAL SIZE: .75 NOMINAL SIZE SOURCE: 01
COMPONENT DESIGN PRESSURE: 600 TEMPERATURE: - COMP. DESIGN SOURCE: 46
SYSTEM DESIGN PRESSURE: 1100 TEMPERATURE: 440 SYSTEM DESIGN SOURCE: 08
SYSTEM WORKING PRESSURE: 850 TEMPERATURE: 440 SYSTEM WORKING SOURCE: 08

* * * * * * * * * * * * * * * * ELECTRICAL EQUIPMENT DATA * * * * * * * * * * * * * * * *

POWER SOURCE: NA POWER PHASE: NA POWER SOURCE SOURCE: NA
ELECTRICAL POWER VOLTAGE: NA POWER CURRENT: NA CURRENT SOURCE: NA
ELECTRICAL CONTROL VOLTAGE: NA CONTROL CURRENT: NA CONTROL SOURCE: NA
NEMA CLASS: NA NEMA CLASS SOURCE: NA
INSULATION CLASS: NA INSULATION CLASS SOURCE: NA
FRAME SIZE: NA FRAME SIZE SOURCE: NA
THERMAL/OVERLOAD PROTECTION: NA THERMAL/OVERLOAD SOURCE: NA

* * * * * * * * * * * * * * * * * * OTHER COMPONENT DATA * * * * * * * * * * * * * * * * * *

NORMAL CAPACITY: NA UNITS: NA CAPACITY SOURCE: N
ROTATION: NA ROTATION SOURCE: NA
BUILDING: RC ELEVATION: 291 BLDG/ELEV. SOURCE:
COLUMN/LOCATION: - COL/LOC SOURCE:
ENVIRONMENTAL ZONE: - ENV. ZONE SOURCE: -
ADDL. REFERENCE: NAPS UFSAR SECTION 10.4.3 NA
NA NA
C(:

COMPONENT ID: 1-FW-104
QA CATEGORY: SR

* * * * * * * * * * * * * * * * COMPONENT IDENTIFICATION DATA * * * * * * * * * * * * * * *

STA' 59 UNIT: 01 SYSTEM: FW COMPONENT CODE: VALVE
WBS 01-FW - -104 EQ HOST: EQML: N
R APPLICABLE: N VERIFIED: Y REFERENCE DWG: 11715-FM-74A
NOMENCLATURE: VALVE, ~~GATE~~ GLOBE REVISION: 16
APPLICATION: STEAM GENERATOR 1-RC-E-1B LT-1485 I ELEC DRAWING:
SOLATION 01 TEST LOOP DIAGRAM:

* * * * * * * * * * * * * * * * QUALITY CLASSIFICATION DATA * * * * * * * * * * * * * * * *

ASSOC. COMPONENT ID: 01-FW -LT -1485 COMPONENT CODE: IXMITR

COMPONENT FUNCTION(S)		QA CATEGORY DEFINITION(S)
SYPB	SYSTEM PRESSURE BOUNDARY	5.1.2.K
IISO	INSTRUMENT ISOLATION	5.4

* EQML DATA *

EQML SOURCE DOC:
FILE: EQML FE DRAWING: REG. GUIDE 1.97:
INTERFACE: EQML LOOP DOC:
FUNCTION: EQML STATUS:
IN DATE: MAINT. CATEGORY: SOUND REASONS TO CONTRARY (SRC)

* * * * * * * * * * * * * * * * COMPONENT PURCHASING DATA * * * * * * * * * * * * * * * *

MANUFACTURER: C419 CONVAL, INC MFGR SOURCE:
MODEL OR CATALOG#: - 0.75-11G31-S16-30 MODEL SOURCE: -
P.O. NO.: - CNT 396432 P.O. SOURCE: -
SERIAL NO.: - SERIAL NO. SOURCE: -
STOCK NO.: STOCK NO. SOURCE:
PURCHASE SPEC.: - NAP-0023 PURCHASE SPEC. SOURCE: -
PURCHASE REQ.: - PURCHASE REQ. SOURCE: -

CODES AND STANDARDS:
-

DOCUMENT: -

REG. REQUIREMENTS: -

REF DOCUMENT: -

* * * * * * * * * * * * * * * * MECHANICAL EQUIPMENT DATA * * * * * * * * * * * * * * * *

FLOW DIAGRAM VALVE DESIGNATOR: ~~VGS-223~~ NOMINAL SIZE: .75 NOMINAL SIZE SOURCE: 01
COMPONENT DESIGN PRESSURE: 600 TEMPERATURE: - COMP. DESIGN SOURCE: 46
SYSTEM DESIGN PRESSURE: 1100 TEMPERATURE: 440 SYSTEM DESIGN SOURCE: 08
SYSTEM WORKING PRESSURE: 850 TEMPERATURE: 440 SYSTEM WORKING SOURCE: 08

* * * * * * * * * * * * * * * * ELECTRICAL EQUIPMENT DATA * * * * * * * * * * * * * * * *

POWER SOURCE: NA POWER PHASE: NA POWER SOURCE SOURCE: NA
ELECTRICAL POWER VOLTAGE: NA POWER CURRENT: NA CURRENT SOURCE: NA
ELECTRICAL CONTROL VOLTAGE: NA CONTROL CURRENT: NA CONTROL SOURCE: NA
NEMA CLASS: NA NEMA CLASS SOURCE: NA
INSULATION CLASS: NA INSULATION CLASS SOURCE: NA
FRAME SIZE: NA FRAME SIZE SOURCE: NA
THERMAL/OVERLOAD PROTECTION: NA THERMAL/OVERLOAD SOURCE: NA

* * * * * * * * * * * * * * * * OTHER COMPONENT DATA * * * * * * * * * * * * * * * *

NORMAL CAPACITY: NA UNITS: NA CAPACITY SOURCE: N
ROTATION: NA ROTATION SOURCE: NA
BUILDING: RC ELEVATION: 291 BLDG/ELEV. SOURCE:
COLUMN/LOCATION: - COL/LOC SOURCE:
ENVIRONMENTAL ZONE: - ENV. ZONE SOURCE: -
ADD REFERENCE: NAPS UFSAR SECTION 10.4.3 NA
NA NA
COMMENTS:

COMPONENT ID: 1-FW-105
QA CATEGORY: SR

* * * * * COMPONENT IDENTIFICATION DATA * * * * *

ST : 59 UNIT: 01 SYSTEM: FW COMPONENT CODE: VALVE
01-FW - -105 EQ HOST: EQML: N
X R APPLICABLE: N VERIFIED: Y REFERENCE DWG: 11715-FM-74A
NOMENCLATURE: VALVE, ~~GATE~~ GLOBE REVISION: 16
APPLICATION: STEAM GENERATOR 1-RC-E-1B LT-1485 ISOLATION ELEC DRAWING:
01 TEST LOOP DIAGRAM:

* * * * * QUALITY CLASSIFICATION DATA * * * * *

ASSOC. COMPONENT ID: 01-FW -LT -1485 COMPONENT CODE: IXMITR

COMPONENT FUNCTION(S)		QA CATEGORY DEFINITION(S)
SYPB	SYSTEM PRESSURE BOUNDARY	5.1.2.K
IISO	INSTRUMENT ISOLATION	5.4

* * * * * EQML DATA * * * * *

EQML SOURCE DOC:
FILE: EQML FE DRAWING: REG. GUIDE 1.97:
INTERFACE: EQML LOOP DOC:
FUNCTION: EQML STATUS:
IN DATE: MAINT. CATEGORY: SOUND REASONS TO CONTRARY (SRC):

* * * * * COMPONENT PURCHASING DATA * * * * *

MANUFACTURER: C419 CONVAL, INC MFGR SOURCE:
MODEL OR CATALOG#: - 0.75-11G34-S16-3D MODEL SOURCE: -
P.O. NO.: - CNT396432 P.O. SOURCE: -
SERIAL NO.: - SERIAL NO. SOURCE: -
STOCK NO.: STOCK NO. SOURCE:
PURCHASE SPEC.: - NAP-0023 PURCHASE SPEC. SOURCE: -
PURCHASE REQ.: - PURCHASE REQ. SOURCE: -

CODES AND STANDARDS:
-

DOCUMENT: -

REG. REQUIREMENTS: -

REF DOCUMENT: -

* * * * * MECHANICAL EQUIPMENT DATA * * * * *

FLOW DIAGRAM VALVE DESIGNATOR: ~~V03-608~~ NOMINAL SIZE: .75 NOMINAL SIZE SOURCE: 01
COMPONENT DESIGN PRESSURE: 600 TEMPERATURE: - COMP. DESIGN SOURCE: 46
SYSTEM DESIGN PRESSURE: 1100 TEMPERATURE: 440 SYSTEM DESIGN SOURCE: 08
SYSTEM WORKING PRESSURE: 850 TEMPERATURE: 440 SYSTEM WORKING SOURCE: 08

* * * * * ELECTRICAL EQUIPMENT DATA * * * * *

POWER SOURCE: NA POWER PHASE: NA POWER SOURCE SOURCE: NA
ELECTRICAL POWER VOLTAGE: NA POWER CURRENT: NA CURRENT SOURCE: NA
ELECTRICAL CONTROL VOLTAGE: NA CONTROL CURRENT: NA CONTROL SOURCE: NA
NEMA CLASS: NA NEMA CLASS SOURCE: NA
INSULATION CLASS: NA INSULATION CLASS SOURCE: NA
FRAME SIZE: NA FRAME SIZE SOURCE: NA
THERMAL/OVERLOAD PROTECTION: NA THERMAL/OVERLOAD SOURCE: NA

* * * * * OTHER COMPONENT DATA * * * * *

NORMAL CAPACITY: NA UNITS: NA CAPACITY SOURCE: N
ROTATION: NA ROTATION SOURCE: NA
BUILDING: RC ELEVATION: 291 BLDG/ELEV. SOURCE:
COLUMN/LOCATION: - COL/LOC SOURCE:
ENVIRONMENTAL ZONE: - ENV. ZONE SOURCE: -
AP REFERENCE: NAPS UFSAR SECTION 10.4.3 NA
NA NA

COMMENTS:

COMPONENT ID: 1-FW-106
QA CATEGORY: SR

* * * * * * * * * * * * * * * * * * COMPONENT IDENTIFICATION DATA * * * * * * * * * * * * * * *

STATION: 59 UNIT: 01 SYSTEM: FW COMPONENT CODE: VALVE
WP 01-FW - -106 EQ HOST: EQML: N
R APPLICABLE: N VERIFIED: Y REFERENCE DWG: 11715-FM-74A
ATURE: VALVE, ~~GATE~~ GLOBE REVISION: 16
APPLICATION: STEAM GENERATOR 1-RC-E-1B LT-1484 I ELEC DRAWING:
SOLATION 01 TEST LOOP DIAGRAM:

* * * * * * * * * * * * * * * * * QUALITY CLASSIFICATION DATA * * * * * * * * * * * * * * * * *

ASSOC. COMPONENT ID: 01-FW -LT -1484 COMPONENT CODE: IXMITR

COMPONENT FUNCTION(S) QA CATEGORY DEFINITION(S)
SYPB SYSTEM PRESSURE BOUNDARY 5.1.2.K
IISO INSTRUMENT ISOLATION 5.4

* EQML DATA *

EQML SOURCE DOC:
FILE: EQML FE DRAWING: REG. GUIDE 1.97:
INTERFACE: EQML LOOP DOC:
FUNCTION: EQML STATUS:
IN DATE: MAINT. CATEGORY: SOUND REASONS TO CONTRARY (SRC):

* * * * * * * * * * * * * * * * * COMPONENT PURCHASING DATA * * * * * * * * * * * * * * * * *

MANUFACTURER: C419 CONVAL, INC MFGR SOURCE:
MODEL OR CATALOG#: - 0.75-11G3J-S16-3D MODEL SOURCE: -
P.O. NO.: - CNT 396432 P.O. SOURCE: -
SERIAL NO.: - SERIAL NO. SOURCE: -
STOCK NO.: STOCK NO. SOURCE:
PURCHASE SPEC.: - NAP-0023 PURCHASE SPEC. SOURCE: -
PURCHASE REQ.: - PURCHASE REQ. SOURCE: -

CODES AND STANDARDS:
-

IMENT: -

REG. REQUIREMENTS: -

REF DOCUMENT: -

* * * * * * * * * * * * * * * * MECHANICAL EQUIPMENT DATA * * * * * * * * * * * * * * * * *

FLOW DIAGRAM VALVE DESIGNATOR: ~~HGS-608~~ NOMINAL SIZE: .75 NOMINAL SIZE SOURCE: 01
COMPONENT DESIGN PRESSURE: 600 TEMPERATURE: - COMP. DESIGN SOURCE: 46
SYSTEM DESIGN PRESSURE: 1100 TEMPERATURE: 440 SYSTEM DESIGN SOURCE: 08
SYSTEM WORKING PRESSURE: 850 TEMPERATURE: 440 SYSTEM WORKING SOURCE: 08

* * * * * * * * * * * * * * * * ELECTRICAL EQUIPMENT DATA * * * * * * * * * * * * * * * * *

POWER SOURCE: NA POWER PHASE: NA POWER SOURCE SOURCE: NA
ELECTRICAL POWER VOLTAGE: NA POWER CURRENT: NA CURRENT SOURCE: NA
ELECTRICAL CONTROL VOLTAGE: NA CONTROL CURRENT: NA CONTROL SOURCE: NA
NEMA CLASS: NA NEMA CLASS SOURCE: NA
INSULATION CLASS: NA INSULATION CLASS SOURCE: NA
FRAME SIZE: NA FRAME SIZE SOURCE: NA
THERMAL/OVERLOAD PROTECTION: NA THERMAL/OVERLOAD SOURCE: NA

* * * * * * * * * * * * * * * * * * OTHER COMPONENT DATA * * * * * * * * * * * * * * * * * *

NORMAL CAPACITY: NA UNITS: NA CAPACITY SOURCE: N
ROTATION: NA ROTATION SOURCE: NA
BUILDING: RC ELEVATION: 291 BLDG/ELEV. SOURCE:
COLUMN/LOCATION: - COL/LOC SOURCE:
ENVIRONMENTAL ZONE: - ENV. ZONE SOURCE: -
ADDI REFERENCE: NAPS UFSAR SECTION 10.4.3 NA
NA NA
CL :

COMPONENT ID: 1-FW-107
QA CATEGORY: SR

* * * * * * * * * * * * * * * * * COMPONENT IDENTIFICATION DATA * * * * * * * * * * * * * * *

ST : 59 UNIT: 01 SYSTEM: FW COMPONENT CODE: VALVE
01-FW - -107 EQ HOST: EQML: N
X R APPLICABLE: N VERIFIED: Y REFERENCE DWG: 11715-FM-74A
NOMENCLATURE: VALVE, ~~GATE~~ GLOBE REVISION: 16
APPLICATION: STEAM GENERATOR 1-RC-E-1B LT-1484 ISOLATION ELEC DRAWING:
01 TEST LOOP DIAGRAM:

* * * * * * * * * * * * * * * * * QUALITY CLASSIFICATION DATA * * * * * * * * * * * * * * * * *

ASSOC. COMPONENT ID: 01-FW -LT -1484 COMPONENT CODE: IXMITR

COMPONENT FUNCTION(S)	QA CATEGORY DEFINITION(S)
SYPB SYSTEM PRESSURE BOUNDARY	5.1.2.K
IISO INSTRUMENT ISOLATION	5.4

* EQML DATA *

EQML SOURCE DOC:
FILE: EQML FE DRAWING: REG. GUIDE 1.97:
INTERFACE: EQML LOOP DOC:
FUNCTION: EQML STATUS:
IN DATE: MAINT. CATEGORY: SOUND REASONS TO CONTRARY (SRC):

* * * * * * * * * * * * * * * * * COMPONENT PURCHASING DATA * * * * * * * * * * * * * * * * *

MANUFACTURER: C419 CONVAL, INC MFGR SOURCE:
MODEL OR CATALOG#: - 0.75-11G32-S16-3D MODEL SOURCE: -
P.O. NO.: - P.O. SOURCE: -
SERIAL NO.: - CNT396432 SERIAL NO. SOURCE: -
STOCK NO.: STOCK NO. SOURCE:
PURCHASE SPEC.: - NAP-0023 PURCHASE SPEC. SOURCE: -
PURCHASE REQ.: - PURCHASE REQ. SOURCE: -

CODES AND STANDARDS:
-

DOCUMENT: -

REG. REQUIREMENTS: -

REF DOCUMENT: -

* * * * * * * * * * * * * * * * * MECHANICAL EQUIPMENT DATA * * * * * * * * * * * * * * * * *

FLOW DIAGRAM VALVE DESIGNATOR: ~~VGS-60B~~ NOMINAL SIZE: .75 NOMINAL SIZE SOURCE: 01
COMPONENT DESIGN PRESSURE: 600 TEMPERATURE: - COMP. DESIGN SOURCE: 46
SYSTEM DESIGN PRESSURE: 1100 TEMPERATURE: 440 SYSTEM DESIGN SOURCE: 08
SYSTEM WORKING PRESSURE: 850 TEMPERATURE: 440 SYSTEM WORKING SOURCE: 08

* * * * * * * * * * * * * * * * * ELECTRICAL EQUIPMENT DATA * * * * * * * * * * * * * * * * *

POWER SOURCE: NA POWER PHASE: NA POWER SOURCE SOURCE: NA
ELECTRICAL POWER VOLTAGE NA POWER CURRENT: NA CURRENT SOURCE: NA
ELECTRICAL CONTROL VOLTAGE: NA CONTROL CURRENT: NA CONTROL SOURCE: NA
NEMA CLASS: NA NEMA CLASS SOURCE: NA
INSULATION CLASS: NA INSULATION CLASS SOURCE: NA
FRAME SIZE: NA FRAME SIZE SOURCE: NA
THERMAL/OVERLOAD PROTECTION: NA THERMAL/OVERLOAD SOURCE: NA

* * * * * * * * * * * * * * * * * * OTHER COMPONENT DATA * * * * * * * * * * * * * * * * * * *

NORMAL CAPACITY: NA UNITS: NA CAPACITY SOURCE: N
ROTATION: NA ROTATION SOURCE: NA
BUILDING: RC ELEVATION: 291 BLDG/ELEV. SOURCE:
COLUMN/LOCATION: - COL/LOC SOURCE:
ENVIRONMENTAL ZONE: - ENV. ZONE SOURCE: -
AI REFERENCE: NAPS UFSAR SECTION 10.4.3 NA
NA NA

COMMENTS:

END OF REPORT

COMPONENT ID: 1-FW-108
QA CATEGORY: SR

* * * * * * * * * * * * * * * * * COMPONENT IDENTIFICATION DATA * * * * * * * * * * * * * * * *

STATI... 59 UNIT: 01 SYSTEM: FW COMPONENT CODE: VALVE
WPT... 01-FW - -108 EQ HOST: EQML: N
... R APPLICABLE: ~~N~~ VERIFIED: Y REFERENCE DWG: 11715-FM-74A
... TURE: VALVE, ~~GATE~~ GLOBE REVISION: 18
APPLICATION: STEAM GENERATOR 1-RC-E-1B LT-1487 ISOLATION ELEC DRAWING:
01 TEST LOOP DIAGRAM:

* * * * * * * * * * * * * * * * * QUALITY CLASSIFICATION DATA * * * * * * * * * * * * * * * * *

ASSOC. COMPONENT ID: 01-FW -LT -1487 COMPONENT CODE: IXMITR

COMPONENT FUNCTION(S)		QA CATEGORY DEFINITION(S)
SYPB	SYSTEM PRESSURE BOUNDARY	5.1.2.K
IISO	INSTRUMENT ISOLATION	5.4

* EQML DATA *

EQML SOURCE DOC:
FILE: EQML FE DRAWING: REG. GUIDE 1.97:
INTERFACE: EQML LOOP DOC:
FUNCTION: EQML STATUS:
IN DATE: MAINT. CATEGORY: SOUND REASONS TO CONTRARY (SRC):

* * * * * * * * * * * * * * * * * COMPONENT PURCHASING DATA * * * * * * * * * * * * * * * * *

MANUFACTURER: C419 CONVAL, INC MFGR SOURCE:
MODEL OR CATALOG#: - 0.75-11G3J-316-3D MODEL SOURCE: -
P.O. NO.: - CNT396432 P.O. SOURCE: -
SERIAL NO.: - SERIAL NO. SOURCE: -
STOCK NO.: STOCK NO. SOURCE:
PURCHASE SPEC.: - NAP-C023 PURCHASE SPEC. SOURCE: -
PURCHASE REQ.: - PURCHASE REQ. SOURCE: -

CODES AND STANDARDS:

R... ...MENT: -

REG. REQUIREMENTS: -

REF DOCUMENT: -

* * * * * * * * * * * * * * * * * MECHANICAL EQUIPMENT DATA * * * * * * * * * * * * * * * * *

FLOW DIAGRAM VALVE DESIGNATOR: ~~VGS-600~~ NOMINAL SIZE: .75 NOMINAL SIZE SOURCE: 01
COMPONENT DESIGN PRESSURE: 600 TEMPERATURE: - COMP. DESIGN SOURCE: 46
SYSTEM DESIGN PRESSURE: 1100 TEMPERATURE: 440 SYSTEM DESIGN SOURCE: 08
SYSTEM WORKING PRESSURE: 850 TEMPERATURE: 440 SYSTEM WORKING SOURCE: 08

* * * * * * * * * * * * * * * * * ELECTRICAL EQUIPMENT DATA * * * * * * * * * * * * * * * * *

POWER SOURCE: NA POWER PHASE: NA POWER SOURCE SOURCE: NA
ELECTRICAL POWER VOLTAGE: NA POWER CURRENT: NA CURRENT SOURCE: NA
ELECTRICAL CONTROL VOLTAGE: NA CONTROL CURRENT: NA CONTROL SOURCE: NA
NEMA CLASS: NA NEMA CLASS SOURCE: NA
INSULATION CLASS: NA INSULATION CLASS SOURCE: NA
FRAME SIZE: NA FRAME SIZE SOURCE: NA
THERMAL/OVERLOAD PROTECTION: NA THERMAL/OVERLOAD SOURCE: NA

* * * * * * * * * * * * * * * * * * OTHER COMPONENT DATA * * * * * * * * * * * * * * * * * * *

NORMAL CAPACITY: NA UNITS: NA CAPACITY SOURCE: N
ROTATION: NA ROTATION SOURCE: NA
BUILDING: RC ELEVATION: 291 BLDG/ELEV. SOURCE:
COLUMN/LOCATION: - COL/LOC SOURCE:
ENVIRONMENTAL ZONE: - ENV. ZONE SOURCE: -
ADDL REFERENCE: NAPS UFSAR SECTION 10.4.3 NA
NA NA

CO... :

COMPONENT ID: 1-FW-109
QA CATEGORY: SR

* * * * * * * * * * * * * * * * COMPONENT IDENTIFICATION DATA * * * * * * * * * * * * * * * *

STATION: 59 UNIT: 01 SYSTEM: FW COMPONENT CODE: VALVE
WPT 01-FW - -109 EQ HOST: EQML: N
R APPLICABLE: N VERIFIED: Y REFERENCE DWG: 11715-FM-74A
ATURE: VALVE, ~~GATE~~ GLOBE REVISION: 16
APPLICATION: STEAM GENERATOR 1 RC-E-1B LT-1487 I ELEC DRAWING:
SOLATION 01 TEST LOOP DIAGRAM:

* * * * * * * * * * * * * * * * QUALITY CLASSIFICATION DATA * * * * * * * * * * * * * * * *

ASSOC. COMPONENT ID: 01-FW -LT -1487 COMPONENT CODE: IXMITR

	COMPONENT FUNCTION(S)	QA CATEGORY DEFINITION(S)
SYPB	SYSTEM PRESSURE BOUNDARY	5.1.2.K
IISO	INSTRUMENT ISOLATION	5.4

* * * * * * * * * * * * * * * * * * * EQML DATA * * * * * * * * * * * * * * * * * * *

EQML SOURCE DOC:
FILE: EQML FE DRAWING: REG. GUIDE 1.97:
INTERFACE: EQML LOOP DOC:
FUNCTION: EQML STATUS:
IN DATE: MAINT. CATEGORY: SOUND REASONS TO CONTRARY (SRC):

* * * * * * * * * * * * * * * * COMPONENT PURCHASING DATA * * * * * * * * * * * * * * * *

MANUFACTURER: C419 CONVAL, INC MFGR SOURCE:
MODEL OR CATALOG#: - 0.75-11GJ-S16-3D MODEL SOURCE: -
P.O. NO.: - P.O. SOURCE: -
SERIAL NO.: - CNT396432 SERIAL NO. SOURCE: -
STOCK NO.: STOCK NO. SOURCE:
PURCHASE SPEC.: - PURCHASE SPEC. SOURCE: -
PURCHASE REQ.: - NAP-0023 PURCHASE REQ. SOURCE: -

CODES AND STANDARDS:
-

JMENT: -

REG. REQUIREMENTS: -

REF DOCUMENT: -

* * * * * * * * * * * * * * * * MECHANICAL EQUIPMENT DATA * * * * * * * * * * * * * * * *

FLOW DIAGRAM VALVE DESIGNATOR: WGS-60B NOMINAL SIZE: .75 NOMINAL SIZE SOURCE: 01
COMPONENT DESIGN PRESSURE: 600 TEMPERATURE: - COMP. DESIGN SOURCE: 46
SYSTEM DESIGN PRESSURE: 1100 TEMPERATURE: 440 SYSTEM DESIGN SOURCE: 08
SYSTEM WORKING PRESSURE: 850 TEMPERATURE: 440 SYSTEM WORKING SOURCE: 08

* * * * * * * * * * * * * * * * ELECTRICAL EQUIPMENT DATA * * * * * * * * * * * * * * * *

POWER SOURCE: NA POWER PHASE: NA POWER SOURCE SOURCE: NA
ELECTRICAL POWER VOLTAGE: NA POWER CURRENT: NA CURRENT SOURCE: NA
ELECTRICAL CONTROL VOLTAGE: NA CONTROL CURRENT: NA CONTROL SOURCE: NA
NEMA CLASS: NA NEMA CLASS SOURCE: NA
INSULATION CLASS: NA INSULATION CLASS SOURCE: NA
FRAME SIZE: NA FRAME SIZE SOURCE: NA
THERMAL/OVERLOAD PROTECTION: NA THERMAL/OVERLOAD SOURCE: NA

* * * * * * * * * * * * * * * * OTHER COMPONENT DATA * * * * * * * * * * * * * * * *

NORMAL CAPACITY: NA UNITS: NA CAPACITY SOURCE: N
ROTATION: NA ROTATION SOURCE: NA
BUILDING: RC ELEVATION: 262 BLDG/ELEV. SOURCE:
COLUMN/LOCATION: - COL/LOC SOURCE:
ENVIRONMENTAL ZONE: - ENV. ZONE SOURCE: -
ADDI REFERENCE: NAPS UFSAR SECTION 10.4.3 NA
NA NA
CC S:

COMPONENT ID: 1-FW-134
QA CATEGORY: SR

* * * * * * * * * * * * * * * * * COMPONENT IDENTIFICATION DATA * * * * * * * * * * * * * * *

STAT**N: 59 UNIT: 01 SYSTEM: FW COMPONENT CODE: VALVE
WP' 01-FW - -134 EQ HOST: EQML: N
R APPLICABLE: N VERIFIED: Y REFERENCE DWG: 11715-FM-74A
..ATURE: VALVE, ~~GATE~~ GLOBE REVISION: 16
APPLICATION: STEAM GENERATOR 1-RC-E-1C LT-1496 I ELEC DRAWING:
SOLATION 01 TEST LOOP DIAGRAM:

* * * * * * * * * * * * * * * * * QUALITY CLASSIFICATION DATA * * * * * * * * * * * * * * * *

ASSOC. COMPONENT ID: 01-FW -LT -1496 COMPONENT CODE: IXMITR

COMPONENT FUNCTION(S) QA CATEGORY DEFINITION(S)
SYPB SYSTEM PRESSURE BOUNDARY 5.1.2.K
IISO INSTRUMENT ISOLATION 5.4

* EQML DATA *

EQML SOURCE DOC:
FILE: EQML FE DRAWING: REG. GUIDE 1.97:
INTERFACE: EQML LOOP DOC:
FUNCTION: EQML STATUS:
IN DATE: MAINT. CATEGORY: SOUND REASONS TO CONTRARY (SRC):

* * * * * * * * * * * * * * * * COMPONENT PURCHASING DATA * * * * * * * * * * * * * * * * *

MANUFACTURER: ~~V010 VOGT, HENRY, MACH. CO., INC.~~ C419 CONVAL, INC MFGR SOURCE: 32
MODEL OR CATALOG#: - 0.75-11G3J-S16-3D MODEL SOURCE: -
P.O. NO.: - P.O. SOURCE: -
SERIAL NO.: - CNT 396432 SERIAL NO. SOURCE: -
STOCK NO.: STOCK NO. SOURCE:
PURCHASE SPEC.: - PURCHASE SPEC. SOURCE: -
PURCHASE REQ.: - NAP-0023 PURCHASE REQ. SOURCE: -

CODES AND STANDARDS:
-

.UMENT: -

REG. REQUIREMENTS: -

REF DOCUMENT: -

* * * * * * * * * * * * * * * * MECHANICAL EQUIPMENT DATA * * * * * * * * * * * * * * * * *

FLOW DIAGRAM VALVE DESIGNATOR: ~~V03-60B~~ NOMINAL SIZE: .75 NOMINAL SIZE SOURCE: 01
COMPONENT DESIGN PRESSURE: 600 TEMPERATURE: - COMP. DESIGN SOURCE: 46
SYSTEM DESIGN PRESSURE: - TEMPERATURE: - SYSTEM DESIGN SOURCE: -
SYSTEM WORKING PRESSURE: - TEMPERATURE: - SYSTEM WORKING SOURCE: -

* * * * * * * * * * * * * * * * ELECTRICAL EQUIPMENT DATA * * * * * * * * * * * * * * * * *

POWER SOURCE: NA POWER PHASE: NA POWER SOURCE SOURCE: NA
ELECTRICAL POWER VOLTAGE: NA POWER CURRENT: NA CURRENT SOURCE: NA
ELECTRICAL CONTROL VOLTAGE: NA CONTROL CURRENT: NA CONTROL SOURCE: NA
NEMA CLASS: NA NEMA CLASS SOURCE: NA
INSULATION CLASS: NA INSULATION CLASS SOURCE: NA
FRAME SIZE: NA FRAME SIZE SOURCE: NA
THERMAL/OVERLOAD PROTECTION: NA THERMAL/OVERLOAD SOURCE: NA

* * * * * * * * * * * * * * * * * * OTHER COMPONENT DATA * * * * * * * * * * * * * * * * * *

NORMAL CAPACITY: NA UNITS: NA CAPACITY SOURCE: N
ROTATION: NA ROTATION SOURCE: NA
BUILDING: RC ELEVATION: 291 BLDG/ELEV. SOURCE:
COLUMN/LOCATION: - COL/LOC SOURCE:
ENVIRONMENTAL ZONE: - ENV. ZONE SOURCE: -
ADD' REFERENCE: NAPS UFSAR SECTION 10.4.3 NA
NA NA
CL S:

COMPONENT ID: 1-FW-135
QA CATEGORY: SR

* * * * * * * * * * * * * * * * * COMPONENT IDENTIFICATION DATA * * * * * * * * * * * * * * *

STA' 59 UNIT: 01 SYSTEM: FW COMPONENT CODE: VALVE
01-FW - -135 EQ HOST: EQML: N
R APPLICABLE: N VERIFIED: Y REFERENCE DWG: 11715-FM-74A
NOMENCLATURE: VALVE, ~~GATE~~ GLOBE REVISION: 16
APPLICATION: STEAM GENERATOR 1-RC-E-1C LT-1496 I ELEC DRAWING:
SOLATION 01 TEST LOOP DIAGRAM:

* * * * * * * * * * * * * * * * QUALITY CLASSIFICATION DATA * * * * * * * * * * * * * * * *

ASSOC. COMPONENT ID: 01-FW -LT -1496 COMPONENT CODE: IXMITR

COMPONENT FUNCTION(S)		QA CATEGORY DEFINITION(S)
SYPB	SYSTEM PRESSURE BOUNDARY	5.1.2.K
IISD	INSTRUMENT ISOLATION	5.4

* EQML DATA *

EQML SOURCE DOC:
FILE: EQML FE DRAWING: REG. GUIDE 1.97:
INTERFACE: EQML LOOP DOC:
FUNCTION: EQML STATUS:
IN DATE: MAINT. CATEGORY: SOUND REASONS TO CONTRARY (SRC):

* * * * * * * * * * * * * * * * COMPONENT PURCHASING DATA * * * * * * * * * * * * * * * *

MANUFACTURER: C419 CONVAL, INC MFGR SOURCE:
MODEL OR CATALOG#: - 0.75-11G3J-S16-3D MODEL SOURCE: -
P.O. NO.: - CNT396432 P.O. SOURCE: -
SERIAL NO.: - SERIAL NO. SOURCE: -
STOCK NO.: STOCK NO. SOURCE:
PURCHASE SPEC.: - NAP-0023 PURCHASE SPEC. SOURCE: -
PURCHASE REQ.: - PURCHASE REQ. SOURCE: -

CODES AND STANDARDS:
-

REG DOCUMENT: -

REG. REQUIREMENTS: -

REF DOCUMENT: -

* * * * * * * * * * * * * * * * MECHANICAL EQUIPMENT DATA * * * * * * * * * * * * * * * *

FLOW DIAGRAM VALVE DESIGNATOR: ~~YB9-60B~~ NOMINAL SIZE: .75 NOMINAL SIZE SOURCE: 01
COMPONENT DESIGN PRESSURE: 600 TEMPERATURE: - COMP. DESIGN SOURCE: 46
SYSTEM DESIGN PRESSURE: - TEMPERATURE: - SYSTEM DESIGN SOURCE: -
SYSTEM WORKING PRESSURE: - TEMPERATURE: - SYSTEM WORKING SOURCE: -

* * * * * * * * * * * * * * * * ELECTRICAL EQUIPMENT DATA * * * * * * * * * * * * * * * *

POWER SOURCE: NA POWER PHASE: NA POWER SOURCE SOURCE: NA
ELECTRICAL POWER VOLTAGE: NA POWER CURRENT: NA CURRENT SOURCE: NA
ELECTRICAL CONTROL VOLTAGE: NA CONTROL CURRENT: NA CONTROL SOURCE: NA
NEMA CLASS: NA NEMA CLASS SOURCE: NA
INSULATION CLASS: NA INSULATION CLASS SOURCE: NA
FRAME SIZE: NA FRAME SIZE SOURCE: NA
THERMAL/OVERLOAD PROTECTION: NA THERMAL/OVERLOAD SOURCE: NA

* * * * * * * * * * * * * * * * * OTHER COMPONENT DATA * * * * * * * * * * * * * * * * *

NORMAL CAPACITY: NA UNITS: NA CAPACITY SOURCE: N
ROTATION: NA ROTATION SOURCE: NA
BUILDING: RC ELEVATION: 291 BLDG/ELEV. SOURCE:
COLUMN/LOCATION: - COL/LOC SOURCE:
ENVIRONMENTAL ZONE: - ENV. ZONE SOURCE: -
ADD REFERENCE: NAPS UFSAR SECTION 10.4.3 NA
NA NA
COMMENTS:

COMPONENT ID: 1-FW-136
QA CATEGORY: SR

* * * * * * * * * * * * * * * * * COMPONENT IDENTIFICATION DATA * * * * * * * * * * * * * * *

STAT...: 59 UNIT: 01 SYSTEM: FW COMPONENT CODE: VALVE
WP... 01-FW - -136 EQ HOST: EQML: N
... R APPLICABLE: N VERIFIED: Y REFERENCE DWG: 11715-FM-74A
...ATURE: VALVE, ~~GATE~~ GLOBE REVISION: 16
APPLICATION: STEAM GENERATOR 1-RC-E-1C LT-1495 I ELEC DRAWING:
SOLATION 01 TEST LOOP DIAGRAM:

* * * * * * * * * * * * * * * * * QUALITY CLASSIFICATION DATA * * * * * * * * * * * * * * * *

ASSOC. COMPONENT ID: 01-FW -LT -1495 COMPONENT CODE: IXMITR

	COMPONENT FUNCTION(S)	QA CATEGORY DEFINITION(S)
SYPB	SYSTEM PRESSURE BOUNDARY	5.1.2.K
IISO	INSTRUMENT ISOLATION	5.4

* EQML DATA *

EQML SOURCE DOC:
FILE: EQML FE DRAWING: REG. GUIDE 1.97:
INTERFACE: EQML LOOP DOC:
FUNCTION: EQML STATUS:
IN DATE: MAINT. CATEGORY: SOUND REASONS TO CONTRARY (SRC):

* * * * * * * * * * * * * * * * * COMPONENT PURCHASING DATA * * * * * * * * * * * * * * * * *

MANUFACTURER: ~~V010 VOGT, HENRY, MACH. CO., INC.~~ C419 CONVAL, INC. MFGR SOURCE: 32
MODEL OR CATALOG: - 0.75-11G3J-S16-3D MODEL SOURCE: -
P.O. NO.: - P.O. SOURCE: -
SERIAL NO.: - CNT396432 SERIAL NO. SOURCE: -
STOCK NO.: STOCK NO. SOURCE:
PURCHASE SPEC.: - PURCHASE SPEC. SOURCE: -
PURCHASE REQ.: - NAP-0023 PURCHASE REQ. SOURCE: -

CODES AND STANDARDS:
-

...JMENT: -

REG. REQUIREMENTS: -

REF DOCUMENT: -

* * * * * * * * * * * * * * * * * MECHANICAL EQUIPMENT DATA * * * * * * * * * * * * * * * * *

FLOW DIAGRAM VALVE DESIGNATOR: ~~VGS-608~~ NOMINAL SIZE: .75 NOMINAL SIZE SOURCE: 01
COMPONENT DESIGN PRESSURE: 600 TEMPERATURE: - COMP. DESIGN SOURCE: 46
SYSTEM DESIGN PRESSURE: - TEMPERATURE: - SYSTEM DESIGN SOURCE: -
SYSTEM WORKING PRESSURE: - TEMPERATURE: - SYSTEM WORKING SOURCE: -

* * * * * * * * * * * * * * * * * ELECTRICAL EQUIPMENT DATA * * * * * * * * * * * * * * * * *

POWER SOURCE: NA POWER PHASE: NA POWER SOURCE SOURCE: NA
ELECTRICAL POWER VOLTAGE: NA POWER CURRENT: NA CURRENT SOURCE: NA
ELECTRICAL CONTROL VOLTAGE: NA CONTROL CURRENT: NA CONTROL SOURCE: NA
NEMA CLASS: NA NEMA CLASS SOURCE: NA
INSULATION CLASS: NA INSULATION CLASS SOURCE: NA
FRAME SIZE: NA FRAME SIZE SOURCE: NA
THERMAL/OVERLOAD PROTECTION: NA THERMAL/OVERLOAD SOURCE: NA

* * * * * * * * * * * * * * * * * OTHER COMPONENT DATA * * * * * * * * * * * * * * * * * *

NORMAL CAPACITY: NA UNITS: NA CAPACITY SOURCE: N
ROTATION: NA ROTATION SOURCE: NA
BUILDING: RC ELEVATION: 291 BLDG/ELEV. SOURCE:
COLUMN/LOCATION: - COL/LOC SOURCE:
ENVIRONMENTAL ZONE: - ENV. ZONE SOURCE: -
ADD... REFERENCE: NAPS UFSAR SECTION 10.4.3 NA
NA NA
C... S:

COMPONENT ID: 1-FW-137
QA CATEGORY: SR

* * * * * * * * * * * * * * * * * COMPONENT IDENTIFICATION DATA * * * * * * * * * * * * * * *

ST N: 59 UNIT: 01 SYSTEM: FW COMPONENT CODE: VALVE
: 01-FW - -137 EQ HOST: EQML: N
X R APPLICABLE: N VERIFIED: Y REFERENCE DWG: 11715-FM-74A
NOMENCLATURE: VALVE, ~~GATE~~ GLOBE REVISION: 16
APPLICATION: STEAM GENERATOR 1-RC-E-1C LT-1495 I ELEC DRAWING:
SOLATION 01 TEST LOOP DIAGRAM:

* * * * * * * * * * * * * * * * * QUALITY CLASSIFICATION DATA * * * * * * * * * * * * * * * * *

ASSOC. COMPONENT ID: 01-FW -LT -1495 COMPONENT CODE: IXMITR

COMPONENT FUNCTION(S)		QA CATEGORY DEFINITION(S)
SYPB	SYSTEM PRESSURE BOUNDARY	5.1.2.K
IISO	INSTRUMENT ISOLATION	5.4

* EQML DATA *

EQML SOURCE DOC:
FILE: EQML FE DRAWING: REG. GUIDE 1.97:
INTERFACE: EQML LOOP DOC:
FUNCTION: EQML STATUS:
IN DATE: MAINT. CATEGORY: SOUND REASONS TO CONTRARY (SRC):

* * * * * * * * * * * * * * * * * COMPONENT PURCHASING DATA * * * * * * * * * * * * * * * * *

MANUFACTURER: C419 CONVAL, INC MFGR SOURCE:
MODEL OR CATALOG#: - 0-75-1163J-S16-9D MODEL SOURCE: -
P.O. NO.: - P.O. SOURCE: -
SERIAL NO.: - CNT396432 SERIAL NO. SOURCE: -
STOCK NO.: STOCK NO. SOURCE:
PURCHASE SPEC.: - NAP-0023 PURCHASE SPEC. SOURCE: -
PURCHASE REQ.: - PURCHASE REQ. SOURCE: -

CODES AND STANDARDS:

DOCUMENT: -

REG. REQUIREMENTS: -

REF DOCUMENT: -

* * * * * * * * * * * * * * * * * MECHANICAL EQUIPMENT DATA * * * * * * * * * * * * * * * * *

FLOW DIAGRAM VALVE DESIGNATOR: ~~VOS-608~~ NOMINAL SIZE: .75 NOMINAL SIZE SOURCE: 01
COMPONENT DESIGN PRESSURE: 600 TEMPERATURE: - COMP. DESIGN SOURCE: 46
SYSTEM DESIGN PRESSURE: - TEMPERATURE: - SYSTEM DESIGN SOURCE: -
SYSTEM WORKING PRESSURE: - TEMPERATURE: - SYSTEM WORKING SOURCE: -

* * * * * * * * * * * * * * * * * ELECTRICAL EQUIPMENT DATA * * * * * * * * * * * * * * * * *

POWER SOURCE: NA POWER PHASE: NA POWER SOURCE SOURCE: NA
ELECTRICAL POWER VOLTAGE: NA POWER CURRENT: NA CURRENT SOURCE: NA
ELECTRICAL CONTROL VOLTAGE: NA CONTROL CURRENT: NA CONTROL SOURCE: NA
NEMA CLASS: NA NEMA CLASS SOURCE: NA
INSULATION CLASS: NA INSULATION CLASS SOURCE: NA
FRAME SIZE: NA FRAME SIZE SOURCE: NA
THERMAL/OVERLOAD PROTECTION: NA THERMAL/OVERLOAD SOURCE: NA

* * * * * * * * * * * * * * * * * * OTHER COMPONENT DATA * * * * * * * * * * * * * * * * * * *

NORMAL CAPACITY: NA UNITS: NA CAPACITY SOURCE: N
ROTATION: NA ROTATION SOURCE: NA
BUILDING: RC ELEVATION: 291 BLDG/ELEV. SOURCE:
COLUMN/LOCATION: - COL/LOC SOURCE:
ENVIRONMENTAL ZONE: - ENV. ZONE SOURCE: -
AP REFERENCE: NAPS UFSAR SECTION 10.4.3 NA
NA NA
C TS:
OF REPORT

COMPONENT ID: 1-FW-138
QA CATEGORY: SR

* * * * * * * * * * * * * * * * * COMPONENT IDENTIFICATION DATA * * * * * * * * * * * * * * *

STATION: 59 UNIT: 01 SYSTEM: FW COMPONENT CODE: VALVE
WP 01-FW - -138 EQ HOST: EQML: N
R APPLICABLE: N VERIFIED: Y REFERENCE DWG: 11715-FM-74A
ATURE: VALVE, ~~GATE~~ GLOBE REVISION: 16
LICATION: STEAM GENERATOR 1-RC-E-1C LT-1494 I ELEC DRAWING:
SOLATION 01 TEST LOOP DIAGRAM:

* * * * * * * * * * * * * * * QUALITY CLASSIFICATION DATA * * * * * * * * * * * * * * * *

ASSOC. COMPONENT ID: 01-FW -LT -1494 COMPONENT CODE: IXMITR

COMPONENT FUNCTION(S)	QA CATEGORY DEFINITION(S)
SYPB SYSTEM PRESSURE BOUNDARY	5.1.2.K
IISO INSTRUMENT ISOLATION	5.4

* * * * * * * * * * * * * * * * * * EQML DATA *

EQML SOURCE DOC:
FILE: EQML FE DRAWING: REG. GUIDE 1.97:
INTERFACE: EQML LOOP DOC:
FUNCTION: EQML STATUS:
IN DATE: MAINT. CATEGORY: SOUND REASONS TO CONTRARY (SRC):

* * * * * * * * * * * * * * * * COMPONENT PURCHASING DATA * * * * * * * * * * * * * * * *

MANUFACTURER: C419 CONVAL, INC MFGR SOURCE:
MODEL OR CATALOG#: - 0.75-1163J-S1630 MODEL SOURCE: -
P.O. NO.: - CNT 396432 P.O. SOURCE: -
SERIAL NO.: - SERIAL NO. SOURCE: -
STOCK NO.: STOCK NO. SOURCE:
PURCHASE SPEC.: - NAP-0023 PURCHASE SPEC. SOURCE: -
PURCHASE REQ.: - PURCHASE REQ. SOURCE: -

CODES AND STANDARDS:
-

MENT: -

REG. REQUIREMENTS: -

REF DOCUMENT: -

* * * * * * * * * * * * * * * * MECHANICAL EQUIPMENT DATA * * * * * * * * * * * * * * * *

FLOW DIAGRAM VALVE DESIGNATOR: VGS-608 NOMINAL SIZE: .75 NOMINAL SIZE SOURCE: 81
COMPONENT DESIGN PRESSURE: 603 TEMPERATURE: - COMP. DESIGN SOURCE: 46
SYSTEM DESIGN PRESSURE: - TEMPERATURE: - SYSTEM DESIGN SOURCE: -
SYSTEM WORKING PRESSURE: - TEMPERATURE: - SYSTEM WORKING SOURCE: -

* * * * * * * * * * * * * * * * ELECTRICAL EQUIPMENT DATA * * * * * * * * * * * * * * * *

POWER SOURCE: NA POWER PHASE: NA POWER SOURCE SOURCE: NA
ELECTRICAL POWER VOLTAGE: NA POWER CURRENT: NA CURRENT SOURCE: NA
ELECTRICAL CONTROL VOLTAGE: NA CONTROL CURRENT: NA CONTROL SOURCE: NA
NEMA CLASS: NA NEMA CLASS SOURCE: NA
INSULATION CLASS: NA INSULATION CLASS SOURCE: NA
FRAME SIZE: NA FRAME SIZE SOURCE: NA
THERMAL/OVERLOAD PROTECTION: NA THERMAL/OVERLOAD SOURCE: NA

* * * * * * * * * * * * * * * * * OTHER COMPONENT DATA * * * * * * * * * * * * * * * * *

NORMAL CAPACITY: NA UNITS: NA CAPACITY SOURCE: N
ROTATION: NA ROTATION SOURCE: NA
BUILDING: RC ELEVATION: 291 BLDG/ELEV. SOURCE:
COLUMN/LOCATION: - COL/LOC SOURCE:
ENVIRONMENTAL ZONE: - ENV. ZONE SOURCE: -
ADDL. REFERENCE: NAPS UFSAR SECTION 10.4.3 NA
NA NA
C( ):

COMPONENT ID: 1-FW-139
QA CATEGORY: SR

* * * * * * * * * * * * * * * * * COMPONENT IDENTIFICATION DATA * * * * * * * * * * * * * * *

ST : 59 UNIT: 01 SYSTEM: FW COMPONENT CODE: VALVE
01-FW - -139 EQ HOST: EQML: N
X R APPLICABLE: N VERIFIED: Y REFERENCE DWG: 11715 4A
ENCLATURE: VALVE, ~~GATE~~ GLOBE REVISION: 18
APPLICATION: STEAM GENERATOR 1-RC-E-1C LT-1494 I ELEC DRAWING:
SOLATION 01 TEST LOOP DIAGRAM:

* * * * * * * * * * * * * * * * * QUALITY CLASSIFICATION DATA * * * * * * * * * * * * * * * * *

ASSOC. COMPONENT ID: 01-FW -LT -1494 COMPONENT CODE: IXMITR

COMPONENT FUNCTION(S)		QA CATEGORY DEFINITION(S)
SYPB	SYSTEM PRESSURE BOUNDARY	5.1.2.K
IISO	INSTRUMENT ISOLATION	5.4

* EQML DATA *

EQML SOURCE DOC:
FILE: EQML FE DRAWING: REG. GUIDE 1.97:
INTERFACE: EQML LOOP DOC:
FUNCTION: EQML STATUS:
IN DATE: MAINT. CATEGORY: SOUND REASONS TO CONTRARY (SRC):

* * * * * * * * * * * * * * * * * COMPONENT PURCHASING DATA * * * * * * * * * * * * * * * * *

MANUFACTURER: ~~V010 VOGT, HENRY, MACH. CO., INC.~~ C419 CONVAL, INC MFGR SOURCE: 32
MODEL OR CATALOG#: - 0.75-11G3J-S16-3D MODEL SOURCE: -
P.O. NO.: - P.O. SOURCE: -
SERIAL NO.: - CNT396432 SERIAL NO. SOURCE: -
STOCK NO.: STOCK NO. SOURCE:
PURCHASE SPEC.: - NAP 0023 PURCHASE SPEC. SOURCE: -
PURCHASE REQ.: - PURCHASE REQ. SOURCE: -

CO ND STANDARDS:
-

DOCUMENT: -

REG. REQUIREMENTS: -

REF DOCUMENT: -

* * * * * * * * * * * * * * * * * MECHANICAL EQUIPMENT DATA * * * * * * * * * * * * * * * *

FLOW DIAGRAM VALVE DESIGNATOR: ~~V05-600~~ NOMINAL SIZE: .75 NOMINAL SIZE SOURCE: 01
COMPONENT DESIGN PRESSURE: 600 TEMPERATURE: - COMP. DESIGN SOURCE: 46
SYSTEM DESIGN PRESSURE: - TEMPERATURE: - SYSTEM DESIGN SOURCE: -
SYSTEM WORKING PRESSURE: - TEMPERATURE: - SYSTEM WORKING SOURCE: -

* * * * * * * * * * * * * * * * * ELECTRICAL EQUIPMENT DATA * * * * * * * * * * * * * * * *

POWER SOURCE: NA POWER PHASE: NA POWER SOURCE SOURCE: NA
ELECTRICAL POWER VOLTAGE: NA POWER CURRENT: NA CURRENT SOURCE: NA
ELECTRICAL CONTROL VOLTAGE: NA CONTROL CURRENT: NA CONTROL SOURCE: NA
NEMA CLASS: NA NEMA CLASS SOURCE: NA
INSULATION CLASS: NA INSULATION CLASS SOURCE: NA
FRAME SIZE: NA FRAME SIZE SOURCE: NA
THERMAL/OVERLOAD PROTECTION: NA THERMAL/OVERLOAD SOURCE: NA

* * * * * * * * * * * * * * * * * OTHER COMPONENT DATA * * * * * * * * * * * * * * * * * *

NORMAL CAPACITY: NA UNITS: NA CAPACITY SOURCE: N
ROTATION: NA ROTATION SOURCE: NA
BUILDING: RC ELEVATION: 291 BLDG/ELEV. SOURCE:
COLUMN/LOCATION: - COL/LOC SOURCE:
ENVIRONMENTAL ZONE: - ENV. ZONE SOURCE: -
A' REFERENCE: NAPS UFSAR SECTION 10.4.3 NA
NA NA
C /S:

COMPONENT ID: 1-FW-140
QA CATEGORY: SR

* * * * * COMPONENT IDENTIFICATION DATA * * * * *

STATION: 59 UNIT: 01 SYSTEM: FW COMPONENT CODE: VALVE
WPT 01-FW - -140 EQ HOST: EQML: N
R APPLICABLE: N VERIFIED: Y REFERENCE DWG: 11715-FM-74A
ATURE: VALVE, ~~GATE~~ GLOBE REVISION: 16
APPLICATION: STEAM GENERATOR 1-RC-E-1C LT-1497 I ELEC DRAWING:
SOLATION 01 TEST LOOP DIAGRAM:

* * * * * QUALITY CLASSIFICATION DATA * * * * *

ASSOC. COMPONENT ID: 01-FW -LT -1497 COMPONENT CODE: IXMITR

COMPONENT FUNCTION(S)	QA CATEGORY DEFINITION(S)
SYPB SYSTEM PRESSURE BOUNDARY	5.1.2.X
IISO INSTRUMENT ISOLATION	5.4

* * * * * EQML DATA * * * * *

EQML SOURCE DOC:
FILE: EQML FE DRAWING: REG. GUIDE 1.97:
INTERFACE: EQML LOOP DOC:
FUNCTION: EQML STATUS:
IN DATE: MAINT. CATEGORY: SOUND REASONS TO CONTRARY (SRC):

* * * * * COMPONENT PURCHASING DATA * * * * *

MANUFACTURER: C419 CONVAL, INC MFGR SOURCE:
MODEL OR CATALOG#: - 0.75-11G3J-S16-3D MODEL SOURCE: -
P.O. NO.: - ONT396432 P.O. SOURCE: -
SERIAL NO.: - SERIAL NO. SOURCE: -
STOCK NO.: STOCK NO. SOURCE:
PURCHASE SPEC.: - NAP-0023 PURCHASE SPEC. SOURCE: -
PURCHASE REQ.: - PURCHASE REQ. SOURCE: -

CODES AND STANDARDS:
-

MENT: -

REG. REQUIREMENTS: -

REF DOCUMENT: -

* * * * * MECHANICAL EQUIPMENT DATA * * * * *

FLOW DIAGRAM VALVE DESIGNATOR: ~~UGS-60B~~ NOMINAL SIZE: .75 NOMINAL SIZE SOURCE: 01
COMPONENT DESIGN PRESSURE: 600 TEMPERATURE: - COMP. DESIGN SOURCE: 46
SYSTEM DESIGN PRESSURE: - TEMPERATURE: - SYSTEM DESIGN SOURCE: -
SYSTEM WORKING PRESSURE: - TEMPERATURE: - SYSTEM WORKING SOURCE: -

* * * * * ELECTRICAL EQUIPMENT DATA * * * * *

POWER SOURCE: NA POWER PHASE: NA POWER SOURCE SOURCE: NA
ELECTRICAL POWER VOLTAGE: NA POWER CURRENT: NA CURRENT SOURCE: NA
ELECTRICAL CONTROL VOLTAGE: NA CONTROL CURRENT: NA CONTROL SOURCE: NA
NEMA CLASS: NA NEMA CLASS SOURCE: NA
INSULATION CLASS: NA INSULATION CLASS SOURCE: NA
FRAME SIZE: NA FRAME SIZE SOURCE: NA
THERMAL/OVERLOAD PROTECTION: NA THERMAL/OVERLOAD SOURCE: NA

* * * * * OTHER COMPONENT DATA * * * * *

NORMAL CAPACITY: NA UNITS: NA CAPACITY SOURCE: N
ROTATION: NA ROTATION SOURCE: NA
BUILDING: RC ELEVATION: 291 BLDG/ELEV. SOURCE:
COLUMN/LOCATION: - COL/LOC SOURCE:
ENVIRONMENTAL ZONE: - ENV. ZONE SOURCE: -
ADDI REFERENCE: NAPS UFSAR SECTION 10.4.3 NA
NA NA
CC :

COMPONENT ID: 1-FW-141
QA CATEGORY: SR

* * * * * * * * * * * * * * * * * COMPONENT IDENTIFICATION DATA * * * * * * * * * * * * * * *

STATION: 59 UNIT: 01 SYSTEM: FW COMPONENT CODE: VALVE
WP ': 01-FW - -141 EQ HOST: EQML: N
(R APPLICABLE: N VERIFIED: Y REFERENCE DWG: 11715-FM-74A
ATURE: VALVE, ~~GATE~~ GLOBE REVISION: 16
PL.CATION: STEAM GENERATOR 1-RC-E-1C LT-1497 I ELEC DRAWING:
SOLATION #1 TEST LOOP DIAGRAM:

* * * * * * * * * * * * * * * * QUALITY CLASSIFICATION DATA * * * * * * * * * * * * * * * *

ASSOC. COMPONENT ID: 02-FW -LT -1497 COMPONENT CODE: IXMITR

COMPONENT FUNCTION(S)	QA CATEGORY DEFINITION(S)
SYPB SYSTEM PRESSURE BOUNDARY	5.1.2.K
IISO INSTRUMENT ISOLATION	5.4

* EQML DATA *

EQML SOURCE DOC:
FILE: EQML FE DRAWING: REG. GUIDE 1.97:
INTERFACE: EQML LOOP DOC:
FUNCTION: EQML STATUS:
IN DATE: MAINT. CATEGORY: SOUND REASONS TO CONTRARY (SRC):

* * * * * * * * * * * * * * * * COMPONENT PURCHASING DATA * * * * * * * * * * * * * * * *

MANUFACTURER: C419 CONVAL, INC MFGR SOURCE:
MODEL OR CATALOG#: - 0.75-11G3J-S16-3D MODEL SOURCE: -
P.O. NO.: - P.O. SOURCE: -
SERIAL NO.: - CNT396432 SERIAL NO. SOURCE: -
STOCK NO.: STOCK NO. SOURCE:
PURCHASE SPEC.: - NAP-0023 PURCHASE SPEC. SOURCE: -
PURCHASE REQ.: - PURCHASE REQ. SOURCE: -

CODES AND STANDARDS:
-

JMENT: -

REG. REQUIREMENTS: -

REF DOCUMENT: -

* * * * * * * * * * * * * * * * MECHANICAL EQUIPMENT DATA * * * * * * * * * * * * * * * *

FLOW DIAGRAM VALVE DESIGNATOR: VBS-608 NOMINAL SIZE: .75 NOMINAL SIZE SOURCE: #1
COMPONENT DESIGN PRESSURE: 600 TEMPERATURE: - COMP. DESIGN SOURCE: 46
SYSTEM DESIGN PRESSURE: - TEMPERATURE: - SYSTEM DESIGN SOURCE: -
SYSTEM WORKING PRESSURE: - TEMPERATURE: - SYSTEM WORKING SOURCE: -

* * * * * * * * * * * * * * * * ELECTRICAL EQUIPMENT DATA * * * * * * * * * * * * * * * *

POWER SOURCE: NA POWER PHASE: NA POWER SOURCE SOURCE: NA
ELECTRICAL POWER VOLTAGE: NA POWER CURRENT: NA CURRENT SOURCE: NA
ELECTRICAL CONTROL VOLTAGE: NA CONTROL CURRENT: NA CONTROL SOURCE: NA
NEMA CLASS: NA NEMA CLASS SOURCE: NA
INSULATION CLASS: NA INSULATION CLASS SOURCE: NA
FRAME SIZE: NA FRAME SIZE SOURCE: NA
THERMAL/OVERLOAD PROTECTION: NA THERMAL/OVERLOAD SOURCE: NA

* * * * * * * * * * * * * * * * OTHER COMPONENT DATA * * * * * * * * * * * * * * * * * *

NORMAL CAPACITY: NA UNITS: NA CAPACITY SOURCE: N
ROTATION: NA ROTATION SOURCE: NA
BUILDING: RC ELEVATION: 262 BLDG/ELEV. SOURCE:
COLUMN/LOCATION: - COL/LOC SOURCE:
ENVIRONMENTAL ZONE: - ENV. ZONE SOURCE: -
ADDL. REFERENCE: NAPS UFSAR SECTION 10.4.3 NA
NA NA
C S:

OF REPORT

COMPONENT ID: 1-FW-256
QA CATEGORY: SR

* COMPONENT IDENTIFICATION DATA * * * * * * * * * * * * * *

ST/ · 59 UNIT: 01 SYSTEM: FW COMPONENT CODE: VALVE
01-FW - -256 EQ HOST: EQML: N
. R APPLICABLE: N VERIFIED: Y REFERENCE DWG: 11715-FM-74A
.ENCLATURE: VALVE, GLOBE REVISION: 16
APPLICATION: STEAM GENERATOR 1-RC-E-1A SUCTION D ELEC DRAWING:
RAIN 01 TEST LOOP DIAGRAM:

* * * * * * * * * * * * * * * * * * QUALITY CLASSIFICATION DATA * * * * * * * * * * * * * * * * * *

ASSOC. COMPONENT ID: 01-FW - -258 COMPONENT CODE: VALVE

COMPONENT FUNCTION(S)		QA CATEGORY DEFINITION(S)
SYPB	SYSTEM PRESSURE BOUNDARY	5.1.2.K
DRN	DRAIN	5.4

* EQML DATA *

EQML SOURCE DOC:
FILE: EQML FE DRAWING: REG. GUIDE 1.97:
INTERFACE: EQML LOOP DOC:
FUNCTION: EQML STATUS:
IN DATE: MAINT. CATEGORY: SOUND REASONS TO CONTRARY (SRC):

* * * * * * * * * * * * * * * * * COMPONENT PURCHASING DATA * * * * * * * * * * * * * * * * *

MANUFACTURER: C 419 Control, Inc MFGR SOURCE:
MODEL OR CATALOG#: - 0.75 - 11G2J MODEL SOURCE: -
P.O. NO.: - NS - 306 304 P.O. SOURCE: -
SERIAL NO.: - SERIAL NO. SOURCE: -
STOCK NO.: STOCK NO. SOURCE:
PURCHASE SPEC.: - PURCHASE SPEC. SOURCE: -
PURCHASE REQ.: - PURCHASE REQ. SOURCE: -

COD* *ND STANDARDS:
-

..DOCUMENT: -

REG. REQUIREMENTS: -

REF DOCUMENT: -

* * * * * * * * * * * * * * * * MECHANICAL EQUIPMENT DATA * * * * * * * * * * * * * * * *

FLOW DIAGRAM VALVE DESIGNATOR: VOS-60C NOMINAL SIZE: .75 NOMINAL SIZE SOURCE: 01
COMPONENT DESIGN PRESSURE: 600 TEMPERATURE: - COMP. DESIGN SOURCE: 46
SYSTEM DESIGN PRESSURE: 1100 TEMPERATURE: 440 SYSTEM DESIGN SOURCE: 08
SYSTEM WORKING PRESSURE: 850 TEMPERATURE: 440 SYSTEM WORKING SOURCE: 08

* * * * * * * * * * * * * * * * ELECTRICAL EQUIPMENT DATA * * * * * * * * * * * * * * * *

POWER SOURCE: NA POWER PHASE: NA POWER SOURCE SOURCE: NA
ELECTRICAL POWER VOLTAGE: NA POWER CURRENT: NA CURRENT SOURCE: NA
ELECTRICAL CONTROL VOLTAGE: NA CONTROL CURRENT: NA CONTROL SOURCE: NA
NEMA CLASS: NA NEMA CLASS SOURCE: NA
INSULATION CLASS: NA INSULATION CLASS SOURCE: NA
FRAME SIZE: NA FRAME SIZE SOURCE: NA
THERMAL/OVERLOAD PROTECTION: NA THERMAL/OVERLOAD SOURCE: NA

* * * * * * * * * * * * * * * * * * OTHER COMPONENT DATA * * * * * * * * * * * * * * * * * *

NORMAL CAPACITY: NA UNITS: NA CAPACITY SOURCE: N
ROTATION: NA ROTATION SOURCE: NA
BUILDING: RC ELEVATION: 291 BLDG/ELEV. SOURCE:
COLUMN/LOCATION: - COL/LOC SOURCE:
ENVIRONMENTAL ZONE: - ENV. ZONE SOURCE: -
AD' 'EFERENCE: NAPS UFSAR SECTION 10.4.3 NA
NA NA
C. S:

COMPONENT ID: 1-FW-258
QA CATEGORY: SR

* * * * * * * * * * * * * * * * * COMPONENT IDENTIFICATION DATA * * * * * * * * * * * * * * *

STATION: 59 UNIT: 01 SYSTEM: FW COMPONENT CODE: VALVE
WP 01-FW - -258 EQ HOST: EQML: N
R APPLICABLE: N VERIFIED: Y REFERENCE DWG: 11715-FM-74A
.TURE: VALVE, GLOBE REVISION: 16
LICATION: STEAM GENERATOR 1-RC-E-1B SUCTION D ELEC DRAWI:
RAIN 01 TEST LOOP DI GRAM:

* * * * * * * * * * * * * * * * * QUALITY CLASSIFICATION DATA * * * * * * * * * * * * * * * *

ASSOC. COMPONENT ID: 01-FW - -260 COMPONENT CODE: VALVE

COMPONENT FUNCTION(S)		QA CATEGORY DEFINITION(S)
SYPB	SYSTEM PRESSURE BOUNDARY	5.1.2.K
DRN	DRAIN	5.4

* EQML DATA *

EQML SOURCE DOC:
FILE: EQML FE DRAWING: REG. GUIDE 1.97:
INTERFACE: EQML LOOP DOC:
FUNCTION: EQML STATUS:
IN DATE: MAINT. CATEGORY: SOUND REASONS TO CONTRARY (SRC):

* * * * * * * * * * * * * * * * * COMPONENT PURCHASING DATA * * * * * * * * * * * * * * * * *

MANUFACTURER: C419 CONVAL, INC. MFGR SOURCE:
MODEL OR CATALOG#: - 0.75-11GZJ MODEL SOURCE: -
P.O. NO.: - NS-244452 P.O. SOURCE: -
SERIAL NO.: - SERIAL NO. SOURCE: -
STOCK NO.: STOCK NO. SOURCE:
PURCHASE SPEC.: - PURCHASE SPEC. SOURCE: -
PURCHASE REQ.: - PURCHASE REQ. SOURCE: -

CODES AND STANDARDS:
-

MENT: -

REG. REQUIREMENTS: -

REF DOCUMENT: -

* * * * * * * * * * * * * * * * * MECHANICAL EQUIPMENT DATA * * * * * * * * * * * * * * * * *

FLOW DIAGRAM VALVE DESIGNATOR: VOS-60C NOMINAL SIZE: .75 NOMINAL SIZE SOURCE: 01
COMPONENT DESIGN PRESSURE: 600 TEMPERATURE: - COMP. DESIGN SOURCE: 46
SYSTEM DESIGN PRESSURE: 1100 TEMPERATURE: 440 SYSTEM DESIGN SOURCE: 08
SYSTEM WORKING PRESSURE: 850 TEMPERATURE: 440 SYSTEM WORKING SOURCE: 08

* * * * * * * * * * * * * * * * * ELECTRICAL EQUIPMENT DATA * * * * * * * * * * * * * * * * *

POWER SOURCE: NA POWER PHASE: NA POWER SOURCE SOURCE: NA
ELECTRICAL POWER VOLTAGE: NA POWER CURRENT: NA CURRENT SOURCE: NA
ELECTRICAL CONTROL VOLTAGE: NA CONTROL CURRENT: NA CONTROL SOURCE: NA
NEMA CLASS: NA NEMA CLASS SOURCE: NA
INSULATION CLASS: NA INSULATION CLASS SOURCE: NA
FRAME SIZE: NA FRAME SIZE SOURCE: NA
THERMAL/OVERLOAD PROTECTION: NA THERMAL/OVERLOAD SOURCE: NA

* * * * * * * * * * * * * * * * * OTHER COMPONENT DATA * * * * * * * * * * * * * * * * * * *

NORMAL CAPACITY: NA UNITS: NA CAPACITY SOURCE: N
ROTATION: NA ROTATION SOURCE: NA
BUILDING: RC ELEVATION: 291 BLDG/ELEV. SOURCE:
COLUMN/LOCATION: - COL/LOC SOURCE:
ENVIRONMENTAL ZONE: - ENV. ZONE SOURCE: -
ADDL REFERENCE: NAPS UFSAR SECTION 10.4.3 NA
NA NA
CC

COMPONENT ID: 1-FW-260
QA CATEGORY: SR

* * * * * * * * * * * * * * * * * * COMPONENT IDENTIFICATION DATA * * * * * * * * * * * * * *

STA... 59 UNIT: 01 SYSTEM: FW COMPONENT CODE: VALVE
... 01-FW - -260 EQ HOST: EQML: N
... R APPLICABLE: N VERIFIED: Y REFERENCE DWG: 11715-FM-74A
...LATURE: VALVE, GLOBE REVISION: 16
...PLICATION: STEAM GENERATOR 1-RC-E-1C SUCTION D ELEC DRAWING:
RAIN 01 TEST LOOP DIAGRAM:

* * * * * * * * * * * * * * * * * QUALITY CLASSIFICATION DATA * * * * * * * * * * * * * * * *

ASSOC. COMPONENT ID: 01-FW - -259 COMPONENT CODE: VALVE

COMPONENT FUNCTION(S)		QA CATEGORY DEFINITION(S)
SYPB	SYSTEM PRESSURE BOUNDARY	5.1.2.K
DRN	DRAIN	5.4

* EQML DATA *

EQML SOURCE DOC:
FILE: EQML FE DRAWING: REG. GUIDE 1.97:
INTERFACE: EQML LOOP DOC:
FUNCTION: EQML STATUS:
IN DATE: MAINT. CATEGORY: SOUND REASONS TO CONTRARY (SRC):

* * * * * * * * * * * * * * * * * COMPONENT PURCHASING DATA * * * * * * * * * * * * * * * *

MANUFACTURER: C419 Conval, Inc MFGR SOURCE:
MODEL OR CATALOG#: - 0.75 - 11G2J MODEL SOURCE: -
P.O. NO.: - P.O. SOURCE: -
SERIAL NO.: - NS 244452 SERIAL NO. SOURCE: -
STOCK NO.: STOCK NO. SOURCE:
PURCHASE SPEC.: - PURCHASE SPEC. SOURCE: -
PURCHASE REQ.: - PURCHASE REQ. SOURCE: -

CODES AND STANDARDS:
-

...CUMENT: -

REG. REQUIREMENTS: -

REF DOCUMENT: -

* * * * * * * * * * * * * * * * * MECHANICAL EQUIPMENT DATA * * * * * * * * * * * * * * * *

FLOW DIAGRAM VALVE DESIGNATOR: VOS-60C NOMINAL SIZE: .75 NOMINAL SIZE SOURCE: 01
COMPONENT DESIGN PRESSURE: 600 TEMPERATURE: - COMP. DESIGN SOURCE: 46
SYSTEM DESIGN PRESSURE: 1100 TEMPERATURE: 440 SYSTEM DESIGN SOURCE: 08
SYSTEM WORKING PRESSURE: 850 TEMPERATURE: 440 SYSTEM WORKING SOURCE: 08

* * * * * * * * * * * * * * * * * ELECTRICAL EQUIPMENT DATA * * * * * * * * * * * * * * * *

POWER SOURCE: NA POWER PHASE: NA POWER SOURCE SOURCE: NA
ELECTRICAL POWER VOLTAGE: NA POWER CURRENT: NA CURRENT SOURCE: NA
ELECTRICAL CONTROL VOLTAGE: NA CONTROL CURRENT: NA CONTROL SOURCE: NA
NEMA CLASS: NA NEMA CLASS SOURCE: NA
INSULATION CLASS: NA INSULATION CLASS SOURCE: NA
FRAME SIZE: NA FRAME SIZE SOURCE: NA
THERMAL/OVERLOAD PROTECTION: NA THERMAL/OVERLOAD SOURCE: NA

* * * * * * * * * * * * * * * * * * OTHER COMPONENT DATA * * * * * * * * * * * * * * * * * *

NORMAL CAPACITY: NA UNITS: NA CAPACITY SOURCE: N
ROTATION: NA ROTATION SOURCE: NA
BUILDING: RC ELEVATION: 291 BLDG/ELEV. SOURCE:
COLUMN/LOCATION: - COL/LOC SOURCE:
ENVIRONMENTAL ZONE: - ENV. ZONE SOURCE: -
AD... REFERENCE: NAPS UFSAR SECTION 10.4.3 NA
NA NA
... S:

COMPONENT ID: 1-RC-E -1A
QA CATEGORY: SR

* * * * * * * * * * * * * * * * * COMPONENT IDENTIFICATION DATA * * * * * * * * * * * * * *

N: 59 UNIT: 01 SYSTEM: RC COMPONENT CODE: HTEXCH
D: 01-RC -E -1A EQ HOST: EQML: N
JIX R APPLICABLE: N VERIFIED: Y REFERENCE DWG: 11715-FM-93A
NOMENCLATURE: GENERATOR, STEAM REVISION: 14
APPLICATION: STEAM GENERATOR 1A ELEC DRAWING:
01 TEST LOOP DIAGRAM:

* * * * * * * * * * * * * * * * * QUALITY CLASSIFICATION DATA * * * * * * * * * * * * * * * *

ASSOC. COMPONENT ID: 01-RC -E -1B COMPONENT CODE: HTEXCH

COMPONENT FUNCTION(S)	QA CATEGORY DEFINITION(S)
XFER HEAT TRANSFER	5.1.1
RCPB REACTOR COOLANT PRESSURE BOUNDARY	5.1.1

* * * * * * * * * * * * * * * * * * * EQML DATA *

EQML SOURCE DOC:
FILE: EQML FE DRAWING: REG. GUIDE 1.97:
INTERFACE: EQML LOOP DOC:
FUNCTION: EQML STATUS:
IN DATE: MAINT. CATEGORY: SOUND REASONS TO CONTRARY (SRC):

* * * * * * * * * * * * * * * * * COMPONENT PURCHASING DATA * * * * * * * * * * * * * * * *

MANUFACTURER: W893 WESTINGHOUSE ELECT CORP MFGR SOURCE: 03
MODEL OR CATALOG#: TYPE 51F MODEL SOURCE: 03
P.O. NO.: BNT-273343 P.O. SOURCE: -
SERIAL NO.: 1261 SERIAL NO. SOURCE: 03
STOCK NO.: STOCK NO. SOURCE:
PURCHASE SPEC: W-120, NAP-0001 PURCHASE SPEC. SOURCE: 33
PURCHASE REQ. - PURCHASE REQ. SOURCE: -

C ND STANDARDS:

F DOCUMENT: -

REG. REQUIREMENTS: -

REF DOCUMENT: -

* * * * * * * * * * * * * * * * * MECHANICAL EQUIPMENT DATA * * * * * * * * * * * * * * * *

FLOW DIAGRAM VALVE DESIGNATOR: NA NOMINAL SIZE: NA NOMINAL SIZE SOURCE: NA
COMPONENT DESIGN PRESSURE: - TEMPERATURE: - COMP. DESIGN SOURCE: -
SYSTEM DESIGN PRESSURE: 2485 TEMPERATURE: 650 SYSTEM DESIGN SOURCE: 08
SYSTEM WORKING PRESSURE: 2235 TEMPERATURE: 614 SYSTEM WORKING SOURCE: 08

* * * * * * * * * * * * * * * * * ELECTRICAL EQUIPMENT DATA * * * * * * * * * * * * * * * *

POWER SOURCE: NA POWER PHASE: NA POWER SOURCE SOURCE: NA
ELECTRICAL POWER VOLTAGE: NA POWER CURRENT: NA CURRENT SOURCE: NA
ELECTRICAL CONTROL VOLTAGE: NA CONTROL CURRENT: NA CONTROL SOURCE: NA
NEMA CLASS: NA NEMA CLASS SOURCE: NA
INSULATION CLASS: NA INSULATION CLASS SOURCE: NA
FRAME SIZE: NA FRAME SIZE SOURCE: NA
THERMAL/OVERLOAD PROTECTION: NA THERMAL/OVERLOAD SOURCE: NA

* * * * * * * * * * * * * * * * * OTHER COMPONENT DATA * * * * * * * * * * * * * * * * *

NORMAL CAPACITY: 3168 UNITS: MJH CAPACITY SOURCE: 0
ROTATION: NA ROTATION SOURCE: NA
BUILDING: RC ELEVATION: 300 BLDG/ELEV. SOURCE: 39
COLUMN/LOCATION: 1.5 COL/LOC SOURCE: 39
ENVIRONMENTAL ZONE: - ENV. ZONE SOURCE: -
A REFERENCE: NAPS UFSAR SEC 5 NA
NA NA
C TS:

COMPONENT ID: 1-RC-E -1B
QA CATEGORY: SR

* * * * * * * * * * * * * * * * * COMPONENT IDENTIFICATION DATA * * * * * * * * * * * * *

: 59 UNIT: 01 SYSTEM: RC COMPONENT CODE: HTEXCH
D: 01-RC -E -1B EQ HOST: EQML: N
..NDIX R APPLICABLE: N VERIFIED: Y REFERENCE DWG: 11715-FM-93A
NOMENCLATURE: GENERATOR, STEAM REVISION: 14
APPLICATION: STEAM GENERATOR 1B ELEC DRAWING:
01 TEST LOOP DIAGRAM:

* * * * * * * * * * * * * * * * QUALITY CLASSIFICATION DATA * * * * * * * * * * * * * * * *

ASSOC. COMPONENT ID: 01-RC -E -1A COMPONENT CODE: HTEXCH

COMPONENT FUNCTION(S)	QA CATEGORY DEFINITION(S)
XFER HEAT TRANSFER	5.1.1
RCPB REACTOR COOLANT PRESSURE BOUNDARY	5.1.1

* * * * * * * * * * * * * * * * * * * EQML DATA * * * * * * * * * * * * * * * * * * *

EQML SOURCE DOC:
FILE: EQML FE DRAWING: REG. GUIDE 1.97:
INTERFACE: EQML LOOP DOC:
FUNCTION: EQML STATUS:
IN DATE: MAINT. CATEGORY: SOUND REASONS TO CONTRARY (SRC):

* * * * * * * * * * * * * * * * * COMPONENT PURCHASING DATA * * * * * * * * * * * * * * * *

MANUFACTURER: W.33 WESTINGHOUSE ELECT CORP MFGR SOURCE: 03
MODEL OR CATALOG#: TYPE 51F MODEL SOURCE: 03
P.O. NO.: BNT-273343 P.O. SOURCE: -
SERIAL NO.: 1262 SERIAL NO. SOURCE: 03
STOCK NO.: STOCK NO. SOURCE:
PURCHASE SPEC.: W-120, NAP-0001 PURCHASE SPEC. SOURCE: 33
PURCHASE REQ.: PURCHASE REQ. SOURCE: -

AND STANDARDS:

REF DOCUMENT: -

REG. REQUIREMENTS: -

REF DOCUMENT: -

* * * * * * * * * * * * * * * * MECHANICAL EQUIPMENT DATA * * * * * * * * * * * * * * * *

FLOW DIAGRAM VALVE DESIGNATOR: NA NOMINAL SIZE: NA NOMINAL SIZE SOURCE: NA
COMPONENT DESIGN PRESSURE: - TEMPERATURE: - COMP. DESIGN SOURCE: -
SYSTEM DESIGN PRESSURE: 2485 TEMPERATURE: 650 SYSTEM DESIGN SOURCE: 08
SYSTEM WORKING PRESSURE: 2235 TEMPERATURE: 614 SYSTEM WORKING SOURCE: 08

* * * * * * * * * * * * * * * * ELECTRICAL EQUIPMENT DATA * * * * * * * * * * * * * * * *

POWER SOURCE: NA POWER PHASE: NA POWER SOURCE SOURCE: NA
ELECTRICAL POWER VOLTAGE: NA POWER CURRENT: NA CURRENT SOURCE: NA
ELECTRICAL CONTROL VOLTAGE: NA CONTROL CURRENT: NA CONTROL SOURCE: NA
NEMA CLASS: NA NEMA CLASS SOURCE: NA
INSULATION CLASS: NA INSULATION CLASS SOURCE: NA
FRAME SIZE: NA FRAME SIZE SOURCE: NA
THERMAL/OVERLOAD PROTECTION: NA THERMAL/OVERLOAD SOURCE: NA

* * * * * * * * * * * * * * * * * OTHER COMPONENT DATA * * * * * * * * * * * * * * * *

NORMAL CAPACITY: 3167 UNITS: MBH CAPACITY SOURCE: 0
ROTATION: NA ROTATION SOURCE: NA
BUILDING: RC ELEVATION: 241 BLDG/ELEV. SOURCE: 39
COLUMN/LOCATION: 13 COL/LOC SOURCE: 39
E NMENTAL ZONE: - ENV. ZONE SOURCE: -
/ EFERENCE: NAPS UFSAR SEC 5 NA
NA NA
ENTS:

COMPONENT ID: 1-RC-E-1C
QA CATEGORY: SR

* * * * * * * * * * * * * * * * * COMPONENT IDENTIFICATION DATA * * * * * * * * * * * * * * *

: 59 UNIT: 01 SYSTEM: RC COMPONENT CODE: HTEXCH
: 01-RC -E -1C EQ HOST: EQML: N
...DIX R APPLICABLE: N VERIFIED: Y REFERENCE DWG: 11715-FM-93A
NOMENCLATURE: GENERATOR, STEAM REVISION: 14
APPLICATION: STEAM GENERATOR 1C ELEC DRAWING:
01 TEST LOOP DIAGRAM:

* * * * * * * * * * * * * * * * QUALITY CLASSIFICATION DATA * * * * * * * * * * * * * * * *

ASSOC. COMPONENT ID: 01-RC -E -A COMPONENT CODE: HTEXCH

COMPONENT FUNCTION(S)		QA CATEGORY DEFINITION(S)
XFER	HEAT TRANSFER	5.1.1
RCPB	REACTOR COOLANT PRESSURE BOUNDARY	5.1.1

* * * * * * * * * * * * * * * * * * EQML DATA *

EQML SOURCE DOC:
FILE: EQML FE DRAWING: REG. GUIDE 1.97:
INTERFACE: EQML LOOP DOC:
FUNCTION: EQML STATUS:
IN DATE: MAINT. CATEGORY: SOUND REASONS TO CONTRARY (SRC):

* * * * * * * * * * * * * * * * * COMPONENT PURCHASING DATA * * * * * * * * * * * * * * * *

MANUFACTURER: W893 WESTINGHOUSE ELECT CORP MFGR SOURCE: 03
MODEL OR CATALOG#: TYPE 51 F MODEL SOURCE: 03
P.O. NO.: BNT-273343 P.O. SOURCE: -
SERIAL NO.: 1263 SERIAL NO. SOURCE: -
STOCK NO.: STOCK NO. SOURCE:
PURCHASE SPEC.: W-120, NAP-0001 PURCHASE SPEC. SOURCE: 33
PURCHASE REQ.: - PURCHASE REQ. SOURCE: -

ND STANDARDS:

REF DOCUMENT: -

REG. REQUIREMENTS: -

REF DOCUMENT: -

* * * * * * * * * * * * * * * * MECHANICAL EQUIPMENT DATA * * * * * * * * * * * * * * * *

FLOW DIAGRAM VALVE DESIGNATOR: NA NOMINAL SIZE: NA NOMINAL SIZE SOURCE: NA
COMPONENT DESIGN PRESSURE: - TEMPERATURE: - COMP. DESIGN SOURCE: -
SYSTEM DESIGN PRESSURE: 2485 TEMPERATURE: 650 SYSTEM DESIGN SOURCE: 08
SYSTEM WORKING PRESSURE: 2235 TEMPERATURE: 614 SYSTEM WORKING SOURCE: 08

* * * * * * * * * * * * * * * * ELECTRICAL EQUIPMENT DATA * * * * * * * * * * * * * * * *

POWER SOURCE: NA POWER PHASE: NA POWER SOURCE SOURCE: NA
ELECTRICAL POWER VOLTAGE: NA POWER CURRENT: NA CURRENT SOURCE: NA
ELECTRICAL CONTROL VOLTAGE: NA CONTROL CURRENT: NA CONTROL SOURCE: NA
NEMA CLASS: NA NEMA CLASS SOURCE: NA
INSULATION CLASS: NA INSULATION CLASS SOURCE: NA
FRAME SIZE: NA FRAME SIZE SOURCE: NA
THERMAL/OVERLOAD PROTECTION: NA THERMAL/OVERLOAD SOURCE: NA

* * * * * * * * * * * * * * * * * OTHER COMPONENT DATA * * * * * * * * * * * * * * * * *

NORMAL CAPACITY: 3168 UNITS: MBH CAPACITY SOURCE: 0
ROTATION: NA ROTATION SOURCE: NA
BUILDING: RC ELEVATION: 300 BLDG/ELEV. SOURCE: 39
COLUMN/LOCATION: 7.5 COL/LOC SOURCE: 39
E... MENTAL ZONE: - ENV. ZONE SOURCE: -
...EFERENCE: NAPS UFSAR SEC NA
NA NA
...ENTS:

END OF REPORT

ERF DESIGN CHECKLIST

ERF DESIGN CHECKLIST
VIRGINIA POWER

TITLE/STATION/UNIT: STEAM GENERATOR REPAIR/NORTH ANNA UNIT 1	[✓] DCP [ ] EWR No.: 90-13-1
PREPARING ENGINEER/ORGANIZATION/LOCATION: G.D. BROWN/BECHTEL/GAITHERSBURG, MD	DATE:

INSTRUCTIONS:

This design checklist is to be completed to ensure that any necessary changes to the ERF system software resulting from a Design Change or Engineering Work Request are identified.

PART I: To be completed by the Preparing Engineer

Based on the questions contained in Section 1.0 of this checklist on the following pages, does this modification have any impact on the ERF computer system?

[✓] Yes [ ] No

If yes, Part II must be completed by Virginia Power NES Nuclear Electrical Engineering Computers and Programs Section.

Comments:

Signature of Preparing Engineer: MS Barth	Date: 6/22/92

PART II: To be completed by Virginia Power NES Nuclear Electrical Engineering Computers and Programs Section only if an impact to the ERF system was identified in Part I above.

The ERF system impact identified by the Preparing Engineer above has been reviewed and there is an associated impact on the ERF system software.

[X] Yes [ ] No

If yes, the required software changes are identified in Section 2.0 of this checklist.

Comments:

Signature of Computers and Programs Section Supervisor: MB L.C. Kidd Jr.	Date: 9/23/92

PART III: To be completed by the Reviewing Engineer

The required ERF system impact reviews have been completed.

Comments:

Signature of Reviewer: SL Wilkie	Date: 9-23-92

ERF DESIGN CHECKLIST
VIRGINIA POWER

1.0 ERF impact questions to be answered by the Preparer of a Design Change Package.

1.1 Does this modification add any inputs to the Validyne multiplexor system? If no, go to 1.4.

[ ] YES [✓] NO

Comments: ______________________________

1.2 If the answer to 1.1 is yes, has the need to add these inputs to the ERF system been discussed with and approved by the NES Nuclear Electrical Engineering Computers and Programs Section?

[ ] YES [ ] NO

Comments: ______________________________

1.3 If the answer to 1.1 is yes, a copy of Attachment 7.4 must be completed for each input being added to the system. (Note: Attachment 7.8 must be completed per Step 1.13 for Reg. Guide 1.97 inputs.)

Comments: ______________________________

ERF DESIGN CHECKLIST
VIRGINIA POWER

1.4 Does this modification delete any inputs from the Validyne multiplexor system? If no, go to 1.7.

[ ] YES [✓] NO

Comments: ______________________________

____

____

1.5 If the answer to 1.4 is yes, has the need to delete these inputs from the ERF system been discussed with and approved by the NES Nuclear Electrical Engineering Computers and Programs Section?

[ ] YES [ ] NO

1.6 If the answer to 1.4 is yes, a copy of Attachment 7.5 must be completed for each input being deleted from the system. (Note: Attachment 7.8 must be completed per Step 1.13 for Reg. Guide 1.97 inputs.)

Comments: ______________________________

____

____

1.7 Does the modification add or delete any equipment or instrumentation to any drawing included in the ERF Design Drawing List (Attachment 7.2)? If no, go to 1.9.

[ ] YES [✓] NO

Comments: ______________________________

____

____

ERF DESIGN CHECKLIST
VIRGINIA POWER

1.8 If the answer to 1.7 is yes, a copy of Attachment 7.6 must be completed for each drawing affected.

Comments: ______________________

1.9 Does the modification affect any instrument included in the ERF Input/Output List in such a way that the input signal characteristics (e.g., instrument range, signal type (linear, square root), contact state, etc.) to the Validyne channel tied into the loop will be affected? Note: A copy of the latest ERF Input/Output List can be obtained from the Virginia Power Project Engineer or from NES Nuclear Electrical Engineering Computers and Programs Section. If no, go to 1.11.

[✓] YES [ ] NO

Comments: THE ALARM SETPOINT FOR THE RWST SWITCHOVER TO SUMP RECIRC NEEDS TO BE REVISED TO 19.4% SPAN

1.10 If the answer to 1.9 is yes, a copy of Attachment 7.6 must be completed for each instrument affected.

Comments: ATTACHED

1.11 Does the modification affect any controlled document section included in the ERF Design Controlled Documents List (Attachment 7.3)? If no, go to 1.13.

[✓] YES [ ] NO

Comments: UFSAR SECTIONS 5,6 AND 10 NEED TO BE UPDATED TO INCORPORATE NEW REPLACEMENT SG LOWER ASSEMBLY DESIGN CHANGES. THE PLS DOCUMENT FOR NAPS 1 ALSO REQUIRES REVISION.

ERF DESIGN CHECKLIST
VIRGINIA POWER

1.12 If the answer to 1.11 is yes, a copy of Attachment 7.6 must be completed for each controlled document affected.

Comments: CONTROLLED DOCUMENT IMPACT DATA SHEETS ARE ATTACHED.

1.13 Does the modification affect the Reg. Guide 1.97 status of any ERF input? The answer to this question is yes if the modification removes an input from the Reg. Guide 1.97 variable list, deletes a Reg. Guide 1.97 input from the ERF data base, adds an ERF input to the Reg. Guide 1.97 variable list, or changes the Reg. Guide 1.97 variable type of an existing ERF input. Note: A copy of the latest ERF Reg. Guide 1.97 input list can be obtained from the Project Engineer or NES Nuclear Electrical Engineering Computers and Programs Section. For assistance in determining Reg. Guide 1.97 impact, contact the Reg. Guide 1.97 Coordinator and refer to STD-GN-0035, "NRC Regulatory Guide 1.97 Compliance Engineering Guidelines for Post-Accident Monitoring."

[ ] YES [✓] NO

Comments:

1.14 If the answer to 1.13 is yes, a copy of Attachment 7.8 must be completed for each Reg. Guide 1.97 input affected. For assistance in completing Attachment 7.8, contact the Reg. Guide 1.97 Coordinator.

Comments:

ERF DESIGN CHECKLIST
VIRGINIA POWER

2.0 ERF computer system software modification description. To be completed by NES Nuclear Electrical Engineering Computers and Programs Section only if an impact on the ERF system was identified in Section 1 above.

2.1 Provide a description of the software changes to be performed to reflect the changes identified in Section 1 above. Attach additional sheets if required.

The following parameters will be changed for this DCP:
- Secondary Heat Removal CSF - steam flow high alarm setpt
- RCS Integrity CSF - pressurizer normal level
- Containment Condition CSF - RWST level low alarm setpt.
- Core Heat Removal CSF - normal Tavg
- Top Level and Core Heat Removal PRIDs - That high alarm setpt & That normal value

The OTΔT, OPΔT, and core power level alarms implemented by EWR 92-092 will be changed to the nominal values. SPDS & Displays Design Algorithm Documents and SPDS Test Analysis Report will be updated. DCP I/O List sorted by system will be re-issued

ERF DRAWING/INSTRUMENT/
CONTROLLED DOCUMENT
IMPACT DATA SHEET

This data sheet is to be used in conjunction with Attachments 7.2 and 7.3 to identify ERF system impacts as defined in Sections 1.7, 1.9 and 1.11 of Attachment 7.1.

1.0 DCP affects the following:

[ ] Drawings [✓] Instruments [✓] Controlled Documents

2.0 If drawings or controlled documents are affected, are mark-ups attached showing the changes?

[ ] YES [✓] NO

3.0 For each affected drawing, instrument, and controlled document, complete Page 2 of this attachment.

**ERF DRAWING/INSTRUMENT/
CONTROLLED DOCUMENT
IMPACT DATA SHEET**

4.0 This data sheet represents a change to a:

[ ] Drawing [ ] Instrument [X] Controlled Document

5.0 Based on the response to 4.0, enter one of the following:

Drawing No. ____________________

Instrument Mark No. ____________________

Controlled Document Name and Section No. UFSAR Sections 5, 6, and 10

6.0 Describe the change to be made:

UFSAR Section 5.2.1 discusses the integrity of the RCS boundary and will be revised.

UFSAR Tables 5.2-4 through 5.2-8 will be revised to show changes to the SG Primary-Secondary Boundary Components.

UFSAR Tables 5.2-9 and 5.2-10 will be revised to show changes in SG usage factors, individual transients primary and secondary boundary components.

UFSAR Table 5.2-11 will be revised to show changes in the tubesheet stress analysis results.

UFSAR Table 5.2-12 will be revised for the limit analysis calculation results.

UFSAR Table 5.2-17 contains faulted condition loads for the reactor coolant pump feet. This table needs to be updated to reflect the new loads on the Unit 1 reactor coolant pump feet.

UFSAR Table 5.2-22 will be revised to show correct material for the new SGs.

UFSAR Figures 5.2-1 through 5.2-4 will be revised to correspond to the new SG design.

UFSAR Section 5.5.2 will be revised to reflect the new SG lower assembly design details.

**ERF DRAWING/INSTRUMENT/
CONTROLLED DOCUMENT
IMPACT DATA SHEET**

6.0 Describe the change to be made: (Continued)

UFSAR Section 5.5.9 describes the RCS equipment supports. Section 5.5.9.2.2 discusses specifically the steam generator and reactor coolant pump supports. This section needs to be updated to incorporate the steam generator upper support ring cold condition radial gap tolerance of 0.050 to 0.060 inches. Section 5.5.9.3.1 discusses the dynamic analyses that determines loads on the subject equipment supports. This section needs to be updated to include reference to the NUPIPE-SW computer program.

UFSAR Table 5.5-3 will be revised to show changes in the SG design data.

UFSAR Figure 5.5-3 will be revised to show the replacement steam generator diagram for unit 1.

UFSAR Figure 5.5-17 provides the dynamic model for the reactor coolant loop including the steam generator and reactor coolant pump supports. This figure (to be identified as figure 5.5-17B) will remain valid for the Unit 2 dynamic model only. An additional figure (to be identified as figure 5.5-17A) for the modified Unit 1 dynamic model needs to be incorporated into the UFSAR.

UFSAR Tables 5.5-18, 5.5-19 and 5.5-20 contain loads for the steam generator and reactor coolant pump equipment supports. These tables need to be revised to incorporate the new Unit 1 equipment support loads.

UFSAR Figures 5.5-24 and 5.5-25 provide the identification of steam generator and reactor coolant pump support members respectively. These figures were previously utilized with UFSAR Tables 5.5-21 and 5.5-22 which have been deleted from the UFSAR. UFSAR Figures 5.5-24 and 5.5-25 are no longer required and should be deleted.

UFSAR Section 6.2 discusses Containment Systems. This section is updated to reflect the revised analysis of peak pressure and depressurization following a LOCA based on the mass and energy releases for the replacement steam generators. Where information for Unit 1 differs from the current analysis for Unit 2, clear references to applicable units are added.

UFSAR Section 6.2 also addresses the impact of the new insulation on the containment response following a LOCA. The impact of insulation debris on sump NPSH is evaluated.

UFSAR tables 6.2-2,5,13-17, 47-53, and 77 are revised to reflect the differences between the Unit 1 and Unit 2 analyses.

**ERF DRAWING/INSTRUMENT/
CONTROLLED DOCUMENT
IMPACT DATA SHEET**

6.0 Describe the change to be made: (Continued)

UFSAR Section 6.3 discusses the Emergency Core Cooling System. This section is updated to reflect the change in the NPSH values and the revision to the RWST switchover setpoint for the LHSI recirculation mode for Unit 1.

UFSAR Figure 6.3-11 is revised to reflect the change in the Unit 1 RWST switchover setpoints.

UFSAR Figures 6.3-12 through 6.3-17 are revised to show the Unit 1 vs. Unit 2 NPSH analysis differences.

UFSAR Section 10.3.2 contains the description of the main steam system. This section needs to be revised to describe the new flow limiting device having a flow area of 1.4 ft^2 to be installed in each of the Unit 1 main steam nozzles of the steam generator dome.

UFSAR Figure 10.4-7 and 10.4-8 detail the chemical feed piping and indicate the pipe class at the feedwater elbow as pipe class 601. The installed piping was changed from pipe class 601 to class IC-N-9 as part of DC-80-S82, but these figures were not revised to reflect the change. The replacement chemical feed piping will be IC-N-9 to match the existing installation. These figures will be revised to indicate use of class IC-N-9.

UFSAR Figures 10.4-8 (sheet 1), 10.4-11, 10.4-13 and 10.4-14 show the configuration of the main feedwater system and a portion of the chemical feed system in the area of the feedwater loop seal piping. Each of these figures needs to be updated to reflect the change in pipe class for the new class 601C feedwater loop seal piping replacement.

UFSAR Figure 10.4-17 details the steam generator blowdown piping system arrangement. This figure needs to be updated to reflect the system modifications being implemented by this Design Change. Specifically, the changes in pipe size and class need to be incorporated in this figure.

ERF DRAWING/INSTRUMENT/
CONTROLLED DOCUMENT
IMPACT DATA SHEET

4.0 This data sheet represents a change to a:

[ ] Drawing [ ] Instrument [✓] Controlled Document

5.0 Based on the response to 4.0, enter one of the following:

Drawing No. __________

Instrument Mark No. __________

Controlled Document Name and Section No. NAPS 1 PLS DOCUMENT, SECTION 1.A and 1.B

6.0 Describe the change to be made:

LEVEL CONTROLLERS LC-QS100 A, B, C AND D NEED TO HAVE REVISED SETPOINTS TO CORRESPOND TO THE REVISED CONTAINMENT ANALYSIS FOR SGR. THE REVISED VALUE IS 19.4% of SPAN.

THE PLS DOCUMENT WILL REFLECT THE CHANGE IN NOMINAL TAVE FROM 583.8°F TO 580.8°F.

ERF DRAWING/INSTRUMENT/
CONTROLLED DOCUMENT
IMPACT DATA SHEET

4.0 This data sheet represents a change to a:

[ ] Drawing [✓] Instrument [ ] Controlled Document

5.0 Based on the response to 4.0, enter one of the following:

Drawing No. ______

Instrument Mark No. LC-QS100A, B, & C

Controlled Document
Name and Section No. ______

6.0 Describe the change to be made:

ALARM SETPOINT CHANGE TO REVISED VALUE OF 19.4% SPAN OF RWST VOLUME.

POST ACCIDENT MONITORING DESIGN CHECKLIST

DRAFT

POST-ACCIDENT MONITORING DESIGN CHECKLIST

PREPARER MSBarth DATE 9/18/92

REVIEWER [illegible] DATE [illegible]

YES	NO		
	X	1.	Does the modification affect any existing Regulatory Guide 1.97 instrumentation? (See 5.7.1)
	X	2.	Does the modification affect any control room indicator or recorder? (See 5.7.2)
	X	3.	Does the modification change any radioactive release path? (See 5.7.3)
	X	4.	Does the modification affect any radiation monitoring equipment? (See 5.7.4)
	X	5.	Does the modification change the qualified status of any equipment? (See 5.7.5)
	X	6.	Does the modification affect any HVAC equipment? (See 5.7.6)
	X	7.	Does the modification affect any containment isolation valve monitoring equipment? (See 5.7.7)
	X	8.	Does the modification affect any process monitored by the instrumentation listed in Appendices 7.2 and 7.3 of reference 3.1.3 and 3.1.4? (See 5.7.8)
	X	9.	Does the modification affect any safety-related or semi-vital electrical buses? (See 5.7.9)
	X	10.	Does the modification affect any hydraulic or pneumatic supply system? (See 5.7.10)
	X	11.	Does the modification affect any meteorological equipment? (See 5.7.11)
	X	12.	Does the modification affect and sampling system, path, or capability? (See 5.7.12)

DRAFT

POST-ACCIDENT MONITORING DESIGN CHECKLIST

YES	NO		
X	___	13.	Does the modification require a change to the Emergency Operating Procedures or Emergency Plan? (See 5.7.13)
___	X	14.	Does the modification change the Environmental Zone Description (FZD) and/or any Qualification Documentation Review (QDR)? (See 5.7.14)
___	X	15.	Does the modification involve a change to the Technical Specifications? (See 5.7.15)

REG. GUIDE 1.97 REVIEW SHEET
VIRGINIA POWER NUCLEAR POWER STATION

Title/Station/Unit DCP 90-13-1 EWR ____

STEAM GENERATOR REPLACEMENT No. ____________

NORTH ANNA POWER STATION - UNIT 1

The referenced DCP/EWR has been reviewed by me and requires a change to the basis for compliance with NRC Regulatory Guide 1.97 as defined in References 3.1.3 and 3.1.4.

____ YES X NO

The justification for this determination is as follows:

Refer to section 3.5 of DCP.

The proposed change to Reference 3.1.3/3.1.4 is described below. (Mark N/A if no change is required. Include copies of marked-up changes as necessary.)

[signature] 9/22/92
R.G. 1.97 Coordinator / Date

The subject DCP/EWR has been implemented without any additional changes that affect the justification described above.

_____ YES   _____ NO

If no, describe the change below:

____
Site Supervisor - Design Engineering / Date

SET POINT CHANGE FORM

Attachment 6.1

SETPOINT IMPLEMENTATION / CHANGE FORM STD-GN-0030, REV. 3

SECTION 1

NORTH ANNA POWER STATION UNIT 1

EQUIPMENT MARK NUMBER 2S-LC-100A, B DCP/EWR NUMBER 90-13-1

EQUIPMENT DESCRIPTION RWST LEVEL

REASON FOR CHANGE LOCA CONTAINMENT ANALYSIS

OTHER REFERENCES, DRAWINGS, ETC ______

(✓) SAFETY RELATED ( ) REGULATORY SIGNIFICANT ( ) NON-SAFETY RELATED

SECTION 2

	VALUE	UNITS
SAFETY/PROCESS DESIGN LIMIT REFERENCE THIS VALUE IS DERIVED (ie → 22.8 - 2.9)	19.9	%
CHANNEL/DEVICE UNCERTAINTY REFERENCE EE-092, REV 0	1.57	%
ADDITIONAL CONSERVATISM REFERENCE SM-471 Addendum B PROVIDED A 2.9% DELTA BETWEEN TECH SPEC AND SAFETY ANAL VOLUME (SEE EE-0093, REV 1)	1.33	%
SETPOINT	22.8	%
SETTING TOLERANCE	±.25	%
RESET POINT	23.8	%

REFERENCE CALC SM-471 PROVIDED SETPOINT IN GALLONS; CALC. EE-0093 PROVIDED CONVERSION INTO %.

COMMENTS ______

PREPARED BY R.A. CUNNEEN / RA Cunneen DATE 9 July 92

REVIEWED BY Gary L. Darden / Gary L. Darden DATE 7/9/92

SECTION 3 VERIFICATION OF SETPOINT IMPLEMENTATION

VERIFIED BY ______ DATE ______

VERIFIED BY ______ DATE ______

ATTACH : (1) APPLICABLE MARKED UP SHEETS OF SETPOINT ... LS DOCUMENTS
(2) ADDITIONAL SHEETS FOR ADDITIONAL COMMENTS, MULTIPLE SETPOINT VALUES, ETC.

SETPOINT IMPLEMENTATION / CHANGE FORM

SECTION 1

NORTH ANNA POWER STATION UNIT 1

EQUIPMENT MARK NUMBER 2S-LC-120 A,B,C,D DCP/EWR NUMBER 92-13-1

EQUIPMENT DESCRIPTION RWST LEVEL

REASON FOR CHANGE LOCA CONTAINMENT ANALYSIS

OTHER REFERENCES, DRAWINGS, ETC

(X) SAFETY RELATED ( ) REGULATORY SIGNIFICANT ( ) NON-SAFETY RELATED

SECTION 2

	VALUE	UNITS
SAFETY/PROCESS DESIGN LIMIT	16.5	%
CHANNEL/DEVICE UNCERTAINTY	1.57	%
ADDITIONAL CONSERVATISM	1.33	%

REFERENCE THIS VALUE IS DERIVED (IE → 19.4 - 2.9)

REFERENCE EE-092, REV 0

REFERENCE SM-471 Addendum B PROVIDED A 2.9% DELTA BETWEEN TECH SPEC AND SAFETY ANAL VALUE (SEE EE0093, REV 1)

	VALUE	UNITS
SETPOINT	19.4	%
SETTING TOLERANCE	±.25	%
RESET POINT	20.4	%

REFERENCE

COMMENTS

PREPARED BY R.A. CUNNEEN / RA Cunneen DATE 9 July 92

REVIEWED BY Gary L. Darden / Gary L. Darden DATE 7/9/92

SECTION 3 VERIFICATION OF SETPOINT IMPLEMENTATION

VERIFIED BY ______ DATE ______

VERIFIED BY ______ DATE ______

ATTACH : (1) APPLICABLE MARKED UP SHEETS OF SETPOINT OR PLS DOCUMENTS
(2) ADDITIONAL SHEETS FOR ADDITIONAL COMMENTS, MULTIPLE SETPOINT VALUES, ETC.

SETPOINT IMPLEMENTATION / CHANGE FORM

SECTION 1

Beaver Valley POWER STATION UNIT 1

EQUIPMENT MARK NUMBER ______ DCP/EWR NUMBER 92-13

EQUIPMENT DESCRIPTION SEE ATTACHED

REASON FOR CHANGE CHANGE PLS DOCUMENT TO REFLECT UNIT 1 CHANGE OF NOMINAL Tavg FROM 573.8F TO [illegible]F

OTHER REFERENCES, DRAWINGS, ETC ______

( ) SAFETY RELATED ( ) REGULATORY SIGNIFICANT (X) NON-SAFETY RELATED

SECTION 2

	VALUE	UNITS
SAFETY/PROCESS DESIGN LIMIT	N/A	N/A
CHANNEL/DEVICE UNCERTAINTY	N/A	N/A
ADDITIONAL CONSERVATISM	N/A	N/A
SETPOINT	N/A	N/A
SETTING TOLERANCE	N/A	N/A
RESET POINT	N/A	N/A

REFERENCE (Safety/Process Design Limit) ______

REFERENCE (Channel/Device Uncertainty) ______

REFERENCE (Additional Conservatism) ______

REFERENCE (Setpoint, Setting Tolerance, Reset Point) ______

COMMENTS ______

PREPARED BY T. A. Nolly DATE 8/31/92

REVIEWED BY [signature] DATE 8/31/92

SECTION 3 VERIFICATION OF SETPOINT IMPLEMENTATION

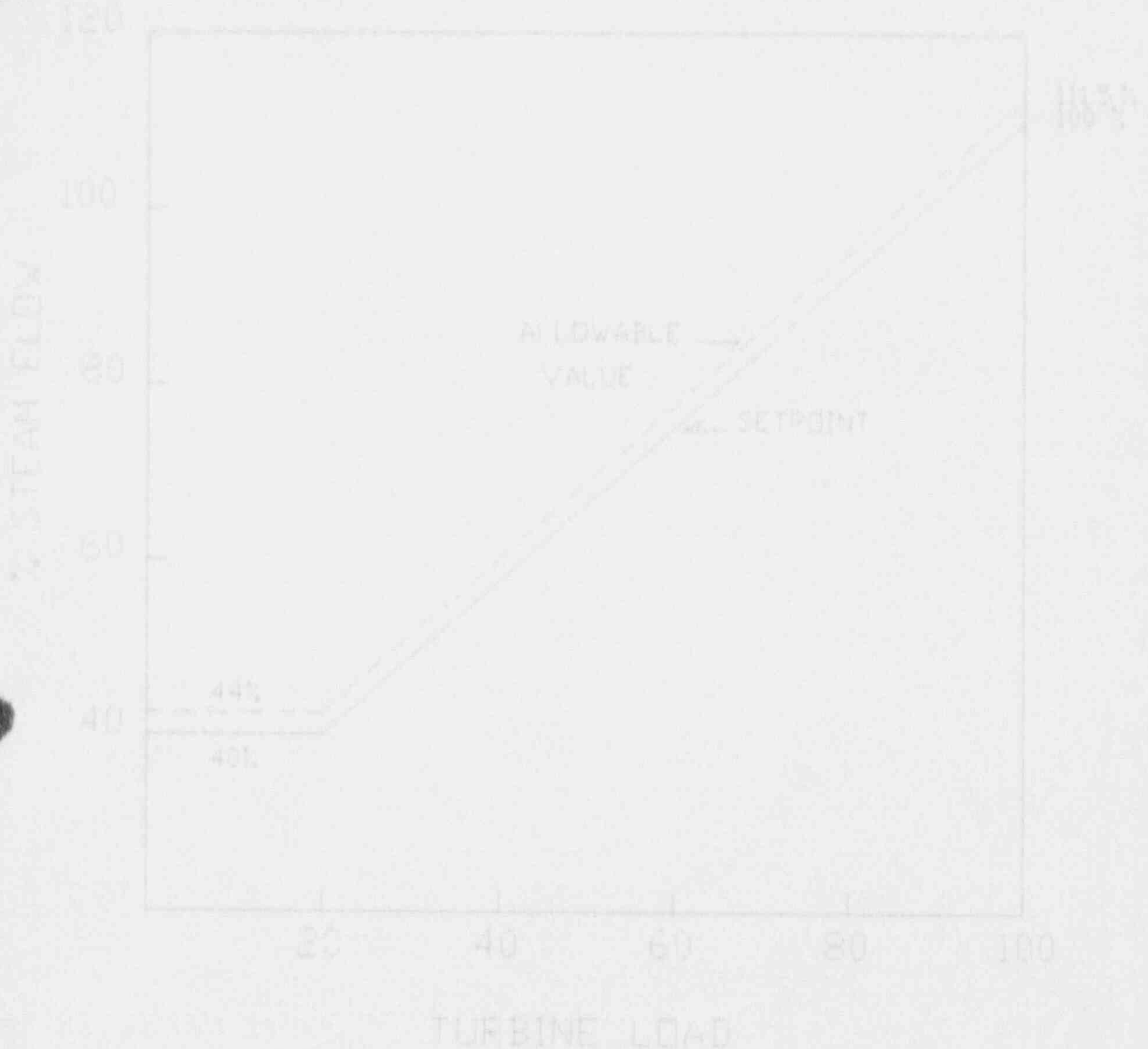
VERIFIED BY ______ DATE ______

VERIFIED BY ______ DATE ______

ATTACH : (1) APPLICABLE MARKED UP SHEETS OF SETPOINT OR PLS DOCUMENTS
(2) ADDITIONAL SHEETS FOR ADDITIONAL COMMENTS, MULTIPLE SETPOINT VALUES, ETC.

FIGURE 1

HIGH STEAM LINE FLOW SAFETY INJECTION SETPOINT VERSUS TURBINE LOAD



NOTE

CALCULATED △ P SPAN = 111.5% OF DESIGN STEAM FLOW (4.76 × 10^6 LBS/HR) AT FULL LOAD STEAM PRESSURE

CALCULATED △ P SPAN = 111.5% OF DESIGN STEAM FLOW (4.49 × 10^6 LBS/HR) AT FULL LOAD STEAM PRESSURE

CALCULATED △ P SPAN = 44% OF DESIGN STEAM FLOW (1.874 × 10^6 LBS/HR) AT NO LOAD STEAM PRESSURE

CALCULATED △ P SPAN = 40% OF DESIGN STEAM FLOW (1.704 × 10^6 LBS/HR) AT NO LOAD STEAM PRESSURE

II CONTROL SYSTEMS

1. REACTOR CONTROL

A. Coolant Average Temperature (Program)

(TM-408F, TM-408G, PM-446, TM-408W)

		Setpoint For* Full Load T_{avg}=586.8°F	Setpoint* For Full Load T_{avg}=580.8°F
1.	High Limit	586.8°F	580.8°F
2.	Low Limit	547°F	547.0°F
3.	Full Power Temperature	586.8°F	580.8°F
4.	Hot Shutdown	547°F	547.0°F
5.	Temperature Gain	0.398°F/% Power Level	0.338°F/% Power Level

6. Lag Time Constant (TM-408C) 30 Seconds[1]

7. Lag Time to PM-446 should be set to "OFF"

B. Coolant Average Temperature (Measured) (TM-408B, TM-408C)

1. Lead Time Constant 20 Seconds [1]

2. Lag Time Constants 10, 1 Seconds [1]

* Setpoints vary linearly from 580°F to 586.8°F

(1) These setpoints may be adjusted during start-up and subsequent operation to optimize control response.

B. Spray Valve Controllers (PC-444C, PC-444D)

Proportional Gain in Percent Spray Valve	4%/% Controller Output(1)
Setpoint where spray is initiated on compensated pressure signal from PC-444A	55% Controller Output(1)

C. Variable Heater Controller

Proportional Gain in Percent Heating Power Per PSI	-6.6%/% Controller Output(1)
Setpoint where proportional heating is full on signal from PC-444A	35% Controller

D.	Power Relief Valve (PCV-455C) operated on compensated pressure signal from PC-444A to PC-444B	92.5% Controller Output
E.	Backup Heaters Turned On, On Compensated Pressure Signal from PC-444A to PC-444F	30% Controller Output
F.	Power Relief Valves (PCV-456 operated on actual pressure) (PC-445A)	2335 PSIG

4. **PRESSURE LEVEL CONTROL**

A. Level Program as Function of T_{avg} (TM-459)

Low Limit For T_{avg} = 547°F

Lower limit will vary linearly for full power T_{avg} between 580.8°F and 586.8°F.

For Full Load T_{avg} = 586.8°F	21.4% of Span
For Full Load T_{avg} = 580.8°F	28.4% of Span

Upper limit for full power T_{avg} between 580.8°F and 586.8°F	64.5% of Span

(1) These setpoints may be adjusted during start-up and subsequent operation to optimize control response.

(4) Unit 2 Valve

$Z_{Lo\text{-}Lo} = K_1 T_{avg} + K_2 \Delta T + K_3$ — $K_1 = 0$

K_2 (See Fig. 9)

K_3 (See Fig. 9)

B. Low Alarm
(TC-409D, TC-409E, TC-409F, TC-409L)

$Z_{Lo} = Z_{Lo\text{-}Lo} + K_4$ — K_4 = 10 Steps (6.25 Inches)

3. <u>Temperature Alarms</u>

A. High T_{avg} (4°F above the nominal T_{avg} at full load) — 584.8°F

(TC-412D, TC-422D, TC-432D)

(3) These alarm setpoints are expected to be adjusted during start-up and subsequent operation such that they are just beyond the range of normal operation variations.

Attachment 6.1

SETPOINT IMPLEMENTATION / CHANGE FORM — STD-GN-0030, REV. 3

SECTION 1

North Anna POWER STATION UNIT 1

EQUIPMENT MARK NUMBER Various DCP/~~EWR NUMBER~~ 92-13

EQUIPMENT DESCRIPTION [illegible]

REASON FOR CHANGE [illegible]

OTHER REFERENCES, DRAWINGS, ETC Tech Spec

(✓) SAFETY RELATED ( ) REGULATORY SIGNIFICANT ( ) NON-SAFETY RELATED

SECTION 2

	VALUE	UNITS
SAFETY/PROCESS DESIGN LIMIT		
REFERENCE		
CHANNEL/DEVICE UNCERTAINTY		
REFERENCE		
ADDITIONAL CONSERVATISM		
REFERENCE		
SETPOINT		
SETTING TOLERANCE		
RESET POINT		
REFERENCE		

COMMENTS See attached markup of PLS & DBD

PREPARED BY [illegible] DATE 9/14/92

REVIEWED BY W Blakes [illegible] DATE 9/14/92

SECTION 3 VERIFICATION OF SETPOINT IMPLEMENTATION

VERIFIED BY ____ DATE ____

VERIFIED BY ____ DATE ____

ATTACH : (1) APPLICABLE MARKED UP SHEETS OF SETPOINT OR PLS DOCUMENTS
(2) ADDITIONAL SHEETS FOR ADDITIONAL COMMENTS, MULTIPLE SETPOINT VALUES, ETC.

4. Automatic reset of manual block on high pressurizer pressure (P-11) — 2000 psig

(PC-455B, PC-456B, PC-457B)

5. High-1 containment pressure — Not Westinghouse Scope, set for about 10% of CDP

6. Time delay on manual reset of SI — 1 minute

B. Steam Line Isolation

1. High steam line flow — See I.1.A.3 above

2. High-2 containment pressure — Not Westinghouse Scope, set for about 33% of CDP

C. Containment Spray Actuation

1. High-3 containment pressure — Not Westinghouse Scope, set for about 50% of CDP

2. Reactor Trips

A. Nuclear Instrumentation

1. Source range high level — 10^5 counts/sec(2)

(NC-31D, NC-32D)

2. Intermediate range high level — Current equivalent to 25% of full power

(NC-35F, NC-36F)

3. Power range, low range, high level — 25% of full power

(NC-41P, NC-42P, NC-43P, N-44P)

4. Power range, high range, high level — 109% of full power

(NC-41R, NC-42R, NC-43R, NC-44R)

5. Power range, high neutron flux rate — +5% of rated power (positive) -3% of rated power (negative)

(NC-41U, NC-42U, NC-43U, NC 44U)

(NC-41K, NC-42K, NC-43K, NC-44K)

Impulse unit time constant — ≥ 2 seconds

(NM-41J, NM-42J, NM-43J, NM-44J)

(2) Values may be adjusted during initial start-up tests.

where,

T = Average temperature, °F

P = Pressurizer pressure, psig

K_1 = 126.4%

K_2 = 2.20% per °F

K_3 = 0.1152% per psi

(TM-412E, TM-422E, TM-432E) t_1 = 25 seconds

t_2 = 4 seconds

(NM-412C, NM-422C, NM-432C) $f(\Delta q)$ = (See item 4, P. 18)

b. Overtemperature ΔT turbine runback and block of rod withdrawal (C-3)

Identical to the reactor trip setpoint above except K_1 becomes 123.4%.

Turbine runback time delay relay — on: 1.5 seconds; off: 28.5 seconds

Turbine load reference reduction rate — 200% per minute

The above values of K_1 are applicable to three loop operation.

For two loop operation, see "Setpoints for Two Loop Operation".

2. Setpoint for overpower ΔT trip and turbine runback

(TM-412G, TC-412B)
(TM-422G, TC-422B)
(TM-432G, TC-432B)

a. Overpower ΔT reactor trip setpoint

$$\Delta T_{sp} = K_4 - K_5 \left(\frac{t_3 S}{1 + t_3 S}\right) T - K_6 (T - T') - f(\Delta q)$$

where,

T = Average temperature, °F

T' = Indicated average temperature at nominal conditions and rated power, for the channel being calibrated (for plant startup, assume T' = 586.8°F)

K_4 = 107.9%

K_5 = 0 for decreasing average temperature
2% per °F for increasing average temperature

K_6 = 0.164% per °F for T ≧ T'
0 for T < T'

(TM-412L, TM-422L, TM-432L) t_3 = 10 seconds

(NM-412B, NM-422B, NM-432B) f(Δq) = (set to zero)

b. Overpower ΔT turbine runback and block of rod withdrawal (4)

Identical to the reactor trip setpoint above except K_4 becomes 104.9%.

Runback rate See Item 1b above

3. Nuclear calibration for ΔT trips

During plant startup tests, all eight calibrated voltage signals from the power range nuclear channels are to be calibrated from core power distribution measurements such that the same signal (8.33 volts at output of NM-41K/L and duplicates) is obtained for the reference flat condition. The reference flat power condition is defined as rated core power with nominal

(4) Unit 2 Value

6. Rod Control System

Control Bank Overlap (Thumbwheel switches in rod control logic cabinet)

S1	(Start B with A)	128 steps
S2	(Stop A)	228 steps
S3	(Start C with B)	256 steps
S4	(Stop B)	356 steps
S5	(Start D with C)	384 steps
S6	(Stop C)	484 steps

128
12

7. Part Length Rod Limit Switches

Lower Electrical

Limit Switch Setpoint — 5 steps

Upper Electrical

Limit Switch Setpoint — 227 steps

III. Alarms

1. Rodstops

In Sections I.2.B.1 and I.2.B.2, automatic and manual rod withdrawal block accompanies turbine runback; also,

A. Power Range High Flux Level — 103% of full power

(NC-41L, NC-42L, NC-43L, NC-44L)

B. Low Turbine Load Cutout of Automatic Rod Withdrawal (C-5) (No alarm provided - permissive status light only) — 15% of full power

(PC-446B)

C. Nuclear Intermediate Range High Flux — Current equivalent to 20% full power

(NC-35E, NC-36E)

2. Insertion Limit Alarms

A. Lo-Lo Alarm

(TM-409A, TM-409B, TM-409C, TM-409D)
(TC-409D, TC-409E, TC-409F, TC-409L)

INSTRUMENT NUMBER	DESCRIPTION	TR	SETPOINT VALUE	UNITS	INC	DEC	RESET	ALARM	LOOP DIAGRAM	SOURCE	REV.
	REACTOR TRIPS										
N/A	MANUAL REACTOR TRIP - 1 OF 2	X	N/A	N/A	-	-		1D-A1		TS-2.2.1B	
NI-NC-41P	LOW POWER RANGE HIGH - RX TRIP 2 OF 4	X	25	PERCENT	X			1L-C1		TS-2.2.1B	
NI-NC-42P	LOW POWER RANGE HIGH - RX TRIP 2 OF 4	X	25	PERCENT	X			1L-C2		TS-2.2.1B	
NI-NC-43P	LOW POWER RANGE HIGH - RX TRIP 2 OF 4	X	25	PERCENT	X			1L-C3		TS-2.2.1B	
NI-NC-44P	LOW POWER RANGE HIGH - RX TRIP 2 OF 4	X	25	PERCENT	X			1L-C4		TS-2.2.1B	
NI-NC-41R	HIGH POWER RANGE HIGH - RX TRIP 2 OF 4	X	109*	PERCENT	X			1L-D1		TS-2.2.1B	
NI-NC-42R	HIGH POWER RANGE HIGH - RX TRIP 2 OF 4	X	109*	PERCENT	X			1L-D2		TS-2.2.1B	
NI-NC-43R	HIGH POWER RANGE HIGH - RX TRIP 2 OF 4	X	109*	PERCENT	X			1L-D3		TS-2.2.1B	
NI-NC-44R	HIGH POWER RANGE HIGH - RX TRIP 2 OF 4	X	109*	PERCENT	X			1L-D4		TS-2.2.1B	
NI-NC-41U	HIGH POWER RANGE NEUT FLUX (+) - RX TRIP 2 OF 4	X	5	PERCENT	X			1L-E1		TS-2.2.1B	
NI-NC-42U	HIGH POWER RANGE NEUT FLUX (+) - RX TRIP 2 OF 4	X	5	PERCENT	X			1L-E2		TS-2.2.1B	
NI-NC-43U	HIGH POWER RANGE NEUT FLUX (+) - RX TRIP 2 OF 4	X	5	PERCENT	X			1L-E3		TS-2.2.1B	
NI-NC-44U	HIGH POWER RANGE NEUT FLUX (+) - RX TRIP 2 OF 4	X	5	PERCENT	X			1L-E4		TS-2.2.1B	
NI-NC-41K	HIGH POWER RANGE NEUT FLUX (-) - RX TRIP 2 OF 4	X	5	PERCENT		X		1L-E1		TS-2.2.1B	
NI-NC-42K	HIGH POWER RANGE NEUT FLUX (-) - RX TRIP 2 OF 4	X	5	PERCENT		X		1L-E2		TS-2.2.1B	
NI-NC-43K	HIGH POWER RANGE NEUT FLUX (-) - RX TRIP 2 OF 4	X	5	PERCENT		X		1L-E3		TS-2.2.1B	
NI-NC-44K	HIGH POWER RANGE NEUT FLUX (-) - RX TRIP 2 OF 4	X	5	PERCENT		X		1L-E4		TS-2.2.1B	
NI-NC-35F	INTERMEDIATE RANGE HIGH - RX TRIP 1 OF 2	X	25	PERCENT	X			1L-B1		TS-2.2.1B	
NI-NC-36F	INTERMEDIATE RANGE HIGH - RX TRIP 1 OF 2	X	25	PERCENT	X			1L-B2		TS-2.2.1B	

* REDUCED TO 103% FOR THE PERIOD OF OPERATION UNTIL STEAM GENERATOR REPLACEMENT PER EWR 92-092 S

INSTRUMENT NUMBER	DESCRIPTION	SR	SETPOINT VALUE	SETPOINT UNITS	INC	DEC	RESET	ALARM	LOOP DIAGRAM	SOURCE	REV.
**NI-NC-31D	SOURCE RANGE HIGH - RX TRIP		1.0E5	CPS	X			1D-C4		TS-2.2.1	
**NI-NC-32D	SOURCE RANGE HIGH - RX TRIP		1.0E5	CPS	X			1D-C4		TS-2.2.1	
NI-NC-35D	PERMISSIVE P-6		1.0E-10	AMPS	X			1L-F1		TS-3.1	
NI-NC-36D	PERMISSIVE P-6		1.0E-10	AMPS	X			1L-F2		TS-3.1	
NI-NC-35E	INTERMEDIATE RANGE - ROD STOP		20	PERCENT	X			1A-C5		PLS(39)	
NI-NC-36E	INTERMEDIATE RANGE - ROD STOP		20	PERCENT	X			1A-C5		PLS(39)	
**NI-NC-35F	INTERMEDIATE RANGE HIGH - RX TRIP		25	PERCENT	X			1D-C3		TS-2.2.1	
**NI-NC-36F	INTERMEDIATE RANGE HIGH - RX TRIP		25	PERCENT	X			1D-C3		TS-2.2.1	
NI-NM-41J	POWER RANGE IMPULSE TIMER		2.25	SECONDS	X			N/A		SP79-05	
NI-NM-42J	POWER RANGE IMPULSE TIMER		2.25	SECONDS	X			N/A		SP79-05	
NI-NM-43J	POWER RANGE IMPULSE TIMER		2.25	SECONDS	X			N/A		SP79-05	
NI-NM-44J	POWER RANGE IMPULSE TIMER		2.25	SECONDS	X			N/A		SP79-05	
**NI-NC-41K	HIGH POWER RANGE NEUTRON FLUX (-) - RX TRIP		3	PERCENT	X			1D-E4		TS-2.2.1	
**NI-NC-42K	HIGH POWER RANGE NEUTRON FLUX (-) - RX TRIP		3	PERCENT	X			1D-E4		TS-2.2.1	
**NI-NC-43K	HIGH POWER RANGE NEUTRON FLUX (-) - RX TRIP		3	PERCENT	X			1D-E4		TS-2.2.1	
**NI-NC-44K	HIGH POWER RANGE NEUTRON FLUX (-) - RX TRIP		3	PERCENT	X			1D-E4		TS-2.2.1	
NI-NC-41L	HIGH POWER RANGE FLUX - RODSTOP INTLK		103*	PERCENT	X			1A-D4		PLS(39)S	
NI-NC-42L	HIGH POWER RANGE FLUX - RODSTOP INTLK		103*	PERCENT	X			1A-D4		PLS(39)S	
NI-NC-43L	HIGH POWER RANGE FLUX - RODSTOP INTLK		103*	PERCENT	X			1A-D4		PLS(39)S	

* REDUCED TO 98 FOR THE PERIOD OF OPERATION UNTIL STEAM GENERATOR REPLACEMENT PER EWR 92-092 S
** DUPLICATE OF INSTRUMENT NUMBER SHOWN IN SECTION A1 DELETED BY EWR 92-141A.
INFORMATION SHOWN WITH THIS ITEM IS NO LONGER EFFECTIVE.

INSTRUMENT NUMBER	DESCRIPTION	SR	SETPOINT VALUE	UNITS	INC	DEC	RESET	ALARM	LOOP DIAGRAM	SOURCE	REV.
NI-NC-44L	HIGH POWER RANGE FLUX - RODSTOP INTLK		103*	PERCENT	X			1A-D4		PLS(39)S	
NI-NC-41M	PERMISSIVE P-7 (RX POWER), P-10		10	PERCENT	X			1L-G1		TS-3.1	
NI-NC-42M	PERMISSIVE P-7 (RX POWER), P-10		10	PERCENT	X			1L-G2		TS-3.1	
NI-NC-43M	PERMISSIVE P-7 (RX POWER), P-10		10	PERCENT	X			1L-G3		TS-3.1	
NI-NC-44M	PERMISSIVE P-7 (RX POWER), P-10		10	PERCENT	X			1L-G4		TS-3.1	
NI-NC-41N	PERMISSIVE P-8		30	PERCENT	X			1L-H1		TS-3.1	
NI-NC-42N	PERMISSIVE P-8		30	PERCENT	X			1L-H2		TS-3.1	
NI-NC-43N	PERMISSIVE P-8		30	PERCENT	X			1L-H3		TS-3.1	
NI-NC-44N	PERMISSIVE P-8		30	PERCENT	X			1L-H4		TS-3.1	
#NI-NC-41P	LOW POWER RANGE HIGH - RX TRIP		25	PERCENT	X			1D-C2		TS-2.2.1	
#NI-NC-42P	LOW POWER RANGE HIGH - RX TRIP		25	PERCENT	X			1D-C2		TS-2.2.1	
#NI-NC-43P	LOW POWER RANGE HIGH - RX TRIP		25	PERCENT	X			1D-C2		TS-2.2.1	
#NI-NC-44P	LOW POWER RANGE HIGH - RX TRIP		25	PERCENT	X			1D-C2		TS-2.2.1	
#NI-NC-41R	HIGH POWER RANGE HIGH - RX TRIP		109**	PERCENT	X			1D-C1		TS-2.2.1B	
#NI-NC-42R	HIGH POWER RANGE HIGH - RX TRIP		109**	PERCENT	X			1D-C1		TS-2.2.1B	
#NI-NC-43R	HIGH POWER RANGE HIGH - RX TRIP		109**	PERCENT	X			1D-C1		TS-2.2.1B	
#NI-NC-44R	HIGH POWER RANGE HIGH - RX TRIP		109**	PERCENT	X			1D-C1		TS-2.2.1B	
#NI-NC-41U	HIGH POWER RANGE NEUTRON FLUX (+) - RX TRIP		5	PERCENT	X			1D-E4		TS-2.2.1	
#NI-NC-42U	HIGH POWER RANGE NEUTRON FLUX (+) - RX TRIP		5	PERCENT	X			1D-E4		TS-2.2.1	

* REDUCED TO 98 FOR THE PERIOD OF OPERATION UNTIL STEAM GENERATOR REPLACEMENT PER EWR 92-092 S
** REDUCED TO 103 FOR THE PERIOD OF OPERATION UNTIL STEAM GENERATOR REPLACEMENT PER EWR 92-092 S
# DUPLICATE OF INSTRUMENT NUMBER SHOWN IN SECTION A1 DELETED BY EWR 92-141A.
INFORMATION SHOWN WITH THIS ITEM NO LONGER EFFECTIVE.

INSTRUMENT NUMBER	DESCRIPTION	SR	SETPOINT VALUE	UNITS	INC	DEC	RESET	ALARM	LOOP DIAGRAM	SOURCE	REV.
TC-RC1409D-2	LOW ROD INSERTION LIMIT ALARM		VARIABLE	DEG.F.		X		N/A	RC-087	PLS(39)	
TC-RC1409E-1	LO-LO ROD INSERTION LIMIT ALARM		VARIABLE	DEG.F.		X		N/A	RC-087	PLS(39)	
TC-RC1409E-2	LOW ROD INSERTION LIMIT ALARM		VARIABLE	DEG.F.		X		N/A	RC-087	PLS(39)	
TC-RC1[illegible]	LO-LO ROD INSERTION LIMIT ALARM		VARIABLE	DEG.F.		X		N/A	RC-087	PLS(39)	
TC-RC1[illegible]	LOW ROD INSERTION LIMIT ALARM		VARIABLE	DEG.F.		X		N/A	RC-087	PLS(39)	
TC-RC1409K	DELTA T CONTROL		N/A	N/A				N/A	RC-087	PLS(39)	
TC-RC1409L-1	LO-LO ROD INSERTION LIMIT ALARM		VARIABLE	DEG.F.		X		N/A	RC-087	PLS(39)	
TC-RC1409L-2	LOW ROD INSERTION LIMIT ALARM		VARIABLE	DEG.F.		X		N/A	RC-087	PLS(39)	
TC-RC1410	RCS LOOP 1 STOP VALVE OPEN INTLK		5	DEG.F.	X			1P-C5	RC-121	PLS(21)	
TC-RC1413	RCS LOOP 1 STOP VALVE OPEN INTLK		5	DEG.F.	X			1P-D5	RC-124	PLS(21)	
TC-RC1420	RCS LOOP 2 STOP VALVE OPEN INTLK		5	DEG.F.	X			1P-C6	RC-122	PLS(21)	
TC-RC1423	RCS LOOP 2 STOP VALVE OPEN INTLK		5	DEG.F.	X			1P-D6	RC-125	PLS(21)	
TC-RC1430	RCS LOOP 3 STOP VALVE OPEN INTLK		5	DEG.F.	X			1P-C7	RC-123	PLS(21)	
TC-RC1433	RCS LOOP 3 STOP VALVE OPEN INTLK		5	DEG.F.	X			1P-D7	RC-126	PLS(21)	
* TC-RC1412B	OVER~~TEMP~~ Power DELTA T - RX TRIP/~~TURB RUNBACK~~ - CKT 1	X	~~(109.1%)~~ VARIABLE*	ΔT	X		[illegible]	1M-G1/1D-A2	RC-091	EWR 91-081	
* TC-RC1412B	OVER~~TEMP~~ Power DELTA T - ~~RX TRIP~~/TURB RUNBACK - CKT 2	X	~~106.1%~~ VARIABLE*	ΔT	X		[illegible]	1M-F1/1P-E4	RC-091	EWR 91-081	
* TC-RC1422B	OVER~~TEMP~~ Power DELTA T - RX TRIP/~~TURB RUNBACK~~ - CKT 1	X	~~(109.1%)~~ VARIABLE*	ΔT	X		[illegible]	1M-G2/1D-A2	RC-093	EWR 91-081	
* TC-RC1422B	OVER~~TEMP~~ Power DELTA T - ~~RX TRIP~~/TURB RUNBACK - CKT 2	X	~~106.1%~~ VARIABLE*	ΔT	X		[illegible]	1M-F2/1P-E4	RC-093	EWR 91-081	
* TC-RC1432B	OVER~~TEMP~~ Power DELTA T - RX TRIP/~~TURB RUNBACK~~ - CKT 1	X	~~109.1%~~ VARIABLE*	ΔT	X			1M-G3/1D-A2	RC-095	EWR 91-081	
TC-RC1432B	OVERPOWER DELTA T - TURB RUNBACK - CKT 2	X	VARIABLE*	ΔT	X			1M-F3/1P-E6	RC-095	EWR 91-081	
**TC-RC1412D	LOW TAVE RX TRIP OVERRIDE - FW VALVE CLOSE		554	DEG.F.		X		N/A	RC-091	PLS(37)	
**TC-RC1412D-2	HIGH TAVE ALARM - FW VALVE CLOSE		587.0	DEG.F.	X			N/A	RC-091	DCP89-15-3	4
**TC-RC1422D-1	LOW TAVE RX TRIP OVERRIDE - FW VALVE CLOSE		554	DEG.F.		X		N/A	RC-[illegible]	PLS(37)	
**TC-RC1422D-2	HIGH TAVE ALARM - FW VALVE CLOSE		587.0	DEG.F.	X			N/A	RC-093	DCP89-15-3	4

*NOTE: DUPLICATE OF INSTRUMENT NUMBER SHOWN IN SECTION A1 DELETED BY DCP 86-02-1. INFORMATION SHOWN WITH THIS ITEM IS NO LONGER EFFECTIVE.

**NOTE: DUPLICATE OF INSTRUMENT NUMBER SHOWN ON PAGE 112 OF SECTION B1 DELETED BY EWR 92-017. INFORMATION SHOWN WITH THIS ITEM IS NO LONGER EFFECTIVE.

* See Section A1 Page 2

DC 90-13-1, Appendix 4-7, Page 10

Place on page 111 as shown and delete on this page

INSTRUMENT NUMBER	DESCRIPTION	SR	SETPOINT VALUE	SETPOINT UNITS	SETPOINT INC	SETPOINT DEC	RESET	ALARM	LOOP DIAGRAM	SOURCE	REV.
*TC-RC1432B	OVERTEMP ΔT - RX TRIP/TURB RUNBACK - CKT 2	X	106.1%	ΔT	X			1M-F3/1P-E4	RC-095	EWR 91-081	
*TC-RC1412C	OVERTEMP ΔT - RX TRIP/~~TURB RUNBACK~~ - CKT 1	X	VARIABLE* ~~(108.5%)~~	ΔT	X			1M-D1/1D-A3	RC-091	EWR 91-081	
*TC-RC1412C	OVERTEMP ΔT - ~~RX TRIP~~/TURB RUNBACK - CKT 2	X	VARIABLE* ~~105.5%~~	ΔT	X			1M-E1/1P-E3	RC-091	EWR 91-081	
*TC-RC1422C	OVERTEMP ΔT - RX TRIP/~~TURB RUNBACK~~ - CKT 1	X	VARIABLE* ~~(108.5%)~~	ΔT	X			1M-D2/1D-A3	RC-093	EWR 91-081	
*TC-RC1422C	OVERTEMP ΔT - ~~RX TRIP~~/TURB RUNBACK - CKT 2	X	VARIABLE* ~~105.5%~~	ΔT	X			1M-E2/1P-E3	RC-093	EWR 91-081	
*TC-RC1432C	OVERTEMP ΔT - RX TRIP/~~TURB RUNBACK~~ - CKT 1	X	VARIABLE* ~~108.5%~~	ΔT	X			1M-D3/1D-A3	RC-095	EWR 91-081	
*TC-RC1432C	OVERTEMP ΔT - ~~RX TRIP~~/TURB RUNBACK - CKT 2	X	VARIABLE ~~105.5%~~	ΔT	X			1M-E3/1P-E3	RC-095	EWR 91-081	
TC-RC1412D	LOW TAVE FW VALVE ISOLATION COINCIDENT WITH P-4 (RX TRIP) - CKT 1	X	554	DEG. F.		X		1L-E5	RC-091	EWR92-017I; PLS (37)S	
TC-RC1412D	HIGH TAVE ALARM - CKT 2	X	587	DEG. F.	X			1B-B4	RC-091	EWR92-017S	
TC-RC1422D	LOW TAVE FW VALVE ISOLATION COINCIDENT WITH P-4 (RX TRIP) - CKT 1	X	554	DEG. F.		X		1L-E6	RC-093	EWR92-017I; PLS (37)S	
TC-RC1422D	HIGH TAVE ALARM - CKT 2	X	587	DEG. F.	X			1B-B4	RC-093	EWR92-017S	
TC-RC1432D	LOW TAVE FW VALVE ISOLATION COINCIDENT WITH P-4 (RX TRIP) - CKT 1	X	554	DEG. F.		X		1L-E7	RC-095	EWR92-017I; PLS (37)S	
TC-RC1432D	HIGH TAVE ALARM - CKT 2	X	587	DEG. F.	X			1B-B4	RC-095	EWR92-017S	
TC-RC1412E	LO-LO TAVG - SI ACTUATION AND P-12 INTLK	X	543	DEG.F.		X		1L-F5	RC-091	EWR92-017I; TS-3.3.2B	
TC-RC1422E	LO-LO TAVG - SI ACTUATION AND P-12 INTLK	X	543	DEG.F.		X		1L-F6	RC-093	EWR92-017I; TS-3.3.2B	
TC-RC1432E	LO-LO TAVG - SI ACTUATION AND P-12 INTLK	X	543	DEG.F.		X		1L-F7	RC-095	EWR92-017I; TS-3.3.2B	
TC-RC1417A	RCTR CLNT PMP MOTOR THRUST BEARNG UPR SHOE-HI ALM		185	DEG.F.	X			1C-H4	RC-001	PLS(65)	
TC-RC1427A	RCTR CLNT PMP MOTOR THRUST BEARNG UPR SHOE-HI ALM		185	DEG.F.	X			1C-H4	RC-006	PLS(65)	
TC-RC1437A	RCTR CLNT PMP MOTOR THRUST BEARNG UPR SHOE-HI ALM		185	DEG.F.	X			1C-H4	RC-011	PLS(65)	
TC-RC1417B	RCTR CLNT PMP MOTOR THRUST BEARNG LWR SHOE-HI ALM		185	DEG.F.	X			1C-H4	RC-002	PLS(65)	
TC-RC1427B	RCTR CLNT PMP MOTOR THRUST BEARNG LWR SHOE-HI ALM		185	DEG.F.	X			1C-H4	RC-007	PLS(65)	
TC-RC1437B	RCTR CLNT PMP MOTOR THRUST BEARNG LWR SHOE-HI ALM		185	DEG.F.	X			1C-H4	RC-012	PLS(65)	

*NOTE: DUPLICATE OF INSTRUMENT NUMBER SHOWN IN SECTION A1 DELETED BY DCP 86-02-1. INFORMATION SHOWN WITH THIS ITEM IS NO LONGER EFFECTIVE.

See Section A1 page 2

ALARA EVALUATION AND DESIGN CHECKLIST

PRELIMINARY ALARA EVALUATION

(To be completed by Design Engineering Organization)

Project Title: STEAM GENERATOR REPAIR - NORTH ANNA UNIT 1

Project Engineer: SHERI L. WILKIE

DCP No. or EWR No.: DC-90-13-1   Phone No. (703) 894-8010

Location: NORTH ANNA POWER STATION, SG REPLACEMENT PROJECT

System: RCS, MS, FW, SG BLOWDOWN

Description of DCP or EWR: RESTORE THE INTEGRITY OF THE STEAM GENERATORS TO A LEVEL EQUIVALENT TO NEW GENERATORS BY REMOVAL AND REPLACEMENT OF THE LOWER ASSEMBLIES.

1. Will this DCP or EWR require activities which must be performed in, or require entry to, a radiation controlled area? [X] Yes [ ] No

2. Will this DCP or EWR involve receiving, shipping, releasing, discharging, processing, conveying, or sampling of radioactive material? [X] Yes [ ] No

3. Will this DCP or EWR create a new radiation source or a new radiation area onsite or cause an increase in dose rates from an existing source? [ ] Yes [X] No

4. Will this DCP or EWR create or increase routine maintenance, operation, service, or surveillance requirements in a radiation area? [ ] Yes [X] No

5. Will this DCP or EWR involve shielding changes, ventilation changes, or materials that contribute to radioactive crud, resin, or sludge treatment systems? [ ] Yes [X] No

6. Does this DCP or EWR involve the replacement of valves or valve internals on the primary system which contain cobalt alloys? [ ] Yes [X] No

Additional ALARA Considerations/Comments:
RE: ITEM #3) A NEW RADIATION AREA WILL BE CREATED BY THE STORAGE FACILITY FOR THE OLD SG'S. THE STORAGE FACILITY IS NOT PART OF THIS DC.
RE: ITEM #6) NO PRIMARY SYS VALVES ARE AFFECTED BY THIS DC.

Completed by: MS Bath   Organization: BECHTEL   Date: 6/22/92

PRELIMINARY ALARA EVALUATION REVIEW

(To be completed by ALARA Coordinator)

Reviewer's Comments (optional): Done

(2)

Reviewed by: James R. Shleser
ALARA Coordinator

6/24/92
Date

ALARA DESIGN CHECKLIST

NOTE: This ALARA design checklist reflects specific items that should be considered by the responsible Discipline Engineer during the initial phases of design. The list is not all-inclusive but rather provides guidance and examples of ALARA design considerations. All sections must be answered. If questions are not applicable, so state. If questions are applicable, provide some explanation to the response.

(To be completed by Discipline Engineer)

Project Title: STEAM GENERATOR REPAIR - NORTH ANNA UNIT 1

Project Engineer: ______

DCP No. or EWR No.: DC-90-13-1 Discipline Engineer: ______

Location: ______ Phone No.: ______

System: RCS, MS, FW, SG BLOWDOWN

SECTION 1 - LOCATION, RADIATION LEVELS, AND SHIELDING

		YES	NO	N/A
1.	Have recent radiation surveys been reviewed to ensure that equipment and components are located away from hot spots and high radiation areas, where practicable? COMPONENTS THAT ARE REMOVED FOR SGR WILL BE REINSTALLED IN THEIR ORIGINAL LOCATION AND CONFIGURATION.	[X]	[ ]	[ ]
2.	Are high occupancy areas established at the lowest practical dose rate? ______	[ ]	[ ]	[X]
3.	Has shielding been placed between serviceable components and substantial radiation source(s) in the area where practical? TEMPORARY SHIELDING, AS REQUIRED WILL BE USED TO REDUCE HIGH DOSE RATES.	[X]	[ ]	[ ]
4.	Is maximum distance provided between serviceable components and substantial radiation source(s) in the area? ______	[ ]	[ ]	[X]

		YES	NO	N/A
5.	Do design features prevent personnel from inadvertently entering areas where a significant dose could be received in a short period of time? SYSTEMS AND COMPONENTS TO BE REINSTALLED TO MEET OR EXCEED EXISTING DESIGN REQUIREMENTS	[X]	[ ]	[ ]
6.	Has the use of permanent shielding been considered to prevent the need for repeated shielding reinstallation? NO PERMANENT PLANT SHIELDING IS INSTALLED BY THIS DC. PERMANENT CLOSURE COVERS FOR THE NOZZLES AND TRANS. CONE ARE PLACED FOR SHIELDING + CONTAM. CONTROL OF THE OLD SG'S.	[ ]	[X]	[ ]
7.	If permanent shielding is not feasible, are provisions incorporated for rapid installation of temporary shielding?	[ ]	[ ]	[X]
8.	Are normal travel patterns established so that workers will not have to frequently go into or near high radiation areas?	[ ]	[ ]	[X]
9.	Has shielding been used to prevent streaming of radiation through pipe and duct penetrations? NO NEW PENETRATIONS ARE ADDED BY THIS DC	[ ]	[ ]	[X]
10.	Are penetrations positioned high in shield walls (i.e., greater than 8 foot from floor? NO NEW PENETRATIONS ARE ADDED BY THIS DC.	[ ]	[ ]	[X]
11.	Will design change avoid producing post-accident radiation fields that may hamper recovery operations or degrade safety equipment?	[ ]	[ ]	[X]

SECTION 2 - MAINTENANCE, OPERATION AND OTHER CONSIDERATIONS

		YES	NO	N/A
1.	Have system components been selected based on long service life, ease of maintenance and low maintenance frequency? IMPROVEMENTS IN REPL. COMPONENT DESIGN AND MATERIALS WILL ENSURE IMPROVEMENTS IN THE ABOVE AREAS.	[X]	[ ]	[ ]

CA1

		YES	NO	N/A
2.	Have reach rods, special tooling or remotely operated valves and controls been used where practicable?	[ ]	[ ]	[X]
3.	Are necessary services (e.g., lighting, air water, electrical) provided or planned for? TEMPORARY FACILITIES HAVE BEEN PROVIDED TO SUPPORT THE DESIGN CHANGE PACKAGE	[X]	[ ]	[ ]
4.	Has the area been designed to accommodate worker and equipment accessibility?	[ ]	[ ]	[X]
5.	Have permanently installed platforms, lighting, ladders, etc., been considered for maintenance access and operations? NO NEW ITEMS PLANNED PER THIS DC (SEE DC 90-06-01)	[ ]	[ ]	[X]
6.	Does insulation design allow for rapid removal and replacement? NEW SG INSULATION IS DESIGNED REMOVABLE FOR EASE OF REMOVAL AND REPLACEMENT.	[X]	[ ]	[ ]
7.	Are components requiring frequent maintenance (e.g., small pumps and valves) designed to permit prompt removal to facilitate repairs in lower radiation areas? REPLACEMENT VALVES & INSTRUMENTS ARE OF THE SAME TYPE AND CONFIGURATION AS CURRENT COMPONENTS	[X]	[ ]	[ ]
8.	Are area communication systems and alarm systems adequate to override expected noise levels?	[ ]	[ ]	[X]
9.	Has equipment been laid out with consideration given to facilitating inspections required by Section XI of the ASME Code and other requirements of the ISI program? INSULATION HAS BEEN DESIGNED REMOVABLE TO ALLOW ACCESS FOR ISI. COMPONENT CONFIGURATIONS ARE UNCHANGED	[X]	[ ]	[ ]

		YES	NO	N/A
10.	Does design allow surveillance to be performed from outside high radiation areas, where practical, through the use of remote readout devices, viewing points, radiation detector ports or TV cameras?	[ ]	[ ]	[X]
11.	Are clearances through doorways, equipment manways, and structural components adequate for installation, maintenance, and inspection? EXISTING CLEARANCES HAVE BEEN EVALUATED AND ARE ADEQUATE.	[X]	[ ]	[ ]
12.	Have provisions been incorporated to allow rigging of component(s), e.g., pad eyes? EXTENSIVE PLANNING AND PREPARATION WILL PRECEDE THE RIGGING ACTIVITIES AS PART OF THE ENGINEERED DESIGN	[X]	[ ]	[ ]
13.	Have lay-down areas and pull-space for equipment been considered? PULL SPACE AND LAY DOWN AREAS FOR THE STEAM DOMES AND LOWER ASSEMBLIES HAVE BEEN EVALUATED AS PART OF THE ENGINEERED DESIGN.	[X]	[ ]	[ ]
14.	Have design features been used that minimized or eliminated required inspection, calibration, testings, and maintenance where practical? REPLACEMENT OF THE SG LOWER ASSEMBLIES, PORTIONS OF FW + BD PIPING AND INSULATION WILL REDUCE MAINTENANCE FREQUENCY AND EFFORT.	[X]	[ ]	[ ]
15.	Have lubricating systems or self lubricating units been considered?	[ ]	[ ]	[X]
16.	Are special tools available for TRIAL use on the component prior to and during installation? MOCK UP TRAINING WILL BE USED IN A NUMBER OF INSTANCES TO FAMILIARIZE WORKERS WITH SPECIAL TOOLS AND PROCEDURES.	[X]	[ ]	[ ]
17.	Does design include catch pans, floor and equipment drains, and drip pans under equipment that have a significant potential for leakage?	[ ]	[ ]	[X]

		YES	NO	N/A
18.	Does design change minimize the production of solid and liquid radwaste during installation, maintenance and operation?	[ ]	[ ]	[X]

SECTION 3 - PIPING SYSTEMS AND VALVES [ ] N/A

		YES	NO	N/A
1.	Are piping run lengths minimized? PIPING THAT IS REMOVED FOR SGR WILL BE REINSTALLED IN ITS ORIGINAL LOCATION AND CONFIGURATION	[X]	[ ]	[ ]
2.	Are horizontal run lengths minimized? SEE COMMENT FOR #1	[X]	[ ]	[ ]
3.	Are drain lines sloped continuously and is backflooding prevented? SEE COMMENT FOR #1	[X]	[ ]	[ ]
4.	Is there sufficient flow velocity to prevent settling?	[ ]	[ ]	[X]
5.	Are sharp constrictions avoided? SEE COMMENT FOR #1	[X]	[ ]	[ ]
6.	Are long radius elbows used when 90° fittings are necessary? SEE COMMENT FOR #1	[X]	[ ]	[ ]
7.	Are stagnant legs avoided? SEE COMMENT FOR #1	[X]	[ ]	[ ]
8.	Are moderately radioactive pipes used to shield highly radioactive pipes in the same chase where practical?	[ ]	[ ]	[X]
9.	Are techniques available to periodically flush, hydrolase, or chemically decontaminate piping?	[ ]	[ ]	[X]

		YES	NO	N/A
10.	Is radioactive piping routed through shielding cases, behind components or structures, close to the floors or ceilings and next to walls and kept away from doors, hatches and entranceways, where practical?	[ ]	[ ]	[X]
11.	Are vent lines with horizontal offsets avoided?	[ ]	[ ]	[X]
12.	Are vent lines designed to preclude the release of radioactive liquids should tank be overfilled?	[ ]	[ ]	[X]
13.	Are field joints minimized? PREFAB WORK ON BLOWDOWN PIPING WILL ELIMINATE SOME FIELD WELDS.	[X]	[ ]	[ ]
14.	Are in-line strainers designed for optimum system capacity to prevent frequent changes?	[ ]	[ ]	[X]
15.	Are check valves included to prevent contaminated fluid backup and cross contamination?	[ ]	[ ]	[X]
16.	Has the use of live-loaded valve packings and bellow valves been considered? SMALL MANUAL VENT, DRAIN AND ROOT VALVES WILL BE REPLACED WITH LIVE-LOAD PACKING VALVES	[X]	[ ]	[ ]
17.	Are valves installed stem up? VALVES WILL BE INSTALLED STEM UP.	[X]	[ ]	[ ]
18.	Have valve designs with bonnet cavities been avoided? NEW GLOBE VALVE SELECTION CONSIDERS VALVE BONNET DESIGN	[X]	[ ]	[ ]

	YES	NO	N/A
19. Has quick access to valve packings been provided? VALVES TO BE INSTALLED IN THEIR ORIGINAL CONFIGURATION.	[X]	[ ]	[ ]
20. Has curbing around valve stations been employed to restrict spread of contaminated liquids?	[ ]	[ ]	[X]
21. Can the valve motor operator be quickly removed?	[ ]	[ ]	[X]
22. Can a solenoid or motor operator be used instead of a manually operated valve? NOT COST EFFECTIVE FOR THESE APPLICATIONS	[ ]	[X]	[ ]

SECTION 4 - TANKS, SUMPS, AND PUMPS [X] N/A

	YES	NO	N/A
1. Is the tank or sump located in a low occupancy and low traffic area (if applicable)?	[ ]	[ ]	[ ]
2. Is the tank or sump bottom sloped to the outlet?	[ ]	[ ]	[ ]
3. Have manway and ladder requirements been considered for inspections and maintenance activities?	[ ]	[ ]	[ ]
4. Has a built in tank spray or sump flush system been included for cleanout?	[ ]	[ ]	[ ]
5. Has curbing been considered to restrict the spread of tank leakage and prevent an unmonitored release?	[ ]	[ ]	[ ]

		YES	NO	N/A
6.	Does tank design include high-level alarms to preclude overflow?	[ ]	[ ]	[ ]
7.	Is the tank's or sump's interior smooth and free of pockets to facilitate decontamination?	[ ]	[ ]	[ ]
8.	Has the use of agitators or mixers to reduce or prevent settling been considered?	[ ]	[ ]	[ ]
9.	Have canned pumps or mechanical seals been considered instead of standard packing glands?	[ ]	[ ]	[ ]
10.	Have tanks and other components located in high radiation areas been designed with spare sampling connections?	[ ]	[ ]	[ ]
11.	Are pump seals easily accessible?	[ ]	[ ]	[ ]

SECTION 5 - INSTRUMENTATION [ ] N/A

		YES	NO	N/A
1.	Is instrumentation located in what will be a low background area? INSTRUMENTS ARE INSTALLED IN THEIR ORIGINAL LOCATIONS	[ ]	[X]	[ ]
2.	Have remote readouts (or CCTV monitoring) been considered?	[ ]	[ ]	[X]
3.	Does the instrument selected operate with a minimum quantity of working fluid? THERE IS NO CHANGE OF INSTRUMENT TYPE IN THIS DC.	[ ]	[ ]	[X]

		YES	NO	N/A
4.	Are transducer pressure gauges used instead of the bellows type? SEE COMMENT #3	[ ]	[ ]	[X]
5.	Is the instrument selected to allow remote reading and calibrations? SEE COMMENT #3	[ ]	[ ]	[X]
6.	Are in-line flow gauges (indicators) installed to allow easy reading from a low dose rate area?	[ ]	[ ]	[X]
7.	Are instruments grouped functionally to minimize time for surveillance and calibrations? SEE COMMENT # 1	[ ]	[ ]	[X]
8.	Are pressure relief valves and isolation valves installed so instruments can safely and easily be removed? SEE COMMENT # 1	[X]	[ ]	[ ]
9.	Are influent and effluent sample ports designed to ensure isokinetic sampling?	[ ]	[ ]	[X]

SECTION 6 - FILTERS AND DEMINERALIZERS [X] N/A

		YES	NO	N/A
1.	Have filters and other routinely serviced items been standardized, where practicable?	[ ]	[ ]	[ ]
2.	Are filter housings designed with quick release mechanism, where applicable?	[ ]	[ ]	[ ]
3.	Have remote or shielded methods for replacing hot filters been considered?	[ ]	[ ]	[ ]

		YES	NO	N/A
4.	Has the use of dual filters been considered so one filter can be allowed to decay prior to filter changeout?	[ ]	[ ]	[ ]
5.	Have single assembly filter cartridges been considered for ease of changing?	[ ]	[ ]	[ ]
6.	Are resin slurry lines shielded where practicable?	[ ]	[ ]	[ ]
7.	Does resin sluicing design minimize deposition of resin in system and provide methods to unclog lines?	[ ]	[ ]	[ ]
8.	Can containment be established to reduce the spread of contamination during filter changes?	[ ]	[ ]	[ ]
9.	Are hot filters and ion-exchangers located in low occupancy and low traffic areas?	[ ]	[ ]	[ ]

SECTION 7 - VENTILATION [ ] N/A

		YES	NO	N/A
1.	Do ventilation ducts have cleanout ports for decontamination?	[ ]	[ ]	[X]
2.	Are the number of direction changes in duct minimized to prevent contamination buildup?	[ ]	[ ]	[X]
3.	Are blower motors located in low-background areas?	[ ]	[ ]	[X]

		YES	NO	N/A
4.	Are sampling ports designed to collect isokinetic samples?	[ ]	[ ]	[X]
5.	Can filters be inspected and tested easily?	[ ]	[ ]	[X]
6.	Does the existing/proposed ventilation maintain a negative pressure, relative to adjacent clean areas?	[ ]	[ ]	[X]
7.	Have provisions been made to balance the system? DURING PERIODS WHEN THE CONTAINMENT EQUIP. HATCH IS OPEN, THE CONTAINMENT VENT SYS WILL BE INTENTIONALLY UNBALANCED TO MAINTAIN A NEG PRESS. IN CONTAINMENT.	[ ]	[X]	[ ]
8.	Are dampers and fittings installed in ventilation ducts to permit use of "elephant trucks" without imbalancing the entire system? A TEMP "T" + DAMPER TO USE AN "ELEPHANT TRUNK" IN THE UPPER CONTAINMENT AREA WILL BE USED.	[X]	[ ]	[ ]
9.	Are filters for low or non-contaminated ventilation systems located in low background areas?	[ ]	[ ]	[X]
10.	Does design provide for remote removal and replacement of hot filters where practicable?	[ ]	[ ]	[X]
11.	Are ventilation exhaust ducts located near the floor and away from entrances?	[ ]	[ ]	[X]
12.	Are filters for highly contaminated ventilation systems located in shielded housings and designed for easy removal?	[ ]	[ ]	[X]

		YES	NO	N/A
13.	Have sampling stations for primary coolant been provided with auxiliary ventilation?	[ ]	[ ]	[X]
14.	Does sample line design and materials utilize minimize plate-out of radioactive particulates?	[ ]	[ ]	[X]

SECTION 8 - MATERIALS OF CONSTRUCTION

		YES	NO	N/A
1.	Have low-cobalt and low-nickel materials been used where practical on the primary system to reduce activated corrosion products? THE STAINLESS STEEL MATERIAL BEING USED ON THE PRIMARY SYSTEM HAS A LOW-COBALT AND LOW-NICKEL CONTENT.	[X]	[ ]	[ ]
2.	Are the materials compatible with primary system chemistry? REPLACEMENT MATERIALS ARE IDENTICAL TO EXISTING MATERIALS EXCEPT FOR CHROME-MOLY PIPING IN BLOWDOWN AND FW SYSTEMS AND ALLOY 690 TT SG TUBES	[X]	[ ]	[ ]
3.	Is the use of stellite material minimized? SEE COMMENTS FOR #2	[X]	[ ]	[ ]
4.	Have radiation-damage resistant materials been used in high radiation areas? SEE COMMENTS FOR #2	[X]	[ ]	[ ]
5.	Are rough surface finishes avoided? SEE COMMENTS FOR #2	[X]	[ ]	[ ]
6.	Are new valves, pipes, and components chemically preconditioned or has passivation been considered prior to use, to minimize corrosion?	[ ]	[ ]	[X]
7.	Has chemical treatment of the system to minimize corrosion been considered?	[ ]	[ ]	[X]

		YES	NO	N/A
8.	Is stainless steel used as practical to reduce corrosion and optimize decontamination?	[ ]	[ ]	[X]
9.	Are crevices, holes, notches, and recesses avoided? REPLACEMENT SG DESIGN ENHANCEMENTS REDUCE CORROSION POTENTIAL.	[X]	[ ]	[ ]
10.	Are sockethead cap screws and knurled finishes avoided? SEE COMMENTS FOR #2	[X]	[ ]	[ ]
11.	Are protective coatings selected for the longest possible life expectancy?	[ ]	[ ]	[X]
12.	Are all materials selected for life expectancy of the plant? SEE COMMENTS FOR #2	[X]	[ ]	[ ]
13.	Are concrete surfaces sealed, smooth and painted for easy decontamination?	[ ]	[ ]	[X]

Prepared by: MS Barth Date: 6/22/92

Organization: BECHTEL

Reviewed by: [signature] Date: 6/24/92
ALARA Coordinator

ENGINEERING CHANGE REQUEST AND MARK-UPS FOR SYSTEM DESIGN BASIS DOCUMENT CHANGES

VIRGINIA POWER

Engineering Change Request

POW 14

Instructions:
Complete items 1 through 11 and forward to the coordinator for the document affected by the Engineering Change Request (ECR). Refer to Nuclear Design Control Manual (NDCM) Procedure 3.4.

1) ECR No. 2081

2) Station Name And Unit: North Anna/Unit 1

3) Page 1 Of 1

4) Subject/Title: Design Basis Documents (DBDs) for Recirculation Spray, Safety Injection, Quench Spray and Service Water Require Revision.

5) Key Words

6) Document Type: Design Basis Document

7) Document Title And Number: Recirculation Spray, SDBD-NAPS-RS, Safety Injection-SDBD-NAPS-SI, Quench Spray SDBD-NAPS-QS, Service Water SDBD-NAPS-SW

8) Change Requested

The existing reflective insulation on the North Anna Unit 1 steam generators and associated piping will be replaced with fiberglass - blanket type insulation when the steam generators are replaced. In addition, the containment analysis has been redone due to the steam generator replacements using the 1979 Westinghouse mass and energy release model.

The Design Basis Documents (DBDs) for Recirculation Spray, Safety Injection, Quench Spray, and Service Water need to be revised to reflect the results of the steam generator insulation debris analysis and containment analysis.

9) Originator (Signature)	Date	10) Reviewed/Approved (Signature)		Date	11) Extension
Michael K Calling /SWEC	7-9-92	RAB for N Hanley /SWEC		7/9/92	
12) Assigned By (Signature)	Date	13) Assigned To	14) Department	15) Assigned Date	16) Due Date

17) Resolution

18) Resolution By (Signature)	Date	19) Reviewed By (Signature)	Date	20) Approved By (Signature)	Date

21) Distribution	22) Received By Records Management (Signature)	Date

Form No. 726249(Dec 90)

NAPS SW

3. The capability to prevent or mitigate the consequences of accidents with potential offsite exposure approaching 10CFR100 levels.

A more detailed discussion of the classification of safety-related functions is presented in Virginia Power Standard No. STD-GN-0003, "Instructions for Identification and Classification of Safety-Related Structures, Systems, and Components."

The following functions of the SW System are safety related.

1. The SW System shall provide cooling water so that the heat removed from safety-related equipment during normal and accident conditions can be transferred to the ultimate heat sink. [25.1.7]

   a. The SW System shall provide an emergency source of cooling water to transfer heat to the ultimate heat sink whenever the Recirculation Spray (RS) System is required to remove heat and depressurize the ~~containment.~~ [25.4.4] [25.4.5] [25.4.38] [25.8.18] [25.4.46]

   b. The SW System shall provide a source of cooling water to transfer heat from the Component Cooling Water (CC) System heat exchangers to the ultimate heat sink during normal and accident conditions. [25.8.18] [24.2.1]

   c. The SW System shall provide, during all modes of operation, a source of cooling water to transfer heat from the Control Room air conditioning condensers to the ultimate heat sink as required by the Control Room Air Conditioning and Pressure (HC) System. [25.8.18] [24.2.2]

   d. The SW System shall provide during all modes of operation, a source of cooling water to transfer heat from the charging pump seal coolers, gear reducers, and lube oil coolers to the ultimate heat sink as required by the Chemical and Volume Control (CH) System. [25.8.18] [24.2.4]

   e. The SW System shall provide a backup source of cooling water to the CC System to transfer heat

2.2 PERFORMANCE CRITERIA

This section identifies the performance criteria for each of the corresponding functions identified in Section 2.1. The performance criteria establish the objective measure of functionality and provide quantification of acceptable performance for satisfying the system functional requirements which are identified. Performance criteria are expressed in the most fundamental manner to allow the designer/engineer flexibility in establishing design features to satisfy the criteria. Generally, the fundamental performance criteria can be satisfied by many designs. The primary objective in establishing performance criteria is to define the fundamental information needed to design the system that is independent of component selection or system configuration.

The performance criteria identified in this chapter are presented "qualitatively." The performance requirements are stated in "quantitative" terms in Chapter 6, "Key System Parameters."

The performance criteria for the SW System are identified in the following sections:

2.2.1 Safety-Related Performance Criteria

This subsection identifies the safety-related performance criteria for the SW System. Performance criteria related to the same function are grouped together. The following are safety-related performance criteria:

1. In support of the heat removal function identified in Section 2.1.1, the following performance criteria shall be met:

   a. The SW System shall deliver, assuming a single failure, the minimum flow to the RS heat exchangers, over a range of water temperatures required to satisfy the heat-removal rate established in the plant safety analysis and the requirem~~ents of the RS~~ System. [25.4.4] [25.4.5] [25.4.38] [25.4.46]

   b. The SW System shall deliver the minimum flow, over a range of water temperatures required, to the CC heat exchangers, as defined by the CC System. [25.12.15]

   c. The SW System shall deliver the minimum flow of maximum temperature water required to the Control Room air conditioning condensers as defined by the HC System. [25.12.22]

5.1.14	Recirculation Spray (RS)	The SW System shall supply cooling for the RS heat exchangers.	[25.4.4] [25.4.5] [25.4.38] ~~[25.8.18]~~ [25.4.46]
5.1.15	Service (SA)	The SW System shall supply cooling for the SA compressor water jackets and after coolers.	[25.8.20]
5.1.16	Steam Generator Blowdown (BD)	The SW System shall supply backup cooling for the BD System containment penetration coolers.	[25.8.19]

5.2 SUPPORTING SYSTEMS

This section identifies the systems that provide support services to the SW System. The interfacing systems must provide support services to the SW System to function as designed. The requirements and/or limitations of the supporting systems are defined, as well as the name of the supporting system and its system code, and the source of the requirement and/or limitation.

The systems identified below provide supporting services to the SW System.

	System	Requirement/Limitation	Source
5.2.1	Auxiliary Building Ventilation (HA)	The HA System shall provide filtered air for ventilation and heating for the SW System components located in the Auxiliary Building.	[25.8.8] [25.8.9] [25.8.62] [25.8.65]
5.2.2	Circulating Water (CW)	The CW System shall provide screened lake water to the ASW pumps, and the CW screenwash pumps shall provide makeup water to the SW reservoir. The CW System also shall provide the discharge path to Lake Anna when the lake is being used as the heat sink.	[25.8.18]
5.2.3	Containment Air Cooling (HR)	The HR System shall provide cooling for the containment where SW System components are located.	[25.8.4] [25.8.5] [25.8.65]

Table 6.1-1. Key System Parameters, Safety Related Functions, SW System

(A) Item No.	(B) System Parameter	(C) Function	(D) Operating Condition	(E) Design Requirement	(F) Reference	(G) Remarks
6	Temperature (max)	SW supply	Condition IV (LOCA one unit, Loss of power two units).	110°F [25.4.40]	Table 11.2-1	
7	Temperature (max)	SW supply to RS heat exchangers	Condition IV (LOCA one unit)	97°F ~~[25.4.5]~~ [25.4.46]	Section 2.2 Table 11.1-5	
8	Level (min)	SW reservoir level	Condition I-IV	EL. 313 ft [25.4.8]	Section 11.2	
9	Level (min)	North Anna Lake Level	Condition I-IV	EL. 244 ft [25.12.4] [25.12.5]	Section 11.2	

DC 90-13-1, Appendix 4-9, Page 6

The design requirement for a Condition IV event is that Condition IV faults shall not cause a release of radioactive material that results in an undue risk to public health and safety exceeding the guidelines of 10CFR100 and a single Condition IV fault shall not cause a consequential loss of required functions of systems needed to cope with the fault including those of the Reactor Coolant System and the reactor containment system.

During the following Condition IV events, which do not result in an energy release to the containment, the affected unit will shut down:

1. Steam generator tube rupture
2. Single reactor coolant pump locked rotor
3. Fuel-handling accident outside containment

The SW System operation is as described in Section 3.3.1.3 for shutdown.

During a fuel handling accident inside containment, a Condition IV event, there are no special requirements for the SW System, and it operates as described in Section 3.3.1., item 2 for refueling.

During the following Condition IV events involving an energy release within the containment, the SW System supplies cooling water to the equipment described in Section 3.3.4:

1. Major Reactor Coolant System pipe rupture (LOCA)

2. Major secondary system pipe rupture (i.e., a main steam line break (MSLB) or a feedwater line break (FWLB))

3. Rupture of a control rod drive mechanism housing (REA) [25.4.46]

The accident design basis for the flow requirements of the SW System is the simultaneous occurrence of a LOCA [25.4.38] or MSLB [25.4.4] [25.4.5] and the loss of offsite power (LOOP) for both units. The most limiting single failure is also assumed concurrent with the postulated event.

The minimum flow requirement for the SW System during the first half hour following a LOCA or MSLB and a loss of offsite power is 25,816 gpm. The flow requirement during the second half hour (t=30 minutes to t=1 hr) following the event is 27,016 gpm. See Table 6.1-4. [24.6.1]

A service water flow of 18,000 gpm is required for four RS heat exchangers (4,500 gpm each) during the first hour following the event. Given the single failure of one engineered safety features (ESF) train, the operator may first choose to identify the inoperative RS heat exchangers and secure the service water flow to the same. The minimum operator action time is assumed to be 30 minutes. This action will reduce the required service water flow to the accident unit by 9,000 gpm.

After the first hour following the event, 9,000 gpm is required for one train of two RS heat exchangers (4,500 gpm each). The SW System flow between 1 and 24 hours following the event is 18,016 gpm with one CC heat exchanger used on the nonaccident unit. See Table 6.1-4. [24.6.2]

The minimum specified service water flow per RS heat exchanger is 4,500 gpm. This flow and rate is used as an input to the containment safety analyses. [25.4.4] [25.4.5] [25.4.38] [25.4.46]

It should be noted that the FWLB and REA are not analyzed for containment temperature and pressure since their consequences are enveloped by the LOCA and MSLB. [24.11.1]

Plant safety analyses are presented in detail in the PDBD and consist of the following calculations: [24.11.2]

1. Calculation No. 14938.37-US(B)-259-0, "LOCA Analysis for Revised Technical Specifications on Containment Air Partial Pressure," SWEC, May 28, 1987. [25.4.5] [25.4.46]

2. Calculation No. 11715-ES-150-2, "LOCTIC Input, Containment Integrity," SWEC, July 31, 1987. [25.4.38]

3. Calculation No. 14938.37-US(B)-260-0, "Main Steam Line Break Analysis," SWEC, May 28, 1987. [25.4.4]

11.2 SUPPLEMENTARY ANALYSES

This section identifies and discusses any supplementary safety analysis performed for the SW System in addition to those analyses required by Section 11.1. Each analysis is described briefly below and is summarized in Table 11.2-1.

11.2.1 SW Reservoir Performance Analysis

A Service Water Reservoir Performance Analysis (ME-062) [25.4.40] was performed to evaluate performance during normal and design basis conditions to ensure reservoir temperatures will be maintained below TS limits during normal conditions, and evaluate system capability during DBA conditions. [24.11.3] [24.11.4]

Table 11.1-5. Plant Safety Analysis Parameters, SW System, Condition IV Events - Limiting Faults, Event: LOCA/MSLB

	Parameter		Reason
1.	The minimum acceptable service water flow rate of each of two RS heat exch[illegible]er shall be 4,500 gpm. [25.4.4] [25.4.5] [25.4.38] [25.4.46]	Why	The containment safety analysis is performed using the minimum acceptable service water flow. Minimum service water flow is conservative for the containment safety analysis.
		Origin	The service water flow rate is documented in the containment safety analyses.
		Impact	Changes in service water flow rate will impact the containment safety analysis, specifically, containment depressurization time and subatmospheric peak pressure.

Table 11.1-5. Plant Safety Analysis Parameters, SW System, Condition IV Events - Limiting Faults, Event: LOCA/MSLB

	Parameter		*Reason*
2.	The service water temperature shall be limited to the range of 35-97°F. All service water temperatures reflect a 2°F margin for pond temperature rise in the first 2 hours. After 2 hours the temperature may rise higher than the 2°F increase, but does not impact the containment safety analysis. ~~[25.4.4] [25.4.5]~~ [25.4.38] [25.4.46]	Why	Service water provides the main path for heat removal from containment during a postulated accident. As part of the containment safety analysis, a special analysis referred to as "Tech-Spec Analysis," which calculates the allowable containment air partial pressure with a given service water temperature, is performed. This is an integral part of the operational technical specifications. [25.12.4] [25.12.5]
		Origin	The origin of the service water temperature requirements is the containment LOCA analyses ~~[25.4.6]~~ [25.4.46]
		Impact	Changes in service water temperature without the proper adjustment in containment air partial pressure could impact the peak calculated containment pressure, depressurization time and ability to maintain a subatmospheric pressure.

1. Calculation No. 14938.37-US(B)-259-0, "LOCA Analysis for Revised Technical Specifications on Containment Air Partial Pressure," SWEC, May 28, 1987. [25.4.5]

2. Calculation No. 11715-ES-150-2, "LOCTIC Input, Containment Integrity," SWEC, July 31, 1987. [25.4.38]

3. Calculation No. 14938.37-US(B)-260-0, "Main Steam Line Break Analysis," SWEC, May 28, 1987. [25.4.4]

4. Calculation No 02072.2010-US(B)-274-0, "Containment LOCA Analysis with New Steam Generators" (SWEC), June 17, 1992. [25.4.46]

16.1 ESTIMATE POSSIBILITY OF SW RESERVOIR FREEZING

This calculation was used in the development of the SW System DBD as a means of defining the SW flow to the individual SW components.

Calculation Number, Revision, and Date - 12050-12.5.33.31, Rev. 0, May 25, 1978 [25.4.11]

Calculation Title - "Estimate Possibility of SW Reservoir Freezing"

Purpose - To determine the extent of the worst case freezing possibility of the SW reservoir since some heat is being added to the system by Unit 1. This determination will help to recommend some type of freeze protection, if needed.

Assumptions - It was assumed that the coldest 30-day period on record for Richmond would be the worst case basis for SW reservoir freezing. The SW pump can be throttled to reduce the height of the spray to conserve heat.

Inputs - The following sources were used as inputs to the calculation:

1. Component Heat Loads and Flows from Nuclear Groups.
2. SWEC Calculation 11715-12.5.3.3.14, July 7, 1976.
3. SWEC Calculation 11715-12.5.3.3.16, July 21, 1976.
4. SWEC Calculation 11715-12.5.3.3.19, August 5, 1976.
5. SWEC Calculation 11715-12.5.3.3.45, April 14, 1978.

Methodology - Expected heat loads on the SW System due to normal operation of Unit 1 were obtained. The flows necessary for each component were then obtained. Computer program EN-127, "Ultimate Heat Sink - Spray Cooling Pond," V00, L00, and various assumptions listed in body of calculation were then used to determine low temperatures in the SW reservoir.

Results and Conclusions - The calculation as performed determined that freezing conditions could occur in the SW reservoir dependent on the heat loads and percentage of spray nozzles being used.

Table 19.2-1. Licensing Basis Surveillance Requirements

2. Design Basis Surveillance Requirement - Service Water Leakage Into RS Heat Exchangers

a. System or Component - RS Heat Exchangers

b. Source - [25.5.5] [25.5.10]

c. Surveillance Required - The SW piping and RS coolers shall be maintained dry.

Reason or Basis - To prevent fouling of the RS cooler tubes and ensure that the RS cooler heat transfer coefficients are in accordance with the assumptions of the safety analyses. [25.4.4] [25.4.5] [25.4.38] [25.4.46]

d. Surveillance Parameter - The RS coolers shall be inspected to determine if service water is leaking into the RS coolers.

Reason or Basis - Refer to item c above.

e. Surveillance Frequency - At least once per week

Reason or Basis - [25.5.5]

Reason or Basis - RS cooler tube fouling has to be prevented regardless of plant operational mode since the tubes have to be chemically cleaned to remove fouling when it occurs.

g. Acceptance Criteria - Verification of the lack of moisture collection when the vents/drains associated with the RS coolers are opened.

Reason or Basis - Refer to item d above.

25.4.38 11715-ES-150-2, Rev. 2, "LOCTIC Input, Containment Integrity," SWEC, July 31, 1987.

25.4.39 ME-0049, Rev. 0, "Service Water Chemical Addition Pumps," Virginia Power (no date available).

25.4.40 ME-062, Rev. 0, "Reservoir Performance Analysis," with Addendum 1-3, Virginia Power, May 1, 1987.

25.4.41 ME-0226, Rev. 0, "SW Flow Increase Due to Simulated Opening of Bypass and Spray Array MOVs," Virginia Power (no date available).

25.4.42 CE-0067, Rev. 0, "Overpressurization of SW Piping," Virginia Power (no date available).

25.4.43 14938.09-5-1, Rev. 3, "Service Water Tie-In Vault," SWEC, May 15, 1987.

25.4.44 CE 199, Rev. 1, "Valve House Structure," Virginia Power, 1985.

25.4.45 CE 272, Rev. 0, "Spray Array Support Structure," Virginia Power, 1985.

25.4.46 02072.2010-US(B)-274, Rev 0, "Containment LOCA Analysis with new Steam Generators", SWEC, June 17, 1992

25.5 LICENSING CORRESPONDING REFERENCES

25.5.1 Technical Report No. PE-0013, "Instrumentation for Light Water Cooled Nuclear Power Plants to Assess Plant Environs Condition During and Following an Accident," December 19, 1989.

25.5.2 Facility Operating License No. NPF-4, North Anna Power Station Unit No. 1, November 26, 1977, as Amended, Section 2.D.

25.5.3 Facility Operating License No. NPF-7 North Anna Power Station Unit No. 2 August 21, 1980, as Amended, Section 2.C.

25.5.4 NUREG-0053, Section 9.2.1, Safety Evaluation Report, June 4, 1976, including Supplement 10, April 10, 1980.

25.5.5 Virginia Power Letter to NRC Serial No. N-89-023, September 8, 1989 (I.D. No. 8909180335-3).

25.5.6 North Anna Power Station Unit 1 TS 3.7.1.2 and 4.7.1.2.

25.5.7 North Anna Power Station Unit 1 TS 3.7.1.3 and 4.7.1.3.

25.5.8 Virginia Power Letter Serial No. 252, "IE Bulletin 81-03," May 21, 1981.

NAPS SI

2.3.11 Hydraulic Requirements

1. The SI system piping, fittings, valves, sizing, and layout shall address system fluid flow requirements (based on the fluid pressure, temperature, and system fluid transients), and fluid velocity requirements (to limit pressure drops and erosion). [25.11.1]

2. To ensure proper operation of the system, the NPSH required by the system pumps shall be less than the NPSH available to the pumps at the maximum required flow. During runout conditions, the NPSH available shall be sufficient to prevent damage to the pump. [25.11.1]

2.3.12 Chemistry and Sampling Requirements

The SI System fluid chemistry shall be maintained within the limits assumed in the safety analyses. [25.4.15] [25.4.16] [25.4.17] [25.4.18] [25.4.19] [25.4.11] [25.4.21] [25.4.20] [25.4.30] [25.4.49]

2.3.13 Electrical Requirements

1. Power Source and Voltage

   a. Safety-related electrical equipment associated with the SI System shall be supplied from both an onsite an offsite source through a highly reliable distribution system at the required voltage. [25.1.34] [25.10.1]

   b. Non-safety-related electrical equipment associated with the SI System shall be supplied from a non-Class 1E power supply. [25.11.1]

2. Electrical Insulation

   SI System electrical equipment voltage insulation shall be adequate for the normal system operating voltages. [25.11.1]

3. Cable and Raceway

   The cable and raceway supplying SI System electrical equipment shall be designed to meet cable and raceway fill, arrangement, and ampacity requirements. [25.11.1]

4. Motors and Loads

DC 90-13-1, Appendix 4-9, Page 16

Table 6.1-1. Key System Parameters, Safety-Related Functions, SI System

(A) Item No.	(B) System Parameter	(C) Function	(D) Operating Condition	(E) Design Requirement	(F) Reference	(G) Remarks
1	Flow, minimum (gpm)	LBLOCA, Injection Phase	Condition IV RC System Pressure (psia)	 Flow [25.12.4] [25.4.11]	Table 11.1-5	Flow rate is a function of RC System pressure. This is total required flow. Note the total flow is the LHSI flow plus the HHSI flow.
			14.7	2558.6		
			24.7	2412.8		
			34.7	2254.5		
			44.7	2080.4		
			54.7	1898.3		
			64.7	1695.2		
			74.7	1481.1		
			84.7	1263.0		
			94.7	1001.9		
			114.7	370.8		
2	Flow, maximum HHSI pumps (gpm)	LOCA, Injection Phase	Condition III, IV	565 [25.4.21] [25.4.22] [25.4.49]	Table 11.1-5 Table 11.1-4	(1) Assumes minimum safeguards. (2) Flow rates are inputs to the containment analysis.
				741 [25.4.21] [25.4.22] [25.4.49]		(1) Assumes normal safeguards. (2) Flow rates are inputs to the containment analysis.
3	Flow, maximum LHSI pumps (gpm)	LOCA, Injection Phase	Condition III, IV Differential Pressure (psi)	 Flow [25.4.21] [25.4.22] [25.4.49]	Table 11.1-5 Table 11.1-4	(1) Differential pressure = Reactor Vessel Pressure (RVP)-RWST Static Head (HRWST). (2) Assumes minimum safeguards. (3) Flow rates are inputs to the containment analysis.
			-100	4200		
			-5	4180		
			0	4160		
			10	4005		
			100	4000		

Table 8.1.1. Key System Parameters, Safety Related Functions, SI System

(A) Item No.	(B) System Parameter	(C) Function	(D) Operating Condition	(E) Design Requirement	(F) Reference	(G) Remarks
4	Flow, maximum LHSI pumps (gpm)	LBLOCA, Injection Phase	Condition IV Differential Pressure (psi)	Flow [25.4.21] [25.4.22] [25.4.49]	Table 11.1-5	
			-20			
			-5	5463 6000		(1) Assumes normal safeguards.
			0	5378		(2) Flow rates are inputs to the containment analysis.
			10	5278		
			80	4800		
5	Flow, minimum (gpm)	LOCA, Recirculation Phase	Condition IV Time after Reactor trip (min)	Flow [25.9.1] [24.6.1]	Table 11.1-5 Table 11.1-4	Minimum required flow varies with time after shutdown.
			14	350		
			30	275		
			60	220		
			120	180		
			300	140		
			800	110		
			1400	95		
			10,000	47		
6	Flow, maximum LHSI pumps (gpm)	LOCA, Recirculation Phase	Condition IV	4030	Table 11.1-5	(1) Assumes minimum safeguards.
				5750	Table 11.1-4	(2) Assumes normal safeguards.
				[25.4.21] [25.4.22] [25.4.49]		(3) Both flow rates are inputs to the containment analysis.

DC 90-13-1, Appendix 4-9, Page 17

Table 6.1-1. Key System Parameters, Safety Related Functions, SI System

(A) Item No.	(B) System Parameter	(C) Function	(D) Operating Condition	(E) Design Requirement	(F) Reference	(G) Remarks
7	Flow, minimum (gpm)	SBLOCA, Injection Phase	Condition III RC System Pressure (psia)	Flow [25.4.10] [24.6.2]	Table 11.1-4	Break size < 5.187 inches.
			214	365.6		
			414	347.5		
			614	329.4		
			814	311.3		
			1014	285.6		
			1214	271.5		
			1414	246.1		
			1614	220.7		
			1814	195.5		
			2014	162.9		
			2214	123.1		
			2414	52.9		
			2454	0		
8	Flow, minimum (gpm)	SBLOCA, Injection Phase	Condition III RC System Pressure (psia)	Flow [25.4.10] [24.6.2]	Table 11.1-4	(1) Break size > 5.187 inches. (2) Two values are listed at 1014 psia. The higher value assumes that the HHSI/Charging pump recirculation line is isolated. The second assumes it is not.
			214	335.2		
			414	311.5		
			614	286.8		
			814	260.8		
			1014	233.1		
			1014	223.6		
			1214	193.1		
			1614	122.4		
			1814	78.7		
			2014	22.2		

DC 90-13-1, Appendix 4-9, Page 18

Table 6.1.1. Key System Parameters, Safety-Related Functions, SI System

(A) Item No.	(B) System Parameter	(C) Function	(D) Operating Condition	(E) Design Requirement	(F) Reference	(G) Remarks
9	Flow, minimum (gpm)	MSLB	Condition IV Pressure (psia) 214 414 614 814 1014 1015 1214 1414 1614 1814 2014 2214 2414 2454	Flow [25.12.4] 553.3 530.3 506.3 481.0 454.0 441.1 411.8 380.0 345.1 305.8 260.0 202.3 110.8 74.4	Table 11.1-5	Flow rate is a function of RC System pressure
10	Temperature, minimum; Accumulator, (F)	LBLOCA	Condition IV	86 [25.4.11]	Table 11.1-5	(1) This temperature is an input assumption to the LBLOCA analysis. (2) The containment analysis assumes a minimum accumulator temperature of 80°F which is conservative. See Table 11.1-5.[25.4.12]
11	Temperature, minimum; BIT (F)	MSLB	Condition IV	111 [25.4.13] [24.6.3]	Table 11.1-5	The reference listed refers to WCAP-1570 as its source for solubility vs. temperature data.
12	Temperature, minimum; RWST (F)	LBLOCA, Injection Phase	Condition IV	40 [25.4.11]	Table 11.1-5	This temperature is an input assumption to the LBLOCA analysis

DC 90-13-1, Appendix 4-9, Page 19

Table 6.1-1. Key System Parameters, Safety Related Functions, SI System

(A) Item No.	(B) System Parameter	(C) Function	(D) Operating Condition	(E) Design Requirement	(F) Reference	(G) Remarks
13	Temperature, maximum, RWST (F)	SBLOCA, Injection Phase	Condition III	45 [25.4.14]	Table 11.1-5	Temperature is assumed in determining NPSH A for the HHSI/Charging pumps during the injection phase.
14	Temperature, RWST (F)	MSLB	Condition IV	[24.6.4]		
15	Water volume, minimum; accumulator (ft^3 or cu ft)	LBLOCA Reflood	Condition IV	1025 [25.4.11]	Table 11.1-5	Water volume required to be maintained in each accumulator by the LBLOCA analysis. Minimum volume of 997.3 ft^3 is assumed in each accumulator by the containment analysis which is conservative. See Table 11.1-5 item 9. [25.4.12]
16	Volume, total; accumulator (ft^3)	LBLOCA	Condition IV	1450 [25.4.11] [25.4.21] [25.4.44]	Table 11.1-5	Total volume to the accumulator (water and nitrogen). Input to both the LBLOCA and Containment analyses.
17	Volume, minimum; BIT (gallons)	LBLOCA	Condition IV	900 [25.4.15] [25.4.16] [25.4.17] [25.4.18] [25.4.19]	Table 11.1-5	Volume is an input into the MSLB analysis.
18	Pressure, minimum; accumulator (psia)	LBLOCA	Condition IV	594 [25.4.11]	[illegible]1-5	Pressure is an input to the LBLOCA analysis.

DC 90-13-1, Appendix 4-9, Page 20

Table 6.1-1. Key System Parameters, Safety Related Functions, SI System

(A) Item No.	(B) System Parameter	(C) Function	(D) Operating Condition	(E) Design Requirement	(F) Reference	(G) Remarks
32	Time, maximum, recirculation path alternation interval (hours)	LOCA	Condition III, IV	10 [24.6.18] [25.6.15]	Table 11.1-5 Table 11.1-4	
33	NPSH-R, LHSI pumps (ft)	LOCA	Condition III, IV	UNIT 2: 13.4 [25.5.6] [25.4.21] UNIT 1: 13.1 [25.4.49] [25.4.49]	Table 11.1-5 Table 11.1-4	Assumes a flow rate of 4130 gpm with minimum safeguards.
34	NPSH-R, HHSI/ charging pumps (ft)	LOCA	Condition III, IV	23.7 [25.3.16]	Table 11.1-5	Assumes a flow rate of 610 gpm with minimum safeguards.

DC 90-13-1, Appendix 4-9, Page 21

In 1978, NAPS Unit 1 introduced a cross-connection between the RS and SI Systems, specifically between the Outside Recirculation Spray (ORS) pump discharge and the LHSI pump discharge, to provide a long-term backup for the LHSI pump. [25.12.12] This feature was not intended to be part of the design bases and was to be allowed only if reactor coolant temperature was less than 200°F. This design change (DCP-78-01) was made only to NAPS Unit 1 and was provided as additional assurance of core cooling (in the event of an accident) prior to completion of the long term pump testing required to demonstrate LHSI and Recirculation Spray (RS) pumps. [25.5.17] [25.10.30] LHSI pump reliability was subsequently demonstrated, thus eliminating the need for the cross-connect. [25.5.18]

Both the HHSI and LHSI pumps rely on the RWST for the initial injection mode. This tank is described in detail in SDBD-NAPS-QS. [24.7.6]

Three accumulators were provided as passive engineered safeguards. No external source of power or signal transmission was needed to obtain fast-acting, high-flow injection of boric acid solution should the need arise. One accumulator was connected to each of the RC System cold legs. [24.2.2] [25.3.5]

The accumulators were pressure vessels filled with boric acid solution and pressurized with nitrogen gas. During normal operation each accumulator was isolated from the RC System by two check valves in series. If the RC System pressure fell below the accumulator pressure, the check valves would open and boric acid solution would be forced into the RC System. Mechanical operation of the swing-disc check valves would be the only action required to open the injection path from the accumulators to the core via the cold leg. [24.2.2] [25.3.5]

The design capacity of the accumulators was based on the assumption that flow from one of the three accumulators spills onto the containment floor through the ruptured loop. The two remaining accumulators provided sufficient water to fill the volume outside of the core barrel below the nozzles, the bottom plenum, and one-half the core. [24.2.2] [25.3.5]

Insert 7.1

There are no significant changes to the original concept of the SI System as submitted in both the PSAR and the FSAR.

7.2 PROGRAMMATIC AND REGULATORY ISSUE HISTORY

This section identifies and discusses programmatic topics or regulatory issues that have influenced the original design basis. This discussion is presented to summarize the overall effect of the programmatic and regulatory issues on the design philosophy and design basis, and not to discuss specific requirements or changes attributable to the issues. A more complete discussion of programmatic issues is presented in Chapter 15.

Insert 7.1

In the late 1980's, it was noted that the SGs at NAPS-1 were experiencing corrosion-related degradation resulting in the requirement for frequent inspection, and plugging of a significant number of SG tubes. Despite improvements in secondary water chemistry, tube degradation continued to occur. In 1990, engineering efforts were initiated to replace the SG lower tube bundle assemblies at NAPS-1. The replacement SGs were to be fabricated and analyzed to standards which were at a minimum, equivalent to the existing equipment. [25.12(11)]. As noted in Design Change Package DC 90-13-1 [25.12(11)], the new Model 51F SG was an improvement over the old Model 51 SG and from a safety analysis perspective, could be considered a "replacement" for the Model 51.

15 15

One of the differences between the old and the replacement SGs was the thermal insulation. The new insulation was a fiberglass blanket type which exceeded the design requirements of the existing insulation. However, due to the direction received in GL 85-22 [25.1.66], analytical models (in accordance with RG 1.82, Rev. 1 [25.1.63]) had to be utilized to estimate post LOCA head loss across the containment sump screens due to insulation debris. This analysis, [25.4(138)], resulted in the development of new values for the NPSH available at the ORS and IRS pump suction and a new containment LOCA analyses for NAPS-Unit 1 [25.4(139)]

LHSI 49 48

01287.B07

Table 11.1-4. Plant Safety Analysis Parameters, SI System, Condition III Events - Infrequent Faults, Event: SBLOCA

	Parameter		Reason
2.	During the injection phase following a LOCA, the maximum HHSI flow shall be limited to 565 gpm with minimum safeguards and 741 gpm with normal safeguards.	Why	These maximum flow rates are used in analysis of mass and energy release rates. Maximum rates are assumed to obtain conservative results.
		Origin	Flow rates are calculated in [25.4.22] and used in containment analysis of [25.4.21] & [25.4.49]
		Impact	Flow rates higher than those presented may invalidate the referenced analyses.

Table 11.1-4. Plant Safety Analysis Parameters, SI System, Condition III Events - Infrequent Faults, Event: SBLOCA

Parameter		Reason
3. During the Injection Phase following a LOCA With a minimum safeguards, the SI System shall deliver a maximum of: Flow (gpm) at DP (psi) 4200 -100 4180 -5 4160 0 4005 10 4000 100 With normal safeguards the SI shall deliver a maximum of: Flow (gpm) at DP (psi) 6000 -20 5463 -5 5378 0 5278 10 4800 80	Why	The maximum flow rates are used in analysis of mass and energy release rates. Maximum rates are assumed to obtain conservative results.
	Origin	Flow rates are calculated in [25.4.22] and used in the containment analysis. [25.4.21] [25.4.49]
	Impact	Flow rates greater than those calculated may invalidate the referenced analyses.

Table 11.1-4. Plant Safety Analysis Parameters, SI System, Condition III Events - Infrequent Faults, Event: SBLOCA

	Parameter	*Reason*	
5.	During the recirculation phase following a LOCA, the maximum SI flow shall be limited to 4030 gpm for minimum safeguards and 5750 gpm for normal safeguards.	Why	These flow rates are used in analysis of mass and energy release rates. Maximum rates are assumed to obtain conservative results.
		Origin	The flow rates stated are calculated in [25.4.22] and used in the containment safety analysis. [25.4.21] [25.4.49]
		Impact	Flow rates greater than those calculated maximums may invalidate the containment safety analysis.

Table 11.1-4. Plant Safety Analysis Parameters, SI System, Condition III Events - Infrequent Faults, Event: SBLOCA

	Parameter		Reason
11.	The net positive suction head available (NPSHA) to the UNIT 2 LHSI pumps shall be a minimum of 13.4 ft.	Why	Sufficient NPSH is required to ensure minimum flow requirements are met.
	The net positive suction head available to the unit 1 LHSI pumps shall be a minimum of 13.1 FT.	Origin	The NPSHR assumes minimum safeguards, recirculation phase, and a flow rate of 4030 gpm. [25.5.6] [25.4.21] [25.4.48] [25.4.49]
		Impact	If NPSH is insufficient, the LHSI pump may not provide adequate flow. Either consequence would invalidate the LBLOCA analyses.

Table 11.1-5. Plant Safety Analysis Parameters, SI System, Condition IV Events - Limiting Faults, Event: LOCA, MSLB

Parameter		Reason
2. During the injection phase following a LOCA, the maximum HHSI flow shall be limited to 565 gpm with minimum safeguards and 741 gpm with normal safeguards.	Why	These flow rates are used in analysis of mass and energy release rates. The maximum rates are assumed to obtain conservative results.
	Origin	Flow rates are calculated in [25.4.22] and used in containment analysis of [25.4.21] [25.4.49]
	Impact	Flow rates higher than those presented may invalidate the referenced analyses.

Table 11.1-5. Plant Safety Analysis Parameters, SI System, Condition IV Events - Limiting Faults, Event: LOCA, MSLB

Parameter		Reason
3. During the Injection Phase following a LOCA.	Why	The maximum flow rates are used in the analysis of mass and energy release rates. The maximum rates are assumed to obtain conservative results.
With a minimum safeguards, the SI System shall deliver a maximum of:	Origin	Flow rates are calculated in [25.4.22] and used in the containment analysis. [25.4.21] [25.4.49]
Flow (gpm) at DP (psi) 4200 -100 4180 -5 4160 0 4005 10 4000 100	Impact	Flow rates greater than those calculated maximums may invalidate the referenced analyses.
With normal safeguards the SI shall deliver a maximum of: Flow (gpm) at DP (psi) 6000 -20 5463 -5 5378 0 5278 10 4800 80		

Table 11.1-5. Plant Safety Analysis Parameters, SI System, Condition IV Events - Limiting Faults, Event: LOCA, MSLB

	Parameter		Reason
5.	During the recirculation phase following a LOCA, the maximum SI flow shall be limited to 4030 gpm for minimum safeguards and 5750 gpm for normal safeguards.	Why	These flow rates are used in analysis of mass and energy release rates. The maximum rates are assumed to obtain conservative results.
		Origin	The flow rates stated are calculated in [25.4.22] and used in the containment safety analysis. [25.4.21] [25.4.49]
		Impact	Flow rates greater than those calculated maximums may invalidate the containment safety analysis.

Table 11.1-5. Plant Safety Analysis Parameters, SI System, Condition IV Events - Limiting Faults, Event: LOCA, MSLB

	Parameter		Reason
23.	The Net Positive Suction Head Available (NPSHA) to the UNIT 1 LHSI pumps shall be a minimum of ~~[illegible]~~ 13.1 ft.	Why	Sufficient NPSH is required to ensure minimum flow requirements are met.
	The Net Positive Suction Head Available ($NPSH_A$) to the Unit 2 LHSI Pumps shall be a minimum of 13.4 Ft.	Origin	The NPSHR assumes minimum safeguards, recirculation phase, and a flow rate c. [illegible]030 gpm. [25.5.6] [25.4.21] [25.4.48] [25.4.49]
		Impact	If NPSH is insufficient, the LHSI pump may not provide adequate flow, invalidating the LBLOCA analyses.

E. Design Requirement - The minimum/maximum design value for the parameter is entered as the "design requirement." This value is normally based directly upon the assumptions, input, or results of the safety analysis or is derived therefrom. Where the value of a design requirement is dependent upon the value of another parameter, the dependency is explained in the "Remarks" column and the dependent parameter listed as a parameter.

F. Design Specification - The value established by the designer for the parameter in the design specification (NAS, NUS, etc.) is recorded.

G. Design Margin - The design margin available for this parameter is presented. Design margin is the difference between the "design requirement" and the "design specification."

H. Installed Value - The actual value for the parameter, specific to the installed component, is recorded. This value must be based on a substantiated source that conclusively demonstrates the stated installed value.

The difference between the "installed value" and the "design specification" represents the performance margin.

I. Total Margin - Total margin is the difference between "design requirements" and "installed value." The total margin is dependent upon the actual installed component and is subject to change if the component is replaced or modified.

J. Reference - References to the applicable sections of this SDBD that contain the reason or basis are provided. Reasons and references for margins may be presented here.

K. Remarks - This column is used to record any pertinent remarks not included elsewhere.

Subsequent sections of this chapter provide discussion of margins presented in Table 12.0-1. A subsection is presented for each major component identified in Table 12.0-1. Key parameters for the component are listed as separate items in the sections.

12.1 ACCUMULATOR TANKS

Design Requirement - Each accumulator shall have a minimum total volume of 1450 ft^3. As described in Table 11.1-5 for the LOCA analysis, a minimum borated water volume and a minimum nitrogen gas pressure (based on remaining volume in accumulator) was assumed. For the containment analysis, a maximum nitrogen volume and pressure was assumed based on an assumed minimum water volume and a total volume of 1450 ft^3. [25.4.21] [25.4.49] requirement ensures that sufficient shutdown reactivity exists during the recirculation phase following a LOCA.

Design Specification - Technical Specifications require that the BIT contain a minimum of 900 gallons of boric acid solution with a minimum concentration of 12950 ppm. [25.5.12] [25.5.13]

Margin Result - The design requirement is met with the margin shown in Table 12.0-1.

5. Design Requirement - The BIT shall contain a maximum of 900 gallons of boric acid solution with a maximum concentration of 15750 ppm. [25.4.30] [24.12.4] This is required to ensure that the ten-hour recirculation path alternation interval is adequate to prevent the precipitation of boric acid solution within the core.

Design Specification - The Technical Specifications limit the maximum concentration to 15750 ppm and the volume to 900 gallons. [25.5.12] [25.5.13]

Margin Result - The design requirement is met with zero margin.

12.3 LHSI PUMPS

A minimum of 13.1 ft of NPSH shall be available to the Unit 1 LHSI Pumps. [25.4.49]

1. Design Requirement - A minimum of 13.4 ft of NPSH shall be available to the Unit 2 LHSI pumps. [25.5.6]

The minimum NPSH available to the Unit 1 LHSI Pumps is 14.1 ft. [25.4.49]

Design Specification - The minimum NPSH available to the Unit 2 LHSI pumps is 13.5 ft. [25.4.21] [24.12.5]

Margin Result - The design requirement is met with the margin shown in Table 12.0-1.

2. Design Requirement - Each LHSI pump shall be capable of supplying boric acid solution to the RC System at the flow rates assumed in the LBLOCA analysis. Refer to Table 11.1-5. [25.4.11]

Design Specification - The flow rates used in the LBLOCA analysis are determined by reducing the LHSI minimum performance curve by five percent of design head over the entire curve. This results in a conservatively low flow rate for input into the LBLOCA analysis. [25.12.4] The LHSI pumps are designed for 3000 gpm with a 250 ft head.

Margin Result - The design requirement is met with margin as shown in Table 12.0-1.

3. Design Requirement - The flow rate provided to the RC System shall not exceed the maximum flow rates assumed in the containment analysis (4030 gpm with minimum safeguards, and 5750 gpm with normal safeguards). [25.4.21] [25.4.49]

   Design Specification - The flow rates used in the containment analysis were calculated based on the LHSI and HHSI pump curves and assumptions which tend to maximize calculated flow. Therefore, it is expected that the actual maximum flow is less than the values stated above. [25.4.21] [25.4.49]

   Margin Result - Actual installed flow rate data is not available; therefore, the requirement is met with zero margin.

12.4 HHSI/CHARGING PUMPS

1. Design Requirement - A minimum of 23.7 ft of NPSH shall be available to the HHSI/Charging pumps. [27.3.18]

   Design Specification - The minimum NPSH available to the HHSI/charging pumps is 39.01 ft. [25.4.14]

   Margin Result - The design requirement is met with the margin stated in Table 12.0-1.

2. Design Requirement - Each HHSI/charging pump shall be capable of supplying boric acid solution to the RC System at the flow rates assumed in the LOCA and MSLB analyses. [25.4.11] [25.4.15] [25.4.16] [25.4.17] [25.4.18] [25.4.19] [24.12.6] [24.12.7]

   Design Specification - The flow rates used in the analyses of concern were calculated by degrading the certified pump curve by 290 ft for the entire curve. This results in a conservatively low flow rate input to these analyses. [25.4.10]

   Margin Result - The requirement is met with margin as shown in Table 12.0-1.

3. Design Requirement - The flow rate provided to the RC System shall not exceed the maximum flow rates assumed in the containment analysis (565 gpm with minimum safeguards and 741 gpm with normal safeguards). [25.4.21] [25.4.49]

   Design Specification - The flow rates used in the containment analysis were calculated based on HHSI pump curve and assumptions which tend to maximize calculated flow. Therefore, it is expected that the actual maximum

Table 12.0-1. Key Component Parameters, SI System

(A) Item No.	(B) System Parameter	(C) Operating Condition	(D) System/ Component	(E) Design Requirement	(F) Design Specification	(G) Design Margin	(H) Installed Value	(I) Total Margin	(J) Reference	(K) Remarks
1	Total volume (ft^3)	Condition IV LBLOCA, Containment Analysis	1-SI-TK-1A 1-SI-TK-1B 1-SI-TK-1C 2-SI-TK-1A 2-SI-TK-1B 2-SI-TK-1C	1450 [25.4.11] [25.4.21] [25.4.49]	1450 [24.12.10]	0	N/A	N/A	Section 12.1	
2	Volume (gal)	Condition IV MSLB	1-SI-TK-2 2-SI-TK-2	900 [25.4.15] [25.4.16] [25.4.17] [25.4.18] [25.4.19]	900 [25.3.5] [24.12.11]	0	N/A	N/A	Section 12.2 item 1	
3	Temperature (°F)	Condition IV MSLB	1-SI-TK-2 2-SI-TK-2	111 [25.4.15] [25.4.16] [25.4.17] [25.4.18] [25.4.19]	111 [25.4.13]	0	115 [25.5.12] [25.5.13]	4	Section 12.2 item 3	Installed value is the minimum allowed by the Technical Specifications
4	Boric Acid Concentration (ppm)	Condition IV MSLB	1-SI-TK-2 2-SI-TK-2	12250 [25.4.15] [25.4.16] [25.4.17] [25.4.18] [25.4.19]	12950 [25.5.12] [25.5.13]	700	N/A	N/A	Section 12.2 item 2	Design specification is the minimum allowed by the Technical Specifications
5	Boric Acid Concentration (ppm)	Condition IV LBLOCA	1-SI-TK-2 2-SI-TK-2	12950 [25.4.20]	12950 [25.5.12] [25.5.13]	0	N/A	N/A	Section 12.2 item 4	Design specification is the minimum allowed by the Technical Specifications

DC 90-13-1, Appendix 4-9, Page 35

Table 12.0-1. Key Component Parameters, SI System

(A) Item No.	(B) System Parameter	(C) Operating Condition	(D) System/ Component	(E) Design Requirement	(F) Design Specification	(G) Design Margin	(H) Installed Value	(I) Total Margin	(J) Reference	(K) Remarks
6	Boric Acid Concentration	Condition IV LBLOCA	1-SI-TK-2 2-SI-TK-2	15750 [25.4.30] [25.6.15]	15750 [25.5.12] [25.5.13]	0	N/A	N/A	Section 12.2 item 5	Design Specification is the maximum allowed by the Technical Specifications
7	NPSH (ft of H_2O)	Condition IV LOCA	1-SI-P-1A UNIT 2 1-SI-P-1B 2-SI-P-1A 2-SI-P-1B UNIT 1	13.4 [25.5.6] 13.1 [25.4.48] [25.4.49]	13.4 [25.5.6] 13.1 [25.4.48] [25.4.49]	0 0	13.5 [25.4.21] 14.1 [25.4.48] [25.4.49]	.1 1.0	Section 12.3 item 1 Section 12.3 item 1	
8	NPSH (ft of H_2O)	Condition IV LOCA	1-CH-P-1A 1-CH-P-1B 1-CH-P-1C 2-CH-P-1A 2-CH-P-1B 2-CH-P-1C	23.7 [25.3.16]	23.7 [25.3.16]	0	39.01 [25.4.14]	15.31	Section 12.4 item 1	Assumes a flow rate of 610 gpm, and minimum safeguards
9	Flow (gpm)	Condition IV MSLB LOCA	1-CH-P-1A 1-CH-P-1B 1-CH-P-1C 2-CH-P-1A 2-CH-P-1B 2-CH-P-1C	[24.12.12]	[illegible] [illegible]ax) [illegible]15]	[24.12.12]	N/A	N/A	Section 12.4	
10	Flow (gpm)	Condition IV LOCA	1-SI-P-1A 1-SI-P-1B 2-SI-P-1A 2-SI-P-1B	4030 [25.4.21] [25.4.22] [25.4.49]	4030 [25.4.21] [25.4.22] [24.12.13] [25.4.49]	0	N/A	N/A	Table 11.1-5	This is the maximum recirculation phase flow assuming minimum safeguards

DC 90-13-1, Appendix 4-9, Page 36

Table 12.0-1 Key Component Paramcters, SI System

(A) Item No.	(B) System Parameter	(C) Operating Condition	(D) System/ Component	(E) Design Requirement	(F) Design Specification	(G) Design Margin	(H) Installed Value	(I) Total Margin	(J) Reference	(K) Remarks
				5750 [25.4.21] [25.4.22] [25.4.49]	5750 [25.4.21] [25.4.22] [24.12.13] [25.4.49]	0	N/A	N/A	Table 11.1.5	This is the maximum recirculation phase flow assuming normal safeguards.
11	Flow (gpm)	Condition IV LBLOCA	1-SI-P-1A 1-SI-P-1B 1-SI-P-1A 2-SI-P-1B	[24.12.14]	3000 gpm [25.3.5]	[24.12.14]	N/A	N/A	Table 11.1.5	
12	Flow (gpm)	Condition IV LOCA	1-CH-P-1A 1-CH-P-1B 1-CH-P-1C 2-CH-P-1A 2-CH-P-1B 2-CH-P-1C	[25.4.21] 565 [25.4.22] [24.12.15] [25.4.49]	565 [25.4.22]	0	N/A	N/A	Table 11.1.5	Maximum calculated flow, assuming minimum safeguards. Injection phase.
				[25.4.21] 741 [25.4.22] [24.12.15] [25.4.49]	741 [25.4.22]	0	N/A	N/A	Table 11.1.5	Maximum calculated flow, assuming normal safeguards. Injection phase.
13	Boric Acid Concentration, minimum (ppm)	Condition IV LBLOCA	1-SI-TK-1A 1-SI-TK-1B 1-SI-TK-1C 2-SI-TK-1A 2-SI-TK-1B 2-SI-tk-1C	1900 [25.4.20]	2200 [25.5.8] [25.5.9] [24.12.16]	300	N/A	N/A	Table 11.1.5	Design Specification is the minimum allowed by the Technical Specifications.

<u>Design Feature</u> - Initial designs of the building structural elements accommodate a limited number of openings and penetrations, providing required configuration and reinforcing steel. The design methodology and acceptance criteria are provided in the PDBD. [24.14.41]

14.1.11 Hydraulic Requirements

This subsection addresses the hydraulic design requirements for the SI System.

1. <u>Requirement</u> - The SI System piping/fittings/valves, sizing, and layout shall address system fluid flow requirements (based on fluid pressure, temperature, and system fluid transients, fluid velocity requirements (to limit pressure drops and erosion). [25.11.1]

   <u>Design Feature</u> - The HHSI suction piping from the RWST is designed for low head losses in order to ensure adequate NPSH to the HHSI/charging pumps. [25.3.5] [25.4.14] [24.14.10]

   The HHSI branch lines are designed for high pressure losses to limit the flow out of a ruptured branch line. [25.3.5] [24.14.10]

   Calculations have been conducted which show that pressure drops within the SI System are sufficient to allow the minimum required flow rates (listed in Table 6.1-1) to be met. [25.4.10] [25.6.19] [25.12.4] [24.14.44]

   The SI System is designed so that the HHSI charging pumps can be isolated on separate headers. This ensures full flow from at least one pump will be delivered to the core should a branch line break. [25.3.5] [24.14.10]

   The LHSI pump suction piping is designed and arranged for low head loss in order to ensure adequate NPSH to the LHSI pumps for the entire period recirculation is required. [25.3.5] [25.4.21] [24.14.10] [25.4.48] [25.4.49]

   Refer to Section 13.2.3 for additional information on SI System pump NPSH.

Table 14.1-1. Process Condition Summary

SYSTEM: Safety Injection | SYSTEM CODE: SI | SDBD NUMBER: SDBD-NAPS | NODE NO.: 1

NODE TYPE: X MASS FLOW _ INTERFACE _ COMPONENT

PARAMETER	DESIGN VALUE	OPERATING CONDITIONS NORMAL	EMERGENCY	SHUTDOWN	OTHER
PRESSURE (psig)	2485.0	Note 1	2235		
TEMPERATURE (°F)	250	Note 1	160		
FLOW RATE (gpm)		Note 1	741		
NPSH REQ'D (ft)		Note 1			
PRESSURE DROP (psi)		Note 1			
HEAT LOAD (Btu/hr)		Note 1			
FLUID TYPE	Boric Acid Solution	Note 1	Boric Acid Solution		
VISCOSITY (centipoise)		Note 1			
FLUID pH					

FLOW IS FROM: RHSI/CHARGING PUMPS TO: BIT

NOTES: (1) The SI System does not operate during normal plant operation. The system is in standby as described in Section 3.3.1.

(2) Pressure and temperature from line designation table.

REFERENCES: [25.4.21] [25.4.22] [25.12.10] [24.14.59] [24.14.80]
[25.4.49]

REVISION NO: 00 DATE: July 1, 1990

Table 14.1-1. Process Condition Summary

SYSTEM: Safety Injection | SYSTEM CODE: SI | SDBD NUMBER: SDBD-NAPS-SI | NODE N 2

NODE TYPE: X MASS FLOW _ INTERFACE _ COMPONENT ____

PARAMETER	DESIGN VALUE	OPERATING CONDITIONS: NORMAL	EMERGENCY	SHUTDOWN	OTHER
PRESSURE (psig)	2485	Note 1	97		
TEMPERATURE (°F)	195	Note 1	160		
FLOW RATE (gpm)		Note 1	6000		
NPSH REQ'D (ft)		Note 1			
PRESSURE DROP (psi)		Note 1			
HEAT LOAD (Btu/hr)		Note 1			
FLUID TYPE	Boric Acid Solution	Note 1	Boric Acid Solution		
VISCOSITY (centipoise)		Note 1			
FLUID pH		Note 1			

FLOW IS FROM: LHSI Pumps TO: RC System Cold Legs

NOTES: (1) The SI System does not operate during normal plant operation. The system is in standby as described in Section 3.3.1.

REFERENCES: [24.14.59] [24.14.80] [25.4.21] [25.12.10]
[25.4.49]

REVISION NO: 00 | DATE: July 1, 1990

Table 14.1-1. Process Condition Summary

SYSTEM: Safety Injection	SYSTEM CODE: SI	SDBD NUMBER: SDBD-NAPS-SI	NODE NO.: 3

NODE TYPE: X MASS FLOW _ INTERFACE _ COMPONENT ____

PARAMETER	DESIGN VALUE	OPERATING CONDITIONS: NORMAL	EMERGENCY	SHUTDOWN	OTHER
PRESSURE (psig)	700.0	601	660		
TEMPERATURE (°F)	150	150	120	86	
FLOW RATE (gpm)		Note 1			
NPSH REQ'D (ft)		Note 1			
PRESSURE DROP (psi)		Note 1			
HEAT LOAD (Btu/hr)		Note 1			
FLUID TYPE	Boric Acid Solution	Note 1	Boric Acid Solution		
VISCOSITY (centipoise)		Note 1			
FLUID pH		Note 1			

FLOW IS FROM: Accumulator TO: RC System

NOTES: (1) The SI System does not operate during normal plant operation. The system is in standby as described in Section 3.3.1.

REFERENCES: [24.14.80] [24.14.82] [25.4.11] [25.4.21] [25.12.10] [25.4.49]

REVISION NO: 00 DATE: July 1, 1990

14.5.2 Performance

Requirement - The accumulators shall be maintained in a condition which satisfies assumptions made in the LOCA and Containment Analyses. [25.4.11] [25.4.20] [25.4.21] [25.4.49]

Design Feature - Each accumulator is maintained under the following conditions, which satisfy assumptions made in the LOCA analyses. [25.4.11] [25.5.8] [25.5.9]

Volume: 7580-7756 gallons
Boration: 2200 ppm - 2400 ppm
Pressure: 599 - 667 psig

14.5.3 Regulations, Codes, and Standards

Requirement - The accumulators shall be designed and installed to generally recognized codes and standards or clearly stated quality requirements to ensure a quality product in keeping with the required safety function.

Design Feature - The accumulators are designed to ASME Section III, Class C. [25.10.5] [24.14.118]

14.5.4 Design Conditions

Requirement - The SI System shall be designed and purchased to be capable of withstanding the pressure, temperature, and flow of the fluid passing through the system.

Design Feature - The accumulators are designed for the following conditions: [25.3.5] [24.14.119]

Capacity:	1450 ft^3
Pressure:	700 psig (internal)
	45 psig (external)
Temperature:	300°F

14.5.5 Loads

1. Requirements - The accumulators shall be designed to withstand seismic loads, wind and snow loads, tornadoes, hurricanes, missiles, and floods due to natural phenomena. [25.1.16] [25.10.1]

12. Virginia Power Letter to NRC, Ser. No. 412, September 16, 1977.

INSERT "A"

13. SWEC Calculation 11715-US(B)-246, "Containment Integrity and Nash Analysis for the Proposed 7.5°F Increase in Tave of Reactor Coolant (LOCA Analysis)," Rev. 1, May 11, 1983. [24.15.3]

16.1 LHSI AND HHSI SYSTEM CURVES

Calculation Number, Revision and Date - 11715-341N, Rev. 0, September 12, 1977 [25.4.22]

Calculation Title - "LHSI and HHSI System Curves (Cold Leg Injection)"

Purpose - To determine LHSI and HHSI flow rates during injection and recirculation phases (cold leg only) of operation for input into LOCTIC.

Assumptions - It was assumed that LHSI pump "can" losses are negligible; LHSI pump, 1-SI-P-1A and HHSI pump, 1-CH-P-1A were used to represent the single and parallel pump operations because they provide the maximum flow/head; linear combination of head loss for the shared HHSI & LHSI path to the R.V., maximum flow through the SI System.

For the injection mode only, the RWST temperature was assumed to be 47°F; head loss datum is calculated at 3,700 gpm (LHSI) and 600 gpm (HHSI); minimum flow from one LH pump; and no seal water injection flow.

For the recirculation mode only, sump temperature is assumed to be 200°F; head loss datum is calculated at 3600 gpm (LHSI) and 600 gpm (HHSI); constant HHSI flow; throttle valves on HHSI discharge lines do not have choked flow making head loss a function of Q^2; sump water height is 4.6 ft.; head loss through screens is negligible; and no return flow through HHSI suction lines during minimum (1 LHSI and 2 HHSI) pump operation.

Inputs - The following sources were used as inputs to the calculation:

1. Crane Technical Paper No. 410, 1976.
2. 11715-SSR-3 "As-Built" Drawings.
3. 11715-FP-4A through D, FP-7A through F, FP-11A,B,C,D,M,N, FP-9B.
4. Westinghouse Data Sheets 9.33 and 9.60.
5. PTR 8-0 Graphical Fluid Systems Analysis.
6. LHSI & HHSI Test Data 1-P.O.-36.3 and 1-PT-57.1B.

Methodology - Graphical analysis was used to determine the system head loss and the LHSI and HHSI flow rates for several modes of operation. The curves used were based on as-built conditions and system piping design.

INSERT
"A"

14. SWEC CALCULATION U2072.1610 - US(B) - 273, "HEAD LOSS ACROSS EMERGENCY SUMP SCREENS DUE TO INSULATION DEBRIS CAUSED BY LOCA EVENT," NAPS UNIT 1, REV. 0, MAY 27, 1992.

15. SWEC CALCULATION 02072.2010 - US(B) - 274, "CONTAINMENT LOCA ANALYSIS WITH NEW STEAM GENERATORS," NAPS UNIT 1, REV. 0, JUNE 17, 1992.

11715/12050-NM(B)-291-DAC, Rev. 0, "Low Head SI Line Pipe Rupture Analysis," SWEC, March 8, 1977.

11715/12050-NM(B)-341-HZQ, Rev. 0, "LHSI Pump turbulence Limiter," SWEC, July 3, 1978.

11715/12050-NM(B)-342-CZQ, Rev. 1, "LHSI Pump Dynamic Analysis," SWEC, January 17, 1978.

11715/12050-NM(B)-346-DAS, Rev. 0, "Nozzle Loads for 6" HPSI Lines," SWEC, May 23, 1978.

11715/12050-NM(B)-354-CBD, Rev. 0, "Stress Analysis of LHSI Pump Motor Support," SWEC, November 1, 1978.

25.4.47 14938.44-UR(B)-012-0, Rev. 0, "North Anna LOCA Dose in Control Room," SWEC, April 26, 1988.

INSERT "B"

25.5 LICENSING CORRESPONDENCE REFERENCES

25.5.1 NRC Letter Serial No. 084/021378, Brown to Case, March 13, 1978.

25.5.2 NRC Letter Serial No. 152, Brown to Case, March 22, 1978.

25.5.3 NRC Letter Serial No. 0701, Ferguson to Denton, August 11, 1980.

25.5.4 Facility Operating License No. NPF-4 (North Anna Power Station Unit No. 1), November 26, 1977, as amended. Section 2.D.

25.5.5 Facility Operating License No. NPF-7 (North Anna Power Station Unit No. 2), August 21, 1980, as amended. Section 2.C.

25.5.6 Letter to NRC, Serial No 412, "Recirculation Spray and LHSI Pump NPSH," Virginia Power, September 16, 1977.

25.5.7 Letter to NRC, Ser. No. 963, Virginia Power, April 15, 1976.

25.5.8 Technical Specification 3/4.5, NAPS Unit 1, April 14, 1987.

25.5.9 Technical Specification 3/4.5, NAPS Unit 2, April 14, 1987.

25.5.10 Technical Specification 3/4.6, NAPS Unit 1, December 14, 1988.

25.5.11 Technical Specification 3/4.6, NAPS Unit 2, December 14, 1988.

INSERT

"B"

25.4.48 02072.1610 - US(B) - 273, REV. 0, "HEAD LOSS ACROSS EMERGENCY SUMP SCREENS DUE TO INSULATION DEBRIS CAUSED BY LOCA EVENT," NAPS UNIT 1, SWEC, MAY 27, 1992.

25.4.49 02072.2010 - US(B) - 274, REV. 0, "CONTAINMENT LOCA ANALYSIS WITH NEW STEAM GENERATORS," NAPS UNIT 1, SWEC, JUNE 17, 1992.

25.12.8 "NAPS Unit 2 In-service Testing Program Plan for Pumps and Valves First Inspection Interval December 14, 1980 - December 14, 1990," Rev. 4, Virginia Power, January 16, 1989.

25.12.9 Line Designation Table, NAPS Unit 1, Rev. 9, June 7 1978.

25.12.10 "In-Service Inspection Manual," Virginia Power, March 31, 1988.

25.12.11 E&DCR No. P-2152, "NA1 LHSI/RS Cross-Connect," January 17, 1978.

25.12.12 Westinghouse Motor Data Sheet, SWEC File No. 11715-1.10-26A and 12050-1.10-26A, March 3, 1971.

25.12.13 DCP-84-55, "RG 1.97 HPSI Flow Transmitter Modifications."

25.12.14 DCP-84-56 "RG 1.97 HPSI Flow Transmitter Modifications."

25.12.15 DC-90-13-1, Steam Generator Replacement North Anna Unit 1.

NAPS QS

in the most fundamental manner to allow the designer/engineer flexibility in establishing design features to satisfy the criteria. Generally, the fundamental performance criteria can be satisfied by many designs. The primary objective in establishing performance criteria is to define the fundamental information needed to design the system that is independent of component selection or system configuration.

The performance criteria identified in this chapter are presented "qualitatively." The performance requirements are stated in "quantitative" terms in Chapter 6, "Key System Parameters."

The performance criteria for the QS System are identified in the following sections:

2.2.1 Safety-Related Performance Criteria

This subsection identifies the safety-related performance criteria for the QS System. Performance criteria related to the same function are grouped together. The following are safety-related performance criteria:

1. In the event of an accidental release of high-energy fluids inside containment:

   a. In support of the heat removal functions in Section 2.1.1 and in order to meet the requirements of the containment safety analysis to limit, in conjunction with the RS System, the containment pressure and temperature to less than design and to depressurize the containment within 1 hour following an accident, the QS System shall meet the following performance criteria. It should be noted that the performance criteria listed below are based on the LOCA and main stream line break (MSLB) since the consequences of these two accidents envelop the consequences of other accidents (e.g., rod ejection accident (REA), feedwater line break (FWLB)) that may cause containment pressurization. [24.2.5]

      - The water from the RWST shall be delivered to the containment structure as atomized spray to cool and depressurize the containment atmosphere at the flow rate assumed in the containment safety analysis. [25.4.3] [25.4.5] [25.4.6] [25.4.7]

        [25.4.63] [24.2.14]

- The temperature of the spray water delivered to the containment atmosphere shall be in accordance with the containment safety analysis. [25.4.3] [25.4.5] [25.4.7]

[25.4.63] [24.2.14]

- The usable volume of water in the RWST shall be as established in the containment safety analysis. [25.4.3] [25.4.5] [25.4.7]

- The QS System shall be initiated by the CDA signal which shall be activated by a containment pressure setpoint in accordance with the safety analysis. [25.4.3] [25.4.5] [25.4.7]

- The maximum time delay allowable between initiation of the CDA signal and obtaining an effective spray shall be within the limits established in the safety analyses. [25.4.3] [25.4.5] [25.4.7]

- The QS containment coverage shall be maximized to support heat removal. [25.4.3] [25.4.5] [25.4.7] [25.4.24] [24.2.6]

- The minimum droplet thermal effectiveness of the sprays shall be as documented in the containment safety analyses. [25.4.3] [25.4.5] [25.4.7] [24.2.7]

b. In support of the iodine removal function in Section 2.1.1, the QS System shall meet the following performance criteria. It should be noted that the performance criteria listed below are based on the LOCA since its consequences envelop the consequences of other accidents (e.g., FWLB, MSLB, REA) that may result in radiological releases to the environment. [25.4.10] [25.4.12]

[24.2.14]

sump. [25.3.30] [25.4.1] [25.11.1] [24.2.2]

c. In support of the pH control function in Section 2.1.1, the QS System shall meet the following performance criteria:

- The performance parameters as listed in Section 2.2.1, with the exception of spray coverage.

- The ultimate sump pH shall be greater than 7.0. [25.1.6] [25.4.1]

d. In support of ensuring that the IRS pumps have adequate NPSH, water from the QS System shall be delivered to the suction of the IRS pump at the flow rate and temperature assumed in the IRS pumps' NPSH analyses. [25.4.3] [25.5.1]

e. In support of the containment isolation function identified in Section 2.1.1, during an accidental release of high-energy fluids inside containment (e.g., LOCA, MSLB, FWLB, REA) the leakage of containment atmosphere via the QS System containment isolation valves shall be within the acceptable limits defined in 10CFR50, Appendix J. [25.1.22] [25.1.52] [25.5.3] [25.5.4] [25.10.8]

f. In support of the emergency core cooling function in Section 2.1.1, borated water from the RWST shall be utilized by the SI System to be injected into the reactor core at the flow rate assumed in the containment safety analysis and documented in SDBD-NAPS-SI. [25.3.28] [25.4.3] [25.4.5] [25.4.7]

[25.4.63] [24.2.14]

g. In support of providing minimum recirculation flow capability to the LHSI pumps (Section 2.1.1), a flow path shall be available between the LHSI pumps' discharges and the RWST. [25.6.1] [25.8.2] [25.8.5]

h. In support of initiating long-term emergency core cooling (Section 2.1.1), the QS System shall provide the initiating signal for re-alignment of the

LHSI pump suction (from the RWST to the containment sump) on a "low" RWST level setpoint, which is in accordance with the containment safety analyses and documented in SDBD-NAPS-SI. [25.3.28] [25.4.3] [25.4.5] [25.4.7] [24.2.1]

[25 .. 03]

2. The RWST shall support the water source function discussed in Section 2.1.1, item 2 without impacting the volume, temperature, and chemistry requirements (identified in Section 2.2.1, item 1) imposed on the RWST fluid by the containment analyses. [25.3.30] [25.4.3] [25.4.5] [25.4.7] [25.5.11] [25.11.1] [24.2.2]

2.2.2 Non-Safety-Related Performance Criteria with Special Regulatory Significance

This subsection identifies the non-safety-related with special regulatory significance (NSQ) performance criteria for the QS System. Performance criteria related to the same function are grouped together. The following are NSQ performance criteria:

1. In support of the fire safe shutdown function discussed in Section 2.1.2, the QS System shall meet the following performance criteria:

   a. The QS System (without chemical addition) shall be used to cool the containment, after a fire inside containment, if it is necessary to enter the containment before the temperature inside has decreased to a suitable level for access. [25.5.5]

      The QS pumps shall be used as needed, at rated flow for 10- to 15-minute periods. The time limit is intended to allow the operator to evaluate whether further spraying via the use of the QS System is necessary. [25.5.5] [25.11.1]

   b. The RWST shall support the water source function discussed in Section 2.1.2 related to safe shutdown after a fire. The performance criteria for the RWST fluid, i.e., temperature, volume, and chemistry, are based on charging system performance criteria for that specific event. It should be noted, however, that the volume, temperature, and boration requirements imposed on the RWST fluid by the containment analyses (see Section 2.2.1) envelop the performance criteria imposed on the RWST by the charging system. [25.3.30] [25.4.3] [25.4.5] [25.4.7] [24.2.2]

[25.4.63]

2. In support of the RG 1.97 function, the RWST water level and the QS flow shall be measured. [25.5.6] [24.2.12]

3. The RWST shall support the water source function discussed in Section 2.1.2 related to safe shutdown after an ATWS on an as needed basis. There are, however, no performance criteria imposed on the QS System due to the ATWS issue.

4. The RWST shall support the water source function discussed in Section 2.1.2 related to safe shutdown after a station blackout. The performance criteria for the RWST fluid are based on charging system performance criteria relative to this issue. The requirements imposed on the RWST fluid by the containment analyses (see Section 2.2.1) envelop the performance criteria imposed on the RWST by the charging system. [25.3.30] [25.4.3] [25.4.5] [25.4.7] [25.5.11] [24.2.2]

[25.4.63][24.2.14]

2.2.3 Non-Safety-Related Performance Criteria

This subsection identifies the non-safety-related performance criteria for the QS System. Performance criteria related to the same function are grouped together. The non-safety-related performance criteria are:

1. In support of the QS System function to maintain the temperature of the RWST during modes 1, 2, 3, and 4 (as identified in Section 2.1.3), the temperature of the fluid stored in the RWST shall be reduced to and automatically maintained within the requirements of the containment safety analysis and as stated in Section 2.2.1. [25.3.32] [25.4.3] [25.4.5] [25.4.7] [25.4.26]

[25.4.63]

2. The RWST shall support the water source functions discussed in Section 2.1.3 without impacting the volume, temperature, and chemistry requirements imposed on the RWST fluid by the containment safety analysis during modes 1, 2, 3, and 4 (see Section 2.2.1) [25.4.3] [25.4.5] [25.4.7] [25.11.1] and the need to supply the backup source of borated water to the CH System to ensure the availability of reactivity control during modes 5 and 6. [25.3.30] [24.2.2] The performance criteria for the RWST for modes 1, 2, 3, and 4 encompass those for modes 5 and 6.

3.0 SYSTEM DESCRIPTION

This chapter contains a brief system description and system operation overview. The primary purpose of this chapter is to familiarize the user with the system and to serve as an introduction to the design basis information which is presented in subsequent chapters. The information presented addresses the broad perspective and major features of the system, with reference to appropriate schematic or block diagrams. The chapter is intended primarily to assure that the SDBD is a stand-alone document and to provide orientation to the user.

The actual configuration of the system, including component arrangement, is not a design basis requirement since many configurations could be established to satisfy design basis requirements. Therefore, the information in this chapter does not constitute part of the design basis but does provide useful information for understanding and applying the design basis requirements.

The controlled configuration drawings define the system configuration and arrangements and should be consulted by the user for current, up-to-date system information. Station procedures should be consulted to determine current system operation practice.

3.1 SYSTEM DESCRIPTION

This section provides a description of the QS System. The system description is not a design basis requirement, but is presented to assist the user in understanding and applying the information contained in the SDBD. The user should refer to controlled configuration management documents for specific information and current conditions.

The QS System is described in terms of performance of its primary functions: the limiting of containment pressure and temperature rise in containment in the event of a high-energy line break inside the containment such as a LOCA or MSLB, and the removal of iodine gas from the containment atmosphere in the event of release of iodine from the RC System during one of these accidents. [25.1.2] [25.1.9] [25.4.2] [25.4.5] [25.4.7] [25.4.63] These two aspects of the QS System's performance affect the release of radioactive materials to the outside atmosphere following an accident. Depressurizing the containment atmosphere decreases the duration and rate of leakage to the outside environment. Removing the iodine reduces the amount of one of the most biologically harmful radioactive fission products from the containment atmosphere, thereby reducing the amount available for release to the environment. [25.4.10] [25.4.11] [25.4.25] The containment depressurization results from the combined effects of the QS System and the RS System, with the RS System removing the heat energy from the containment by transferring it to the ultimate heat sink supplied by the Service Water (SW) System. [25.4.5]

The QS System is shown schematically in Figure 3.1-1. It is composed of a single large yard tank containing the emergency supply of borated water for all Engineered Safety Feature (ESF) Systems; a second, smaller, yard tank containing a sodium hydroxide solution (NaOH); and two independent QS flow paths, pumped by two independent pumps from the main tank to spray rings inside the containment. [25.8.1] [25.8.2]

The main tank (RWST) is maintained full of cooled, borated water. The temperature is maintained by means of a dedicated recirculation flow path through heat exchangers cooled by either plant chilled water or refrigeration units. The chemistry of the RWST fluid is maintained by means of a chemistry-controlled makeup source. [25.4.26] [25.8.1] [25.8.2] The piping from the tank to the QS pumps is maintained flooded, with the suction motor-operated valves (MOVs) open. [25.8.1] [25.8.2]

If the containment pressure rises to the high-high setpoint, redundant signals (CDA) initiate startup of the QS pumps. The CDA signal can also be initiated manually. The signals also initiate opening all MOVs along the flow paths to the containment spray rings. The signals also start a 5-minute delay timer to open redundant isolation MOVs in a pipe connection between the RWST and the chemical addition tank (CAT). [25.8.10] [25.8.1] [25.7.1] The opening of the MOVs allows gravity flow to empty the CAT's contents to change the pH of the RWST sprayed borated water to the basic range to enhance iodine gas removal from the containment atmosphere and to inhibit chloride stress corrosion cracking of equipment inside the containment.

The optimum pH of 8.5 to 11 for iodine removal is accomplished by the initial mixing of the NaOH solution from the CAT with the RWST water within the RWST's weir while the water is being pumped and sprayed into the containment. [25.1.9] [25.4.1] The optimum pH of 7 to 8 for corrosion protection is accomplished by the QS spray water mixing with the spilled reactor coolant and SI water in the containment sump. [25.1.8] [25.4.1] The water that is pumped by the SI System from the RWST has the pH of the borated water in the RWST without the NaOH. The low head safety injection (LHSI)/high-head safety injection (HHSI) water from the RWST eventually passes through the RC System to the containment and to the containment sump where it mixes with the drained water from the QS System sprays causing the pH to be reduced from 8.5 - 11 to 7 - 8. [25.4.1] This mixture of water spilled from the RC System break, sprayed by the QS pumps through the QS ring headers, and added directly to the IRS pump casing for NPSH reasons is sprayed by the RS System. [25.4.5]

The QS pumps, located in the Quench Spray Pump Area (QSPA), discharge through strainers capable of removing solid material that might clog the spray nozzles. [25.6.2] After the strainers, the pumps discharge through MOVs used to isolate the pump and strainer during maintenance, to isolate the spray rings during pump testing, and to isolate the containment atmosphere during operation of the reactor. [25.3.2] [25.10.6]

The QS fluid, on passing through the isolation MOVs, is sprayed inside the containment in the dome area through the spray nozzles mounted on a single 360-degree ring header per pump. [25.4.23] [25.8.6] In the event of an MSLB or LOCA, the total transit time from CDA to spray is no greater than 66 seconds. [25.4.5] [25.4.9]

[25.4.63]

The sprayed QS liquid from either of the ring headers forms a fine mist of droplets, with low temperature and pH of 8.5 to 11.0, which covers approximately 38 percent of the containment air space. [25.4.24] This mist interacts with the steam and other containment gases to absorb heat and remove iodine. [25.1.2] [25.1.9] The QS fluid eventually drains to the containment sump from where it is circulated, cooled, and sprayed by the RS System. [25.8.1] [25.8.2]

The QS pumps complete their function when the water in the RWST is pumped to the lowest possible level. [25.3.7] [25.3.8]

The equipment in the QS System is discussed in more detail below and summarized in Section 3.2.

The RWST stores borated water maintained at a temperature between 40°F and 50°F for use by the QS System and the SI System during an accident inside the containment. [25.4.5] [25.4.26] [25.4.63] [25.6.9] [25.6.10] The RWST contents are also used by the SI System to flood the reactor containment refueling cavity for refueling operations. [25.4.26] [25.3.28] [24.2.1]

The RWST is a large, flat-bottomed, cylindrical stainless steel, free-standing tank with a hemispherical roof. The tank contains an 8-ft 3-in.-high internal weir around the QS pump suction nozzles, which enhances mixing of the NaOH and limits the amount of NaOH that disperses into the rest of the tank. The weir has openings to increase the usable volume of water from the RWST. [25.3.2] [25.8.7] [25.6.5]

The RWST internals include turned down nozzles with tail extensions on the two QS pump suction lines and on the CAT discharge inside the weir volume. [25.8.7] Nozzles enhance the NaOH mixing at the QS pump suction. Spray nozzles for recirculating QS pump discharge are provided for testing the pumps and determining the cleanliness of the water. [25.10.6]

The temperature of the water in the RWST is initially lowered by recirculation of the fluid using the dedicated recirculation pumps and two heat exchangers cooled by the plant's Chilled Water (CD) System. The temperature of the water is maintained by recirculation by the same pumps through the refrigeration units. [25.4.26] [25.10.6] The recirculation pumps are parallel-piped vertical, two-speed centrifugal pumps located near the QS pumps in the QSPA. The pumps are not safety related, but the redundancy is provided for reliability.

The recirculation loop contains parallel redundant heat exchangers of the shell and tube design. Chilled water from the turbine plant is used on the shell side to cool the water returned from refueling down to 43°F. These heat exchangers are located in the yard adjacent to the RWST. In an alternate path to the heat exchangers, a parallel pair of refrigeration units is provided to further cool the RWST water from 43°F to 40°F. [25.6.4]

Table 3.2-1. Major Component Summary

Item No.	Component Name	Function
	Refueling Water Storage Tank (RWST)	Stores borated water at a specified temperature for use in the QS and SI Systems during a LOCA, MSLB, or a transient that initiates SI. [25.4.5] [25.4.63]
		Backup source of water to support makeup to the volume control tank, SI accumulator, and casing cooling tank. [25.3.2]
		Backup source of borated water to the CH System to support safe shutdown. Refer to Section 2.1.1 and 2.1.2.
		Source of water for RC System hydrostatic test, refueling operations, and LHSI pump testing and accumulator check valve leak testing. [25.3.2] [25.4.26]
	RWST Coolers	Provide initial cooling of the RWST water to a temperature of about 45°F. [25.3.2] [25.4.26]
	RWST Refrigeration Units	Maintain the RWST water temperature between 40°F and 43°F. [25.4.26] [25.6.4]
	RWST Recirculation Pumps	Recirculate the RWST water through the RWST coolers or refrigeration units, as required. [25.4.26] [25.4.32]
	QS Pump Suction Isolation Valves	Normally open MOVs to the suction of the QS pumps. Isolate QS pump suction. [25.3.2] [25.10.6]
	QS Pumps	Provide the motive force for the flow from the RWST to the spray rings and provide cool water to the IRS pumps for NPSH requirements. [25.3.2] [25.8.1] [25.8.6]
	QS Pump Discharge Strainers	Remove debris in the pump discharge that could clog the spray nozzles. [25.3.2] [25.10.6]
	QS Pump Discharge Isolation Valves	Normally closed MOVs that open upon a CDA signal to allow flow of QS water to the spray rings. [25.8.1] [25.8.2]
		Close to provide containment isolation. [25.10.6]
	Chemical Addition Tank (CAT)	Stores NaOH solution for addition to the RWST QS water by gravity feed. [25.4.1] [25.4.2]
	CAT Recirculation Pump	Fills, recirculates, and samples the contents of the CAT. [25.4.31]
	CAT Isolation Valves	Normally closed MOVs that prevent premature mixing of the NaOH in the CAT with the RWST water. [25.7.1]

		CH System to achieve and maintain shutdown reactivity, temperature, and pressure in the reactor. RWST water may also be used to bring the plant to safe shutdown after an ATWS event	
5.1.2	Electrical Instrumentation (EI)	The QS System shall provide the EI System process instrumentation cabinets with signals to enable the process instrumentation cabinets to control and monitor QS System parameters.	[25.3.50] [24.5.2]
5.1.3	Emergency Response Facilities Computer (ERF)	The QS System shall provide input for the ERF System indication.	[25.12.4] [25.12.5] [24.5.3]
5.1.4	Plant Computer (CM)	The QS System shall provide the CM System with input to aid the operator in tracking key system paramaters.	[25.3.51] [24.5.4]
5.1.5	Recirculation Spray (RS)	1. The QS System shall provide low-temperature water to the suction of the IRS pumps to increase available NPSH.	[25.4.63] [25.3.29] [25.4.5] [25.5.1] [24.5.5]
		2. The QS System shall provide makeup water to the RS System casing cooling tank (as necessary) via the use of hose connections.	[25.3.27] [24.2.4]
		3. The QS System shall provide water for IRS pump testing. A portable pump and piping shall be used to transfer the water from the refueling cavity to the diked sump and back after test completion.	[25.10.6] [24.5.6]
5.1.6	Safety Injection (SI)	1. The QS System shall provide water to the SI System for	[25.3.28] [24.2.1]

injection into the reactor core for postaccident heat removal.

2. The QS System RWST shall provide water to the SI System to fill or raise the water level of the accumulators. [25.3.28]

3. The QS System shall provide water to fill the refueling cavity by way of the SI System pumps during refueling operations. [25.3.28] [24.2.1]

4. The QS System shall provide water to the SI System to support the hydrostatic test of the RC System. [25.3.28] [24.2.1]

5. The QS System shall provide water for the LHSI pump testing. This flow path also supports minimum recirculation flow from the LHSI pumps during injection mode. [25.3.28] [24.2.1]

6. The QS System shall receive excess water via the accumulator test line during accumulator check valve leak testing. [25.3.28] [24.2.1]

7. The QS System shall realign the LHSI pumps from taking suction from the RWST to taking suction from the containment sump when the water level in the RWST reaches the required switchover level. [25.3.28] [25.4.3] [25.4.5] [25.4.7] [25.4.63] [24.2.1]

5.2 SUPPORTING SYSTEMS

This section identifies the systems that provide support services to the QS System. The interfacing systems must provide support services to the QS System to function as designed. The requirements and/or limitations of the supporting systems are defined,

Table 6.1-1. Key System Parameters, Safety-Related Functions, QS System

(A) Item No.	(B) System Parameter	(C) Function	(D) Operating Condition	(E) Design Requirement	(F) Reference	(G) Remarks
1	Volume (gal)	RWST volume to support accident mitigation	Condition IV • LOCA • MSLB • FWLB • REA	Min: 448,091 [25.4.47]	Section 2.2.1	Includes volume required to support containment safety analyses plus unusable volume. To ensure postaccident availability, this volume of water is maintained in the RWST during Modes 1, 2, 3, and 4. It encompasses the RWST volume requirement of 80,456 gallons at 2300 ppm [25.4.47] relative to providing a backup source of borated water to support negative reactivity control during Modes 1, 2, 3, and 4.
2	Volume (gal)	RWST volume to support negative reactivity control	Condition I • Shutdown • Refueling	Min: 26,254 (At 2300 ppm boron) [25.4.47]	Section 2.2.1	Includes volume to support negative reactivity control plus unusable volume.
3	Temperature (°F)	RWST temperature to support accident mitigation	Condition IV • LOCA • MSLB • FWLB • REA	Min: 40 [25.4.13] Max: 50 [25.4.3] [25.4.5] [25.4.7] [25.4.63]	Section 2.2.1	Also see Table 6.3-1, Item 1.
4	Temperature (°F)	RWST temperature to prevent freezing	Condition I • Shutdown • Refueling	Min: 35 [25.3.30] [24.6.2]	Section 2.2.1	
5	Flow Rate (gpm)	QS flowrate to support accident mitigation	Condition IV • LOCA • MSLB • FWLB • REA	2000 [25.4.3] [25.4.5] [25.4.7] [25.4.63]	Section 2.2.1	The flow of 2000 gpm is a designated design point. QS flow follows a system curve. 150 gpm for IRS pumps is included in flow.

DC 90-13-1, Appendix 4-9, Page 60

Table 6.1-1. Key System Parameters, Safety Related Functions, QS System

(A) Item No.	(B) System Parameter	(C) Function	(D) Operating Condition	(E) Design Requirement	(F) Reference	(G) Remarks
6	Flow Rate (gpm)	QS Flow to IRS pump suction	Condition IV • LOCA • MSLB • FWLB • REA	Min: 150 [25.4.3] [25.4.5] [25.4.63]	Section 2.2.1	At 50°F as set by parameter No. 3.
7	Spray Actuation Time (sec)	Time after CDA signal when full flow QS spray is delivered in containment	Condition IV • LOCA • MSLB • FWLB • REA	66 [25.4.3] [25.4.5] [25.4.7] [25.4.63]	Section 2.2.1	Spray begins within 66 seconds of a CDA signal after a LOCA or MSLB which are the controlling accidents.
8	Spray Coverage Volume (cubic ft)	Containment volume covered by QS spray	Condition IV • LOCA • MSLB • FWLB • REA	721,000 [25.4.12]	Section 2.2.1	Spray coverage based on existing configuration and used in control room dose analyses. [24.2.6]
9	Droplet thermal effectiveness	Heat removal capability of QS droplets	Condition IV • LOCA • MSLB • FWLB • REA	0.9 [25.4.3] [25.4.5] [25.4.7] [24.2.7] [25.4.63]	Section 2.2.1	
10	Volume (gal)	CAT volume	Condition IV • LOCA • MSLB • FWLB • REA	Min: 4800 [25.[illegible]1]	Section 2.2.1	Basis is spray and sump water pH requirements.

Insert 7.1

spray ring headers to divert cool water from the RWST to the suction of the RS pumps. [25.5.1] [25.7.12]

7.2 PROGRAMMATIC AND REGULATORY ISSUE HISTORY

This section identifies and discusses programmatic topics or regulatory issues that have influenced the original design basis. This discussion is presented to summarize the overall effect of the programmatic and regulatory issues on the design philosophy and design basis, and not to discuss specific requirements or changes attributable to the issues. A more complete discussion of programmatic issues is presented in Chapter 15.

7.2.1 Appendix R

In response to the 10CFR50, Appendix R requirements, the QS System was evaluated for lowering the temperature inside the containment following a fire. The QS pumps can be used for short periods of 10 to 15 minutes to cool the containment, after a fire inside containment, if necessary for access. The QS System can be operated in this manner with the CAT isolated to prevent introduction of NaOH spray into the containment. No design changes to the QS System were necessary. [25.5.5]

7.2.2 RG 1.97

In response to RG 1.97 requirements, the RWST fluid level and the QS System flow are measured to comply with RG 1.97. No changes to the QS System design were necessary. [25.5.6]

7.3 DESIGN CHANGES

There have been no design changes implemented on the QS System that have affected the design basis.

7.4 CURRENT DESIGN PHILOSOPHY

The design basis presented in this SDBD constitutes the current design philosophy for the QS System.

<u>Insert 7.1</u>

In the late 1980's, it was noted that the SGs at NAPS-1 were experiencing corrosion-related degradation resulting in the requirement for frequent inspection, and plugging of significant number of SG tubes. Despite improvcments in secondary water chemistry, tube degradation continued to occur. In 1990, engineering efforts were initiated to replace the SG lower tube bundle assemblies at NAPS-1. The replacement SGs were to be fabricated and analyzed to standards which were at a minimum, equivalent to the existing equipment. 8 [25.12.1]. As noted in Design Change Package DC 90-13-1 [25.12.11], the new Model 51F SG was an improvement over the old Model 51 SG and from a safety analysis perspective, could be considered a "replacement" for the Model 51. 8

However, since some of the SG parameters were slightly different and due to the use of a different type of insulation, the containment LOCA analysis was revised for NAPS-Unit 1 [25.4.63]. The system parameters for the QS system were however not impacted with the following exception. The LHSI pump switchover setpoint had to be revised to address the NPSH requirements for the LHSI pumps.

01287.B07

Table 9.4-1. Nuclear Industry Codes and Standards, QS System

Number	Rev. No./Date	Issuing Organization	Title	Reference	Compliance	Type	Code Reconciliation
5.1	1979	ANSI	Decay Heat Power in Light Water Reactors	[25.2.15] [25.4.5] [25.4.63]	Complete	Self-imposed	None
N18.2	1973	ANSI	"Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants"	[25.2.6] [25.10.16]	Complete	Licensing Commitment	None
N45.2.11	1974	ANSI	"Quality Assurance Requirements for the Design of Nuclear Power Plants"	[25.5.12] [25.1.62] [25.2.20]	Complete	Licensing Commitment	None

DC 90-13-1, Appendix 4-9, Page 64

11.1.2 Condition II Events - Faults of Moderate Frequency

This subsection identifies the Condition II events applicable to the QS System. Condition II occurrences include incidents, any one of which may occur during a calendar year for a plant. Each event is briefly described below and is summarized in Table 11.1-3.

The design for Condition II events is that the event shall be accommodated with, at most, a shutdown of the reactor with the plant capable of returning to operation after corrective action and any release of radioactive materials in effluents to unrestricted areas shall be in conformance with 10CFR20.

For all Condition II events, the QS System is on standby (with the exception of the components needed to automatically maintain the temperature of the RWST fluid) and is aligned as described in Section 3.3.1.

There are no safety analyses associated with QS System operation during Condition II events.

11.1.3 Condition III Events - Infrequent Faults

This subsection identifies Condition III events applicable to the QS System. Condition III occurrences include incidents, any one of which may occur during the lifetime of the plant. Each of the events is described briefly below and is summarized in Table 11.1-4.

The design requirement for a Condition III event is that (1) the incident shall not cause more than a small fraction of the fuel elements in the reactor to be damaged, although sufficient fuel element damage might occur to preclude resumption of operation for a considerable outage time, (2) the release of radioactive materials may exceed 10CFR20 but shall not be sufficient to interrupt or restrict public use of those areas beyond the exclusion radius, and (3) the Condition III events shall not, by themselves, generate a Condition IV fault or result in a consequential loss of function of the Reactor Coolant System or reactor containment barriers.

For all Condition III events, the QS System is on standby (with the exception of the components needed to automatically maintain the temperature of the RWST fluid) and is aligned as described in Section 3.3.1.

The QS System may be activated automatically in the event of a loss of reactor coolant from small ruptured pipes, or minor secondary system pipe breaks if the containment "high-high" pressure setpoint is reached. However, these accidents are not analyzed since their consequences are encompassed by more severe accidents like the LOCA or the MSLB. [25.7.8]

[24.2.14]

There are no safety analyses associated with QS System operation during Condition III events.

11.1.4 Condition IV Event - Limiting Faults

This subsection identifies the Condition IV events applicable to the QS System. Condition IV occurrences are faults that are not expected to occur, but are postulated because their consequences would include the potential for the release of significant amounts of radioactive material. Condition IV faults are the most drastic that must be designed against, and thus normally represent the limiting design case. Each event is described briefly below and is summarized in Table 11.1-5.

The design requirement for a Condition IV event is that Condition IV faults shall not cause a release of radioactive material that results in an undue risk to public health and safety exceeding the guidelines of 10CFR100 and a single Condition IV fault shall not cause a consequential loss of required functions of systems needed to cope with the fault including those of the Reactor Coolant System and the reactor containment system.

During the Condition IV events listed below, the QS System is automatically initiated and provides cool borated water, in the form of atomized spray, to cool and depressurize the containment atmosphere to subatmospheric conditions.

1. Major RC System Pipe Rupture (LOCA).
2. Rupture of a Main Steam Pipe (MSLB inside containment)
3. Major Secondary System Pipe Rupture (FWLB)
4. Rupture of a Control Rod Drive Mechanism Housing (REA)

(It should be noted that the FWLB and REA are not analyzed for containment temperature and pressure since their consequences are enveloped by the MSLB and LOCA.) [24.2.14]

1. The QS System is actuated by a CDA signal from the RPS System. [25.4.3] [25.4.63]
2. Spray begins within 66 seconds after the CDA signal in the event of a LOCA or MSLB. [25.4.3] [25.4.63]
3. Flow through the system is based on the QS pump curve. [25.4.3] [25.4.6] [25.4.63]

4. The temperature of the spray water is between 40° and 50°F. [25.4.5] [25.4.63]

5. NaOH is added to the spray water 5 minutes after the CDA signal. [25.4.2] [25.7.1] [25.7.2]

The QS pumps are capable of delivering a flow of 2000 gpm at a total discharge head of 265.5 ft. [25.4.14]

The QS System flow is based on the QS pump flow curve developed in calculation [25.4.6]. This curve is used as part of the input data to the containment LOCA analysis and the MSLB analysis. [25.4.5] [25.4.7] [25.4.63]

The maximum temperature of the QS water from the RWST is established and verified by the containment LOCA analysis. [25.4.5] [25.4.63] The maximum allowable RWST water temperature is controlled at ≤ 50°F. [25.6.9] [25.6.10] [25.4.63]

The minimum allowable QS water temperature is calculated to be < °F. [25.4.13] The minimum temperature is established by considering an accident that results in the inadvertent actuation of one QS pump.

Water for the QS System is stored in the RWST. The RWST water is not totally dedicated to QS operation; a portion of the water is required for safety injection. When a containment analysis is performed, the minimum quantity of RWST water is conservatively used in the analysis.

[25.4.63] For NAPS Units 1 and 2, 435,361 gallons are required for QS and SI. The various system volumes and the corresponding RWST elevations are documented in calculations [25.4.3] The minimum usable tank volume is used as input to the containment analysis. This analysis establishes and documents the minimum required volume of water to be available for the QS System. The minimum RWST water level is controlled at elevation 326 ft, 6 in. [25.4.3] [25.4.63]

The RWST elevation at which low head SI (LHSI) suction is switched from the RWST to the containment recirculation sump directly impacts the total volume available to the QS System. The water volume removed from the RWST prior to manual LHSI switchover is 315,252 gal at the low-level alarm of the RWST. For unit 2 The LHSI auto-switchover setpoint is at elevation 287 ft, 5 in. [25.4.3], For unit 1 the LHSI auto-switchover setpoint was revised by reference [25.4.63].

The QS System spray headers are located in the upper portion of the containment dome at elevations 393 ft, 3 in. and 391 ft, 10 in. (approximately 100 ft above the operating floor). [25.4.23] The QS nozzles are positioned to spray as much containment volume as practical for effective heat and fission-product removal. The total volume of containment that is covered by the QS is 721,000 cubic ft. [25.4.24] A conservatively low QS heat removal effectiveness of 90 percent is assumed in the containment analysis. [25.4.5] The assumption of low QS heat removal effectiveness minimizes heat rejection to the containment sump while maximizing containment pressure. A 100-percent heat removal efficiency is conservatively assumed for the NPSH analysis. [25.4.5] This assumption maximizes the amount of heat rejection to the containment sump while minimizing containment pressure. Low containment pressure and high sump water temperature reflect a conservative prediction of NPSH.

The QS System is used for the addition of NaOH to the containment sump. This chemical is added both to enhance fission product removal and to provide a spray environment that minimizes the potential for stress-corrosion cracking. The QS pH is calculated in [25.4.1] [25.4.9] [25.7.1] [25.7.2] [25.7.3]

[25.4.03]

The safety analyses require that the QS pumps be started immediately upon receipt of a CDA signal. The time between the start of a QS pump and the time the QS becomes effective is 66 seconds, maximum. [25.4.3] The major part of this time is attributed to system fill; the remaining time is devoted to sequencing and margin times.

The QS System has been designed to meet single failure criteria as defined in Appendix A to 10CFR50. [25.4.4]

The QS System consists of two identical, redundant subsystems. Each subsystem is designed to supply 100 percent of the cooling capacity required to assist in mitigating a postulated DBA. The containment analysis identifies the most limiting single failure to be the loss of one emergency bus. This failure reduces QS flow by about 50 percent.

11.2 SUPPLEMENTARY ANALYSES

This section identifies and discusses any supplementary safety analysis performed for the QS System, in addition to those analyses required by Section 11.1. Each analysis is described briefly below and is summarized in Table 11.2-1.

The RWST provides borated makeup water to the CH System during a fire safe shutdown. The QS System (without chemical addition) can be used to cool the containment, after a fire inside containment, if it is necessary to enter the containment before the temperature inside has decreased to a suitable level for access. The QS pumps shall be used as needed, at rated flow for 10- to 15-minute periods. The time limit is intended to allow the operator to evaluate whether further spraying via the use of the QS System is necessary. The RWST performance requirements are enveloped by the Condition IV events and parameters discussed in Section 11.1.4. The Appendix R fire safe shutdown analysis is documented in the Appendix R Report. [25.5.5]

Table 11.1-5. Plant Safety Analysis Parameters, QS System, Condition IV Events - Limiting Faults, Event: LOCA, MSLB

Parameter		Reason	
1. RWST Volume The minimum useable RWST water volume shall be 435,361 gal.	Why	The usable volume of water in the RWST is required to ensure that the containment pressure does not exceed the design value and that the containment is depressurized within 1 hour. [25.4.5] [25.4.7], [25.4.63]	
	Origin	The volume of water required is estimated and then used in the performance of the safety analyses: [25.4.63] [25.4.2] [25.4.5] [25.4.7] therefore, the origin of this parameter value is the safety analyses.	
	Impact	The parameter value, as stated, is necessary to meet the containment design basis objectives regarding peak pressure of 45 psig, depressurization within 1 hour, and maintenance of subatmospheric conditions. [25.1.47] [25.4.27]	

Table 11.1-5. Plant Safety Analysis Parameters, QS System, Condition IV Events - Limiting Faults, Event: LOCA, MSLB

Parameter		Reason
2. RWST Temperature The maximum/minimum RWST water temperature shall be 50°F/40°F.	Why	The limits on the RWST water temperature ensure containment integrity in the event of both pressurization due to a DBA [25.4.5] [25.4.7] or — [25.4.63] depressurization through inadvertent operation of the QS System. [25.4.13]
	Origin	The tank fluid temperatures are estimated and then used in the performance of the safety analyses; the range of fluid temperature is based on the limits which will ensure containment integrity. [25.4.5] [25.4.7]
	Impact	The maximum value, as stated, is necessary to meet the containment design basis objectives regarding peak pressure of 45 psig, [25.4.27] depressurization within 1 hour, and maintenance of subatmospheric conditions. [25.1.47] The minimum value is necessary to prevent lifting of the containment mat liner. [25.4.13]

Table 11.1-5. Plant Safety Analysis Parameters, QS System, Condition IV Events - Limiting Faults, Event: LOCA, MSLB

Parameter			Reason
3.	QS Effective Time The maximum time for the QS to become effective following CDA signal after a LOCA or MSLB shall be 66 seconds.	<u>Why</u>	The QS System effective time is required to be no greater than the stated value to ensure that the containment pressure does not exceed the design value and that the containment is depressurized within the required time. [25.4.3] [25.4.5] [25.4.7] [25.4.63]
		<u>Origin</u>	Given the occurrence of a loss of offsite power, QS is realized on the first diesel generator load sequence step. The QS effective time is based on the time to first diesel generator load sequence step plus system fill time. [25.4.3] [25.4.8]
		<u>Impact</u>	The start time for the QS System has a significant impact on the containment safety analyses, specifically on the second peak pressure: There is a lesser impact on depressurization time and subatmospheric peak pressure. [25.4.3] [25.4.5] [25.4.7] [25.4.63]

Table 11.1-5. Plant Safety Analysis Parameters, QS System, Condition IV Events - Limiting Faults, Event: LOCA, MSLB

Parameter		Reason	
4.	QS Heat Removal Effectiveness QS heat removal effectiveness for the containment safety analyses shall be 0.9. The QS heat removal effectiveness for the LHSI and RS pumps NPSH analyses shall be 1.0.	Why	A heat removal effectiveness of 0.9 is considered to be conservatively low for heat removal and containment depressurization. An effectiveness of 1.0 is conservatively high for RS and LHSI pump NPSH analyses. Each is conservative for the particular analysis performed. [25.4.5] [25.4.7] [25.4.63]
		Origin	The heat removal effectiveness is an assumed conservative value based on the arrangement of the spray nozzles, droplet size, total spray coverage, and droplet fall height. [24.2.7]
		Impact	Spray heat removal effectiveness has a direct impact on heat removal. Changes in the values used could impact the calculated containment peak pressure, depressurization time, invalidating the Plant Safety Analyses. [25.4.3] [25.4.5] [25.4.7] [25.4.63]

Table 11.1-5. Plant Safety Analysis Parameters, QS System, Condition IV Events - Limiting Faults, Event: LOCA, MSLB

	Parameter		Reason
6.	QS Flow Rate The minimum acceptable QS flow rate shall be the system flow curve calculated in [25.4.6] based on a design point flow rate of 2000 gpm. A constant bleed flow of 150 gpm shall be subtracted from the QS flow. The bleed flow shall be input to each IRS pump suction point.	Why	The containment safety analyses are performed based on these flow rates. The ability to depressurize the containment, in conjunction with the RS System, within the required time establishes the requirement for the parameters. [25.4.3] [25.4.63]
		Origin	The rate at which the containment is cooled and depressurized is dependent upon the rate at which spray is injected into the containment following a DBA. [25.4.63] [25.4.5] [25.4.6] [25.4.7] [25.5.1] The QS flow rate is a function of the containment pressure and the elevation of the water in the RWST. [25.4.6] The QS System flow curve originates from calculation [25.4.6] and is verified by the performance of the safety analyses.
		Impact	A change in the QS flow rate from that used in the safety analyses will impact the results of the containment safety analysis as to second peak pressure, depressurization time, and subatmospheric pressure. [25.4.3] [25.4.5] [25.4.7] [25.4.63]

Table 11.1-5. Plant Safety Analysis Parameters, QS System, Condition IV Events - Limiting Faults, Event: LOCA, MSLB

Parameter		Reason	
11.	Containment "High-High" Signal Setpoint Pressure Containment "high-high" pressure signal setpoint shall be 27.75 psia.	Why	This pressure setpoint is to be used in the plant safety analyses to activate all QS System MOVs and to start the delay timer associated with NaOH injection from the CAT. [25.4.5] & [25.4.63]
		Origin	A pressure setpoint of 24.7 psia is used in the safety analyses. However (as documented in the analyses), it has been determined that a CDA initiating pressure of 30 psia is acceptable. [25.4.5] [25.4.7] [25.4.63]
		Impact	A change in the CDA setpoint may impact the safety analyses depending on the magnitude of the change. Consequently, any change in the setpoint would have to be evaluated for impact on the safety analyses. Setpoint changes should also address the need to avoid spurious alarms. [25.4.3] [25.4.5] [25.4.7] [25.7.10] [25.4.63]

E. Design Requirement - The minimum/maximum design value for the parameter is entered as the "design requirement." This value is normally based directly upon the assumptions, input, or results of the safety analysis or is derived therefrom. Where the value of a design requirement is dependent upon the value of another parameter, the dependency is explained in the "Remarks" column and the dependent parameter listed as a parameter.

F. Design Specification - The value established by the designer for the parameter in the design specification (NAS, NUS, etc.) is recorded.

G. Design Margin - The design margin available for this parameter is presented. Design margin is the difference between the "design requirement" and the "design specification."

H. Installed Value - The actual value for the parameter, specific to the installed component, is recorded. This value must be based on a substantiated source that conclusively demonstrates the stated installed value.

The difference between the "installed value" and the "design specification" represents the performance margin.

I. Total Margin - Total margin is the difference between "design requirement" and "installed value." The total margin is dependent upon the actual installed component and is subject to change if the component is replaced or modified.

J. Reference - References to the applicable sections of this SDBD that contain the reason or basis are provided. Reasons and references for margins may be presented here.

K. Remarks - This column is used to record any pertinent remarks not included elsewhere.

Subsequent sections of this chapter provide discussion of margins presented in Table 12.0-1. A subsection is presented for each major component identified in Table 12.0-1. Key parameters for the component are listed as separate items in the sections.

12.1 RWST VOLUME

1. Design Requirement During Modes 1, 2, 3, and 4, the RWST shall contain a usable volume of 435,361 gal. of water. [25.4.3] [25.4.5] [25.4.7] [25.4.63]

   Design Specification - During Modes 1, 2, 3, 4, the volume of water in the RWST is controlled by the Technical Specifications between 466,200 gal and 487,000 gal. [25.6.9] [25.6.10] The volume of water required in the RWST to ensure a usable volume of 435,361 gal is 448,091 gal. [25.4.47]

Margin Result - The design requirement is met with a margin of 18,109 gal based on the minimum Technical Specification volume of 466,200 gal.

2. Design Requirement - During Modes 5 and 6, the RWST shall contain a usable volume of 3351 gal of water with a boration level of 2300 ppm or 3065 gal of water with a boration level of 2400 ppm. [25.4.47]

Design Specification - During Modes 5 and 6, the volume of water in the RWST is controlled by Technical Specification and maintained at 5,000 gal with a boration level between 2300 ppm and 2400 ppm. [25.6.9] [25.6.10] The volume of water required in the RWST to ensure a usable volume of 3351 gal at 2300 ppm of boron (worst case) is 26,264 gal. [25.4.47]

Margin Result - The design requirement is met with a margin of 24,736 gal based on the minimum Technical Specification volume of 51,000 gal.

12.2 RWST FLUID TEMPERATURE

Design Requirement - The design requirement for the temperature of the water in the RWST shall be a minimum of 40°F and a maximum of 50°F. [25.4.3] [25.4.5] [25.4.7] & [25.4.13] [25.4.63]

Design Specification - The temperature of the water in the RWST is kept between 40°F and 43°F by the RWST cooling subsystem. [25.3.32]

Margin Result - The design requirement is met with a margin of 7°F.

12.3 QS PUMP FLOW RATE

Design Requirement - Each QS pump shall be capable of delivering the flow (QS flow as well as bleed flow of 150 gpm to the IRS pumps) used in the containment analysis, with a minimum available NPSH at the QS pump suction of 25.4 ft. [25.4.7] [25.4.15] & [25.4.63]

The containment analyses, calculations [25.4.3], [25.4.5], and [25.4.7], assume that QS is effective 66 seconds after a CDA signal, which is activated at a containment high-high pressure setpoint of 27.75 psia. The QS flow is based on the system flow curve, calculation [25.4.6], which shows a flow rate dependent on the containment pressure and the height of water in the RWST. The minimum heat removal effectiveness of the QS spray, which is dependent on droplet fall height, nozzle size, and pressure drop across the nozzle, is assumed to be 0.9. [24.12.1] Based on the location of the QS System spray headers, the size and orientation of the spray nozzles, and QS flow, the dose analyses use a QS volume coverage of 721,000 cubic ft. [25.4.10] [25.4.11] [25.4.12] [25.4.24] [24.2.6]

Table 12.0-1. Key Component Parameters, QS System

(A) Item No.	(B) System Parameter	(C) Operating Condition	(D) System/ Component	(E) Design Requirement	(F) Design Specification	(G) Design Margin	(H) Installed Value	(I) Total Margin	(J) Reference	(K) Remarks
1	Volume (gal)	Condition IV LOCA MSLB FWLB REA	RWST (1-QS-TK-1) (2-QS-TK-1)	448,091 [25.4.47]	Min 466,200 [25.6.9]; [25.6.10]	18,109 [25.4.47]	Not available (NA)	Not available (NA)	Section 2.2.1; Table 11.1-5, Item 1	Includes volume for QS and SI Systems and for IRS pump suction feed.
2	Temperature (°F)	Condition IV LOCA MSLB FWLB REA	RWST (1-QS-TK-1) (2-QS-TK-1)	Min: 40°F [25.4.13] Max: 50°F [25.4.3] [25.4.5] [25.4.7] [25.4.63]	40 to 43 [25.3.32]	7	NA	NA	Section 2.2.1; Table 11.1-5, Item 2	Minimum temperatures based on maximum allowable depressurization of containment.
3	Flow Rate (gpm)	Condition IV LOCA MSLB FWLB REA	QS Pumps (1-QS-P-1A) (1-QS-P-1B) (2-QS-P-1A) (2-QS-P-1B)	2000 [25.4.3] [25.4.5] [25.4.7] [25.4.63]	2000 [25.4.6] [25.4.8] [25.6.1]	0	NA	NA	Section 2.2.1; Table 11.1-5, Item 6	2000 gpm is a designated design flow rate. QS flow follows a system curve which is delivered within 66 sec of a CDA. 150 gpm for IRS pumps is included in flow.
4	Volume (gal)	Condition IV LOCA MSLB FWLB REA	CAT (1-QS-TK-2) (2-QS-TK-2)	4800 [25.4.1]	4800 [25.6.9] [25.6.10]	0	NA	NA	Section 2.2.1; Table 11.1-5, Item 8	Basis is spray and sump water pH requirements. [24.12.3]

DC 90-13-1, Appendix 4-9, Page 77

resulting from any loss-of-coolant-accident. This margin shall reflect consideration of (1) the effects of potential energy sources that have not been included in the determination of the peak conditions, such as energy in steam generators and as required by § 50.44, energy from metal-water and other chemical reactions that may result from degradation but not total failure of emergency core cooling functioning; (2) the limited experience and experimental data available for defining accident phenomena and containment responses; and (3) the conservation of the calculation model and input parameters." [25.1.30]

Type - Licensing basis commitment. [25.10.8]

Design Basis Commitment - The entire GDC 50. [25.10.8]

Design Basis Feature - The QS System is designed so that the containment structure can accommodate without exceeding the design leakage rate, and with sufficient margin, the calculated pressure and temperature conditions resulting from a LOCA. [25.4.3] [25.4.5] [25.4.27] [25.8.1] [25.8.2]
[25.4.63]

16. GDC 54, "Piping Systems Penetrating Containment," May 21, 1971 [25.1.22]

Requirement - "Piping systems penetrating primary reactor containment shall be provided with leak detection, isolation, and containment capabilities having redundancy, reliability, and performance capabilities that reflect the importance to safety of isolating these piping systems. Such piping systems shall be designed with a capability to test periodically the operability of the isolation valves and associated apparatus and to determine if valve leakage is within acceptable limits." [25.1.22]

Type - Licensing basis commitment. [25.10.8]

Design Basis Commitment - The entire GDC 54. [25.10.8]

Design Basis Feature - The containment isolation system provides, during accident conditions, at least two barriers between the atmosphere outside the containment structure and either the fluid inside the reactor coolant pressure boundary or the atmosphere inside the containment structure. The operation of the containment isolation system is automatic, and failure of one valve or barrier does not prevent isolation. Means are

Design Basis Commitment - The QS System design uses the acceptance criteria provided in SRP 6.2.1.1.A related to subatmospheric containments. [24.13.4]

Design Basis Feature - The QS System (in conjunction with the RS System) reduces and maintains containment pressure to subatmospheric within 1 hour of the accident. [25.4.3] [25.4.7] [25.4.63]

3. SRP 6.2.2, "Containment Heat Removal System," 1975 [25.1.48]

Requirement - The acceptance criteria detailed in this document amplify the requirements of GDC 38, 39, 40, and 50 of 10CFR50, Appendix A, and provide the basis of the containment heat removal system design. [25.1.48]

Type - Self-imposed requirement. [24.13.5]

Design Basis Commitment - The entire SRP 6.2.2 with the exception of acceptance criteria 11(6) related to sump design and compliance with the analyses requirements of RG 1.82. [24.13.5]

[25.4.63]

Design Basis Feature - The QS System design is based on the guidance of SRP 6.2.2. [25.4.5] [25.4.7] Acceptability of the containment sump design was based on actual hydraulic model studies/tests done using a model of the NAPS Units 1 and 2 reactor containment sumps. [25.12.6]

4. SRP 6.5.2, "Containment Spray as a Fission Product Cleanup System" [25.1.9] [25.1.60]

Requirement - The spray system shall be designed such that the spray solution maintains the highest possible pH, within materials compatibility constraints. This requirement is satisfied by a spray pH in the range of 8.5 to 11.0. [25.1.60] [25.1.9]

Type - Self-imposed requirement. [24.13.6]

Design Basis Commitment - The portion of the SRP related to injection spray pH. [24.13.6]

Design Basis Feature - The QS System design uses the acceptance criteria listed in SRP 6.5.2 relating to pH of QS water. The calculated minimum QS pH is 8.8. [25.4.1] [25.4.33]

Design Feature - Upon receiving a CDA signal, the QS System sprays water from the RWST into the containment atmosphere. [25.4.5] [25.4.65]

b. Requirement - After a DBA, the QS System shall provide the capability to remove iodine from the containment atmosphere to minimize public exposure levels due to leakage of radioactive gases from the containment. [25.1.9] [25.1.60]

Design Feature - The QS System injects NaOH solution into the spray water to enhance removal of iodine from the containment atmosphere and to minimize chloride stress-corrosion inside containment. The spray water ultimately ends up in the sump water. [25.4.1] [25.4.2] [25.4.33]

c. Requirement - The QS System shall provide the means to control the pH level of water in the containment sump to minimize chloride stress - corrosion cracking inside containment and to ensure retention of radioactive iodine by sump water. [25.1.8] [25.1.60]

Design Feature - The QS System injects NaOH solution into the spray water which ultimately ends up in the sump water. [25.4.1] [25.4.33] [25.7.1] [25.7.2] [25.7.3]

d. Requirement - The QS System shall provide a source of cool water to the RS System in support of the IRS pump net positive suction head (NPSH) requirements. [25.3.29] [24.2.4]

Design Feature - Upon receiving a CDA signal, the QS System provides bleed flow to the IRS pumps suction point to increase the available NPSH. [25.4.5] [25.4.7] [25.4.65]

e. Requirement - The QS System shall provide containment isolation (relative to containment atmosphere leakage via the QS System containment isolation valves) to reduce the leakage of containment atmosphere so that the site boundary dose from any outleakage prior to the containment becoming subatmospheric is within the limits of 10CFR100. [25.1.22] [25.1.29] [25.10.8]

Design Feature - QS System piping which connects directly to the containment atmosphere and penetrates the containment is provided with redundant containment isolation valves. [25.8.1] [25.8.2]

f. Requirement - The QS System shall provide a source of cool, borated water to the SI System in support of initial emergency core cooling [25.3.2?] [24.2.1]

Design Feature - Upon receiving an SI signal, the QS System supplies water from the RWST to the SI System for injection into the core to remove heat. [25.4.5] [25.4.7] [25.4.63]

g. Requirement - The QS System shall support minimum recirculation flow for the LHSI pumps during postaccident safety injection mode. [25.3.28] [24.2.1]

Design Feature - A flow path is provided between the LHSI pump discharge points and the RWST. [25.8.1] [25.8.2] [25.8.3]

h. Requirement - The QS System shall initiate the realignment of the LHSI pump suction from the RWST to the containment sump when the RWST fluid contents are close to depletion. [25.3.28] [24.2.1]

Design Feature - An RWST low-level alarm signal initiates operator action to realign the LHSI pump suction from the RWST to the containment sump. If realignment is not performed by operator action at the low-level alarm signal, automatic switchover occurs at the low-low level signal. [25.3.28] [24.2.1]

i. Requirement - The QS System RWST shall provide a backup source of borated water to the CH System in support of ensuring the availability of when the tank level drops below its low-level set point. [25.3.29] [24.2.4]

- The SI System to support the hydrostatic test of the RC System. [25.3.28] [24.2.1]
- The SI System to support the LHSI pump testing. [25.3.28] [24.2.1]
- The SI System to support accumulator check valve leak testing. [25.3.28] [24.2.1]

Design Feature - A connection is provided from the RWST to the charging pumps and SI pumps. [25.8.1] [25.8.2] CCT tank makeup is provided using portable equipment. [25.3.29]

14.1.2 Performance

The performance criteria for the system are identified in Section 2.2. The operational performance of the system under normal and accident conditions is addressed in Chapter 3. Chapter 6 identifies major system performance parameters and their bases. Chapter 11 addresses the operation of the system under accident conditions.

1. Safety-Related Performance Criteria

   a. Requirement - The QS System, in conjunction with the RS System, shall meet the requirements of the containment safety analysis to depressurize the containment within 1 hr following a DBA. [25.4.5] [25.4.7] [25.4.63]

      Design Feature - The QS System delivers 435,361 gal of 40-50°F water from the RWST to the containment by providing: [25.4.3] [25.4.5] [25.4.7] [25.4.63]

      - 150 gpm per QS pump to the suction of the IRS pumps. [25.4.3] [25.4.5] [25.5.1] [25.4.63]
      - A minimum 4610 gpm to the SI System for injection into the reactor core until SI switchover to the containment sump. [25.4.3] [25.4.5] [25.4.33] [25.4.63]

- An average 2000 gpm per QS pump as atomized spray to the containment atmosphere and bleed to the RS pumps. [25.4.3] [25.4.5] [25.4.6] [25.4.15] [25.4.63]

  The actual flow rate at any time is based on the system flow curve and depends on the containment pressure and the height of water in the RWST. However, the QS pumps are designed to deliver 2000 gpm at a total developed head of 265.5 ft. [25.4.6] [25.4.14] [25.4.15]

- The QS System is initiated by a CDA signal which is activated by a containment pressure setpoint of 27.75 psia [25.4.5] [25.4.7], [25.4.63] or by manual initiation.

- The QS becomes effective within 66 seconds after a CDA signal. [25.4.3] [25.4.8]

- The heat removal effectiveness of the QS is a minimum 0.9 with a containment spray volume coverage of 721,000 cubic ft. [25.4.5] [25.4.7] [25.4.24] [25.4.63]

b. Requirement - The QS System shall be used to remove iodine from the containment atmosphere following a LOCA in order to minimize public exposure levels due to leakage of radioactive gases from the containment. [25.1.9] [25.4.1] [25.10.7]

   Design Feature - The QS System injects NaOH solution from the CAT into the spray water to produce a spray with a minimum pH of 8.8. [25.4.1] The addition of the caustic NaOH into the spray water enhances the removal of radioactive iodine from the containment atmosphere. A pH level between 8.5 and 11.0 is optimal for iodine removal. [25.1.9]

   The NaOH from the CAT is automatically introduced into the spray water 5 minutes after a CDA signal. [25.8.10] [25.8.11] The spray coverage is 721,000 cubic feet. [25.4.24]

c. Requirement - The QS System shall be used to control the final pH level of the sump water in the containment to minimize chloride stress-corrosion inside the containment. [25.1.8] [25.4.1]

Design Feature - The QS System injects NaOH solution into the spray water, which ultimately ends up in the sump water. The final containment sump minimum pH value is approximately 7.7. [25.4.1] Maintaining the pH of a borated solution at greater than 7.0 helps inhibit initiation of stress corrosion cracking of austenitic stainless steel components, and limits the revolution of iodine from the sump water. [25.1.8] [25.1.60]

d. Requirement - The QS System shall provide cool water to the suction of the IRS pumps to increase the available NPSH. [25.3.29] [25.5.1]

Design Feature - A flow path is provided to bleed 150 gpm per QS pump from the QS flow to the spray headers for input to the IRS pump suction. [25.4.3]]25.8.1] [25.8.2]

e. Requirement - The QS System shall provide containment isolation capability to keep leakage of containment atmosphere via the QS System containment isolation valves within the acceptable limits defined in 10CFR50, Appendix J. [25.1.22] [25.1.52] [25.5.3] [25.5.4]

Design Feature - The QS pump discharge lines are each provided with one normally closed MOV (located outside containment) and one weight-loaded check valve (located inside containment) to provide containment isolation functions. [25.8.1] [25.8.2] During the initial phase of a DBA, to support system function, the QS System isolation valves are in an open position. On completion of the QS System function of injecting spray into the containment, the QS pump discharge MOVs are manually closed to support containment isolation. [25.8.1] [25.8.2] [25.8.10] [25.8.11]

Limiting containment atmosphere leakage due to the isolation valve leakage is ensured by the testing/surveillance requirements imposed on containment isolation valves by the NAPS Technical Specifications. [25.6.9] [25.6.10]

f. Requirement - The QS System shall provide cool borated water to the SI System to be injected into the reactor core to remove heat. [25.3.28] [25.4.3] [25.4.5] [25.4.7] [24.2.1]

[25.4.43]

Design Feature - Connections are provided from the RWST to the LHSI and HHSI (CH System charging pumps) pumps. Upon receipt of an SI signal, water from the RWST at 40 to 50°F is injected into the core at a minimum 4610 gpm. [25.4.3] [25.4.33] [25.8.1] [25.8.2] [25.8.5]

The RWST low-level alarm provides a signal for manual switchover of the SI System from the RWST to the containment sump. If manual switchover is not performed on the low-level signal, automatic switchover occurs on an RWST low-low-level alarm signal. [25.3.28] [25.4.3] [25.4.5] [25.4.7] [25.4.63] [24.2.1]

2. NSQ Performance Criteria

a. Requirement - The QS System shall be used to cool, without admitting NaOH from the CAT, the containment after a fire inside the containment, if it is necessary to enter the containment before the temperature inside has decreased to a suitable level for access. [25.1.23] [25.5.5]

Design Feature - The QS pumps can be cycled for 10- to 15-minute periods, as needed. The time limit is to allow the operator to evaluate whether further spraying is required. [25.5.5]

b. Requirement - The QS System shall provide information on RWST water level and QS flow in support of RG 1.97 requirements. [25.1.11] [25.5.6]

Design Feature - RWST level instrumentation and QS flow instrumentation are provided to monitor QS System RG 1.97 variables. [25.5.6] [25.8.1] [25.8.2] [24.2.12]

c. Requirement - The QS System shall support safe shutdown in the event of an ATWS or station blackout by providing a source of borated water to the CH System. [25.3.30] [25.5.9] [25.5.10]

Design Feature - Connections are provided between the RWST and the CH System. [25.8.1] [25.8.2] The performance criteria imposed on the RWST fluid by the CH System for these events are enveloped by the criteria imposed by the containment analyses. [25.3.30] [25.5.9] [25.5.10] [24.2.2] [24.2.3]

3. Non-Safety Performance Criteria

a. Requirement - During modes 1, 2, 3, and 4, the temperature of the RWST fluid shall be maintained between 40°F and 50°F. [25.4.3] [25.4.5] [25.4.7] [25.4.13] [25.4.63]

Design Feature - The refueling water cooling subsystem automatically functions to maintain the temperature of the RWST fluid between 40°F and 43°F. The recirculation pumps and coolers initially cool the RWST to 43°F. The recirculation pumps and refrigeration units automatically maintain the temperature between 40°F and 43°F. [25.4.26] [25.8.1] [25.8.2]

b. Requirement - During normal operations (on an as needed basis), the QS System shall provide sufficient water to meet the requirements of the following:

- The CH system in response to a low level in the volume control tank. [25.3.30] [24.2.2]
- The SI System to fill the refueling cavity for refueling by means of the SI pumps. [25.3.28] [24.2.1]
- The SI System to fill the accumulators or to raise the level of the water in the accumulators. [25.3.28] [24.2.1]
- The RS System casing cooling tank for original fill and for subsequent makeup when the tank level drops below its low-level setpoint. [25.3.29] [24.2.4]
- The SI System to support the hydrostatic test of the RC System. [25.3.28] [24.2.1]
- The SI System to support the LHSI pump testing. [25.3.28] [24.2.1]
- The SI System to support accumulator check valve leak testing. [25.3.28] [24.2.1]

Design Feature - Piping connections from the RWST to the CH and SI Systems are provided. [25.8.1] [25.8.2] [25.8.4] [25.8.5]. The RWST water volume is sufficient to meet the needs of the CH, RS, and SI Systems. [25.3.28] [25.3.30] [24.2.2] [24.2.1] [25.4.22]

14.1.6 Environmental Conditions

Requirement - The QS System shall be capable of withstanding the pressure, temperature, humidity, corrosiveness, radiation, and electromagnetic levels of the environments to which it is exposed. The system shall be capable of operating at the design capacity in the above environments for the duration of time it is required to operate. [25.1.14] [25.1.26] [25.1.32] [25.1.33] [25.3.17] [25.10.8]

Design Feature - QS System components, except for the QS pump discharge MOVs, are required to operate for only 1 hour following an accident and are located in mild environments, except for stainless steel piping. [25.3.17] [25.4.5] [25.4.7] [25.8.1] [25.8.2]
[25.4.63]

QS System electrical and I&C components located in mild environments are designed for the expected environments. In those areas where ventilation failure may cause the ambient temperature to exceed the temperature for which the equipment is qualified, the Ambient Temperature Monitoring (AM) System provides an alarm in the Control Room and the impact of the temperature is evaluated. [25.10.11] [24.14.3]

The QS pump discharge MOVs, which are also containment isolation valves, are located in environmental zone SFGD-1, which is considered a harsh environment. The QS pump discharge MOVs, as identified in the Equipment Qualification Master List (EQML), are qualified to their environment. [25.3.17] [25.3.38] [25.6.3] [25.6.10]

QS System components are located in areas of the following buildings:

1. Containment (piping)
2. Safeguards Building
3. Quench Spray Pump Area
4. Auxiliary Building (electrical)
5. Cable Tunnel (electrical)
6. Switchgear Room (electrical)
7. Control Room (I & C)

14.1.7 Interface Requirements

This section addresses the interface requirements between the QS System and other plant systems. The fundamental system interface requirements are identified in Chapter 5. This section provides additional design basis information for the functional and physical interfaces.

calculations [25.4.39] [25.4.40] qualify the QS System piping for the loadings specified above.

d. Loads for evaluation of the equipment nozzles and pipe penetrations associated with the QS System piping are generated in the pipe stress analysis calculations. [25.4.36] [25.4.37] [25.4.38] The evaluation of the following nozzle loads are documented in mechanical analysis calculations. [25.4.41]

Calculation No./Component Identification No.

11715-NM(B)-318	/1-QS-TK 1
	/1-QS-TK 2
	/1-QS-P 1A, B
	/1-QS-P 2A

Reaction loads on those components are further discussed in Sections 14.2 through 14.11.

Reaction loads on other components (e.g., valves) are considered acceptable when the adjoining piping has been shown to meet the code requirements.

2. Requirement - The QS System and its components shall be sized to account for any anticipated corrosion or erosion. [25.11.1]

   Design Feature - Following an accident, the QS System is required to be operable for approximately 1 hour, consequently limiting the probability of erosion. [25.4.5] [25.4.7] [25.4.63]

   During normal operation, the recirculation portion of the QS System operates to maintain the temperature of the RWST water. However, the system is a low-energy/velocity system and is not anticipated to cause erosion. [24.14.9] [24.14.124]

3. Requirement - Thermal insulation shall be provided to improve system thermal performance.

   Design Feature - The RWST is provided with 2 inches of polyurethane to limit the temperature rise of the RWST fluid to 0.5°F per day when the cooling subsystem is not operating. [25.4.26] [25.8.7]

14.1.11 Hydraulic Requirements

This subsection addresses the hydraulic design requirements for the QS System.

Requirement - The QS System piping, fittings, valves, sizing, and layout shall address system fluid flow requirements (based on the fluid pressure, temperature, and system fluid transients) and fluid velocity requirements to limit pressure drops and erosion.

To ensure proper operation of the system, the NPSH required by the system pumps shall be less than the NPSH available at the pumps. [25.1.34] [25.10.9]

Design Feature - The NPSH requirements of QS System pumps are addressed in the respective components' section of Sections 14.2 to 14.11.

Piping sizing and layout is based on the flow requirements of the QS System.

The QS System is required to deliver an average 2000 gpm per QS pump. The actual flow per pump as used in the containment safety analyses is based on the system flow curve. The flow rate of water to the containment spray rings during a DBA is dependent upon the difference between the containment atmospheric pressure and the RWST fluid static pressure, i.e., the height of water in the RWST. [25.4.3] [25.4.5] [25.4.6] [25.4.7] [25.4.15] [25.4.63]

The CAT feeds NaOH solution into the spray water by balanced gravity feed. As the CAT and RWST empty, the levels in both tanks remain in hydrostatic equilibrium until the CAT is empty and the QS water volume from the RWST is depleted. [25.4.2]

The sizing of piping results in an appropriate fluid flow velocity, and pressure drops within the QS System piping are low enough for system operation. [25.4.16] [25.4.17] [25.4.18] [25.4.19] [24.14.13]

14.1.12 Chemistry and Sampling Requirements

This subsection identifies chemistry or chemical requirements for the QS System that are part of the design basis, including design requirements that facilitate sampling or other chemistry requirements.

Requirement - The QS System shall maintain the required boron concentration in the RWST in support of the CH System and SI System requirements. [25.3.28] [25.3.30] [25.5.11] [24.2.2] [24.2.1]

3. For piping systems penetrating containment, containment isolation capability shall be maintained. [25.1.22] [25.1.28] [25.10.8]

4. In the event of loss of offsite power, the safety-related portion of the QS System shall be powered by redundant emergency power sources. [25.1.2] [25.10.8]

Design Feature - A QS System is provided for each reactor unit, and each QS System is independent of the other. There are no cross-connects between the units and no dependence of one unit on the opposite unit's QS System. The Unit 1 or Unit 2 RWST can be used to supply borated water to the SI hydrostatic test pump suction via a locked closed cross-connection valve in the SI System. Operator action is necessary to unlock the valve to perform this function. Cross-flow between the two RWSTs through the SI System is not possible due to in-line check valves. Administrative controls prevent cross-flow between the two RWSTs through the RP pumps. [25.8.1] [25.8.2]

The QS System contains sufficient component redundancy so that under accident conditions, the minimum performance requirements (Section 2.2) can be achieved, assuming a single failure. The QS System is designed to meet the single failure criterion as defined in 10CFR50 Appendix A. [25.1.1] [25.4.4] [25.4.5] [25.4.6]

A weight-loaded check valve and a remotely-operated MOV provide the redundant barrier in each QS flow path penetrating the containment structure. [25.8.1] [25.8.2]

The QS System includes redundant power sources so that each separate train is powered independently. [25.8.8] [25.8.9] In the event of a loss of offsite power, each motor-driven pump is powered from separate emergency buses. [25.8.8] [25.8.9]

14.1.19 Reliability

This section identifies the design bases that specifically affect the reliability of the QS System or system components, including the definition of required design features.

Requirement - Failure modes and effects shall be identified and analyzed for the QS System to ensure its availability following a DBA. [25.1.1] [25.10.8]

Design Feature - The QS System reliability is ensured by providing two separate and parallel trains. Each train is capable of performing the intended function of the QS System. [25.4.4] [25.4.5] [25.8.1] [25.8.2]
[25.4.6]

The QS System can perform its intended function in the event of a single failure. [25.4.4] [25.4.5] [25.4.5]

14.1.20 Test and Surveillance Requirements

This subsection identifies the testing and surveillance requirements applicable to the QS System, including the specific design features incorporated into the design to facilitate system or component testing and surveillance. The design bases for the requirements are identified.

Requirement - The QS System shall be designed to permit appropriate periodic inspection and testing of electrical, mechanical, and structural features associated with the system to assess and ensure the reliability of the system and its capability to continue to support its minimum performance requirements. [25.1.3] [25.1.4] [25.1.12] [25.1.14] [25.10.8]

Design Feature - The QS System design includes test connections, piping, and instrumentation, which will permit appropriate periodic testing of electrical and mechanical components to assess system reliability and capability of supporting its minimum performance requirements. [25.8.1] [25.8.2]

The QS System includes an RWST recirculation flow path with test nozzles in the RWST to allow testing of the QS pumps and to determine if the RWST fluid contains particulate material that might clog the spray header nozzles. [25.8.1] [25.8.2]

The chemical addition subsystem includes a recirculation loop with a CAT pump that has sampling connections that permit sampling and testing of the CAT contents. [25.8.1] [25.8.2]

The QS System layout permits accessibility and allows for periodic inspection of electrical, mechanical, and structural features to assess system reliability. [25.8.6] [25.8.21] [25.8.34]

The QS System design provides leakage monitoring connections for testing the containment isolation valves. [25.8.1] [25.8.2]

The QS System testing/surveillance requirements are addressed in Chapters 19 and 21.

14.1.21 Maintenance and Repair Requirements

This subsection identifies the maintenance and repair requirements that are part of the design bases of the QS System, including the design features, and their bases, required to facilitate maintenance.

2. Requirement - The QS System shall maintain containment isolation capability. [25.1.22] [25.1.26] [25.10.8]

   Design Feature - The QS pump discharge lines are each provided with one normally closed MOV (located outside containment) and one weight-loaded check valve (located inside containment) to provide containment isolation function. Means are provided to close the valves manually after completion of the QS System function of injection of spray into the containment. [25.8.1] [25.8.2]

   The QS pump discharge MOVs are included in the plant ISI Program. Limiting containment atmosphere leakage due to isolation valve leakage is ensured by the testing/surveillance requirements imposed by the Technical Specifications. [25.5.7] [25.5.8] [25.6.9] [25.6.10]

3. Requirement - Radioactive releases to the environment due to operation of the QS System shall be monitored. [25.1.? [25.10.5]

   Design Feature - There are no radiation monitors included in the QS System design. QS System fluid is injected directly into the containment following a DBA. [25.4.5] [25.4.7] [25.8.1] [25.8.2] [25.4.63]

14.1.27 Materials, Processes, Parts, and Equipment

This subsection identifies any special system or component features that are required to satisfy the design basis, including design features, and their bases, required to address concerns associated with materials, processes, or parts for the QS System.

Requirement - Special materials, processes, parts, and equipment requirements necessary for the operations and maintenance of the QS System shall be specified. [25.11.1]

Design Feature - There are no special materials, processes, parts, or equipment requirements for the QS System.

14.1.28 Personnel Safety

This subsection identifies the personnel safety requirements that are included in the design basis, including design features, and their bases, required to address personnel safety requirements for the QS System. The regulations

Table 14.1-1. Process Condition Summary

SPRAY: Quench Spray	SYSTEM CODE: QS	SDBD NUMBER: SDBD-NAPS-QS	NODE NO.: 1

NODE TYPE: X MASS FLOW _INTERFACE _ COMPONENT ______

PARAMETER	DESIGN VALUE	OPERATING CONDITIONS NORMAL	EMERGENCY	SHUTDOWN	OTHER
PRESSURE (psig)	30(a)		25(a)		
TEMPERATURE (°F)	75(a)		40-50(d) (c)		
FLOW RATE (gpm)	2000(b) (c)		2000(b) Note 1		
NPSH REQ'D (ft)	14(c)		N/A		
PRESSURE DROP (psi)	1.61(b) Note 3		1.61(b) Note 2 Note 3		
HEAT LOAD (Btu/hr)	N/A		N/A		
FLUID TYPE	Water/NaOH mix		Water/NaOH mix		
VISCOSITY (centipoise)	1.5(b)		1.5(b)		
FLUID pH					

FLOW IS FROM: 1-QS-TK-1 T 1-QS-P-1A

NOTES:
1. Actual flow rate in safety analyses is based on system flow curve; however, pressure drop values are based on 2000 gpm.
2. System pressure drops are based on fluid temperature of 40°F (b).
3. Value given is for Unit 1, Unit 2 value is 1.57 psig (f).
4. Refer to Section 14.1.12.
5. Ref (D) is for unit 1 only

REFERENCES: (a): [25.3.5] [24.14.43]; (b): [25.4.18]; (c): [25.6.1]; (d): [25.4.5]; (e): [25.6.1]; (f): [25.4.18]
(D) [25.4.63] Notes

REVISION NO: 00 DATE: July 1, 1990

Table 14.1-1. Process Condition Summary

SYSTEM: Quench Spray	SYSTEM CODE: QS	SDBD NUMBER: NAPS-QS	NODE NO.: 2

NODE TYPE: X MASS FLOW _INTERFACE _ COMPONENT ____

PARAMETER	DESIGN VALUE	OPERATING CONDITIONS NORMAL	EMERGENCY	SHUTDOWN	OTHER
PRESSURE (psig)	30(a)		25(a)		
TEMPERATURE (°F)	75(a)		40-50(d) (F)		
FLOW RATE (gpm)	2000(b) (c)		2000(b) Note 1		
NPSH REQ'D (ft)	14(c)		N/A		
PRESSURE DROP (psi)	1.84(b) Note 3		1.84(b) Note 2 Note 3		
HEAT LOAD (Btu/hr)	N/A		N/A		
FLUID TYPE	Water/NaOH mix		Water/NaOH mix		
VISCOSITY (centipoise)	1.5(b)		1.5(b)		
FLUID pH					

FLOW IS FROM: 1-QS-TK-1 TO: 1-QS-P-1B

NOTES:
1. See Table 14.1-1, Node No. 1, Note 1.
2. See Table 14.1-1, Node No. 1, Note 2, and Reference (b).
3. Value given is for Unit 1; Unit 2 value is 1.36 psi (e).
4. Refer to Section 14.1.12
5. Reference (F) is for Unit 1 only

REFERENCES: (a): [25.3.5] [24.14.43]; (b): [25.4.17]; (c): [25.6.1]; (d): [25.4.5]; (e): [25.4.18] (F): [25.4.63] Note 5

REVISION NO: 00 DATE: July 1, 1990

Table 14.1-1. Process Condition Summary

SYSTEM: Quench Spray	SYSTEM CODE: QS	SDBD NUMBER: NAPS-QS	NODE NO.: 3

NODE TYPE: X MASS FLOW _ INTERFACE _ COMPONENT ____________

PARAMETER	DESIGN VALUE	OPERATING CONDITIONS			
		NORMAL	EMERGENCY	SHUTDOWN	OTHER
PRESSURE (psig)	160(a)		100-155(a)		
TEMPERATURE (°F)	75(a)		40-50(c) (F)		
FLOW RATE (gpm)	[illegible]000(b) (c)		2000 [illegible] Note 1		
NPSH REQ'D (ft)	N/A		N/A		
PRESSURE DROP (psi)	18.8(b) Note 3		18.3(b) Note 2 Note 3		
HEAT LOAD (Btu/hr)	N/A		N/A		
FLUID TYPE	Water NaOH mix		Water NaOH mix		
VISCOSITY (centipoise)	1.5(b)		1.5(b)		
FLUID pH					

FLOW IS FROM: 1-QS-P-1A TO: Spray Header

NOTES:
1. See Table 14.1-1, Node No. 1, Note 1.
2. See Table 14.1-1, Node No. 1, Note 2, and Reference (b).
3. Value given is for Unit 1; Unit 2 value is 31.3 psi (e).
4. Refer to Section 14.1.12.
5 Reference (F) is for unit 1 only

REFERENCES: (a): [25.3.5] [24.14.43]; (b): [25.4.16]; (c): [25.6.1]; (d): [25.4.5]; (e): [25.4.18]; (F): [25.4.63] Note 5

REVISION NO: 00	DATE: July 1, 1990

Table 14.1-1. Process Condition Summary

SYSTEM: Quench Spray | SYSTEM CODE: QS | SDBD NUMBER: NAPS-QS | NODE NO.: 4

NODE TYPE: X MASS FLOW _ INTERFACE _ COMPONENT ____

PARAMETER	DESIGN VALUE	OPERATING CONDITIONS NORMAL	EMERGENCY	SHUTDOWN	OTHER
PRESSURE (psig)	160(a)		100-155(a)		
TEMPERATURE (°F)	75(a)		40-50(d) (F)		
FLOW RATE (gpm)	2000(b) (c)		Note 1		
NPSH REQ'D (ft)	N/A		N/A		
PRESSURE DROP (psi)	31.5(b) Note 3		31.5(b) Note 2 Note 3		
HEAT LOAD (Btu/hr)	N/A		N/A		
FLUID TYPE	Water/NaOH mix		Water NaOH mix		
VISCOSITY (centipoise)	1.5(b)		1.5(b)		
FLUID pH					

FLOW IS FROM: 1-QS-P-1B TO: Spray Header

NOTES:
1. See Table 14.1-1, Node No. 1, Note 1.
2. See Table 14.1-1, Node No. 1, Note 2, and Reference (b).
3. Value given is for Unit 1; Unit 2 value is 34.2 psi (e).
4. Refer to Section 14.1.12.
5. Reference (F) is for Unit 1 only

REFERENCES: (a): [24.3.5] [24.14.43]; (b): [25.4.17]; (c): [25.6.1]; (d): [25.4.5]; (e): [25.4.19] (F): [25.4.63] Note 5

REVISION NO: 00 | DATE: July 1, 1990

Table 14.1-1. Process Condition Summary

SYSTEM: Quench Spray	SYSTEM CODE: QS	SDBD NUMBER: NAPS-QS	NODE NO.: 5

NODE TYPE: X MASS FLOW _ INTERFACE _ COMPONENT ____

PARAMETER	DESIGN VALUE	OPERATING CONDITIONS NORMAL	EMERGENCY	SHUTDOWN	OTHER
PRESSURE (psig)	35(a)		35(a)		
TEMPERATURE (°F)	50(a)		40-50(c) (e)		
FLOW RATE (gpm)	150(a)		150(b)		
NPSH REQ'D (ft)	N/A		N/A		
PRESSURE DROP (psi)	3.33(b) Note 2		3.33(b) Note 1 Note 2		
HEAT LOAD (Btu/hr)	N/A		N/A		
FLUID TYPE	Water/NaOH mix		Water/NaOH mix		
VISCOSITY (centipoise)	1.66 x 10^{-5}(b)		1.66 x 10^{-5}(b)		
FLUID pH					

FLOW IS FROM: 1-QS-A86 TO: 1-RS-P-1A

NOTES:
1. See Table 14.1-1, Node No. 1, Note 2, and Reference (b).
2. Value given is for Unit 1; Unit 2 value is 2.08 psi (d).
3. Refer to Section 14.1.12.
4. Reference (e) is for Unit 1 only

REFERENCES: (a): [25.3.5] [24.14.43]; (b): [25.4.20]; (c): [25.4.5]; (d): [25.4.21]; (e); [25.4.63] Note 4

REVISION NO: 00	DATE: July 1, 1990

Table 14.1-1. Process Condition Summary

SYSTEM: Quench Spray | SYSTEM CODE: QS | SDBD NUMBER: NAPS-QS | NODE NO.: 6

NODE TYPE: X MASS FLOW _ INTERFACE _ COMPONENT ______

PARAMETER	DESIGN VALUE	OPERATING CONDITIONS: NORMAL	EMERGENCY	SHUTDOWN	OTHER
PRESSURE (psig)	35(a)		35(a)		
TEMPERATURE (°F)	50(a)		40-50(c) (e)		
FLOW RATE (gpm)	150(b)		150(b)		
NPSH REQ'D (ft)	N/A		N/A		
PRESSURE DROP (psi)	3.42(b) Note 2		3.42(b) Note 1 Note 2		
HEAT LOAD (Btu/hr)	N/A		N/A		
FLUID TYPE	Water/NaOH mix		Water/NaOH mix		
VISCOSITY (centipoise)	1.66×10^{-5}(b)		1.66×10^{-5}(b)		
FLUID pH					

FLOW IS FROM: Line 1-QS-A87 TO: 1-RS-P-1B

NOTES:
1. See Table 14.1-1, Node No. 1, Note 2, and Reference (b).
2. Value given is for Unit 1; Unit 2 value is 2.04 psi (d).
3. Refer to Section 14.1.12.
4. Reference (e) is for unit 1 only

REFERENCES: (a): [25] [24.14.43]; (b): [25.4.20]; (c): [25.4.5]; (d): [25.4.21]; (e): [25.4.63] Note 4

REVISION NO: 00 DATE: July 1, 1990

14.2 RWST DESIGN REQUIREMENTS

14.2.1 Basic Function

1. Requirement - The RWST shall be the source of water in support of the QS System function to provide heat removal capability to be used, in conjunction with the RS System, to cool and depressurize the containment following a DBA. [25.1.2] [25.10.8]

   Design Feature - In support of the QS System function, the RWST is sized to contain the required volume of water for the following: [25.4.3] [25.4.5] [25.4.7] [25.4.47] [25.4.63]

   a. QS System for spraying into the containment
   b. SI System for core cooling
   c. RS System to increase available NPSH for the IRS pumps.

2. Requirement - The RWST shall provide a backup source of borated water to the CH System in support of ensuring the availability of negative reactivity control of the reactor. [25.1.23] [25.3.30] [25.5.11] [24.2.2]

   Design Feature - A connection is provided from the RWST to the CH System charging pumps. [25.8.1] [25.8.2]

3. Requirement - The RWST shall be the source of water to the following:

   a. The operating charging pump of the CH System when the volume control tank level drops below its low-level setpoint. [25.3.30] [24.2.2]

   b. The SI System to fill or raise the level of the SI accumulators. [25.3.28] [25.2.1]

   c. The refueling cavity, by means of the SI pumps, to support refueling operations. [25.3.28] [24.2.1]

   d. The RS System casing cooling tank for original fill and for subsequent makeup when the tank level drops below its low level set point. [25.3.29] [24.2.4]

   e. The SI System to support the hydrostatic test of the Reactor Coolant (RC) System. [25.3.28] [24.2.1]

f. The SI System to support the LHSI pump testing. [25.3.28] [24.2.1]

g. The SI System to support accumulator check valve leak testing. [25.3.28] [24.2.1]

Design Feature - Piping connections from the RWST to the CH and SI Systems are provided. [25.8.1] [25.8.2] [25.8.4] [25.8.5] Makeup water to the RS System CCT is provided using portable equipment. [25.3.29]

14.2.2 Performance

Requirement - The RWST shall provide sufficient volume of water for the QS System, in conjunction with the RS System, to meet the requirements of the containment safety analysis to depressurize the containment within 1 hour following a DBA. [25.4.5] [25.4.63]

[25.4.63] Design Feature - In support of the containment safety analysis, the volume of water provided by the RWST is 435,361 gal. [25.4.3] [25.4.5] [25.4.7] [25.4.63] To supply the stated volume, the minimum volume required in the RWST shall be 448,091 gal. [25.4.47] The Technical Specifications require the RWST to contain between 466,200 gal and 487,000 gal of 2300 ppm to 2400 ppm borated water at a temperature between 40°F and 50°F. [25.6.9] [25.6.10]

14.2.3 Regulations, Codes, and Standards

Requirement - The RWST shall be designed, manufactured, examined, inspected, tested, and certified to generally recognized codes and standards or clearly stated quality requirements to ensure a quality product in keeping with the required safety function. [25.1.12] [25.1.24] [25.5.3] [25.5.4] [25.10.8]

Design Feature - The RWST is designed in accordance with API Standard 650, "Welded Steel Tanks for Oil Storage." [25.2.4] [25.6.5] [25.8.7] [24.14.44]

14.2.4 Design Conditions

Requirement - The RWST shall be designed to be capable of operating under the pressure, temperature, and chemistry conditions of the fluid passing through the system. [25.2.5]

Design Feature - The RWST was designed to withstand the following: [25.8.7]

1. Pressure = atmospheric

2. Temperature = 150°F

14.2.13 Electrical Requirements

Requirement - The RWST has no electrical requirements.

Design Feature - Not applicable.

14.2.14 Layout and Arrangement

Requirement - The layout and arrangement of the QS System piping and components shall ensure that system performance criteria are met and shall facilitate system operation, maintenance, and testing. [25.11.1]

Design Feature - The RWST is located in the yard near the QSPA. The tank includes an internal weir into which the CAT solution is injected and from which the QS pump suction are taken. Refer to Section 14.2.11. The suction nozzles for the SI System are outside the weir. The tank has suction nozzles for the cooling subsystem at two elevations: one for when the coolers are operating and one for the refrigeration units. The return to the tank is the same for both loops. The tank also has return piping for the QS pump recirculation loop. A small portion of the return flow is through nozzles at the top of the tank. The nozzles serve to check for particulate matter in the RWST water. The RWST has an overflow line which is piped to the valve pit in the Safeguards Building. [25.8.1] [25.8.2] [24.14.47]

The RWST has external ladders and platforms as well as an internal ladder and access ways for accessibility. [25.8.7]

14.2.15 Operational Requirements

Requirements - The required volume, temperature, and boration level shall be maintained in the RWST. [25.3.30] [25.4.5] [25.4.7] [25.5.11] [24.2.2]

[25.4.63]

Design Feature - Upon initial fill or after return of refueling cavity water to the RWST, the temperature of the water in the RWST is lowered by aligning the RWST cooling subsystem to recirculate the water through the coolers. The temperature of the RWST water is maintained by aligning the cooling subsystem to recirculate the water through the refrigeration units. [25.4.26]

The boration level in the RWST is controlled and maintained during makeup by adjusting the boric acid concentration coming from the boric acid blender in the CH System. [25.3.30] [25.5.11] [24.2.2]

14.2.16 Instrumentation and Control Requirements

Requirement - Instrumentation and controls shall be provided to monitor RWST operation during normal and accident conditions, and to maintain the RWST fluid within prescribed operating ranges. [25.1.16] [25.10.8]

Design Feature - In support of the QS System functions and performance requirements, instrumentation is provided on the RWST to monitor fluid level and temperature. The RWST level instrumentation is used in support of the QS System postaccident monitoring requirements. [25.5.6] [25.8.1] [25.8.2]

14.2.17 Access, Administrative Control, and Security

Requirement - The access control, carding, and zoning for the area in which the RWST is located shall be in accordance with the requirements identified in 10CFR73 and 10CFR50.54(p) and incorporated by Virginia Power in the plant-specific security plan. [25.1.37] [25.1.57] [25.5.3] [25.5.4]

Design Feature - The yard where the RWST is located is included within the site protected area. [25.8.34]

14.2.18 Redundancy, Diversity, and Separation

1. Requirement - The RWST shall not be shared among nuclear units unless it can be shown that such sharing will not significantly impair safety functions. [25.1.15] [25.10.8]

   Design Feature - One RWST is provided per nuclear unit. There is no cross-connection between the RWST of Unit 1 and the RWST of Unit 2. [25.8.1] [25.8.2]

2. Requirement - The QS System safety-related components shall contain sufficient redundancy so that under accident conditions the minimum performance criteria of the QS System can be met assuming a single failure. [25.1.2] [25.10.8]

   Design Feature - During the short term, single failure is limited to a failure of an active component. [25.10.8] The RWST is a passive component that completes its function within approximately 1 hour. [25.4.5] [25.4.7] [25.4.63]

14.2.19 Reliability

Requirement - Failure modes and effects shall be identified and analyzed for the RWST to ensure its availability following a DBA. [25.1.1] [25.10.8]

Design Feature - The RWST is a passive component and is required for only 1 hour following a DBA. [25.4.3] [25.4.7] [25.4.7]

14.2.20 Test and Surveillance Requirements

Requirement - The RWST shall be designed to permit appropriate periodic inspection and testing of electrical, mechanical, and structural features to assess and ensure the reliability of the RWST and its capability to continue to support its minimum performance requirements. [25.1.3] [25.1.4] [25.10.8]

Design Feature - The RWST design includes test connections and piping that permit periodic testing to ensure that the RWST fluid parameters are within the values prescribed by the QS System performance requirements. [25.8.1] [25.8.2]

The RWST also has internal test nozzles similar to the QS spray header nozzles. These test nozzles are used during RWST recirculation testing to determine if there is particulate matter in the RWST water that might plug the spray header nozzles. [25.8.1] [25.8.2]

Specific testing and surveillance requirements are addressed in Chapters 19 and 21.

The RWST has ladders and platforms plus access openings that allow for inspection of the tank. [25.8.7]

14.2.21 Maintenance and Repair Requirements

Requirement - The RWST shall be located to promote accessibility for maintenance and repair and to allow for equipment removal and laydown. Corrective and preventive maintenance required to support the design basis shall be specified. [25.11.1]

The RWST shall be included in the plant ISI Program to ensure that operation or age does not alter the RWST beyond the established design basis. [25.1.18] [25.5.7] [25.5.8]

Design Feature - The RWST is located in the yard and is easily accessible. There is sufficient space for equipment removal and laydown. External ladders

14.3 QS PUMP DESIGN REQUIREMENTS

14.3.1 Basic Function

Requirement - The QS pumps shall support the QS System in providing heat removal capability to be used, in conjunction with the RS System, to cool and depressurize the containment following a DBA. [25.1.2] [25.10.8]

Design Feature - In support of the QS System function, two QS pumps provide the motive force to deliver the water from the RWST to the QS spray headers in the containment. [25.8.1] [25.8.2]

14.3.2 Performance

Requirement - The QS pumps shall deliver the water for the QS System to meet, in conjunction with the RS system, the requirements of the containment safety analysis to depressurize the containment within 1 hour following a DBA. [25.4.6]

[25.4.63]

Design Feature - The QS pumps start upon receipt of a CDA signal from the RPS or from manual initiation. Each pump delivers the flow rate as given by the system flow curve. The actual flow depends on the containment pressure and the height of water in the RWST. At the designated pump design point, the QS pumps deliver 2000 gpm at a total discharge head of 265.5 ft. [25.4.5] [25.4.6] [25.4.14] [25.4.15]

[25.4.63]

A small fraction of the flow, 150 gpm, is bled off to the suction point of the IRS pumps. [25.4.5] [25.4.7] [25.5.1]

14.3.3 Regulations, Codes, and Standards

Requirement - The QS pumps shall be designed, manufactured, examined, inspected, tested, and certified to generally recognized codes and standards or clearly stated quality requirements to assure a quality product in keeping with the required safety function. [23.1.12] [25.1.24] [25.5.3] [25.5.4] [25.10.8]

Design Feature - Each QS pump is designed, manufactured, examined, inspected, tested, and certified to meet the Class II requirements of the "Draft ASME Code for Pumps and Valves for Nuclear Power," November 1968 and applicable portions of ASME Section VIII and IX as required by the specification. The motors are designed and tested in accordance with ANSI, IEEE, and NEMA as required by the specification. [25.2.3] [25.6.1] [25.10.6] [24.14.49] [24.14.50] no dependence of one unit on the other unit's QS pumps. [25.8.1] [25.8.2]

2. Requirement - The safety-related components of the QS System shall contain sufficient redundancy and separation so that under accident conditions the minimum performance criteria can be met assuming a single failure. [25.1.2] [25.10.8]

   Design Feature - One QS pump is provided per QS System train. [25.8.1] [25.8.2] Each pump is powered from reliable, redundant EP System trains. [25.8.8] [25.8.9]

   The function of the QS System can be accomplished with only one QS System train, i.e., one QS pump operating. [25.4.4]

14.3.19 Reliability

Requirement - Failure modes and effects shall be identified and analyzed for the QS pumps to ensure their availability following a DBA. [25.1.1] [25.10.9]

Design Feature - Two separate independent QS pumps are provided for the QS System. The function and performance requirements of the QS System can be met with only one pump working. [25.4.4] [25.4.5], [25.8.1] [25.8.2]

[25.4.63]

14.3.20 Test and Surveillance Requirements

Requirement - The QS pumps shall be designed to permit appropriate periodic inspection and testing of electrical, mechanical, and structural features associated with the pumps to assess and ensure the reliability of the pumps and their capability to continue to support their minimum performance requirements. [25.1.3] [25.5.7] [25.5.8]

Design Feature - The QS System includes a recirculation flow path from the QS pumps to the RWST, which allows for testing of the QS pumps. [25.8.1] [25.8.2]

14.3.21 Maintenance and Repair Requirements

Requirement - The following are maintenance and repair requirements for the QS pumps:

1. The QS pumps shall be located to promote accessibility for maintenance and repair and to allow for equipment removal and laydown. Corrective and preventive maintenance required to

Design Feature - During the short term, single failure is limited to a failure of an active component. [25.10.8] The CAT is a passive component that completes its function within 1 hour. [25.4.5] [25.4.7]

[25.4.63]

14.4.19 Reliability

Requirement - Failure modes and effects shall be identified and analyzed for the CAT to ensure its availability following a DBA. [25.1.1] [25.10.8]

Design Feature - The CAT is a passive component which is required for only a short time following a DBA. [25.4.5] [25.4.7]

[25.4.63]

14.4.20 Test and Surveillance Requirements

Requirement - The CAT shall be designed to permit appropriate periodic inspection and testing of electrical, mechanical, and structural features to assess and ensure the reliability of the CAT and its capability to continue to support its minimum performance requirements. [25.1.3] [25.1.4] [25.10.6]

Design Feature - The CAT design includes test connections and piping, which permit periodic testing to ensure that the CAT fluid parameters are within the values prescribed by the QS System performance requirements. [25.8.1] [25.8.2]

Specific testing and surveillance requirements are addressed in Chapters 19 and 21.

The CAT has ladders, platforms, and access openings to allow for accessibility for inspection and maintenance. [25.8.20]

14.4.21 Maintenance and Repair Requirements

Requirement - The following are maintenance and repair requirements for the CAT:

1. The CAT shall be located to promote accessibility for maintenance and repair and to allow for equipment removal and laydown. Corrective and preventive maintenance required to support the design basis of the CAT shall be specified. [25.11.1]

2. The CAT shall be included in the plant ISI Program to ensure that operation or age do not alter the CAT beyond the established design basis. [25.1.18] [25.5.7] [25.5.8]

14.6 REFUELING WATER RECIRCULATION PUMP DESIGN REQUIREMENTS

14.6.1 Basic Function

Requirement - The refueling water recirculation pumps shall support the QS System in providing the heat removal capability to be used, in conjunction with the RS System, to cool and depressurize the containment following a DBA. [25.1.2] [25.10.8]

Design Feature - In support of the QS System function, the refueling water recirculation pumps circulate the water from the RWST through the RWST coolers or through the refueling water refrigeration units to keep the water at the required temperature. [25.4.26]

14.6.2 Performance

Requirement - The refueling water recirculation pumps shall support the QS System, in conjunction with the RS System, to meet the requirements of the containment safety analysis to depressurize the containment within 1 hour following a DBA. [25.4.5] [25.4.63]

Design Feature - In support of the QS System performance criteria, the refueling water recirculation pumps perform the following functions: [25.4.26] [25.4.32]

1. Upon filling of the RWST, either initially at plant startup or from the refueling cavity after a refueling, each pump is capable of circulating warm water from the RWST through both RWST coolers in parallel and back to the RWST at a rate of 650 gpm or if one cooler is out of service, through the remaining cooler at 520 gpm.

2. When the water in the tank has reached 43°F, the water from the RWST is circulated through the refrigeration units at 72 gpm until the temperature of the water reaches 40°F.

3. The pumps cycle to keep the RWST water between 40°F and 43°F.

14.6.3 Regulations, Codes, and Standards

Requirement - The refueling water recirculation pumps shall be designed, purchased, and installed to generally recognized codes and standards or clearly stated quality requirements to assure a quality product in keeping with the required safety function. [25.1.12] [25.1.24] [25.5.3] [25.5.4] [25.10.8]

14.8 REFUELING WATER REFRIGERATION UNIT DESIGN REQUIREMENTS

14.8.1 Basic Function

Requirement - The refueling water refrigeration units shall support the QS System in providing heat removal capability to be used, in conjunction with the RS System, to cool and depressurize the containment following a DBA. [25.1.2] [25.10.8]

Design Feature - The refueling water refrigeration units are used to keep the temperature of the water in the RWST between 40°F and 43°F. [25.4.26]

14.8.2 Performance

Requirement - The refueling water refrigeration units shall support the QS System, in conjunction with the RS System, to meet the requirements of the containment safety analysis to depressurize the containment within 1 hour following a DBA. [25.4.5]

[25.4.63]

Design Feature - Each refueling water refrigeration unit delivers 7.5 tons of refrigeration to keep the RWST water between 40°F and 43°F. Flow through each unit is 36 gpm with a pressure drop not exceeding 10 psi. [25.4.26] [25.6.4] [24.14.94]

14.8.3 Regulations, Codes, and Standards

Requirement - The refueling water refrigeration units shall be designed and installed to generally recognized codes and standards or clearly stated quality requirements to assure a quality product in keeping with the required safety function. [25.1.12] [25.1.24] [25.5.3] [25.5.4]

Design Feature - The refueling water refrigeration units are fabricated and wired according to the National Electric Code. Vessels are constructed according to Section VIII, Division 1 of the ASME Boiler and Pressure Vessel Code including the 1971 Summer Addendum. [25.2.2] [25.6.4] [25.12.1] [24.14.94]

Each unit is tested and rated in accordance with the American Society of Heating, Refrigerating and Air Conditioning Engineers Guide and Data Book. [25.2.2] [25.6.4] [25.12.1] [24.14.94]

14.8.27 Materials, Processes, Parts, and Equipment

Requirement - There are no special materials, processes, parts, or equipment requirements for the QS System.

Design Feature - Not applicable.

14.8.28 Personnel Safety

Requirement - Safety requirements associated with hazardous materials that impose design requirements on the refueling water refrigeration units shall be specified. [25.1.59]

Design Feature - The refueling water refrigeration units have RWST water flowing through them. The RWST water contains boric acid, which may constitute a hazard if leakage occurs. The refrigeration units contain freon, which may also constitute a hazard. [24.14.42]

14.9 QS PUMP DISCHARGE STRAINER DESIGN REQUIREMENTS

14.9.1 Basic Function

Requirement - The QS pump discharge strainers shall support the QS System to provide heat removal capability to be used, in conjunction with the RS System, to cool and depressurize the containment following a DBA. [25.1.2] [25.10.8]

Design Feature - The QS pump discharge strainers are designed to prevent debris that might clog the spray nozzles from reaching the spray rings. [25.6.2] [25.8.1] [25.8.2]

14.9.2 Performance

Requirement - The QS pump discharge strainers shall support the QS System, in conjunction with the RS System, to meet the requirements of the containment safety analysis to depressurize the containment within 1 hour following a DBA. [25.4.5] [25.4.6]

Design Feature - The discharge strainers were designed to prevent particles larger than 3/16 in. from reaching the spray ring nozzles [25.6.2] [24.14.102].

14.9.27 Materials, Processes, Parts, and Equipment

Requirement - There are no special materials, processes, parts, or equipment requirements for the QS pump discharge strainers.

Design Feature - Not applicable.

14.9.28 Personnel Safety

Requirement - There are no personnel safety requirements for the QS pump discharge strainers.

Design Feature - Not applicable.

14.10 QS SYSTEM SPRAY HEADER DESIGN REQUIREMENTS

14.10.1 Basic Function

Requirement - The QS System spray headers shall support the QS System to provide heat removal capability to be used, in conjunction with the RS System, to cool and depressurize the containment following a DBA. [25.1.2] [25.10.8]

The QS System shall also be used to remove iodine from the containment atmosphere and to control the final pH of the sump water. [25.1.8] [25.1.9] [25.1.60]

Design Feature - The QS spray rings and nozzles distribute the QS water around the upper containment area producing the proper spray pattern and atomizing the water to maximize the surface area available for heat transfer and fission product removal from the containment atmosphere. [25.4.9] [25.4.24]

14.10.2 Performance

Requirement - The QS System spray headers shall support the QS System to meet, in conjunction with the RS System, the requirements of the containment safety analysis to depressurize the containment within 1 hour following a DBA. [25.4.5] & [25.4.63]

Design Feature - The QS spray rings and nozzles distribute and atomize the spray water providing:

1. A thermal effectiveness for heat removal of at least 0.9 and [25.4.5]. & [25.4.63]

2. A containment air volume coverage of 38 percent of the total air volume, including 44 percent of the air volume above the operating floor. [25.4.[illegible]]

14.10.3 Regulations, Codes, and Standards

Requirement - The QS System spray headers shall be designed and installed to generally recognized codes and standards or clearly stated quality requirements to assure a quality product in keeping with the required safety function. [25.1.12] [25.1.24] [25.5.3] [25.5.4]

Design Feature - The QS System spray rings are designed and fabricated according to the requirements of ANSI B31.7, "Code for Nuclear Power Piping." [25.2.5] [25.6.7] [25.8.1] [25.8.2] [24.14.104]

The spray nozzles meet the requirements of Category I equipment. [25.3.16] [25.6.11] [24.14.104]

14.10.4 Design Conditions

Requirement - The QS System spray headers shall be designed to be capable of operating under the pressure, temperature, and chemistry conditions of the fluid passing through the system. [25.11.1]

Design Feature - The following are design conditions for which the QS System spray rings and nozzles are designed. [25.4.5] [25.4.29] [25.4.33] [25.4.63]

Parameter	Value
Flow (gpm)	
Ring	1600 - 2000*
Nozzles - each	6.7 - 8.4
Temperature (°F)	40 - 50
Nozzle pressure (psi)	23 - 35
Fluid chemistry	See Section 14.1.12

*QS System flow minus bleed to IRS pumps. See Table 6.1-1.

14.10.5 Loads

1. Requirement - The QS System spray headers and associated supports shall be designed to withstand or be protected from the effects of natural phenomena such as earthquakes, hurricanes, missiles, and floods due to natural phenomena. [25.1.13] [25.10.8] [25.1.53] [25.10.26] [24.14.128]

14.10.28 Personnel Safety

Requirement - There are no personnel safety requirements for the QS System spray headers.

Design Feature - Not applicable.

14.11 QS SYSTEM MOTOR-OPERATED VALVE DESIGN REQUIREMENTS

14.11.1 Basic Function

Requirement - The QS System MOVs shall be used to align the QS System such that:

1. The QS System shall provide heat removal capability to be used, in conjunction with the RS System, to cool and depressurize the containment following a DBA. [25.1.2] [25.4.5] [25.4.7] [25.4.63]

2. The QS System shall also be used to remove iodine from the containment atmosphere and to control the final pH of the sump water. [25.1.8] [25.1.9] [25.4.1]

Design Feature - The QS System MOVs function to align the QS System in the required mode of operation (see Section 3.3).

The QS System MOVs comprise the following. [25.8.1] [25.8.2]

1. The QS pump suction MOVs,

2. The QS pump discharge isolation or spray header isolation MOVs, and

3. The CAT isolation MOVs.

The QS pump discharge isolation MOVs also serve as containment isolation valves.

14.11.2 Performance

Requirement - The QS System MOVs shall be used to align the QS System such that the QS System, in conjunction with the RS System, shall meet the requirements of the containment safety analysis to depressurize the containment within 1 hr following a DBA. [25.4.5]

[25.4.63]

14.11.19 Reliability

Requirement - Failure modes and effects shall be identified and analyzed for the QS System MOVs to ensure their availability following a DBA. [25.1.1] [25.10.8]

Design Feature - The QS System has two separate and parallel trains from the RWST to the spray headers. Each train has QS pump suction and discharge MOVs. The function and performance requirements of the QS System can be met with only one train operating. [25.4.4] [25.4.5] [25.4.63] [25.8.1] [25.8.2]

There is only one feed line from the CAT to the RWST for injection of NaOH into the spray water. However, there are two CAT discharge MOVs in parallel to ensure that a path for NaOH injection into the spray water is available should one MOV fail. [25.8.1] [25.8.2]

14.11.20 Test and Surveillance Requirements

Requirement - The QS System shall be designed to permit appropriate periodic inspection and testing of electrical, mechanical, and structural features associated with the system to assess and ensure the reliability of the system and its capability to continue to support its minimum performance requirements. [25.1.18] [25.5.7] [25.5.8]

Design Feature - The QS System MOVs are included in the ISI Program. [25.5.7] [25.5.8]

14.11.21 Maintenance and Repair Requirements

Requirement - The following are maintenance and repair requirements for the QS System MOVs:

1. The QS System MOVs shall be located to promote accessibility for maintenance and repair and to allow for equipment removal and laydown. Corrective and preventive maintenance required to support the design basis of the QS System MOVs shall be specified. [25.11.1]

2. The QS System MOVs shall be included in the plant ISI Program to ensure that plant operations or age do not alter the QS System MOVs to a point beyond the established design basis. [25.1.18] [25.5.7] [25.5.8]

The major QS System design parameters are based on plant-level analyses which are summarized in the PDBD. [24.16.1] A list of the plant-level analyses follows.

1. SWEC Calculation 11715-ES-150, Rev. 2, "Loctic Input (Containment Integrity)," July 31, 1987.

2. SWEC Calculation 14938.37-US(B)-259, Rev. 0, "LOCA Analysis for Revised Technical Specification on Containment Air Partial Pressure," May 28, 1987.

3. SWEC Calculation 14938.37-US(B)-260, Rev. 0, "Main Steam Line Break Analysis," May 28, 1987.

4. SWEC Calculation 11715-ES-113, Rev. 0, "Inadvertent Operation of Containment Sprays," August 28, 1975.

5. SWEC Calculation 14938.44-UR(B)-012-0, "North Anna LOCA Dose in Control Room," April 22, 1988.

6. SWEC Calculation 11715-RP-A101-0, "DBA Thyroid Dose at E.B., N.O.R. and LPZ," August 30, 1972.

7. SWEC Calculation 14237.02-UR(B)-001-1, "Determination of Accident Analysis Results after Accounting for 2% Safety Margin for Instrument Error," March 21, 1984.

8. SWEC Calculation 14799.02-UR(B)-001-0, "Update of UFSAR Accident Dose to Account for 2% Instrument Error," March 22, 1984.

9. SWEC Calculation 13644.01-US-241-0, "Quench Spray Iodine Removal Coefficients," April 1, 1981.

10. Virginia Power Calculation SM-429, "Shutdown Reactivity following a LOCA", October 19, 1986.

11. SWEC CALCULATION 02072.7010-US(B)-274 REV 0 "CONTAINMENT LOCA ANALYSIS WITH NEW STEAM GENERATOR" JUNE 17, 1992

16.1 QS SYSTEM - FLOW VS. PRESSURE DROP (H_c - H_t) WITH AND WITHOUT 150 GPM FLOW TO SUMP

Calculation Number, Revision, and Date - 11715-361N, Rev. 0, October 4, 1977 [25.4.6]

Calculation Title - "QS System - Flow vs. Pressure Drop (H_c-H_t) with and without 150 GPM Flow to Sump"

Purpose - To develop system flow curves as a function of containment pressure and RWST level

6. Crane Technical Paper No 410, 1980

7. Virginia Power NAPS Unit 1 and 2, Plant Manual, Vol. 4

8. a. SWEC Drawing 12050-FP-4B, Rev. 12
   b. SWEC Drawing 12050-FP-14A, Rev. 9
   c. SWEC Drawing 12050-FP-14B, Rev. 9
   d. SWEC Drawing 12050-FP-14C, Rev. 7
   e. SWEC Drawing 12050-FP-14D, Rev. 7
   f. SWEC Drawing 12050-FP-14E, Rev. 8

9. SWEC Calculation 11715-US(B)-247, Rev. 1 (LOCTIC Run No. 485, (Job No. 2866), dated April 26, 1983) This requires confirmation for LOCTIC Code.

10. SWEC Calculation 11715-ES-150-1

11. SWEC Calculation 11715-390N-0

12. SWEC Calculation 11715-356N-0

13. Crane Technical Paper, 13th Rev., 1969

14. Logic diagram 11715-ESK-6K and 6L 12050-ESK-6K and 6L

15. NAPS UFSAR 6.2-89

Methodology - The Bernoulli Equation was applied to the QS System to determine flow rates at various points along the flow path. The fill time of the QS System was determined based on the flow pattern. The total delay time for the system (diesel and sequencer delay and fill time) was determined.

Results and Conclusions - The calculated QS System effective time was determined to be 63 seconds. The calculated value is within the value of 66 seconds used in the plant-level analysis calculation. [25.4.5] [25.4.63]

16.3 SUMP pH WITH 2400 PPM BORON IN RWST

Calculation Number, Revision, and Date - Virginia Power Calculation SM-415, Rev. 1, September 21, 1986 [25.4.1]

Calculation Title - "Sump pH with 2400 ppm Boron in RWST"

1

IMAGE EVALUATION
TEST TARGET (MT-3)

150mm

6"

1

IMAGE EVALUATION
TEST TARGET (MT-3)

150mm

6"

1

IMAGE EVALUATION TEST TARGET (MT-3)

150mm

6"

1

IMAGE EVALUATION
TEST TARGET (MT-3)

1.0 1.1 1.25 1.4 1.6 1.8 2.0 2.2 2.5 2.8 3.2

150mm

6"

Table 23.1-3. Design Basis Setpoints, Safety Analysis Setpoint Parameters, QS System

	Parameter	Basis/Reason		Reference
1.	RWST Level Alarm	The RWST level setpoints shall be set such that:		[25.4.3] [25.4.5] [25.4.7] [25.4.47] [25.4.63]
		a.	The minimum usable volume of water available from the RWST shall be in accordance with the containment safety analysis.	
		b.	The low-level alarm, which alerts the operator to manually shift the SI System to recirculation mode, shall be in accordance with the minimum volume of water expended from the RWST at SI System switchover.	[25.4.3] [25.4.5] [25.4.7]
		c.	The LHSI auto switchover setpoint, which automatically causes the SI System to shift to the recirculation mode, shall be in accordance with the maximum volume of water expended from the RWST at the SI System switchover.	[25.4.3] [25.4.5] [25.4.7]
2.	Chemical Addition Tank Level	The low level setpoint for the CAT ensures that sufficient NaOH is available to be added to the containment spray in the event of a LOCA, thus maintaining a sump water pH of greater than 7.0.		[25.1.8] [25.1.60] [25.4.1]

24.0 OPEN ITEMS

This chapter identifies open items (as defined in Section 1.13), including errors and omissions that have been identified during the preparation of this SDBD. Each open item is noted in the body of the text with an open item number, rather than stating the full description of the open item within the text. The number used in the text [24.X.X] corresponds with the number in this chapter. The combination of the SDBD number and the open item establishes a unique number, e.g., SDBD-NAPS-AFW-24.6.2, that is used to monitor the status of the open item in the Design Basis Document Item Resolution System (DBD IRS).

24.2.1 Sections 2.1, 3.1, 5.1.6, 14.1, and 14.2.2. The requirements imposed on the QS System by the SI System should be addressed in SDBD-NAPS-SI. When SDBD-NAPS-SI is issued, the requirements as stated should be verified.

24.2.2 Sections 2.1, 5.1.1, 14.1, and 14.2.1. The requirements imposed on the QS System by the CH System should be addressed in SDBD-NAPS-CH. When SDBD-NAPS-CH is issued, the requirements as stated should be verified.

24.2.3 Sections 2.1.2, 5.1.1, and 14.1 refer to Virginia Power NAPS Units 1 and 2 ATWS Report [25.5.9], which is not a controlled document. The statement that the RWST may be used as a supply of borated water to the CH System to support safe shutdown after an ATWS event needs to be confirmed. Consequently, when SDBD-NAPS-CH is issued, this requirement needs to be confirmed and the appropriate section of SDBD-NAPS-CH referenced in this SDBD.

24.2.4 Sections 2.1.3, 5.1.5, Item 2, 14.1, and 14.2.1. Reference [25.3.27] (training module) indicates that fill and makeup water to the casing cooling tank is provided by the RWST via the use of hose connections. This statement needs to be verified when SDBD-NAPS-RS [25.3.29] is issued.

24.2.5 Section 2.2.1. There is currently no evaluation that documents that the consequences (i.e., temperature/pressures) of a LOCA or a MSLB envelop the consequences of all other accidents.

[25.4.63]

24.2.6 Section 2.2.1, Table 6.1-1, Table 11.1-5, Item 7, and Section 12.3. The containment safety analyses [25.4.5] [25.4.7] and the site boundary dose analyses [25.4.10] [25.4.11] [25.4.25] assume 100-percent containment atmosphere spray coverage instead of 38 percent for QS and 74 percent for RS as documented in calculation [25.4.24]. The site boundary dose analyses may need to be revised to address actual spray coverage.

24.2.7 Section 2.2.1, Table 6.1-1, and Table 11.1-5. The spray heat removal effectiveness, (which is based on nozzle data, droplet fall height, etc.) is assumed to be 0.9 in the containment analyses. A calculation or other form of documentation to determine the spray heat removal effectiveness may be required.

24.2.8 Section 2.2.1 and Table 6.1-1. The bases for the lambda removal coefficients used in the dose analyses and their relationship to the QS and sump water pH values need to be documented.

24.2.9 Sections 2.2.1, Table 6.1-1, and Table 11.1-5. Technical Specification basis 3/4.5.5 refers to the pH value of the QS spray water as between 8.5 and 11.0 and a sump water pH value between 7.7 and 9.0. The upper limit values for pH as quoted are not documented in calculation [25.4.1].

24.2.10 Sections 2.2.1, Table 6.1-1, and Table 11.2-1. Technical Specification basis 3/4.1.2 refers to the pH value of the solution recirculating within the containment after a LOCA as between 8.5 and 11.0. This value is not in agreement with basis 3/4.5.5 (see Section 21.3), nor with the value calculated in reference [25.4.1]. The discrepancy should be resolved.

24.2.11 Section 2.3.28 refers to safety requirements associated with hazardous materials. Documents stating Virginia Power's commitment to 29CFR1910 requirements have not been located.

24.2.12 Sections 2.2.2, 4.3, 6.2, 13.2.1, 14.1.1, 14.1.2,14.1.16, and 15.1.3. It is not clear why the NAPS Units 1 and 2 RG 1.97 Compliance Report [25.3.48] does not list instrumentation to monitor QS System containment isolation valve position as instrumentation required to meet RG 1.97 Type B variables.

24.2.13 Sections 2.3.28 and 14.1.28 refer to the QS System design including industrial safety requirements specified in 29CFR1910. Documents referencing such a commitment, if any, were not available for review when this SDBD was written. When an appropriate reference is found, it should be included in this system as a reference. If no reference is found, this requirement statement may need to be deleted or considered applicable based on good engineering practice.

INSERT 24.2

24.4.1 Section 4.1. The UFSAR states that all QS System piping, except for flow test lines, is Seismic Category I. Since the UFSAR is not a referenceable design basis document, the statement should be confirmed.

24.4.2 Sections 4.2.4, Item 1 and 14.1.7, Item 14 refer to the HO System. When SDBD-NAPS-HO is issued, it should be verified that the HO System maintains the environment in QS System component locations.

24.4.3 Sections 4.2.4 and 5.2.2 refer to the HA System. When SDBD-NAPS-HA is issued, it should be confirmed that the HA System helps maintain the environment in the Auxiliary Building and in the cable tunnel where QS System components are located.

24.4.4 Sections 4.2.4 and 5.2.5 refer to the HR System. When SDBD-NAPS-HR is issued, it should be confirmed that the containment air recirculation coolers of the HR

Insert 24.?

24.2.14 Sections 2.2, 11.1.5, 11.1.4

Calculation [25.4.7] should be revised to address SG replacement at NAPS-1 and to document why the calculation (as is) is still considered the bounding MSLB analysis for NAPS Units 1 and 2.

01287.B07

25.4.62 SM-429 (no revision number available), "Shutdown Reactivity Following a LOCA," Virginia Power, October 19, 1986

INSERT 25.4

25.5 LICENSING CORRESPONDENCE

25.5.1 No. 412, "Analysis and System Modification for Recirculation Spray Pumps Net Positive Suction Head, North Anna Power Station Unit 1," Virginia Power, September 16, 1977.

25.5.2 No. 701, "North Anna 2 Compliance with NRC Regulations," Virginia Power, August 11, 1980.

25.5.3 Facility Operating License No. NPF-4 (North Anna Power Station Unit No. 1) November 26, 1977, as amended. Section 2.D.

25.5.4 Facility Operating License No. NPF-7 (North Anna Power Station Unit No. 2) August 21, 1980, as amended. Section 2.C.

25.5.5 "10CFR50 Appendix R Report NAPS - Units 1 and 2," Rev. 6, Virginia Power, June 1988.

25.5.6 Technical Report No. PE-0013, Regulatory Guide 1.97 Compliance for NAPS Rev. 0, Virginia Power, December 19, 1989.

25.5.7 "NAPS Unit 1 Inservice Testing Program Plan for Pumps and Valves First Inspection Interval June 6, 1978 - December 14, 1990," Rev. 4, Virginia Power, January 16, 1989.

25.5.8 "NAPS Unit 2 Inservice Testing Program Plan for Pumps and Valves First Inspection Interval December 14, 1980 - December 14, 1990," Rev. 4, Virginia Power, January 16, 1989.

25.5.9 "NAPS Units 1 & 2 Anticipated Transients without Scram," Virginia Power, January, 1987.

25.5.10 "NAPS Units 1 & 2 Station Blackout Safe Shutdown Analysis, Virginia Power, June, 1987.

25.5.11 NE Technical Report No. 539, Rev. 0, "Boron Concentration Increase for North Anna Units 1 and 2," Virginia Power, October 1986.

25.5.12 NRC Letter to Virginia Power, ACN 8211060248, "VEPCO Topical Report, Quality Assurance Program Operations Phase," October 6, 1982, documenting acceptance of Amendment 4 of Virginia Power QA Program VEP-1-4A.

INSERT 25.4

25.4.63 02072.2010-US(B)-274, Rev. 0, "Containment LOCA Analysis with New Steam Generator," June 17, 1992

01287.B07

25.10.27 UFSAR Section 3.2.2, "System Quality Group Classification."

25.11 ACCEPTED ENGINEERING PRACTICE

25.11.1 Some features of the original design of the North Anna and Surry plants were based on requirements that were not documented in a source that can be identified or referenced. These requirements were employed to ensure a safe and reliable design and were based engineering experience, industry practice, and previously proven design that existed during the time the plants were destroyed. DESIGNED

25.12 MISCELLANEOUS REFERENCES

25.12.1 American Society of Heating, Refrigerating, and Air Conditioning Engineers Guide and Data Book, "Equipment," Chapter 22, Pages 270-275, 1969.

25.12.2 Design Change Log for NAPS, July 6, 1988.

25.12.3 PSE-185/Task Item 328, System Design Basis Documentation, Virginia Power, May 5, 1989.

25.12.4 NUREG-0696, Analog Computer Input/Output List, NAPS, September 24, 1987.

25.12.5 NUREG-0696, Digital Computer Input/Output List, NAPS, October 14, 1987.

25.12.6 "Hydraulic Model Studies of the Reactor Containment Building Sump," Alden Research Laboratories, July 1977.

25.12.7 Task Item 107, "MOV Rerate Evaluation - GDC 17," NAPS Units 1 and 2, SWEC (no date available).

25.10.27 UFSAR Section 3.2.2, "System Quality Group Classification."

25.11 ACCEPTED ENGINEERING PRACTICE

25.11.1 Some features of the original design of the North Anna and Surry plants were based on requirements that were not documented in a source that can be identified or referenced. These requirements were employed to ensure a safe and reliable design and were based engineering experience, industry practice, and previously proven design that existed during the time the plants were destroyed.

25.12 MISCELLANEOUS REFERENCES

25.12.1 American Society of Heating, Refrigerating, and Air Conditioning Engineers Guide and Data Book, "Equipment," Chapter 22, Pages 270-275, 1969.

25.12.2 Design Change Log for NAPS, July 6, 1988.

25.12.3 PSE-185/Task Item 328, System Design Basis Documentation, Virginia Power, May 5, 1989.

25.12.4 NUREG-0696, Analog Computer Input/Output List, NAPS, September 24, 1987.

25.12.5 NUREG-0696, Digital Computer Input/Output List, NAPS, October 14, 1987.

25.12.6 "Hydraulic Model Studies of the Reactor Containment Building Sump," Alden Research Laboratories, July 1977.

25.12.7 Task Item 107, "MOV Rerate Evaluation - GDC 17," NAPS Units 1 and 2, SWEC (no date available).

25.12.8 DC-90-13-1, Steam Generator Replacement, North Anna Unit 1.

NAPS RS

During all modes of normal operation except cold shutdown and refueling, the RS System shall provide the means to cool and maintain the temperature of the water in the casing cooling tank. [25.11.1] (Casing cooling water is injected into the outside recirculation spray (ORS) pump suction following an accident to ensure that the net positive suction head available (NPSHA) to the ORS pumps is greater than the net positive suction head required (NPSHR).) [25.1.22] [25.10.2]

2.2 PERFORMANCE CRITERIA

This section identifies the performance criteria for each of the corresponding functions identified in Section 2.1. The performance criteria establish the objective measure of functionality and provide quantification of acceptable performance for satisfying the system functional requirements which are identified. Performance criteria are expressed in the most fundamental manner to allow the designer/engineer flexibility in establishing design features to satisfy the criteria. Generally, the fundamental performance criteria can be satisfied by many designs. The primary objective in establishing performance criteria is to define the fundamental information needed to design the system that is independent of component selection or system configuration.

The performance criteria identified in this chapter are presented "qualitatively." The performance requirements are stated in "quantitative" terms in Chapter 6, "Key System Parameters."

The performance criteria for the RS System are identified in the following sections:

2.2.1 Safety-Related Performance Criteria

This subsection identifies the safety-related performance criteria for the RS System. Performance criteria related to the same function are grouped together. The following are safety-related performance criteria:

1. In support of the heat-removal/pressure-suppression function identified in Section 2.1.1, during an accidental release of high-energy fluids inside containment, the RS System performance requirements shall meet those stated in the containment safety analyses in order to depressurize the containment to subatmospheric conditions within 1 hour following the accident. [25.10.6] [25.10.7] The RS System design parameters are input into the containment safety analyses, [25.4.1] [25.4.2] [25.4.4] [25.4.72] [24.2.1] thus establishing the following as its safety-related performance criteria. It should be noted that the performance criteria listed below are based on the loss-of-coolant accident (LOCA) and main steam line break (MSLB) since the consequences of these two accidents envelop the

[25.4.139]
[24.2.17]

consequences of other accidents (rod ejection accident (REA), feedwater line break (FWLB)) that will cause containment pressurization. [24.2.2]

[25.4.139]
[24.2.17]

a. The RS System shall be initiated automatically by the containment depressurization actuation (CDA) signal, which shall be initiated by a containment "high-high" pressure signal activated at a containment pressure established in the safety analyses. [25.4.2] [25.4.4] [25.4.72]

b. The CDA signal shall activate timers, which will delay the start of the RS pumps. This delay will allow for the adequate buildup of water (resulting from the break and the operation of the QS System) in the containment sumps and prevent IRS/ORS pump cavitation at startup. The maximum time delay allowable prior to obtaining an effective full flow spray shall be as established in the safety analyses. [25.4.2] [25.4.4] [25.4.72]

c. The RS System shall pump containment sump water through the shell side of the RS coolers to the RS rings at a minimum flow rate established in the safety analyses. [25.4.2] [25.4.4] [25.4.72]

d. The minimum overall heat removal coefficient for the RS coolers shall be as established in the safety analyses. [25.4.2] [25.4.4] [25.4.72]

e. The location of the RS rings, the size and arrangement of the nozzles in the ring, and the pressure difference between the fluid in the spray header and containment atmosphere shall support the minimum droplet thermal effectiveness of the sprays assumed in the containment safety analyses. [25.7.29] [25.4.4] [25.4.2] [25.4.72] [24.2.3]

f. The location of the RS rings and the size and arrangement of the nozzles in the ring shall maximize the volume of containment atmosphere covered by the recirculation spray to support heat removal. [25.4.31] [24.2.4] [25.4.4] [25.4.2] [25.4.72]

[25.4.139]

[25.4.139]
[24.2.17]

g. To support the postaccident NPSH requirement of the ORS pumps, cold water injection shall be provided at the ORS pump suction at the rate, temperature, and for the duration required by the NPSH calculations included with the containment safety analyses. [25.4.4] [25.4.2] [24.2.5] Note that since casing cooling flow is available, credit is taken for it in the containment analyses. [25.4.4] [25.4.2] [25.4.72]

h. To ensure that the RS System performance is in accordance with the assumptions of the containment safety analyses, the containment sump shall be designed to discourage debris transport. Consequently, the total sump screen area shall be large enough to ensure low sump fluid velocity (thus reducing debris transport to the RS pump suction), and the sump screen mesh opening size shall be smaller than the RS nozzles (to prevent nozzle blockage). [25.12.2] [25.8.58] [24.2.6] [25.4.45] [25.4.138]

For completeness, safety-related performance criteria applicable to systems interfacing with the RS System (but necessary for proper operation of the RS System) are listed below. [24.2.7]

a. The QS System and the Safety Injection (SI) System shall operate as established in the containment safety analyses. [25.4.2] [25.4.4] [25.4.72]. The above includes cold water injection from the QS System to the inside recirculation spray (IRS) pump suction at the rate and for the duration assumed in the NPSH calculations documented in calculation [25.4.2]. [25.4.139]

b. The Service Water (SW) System shall provide cooling water to the tube side of the RS coolers at the minimum rate established in the safety analyses. [25.4.4] [25.4.2] [25.4.72] [24.2.1]

c. The maximum allowable primary containment air partial pressure versus service water temperatures and refueling water storage tank (RWST) water temperature shall be in accordance with the containment safety analysis [25.4.2], which is also

[25.4.139]

in 10CFR50, Appendix J. [25.1.19] [25.1.53] [25.10.1] [25.5.4] [25.5.5]

2.2.2 Non-Safety-Related Performance Criteria with Special Regulatory Significance

This subsection identifies the non-safety-related with special regulatory significance (NSQ) performance criteria for the RS System. Performance criteria related to the same function are grouped together. The following are NSQ performance criteria:

In support of providing the means to assess plant conditions during and following an accident (as identified in Section 2.1.2), the RS System design shall include instrumentation that will measure RS flow rate, sump water level, and sump water temperature in accordance with the performance criteria (i.e., range, sensitivity, etc.) required by RG 1.97. [25.1.33] [25.3.48] [24.2.12] [24.2.16]

2.2.3 Non-Safety-Related Performance Criteria

This subsection identifies the non-safety-related performance criteria for the RS System. Performance criteria related to the same function are grouped together. The non-safety-related performance criteria are:

[25.4.139]

In support of the RS System function to automatically maintain the temperature of the CCT within the requirements of the ORS pump NPSH analyses [25.4.2] during all modes of operation except shutdown, refueling, and accident, the following performance criteria shall be met: [25.11.1]

1. The temperature setpoints for the CCT temperature sensors (which initiate the CCT cooling loop) shall be well below that required by the ORS pump NPSH analyses. [25.11.1]

2. The heat-removal capability of the refrigeration units associated with the CCT cooling loop shall be greater than the normal operation heat gain of the CCT fluid during the worst ambient conditions. [25.4.35]

2.3 SYSTEM DESIGN CRITERIA (ANSI N45.2.11)

The purpose of this section is to identify the fundamental design criteria of the system based on the application of ANSI N45.2.11, "Quality Assurance Requirements for the Design of Nuclear Power Plants," as related to design input requirements. Virginia Power is committed to follow the requirements contained in ANSI N45.2.11 in fulfillment of its quality assurance program requirements.

The RS System consists of four 50-percent subsystems that pump the sump water resulting from the accident through heat exchangers to spray headers located in the containment dome. [25.10.6] [25.8.1] [25.8.2] The pumps for two subsystems are inside the containment and two are outside, and all four subsystems are independent of each other in the piped portion of the recirculation flow path. [25.10.6] [25.8.1] [25.8.2]

If a LOCA, a rod ejection accident, a steam line break, or a feedwater line break occurs inside the containment, the pressure in the containment atmosphere will increase from the resulting energy release. Depending on the type of accident, some of the fluid released from the RC System or the steam generator will be liquid and will drain from the upper floors of the containment down to the basemat. [25.4.81] Thus, the normal containment sump will fill with the drained water, and overflow from its location adjacent to the primary shield wall into the diversion channel in the basemat floor and onto the basemat floor itself. [25.8.9] The released energy will increase the pressure and temperature inside the containment. When the containment "high-high" pressure setpoint is reached, it will actuate the CDA signal, which will in turn initiate the operation of the QS System. The QS System will spray water from the RWST (mixed with NaOH solution from the chemical addition tank) into the containment, thus increasing the sump water inventory. [25.3.6] [25.10.6] [24.2.7] This high pH QS System water, at a temperature of 40 to 50°F, is also pumped directly to the suction of the IRS pumps to reduce the temperature of the sump water and increase the available NPSH at the IRS pumps. [25.3.6] [25.4.2] [25.4.139] [24.2.7] The remainder of the RWST water is pumped into the RC System by the high-pressure or low-pressure safety injection pumps where, along with the remaining RC System inventory and accumulator discharge, it cools the fuel during the accident and eventually migrates to the containment sump. [25.3.10] [24.2.7] The combination of the water from all these sources eventually floods the basemat floor of the containment and the RS sump screen well. The water from different sources, with different pHs, is mixed in the sump to provide an ultimate sump pH of 7.7. [25.4.76]

The CDA signal also ensures that all RS MOVs are open [25.8.43] [25.8.44] [25.8.48] [25.8.49] and initiates timers that delay the start of the RS pumps. [25.8.31] [25.8.11] [25.8.12] [25.8.14] [25.8.15] [25.8.28] [25.8.27] [25.8.30] This delay allows for adequate buildup of water in the containment sumps and prevents IRS/ORS pump cavitation at startup. [25.4.81] [24.3.1] The IRS subsystems contain vertical pumps located inside containment. The pumps take suction from within the screen well. [25.8.58] [25.8.1] [25.8.2] The sump liquid is pumped through the shell side of the RS heat exchanger, also located inside the containment, to a spray header in the containment dome that covers a 180-degree arc. The spray headers of each IRS subsystem cover a different 180-degree arc, thus providing 360-degree spray coverage. [25.8.8] The RS coolers transfer the heat from the hot sump liquid to the water of the ultimate heat sink supplied by the SW System. [25.3.7] [24.2.7]

The cooled water is sprayed through two types of spray nozzles to provide proper containment atmosphere coverage (85 percent when combined with QS System coverage), and to provide proper droplet size for heat and iodine removal. The larger droplets, while less effective for heat removal, ensure (due to momentum) that the sprayed droplets reach the extreme areas. The small fog droplets increase the overall heat transfer and steam condensation and are effective in removal of iodine. [25.4.31] [25.4.46] [24.2.3] [25.1.8]

The ORS subsystems perform the same functions as the IRS subsystem. [25.8.1] [25.8.2] The suction piping to the ORS pumps runs from the containment basemat RS sump liner, under the liner embedded in the mat, to the Safeguards Building valve pit. [25.8.8] The suction pipe has a containment isolation valve, which is normally open, and is routed to a vertical cylindrical pressure vessel that serves as the casing of the ORS pump. The ORS pump discharges through an outside containment isolation valve (normally open) and an inside containment check valve to its RS cooler and spray header. [25.10.6] [25.8.2] [25.8.1]

The casing cooling subsystem is a part of the RS System and is provided to increase the available NPSH at the ORS pump suction to ensure that the ORS pumps perform in accordance with the safety analyses. Upon receipt of a CDA signal, the casing cooling pumps start up and inject cold water from the CCT into the ORS pump suction to reduce the temperature of the containment sump water and increase the available NPSH at the ORS pumps. The QS System provides a similar cooling flow to the IRS pump suction. [25.8.1] [25.8.2] [25.4.2] [25.4.139] X

All electrical components in the RS System have emergency power except for the casing cooling subsystem recirculation and cooling equipment. (See Figure 4.2-2.) Since there are four 50-percent RS subsystems being powered by two emergency electrical subsystems (or trains), the association of the electric source and the RS subsystems is designed to ensure that minimum safeguards are available on the loss of an electrical train. Consequently, all components (i.e., pumps, valves, etc.) supporting an RS subsystem are powered by the same electrical train. The QS subsystem that provides cooling flow to an IRS pump is powered by the same electrical train as the IRS subsystem it supports. Similar electrical compatibility is maintained between the RS subsystems and associated casing cooling subsystem and the SW System. Also, one inside and one outside RS subsystem powered by the same electrical train combine to provide complete 360-degree spray coverage. [25.8.1] [25.8.2] [25.10.6] [25.3.8]

The RS equipment is discussed in more detail below.

The RS sump screens are provided to protect the suction piping to the RS pumps against blockage. The screen area is designed to discourage debris transport and the screen mesh is sized to protect the spray nozzles from clogging. The screen well is an enclosure against the containment wall opposite the RS valve pit. [25.8.8] Coarse and fine mesh screens are provided on the three vertical sides and its solid top deck supports the IRS pumps. The RS screen well is divided by screens into two 100-percent sections. [25.8.58] Four suction pipes are located inside the sumps. One ORS pump and one low-head safety injection (LHSI) pump suction pipe are located in each side of the divided sump. The suction lines embedded in the containment mat have cylindrical intake screens to eliminate vortexing. [25.8.8] [25.10.6]

The IRS pumps are vertical, two-stage turbine pumps. They are capable of providing a flow of 3300 gpm against a head of 275 ft. [25.4.11] The inside spray pumps are within a cylindrical screen extending from the deck of the screen structure to the bottom of the sump. The pump casing itself is about 13 ft long, extending into a shallow well in the sump. The motor and discharge nozzle part of the pumps' casing are on top of the sump screen deck. [25.10.6] The QS pump discharges 150 gpm to a ring header around the suction of the IRS pumps in order to supply cool water during the initial phase of the accident for NPSH enhancement. [25.3.6] [24.2.7] [25.4.2] [25.4.139]

The ORS pumps are located in the Safeguards Building and are also vertical, two-stage turbine pumps. They are capable of providing a flow of 3700 gpm against a head of 290 ft. [25.4.12] The outside pumps have tandem mechanical seals to prevent leakage of containment water. [25.6.1] [25.10.6] The pumps are mounted on top of a vertical pressure vessel that serves as the well casing. [25.6.6] These casings are approximately 50 ft high in order to place the motors at an elevation above the ground grade for protection against flooding. The discharge nozzle section of the casing and the vertical motor are on top of this casing. The suction line from the RS sump in the containment enters the bottom of the casing. The casings are, in effect, extensions of the containment liner. [25.10.6] The ORS pumps have 800 gpm of casing cooling water injected into the suction piping during the initial phases of the accident for NPSH enhancement. [25.4.53] [25.4.57] [25.4.83] [25.4.82]

The four RS heat exchangers are located inside the containment. [25.8.9] They are of vertical shell and straight tube design. The higher pressure RS fluid is cooled on the shell side by lower pressure service water in the tubes. The four heat exchangers are identical. [25.6.9]

The RS headers are mounted on the dome liner below and farther from the containment center than the QS ring headers. Each 180-degree RS header contains 390 nozzles: 293 nozzles are of the fine atomizing type and 97 are of a coarse droplet type. [25.4.62] [25.8.8] [24.3.2] Nozzle orientation serves to maximize the volume of the containment being sprayed.

Piping from the IRS pumps to the headers does not contain valves. A blank flanged tee at the headers allows for inserting a spool piece, to connect to a drain line back to the sump that allows full-flow testing of the IRS pumps. The nozzles must be plugged for the test. [25.8.1] [25.8.2] Spool pieces can also be installed in the discharge lines of the IRS pumps to direct flow back into the diked sump for pump testing.

The RS System piping (other than the ORS pump suction lines), which connects directly to the containment atmosphere and penetrates reactor containment, is provided with redundant containment isolation valves, i.e., an ORS pump discharge MOV and an inside containment weight-loaded check valve. A single isolation valve outside containment is provided for the ORS pump suction to promote reliability and low suction losses. [25.8.1] [25.8.2] The suction MOV is in the valve pit in the Safeguards Building, with a long extension stem to the motor and handwheel above grade level to protect against flooding. The ORS pump discharge MOVs are at elevation 258 ft-3 in. of the Safeguards Building. [25.8.57] The ORS pump discharge valve motor is also above grade. Both suction and discharge MOVs can be operated manually and are handwheel operable from one floor above the piping. [25.8.57] Both MOVs are open during plant operation. [25.8.1] [25.8.2] Any containment leakage during normal operation is into the subatmospheric containment. In the event of a leak during an accident, the safety-related Safeguards Building ventilation system processes the air through safety-grade charcoal filters. [25.8.64] [25.8.67] [25.10.17] [25.3.30] [25.7.24] If major leakage occurs at the suction valve, the valve pit fills with water and prevents containment atmosphere from escaping into the Safeguards Building. The weight-loaded check valves inside the containment provide positive closure during normal operation. Furthermore, the ORS headers include fittings similar to the IRS headers to support full-flow testing. [25.8.1] [25.8.2]

The casing cooling subsystem consists of a outdoor cylindrical tank (that provides a supply of borated water at 42°F to 45°F for about 1 hour following an accident), two pumps to inject the contents of the tank into the ORS pump suction (to enhance the NPSH during the initial phase of operation of the RS System), MOVs to be activated by the CDA signal, piping, fittings, and other small components. [25.4.53] [25.4.57] [25.4.82] [25.4.83] [25.8.1] [25.8.2] [25.3.47] [25.4.2] [25.4.4] [25.4.72] [25.4.139] [24.2.17] The water is pumped to the ORS pumps via redundant flow paths. [25.8.1] [25.8.2] The casing cooling pumps are safety-related, centrifugal pumps, with a capacity of 960 gpm against a head of 101 ft, and are located in the casing cooling pumphouse. [25.4.53] [25.4.57] [25.4.82] [25.4.83] [25.8.61] [25.8.62] The pumps are started by the CDA signal. [25.8.1] [25.8.2] The discharge from each casing cooling pump is tied to the suction line from the containment to the ORS pumps, thus making the line an extension of the containment boundary. [25.8.1] [25.8.2] A check valve, a normally closed MOV, and a second normally open MOV are provided in the line to support containment isolation. [25.8.1] [25.8.2] A CDA signal ensures that the casing cooling subsystem MOVs are open. Containment isolation is maintained after the tank has drained down by the motor-operated casing cooling pump discharge valves, which close automatically on receipt of a low water level in the CCT and a low pump discharge flow as measured across the recirculation branch line off the casing cooling pump discharge. [25.8.1] [25.8.2]

The CCT contents are required to be of adequate level and temperature during plant operation. [25.4.2] [25.4.139] The temperature of the CCT fluid is reduced and maintained within a band of 42°F to 45°F by a non-safety-related casing cooling recirculation subsystem, which consists of a centrifugal pump and refrigeration unit and is activated by a CCT fluid temperature sensor. [25.3.47] [25.8.1] [25.8.2] [25.3.11] [25.3.12] Since the temperature control of the fluid contents of the CCT is a part of the <u>Technical</u>

When a periodic recirculation flow test is performed on the outside RS pumps, it is necessary to fill and vent the pump casing and ensure that the pump seal water lubrication system is filled. A fill station is used when filling the outside RS pump seal water system with primary grade water. The fill station helps prevent damage to the seal head tank diaphragm by using PCV-RS-107A and 107B (207A and B for Unit 2) to limit the filling pressure to 4 psig. RV-RS-100A and 100B (200A and B for Unit 2) set at 5 psig provide a backup to limit the fill pressure. [24.3.7] Closing of the suction line valve and the isolation valve between the pump discharge and the containment penetration will allow the pump casing to be filled with water and the pump to recirculate water through a test line from the pump discharge back to the pump casing.

Periodic recirculation flow testing of the IRS pumps is accomplished by removing the straight spool piece in the pump discharge line to the RS coolers and inserting an elbow spool piece to allow the water to flow through a test line back to the containment sump. [25.8.1] [25.8.2] A portable dike secured by permanently mounted support brackets is installed around the sump during flow testing. The dike will contain approximately [illegible]000 gallons of water, which is adequate for filling an RS subsystem and properly testing the associated pump. A minimum water level is required in the containment sump and portable dike in order to provide the pump with a minimum acceptable NPSH. [25.4.2] [25.4.139] Water is supplied to the sump from the refueling cavity by a portable pump and piping and is returned to the cavity following completion of the test. Casing cooling fluid or primary grade water can also be used to fill the diked sump. The dike is suitably sealed to the walls and floor of the containment during testing and is removed and stored elsewhere during normal station operation. [25.7.28] [25.10.6] [24.3.8]

Following completion of the test, strict administrative procedures ensure that the dike and elbow spool pieces are removed and that the straight spool piece is inserted to provide the proper flow path from the pump through the coolers and out to the spray headers.

The throttle valves in the test lines [25.8.1] [25.8.2] are used to vary the RS pump disc[illegible] pressure so that flow readings at various discharge pressures are obtained during the flow tests. These points are compared to the pump operating curve, which

		System shall also provide water during ORS pump testing through the use of hose connections.	
5.2.16	Quench Spray (QS) [25.4.139] [24.2.17]	1. In the event of an accidental release of high-energy fluids in containment (LOCA, MSLB, REA, FWLB), the QS System, in conjunction with the RS System, shall cool and depressurize containment to subatmospheric within 1 hour.	[25.1.16] [25.4.1] [25.4.2] [25.4.4] [25.4.72] [25.1.21] [25.1.1]
		2. During Condition IV events (LOCA, MSLB, REA, FWLB), the QS System shall provide sufficient flow of low-temperature water to the suction of the IRS pumps to increase available NPSH.	[25.1.22] [25.4.2] [25.4.139]
		3. During normal operations, the QS System RWST shall provide makeup water to the CCT, as necessary, using a hose connection. RWST water shall also be used for IRS pump testing. A portable pump (and piping) shall be used to transfer the water from the refueling cavity to the dike and back after test completion.	[25.3.8] [24.5.3]
5.2.17	Reactor Protection (RPS)	During Condition IV events (LOCA, MSLB, REA, FWLB), the RPS Systems' Engineered Safety Features (ESF) Actuation subsystem shall provide the CDA initiation signal to the RS System.	[25.1.1]
5.2.18	Security (SE)	During all condition events, the SE System shall provide access control to the cubicles containing safety-related RS components.	[24.5.4] [25.1.36] [25.5.4] [25.5.5]

5.2.19	Service Water (SW)	During Condition IV events (LOCA, MSLB, REA, FWLB), the SW System shall provide cooling water to the RS coolers at the rate established in the containment safety analyses. (The safety analyses provide for a range of SW temperatures in conjunction with other parameters and are also the basis for Technical Specifications Figure 3.6-1.)	[25.4.4] [25.3.11] [25.3.12] [25.4.79] [25.4.77] [25.4.78] [25.4.139] [24.2.17]
5.2.20	Station Service Power (ESS)	During all Condition I events, the ESS System shall provide non-Class 1E 480V power to the CCT recirculation pump and refrigeration unit motors.	[25.2.8] [25.12.10]

Table 6.1-1. Key System Parameters, Safety Related Functions, RS System

(A) Item No.	(B) System Parameter	(C) Function	(D) Operating Condition	(E) Design Requirement	(F) Reference	(G) Remarks
1	Spray Effective Time	Time after CDA signal when full flow RS spray is delivered in containment	Condition IV • LOCA • MSLB • FWLB • REA	Max: 300 sec [25.4.2] [25.4.4] [25.4.72] [24.2.1] [25.12.2] [24.3.1] [25.4.139] [24.2.17]	Containment Safety Analyses (see Table 11.1-5, item 6 and Sec. 2.2.1)	Time after CDA by which RS sprays shall be delivered at the rated flow, in containment atmosphere, per safety analyses. The spray effective time addresses diesel sequencing, delay in pump startup to avoid cavitation, pump "up-to speed" and system fill time. The IRS and ORS pumps' startup (at the receipt of the CDA signal) are delayed by means of timers by 195 and 210 secs, respectively, to allow borated water to accumulate in the sump. This delay is necessary for RS pump operation without cavitation (or creation of a vortex core) to the extent that would impact the system operation. (It should be noted that to ensure containment depressurization in accordance with the safety analyses, the CDA signal is initiated prior to an in-containment pressure of 30 psia.)
2	Flowrate	IRS subsystem flow rate to RS headers	Condition IV • LOCA • MSLB • FWLB • REA	Min: 3300 gpm [25.4.2] [25.4.4] [25.4.72] [24.2.1] [25.4.139] [24.2.17]	Containment Safety Analyses (see Table 11.1-5, item 2 and Sec. 2.2.1)	IRS pump flow rate used in safety analyses. This flow rate also is used in the RS pumps NPSH analysis [25.4.2] [25.4.137] A flow greater than that used in the NPSH analysis could reduce available NPSH and impact system operation (Sec. 11.1.4 and [24.6.1]). Note that to ensure that this flowrate is available, NPSH and head requirements are addressed (see Chapter 12.) Furthermore, a containment sump screen mesh opening size that prevents passage of particles larger than 0.25 inches in diameter (to prevent RS nozzle blockage) and a sump screen surface area of 168 sq ft (to reduce sump debris transport) are design features provided to ensure that the IRS flowrate remains in accordance with the safety analyses. [25.4.4] [25.8.58] [25.12.2] [25.4.1..] [25.4.137]
3	Flowrate	ORS subsystem flowrate to RS headers	Condition IV • LOCA • MSLB • FWLB • REA	Min: 3640 gpm [25.4.2] [25.4.4] [25.4.72] [24.2.1] [25.4.139] [24.2.17]	Containment Safety Analyses (see Table 11.1-5, item 2 and Sec. 2.2.1)	ORS pump flow rate used in safety analyses. (Remarks provided for IRS subsystem flowrate are also applicable to ORS subsystem.)

DC 90-13-1, Appendix 4-9, Page 136

Table 6.1-1. Key System Parameters, Safety-Related Functions, RS System

(A) Item No.	(B) System Parameter	(C) Function	(D) Operating Condition	(E) Design Requirement	(F) Reference	(G) Remarks
4	Flowrate	Casing cooling fluid injection into ORS pump suction	Condition IV • LOCA • MSLB • FWLB • REA	Min: 800 gpm [25.4.2] [25.4.4] [24.4.72] 25 [25.5.1]	ORS Pump NPSH Analysis (see Table 11.1-5, Item 13 and Sec. 2.2.1)	To support ORS Pump NPSH requirements, the casing cooling pumps provide cooling flow to the pumps' suction. [24.6.2] Note that since casing cooling flow is available, credit is taken for it in the containment analyses. (The QS System provides the cooling flow required at the IRS pumps' suction to support the IRS pump NPSH requirements.)
5	Temperature	CCT temperature	All Conditions except • Cold Shutdown • Refueling	Max: 50°F [25.4.2] [25.4.4] [24.4.72] 25 [25.5.1] [24.2.5]	ORS Pump NPSH Analysis (see Table 11.1-5, Item 11 and Secs. 2.2.3 and 2.2.1)	The plant Technical Specifications ensure that the CCT temperature is maintained below 50°F to support the ORS Pump NPSH requirements. Also see Item 1, Table 6.3-1. (Note that since casing cooling flow is available, credit is taken for it in the containment analyses.)
6	Volume	CCT volume	All Conditions except • Cold Shutdown • Refueling	Min: 96,000 gal [25.4.2] [25.4.4] [25.4.72] [25.5.1] [24.2.5]	ORS Pump NPSH Analysis (see Table 11.1-5, Item 10 and Secs. 2.2.3 and 2.2.2)	Useable volume of water in the CCT required to support ORS Pump NPSH requirements. Casing cooling tank volume is maintained by the plant Technical Specifications at a fixed volume greater than this design requirement. Increasing the volume of available fluid in the casing cooling tank to greater than that specified by Technical Specifications could impact ultimate sump pH and maximum flood level in containment. (Note that since casing cooling flow is available, credit is taken for it in the containment analyses.)

[25.4.139]
[24.2.17]

DC 90-13-1, Appendix 4-9, Page 137

Table 6.1-1. Key System Parameters, Safety Related Functions, RS System

(A) Item No.	(B) System Parameter	(C) Function	(D) Operating Condition	(E) Design Requirement	(F) Reference	(G) Remarks
7	Chemistry	CCT boron concentration	All Conditions except • Cold Shutdown • Refueling	Max: 2400 ppm Min: 2300 ppm [25.4.76] [24.2.10] [24.2.9] [24.2.11] [25.4.129] [25.1.60] [25.5.8]	Support sump pH and SI System function (Table 11.1-5, item 12 and Secs. 2.2.2 and 2.2.1)	The CCT boration level is intended to be the same as the RWST. Note that the "minimum" value is based on SI System performance requirements and the "maximum" value on sump pH (or iodine removal) requirements.
8	Heat Transfer Coefficient	Heat removal capability of IRS coolers	Condition IV • LOCA • MSLB • FWLB • REA	Min: 3.55 E6 Btu/hr°F [25.4.2] [25.4.4] [25.4.72] [24.2.1]	Containment Safety Analyses (see Table 11.1-5, item 3 and Sec. 2.2.1)	Required heat removal capability of the IRS coolers to support containment analyses. (The temperature and flow rate of the tube side of the RS cooler as required by the containment analyses is documented in SDBD-NAPS-SW. [24.6.3])
9	Heat Transfer Coefficient	Heat removal capability of ORS coolers	Condition IV • LOCA • MSLB • FWLB • REA	Min: 3.65 E6 Btu/hr°F [25.4.2] [25.4.4] [25.4.72] [24.2.1]	Containment Safety Analyses (see Table 11.1-5, item 3 and Sec. 2.2.1)	Required heat removal capability of the ORS coolers. (Remarks provided for the IRS cooler heat removal capability are also applicable to the ORS subsystem.)
10	Droplet Thermal Effectiveness	Heat removal capability of RS droplets	Condition IV • LOCA • MSLB • FWLB • REA	Min: 0.9 [25.4.2] [25.4.4] [25.4.72] [24.2.1]	Containment Safety Analyses (see Table 11.1-5, item 4 and Sec. 2.2.1)	Droplet thermal effectiveness (based on droplet size, spray nozzle arrangement, and droplet fall height) assumed in safety analyses. [24.6.4] [24.2.3] [24.3.2]

[25.4.139]
[24.2.17]

DC 90-13-1, Appendix 4-9, Page 138

[25.4.139]
[24.2.17]

Table 6.1-1. Key System Parameters, Safety Related Functions, RS System

(A) Item No.	(B) System Parameter	(C) Function	(D) Operating Condition	(E) Design Requirement	(F) Reference	(G) Remarks
11	Spray Coverage Volume	Containment volume covered by RS sprays	Condition IV • LOCA • MSLB • FWLB • REA	Min: 1,401,000 ft^3 [25.4.75]	Fission product removal (see Table [illegible]-5, Item 5 and Sec. 2.2.1)	Spray coverage based on existing configuration [25.4.31] and used in control room dose analysis. Note that the containment safety analyses [25.4.2] [25.4.4] [25.4.72] and the site boundary dose analyses [25.4.3] [25.4.73] [25.4.74] assume an effective 100% of containment volume covered by recirculation spray. [24.2.4]

DC 90-13-1, Appendix 4-9, Page 139

length or the setting of the pump raised about 1 ft. [25.7.27] [24.7.1]

The casing cooling systems for Units 1 and 2 were originally designed for one chiller for each CCT. The maximum temperature to be allowed by the <u>Technical Specifications</u> was 65°F; accordingly, the chillers would be set at 55°F, which would allow approximately 8 days for the water to rise from 55° to 65°F following a chiller failure. Subsequent NPSH requirements forced the design basis temperature down to 50°F, thus requiring the chiller to be reset to 47°F, which allowed less than 2 days for the water to rise from 47°F to 50°F. Consequently, a second chiller unit was provided as backup for the casing cooling systems for both NAPS Units 1 and 2. [25.7.13] (Currently the chillers are set at 42°F.)

It should be noted that in accordance with the system classifications used by SWEC (which corresponds to the ANSI B31.7 Class II and Class III classifications) the RS System safety-related mechanical components (and piping) were designated Q2 with the following exception. The portion of the casing cooling subsystem that lay beyond the normally open casing cooling pump discharge isolation valves up to and including the CCT was designated Q3. The remainder of the system (i.e., the portion of the casing cooling subsystem that supported cooling of the CCT fluid) was designated non-safety related. [25.8.1] [25.8.2]

In 1978, NAPS Unit 1 introduced a cross-connection between the RS and SI Systems, (specifically between the ORS pump discharge point and the LHSI pump discharge) to provide a long-term backup for the LHSI in case of LHSI pump failure. [25.12.1] This connection was not intended to be a part of the design bases and was to be allowed only if reactor temperature was less than 200°F. This design change (DCP-78-01) was made only to NAPS Unit 1 and was provided as additional assurance of core cooling (in the event of an accident) prior to completion of the required long-term pump testing to demonstrate the reliability of the RS and LHSI pumps. [25.5.3] [25.10.12] [24.5.1] The LHSI pump's reliability was subsequently demonstrated, thus eliminating the need for the cross-connect. [25.5.10] Consequently, the RS System design at NAPS Unit 2 does not include the cross-connect. [25.8.1] [25.8.2]

Insert 7.1

Insert 7.1

In the late 1980's, it was noted that the SGs at NAPS-1 were experiencing corrosion-related degradation resulting in the requirement for frequent inspection, and plugging of a significant number of SG tubes. Despite improvements in secondary water chemistry, tube degradation continued to occur. In 1990, engineering efforts were initiated to replace the SG lower tube bundle assemblies at NAPS-1. The replacement SGs were to be fabricated and analyzed to standards which were at a minimum, equivalent to the existing equipment. [25.12.11]. As noted in Design Change Package DC 90-13-1 [25.12.11], the new Model 51F SG was an improvement over the old Model 51 SG and from a safety analysis perspective, could be considered a "replacement" for the Model 51.

One of the differences between the old and the replacement SGs was the thermal insulation. The new insulation was a fiberglass blanket type which exceeded the design requirements of the existing insulation. However, due to the direction received in GL 85-22 [25.1.66] analytical models (in accordance with RG 1.82, Rev. 1 [25.1.63]) had to be utilized to estimate post LOCA head loss across the containment sump screens due to insulation debris. This analysis, [25.4.138], resulted in the development of new values for the NPSH available at the ORS and IRS pump suction and a new containment LOCA analyses for NAPS-Unit 1. [24.2.17] [25.4.139]

01287.B07

Table 8.2-1. Regulatory/Safety Guides, Regulatory Guidance Documents, RS System, North Anna Power Station

Regulatory Guide	Title
1.1	"Net Positive Suction Head for Emergency Core Cooling and Containment Heat Removal System Pumps," (Safety Guide 1), November 1970. [25.1.22] [25.10.2]
1.4	"Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss-of-Coolant Accident for Pressurized Water Reactors," (Safety Guide 4), November 1970. [25.1.5] [25.10.2]
1.26	"Quality Group Classifications and Standards," (Safety Guide 26), March 1972. [25.1.25] [25.10.2]
1.46	"Protection Against Pipe Whip Inside Containment," May 1973. [25.1.47] [25.10.2]
1.48	"Design Limits and Loading Combinations for Seismic Category 1 Fluid System Components," May 1973. [25.1.48] [25.10.2]
1.64	"Quality Assurance Requirements for the Design of Nuclear Power Plants," June 1976. [25.1.54] [25.5.6]
1.97	"Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident," May 1983. [25.1.33] [25.10.2] [25.3.48]
1.82 (NAPS Unit 1)	"Water Source for Long-term Recirculation Cooling Following a Loss-of-Coolant Accident", November 1985 [25.1.63] [25.10.6] [24.8.1]

Table 8.2-4. Other Regulatory/Information Guidance Documents, Regulatory Guidance Documents, RS System, North Anna Power Station

Document	Title
NUREG-0588 (NAPS Unit 2)	"Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment," July 1981 [25.1.28] [25.5.2] [25.3.56] [25.10.28]
TID-14844	"Calculation of Distance Factors for Power and Test Reactor Sites," March 1962. [25.1.31] [25.10.7]
NUREG 0737 (NAPS Unit 2)	"Clarification of TMI Action Plan Requirements," November 1980. [25.1.23] [25.5.5]
NUREG/CR-2791 (NAPS Unit 1)	"Methodology for Evaluation of insulation Debris Effects", September 1982 [25.1.64] [25.10.6] [24.8.1]
NUREG-0897 (NAPS-Unit 1)	"Containment Emergency Sump Performance", October 1985 [25.1.65] [25.10.6] [24.8.1]

Table 8.3-1. Licensing Design Basis Requirements and Commitments. Updated Final Safety Analysis Report, RS System, North Anna Power Station

UFSAR Section No.		Requirement	
6.2.2 [25.10.6]	1.	In order to ensure long-term reliability of the RS coolers, following each periodic test, the RS coolers shall be put in dry layup.	Refer to Section 13.3.1, item 1.
	2.	The containment depressurization systems shall provide the ECCS System with water for effective core cooling on a long-term basis after a LOCA.	
6.2.2.2.2 through 6.2.2.2.5 [25.10.9]	1.	RS pumps, MOVs, and check valves shall be fabricated in accordance with Draft ASME Code for Pumps and Valves for Nuclear Power Plants (1968). [25.2.2]	Refer to Section 13.3.1, item 7.
	2.	Piping fabrication, installation and testing shall be in accordance with ANSI B 31.7 (1969). [25.2.4]	
	3.	RS coolers shall be designed to ASME Code Section III C (1968). [25.2.16]	
	4.	ORS pump casing shall be designed to ASME Code Section III B (1968). [25.2.5]	
6.2.2 [25.10.6] 15.4.1 [25.10.7] 12.1.1 [25.10.23]		The containment depressurization systems shall reduce the concentration of radioactive iodine in containment atmosphere quickly so that for any outleakage during the time the containment is above atmospheric, the resulting dose is within the limits of 10CFR100.11 [25.1.3]	Refer to Section 13.1.1, item 5.
6.2.2 [25.10.6] 15.4.1 [25.10.7]		The containment depressurization systems shall cool and depressurize the containment structure to subatmospheric pressure in less than 1 hour following a LOCA.	Refer to Section 13.3.1, item 3.
6.2.2 [2..10.6]		The insulation debris inventory and transport analysis performed for the NAPS Unit 1 containment emergency sump shall follow the guidance of 1. RG 1.82, Rev 1 2. NUREG/CR-2791 3. NUREG-0897, Rev 1	Refer to Sec. 13.2.1 item 8 Refer to Sech. 13.2.4 item i Refer to Sec 13.2.4 item E

Identified below are some of the safety analyses that established RS System design and that were necessary to ensure proper operation of the RS System during a LOCA or a MSLB.

1. Minimum RS System Flow Requirements (and associated parameters) to Support Containment Analyses

   The minimum RS and SW flow requirements are based on the accident design objectives for the Containment Building. A number of postulated primary and secondary high-energy line breaks are considered, all of which discharge high-energy fluids into the containment. The rate of heat removal from the containment via the RS, QS, and SW Systems must be of sufficient magnitude to maintain the containment peak pressure below 45 psig [25.4.1] and reduce the pressure to below atmospheric conditions within 1 hour. After 1 hour, the RS and SW Systems must remove sufficient heat to maintain subatmospheric conditions in the containment in the long term. RS System operation in the long term is addressed in calculation [25.4.85]. The minimum RS and SW flows are calculated in [25.4.11], [25.4.12], [25.4.13], [25.4.32], [25.4.49], and [25.4.51].

   The flow rates are used in part to calculate the minimum overall heat transfer coefficient and other specifications for the RS coolers. [25.4.14] [25.4.37] [25.7.11] Throttling down the SW flows to the RS coolers in the long term is addressed in calculation [25.4.77].

   [25.4.139]
   [24.2.17]

   The calculated minimum flow rates, spray droplet thermal effectiveness, and overall heat removal coefficients are used as inputs in the containment safety analyses. [25.4.2] [25.4.4] [25.4.72] [24.2.1] The containment safety analyses confirm the acceptability of these parameters and thereby establish their minimum allowable value. It has been determined in [25.4.20] that, after 24 hours (with minimum safeguards), one IRS pump is sufficient to remove decay heat and the remaining pump (ORS) may be shut down.

2. Maximum RS and SW Flow Requirements

   There are no documented requirements on maximum RS flow rates. However, the use of an RS flow rate higher than that addressed in the NPSH analysis may create a situation where the RS pumps could operate with insufficient NPSH. [24.6.1]

The safety analyses do not impose a requirement on the maximum SW flow since any flow greater than that used in the "minimum flow" safety analyses would reduce containment temperature and pressure at a faster rate.

3. RS Pump NPSH Analysis

In addition to the containment analyses (see item 1) and as part of the system's safety analyses, NPSH analyses are performed on the RS pumps. [25.4.2] [25.4.139] These analyses determine that minimum available NPSH is adequate to ensure that the RS System can accomplish its intended function. These analyses perform a spectrum of sensitivity studies that address the following:

a. Break location
b. Break size
c. Minimum/normal ESF
d. SW temperature
e. RWST temperature

RS pump NPSH analyses are performed for the LOCA. [25.4.2] [25.4.138] [25.4.139] [24.2.5] To ensure proper operation of the RS System, the RS pumps are delayed to allow water to accumulate on the containment floor. [25.4.33] [25.4.43] [25.4.81] [25.12.2] [24.3.1] A delay time of approximately 3.5 minutes was assumed from the time the containment "high-high" pressure signal is reached to pump startup. The NPSH analyses verified that the combination of the minimum containment atmosphere pressure and the water accumulation in the sump at 3.5 minutes following a CDA was sufficient for the purpose of meeting the RS pumps' NPSH requirements. The containment analyses (see item 1) verify that this delay (which corresponds to a spray effective time of 5 minutes [25.4.111] [24.3.1] after the CDA signal and accounts for system fill time) [25.4.56] is acceptable and meets the required performance necessary in the containment safety analyses. [25.4.81] [24.3.1]

Some of the early NPSH analyses performed on the RS pumps showed that the available NPSH was less than the required NPSH. Consequently, both hardware changes and a series of analyses referred to as "NPSH fixes" were required. Results of these analyses are discussed in detail in reference [25.5.1].

[25.4.139]

The NPSH analysis [25.4.2] established a second set of requirements for flow and a dedicated water storage capacity in order to achieve adequate NPSH results. The IRS pumps require a minimum bleed flow of 150 gpm per pump at a maximum temperature of 50°F from the QS System. [25.4.23] This increases the available NPSH by decreasing the water temperature entering the IRS pump suction. The ORS pumps require a much higher flow of 800 gpm per pump at a maximum temperature of 50°F to satisfy NPSH requirements. [25.4.24] [25.4.25] [25.4.53] [25.4.57] [25.4.61] [25.4.82] [25.4.83] [24.6.2]. The water for the ORS pumps is taken from a dedicated casing cooling tank with a required volume of 96,000 gallons, the impact of which is included in the containment analyses. [25.4.2] [25.4.4] [25.4.72] The sizing of the chiller units required to maintain the casing cooling tank temperature is documented in calculation [25.4.35].

[25.4.139]
[24.2.17]

In addition, the availability of sump water to support RS pump operation is analyzed for the LOCA and MSLB accident. These analyses address break effluent water "lost" in the reactor cavity as well as entrainment of containment sprays in the atmosphere or on surfaces (as a film) and calculate a "delay" in the transport time of the spray water to the sump. [25.4.127] [25.4.128] [25.4.81] It has been determined in calculation [25.4.81] [24.3.1] that there may be insufficient sump water inventory during the initial startup phase of the RS pumps during an MSLB, which could result in some RS pump cavitation. However, manufacturer's data and tests show that the period of degraded flow will not impact the long-term operation of the RS pumps.

4. RS Coverage for Heat Removal and Fission-Product Removal

The RS nozzles and nozzle headers are arranged in general to accommodate the requirements set forth in 10CFR50, Appendix A, GDC 38. The basic requirements deal with postaccident heat and fission-product removal. Spray header rings and spray nozzles are designed to maximize the sprayed containment volume. Sprayed volumes are discussed in Section 6.2.2 of the FSAR and are documented in [25.4.31] and [25.4.59]. The containment safety analyses [25.4.2] [25.4.4] [25.4.72] and the site boundary dose analyses [25.4.3] [25.4.73] [25.4.74] assume an effective 100 percent of containment volume covered by recirculation spray. The Control Room dose analysis [25.4.75]

[25.4.139]
[24.2.17]

uses the spray coverage based on existing configuration as documented in calculation [25.4.31] [24.2.4].

The RS droplet effectiveness directly controls the rate of heat removal from the containment during a postulated LOCA or MSLB. The RS droplet effectiveness depends on droplet size [25.4.30] [24.6.4] fall height, [25.4.29] and environment and spray temperature and pressure, and is assumed [24.2.3] to be 0.9 and used as input in the containment safety analyses. [25.4.2] [25.4.4] [25.4.72] [25.4.139] [24.2.17] The RS nozzle design is discussed in [25.4.34], [25.4.46], [25.8.8], and [24.2.3].

The containment sump water pH is controlled by the addition of NaOH solution via the QS System. [25.4.76] [25.7.3] [25.7.4] The NaOH addition enhances fission-product removal and provides an environment that minimizes the potential for stress corrosion cracking of components.

5. Single Failure Analysis

The RS System has been designed to withstand a single failure, as defined in Appendix A of 10CFR50.

The RS System contains two identical redundant subsystems, each powered by a separate emergency diesel generator. Each subsystem (composed of one ORS and one IRS pump) is designed to have the required performance needed for heat removal during a postulated LOCA concurrent with the most limiting single failure. The containment LOCA analysis [25.4.2] [25.4.139] establishes the most limiting failure as the loss of one train.

The single failure analyses for the MSLB [25.4.72] [24.2.17] identify the failure of one emergency bus coincident with the failure of the nonreturn valve in the ruptured steam line as the worst case scenario. While the assumption of two failures is conservative, it may not be overly conservative since the nonreturn valve failure results in additional steam releases early in the transient before the containment heat removal system starts; whereas the emergency bus failure results in a conservative prediction of the long-term pressure and temperature response. This approach simplifies the single failure analysis since the number of break size/power level/single failure combinations that must be analyzed is reduced.

The RS pump NPSH analysis establishes that full operation of the RS System (i.e., no failure) results in the most limiting or lowest available NPSH.

6. Containment Recirculation Sump Requirement

To promote proper operation of the RS System during a postulated accident, the containment and containment recirculation sumps are designed to the following requirements:

a. Sump screens designed to prevent the flow of particles larger than the diameter of the smallest RS nozzle orifice to prevent recirculation spray blockage. [25.4.30] [25.8.58]

b. Area of sump screens large enough to support low water velocity in the sump so as not to encourage debris transport and assure that the sump screens will not clog to the extent of creating a major reduction in system flow rate or causing pump cavitation. [25.4.42] [25.4.45] [25.4.56] [25.12.2] [25.4.138]

c. The minimum acceptable containment sump water level necessary for RS pump operation without cavitation or creation of a vortex core. [25.4.33] [25.4.43] [25.4.67] [25.4.68] [25.4.81] [25.12.2] [25.4.13 [24.3.1] (Maximum sump water level is documented in [25.4.84].)

d. An ultimate containment sump water pH of approximately 7.7 to support iodine removal from the containment atmosphere, [25.1.8] [25.4.76] and minimize the potential of stress corrosion cracking in safety-related equipment [25.7.3] [25.7.4].

Insert A (from Pg 11-10)

It should be noted that, in general, debris formation in containment is reduced by:

a. The use of qualified paint inside containment to the extent practical to minimize the potential of sump screen blockage by dislodged paint chips. [25.6.15] [24.11.3] Surfaces painted with unqualified paint are covered with a stainless steel screen to retain the dislodged paint chips. [24.11.4]

[25.12.11]

b. The use of thermal insulation which, to a degree, resists shredding and transport to the sump screens. [25.6.16] [24.11.5] It should be noted that all exposed insulation in containment is jacketed with stainless steel or silicone rubber impregnated fiberglass cloth encased in a stainless steel mesh and is designed to withstand a post-LOCA environment. [25.5.34]

Insert 11.1.4

c. General containment design, such as the use of grating in subcompartments, which resists transport of large debris. [25.8.9]

Additionally, the containment recirculation sump is divided by a fine mesh into two redundant sumps, each sump supporting one RS subsystem. Consequently, no single failure could result in the clogging of all suction points to t[illegible]S System.

Insert A
(move to Pg 11-9)

11.2 SUPPLEMENTARY ANALYSES

This section identifies and discusses any supplementary safety analysis performed for the RS System, in addition to those analyses required by Section 11.1. Each analysis is described briefly below and is summarized in Table 11.2-1.

There are no supplementary safety analyses performed for the RS System.

11.3 HEAT BALANCE ANALYSES

This section identifies heat balance analyses that have been performed for the RS System. Each analysis is described briefly below and is summarized in Table 11.3-1.

There are no heat balance analyses applicable to the RS System.

Insert 11.1.4

The calculation that determines the available NPSH at the RS pumps for NAPS-2 does not address head loss across the sumps screens resulting from accumulation of post-accident insulation debris. [25.4.2] This calculation reflects the original licensing basis for NAPS 1 and 2 which does not include a commitment to Reg Guide 1.82. (The issue of sump blockage was addressed via experimental results. Prior to initial operation, Alden Research Labs was hired to build a model of the NAPS containment sump to ascertain the absence of hydraulic phenomena that could affect NPSH. Tests incorporating various pump and flow combinations under different possible restrictions, such as screen blockage, were undertaken. Design modifications were made as necessary to address problem areas.) However, subsequent to SG replacement at NAPS-1, and a self-imposed commitment by Virginia Power to GL 85-22 [25.1.66], an analytical approach was used to determine the head loss across the NAPS-1 emergency sump screens. The analyses uses the guidance of Reg Guide 1.82, Rev. 1, and develops a plant specific model that identifies the break that potentially generates the largest quantity of debris, determines the portion of the debris that reaches the sump screens and determines the head loss across the sump screen due to this accumulation of debris. [25.4.138] This head loss is then factored into the RS pump NPSH analysis for NAPS-1. [25.4.139]

01287.B07

Table 11.1-5. Plant Safety Analysis Parameters, RS System, Condition IV Events - Limiting Faults, Event: LOCA, MSLB

	Parameter		*Reason*
1.	Containment "High-High" Pressure Signal Setpoint Containment "high-high" pressure signal setpoint shall be such that the CDA signal is initiated prior to an in-containment pressure of 30 psia.	Why	This pressure setpoint is used in the plant safety analyses [25.4.2] [25.4.139] [24.2.17] [25.4.4] [25.4.72] [25.7.41] [24.2.1] to start the "delay" timers associated with ORS and IRS pump startup, initiate casing cooling injection, and activate all RS System MOVs.
		Origin	A pressure setpoint of 24.7 psia is used in the safety analyses. However (as documented in the analyses) it has been determined that a CDA initiating pressure of 30 psia is acceptable.
		Impact	A change in the CDA setpoint may impact the safety analyses depending on the magnitude of the change. Consequently, any change in the setpoint would have to be evaluated for impact on the safety analyses. Setpoint changes should also address the need to avoid spurious alarms.

Table 11.1-5. Plant Safety Analysis Parameters, RS System, Condition IV Events - Limiting Faults, Event: LOCA, MSLB

	Parameter		Reason
2.	RS System Flow Rate The minimum IRS flow shall be 3300 gpm. The minimum ORS flow shall be 3640 gpm. [25.4.139] [24.2.17]	Why	These are the minimum required flow rates that satisfy the plant safety analyses [25.4.2] [25.4.4] [25.4.13] [25.4.14] [25.4.72] [25.4.77] [24.2.1] design basis of depressurizing containment within 1 hour, maintaining containment pressure at less than 45 psig, and maintaining subatmospheric conditions after 1 hour.
		Origin	The flow rate originated as an assumption that was verified by the performance of the containment safety analyses.
		Impact	The flow rate directly impacts the heat removal rate from containment. A decrease in the flow rate could result in not meeting the design basis objectives. It should be noted that the heat removal rate is dependent on the heat transfer coefficient of the RS coolers, which is in part also a function of SW flow (see the SDBD for the SW System).

Table 11.1-5. Plant Safety Analysis Parameters, RS System, Condition IV Events - Limiting Faults, Event: LOCA, MSLB

	Parameter		Reason
3.	RS Cooler Heat Transfer Coefficient The minimum containment heat transfer capability of an IRS heat exchanger shall be 3.55 E6 Btu/hr-°F. The minimum heat removal capability of an ORS heat exchanger shall be 3.65 E6 Btu/hr-°F.	Why	These values represent the minimum required heat removal capability to satisfy the plant safety analyses' [25.4.2] [25.4.4] [25.4.72] [24.11.6] [25.4.139] [24.2.1] [24.2.1] design basis of maintaining containment pressure below 45 psig, depressurizing containment within 1 hour, and maintaining subatmospheric conditions after 1 hour.
		Origin	The heat removal capability of the RS heat exchangers originated as an assumption and was verified by the performance of the safety analyses.
		Impact	The heat removal rate from the containment directly impacts the system's design basis objectives. It should also be noted that the RS cooler heat transfer coefficient impacts sump water temperature, which could affect SI System performance (i.e., available NPSH at LHSI pump suction and temperature of fluid used for emergency core cooling in the recirculation phase). [24.11.7]

Table 11.1-5. Plant Safety Analysis Parameters, RS System, Condition IV Events - Limiting Faults, Event: LOCA, MSLB

	Parameter		Reason
4.	RS Droplet Thermal Effectiveness The minimum IRS and ORS droplet thermal effectiveness shall be 0.9.	Why	This value of spray effectiveness was used in the containment safety analyses to account for the thermal effectiveness of the spray. [25.4.2] [25.4.4] [25.4.72] [24.2.1] [25.4.139] [24.2.17]
		Origin	An assumed conservative value based on the arrangement of the spray nozzles, droplet size, total spray coverage, pressure drop across the spray nozzle, and droplet fall height. [25.4.34] [25.4.46] [25.8.8] [24.2.3] [24.6.4] [25.4.30]
		Impact	A change in the spray droplet effectiveness impacts the rate of containment heat removal which, in turn, impacts the calculated containment peak temperature, pressure, depressurization time, and maintenance of subatmospheric pressure in containment. Sump water temperature is also impacted, which could affect SI System performance (i.e., available NPSH at LHSI pump suction and temperature of fluid used for emergency core cooling in the recirculation phase). [24.11.7]

Table 11.1-5. Plant Safety Analysis Parameters, RS System, Condition IV Events - Limiting Faults, Event: LOCA, MSLB

Parameter		Reason	
6.	RS Effective Time A full-flow recirculation spray shall be available in containment within 300 seconds after the CDA signal. (To support this feature, from the time CDA is reached, the IRS pump start is delayed by 195 seconds, and the ORS pumps are delayed by 210 seconds.)	Why	RS pump starts must be delayed to allow time for water to accumulate in the containment recirculation sump to avoid pump cavitation at startup. This time delay is therefore critical to both the containment safety analyses and pump NPSH analysis.
		Origin	The safety analyses assume a delay of 5 minutes between the CDA signal and the time when the spray becomes effective. The delay time accounts for diesel start time, diesel sequencing time, valve operating times, pump up-to-speed time, and the system fill time. [25.4.2] [25.4.4] [25.4.33] [25.4.43] [25.4.56] [25.4.139] [25.4.67] [25.4.68] [25.4.72] [24.2.17] [25.4.80] [25.4.81] [25.4.111] [25.4.138] [25.12.2] [24.2.1] [24.3.1] [24.11.8] [24.11.10]
		Impact	This parameter is critical to the results of the plant safety analyses and the RS pump NPSH analysis.

Table 11.1-5. Plant Safety Analysis Parameters, RS System, Condition IV Events - Limiting Faults, Event: LOCA, MSLB

	Parameter		Reason
7.	Containment Recirculation Sump Screen Size Total sump screen surface area shall be at least 168 sq. ft.	Why	The sump screen area needs to be large enough to support low sump water velocity and discourage debris transport. [25.4.42] [25.4.58]
		Origin	Sump screen area was based on just enclosing the sumps. Acceptability of sump screen area was based on actual hydraulic model studies/tests done using a model of the NAPS Units 1 and 2 reactor containment sumps. [25.4.42] [25.12.2]
		Impact	Changes in the above parameters could impact performance of the RS pumps.

To address SG replacement at NAPS-1, a debris transport analysis was performed which used the existing sump design as input [25.4.138]

Table 11.1-5. Plant Safety Analysis Parameters, RS System, Condition IV Events - Limiting Faults, Event: LOCA, MSLB

Parameter			Reason
9.	RS Pump NPSH The NPSHA to the ORS pumps shall exceed the NPSHR of 11.0 ft. The NPSHA to the IRS pumps shall exceed the NPSHR of 9.4 ft.	Why	Verification that NPSHA is greater than NPSHR is essential to ensure that the RS System performance is in accordance with the assumptions in the safety analyses.*
		Origin	Safety Guide 1 requires that the RS pumps' NPSH be verified by analyses. Consequently, NPSH verification analyses are performed as part of the LOCA safety analysis. [25.4.2] [25.4.23] [25.5.1] [24.2.5] [25.4.139]
		Impact	Failure to have sufficient NPSH could cause pump damage or less flow than required by the plant safety analyses.

* To ensure that the RS pumps' NPSHA is greater than the NPSHR, the QS System provides cooling water to the suction of the IRS pumps and the casing cooling system provides cooling water to the ORS pump suction. The required injection fluid temperature and flow rate for the ORS and IRS pumps is documented in the NPSH analysis and in the SDBD-NAPS-RS and SDBD-NAPS-QS, respectively.

DC 90-13-1, Appendix 4-C, Page 158

Table 11.1-5. Plant Safety Analysis Parameters, RS System, Condition IV Events - Limiting Faults, Event: LOCA, MSLB

Parameter		Reason	
10.	Usable Volume of Fluid in the Casing Cooling Tank Minimum usable volume of fluid in the CCT shall be 96,000 gallons.	Why	Casing cooling is required to ensure proper operation of the ORS pumps. Without this cooling, the ORS pumps may not have sufficient available NPSH.
		Origin	The system is sized to provide flow for approximately 1 hour. [25.4.2] [25.4.139]
		Impact	The ORS pump NPSH analysis requires casing cooling fluid injection to support proper performance of the ORS pumps during a postulated accident. [25.4.2] [25.4.4] [25.4.53] [25.4.57] [25.4.72] [25.4.82] [25.4.93] [25.4.110] [25.5.1] [25.4.139] [24.2.17] It should also be noted that credit is taken for casing cooling flow when evaluating containment pressures and temperatures following a LOCA or MSLB and in supporting SI System performance (i.e., available NPSH at LHSI pump suction and temperature of fluid used for emergency core cooling during the recirculation phase). [24.11.7]

Table 11.1-5. Plant Safety Analysis Parameters, RS System, Condition IV Events - Limiting Faults, Event: LOCA, MSLB

	Parameter		Reason
11.	Casing Cooling Fluid Temperature Maximum temperature of the casing cooling fluid shall be 50°F.	Why	Casing cooling is required to ensure proper operation of the ORS pumps. Without this cooling, the ORS pumps may not have sufficient available NPSH.
		Origin	The required temperature of the casing cooling fluid is calculated in the ORS pumps' NPSH analysis. [25.4.2] [25.4.139]
		Impact	The ORS pump NPSH analysis fixes the maximum temperature of the casing cooling fluid to ensure proper performance of the ORS pumps during a postulated accident. [25.4.2] [25.4.4] [25.4.35] [25.4.53] [25.4.139] [24.2.17] [25.4.57] [25.4.72] [25.4.82] [25.4.83] [25.5.1] It should also be noted that credit is taken for casing cooling flow when evaluating containment pressures and temperatures following a LOCA or MSLB and in supporting SI System performance (i.e., available NPSH at LHSI pump suction and temperature of sump fluid used for emergency core cooling during the recirculation phase). [24.11.7]

Table 11.1-5. Plant Safety Analysis Parameters, RS System, Condition IV Events - Limiting Faults, Event: LOCA, MSLB

	Parameter		Reason
13.	Casing Cooling Injection Flow Rate Casing cooling flow to each ORS suction point shall be a minimum of 800 gpm.	Why	Casing cooling is required to ensure proper operation of the ORS pumps. Without this cooling, the ORS pumps may not have sufficient available NPSH.
		Origin	The required flow of the casing cooling fluid is calculated in the ORS pump NPSH analysis. [25.4.2] [25.4.139]
	[25.4.139] [24.2.17]	Impact	The ORS pump NPSH analysis fixes the minimum casing cooling fluid injection flow rate to ensure proper performance of the ORS pumps during a postulated accident. [25.5.1] [25.4.53] [25.4.57] [25.4.2] [25.4.4] [25.4.82] [25.4.83] [25.4.61] [25.4.72] [24.6.2] It should be noted that credit is taken for casing cooling flow when evaluating pressures and temperatures in containment following a LOCA or MSLB and in supporting SI System performance (i.e., available NPSH at LHSI pump suction and temperature of sump fluid used for emergency core cooling in the recirculation phase). [24.11.7]

E. Design Requirement - The minimum/maximum design value for the parameter is entered as the "design requirement." This value is normally based directly upon the assumptions, input, or results of the safety analysis or is derived therefrom. Where the value of a design requirement is dependent upon the value of another parameter, the dependency is explained in the "Remarks" column and the dependent parameter listed as a parameter.

F. Design Specification - The value established by the designer for the parameter in the design specification (NAS, NUS, etc.) is recorded.

G. Design Margin - The design margin available for this parameter is presented. Design margin is the difference between the "design requirement" and the "design specification."

H. Installed Value - The actual value for the parameter, specific to the installed component, is recorded. This value must be based on a substantiated source that conclusively demonstrates the stated installed value.

The difference between the "installed value" and the "design specification" represents the performance margin.

I. Total Margin - Total margin is the difference between "design requirements" and "installed value." The total margin is dependent upon the actual installed component and is subject to change if the component is replaced or modified.

J. Reference - References to the applicable sections of this SDBD that contain the reason or basis are provided. Reasons and references for margins may be presented here.

K. Remarks - This column is used to record any pertinent remarks not included elsewhere.

Subsequent sections of this chapter provide discussion of margins presented in Table 12.0-1. A subsection is presented for each major component identified in Table 12.0-1. Key parameters for the component are listed as separate items in the sections.

12.1 IRS PUMP FLOW RATE

[25.4.139] [24.2.17]

for Unit 2 and 10.4 ft for Unit 1.

Design Requirement - The IRS pumps shall be capable of delivering a flow rate of 3300 gpm against a head of 273 ft [25.4.2] [25.4.4] [25.4.72] [24.2.1] [25.4.49] [25.4.11] and a minimum available NPSH of 11.9 ft. (Note that the minimum available NPSH is based on 150 gpm of QS fluid being injected into the IRS pump suction.) [25.4.2] [24.12.1] [25.4.139]

Design Specification - The IRS pumps are purchased to deliver 3300 gpm against a head of 275 ft (see pump curves attached to [25.4.49]) and an NPSH of 9.4 ft. [25.5.1]

Subsequent to SG replacement at Unit 1, a calculation determining head loss across the sump screens due to accumulated debris was performed for Unit 1. [25.4.138] The calculated head loss is included in the NPSH analysis for Unit 1 [25.4.139]

(It should be noted that the RS System design includes 150 gpm of QS fluid being injected into the IRS pump suction. [25.8.1] [25.8.2] [25.3.6]) Furthermore, to ensure that the above flow rate is available without blockage, the containment sump screen mesh opening is made smaller than the diameter of the RS nozzles. [25.8.58] [24.2.6] ~~and~~ a hydraulic "sump model" test ~~is~~ was conducted (prior to plant startup) to determine acceptability of sump design and evaluate the blocking effect on RS flow due to debris transport in the sump. [25.12.2]

Margin Result - There can be no margin on the flow rate since the configured flow rate is the basis of the safety analyses. [24.6.1] The containment analyses and the resultant design of the containment depressurization systems follow an iterative process that relies on previous SWEC experience in accident analyses. As a result, the final, documented containment analyses of record use as input the actual design parameters of the containment depressurization systems, and the acceptability of the results of the containment analyses establishes the acceptability of that combination of RS/QS System design parameters. Consequently, there can be no margin for parameters like RS and casing cooling flowrate, minimum droplet thermal effectiveness, and containment spray coverage.

Margin evaluation for each of the above parameters relative to the results (i.e., peak pressure, depressurization time, etc.) of the containment analyses (assuming different combinations of the remaining parameters whose individual values may vary within a range) is not addressed in this SDBD since it involves several systems (e.g., QS, SI, RS, containment structure, etc.). Furthermore, no documentation was found that addressed the above issue.

However, at a flow rate of 3300 gpm, the purchased pump head exceeds the required flow rate by 2 ft. Also, total margin on NPSH is 2.5 ft for Unit 2 and 1 ft for Unit 1. No design margin was specified. [25.4.139] [24.2.17]

12.2 ORS PUMP FLOW RATE

Design Requirement - The ORS pumps shall be capable of delivering a flow rate of 3640 gpm against a head of 285 ft [25.4.2] [25.4.4] [25.4.72] [24.2.1] [25.4.51] [25.4.12] and a minimum available NPSH of 16.8 ft for Unit 2 and 15.3 ft for Unit 1. (Note that the minimum available NPSH is based on 800 gpm of casing cooling fluid being injected into the ORS pump suction. Refer to Section 12.3.) [25.4.2] [24.12.1] [25.4.139]

Design Specification - The ORS pumps are purchased to deliver 3640 gpm against a head of 290 ft (see pump curves attached to [25.4.51]) and an NPSH of 11 ft. [25.5.1] For additional information refer to the Design Specification portion of Section 12.1 and the entire Section 12.3.

Margin Result - There can be no margin on the flow rate, since the configured flow rate is the basis of the safety analyses. [24.6.1] (For additional information relative to margin, refer to the Margin Result portion of Section 12.1.) However, at a flow of 3640 gpm, the for Unit 2 and 4.3 ft for Unit 1

purchased pump head exceeds the required flow rate by 5 ft. Also, total margin on NPSH is 5.8 ft. No design margin was specified.

X

12.3 CASING COOLING PUMP FLOW RATE

Design Requirement - The casing cooling pumps shall be capable of delivering a flow rate of 960 gpm (800 gpm to ORS pump injection, 160 gpm to recirculate back to the casing cooling tank) against maximum head of 101 ft [25.4.53] [25.4.57] [25.4.82] [25.4.83] and a minimum available NPSH of 32.7 ft (Unit 1) and 33.1 ft (Unit 2). [25.4.110] The safety analyses assume that an 800 gpm injection flow is available at the ORS pump suction immediately after receipt of a CDA signal and continues for approximately 1 hour. [25.4.4] [25.4.2] [25.4.72] [25.4.139] [24.2.17] X

Design Specification - The casing cooling pumps are purchased to deliver 960 gpm against a head of 101 ft (see pump curves attached to [25.4.53] [25.4.57] [25.4.82] [25.4.83]) and an NPSH of 8.3 ft. [25.4.110] The pumps are actuated by a CDA signal and are capable of delivering 800 gpm within 610 seconds following a LOCA. [25.4.61] [24.6.2]

Margin Result - There can be no margin on the flow rate since the configured flow rate is the basis of the safety analyses. (For additional information relative to margin, refer to the Margin Result portion of Section 12.1.) [24.6.2]. The orifice in the line is sized to support a casing cooling flow rate of 800 gpm against a head of 101 ft at 610 seconds. Also, total margin on NPSH is 24.4 ft (Unit 1) and 24.8 ft (Unit 2). No design margin was specified.

12.4 CASING COOLING TANK FLUID TEMPERATURE

Design Requirement - The CCT fluid temperature shall be maintained below 50°F. [25.4.4] [25.4.2] [25.4.72] [25.4.139] [24.2.17] X

Design Specification - The CCT instrumentation, recirculation pumps, and refrigeration units are designed such that the tank fluid temperature is automatically maintained between 42°F and 45°F. [25.3.47] (The temperature of the tank fluid is also maintained between 35°F and 50°F via the plant Technical Specifications [25.3.11] [25.3.12].)

Margin Result - A design margin of 5°F is provided.

12.5 CASING COOLING TANK USABLE FLUID VOLUME

Design Requirement - The CCT shall contain a usable volume of 9?,?00 gallons of water to be injected into the ORS pump suction line. [25.4.4] [25.4.2] [25.4.72] [25.4.139] [24.2.1 X

Design Specification - The volume of water in the tank is maintained via the plant Technical Specifications [25.3.11] [25.3.12] at 116,500 gallons. The volume of water required in the CCT to ensure a usable volume of 96,000 gallons is 113,904 gallons. [25.4.110]

Margin Result - design margin of 2596 gallons is provided.

12.6 CASING COOLING TANK FLUID CHEMISTRY

Design Requirement - The CCT fluid shall have a boron concentration between 2300 and 2400 ppm. [25.4.76] [25.4.129] [24.2.11] [25.1.60] [25.5.8]

Design Specification - The plant Technical Specifications [25.3.11] [25.3.12] ensure that the boron concentration in the CCT is maintained between 2300 ppm and 2400 ppm. [24.12.2]

Margin Result - Maintenance of the Technical Specifications [24.12.2] is consistent with Calculation [25.4.76] and an ultimate postaccident sump pH of 7.7 to support iodine removal and a minimum sump boron concentration to support core cooling by the SI System.

12.7 IRS COOLER HEAT TRANSFER COEFFICIENT

Design Requirement - The IRS coolers shall each have a minimum heat transfer coefficient of 3.55 E6 Btu/hr-°F. [25.4.4] [25.4.2] [25.4.72] [24.2.1] [25.4.139] [24.2.17]

Design Specification - The IRS coolers are purchased to a minimum heat transfer coefficient of 3.82 E6 Btu/hr-°F assuming zero fouling resistance. [25.6.9] [24.9.10]

Margin Result - An apparent design margin of 0.27 E6 Btu/hr-°F is provided. However, since tube fouling is inevitable, periodic cleaning of the tubes is required to maintain the required value of 3.55 E6 Btu/hr-°F. [24.11.6]

12.8 ORS COOLER HEAT TRANSFER COEFFICIENT

Design Requirement - The ORS coolers shall each have a minimum heat transfer coefficient of 3.65 E6 Btu/hr-°F. [25.4.4] [25.4.2] [25.4.72] [24.2.1] [25.4.139] [24.2.17] x

Design Specification - The ORS coolers are purchased to a minimum heat transfer coefficient of 3.82 E6 Btu/hr-°F assuming zero fouling resistance. [25.6.9] [24.9.10]

Margin Result - An apparent design margin of 0.17 E6 Btu/hr-°F is provided. However, since tube fouling is inevitable, periodic cleaning of the tubes is required to maintain the required value of 3.65 E6 Btu/hr-°F. [24.11.6]

12.9 RECIRCULATION SPRAY EFFECTIVE TIME

Design Requirement - The RS System shall be capable of producing a full-flow recirculation spray in containment within 300 seconds following a CDA signal (which shall be activated at a containment pressure below 30 psia). [25.4.2] [25.4.4] [25.4.72] [24.2.1] [25.4.139] [24.2.17]

Design Specification - Assuming the availability of an adequate supply of sump water [24.3.1], the IRS and ORS subsystems will deliver a full-flow recirculation spray within 255.3 seconds and 292 seconds, respectively, following a CDA signal (which is activated at a containment pressure setpoint of 27.75 + 1.5 psia). [25.4.111] (Refer to Technical Specification Table 3.3-4 for CDA signal setpoint.) [25.3.11] [25.3.12] [25.3.47]

Margin Result - A design margin of 44.7 seconds and 8 seconds is provided for the IRS and ORS subsystems, respectively. [24.11.8] [24.11.10]

12.10 RS DROPLET THERMAL EFFECTIVENESS

Design Requirement - The RS droplet thermal effectiveness shall be a minimum of 0.9. [25.4.4] [25.4.2] [25.4.72] [24.2.1] [25.4.139] [24.2.17]

Design Specification - No calculation specific to NAPS Units 1 and 2 was found documenting the bases of the assumption that the RS droplet thermal effectiveness is 0.9. Per calculation [25.4.34], the RS nozzle design and orientation appear to be based on Beaver Valley Unit 1. A calculation performed for Beaver Valley Unit 1 [25.4.46] documents droplet thermal efficiencies between 0.99 to 0.8 for a set of nozzles similar in arrangement to NAPS. [24.2.3] [24.3.2]

Margin Result - There can be no margin on RS droplet thermal effectiveness since the calculated spray droplet effectiveness is the basis of the safety analyses. For additional information relative to margin, refer to the Margin Result portion of Section 12.1.

12.11 RS CONTAINMENT COVERAGE

Design Requirement - The RS System shall produce a spray distribution that maximizes the coverage of the containment volume (in order to maximize heat and fission product removal) and shall, at a minimum, meet the requirements of the containment safety and plant dose analyses.

1. With respect to heat removal, the recirculation spray shall cover an effective 100 percent of containment volume. [25.4.2] [25.4.4] [25.4.72] [24.2.1] [25.4.139] [24.2.17]

[25.4.139] [24.2.17]

Table 12.0-1. Key Component Parameters, RS System

(A) Item No.	(B) System Parameter	(C) Operating Condition	(D) System Component	(E) Design Requirement	(F) Design Specification	(G) Design Margin	(H) Installed Value	(I) Total Margin	(J) Reference	(K) Remarks
1	Flow Rate (gpm)	Condition IV events • LOCA • MSLB • FWLB • REA	IRS Pumps (1-RS-P-1A,B) (2-RS-P-1A,B)	Min: 3300 [25.4.4] [25.4.2] [25.4.72] [25.4.49] [25.4.11]	3300 [25.6.1]	0	Not available (NA)	NA	Section 2.2.1, Table 11.1-5, Item 2	Required system flow to support containment depressurization. (See Section 12.1.)
2	Flow Rate (gpm)	Condition IV events • LOCA • MSLB • FWLB • REA	ORS Pumps (1-RS-P-2A,B) (2-RS-P-2A,B)	Min: 3640 [25.4.2] [25.4.4] [25.4.12] [25.4.51] [25.4.72]	3640 [25.6.1]	0	NA	NA	Section 2.2.1, Table 11.1-5, Item 2	Required system flow to support containment depressurization. (See Section 12.2.)
3	Flow Rate (gpm)	Condition IV events • LOCA • MSLB • FWLB • REA	Casing Cooling Pumps (1-RS-P-3A,B) (2-RS-P-3A,B)	Min: 800 [25.4.2] [25.4.4] [25.4.53] [25.4.57] [25.4.61] [25.4.72] [25.4.82] [25.4.83]	800 [25.6.8]	0	NA	NA	Section 2.2.1, Table 11.1-5, Item 13	Required flow rate to ORS pump suction to support ORS pump NPSH requirements. (See Section 12.3.)
4	Temperature (°F)	All conditions except: • cold shutdown • refueling	Casing Cooling Tanks (1-RS-TK-1) (2-RS-TK-1)	Max: 50 [25.4.2] [25.4.4] [25.4.72]	Max: 45 [25.3.47]	5	NA	NA	Section 2.2.1, Section 2.2.3, Table 11.1-5, Item 11	Required to support ORS pump NPSH requirements. (See Section 12.4.)

DC 90-13-1, Appendix 4-9, Page 167

[25.4.139] [24.2.17]

Table 12.0-1. Key Component Parameters, RS System

(A) Item No.	(B) System Parameter	(C) Operating Condition	(D) System/ Component	(E) Design Requirement	(F) Design Specification	(G) Design Margin	(H) Installed Value	(I) Total Margin	(J) Reference	(K) Remarks
5	Volume (gallons)	All conditions except: • cold shutdown • refueling	Casing Cooling Tanks (1-RS-TK-1) (2-RS-TK-1)	Min: 113,904 [25.4.110]	116,500 [25.3.11] [25.3.12]	2596	NA	NA	Section 2.2.1, Section 2.2.3, Table 11.1-5, item 10	Required to support ORS pump NPSH requirements. (See Section 12.5.)
6	Boron Concentra tion (ppm)	All conditions except: • cold shutdown • refueling	Casing Cooling Tanks (1-RS-TK-1) (2-RS-TK-1)	Min: 2300 Max: 2400 [25.4.76] [25.4.129] [24.2.11]	Min: 2300 Max: 2400 [25.3.11] [25.3.12]	0	NA	NA	Section 2.2.1, Table 11.1-5, item 12	Support sump pH and SI System function. (See Section 12.6.)
7	Heat removal coefficient (Btu/hr-°F)	Condition IV events • LOCA • MSLB • FWLB • REA	IRS Coolers (1-RS-E-1A,B) (2-RS-E-1A,B)	Min: 3.55 E6 [25.4.4] [25.4.2] [25.4.72]	3.82 E6 [25.6.9] [24.9.10]	Refer to Section 12.7	NA	NA	Section 2.2.1, Table 11.1-5, item 3	To support containment depressurization. (See Section 12.7.) Design margin is in effect an allowance for fouling.
8	Heat removal coefficient (Btu/hr-°F)	Condition IV events • LOCA • MSLB • FWLB • REA	ORS Coolers (1-RS-E-1C,D) (2-RS-E-[illegible]D)	Min: 3.6[illegible] E6 [25.4.4] [25.[illegible]2] [25.4.72]	3.82 E6 [25.6.9] [24.9.10]	Refer to Section 12.8	NA	NA	Section 2.2.1, Table 11.1-5, item 3	To support containment depressurization. (See Section 12.8.) Design margin is in effect an allowance for fouling.
9	RS effective time (seconds)	Condition IV events • LOCA • MSLB • FWLB • REA	RS System	Max: 300 [25.4.4] [25.4.2] [25.4.72]	IRS: 255.3 ORS: 292 [25.4.111]	IRS: 44.[illegible] ORS: 8	NA	NA	Section 2.2.1, Table 11.1-5, item 6	To support containment depressurization. (See Section 12.9.)

DC 90-13-1, Appendix 4-9, Page 168

Table 12.0-1. Key Component Parameters, RS System

(A) Item No.	(B) System Parameter	(C) Operating Condition	(D) System/ Component	(E) Design Requirement	(F) Design Specification	(G) Design Margin	(H) Installed Value	(I) Total Margin	(J) Reference	(K) Remarks
10	RS droplet thermal effectiveness	Condition IV events • LOCA • MSLB • FWLB • REA	RS System	Min: 0.9 [25.4.4] [25.4.2] [25.4.72] [25.4.139] [24.2.17]	0.9 [24.2.3]	0	NA	NA	Section 2.2.1, Table 11.1-5, item 4	To support containment depressurization. (See Section 12.10.)
11	RS containment coverage (cu ft)	Condition IV events • LOCA • MSLB • FWLB • REA	RS System	Min: 1,401,000 [25.4.75]	1,401,200 [25.4.31]	Refer to Section 12.11	NA	NA	Section 2.2.1, Table 11.1-5, item 5	To support heat and fission product removal. (See Section 12.11.)
12	Heat load (Btu/hr)	All conditions except: • cold shutdown • refueling	CCT Refrigeration Units (1-RS-MR-1,2) (2-RS-MR-1,2)	>33,549 [25.4.35]	72,000 [25.6.13] [24.9.13]	38,451	NA	NA	Section 2.2.3	To support CCT fluid cooling. (See Section 12.12.) Design margin is in effect an allowance to support removal of sensible heat, i.e., a temperature reduction of the tank at 0.92°F/day.

DC 90-13-1, Appendix 4-9, Page 169

7. GDC 16, "Containment Design," May 21, 1971 [25.1.50]

<u>Requirement</u> - "Reactor containment and associated systems shall be provided to establish an essentially leaktight barrier against the uncontrolled release of radioactivity to the environment and to ensure that the containment design conditions important to safety are not exceeded for as long as postulated accident conditions require."

<u>Type</u> - Licensing basis commitment. [25.10.1]

<u>Design Basis Commitment</u> - The entire GDC 16. [25.10.1]

<u>Design Basis Feature</u> - The RS System (in conjunction with the QS System) is provided to return the containment to subatmospheric pressure (thus terminating the driving force for the release of radioactivity), and maintain the containment at subatmospheric pressure for as long as the situation requires. The containment and its associated engineered safety features, therefore, meet the required functional capability of protecting the public from the consequences of gross equipment failures. [25.4.2] [25.4.4] [25.4.72] [25.4.139] [24.2.17]

8. GDC 17, "Electric Power Systems," July 7, 1971 [25.1.35]

<u>Requirement</u> - "An onsite electric power system and an offsite electric power system shall be provided to permit functioning of structures, systems, and components important to safety. The safety function for each system (assuming the other system is not functioning) shall be to provide sufficient capacity and a capability to ensure that (1) specified acceptable fuel design limits and design conditions of the reactor coolant pressure boundary are not exceeded as a result of anticipated operational occurrences, and (2) the core is cooled and containment integrity and other vital functions are maintained in the event of postulated accidents.

"The onsite electric power supplies, including the batteries and the onsite electric distribution system, shall have sufficient independence, redundancy, and testability to perform their safety functions assuming a single failure.

"Electric power from the transmission network to the onsite electric distribution system shall be supplied by two physically independent circuits (not necessarily on separate rights of way)

[25.4.139] [24.2.17]

containment depressurization systems (i.e., QS and RS) to return containment pressure to subatmospheric within 1 hour and maintain that condition. [25.4.2] [25.4.72] [25.4.4] SRP 6.2.1.1A [25.1.4], issued after NAPS Units 1 and 2 obtained their construction permits, has in its "acceptance criteria" section confirmed the NRC acceptance of the SWEC depressurization requirements for subatmospheric containments.

12. GDC 39, "Inspection of Containment Heat Removal System" July 7, 1971 [25.1.9]

    Requirement - "The containment heat removal system shall be designed to permit appropriate periodic inspection of important components, such as the torus, sumps, spray nozzles, and piping to assure the integrity and capability of the system."

    Type - Licensing basis commitment. [25.10.1]

    Design Basis Commitment - The entire GDC 39. [25.10.1]

    Design Basis Feature - Equipment comprising the RS System is situated so that periodic physical inspections can be made. All RS equipment can be inspected during planned refueling shutdowns. [25.8.8] [25.8.9] [25.8.57] [25.8.61] [25.8.62] (Refer to Sections 14.1.20 and 14.1.21.)

13. GDC 40, "Testing of Containment Heat Removal System," May 21, 1971 [25.1.10]

    Requirement - "The containment heat removal system shall be designed to permit appropriate periodic pressure and functional testing to assure (1) the structural and leaktight integrity of its components, (2) the operability and performance of the active components of the system, and (3) the operability of the system as a whole, and, under conditions as close to the design as practical, the performance of the full operational sequence that brings the system into operation, including operation of applicable portions of the protection system, the transfer between normal and emergency power sources, and the operation of the associated cooling water system."

    Type - Licensing basis commitment. [25.10.1]

    Design Basis Commitment - The entire GDC 40. [25.10.1]

Type - Licensing basis commitment. [25.10.1]

Design Basis Commitment - The entire GDC 41. [25.10.1]

Design Basis Feature - Containment depressurization systems are provided to control fission products generated by a DBA. These systems are sufficiently redundant to withstand a single failure and are operable with either onsite or offsite power.

The caustic sprays from the QS System remove radioactive and particulate fission products from the containment atmosphere by dissolving and transferring the airborne iodine to the sump. The RS System continues this washing action by recirculating this caustic sump water and spraying it back to the containment atmosphere in the long term. [25.8.1] [25.8.2]

15. GDC 50, "Containment Design Bases," May 21, 1971 [25.1.18]

Requirement - "The reactor containment structure, including access openings, penetrations, and the containment heat removal system shall be designed so that the containment structure and its internal compartments can accommodate, without exceeding the design leakage rate and with sufficient margin, the calculated pressure and temperature conditions resulting from any loss-of-coolant accident. This margin shall reflect consideration of (1) the effects of potential energy sources that have not been included in the determination of the peak conditions, such as energy in steam generators and as required by § 50.44 energy from metal-water and other chemical reactions that may result from degradation but not total failure of emergency core cooling functioning, (2) the limited experience and experimental data available for defining accident phenomena and containment responses, and (3) the conservatism of the calculation model and input parameters."

Type - Licensing basis commitment. [25.10.1]

Design Basis Commitment - The entire GDC 50. [25.10.1]

Design Basis Feature - The RS System is designed so that the containment structure can accommodate, without exceeding the design leakage rate and with sufficient margin, the calculated pressure and temperature conditions resulting from a LOCA. [25.4.2] [25.4.1] [25.4.4] [25.8.1] [25.8.2]

[25.4.139]

X

Design Basis Feature - The RS System design follows the guidance of Safety Guide 1 with the exception that the NPSH analyses include conservative predictions of the containment atmosphere pressure and sump water temperature transients. [25.4.2] [25.4.139]

2. Safety Guide 4, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss-of-Coolant Accident for Pressurized Water Reactors," November 1970 [25.1.5]

Requirement - Section 50.34 of 10CFR50 requires that each applicant for a construction permit or operating license provide an analysis and evaluation of the design and performance of structures, systems, and components of the facility. This analysis is provided to assess the risk to public health and safety resulting from operation of the facility. The design basis LOCA is one of the postulated accidents used to evaluate the adequacy of these structures, systems, and components with respect to the public health and safety. This safety guide gives acceptable assumptions that may be used in evaluating the radiological consequences of this accident for a pressurized water reactor.

Type - Licensing basis commitment. [25.10.2]

Design Basis Commitment - The entire Safety Guide 4 except Regulatory Position C.1.a. [25.10.2]

Design Basis Feature - Parameters pertaining to the RS System used for evaluating the potential radiological consequences of a LOCA are consistent with Safety Guide 4 except for Regulatory Position C.1.a.

In calculating the potential thyroid dose, Regulatory Position C.1.a suggests use of the assumption that the released iodine be 85 percent in elemental form, 5 percent in particulate form, and 10 percent in organic iodides. The potential thyroid dose was calculated with the assumption of 90 percent of the iodine in elemental form and 10 percent in organic iodides form. [25.4.3] [25.4.73] [25.4.74] Since it is unlikely that 10 percent of the iodine could be converted to the organic form within the 60-minute depressurization time and there is considerable question as to the amount of aerosols that would be present and unaffected by sprays, the assumption of 10 percent nonremovable radioiodine is conservative.

identified in Section 3.2 of ANSI N45.2.11 and applicable to the RS System design are listed in Section 2.3 and further discussed in Section 14 of this SDBD.

7. RG. 1.97, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident," Rev. 3, May 1983 [25.1.33]

<u>Requirement</u> - RG 1.97 describes a method acceptable to the NRC staff for complying with the Commission's regulations to provide instrumentation to monitor plant variables and systems during and following an accident in a light-water-cooled nuclear power plant.

<u>Type</u> - Licensing basis commitment. [25.10.2]

<u>Design Basis Commitment</u> - The entire RG 1.97 with exceptions as listed in the design basis feature section. [25.3.48] [24.2.12] [25.10.2]

<u>Design Basis Feature</u> - Except as noted, the RS System design includes all the instrumentation identified in RG 1.97 [24.2.16] (i.e., sump level and temperature indicators). Credit is taken for RS pump discharge pressure indicators as alternate instrumentation in place of the required containment spray flow indicators. [25.8.1] [25.8.2], [25.3.48] [24.2.12]

Insert 13.2.1

13.2.2 Standard Review Plan and Branch Technical Positions

This subsection identifies Standard Review Plan (SRP) and Branch Technical Positions (BTP) commitments that are an integral part of the design basis of the RS System. The NRC issues an SRP (NUREG 0800) that establishes guidelines for preparing and reviewing a Final Safety Analysis Report. The NRC has established positions on certain subjects that are documented in the SRP and identified as BTPs. If a utility commits to comply with the requirements of the SRPs or BTPs in the licensing basis, the requirements become a part of the plant's design basis.

The SRPs and BTPs applicable to the design basis of the RS System are identified below:

Insert 13.2.1

8. RG 1.82, "Water Sources for Long-Term Recirculation Cooling Following a Loss-of-Coolant Accident," Rev. 1, November 1985. [25.1.63]

   Requirement - This guide provides both design recommendations and analytical methods acceptable to the NRC staff for the design, fabrication and testing of [25.1.63]:

   - the sumps performing the water source function for the emergency core cooling system and the containment heat removal system.

   - the suction inlet conditions for the pumps associated with the above systems.

   Type - Licensing basis commitment. [25.10.6] [24.8.1]

   Design Basis Commitment - The insulation debris inventory and transport analysis performed on the Unit 1 containment emergency sump, (to evaluate the insulation installed on the replacement SG's), follows the guidance of RG 1.82. [25.10.6] [24.8.1]

   Design Basis Feature - Subsequent to the direction received in GL 85-22 and as a result of SG replacement at NAPS-1, analytical models, in accordance with the guidance of RG 1.82, Rev. 1 were used to estimate the head loss across the sump screens at Unit 1 due to insulation debris [25.4.138]. Accordingly, calculation [25.4.138]

   - identifies the break that potentially generates the largest quantity of debris.

   - determines the portion of the debris that reaches the sump screen.

   - determines the impact of this debris on the performance of the RS and LHSI pumps.

01287.B07

1. SRP 6.2.1.1.A, "PWR Dry Containment Including Subatmospheric Containment," 1975 [25.1.4]

   Requirement - The acceptance criteria detailed in this document complement GDC 50 and apply to the maximum temperature and pressure conditions in containment due to a spectrum of breaks and the relationship with containment design pressure. It also provides guidance on the effectiveness of heat removal systems and the required instrumentation needed to monitor containment following an accident.

   Type - Self-imposed requirement. [24.13.12]

   Design Basis Commitment - The entire SRP 6.2.1.1.A. [24.13.12]

   Design Basis Feature - The RS System design uses the acceptance criteria provided in SRP 6.2.1.1.A related to subatmospheric containments and (in conjunction with the QS System) reduces and maintains containment pressure to subatmospheric within 1 hour of the accident. [25.4.2] [25.4.10] [25.4.72] [25.4.139] [24.2.17]

2. SRP 6.2.2, "Containment Heat Removal System," 1975 [25.1.7]

   Requirement - The acceptance criteria detailed in this document amplify the requirements of GDC 38, 39, 40, and 50 of 10CFR50, Appendix A, and provide the basis of the containment heat removal system design.

   Type - Self-imposed requirement. [24.13.13]

   Design Basis Commitment - The entire SRP 6.2.2 with the exception of acceptance criteria 11(6) related to sump design and compliance with the analyses requirements of RG 1.82 [24.13.13]

   (as required by SRP 6.2.2) is committed to for NAPS-1 only (subsequent to SG replacement at that unit).

   Design Basis Feature - The RS System design is based on the guidance of SRP 6.2.2. [25.4.2] [25.4.72] [25.4.139] [24.2.17] Acceptability of the containment sump design was based on actual hydraulic model studies/tests done using a model of the NAPS 1 and 2 reactor containment sumps. [25.12.2]. Subsequent to SG replacement at NAPS-1, calculations following the analyses requirements of RG 1.82, Rev 1 were performed to determine head loss across the emergency sump screens due to debris accumulation [25.4.138]

Design Basis Feature - The ORS and IRS pumps are "deep draft pumps" as defined by the Commission in IE Bulletin 79-15. As required by the bulletin, the appropriate information relative to these pumps was provided to the NRC in Virginia Power letter [25.5.7]. The review for deficiencies identified excessive bearing wear in the ORS and IRS pump motors. To resolve the issue, the ORS and IRS pump motors were fitted with a single-row angular contact split inner ring bearing which had ratings exceeding the original. The end shield opening was increased in size to give a greater radial clearance around the shaft. Also, to minimize this increase, the labyrinth sealing slinger on the lower end of the end shield was made larger in diameter, one labyrinth surface was added, the labyrinth clearances were made closer, and the distance the lab seal sits on the seal was made longer. [24.13.15]

Insert 13.2.3

The results of the previous long-term ORS and IRS pump testing were also transmitted to the NRC. [25.5.7] (Also see Section 3.3.1.)

13.2.4 Other Regulatory Guidance/Information Documents

This subsection identifies any other NRC regulatory guidance document not identified in Sections 13.2.1, 13.2.2, and 13.2.3 that are design basis commitments identified in the licensing basis.

Other regulatory guidance/information documents applicable to the design basis of the RS System are identified below:

1. NUREG-0588 "Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment," July 1981 [25.1.28]

   Requirement - Equipment that is used to perform a necessary safety function shall be capable of maintaining functional operability under all service conditions postulated to occur during the installed life for the time it is required to operate. This requirement, which is embodied in GDC 1, 2, 4, and 23 and Sections III and XI of Appendix B to 10CFR Part 50, is applicable to equipment located inside as well as outside containment. More detailed guidance related to the methods, procedures and guidelines for demonstrating this capability has been set forth in IEEE Std. 323 and ancillary daughter standards (e.g., IEEE Stds. 317, 334, 382, 383) and has been endorsed

4. GL 85-22, "Potential for Loss of Post-LOCA Recirculation Capability due to Insulation Debris Blockage," December 3, 1985. [25.1.66]

   Requirement - This notification to all licensees stated that based on the staff's technical findings, debris blockage effects are dependent on types and quantities of insulation employed, primary system layout within containment, post-LOCA recirculation patterns and velocities and the post-Loca recirculation flow rates. It stated that no single generic solution was possible and required plant specific sump debris transport/accumulation analyses in accordance with the guidance of RG 1.82, Rev. 1. Though this issue has not been classified as a required "back fit" action, it strongly recommended that RG 1.82, Rev. 1 be used as guidance for the conduct of 10CFR50.59 reviews dealing with the changeout and/or modification of thermal insulation installed on primary coolant system piping and components. [25.1.66]

   Type - Self-imposed requirement [25.4.138].

   Design Basis Commitment - The entire GL 85-22 with the exception of compliance with the requirements of SRP 6.2.2, Rev. 4. [25.4.138] (NAPS Unit 1 only)

   Design Basis Feature - As a result of SG replacement at NAPS Unit 1, analytical models using the guidance of RG 1.82, Rev. 1, NUREG/CR-2791 and NUREG-0897 were used to establish insulation debris inventory and address debris accommodation/transport to develop head loss across the emergency sump screens. [25.4.138]

<u>Type</u> - Licensing basis commitment. [25.10.7] [25.4.3] [25.4.73] [25.4.74]

<u>Design Basis Commitment</u> - Only that portion related to dose conversion factors. The dose conversion factors used in the radiological analyses are obtained from TID-14844. [25.4.3] [25.4.73] [25.4.74] [25.10.7]

<u>Design Basis Feature</u> - One of the criteria that determine the acceptability of the containment depressurization systems' (RS and QS) design is the result of the radiological dose analyses. The dose conversion factors used in the radiological analyses are obtained from TID-14844. [25.4.3] [25.4.73] [25.4.74]

3. NUREG-0737, "Clarification of TMI Action Plan Requirements," November 1980

<u>Requirement</u> - Items developed by the NRC as a result of review of the TMI incident shall be implemented. [25.1.23]

<u>Type</u> - Licensing basis commitment. [25.5.5]

<u>Design Basis Commitment</u> - In support of providing additional accident monitoring instrumentation as required by Section II.F.1 of NUREG-0737, the RS System design shall include containment sump water level indication with a measurement capability of 600,000 gallons. [25.5.5]

Insert 13.2.4

<u>Design Basis Feature</u> - Refer to Section 13.3.3.

13.3 OPERATING LICENSE DOCUMENTS

Operating licenses were issued to the NAPS Station, Units 1 and 2, on November 26, 1977, and August 21, 1980, respectively. The Company is required to maintain compliance with the conditions set forth in the Operating License (OL) as contained in the <u>Updated Final Safety Analysis Report</u> (UFSAR) and the <u>Technical Specifications</u> (TS) as well as other docketed documents that constitute the licensing basis. While the UFSAR and TS contain significant design information, they are considered to be "output" documents that contain results of the design process. The basis of the UFSAR and TS is the design basis contained in this SDBD. The UFSAR and TS must be consistent with the design basis.

4. NUREG/CR-2791, "Methodology for Evaluation of Insulation Debris Effects," September 1982. [25.1.64]

   Requirement - This document describes a methodology for estimating the insulation debris generation and transport associated with postulated piping failure within primary containment. In addition, guidance is also provided to support assessment of the consequence of recirculation sump screen blockage following an accident. [25.1.64]

   Type - Licensing basis commitment. [25.10.6] [24.8.1]

   Design Basis Commitment - The entire requirement as stated above, for Unit 1 only. [24.8.1] [25.10.6]

   Design Basis Feature - Subsequent to the direction received in GL 85-22 and as a result of SG replacement at NAPS-1, analytical models in accordance with (to the extent practicable) the guidance of NUREG/CR-2791 were used to assess insulation damage (following a LOCA), accumulation/transport of debris in the NAPS-1 emergency sump, and head loss across the sump screens. To simplify the model, rigorous analyses (as discussed in the NUREG) were replaced in some areas, by conservative assumptions. [25.4.138]

5. NUREG-0897, "Containment Emergency Sump Performance," October 1985. [25.1.65]

   Requirement - This document compiles the technical findings relative to the issue of containment emergency sump performance which were derived from extensive experimental studies, generic plant studies and assessments of sumps used for long term cooling. Based on test data, empirical equations for debris transport/accumulation were established and are recommended for use when analyzing sump performance.

   Design Basis Commitment - The entire requirement as stated above, for Unit 1 only. [24.8.1] [25.10.6]

   Design Basis Feature - Subsequent to the direction received in GL 85-22 and as a result of SG replacement at NAPS-1, analytical models (which used the technical findings outlined in NUREG 0897) were used to evaluate sump debris accumulation and transport at NAPS - Unit 1. [25.4.138]

01287.B07

design includes vents and drains to allow maintaining the RS coolers dry.

b. The primary function of the RS System is to support containment depressurization. In performing its intended function, the RS System indirectly supports the ECCS. Following an accident, the containment sump water volume is made up from several sources, the main contributor is the QS System's RWST. The CCT fluid is also added to the containment sump water inventory. However, its design bases is to support the RS pumps NPSH; and providing the ECCS with water is an indirect result. Similarly, the design bases for the RS coolers is to support cooling of sump water for containment depressurization purposes. As an indirect result, this cooling supports the SI System in its core cooling function. These indirect results of the RS System function are, however, not a part of the RS System design bases. It should be noted that the RS System design bases allow the operator to throttle down the recirculation spray after the containment has reached subatmospheric conditions. The long-term (beyond 1 hour and up to 30 days) need for recirculation spray is only on an as-needed basis and, specifically, to maintain the containment subatmospheric. [25.1.4] [25.4.2] [25.4.4] [25.4.72] [25.4.20] [25.4.77]

[25.4.139] [24.2.17]

2. UFSAR Section 7.1.3.2.4, "Environmental Requirements" [25.10.39]

<u>Requirement</u> - A temperature monitoring system shall be provided in spaces where Class 1E equipment are located but where non-safety-related HVAC systems are provided.

<u>Type</u> - Licensing basis commitment. [25.10.39]

<u>Design Basis Commitment</u> - The entire requirement as stated above. [25.10.39]

<u>Design Basis Feature</u> - The AM System was provided in support of the above requirement. Relative to the RS System, the AM System provides an alarm in the Control Room when the ambient temperature in the Casing Cooling Pumphouse or the cable tunnel area (both areas have safety-related RS equipment and non-safety-related HVAC systems) exceeds the qualified

temperature range of the safety-related RS equipment located in these areas. [25.8.95] [25.8.96] [25.8.97] [25.8.98]

3. UFSAR Section 6.2.2, "Containment Heat Removal Systems - Containment Depressurization Systems" [25.10.6]; UFSAR Section 15.4.1, "Major Reactor Coolant System Pipe Rupture" [25.10.7]

<u>Requirement</u> - The containment depressurization systems shall be designed to ensure that the containment structure is depressurized to subatmospheric in less than 1 hour following a LOCA.

<u>Type</u> - Licensing basis commitment. [25.10.6] [25.10.7]

<u>Design Basis Commitment</u> - The entire requirement as stated above. [25.10.6] [25.10.7]

<u>Design Basis Feature</u> - The containment depressurization systems are designed to ensure that the containment structure is depressurized to subatmospheric in less than 1 hour following a LOCA. [25.4.2] [25.4.4] [25.4.139]

4. UFSAR Section 15.4.1, "Major Reactor Coolant System Pipe Rupture" [25.10.7]

<u>Requirement</u> - Post-accident ESF leakage outside containment shall be limited to a maximum of:

a. Continuous leakage of 900 cc/hr
b. A single pump failure resulting in a 10 minute leak at 50 gpm

<u>Type</u> - Licensing basis commitment. [25.10.7]

<u>Design Basis Commitment</u> - The entire requirement as stated above. [25.10.7]

<u>Design Basis Feature</u> - The NAPS Units 1 and 2 post LOCA site boundary (LPZ) [25.4.74] and Control Room dose analysis [25.4.75] address a maximum ESF leakage of 900 cc/hr (continuous) and 50 gpm for 10 minutes due to pump failure. [24.2.14] The RS System is a part of the NAPS Units 1 and 2 engineered safeguards. Consequently, RS System leakage is controlled and monitored by TS 6.8.4, item a. [24.3.9] Refer to

Design Feature - The portion of the RS System required for accident mitigation is on standby during Modes 1, 2, 3, and 4. Upon receiving a CDA signal, the RS System is designed to automatically start up and provide containment cooling/depressurization by recirculating containment sump water at sufficient flow and pressure to create an atomized spray in containment. [25.8.1] [25.8.2] The system is designed to maintain containment pressure at values less than its design pressure and minimize containment leakage by depressurizing containment structure to subatmospheric within 1 hour. [25.4.2] [25.4.4] [24.2.1] [25.4.72]

[25.4.139]
[24.2.17]

b. Requirement - In the event of an accidental release of high-energy radioactive fluids inside containment and subsequent containment pressurization, the RS System, in conjunction with the QS System, shall reduce the concentration of radioactive iodine in containment atmosphere so that the site boundary dose from any containment leakage prior to the containment becoming subatmospheric is within the limits of 10CFR100. [25.1.2] [25.1.3] [25.1.5] [25.10.1]

Design Feature - The RS System recirculates containment sump water through the recirculation spray rings for the duration of the accident. [25.8.1] [25.8.2] The containment sump water pH, which is controlled by the QS System, facilitates removal of iodine from the containment atmosphere to the sump, [25.1.6] [24.2.9] and minimizes the potential for stress corrosion cracking for components located inside containment. [25.7.3] [25.7.4]

c. Requirement - In the event of an accidental release of high-energy fluids inside containment and subsequent containment pressurization, the RS System shall provide containment isolation capability (relative to containment atmosphere leakage via the RS System containment isolation valves) to reduce the leakage of containment atmosphere, so that the site boundary dose from

Requirement - To support the postaccident NPSH requirement of the ORS pumps, the RS System shall cool and maintain the temperature of the water in the CCT during all modes of normal operation except cold shutdown and refueling. [25.1.22] [25.11.1] [25.10.2]

Design Feature - A CCT recirculation pump and refrigeration unit are used to maintain the water in the CCT at the temperature required by the ORS pump NPSH analysis. If necessary, a second recirculation pump and refrigeration unit are available for use. [25.8.1] [25.8.2]

14.1.2 Performance

The performance criteria for the RS System are identified in Section 2.2. The operational performance of the system under normal and accident conditions is addressed in Chapter 3. Chapter 6 identifies major system performance parameters and their bases. Chapter 11 addresses the operation of the system under accident conditions.

1. Safety-Related Performance Criteria

   a. Requirement - The RS System shall be initiated automatically by the containment depressurization activation (CDA) signal, which shall be initiated by a containment "high-high" pressure signal activated at a containment pressure established in the safety analyses. [25.4.2] [25.4.4] [25.4.72]

   [25.4.139]
   [24.2.17]

   Design Feature - The RS System is initiated automatically by the containment depressurization actuation (CDA) signal which is initiated by the containment "high-high" pressure signal. [25.8.1] [25.8.2] [25.3.34] [25.3.35]. A containment "high-high" pressure setpoint of 24.7 psia is used in the safety analyses. However, as documented in the analyses, it has been determined that a value of 30 psia is acceptable. [25.4.2] [25.4.72] [25.7.41] The RS System is initiated automatically by the CDA signal, which is activated by the containment "high-high" pressure signal, which has a setpoint of 27.75 psia and an error band that could result in a containment pressure up to a maximum of 29.25 psia. [25.3.11] [25.3.12] [25.3.47]

The CDA signal can also be initiated manually.

b. Requirement - The CDA signal shall activate timers which will delay the start of the RS pumps. This delay will allow for the adequate buildup of water in the containment sumps (resulting from the break and the operating of the QS System) and prevent IRS/ORS pump cavitation at startup. The maximum time delay allowable prior to an effective recirculation spray shall be as established in the safety analyses. [25.4.4] [25.4.72] [25.4.2]

[25.4.139]
[24.2.17]

Design Feature - The CDA signal activates timers (purchased via specification [25.6.30]) that delay the start of the recirculation pumps. [25.8.11] [25.8.12] [25.8.27] [25.8.28] [25.8.14] [25.8.15] [25.8.30] [25.8.31] The safety analyses use a maximum delay time of 300 seconds. [25.4.4] [25.4.2] [25.4.72] Calculation [25.4.111] indicates that recirculation spray will be available and effective at 300 seconds following a CDA. [25.7.42] [24.3.1] [25.4.81] [24.1.10]

c. Requirement - The RS System shall pump containment sump water through the shell side of the RS coolers to the RS rings at a minimum flow rate established in the safety analyses. [25.4.4] [25.4.2] [25.4.72]

Design Feature - The safety analyses and the system design analyses establish an ORS and IRS minimum flow requirement of 3640 gpm against a head of 285 ft and 3300 gpm against a head of 273 ft, respectively. [25.4.2] [25.4.4] [24.2.1] [25.4.51] [25.4.49] [25.4.12] [25.4.11] [25.4.72] The ORS and IRS pumps are purchased via specification [25.6.1] to be able to deliver 3640 gpm against a head of 290 ft (see pump curves included in calculation [25.4.51]) and 3300 gpm against a head of 275 ft (see pump curves included in calculation [25.4.49]), respectively, of containment sump water to the RS rings.

d. Requirement - The minimum overall heat removal coefficient for the RS coolers shall be as established in the safety analyses. [25.4.4] [25.4.2] [25.4.72]

[25.4.139]
[24.2.17]

Design Feature - The safety analyses establish an IRS and ORS minimum heat removal coefficient requirement of 3.55E6 Btu/hr-°F and 3.65E6 Btu/hr-°F, respectively. [25.4.2] [25.4.4] [25.4.72] [24.2.1] [24.11.6] The RS coolers are purchased with a heat removal capacity of 3.82E6 Btu/hr-°F assuming a zero fouling resistance. [25.6.9] [24.9.10]

e. Requirement - The location of the RS rings, the size and arrangement of the nozzles in the rings, and the pressure drop across the RS nozzles shall support the minimum droplet thermal effectiveness of the sprays assumed in the containment safety analysis. [25.4.2] [25.4.4] [25.4.72]

Design Feature - The safety analyses established a droplet thermal effectiveness requirement of 0.9 for the recirculation sprays. [25.4.2] [25.4.4] [25.4.72] [24.2.1] NAPS Units 1 and 2 drawings [25.8.8] and calculations [25.4.30] [25.4.34] [25.4.46] [24.6.4] [24.2.3] document that, based on the purchased RS nozzles, an assumption of 0.9 for the droplet thermal effectiveness is appropriate. (For further information, see Section 14.1.11 and 14.4.2.)

f. Requirement - The location of the RS rings and the size and arrangement of the nozzles in the rings shall maximize the volume of containment atmosphere covered by the recirculation spray to support heat removal. [25.4.2] [25.4.4] [25.4.72] [24.2.4]

Design Feature - The containment safety analyses assume 100-percent containment coverage by sprays. [25.4.2] [25.4.4] [25.4.72] The RS rings are located near the top of the dome to maximize the volume of containment atmosphere covered. The IRS and ORS subsystems form an approximately 100-ft diameter, 360 degree ring at EL 377 ft-10 in., and EL 376 ft-10 in., respectively.

[25.8.8] [25.8.1] [25.8.2] [25.4.29] There are two sizes of nozzles (1HH30100 and 1/2B60) in the spray rings. [25.6.10] [24.2.6] The size and arrangement of these nozzles provide a spray pattern that maximizes coverage. [25.4.59] [25.4.31] From a heat removal standpoint, as a result of the maximized coverage and high RS flow rate, sufficient mixing would occur (due to the temperature and pressure gradients caused) to support the 100-percent coverage assumption in the containment analyses. [24.2.4]

[25.4.139]

g. Requirement - To support the postaccident NPSH requirements of the ORS pumps, cold water injection shall be provided to the suction piping of the ORS pumps at the rate, temperature, and for the duration required by the safety analyses. [25.4.2] [25.4.4] [24.2.5] Note that since casing cooling flow is available, credit is taken for it in the containment analyses. [25.4.2] [25.4.72] [25.4.4]

[25.4.139]
[24.2.17]

Design Feature - To support the postaccident NPSH requirements of the ORS pumps, the NPSH analyses requires cold water injection at a flow rate of 800 gpm for 1 hour following a LOCA at a maximum temperature of 50°F to the ORS pump suction. [25.4.2] The analysis assumes the availability of 96,000 gallons of casing cooling fluid. In accordance with the above requirement, a casing cooling subsystem is provided [25.8.1] [25.8.2] (which consists of a 124,932 gallon (design capacity) tank [25.8.59] [25.4.110] and two 960 gpm casing cooling pumps purchased via specification [25.6.8]) which, in the event of an accident and the receipt of a CDA signal, will automatically inject the CCT water into the ORS pump suction, as specified by the analyses. [25.8.1] [25.8.2] [25.4.53] [25.4.57] [25.4.82] [25.4.83] The NAPS Technical Specification [25.3.11] [25.3.12] maintains a CCT volume of 116,500 gallons. Also provided are two CCT recirculation pumps [25.6.14] and two CCT refrigeration units [25.6.13] which, during all modes of operation except refueling and cold shutdown, maintain the temperature of the CCT at

[25.4.139]

a temperature below 50°F. [25.3.11] [25.3.12] [25.3.47] (See Section 14.1.2, Item 3 for further information.)

To support the NPSH requirements of the IRS pumps, the NPSH analyses require an injection of water at a flow rate of 150 gpm and a maximum temperature of 50°F for the time it takes the RWST to drain down. [25.4.2] [25.4.139] This cooling flow is provided by the QS system. [25.8.1] [25.8.2] [25.3.6] [24.2.7]

Since casing cooling and QS injection are required for the ORS and IRS pump NPSH requirements following a LOCA and are made available by system design, credit is taken for the addition of 800 gpm (of 50°F water) and 150 gpm (of 50°F water) of cooling fluid into the ORS and IRS pump suction, respectively, for the containment depressurization evaluation in the containment safety analyses. [25.4.2] [25.4.4] [25.4.72] [25.4.139] [24.2.17]

h. <u>Requirement</u> - To ensure that the RS System performance is in accordance with the assumptions of the containment safety analysis, the containment sump shall be designed to discourage debris transport. Consequently, the total sump screen area shall be large enough to ensure low sump fluid velocity (thus reducing debris transport to the RS pump suction) and the sump screen mesh opening size shall be smaller than the RS nozzles (to prevent nozzle blockage). [25.12.2] [25.8.58] [24.2.6] [25.4.45]

<u>Design Feature</u> - The containment recirculation sumps are provided with screens to prevent blockage of the RS pumps' suction points and ring nozzles with sump debris. [25.10.15] [25.8.58] The sump screens are designed seismically to remain functional during an accident. [24.14.1] They consist of gratings which act as a trash screen for large debris (and vortex suppressants) and three stages of mesh screening which prevent particles larger than the smallest nozzle orifice of the RS rings from entering the RS pump suction. The

water entering the containment sump first passes through inclined 1-in. x 1/8-in. grating bars on 1-3/16-in. centers which remove large debris. [25.6.11] [24.14.2] Downstream of the grating, there are three stages of screening. The first stage is a coarse mesh with 0.558in. square opening. [24.11.9] The second and third stages are fine meshes with square openings of 0.05-in. [25.8.58] [24.11.9] The screen mesh sizes are smaller than the diameter of the RS nozzles (1/2 B60 and 1HH30100 nozzles) or 0.5156 in. and 0.25 in., respectively. [25.4.30] The third stage consists of cylindrical screens mounted over the intake to the RS Subsystem pumps. One cylindrical screen surrounds the intake of each pump. The design includes perforated vortex breakers inside the cylindrical screens as vortex suppressants.

Each containment sump assembly is divided into two separate halves by a fine mesh divider so that the failure of either half will not adversely affect the other half. Consequently, no single failure could result in the clogging of all four suction points for the RS pumps. [25.8.58]

The containment sumps are designed to direct water to the suction of the pumps. [25.8.66] [25.8.8] [25.8.57] A total sump screen area of 168 ft^2 is provided to support low water velocity in the sump to discourage debris transport and assure that the sump screens will not clog and reduce system flow rate or cause pump cavitation. Acceptability of the sump screen area was based on actual hydraulic model studies/tests done prior to plant operation using a model of NAPS Units 1 and 2 containment sumps. [25.4.42] [25.4.45] [25.4.58] [25.12.2]

Subsequent to SG replacement at NAPS-1 (and change of the type of insulation used) a head loss analysis (across the sump screens) due to debris accumulation was performed. [25.4.138]. The calculated head loss was factored into the new LOCA analysis for NAPS-1 [25.4.139]

Layout and arrangement of equipment/piping located in the containment recirculation sump (e.g., dewatering pump and associated piping) are based on ensuring that RS pump fluid flow is not impacted (e.g., no vortices are created, etc.). [25.7.20] taken for RS pump discharge pressure indicators as alternate instrumentation in place of the required containment spray flow indicators. [25.3.48] [24.2.12] The performance criteria of the above instrumentation meets the requirements of RG 1.97. [24.14.3] [24.2.16]

[25.4.139]

3. Non-Safety-Related Performance Criteria

Requirement - In support of the RS System function to automatically maintain the temperature of the CCT within the requirements of the ORS pump NPSH analyses [25.4.2] during all modes of operation except shutdown, refueling, and accident, the following performance criteria shall be met: [25.11.1]

a. The temperature setpoints for the CCT temperature sensors (which initiate the CCT cooling loop) shall be well below that required by the ORS pump NPSH analyses. [25.11.1]

b. The heat-removal capability of the refrigeration units associated with the CCT cooling loop shall be greater than the normal operation heat gain of the CCT fluid during the worst ambient conditions. [25.4.35]

[25.4.139]

Design Feature - The ORS pump NPSH analysis [25.4.2] takes credit for casing cooling fluid at 50°F being injected into the ORS pump suction following an accident. To ensure the availability of CCT fluid at the required temperature, the setpoints for the CCT temperature indicators (which initiate the CCT cooling loop) are set to maintain the CCT fluid between 42°F and 45°F during normal plant operation. [25.3.47] Furthermore, the normal operation heat gain of the CCT fluid during the worst ambient conditions is 33,549 btu/hr. [25.4.35] The CCT refrigeration units are purchased with a 6-ton capacity (or 72,000 Btu/hr). [25.6.13] [24.9.13] If used continuously, each unit is capable of removing sensible heat to the extent of reducing the temperature of the contents of the CCT by 0.92°F/day. [25.4.35]

System is not required to operate during normal plant operations. In accordance with SRP 6.2.1.1.A, the RS System is required to be operable for 30 days following an accident [24.13.12]; per Virginia Power commitment to NUREG 0588 for Unit 2, the RS System including the casing cooling subsystem is required to be available for 120 days [25.3.56]; and in accordance with the safety analyses, the casing cooling injection loop is required only for 1 hour after the accident [25.4.2] [25.4.139]

The environmental requirements for RS equipment are satisfied via the purchase specifications which identify the normal operation as well as the postaccident environmental conditions (including time duration) to which the equipment must be designed. (Refer to Sections 14.2.6, 14.3.6, 14.4.6, 14.5.6, 14.6.6, and 14.8.6.)

Environmental Qualification of safety-related components is treated as follows:

a. (i) Safety-related electrical, instrumentation, and control RS components required for postaccident mitigation and located in harsh environments (i.e., the IRS and ORS pump motors, the RS and casing cooling MOV motors) are listed in the EQ Master List [25.3.64] and are qualified in accordance with the EQ program (refer to Section 15.1.2). In accordance with this program, to demonstrate qualification, QDRs (which include the supporting qualification information such as engineering/vendor analyses, vendor qualification documentation (i.e., test reports etc.), similarity analyses etc.) are prepared and maintained for each of the above equipment [25.3.49] [25.3.2] [24.13.10].

(ii) Safety-related electrical (including instrumentation and controls) RS equipment located in mild environments (e.g., the casing cooling pump motors) are not included in the EQ program. [25.3.49] The general quality and surveillance requirements applicable to the equipment as

PG System water is also used to provide water during ORS pump testing. [25.3.8] [24.14.13]

- In the event of an accidental release of high-energy fluids inside containment, (LOCA, MSLB, REA, FWLB) the QS System, in conjunction with the RS System, cools and depressurizes containment via sprays. [25.8.1] [25.8.2] [25.4.4] [25.4.2] [25.4.72] [25.4.139] [24.2.17]

  The QS System piping provides 150 gpm of RWST water to the IRS pumps' suction inlet in the containment sump. [25.8.1] [25.8.2]

  The QS System provides CCT makeup water and water for the IRS pump flow tests using the RWST via hose connections. [25.3.8] [24.5.3]

- The SW System piping is connected to the tube side of the RS coolers to cool the containment sump water which passes through the shell side of the coolers. [25.8.1] [25.8.2]

b. Electrical

- The RS System Class 1E 125V dc control circuits are supplied from the 125V dc ED System. [25.8.44] [25.8.49] [25.8.43] [25.8.48]

- The safety-related RS System electrical equipment is supplied from the Class 1E EP System at 4160V and 480V. [25.8.46] [25.8.42] [25.8.47] [25.8.51]

  A loss of the EP System bus voltage will actuate the trip and the subsequent reclosure of the IRS and ORS pump motors to facilitate loading of the diesel generator. [25.8.27] [25.8.28] [25.8.11] [25.8.12]

RS-MOV-155A,B, 156A,B, 100A,B, and 101A,B (255A,B, 256A,B, 200A,B, and 201A,B). [25.8.43] [25.8.44] [25.8.48] [25.8.49]

d. Civil/Structural

- The Auxiliary Building provides the supports and protection from natural phenomena for non-safety-related RS System MCCs. [25.8.42] [25.8.47] (See Section 14.1.5.)

- The Cable Tunnel Area provides the supports, and protection from natural phenomena for safety and non-safety-related RS System MCCs. [25.8.42] [25.8.47] (See Section 14.1.5.)

- The Casing Cooling Pumphouse provides the foundations/supports, and protection from natural phenomena for the safety and non-safety-related portions of the casing cooling subsystem. [25.8.61] (See Section 14.1.5.)

- The containment sump and sump screens are designed to prevent trash and debris from clogging the RS pumps suction and spray nozzles. [25.8.58] [25.4.30] [25.12.2] Paint used inside containment is qualified to accident conditions to minimize peeling. [25.6.15] [24.11.3] Surfaces painted with unqualified paint are covered with a stainless steel screen to retain dislodged paint chips. [24.11.4] Insulation used in containment is resistant to shredding. [25.6.16] [24.11.5]

Subsequent to SG replacement at NAPS-1 (and the use of fiberglass insulation) head loss due to post LOCA debris accumulation in the emergency sump was analyzed and factored into the NAPS-1 LOCA analysis. [25.4.138] [25.4.139] [25.12.11]

The containment structure provides the foundations/supports, and protection from natural phenomena and internal plant hazards for the IRS pumps, the RS coolers, spray rings, and associated piping. [25.8.9] (See Section 14.1.5.)

b. Electrical

None.

c. Instrumentation and Control

- The RS System provides signals to the EI system to control and monitor RS system operating parameters identified in Figure. 4.2-2. [25.8.52] [25.8.53] [25.8.54] [25.8.55] [25.8.56]

- The RS System parameters provided to the ERF computer are identified in Figure 4.2-2 and listed in [25.3.40] [25.3.41] [24.4.5]. (See Section 14.1.16.)

- The RS System input to the CM System is identified in Figure. 4.2-2. and listed in [25.3.59] [24.14.17]. (See Section 14.1.16.)

d. Civil/Structural

The RS System (in conjunction with the QS System) supports the structural integrity of the containment (in the event of an accidental release of high-energy fluids inside containment) by delivering a cooling spray to the containment structure. [25.8.1] [25.8.2] [25.4.2] [25.4.4] [25.4.72] [25.4.139] [24.2.17]

e. Other

None.

14.1.8 Material Requirements

This subsection identifies the specific material requirements associated with the RS System that are part of the design basis. The material requirements identified are based on the system function, process fluid and environmental conditions, and are <u>independent</u> of the actual component installed. This subsection identifies the functional material requirements, not necessarily a specific material. Usually, the functional design basis material requirements can be accommodated by equivalent materials, providing flexibility for changes or replacements.

Design Feature

1. The RS System piping/fittings (including orifices)/valve size, and layout are designed based on the flow requirements of the RS System as established in the safety and pump NPSH analyses. [25.4.2] [25.4.4] [25.4.72] [25.4.139]

   a. The IRS subsystems require a minimum flow rate of 3300 gpm.

   b. The ORS subsystems require a minimum flow rate of 3640 gpm.

   c. The casing cooling injection flow into the ORS pump suction is required to be 800 gpm.

   The "Process Condition Summary" sheets included with this chapter, as well as Sections 14.2.2 and 14.6.2, indicate that the flow rates in different portions of the RS System (based on the existing system layout) meet the flow requirements of the safety analyses.

2. Fluid velocity requirements associated with proper operation of the RS components are addressed during pipe sizing and are in accordance with the mechanical section of the PDBD. [24.14.37]

3. Allowable pressure drop in the system is limited such that the pressure difference between the fluid in the spray header and the containment atmosphere is approximately 25 psid. [25.7.29] Calculation [25.4.48] shows that, based on the current RS System design/layout, the RS maximum spray pressure for the IRS and ORS fluid is 29.4 psid and 24.1 psid, respectively.

4. The NPSH requirements for the pumps within the RS System are addressed in Sections 14.2.11, 14.6.11, and 14.7.11.

5. The containment sumps are designed to direct the water to the suction of the pumps. (Refer to Section 14.1.2, item h.) Screens are provided to remove debris that may be detrimental to pump performance. (Refer to Section 14.1.2, item h.) Pump startup is delayed by timers to assure that sufficient time is allowed for the containment sump to fill up. (Refer to Section 14.1.2, item b.)

6. The minimum submergence required to avoid vortex formation at the IRS and ORS pump suction pipe configuration is 3.5 ft. [24.14.73] The pumps suction lines are located at about El. 212 ft. [25.8.57] Prior to postaccident pump startup, the minimum sump water level is about 216 ft-10 in. [25.4.43]

Additionally, Alden Research Laboratories modeled NAPS' containment sump to ascertain the absence of adverse hydraulic phenomena that could affect NPSH. n A 1:3 model was designed and constructed to include the sump and the surrounding area of the Containment Building with all the structures that could influence the approach flow. Tests incorporating various possible flow and pump combinations under different possible restrictions, such as screen blockage, were undertaken. It was determined that the original design of the sump was satisfactory for all design conditions, with the exception of asymmetrical screen blockages. A modified design incorporating vortex-suppressing devices was developed. Consequently, design modifications involving two levels of floor gratings and perforated vortex breakers inside the cylindrical screens were installed in the containment sump. This design performed satisfactorily under all possible conditions. [25.12.2] [25.10.6]

Subsequent to SG replacement at NAPS-1 and a self imposed commitment by Virginia Power to GL 85-22, head loss across the emergency sump screens (due to debris accumulation) was analyzed in accordance with RG 1.82, Rev 1 and factored into the RS pump NPSH and NAPS-1 LOCA analysis [25.4.138] [25.4.139]

14.1.12 Chemistry and Sampling Requirements

This subsection identifies chemistry or chemical requirements that are part of the design basis for the RS System, including design requirements that facilitate sampling or other chemistry requirements.

1. Requirement - The fluids of the RS System shall be compatible with the function of the system and its supported systems. [25.2.8] [25.1.54] [25.5.6]

   The chemistry of the RS System fluids shall be controlled and maintained so that the system performance requirements are met. [25.2.8] [25.1.54] [25.5.6]

   Design Feature

   a. The chemistry of the containment sump water (which is the RS fluid) is controlled by the QS System and has the following concentration. [25.4.76].

There are no special design features (other than providing easy access) incorporated into the RS System design to support inspection and surveillance. [24.14.84]

4. Requirement - The RS system/component design shall include features to allow testing/surveillance of the system to ensure its performance in accordance with the requirements of the plant safety analyses [25.2.8] [25.5.6] [25.1.54]

Design Feature - Refer to Section 14.1.20, item 1, 2, 3 for design features provided to ensure RS system/component performance in accordance with the requirements of the safety analyses. The surveillance requirements (originating from the safety analyses [25.4.2] [25.4.4] [25.4.72]) which are associated with the CCT fluid temperature and volume are facilitated by the CCT fluid temperature and level indicators [25.8.1] [25.8.2]. The boron concentration of the CCT fluid is determined by taking a sample from the drain taps provided in the system and performing the appropriate testing [25.8.1] [25.8.2]. (Note that limitations on the CCT boron concentration is required by calculation [25.4.76] [25.4.129]). The radioactive concentration of the CCT fluid may be determined in a similar fashion. [24.14.88]

[25.4.139]
[24.2.17]

5. Requirement - The RS System/components design shall include features to allow testing/surveillance of the system in accordance with the requirements of the NAPS Units 1 & 2 Technical Specifications. [25.3.11] [25.3.12]

Design Feature - The NAPS TS Sections 3/4 3.2.1, 3/4 6.1.2, 3/4 6.2.2, 3/4 6.3.1, 3/4 11.1.4, 4.0.5, 4.5.2d, and 6.8.4, item a impose testing and surveillance requirements on the NAPS RS Systems. Refer to Section 13.3.2 for information relative to design features provided to address these requirements.

6. Requirement - To maximize reliability, the non-safety-related portion of the RS System shall be designed to permit appropriate periodic testing. [25.5.6] [25.1.54] [25.2.8]

Design Feature - Refer to Section 14.7.20.

Table 14.1-1. Process Condition Summary

SYSTEM: Recirculation Spray (ORS Train A)	SYSTEM CODE RS	SDBD NUMBER SDBD-NAPS-RS	NODE NO. 2

NODE TYPE: X MASS FLOW __ INTERFACE __ COMPONENT

PARAMETER	DESIGN VALUE	OPERATING CONDITIONS: NORMAL	EMERGENCY	SHUTDOWN	OTHER
PRESSURE (psig)	55 (v)	Note 1	45 (i)	Note 1	Note 1
TEMPERATURE (°F)	285 (v)	Note 1	218.4 (i) Note 2	Note 1	Note 1
FLOW RATE (gpm)	3640 (iv)	Note 1	3640 (ii) Note 3	Note 1	Note 1
NPSH REQ'D (ft)	11.0 (vi)	Note 1	16.8 (vii) Note 6 15.3 (vii) Note 6	Note 1	Note 1
PRESSURE DROP (psi)	N/A	Note 1	1.97 (ii) Notes 4,5	Note 1	Note 1
HEAT LOAD (Btu/hr)	N/A	Note 1	N/A	Note 1	Note 1
FLUID TYPE	Borated Water	Note 1	Borated Water	Note 1	Note 1
VISCOSITY (centipoise)	N/A	Note 1	0.3 (iii)	Note 1	Note 1
FLUID pH	N/A	Note 1	7.7 (viii)	Note 1	Note 1

FLOW IS FROM: Casing Cooling System interface with ORS TO: 2-RS-P-2A

NOTES: The values presented in the Table are estimates based on Unit 2 (Unit 2 design documentation is more complete) but is representative of Unit 1. Exact values for Unit 1 would have to be obtained from the appropriate reference. If not available, Unit 2 values may be used depending on application.

1. System on standby and only operates during LOCA/MSLB/REA/FWLB.
2. Stress analysis to be done at 253°F (i.e., assuming failure of Casing Cooling System).
3. Plant safety analysis uses 3640 gpm.
4. Includes pressure drop for Node 1.
5. Does not include head loss due to elevation loss = 66.7 psi (ii).
6. This value is minimum NPSH available. Note that Unit 1 NPSHA = 15.3 ft and Unit 2 NPSHA = 16

REFERENCES:

Unit 1:		Unit 2:	
(i) ____	(v) [25.3.24]	(i) [25.4.48]	(v) [25.3.25]
(ii) [25.4.12]	(vi) [25.5.1]	(ii) [25.4.51]	(vi) [25.5.51]
(iii) ____	(vii) [25.4.2] [25.4.139]	(iii) [25.4.52]	(vii) [25.4.2]
(iv) [25.6.1] pump curve	(viii) [25.4.76]	(iv) [25.6.1] pump curve	(viii) [25.4.76]

REVISION NO: 0 DATE: July 1, 1990

Table 14.1-1. Process Condition Summary

SYSTEM: Recirculation Spray (ORS,Train A)	SYSTEM CODE RS	SDBD NUMBER SDBD-NAPS-RS	NODE NO. 3

NODE TYPE: X MASS FLOW ___ INTERFACE ___ COMPONENT

PARAMETER	DESIGN VALUE	OPERATING CONDITIONS NORMAL	EMERGENCY	SHUTDOWN	OTHER
PRESSURE (psig)	150 (vii)	Note 1	153.3 (i) Note 5	Note 1	Note 1
TEMPERATURE (°F)	285 (vii)	Note 1	218.4 (i) Note 2	Note 1	Note 1
FLOW RATE (gpm)	3640 (vi)	Note 1	3640 (ii) Note 3	Note 1	Note 1
NPSH REQ'D (ft)	N/A	Note 1	N/A	Note 1	Note 1
PRESSURE DROP (psi)	N/A	Note 1	20.12 (iii) Note 4	Note 1	Note 1
HEAT LOAD (Btu/hr)	—	Note 1	Various	Note 1	Note 1
FLUID TYPE	Borated Water	Note 1	Borated Water	Note 1	Note 1
VISCOSITY (centipoise)	N/A	Note 1	0.3 (iii)	Note 1	Note 1
FLUID pH	N/A	Note 1	7.7 (viii)	Note 1	Note 1
HEAT TRANSFER COEFFICIENT-UA (Btu/hr-°F)	3.82E6 (v)	Note 1	3.65E6 (iv)	Note 1	Note 1

FLOW IS FROM: 2-RS-P-2A TO: 2-RS-E-1D

NOTES: The values presented in the Table are estimates based on Unit 2 (Unit 2 design documentation is more complete) but is representative of Unit 1. Exact values for Unit 1 would have to be obtained from the appropriate reference. If not available, Unit 2 values may be used depending on application.

1. System on standby and only operates during LOCA/MSLB/REA/FWLB.
2. Stress analysis to be done at 253°F (i.e., assuming failure of Casing Cooling System).
3. Plant safety analysis uses 3640 gpm.
4. Does not include head loss due to elevation loss. Total system head loss due to elevation loss is 66.7 psi (ii).
5. See [24.14.4].

REFERENCES:

Unit 1:
(i) ____
(ii) [25.4.12]
(iii) ____
(iv) [25.4.4] [25.4.139]
(v) [25.5.8]
(vi) [25.6.1] pump curve
(vii) [25.3.24]
(viii) [25.4.76]

Unit 2:
(i) [25.4.48]
(ii) [25.4.51]
(iii) [25.4.52]
(iv) [25.4.4]
(v) [25.6.9]
(vi) [25.6.1] pump curve
(vii) [25.3.25]
(viii) [25.4.76]

X

REVISION NO: 0 DATE: July 1, 1990

Table 14.1-1. Process Condition Summary

SYSTEM: Recirculation Spray (IRS,Train A)	SYSTEM CODE RS	SDBD NUMBER SDBD-NAPS-RS	NODE NO. 5

NODE TYPE: X MASS FLOW ___ INTERFACE ___ COMPONENT

PARAMETER	DESIGN VALUE	OPERATING CONDITIONS			
		NORMAL	EMERGENCY	SHUTDOWN	OTHER
PRESSURE (psig)	150 (ix)	Note 1	111.6 (i)	Note 1	Note 1
TEMPERATURE (°F)	285 (ix)	Note 1	244.2 (i)	Note 1	Note 1
FLOW RATE (gpm)	3300 (viii)	Note 1	3300 (ii) Note 5	Note 1	Note 1
NPSH REQ'D (ft)	9.4 (vii) Note 3	Note 1	11.9 (vi) Notes 4,3 10.4 (vi) Note 4,3	Note 1	Note 1
PRESSURE DROP (psi)	N/A	Note 1	12.96 (ii) (iii) Note 2	Note 1	Note 1
HEAT LOAD (Btu/hr)	—	Note 1	Various	Note 1	Note 1
FLUID TYPE	Borated Water	Note 1	Borated Water	Note 1	Note 1
VISCOSITY (centipoise)	N/A	Note 1	0.3 (iii)	Note 1	Note 1
FLUID pH	N/A	Note 1	7.7 (x)	Note 1	Note 1
HEAT TRANSFER COEFFICIENT-UA (Btu/hr-°F)	3.82×10^5 (v)	Note 1	3.55×10^5 (iv)	Note 1	Note 1

X

FLOW IS FROM: 2-RS-P-1A TO: 2-RS-E-1A

NOTES: The values presented in the Table are e... ates based on Unit 2 (Unit 2 d... ign documentation is more complete) but is representative of Unit 1. Exact values for U... 1 would have to b... obtai... ...m the appropriate reference. If not available, Unit 2 values may be used depending on application.

1. System on standby and only operates ...ring LOCA/MSLB/RL... ...
2. Does not include head loss due to elevation loss. Total system hea... ... due to elevation loss = 67.1 psi (ii)
3. This data applies to the suction side of 2-RS-P-1A
4. This value is minimum NPSH available. Note that NPSHA for Unit 1 = 10.4 ft, Unit 2 NPSHA = 11.9 ft. X
5. Plant safety analysis uses 3300 gpm.

REFERENCES:

Unit 1:
(i) ______
(ii) [25.4.11]
(iii) ______
(iv) [25.4.4] [25.4.139] X
(v) [25.6.9]
(vi) ~~[25.4.2]~~ [25.4.139] X
(vii) [25.5.1]
(viii) [25.6.1] pump curve
(ix) [25.3.24]
(x) [25.4.76]

Unit 2:
(i) [25.4.48]
(ii) [25.4.49]
(iii) [25.4.50]
(iv) [25.4.4]
(v) [25.6.9]
(vi) [25.4.2]
(vii) [25.5.1]
(viii) [25.6.1] pump curve
(ix) [25.3.25]
(x) [25.4.7...]

REVISION NO: 0 DATE: July 1, 1990

Table 14.1-1. Process Condition Summary

SYSTEM: Recirculation Spray (IRS,Train A)	SYSTEM CODE RS	SDBD NUMBER SDBD-NAPS-RS	NODE NO. 7

NODE TYPE: X MASS FLOW ___ INTERFACE ___ COMPONENT

PARAMETER	DESIGN VALUE	OPERATING CONDITIONS NORMAL	EMERGENCY	SHUTDOWN	OTHER
PRESSURE (psig)	150 (v)	Note 1	102.9 (i)	Note 1	Note 1
TEMPERATURE (°F)	150 (v)	Note 1	131.7 (i)	Note 1	Note 1
FLOW RATE (gpm)	3300 (iv)	Note 1	3300 (ii) Note 3	Note 1	Note 1
NPSH REQ'D (ft)	N/A	Note 1	~~11.9 (iv)~~ N/A	Note 1	Note 1
PRESSURE DROP (psi)	N/A	Note 1	7.39 (iii) Note 2	Note 1	Note 1
HEAT LOAD (Btu/hr)	N/A	Note 1	Various	Note 1	Note 1
FLUID TYPE	Borated Water	Note 1	Borated Water	Note 1	Note 1
VISCOSITY (centipoise)	N/A	Note 1	0.56 (iii)	Note 1	Note 1
FLUID pH	N/A	Note 1	7.7 (vi)	Note 1	Note 1

FLOW IS FROM: 2-RS-E-1A TO: IRS Spray header inlet

NOTES: The values presented in the Table are estimates based on Unit 2 (Unit 2 design documentation is more complete) but is representative of Unit 1. Exact values for Unit 1 would have to be obtained from the appropriate reference. If not available, Unit 2 values may be used depending on application.

1. System on standby and only operates during LOCA/MSLB/REA/FWLB.
2. Does not include head loss due to elevation loss. Total system head loss due to elevation loss = 67.1 psi (ii).
3. Plant safety analysis uses 3300 gpm.

REFERENCES:

Unit 1:
(i) ______
(ii) [25.4.11]
(iii) ______
(iv) [25.6.1] pump curve
(v) [25.3.24]
(vi) [25.4.76]

Unit 2:
(i) [25.4.48]
(ii) [25.4.49]
(iii) [25.4.50]
(iv) [25.6.1] pump curve
(v) [25.3.25]
(vi) [25.4.76]

REVISION NO: 0 DATE: July 1, 1990

Table 14.1-1. Process Condition Summary

SYSTEM: Recirculation Spray (ORS,Train B)	SYSTEM CODE RS	SDBD NUMBER SDBD-NAPS-RS	NODE NO. 2

NODE TYPE: X MASS FLOW __ INTERFACE __ COMPONENT

PARAMETER	DESIGN VALUE	OPERATING CONDITIONS			
		NORMAL	EMERGENCY	SHUTDOWN	PUMP TEST
PRESSURE (psig)	55 (v)	Note 1	45 (i)	Note 1	Note 1
TEMPERATURE (°F)	285 (v)	Note 1	218.4 (i) Note 2	Note 1	Note 1
FLOW RATE (gpm)	3640 (iv)	Note 1	3640 (ii) Note 3	Note 1	Note 1
NPSH REQ'D (ft)	11.0 (vi)	Note 1	16.8 (vii) Note 6 15.3 (vii) Note 6	Note 1	Note 1
PRESSURE DROP (psi)	N/A	Note 1	1.97 (ii) Notes 4, 5	Note 1	Note 1
HEAT LOAD (Btu/hr)	N/A	Note 1	N/A	Note 1	Note 1
FLUID TYPE	Borated Water	Note 1	Borated Water	Note 1	Note 1
VISCOSITY (centipoise)	N/A	Note 1	0.3 (iii)	Note 1	Note 1
FLUID pH	N/A	Note 1	7.7 (viii)	Note 1	Note 1

FLOW IS FROM: Casing cooling system interface with ORS TO: 2-RS-P-2B

NOTES: The values presented in the Table are estimates based on Unit 2 (Unit 2 design documentation is more complete) but is representative of Unit 1. Exact values for Unit 1 would have to be obtained from the appropriate reference. If not available, Unit 2 values may be used depending on application.

1. System on standby, only operates during LOCA/MSLB/REA/FWLB.
2. Stress analysis to be done at 253°F (i.e., assuming failure of Casing Cooling System).
3. Plant safety analysis uses 3640 gpm.
4. Includes pressure drop for node 1.
5. Does not include head loss due to elevation loss. Total system head loss due to elevation loss = 66.7 psi (ii)
6. This value is minimum NPSH available. Note that NPSHA Unit 1 = 15.3 ft, Unit 2 NPSHA = 16.8 ft

REFERENCES:

Unit 1:	Unit 2:
(i) ______	(i) [25.4.48]
(ii) [25.4.12]	(ii) [25.4.51]
(iii) ______	(iii) [25.4.52]
(iv) [25.6.1] pump curve	(iv) [25.6.1] pump curve
(v) [25.3.24]	(v) [25.3.25]
(vi) [25.5.1]	(vi) [25.5.1]
(vii) ~~[25.4.2]~~ [25.4.139]	(vii) [25.4.2]
(viii) [25.4.76]	(viii) [25.4.76]

REVISION NO: 0 DATE: July 1, 1990

Table 14.1-1. Process Condition Summary

SYSTEM: Recirculation Spray (OPS Train B)	SYSTEM CODE RS	SDBD NUMBER SDBD-NAPS-RS	NODE NO. 3

NODE TYPE: X MASS FLOW ___ INTERFACE ___ COMPONENT

PARAMETER	DESIGN VALUE	OPERATING CONDITIONS NORMAL	EMERGENCY	SHUTDOWN	PUMP TEST
PRESSURE (psig)	150 (vii)	Note 1	153.3 (i) Note 5	Note 1	Note 1
TEMPERATURE (°F)	285 (vii)	Note 1	218.4 (i) Note 2	Note 1	Note 1
FLOW RATE (gpm)	3640 (vi)	Note 1	3640 (ii) Note 3	Note 1	Note 1
NPSH REQ'D (ft)	N/A	Note 1	N/A	Note 1	Note 1
PRESSURE DROP (psi)	N/A	Note 1	20.9 (iii) Note 4	Note 1	Note 1
HEAT LOAD (Btu/hr)	___	Note 1	Various	Note 1	Note 1
FLUID TYPE	Borated Water	Note 1	Borated Water	Note 1	Note 1
VISCOSITY (centipoise)	N/A	Note 1	0.3 (iii)	Note 1	Note 1
HEAT TRANSFER	3.82×10^6 (v)	Note 1	3.65×10^6 (iv)	Note 1	Note 1
FLUID pH	N/A	Note 1	7.7 (viii)	Note 1	Note 1

FLOW IS FROM: 2-RS-P-2B TO: 2-RS-E-1C

NOTES: The values presented in the Table are estimates based on Unit 2 (Unit 2 design documentation is more complete) but is representative of Unit 1. Exact values for Unit 1 would have to be obtained from the appropriate reference. If not available, Unit 2 values may be used depending on application.

1. System on standby, only operates during LOCA/MSLB/REA/FWLB.
2. Stress analysis to be done at 253°F (i.e., assuming failure of Casing Cooling System).
3. Plant safety analysis uses 3640 gpm.
4. Does not include head loss due to elevation loss. Total system head loss due to elevation loss = 56.7 psi (ii).
5. See [24.14.4].

REFERENCES:

Unit 1:
(i) ___
(ii) [25.4.12]
(iii) ___
(iv) ~~[25.4.4]~~ [25.4.139
(v) [25.6.9]
(vi) [25.6.1] pump curve
(vii) [25.3.24]
(viii) [25.4.76]

Unit 2:
(i) [25.4.48]
(ii) [25.4.51]
(iii) [25.4.52]
(iv) [25.4.4]
(v) [25.6.9]
(vi) [25.6.1] pump curve
(vii) [25.3.25]
(viii) [25.4.78]

REVISION NO: 0 DATE: July 1, 1990

Table 14.1-1. Process Condition Summary

SYSTEM: Recirculation Spray (RS Train B)	SYSTEM CODE RS	SDBD NUMBER SDBD-NAPS-RS	NODE NO. 6

NODE TYPE: X MASS FLOW ___ INTERFACE ___ COMPONENT

PARAMETER	DESIGN VALUE	OPERATING CONDITIONS NORMAL	EMERGENCY	SHUTDOWN	PUMP TEST
PRESSURE (psig)	150 (ix)	Note 1	111.6 (i)	Note 1	Note 1
TEMPERATURE (°F)	285 (ix)	Note 1	244.2 (i)	Note 1	Note 1
FLOW RATE (gpm)	3300 (viii)	Note 1	3300 (ii) Note 5	Note 1	Note 1
NPSH REQ'D (ft)	9.4 (vii)	Note 1	11.9 (vi) Note 4 10.4 (vi) Note 4	Note 1	Note 1 X
PRESSURE DROP (psi)	N/A	Note 1	12.43 (ii), (iii)	Note 1	Note 1
HEAT LOAD (Btu/hr)	—	Note 1	Various	Note 1	Note 1
FLUID TYPE	Borated Water	Note 1	Borated Water	Note 1	Note 1
VISCOSITY (centipoise)	N/A	Note 1	0.3 (iii)	Note 1	Note 1
HEAT TRANSFER	3.82×10^6 (v)	Note 1	3.55×10^6 (iv)	Note 1	Note 1
FLUID pH	N/A	Note 1	7.7 (x)	Note 1	Note 1

FLOW IS FROM: 2-RS-P-1B TO: 2-RS-E-1B

NOTES: The values presented in the Table are estimates based on Unit 2 (Unit 2 design documentation is more complete) but is representative of Unit 1. Exact values for Unit 1 would have to be obtained from the appropriate reference. If not available, Unit 2 values may be used depending on application.

1. System on standby, only operates during LOCA/MSLB/REA/FWLB.
2. Does not include head loss due to elevation loss. Total system head loss due to elevation loss = 67.1 psi (ii).
3. This data applies to the suction side of 2-RS-P-1B.
4. This value is minimum NPSH available. Note that Unit 1 NPSHA = 10.4 ft, Unit 2 NPSHA = 11.9 ft X
5. Plant safety analysis uses 3300 gpm.

REFERENCES:

Unit 1:	Unit 2:
(i) ______	(i) [25.4.48]
(ii) [25.4.11]	(ii) [25.4.49]
(iii) ______	(iii) [25.4.50] X
(iv) [25.4.4] [25.4.139]	(iv) [25.4.4]
(v) [25.6.9]	(v) [25.6.9] X
(vi) [25.4.2] [25.4.139]	(vi) [25.4.2]
(vii) [25.5.1]	(vii) [25.5.1]
(viii) [25.6.1] pump curve	(viii) [25.6.1] pump curve
(ix) [25.3.24]	(ix) [25.3.25]
(x) [25.4.76]	(x) [25.4.76]

REVISION NO. 0 DATE: July 1, 1990

14.2 RECIRCULATION SPRAY PUMPS

14.2.1 Basic Function

Requirement - In the event of an accidental release of high-energy fluids inside containment, the ORS and IRS pumps shall support the RS System function to cool, depressurize and reduce the levels of airborne iodine in containment. [25.1.1] [25.1.2] [25.10.1]

Design Feature - The RS System design includes two IRS and two ORS pumps which are provided to recirculate containment sump water from the containment sumps to the RS rings. An atomized spray is thus delivered into the containment atmosphere to cool and depressurize containment. The pH of the containment sump water (which is controlled by the QS System) facilitates the transfer of the airborne radioiodines to the sump. [25.8.1] [25.8.2]

14.2.2 Performance

Requirement - The ORS and IRS pumps, in combination with other RS System components, shall meet the requirements of the containment safety analysis to depressurize the containment within 1 hour after an accident and maintain the containment subatmospheric. [25.4.2] [25.4.4] [25.4.72] [25.4.139] [24.2.17]

Design Feature - In accordance with the containment safety analysis [25.4.2] [25.4.4] [25.4.72] [24.2.1], the RS pumps have the following design features:

[25.4.139]
[24.2.17]

1. The CDA signal activates timers (purchased via specification [25.6.30]) that delay the start of the recirculation pumps. The IRS pumps are delayed by 195 seconds and the ORS pumps by 210 seconds. [25.8.27] [25.8.11] [25.8.12] [25.8.28] [25.8.14] [25.8.15] [25.8.30] [25.8.31] The safety analysis uses a maximum delay time of 300 seconds for full recirculation spray. Calculation [25.4.111] indicates that full recirculation spray will be available and effective at 300 seconds following a DBA. [25.4.81] [25.7.42] [24.3.1] [24.11.8] [24.11.10]

2. The safety analyses assume an IRS and ORS flow of 3300 gpm and 3640 gpm, respectively, of containment sump water to the RS rings. [25.4.2] [25.4.4] [25.4.72] The IRS pumps and ORS pumps are purchased to be able to deliver 3300 gpm against a head of 275 ft and 3640 gpm against a head of 290 ft, respectively. [25.6.1] [24.9.1] (See pump curves attached to [25.4.11] [25.4.12] [25.4.51] [25.4.49].) The RS System design

results in an IRS flow rate of 3300 gpm against a head of 273 ft and an ORS flow rate of 3640 gpm against a head of 285 ft. [25.4.49] [25.4.51] [25.4.12] [25.4.11]

14.2.3 Regulations, Codes, and Standards

Requirement - The ORS and IRS pumps shall be designed, manufactured, examined, inspected, tested, and certified to generally recognized codes and standards or clearly stated quality requirements to assure a quality product in keeping with the required safety function. [25.1.11] [25.10.1] [25.1.34] [25.1.29] [25.5.4] [25.5.5] [25.1.25] [25.2.8] [25.1.54] [25.5.6]

Design Feature - The RS pumps are designed, manufactured, examined, inspected, tested, and certified to meet the Class II requirements of the Draft ASME Code for Pumps and Valves for Nuclear Power, November 1968. [25.6.1] [25.2.2] [25.10.9] [24.9.1]

The ORS pump casings are tested in accordance with the ASME Boiler and Pressure Vessel Code, Section III, Nuclear Vessels. [25.6.6] [24.9.6] [25.2.5]

The pumps and their motors are designed, engineered, manufactured, and inspected in accordance with the ANSI, IEEE, and NEMA standards applicable at the time of purchase of the pumps. [25.6.1] [24.9.1]

The RS pumps are located within the system safety-related boundary. [25.8.1] [25.8.2]

14.2.4 Design Conditions

Requirement - The ORS and IRS pumps shall be designed and purchased to be capable of withstanding the pressure, temperature, flow, and chemistry of the fluid passing through the system. [25.2.4] [25.2.8] [25.1.54] [25.5.6]

Design Features

[25.4.139] [24.2.17]

1. As indicated in Section 14.1.4 (process condition summary tables), the calculated pressure at the suction of the IRS and ORS pumps is 45 psig. [25.4.2] [25.4.4] [25.4.72] The calculated pressure at the discharge of the IRS and ORS pumps is 111.6 psig and 153.3 psig, respectively. [25.4.48] The ORS and IRS pumps are designed to 251 psig. [25.6.1] [24.9.1] Appropriate allowances are provided in the ORS pump casing to address the static head of water up to the check valve [25.8.87]. The ORS pump casings are therefore designed for a maximum pressure of 67.5 psig. [25.6.6] [24.9.6]

and ORS pumps are located. [25.8.66] [24.14.12] [25.6.28] [25.6.29] [24.14.78]

9. Requirement - In accordance with the requirements of the containment safety analyses, the ORS and IRS pumps shall be automatically activated when the containment pressure exceeds the value used in the containment analyses to initiate the RS System [25.1.1] [25.4.2] [25.4.4] [25.4.72]. To support the RS pumps NPSH analysis [25.4.2], [25.4.139 the availability of sump water [25.4.31] as well as the assumptions of the containment safety analyses, on receipt of this signal, the IRS pumps and ORS pumps start shall be delayed by 195 seconds and 210 seconds, respectively [25.4.2] [25.4.4] [25.4.72] [25.4.81]

[25.4.139]
[24.2.17]

Design Feature - The RPS system design includes initiation of the ORS and IRS pump delay timers on a containment 'high-high" pressure signal whose setpoint is based on the assumptions of the containment safety analyses. For details refer to Section 14.1.2, item 1a. The CDA signal activates timers that delay the start of the RS pumps (IRS by 195 seconds and ORS by 210 seconds) [25.8.27] [25.8.11] [25.8.12] [25.8.28] [25.8.14] [25.8.15] [25.8.30] [25.8.31]

10. Requirement - The emergency diesel generator loading sequence shall ensure that in the event of a LOOP coincident with or following an accident, the ORS and IRS pumps are powered within the time requirements of the safety analyses [25.4.2] [25.4.4] [25.4.72] [25.1.35] without overloading the emergency diesel [25.4.130] [24.11.10]

[25.4.139]
[24.2.17]

Design Feature - In the event of a LOOP coincident with the accident, the IRS and ORS pumps receive emergency power within the time requirements or the safety analyses. [25.4.111] [25.8.27] [25.8.28] [25.8.30] [25.8.31] [25.8.11] [25.8.12] 25.8.14] [25.8.15] Additional delay times are established for loading the pumps on the EP System following a LOOP. These delay times (ORS-35 seconds IRS-20 seconds) are implemented via additional timers and are necessary to avoid overloading the diesel. [25.8.27] [25.8.28] [25.8.30] [25.8.31] [25.8.11] [25.8.12] [25.8.14] [25.8.15] [25.4.130] [24.11.10]

For system-level interface requirements, see Section 14.1.7.

14.2.3 Material Requirements

Requirement - The ORS and IRS pumps shall be fabricated with materials compatible with the fluids carried in order to minimize corrosion or shall have a protective coating. [25.1.54] [25.2.8] [25.5.6]

Design Feature - The RS pumps are mounted on seismically designed foundations. The foundations are designed to resist operating and seismic loads. [25.4.1] [25.4.86] [25.4.87] [24.14.109]

14.2.11 Hydraulic Requirements

Requirement - To ensure proper operation of the RS System, the NPSH required by the ORS and IRS pumps shall be less than the NPSH available at the pumps at the maximum required flow rate. [25.1.22] [25.10.2] [25.2.9] [25.1.54] [25.5.6]

Design Feature - The ORS and IRS subsystems are designed to provide a flow rate of 3640 gpm and 3300 gpm, respectively. At this flow rate, the available NPSH at the ORS and IRS pumps suction is 16.8 ft and 11.9 ft, respectively, for Unit 2 [25.4.2] and the required NPSH at the ORS and IRS pump suction is 11 ft and 9.4 ft, respectively. [25.5.1] [24.12.1] The NPSHA is based on a post LOCA scenario with 150 gpm of 50°F RWST water being provided at the IRS pump suction and 800 gpm of casing cooling fluid (at 50°F) being injected into the ORS pump suction piping. [25.4.2] [25.8.1] [25.8.2]

and 15.3 ft and 10.4 ft, respectively, for Unit 1. [25.4.139]

[25.4.139]

Since the ORS and IRS Systems are maintained empty and dry during normal plant operation, when the pumps are started up following an accident, due to the lack of system resistance, the pumps will operate at runout flow (i.e., in excess of maximum required flow) for the period of time necessary to fill the system. This could result in NPSHA being less than NPSHR at runout flow. Per calculation [25.4.115] [25.4.2] [25.4.139] and the RS pumps test data, the pumps will suffer no damage due to this start-up transient. (This is a problem shared by all systems that are empty prior to startup. Usually, a closed valve in the system with a stroke time compatible with system fill time, would prevent the pump from operating at runout flow. However, to promote reliability, the RS System valves are maintained open.)

14.2.12 Chemistry and Sampling Requirements

Requirement - The fluids of the RS System shall be compatible with the function of the system and its supported systems. [25.2.8] [25.1.34] [25.5.6]

The chemistry of the RS System fluids shall be controlled and maintained so that the system performance requirements are met. Provisions shall be provided for sampling of RS fluids. [25.2.8] [25.1.54] [25.5.6]

Design Feature - There are no chemistry requirements for the RS pumps other than material compatibility addressed in Section 14.2.8. Refer to Section 14.1.12, item 2 for design features relative to sampling of RS fluids.

14.2.18 Redundancy, Diversity, and Separation

1. Requirement - The ORS and IRS pumps shall not be shared among nuclear units unless it can be shown that such sharing will not significantly impair safety functions, including, in the event of an accident in one unit, an orderly shutdown and cooldown of the remaining units. [25.1.15] [25.10.1]

   Design Feature - Two ORS pumps and two IRS pumps are provided for each reactor unit. There are no cross-connects between the units and no dependence on the opposite unit's RS pumps. [25.8.1] [25.8.2]

2. Requirement - Redundant ORS and IRS pumps shall be provided to ensure that under accident conditions, the minimum performance criteria can be met assuming single failure. [25.1.1] [25.1.37] [25.10.1]

   [25.4.139]

   Design Feature - The minimum performance criteria of the RS System will be met in the event of a loss of one train (i.e., one ORS pump and one IRS pump). [25.4.2] [25.8.1] [25.8.2] Refer to Section 11.1.4, item 5 for further information.

3. Requirement - In the event of a loss of offsite power, redundant IRS and ORS pumps shall be powered by separate emergency power trains such that each RS System train is powered independently. [25.1.1] [25.10.1]

   Design Feature - In the event of a loss of offsite power, one IRS pump and one ORS pump are powered from the same emergency diesel unit. The remaining IRS and ORS pumps are powered by a different emergency diesel unit. [25.8.4[illegible]] [25.8.47]

4. Requirement - Redundant ORS and IRS pumps shall be separated to ensure availability in case of damage due to internally generated missiles, pipe whip, or jet impingement from high-energy line breaks. [25.1.14] [25.10.1]

   Design Feature - The ORS pumps are located in separate cubicles at elevation 256 ft of the Safeguards Building. [25.8.9] This separation will ensure that at least one ORS pump will remain operational in the event of an internally generated missile. It should be noted that there are no high energy lines in this area. [25.10.46] [24.13.8]

   The IRS pumps are located without separating walls at elevation 217 ft of the containment in the containment recirculation sumps. [25.8.9] However, a hazards review was performed for all safety-related to depressurize the containment within 1 hour following an accident and maintain the containment sub-atmospheric. [25.4.2] [25.4.4] [25.4.72]

[25.4.139] [24.2.17]

Design Feature - In accordance with the containment safety analyses [25.4.4] [25.4.2] [25.4.72] [24.2.1], the RS coolers have the following design feature:

The IRS and ORS coolers are purchased to a heat-removal capability of 3.82 E6 Btu/hr-°F, assuming zero fouling resistance. [25.6.9] [24.9.10] Note that the safety analyses assume a minimum heat-removal capability for the ORS and IRS coolers to be 3.65 E6 Btu/hr-°F and 3.55 E6 Btu/hr-°F, respectively. [24.11.6] (In 1988, a Virginia Power letter [25.7.36] indicated that, due to excessive tube fouling, the IRS and ORS coolers heat removal coefficients had reduced to 2.16 E6 and 2.18 E6 Btu/hr-°F, respectively. Calculations [25.4.78] [25.4.79] were performed to evaluate the impact on containment analyses and imposed some restrictions on service water temperature until the next outage when the RS cooler tubes were cleaned to regain their design basis heat removal coefficients. [25.7.37] [25.7.38] [25.7.39] [25.7.40].)

14.3.3 Regulations, Codes, and Standards

Requirement - The RS coolers shall be designed, manufactured, examined, inspected, tested, and certified to generally recognized codes and standards or clearly stated quality requirements to assure a quality product in keeping with the required safety function. [25.1.11] [25.10.1] [25.1.34] [25.1.29] [25.5.4] [25.5.5] [25.1.25] [25.2.8] [25.1.54] [25.5.6]

Design Feature - The RS coolers are designed, manufactured, examined, inspected, tested, and certified to meet the Class C type vessel in Section III entitled "Nuclear Vessels" of ASME Unfired Boiler and Pressure Vessel Code, 1968. [25.2.16] [25.10.9] [25.6.9] [24.9.10]

The RS coolers are located within the system safety-related boundary. [25.8.1] [25.8.2]

14.3.4 Design Conditions

Requirement - The RS coolers shall be designed and purchased to be capable of withstanding the pressure, temperature, flow, and chemistry of the fluid passing through the system. [25.2.4] [25.2.8] [25.1.54] [25.5.6]

Design Features

1. Calculation [25.4.48] establishes the maximum operating temperatures and pressures throughout the RS System. Accordingly, the maximum operating pressures and temperatures expected in the RS coolers are:

Operating pressure		(psig)
Tube		0-100
Shell	ORS	150
	IRS	112

Note that the pressure of the RS fluid in the shell side of the cooler is maintained higher than the SW fluid in the tube side to prevent dilution of the borated sump water by the SW fluid.

Operating temperature		(°F)
Tube	IRS	95-176
	ORS	95-165
Shell	IRS	245-105 [24.14.122]
	ORS	218-105 [24.14.1[illegible]]

The RS coolers were purchased to withstand the above operating conditions in accordance with the specification [25.6.9] [24.9.10].

Design pressure	(psig)
Tube	150
Shell	150

Design temperature	(°F)
Tube	280
Shell	280

Note that the inlet/outlet fluid temperature versus time in the shell side of the RS coolers was calculated in [25.4.32] (which was based on Beaver Valley Power Station) and provided in specification [25.6.9]. [25.4.139] A review of the computer runs for the latest LOCA containment analyses for NAPS Units 1 and 2 [25.4.2] indicates that the results of calculation [25.4.32] are still representative. Unit

2. The safety analyses [25.4.2] [25.4.4] [25.4.72] [25.4.139] [24.2.17] require the ORS cooler to process 3640 gpm and the IRS cooler to process 3300 gpm of containment sump fluid. The RS cooler specification, [25.6.9] [24.9.10], indicates that the expected flow through the as-purchased equipment is 3500 gpm. [24.14.123]

3. The RS coolers are located above the maximum submergence level in containment. (Refer to Section 14.1.6.)

14.3.7 Interface Requirements

1. Requirement - Location of loads and forces imposed on structures due to RS cooler installation/operation shall be included in the associated building design. [25.2.8] [25.1.54] [25.5.6]

   Design Feature - The information in Sections 14.3.5, 14.3.9, 14.3.10, and 14.3.28 is used as structural interface design criteria to ensure that the containment building design includes the corresponding feature, e.g., shield walls, supporting beams, foundations, etc.

2. Requirement - The SW System shall provide cooling water to the RS coolers at the rate established in the safety analyses. The safety analyses also address a range of SW temperatures in conjunction with other parameters. [25.4.4] [25.4.139]

   Design Feature - The above requirement is provided to the SW System to ensure that the proper flow rate is obtained. [25.3.7] The acceptable range of SW temperatures in conjunction with other parameters as required by the safety analyses is controlled by Figure 3.6-1 of the plant Technical Specifications.

3. Requirement - Information related to the temperature of sump fluid being processed by the RS coolers, and insulation (if available) around equipment shall be provided to the building ventilation system to support the HVAC design of that cubicle. [25.1.28] [25.1.14]

   Design Feature - The only source of heat from RS cooler operation is the fluid it processes. The RS coolers do not operate during normal plant operation and, therefore, do not provide a heat load to the containment air cooling system. Following a Phase B isolation, the containment air cooling system does not operate [25.10.18] and the coolers are qualified to operate in the postaccident environment of the containment without the benefit of ventilation (see Section 14.3.6).

For system-level interface requirements, see Section 14.1.7.

14.3.8 Material Requirements

Requirement - The RS coolers shall be fabricated with materials compatible with the fluids carried in order to minimize corrosion or shall have a protective coating. [25.2.8] [25.1.54] [25.5.6]

[25.4.139]

Design Feature - The minimum performance criteria of the RS System will be met in the event of a loss of one train. [25.4.2] [25.8.1] [25.8.2]. Refer to Section 11.1.4, Item 5 for further information.

3. Requirement - in the event of a loss of offsite power, redundant RS System components important to safety shall be powered by separate emergency power trains such that each RS System train is powered independently. [25.1.[illegible]] [25.10.1]

   Design Feature - The above requirement is not applicable to coolers.

4. Requirement - Redundant RS coolers shall be separated to ensure availability in case of damage due to internally generated missiles, pipe whip, or jet impingement from high-energy line breaks. [25.1.[illegible]4] [25.10.1]

   Design Feature - The RS coolers are vertically supported between elevation 259 ft and elevation 216 ft of the containment in the pressurizer relief tank cubicle. (Refer to Section 14.1.5, Item 2 for design features relative to ensuring availability in the event of a HELB or internally generated missile.)

14.3.19 Reliability

Requirement - Failure modes and effects shall be analyzed for the RS System to ensure its availability following an accident. [25.2.8] [25.1.54] [25.5.6]

Design Feature - Refer to Section 14.1.19.

14.3.20 Test and Surveillance Requirements

Requirement - The RS coolers shall be designed to permit appropriate periodic inspection and testing of the mechanical and structural features associated with the coolers to assess and ensure the reliability of the coolers and its capability to continue to support its minimum performance requirements. [25.1.9] [25.1.10] [25.10.1] [25.1.11] [25.2.8] [25.1.54] [25.5.6] The RS coolers shall be included in the plant In-Service Inspection Program which incorporates the requirements of ASME XI. [25.1.46] [25.3.45] [25.3.52] [25.3.53] [25.2.21] [25.2.22]

Design Feature - After the original system startup tests, the RS coolers are maintained empty and dry to prevent fouling of the tubes. Normally closed MOVs are provided to ensure that the RS coolers are isolated from the SW System. [25.8.105] Vents and drains are provided to drain out the water and blow the tube dry with compressed air after periodic tests of the heat

14.3.27 Materials, Processes, Parts, and Equipment

Requirement - Special material, process, part, and equipment requirements necessary for RS cooler operation and maintenance shall be specified and controlled by plant procedures. [25.2.8] [25.1.54] [25.5.6]

Design Feature - Refer to Section 14.1.27.

14.3.28 Personnel Safety

Requirement - Onsite radiation exposure due to normal operation and maintenance of RS coolers shall be monitored and controlled such that operator doses are maintained ALARA. Postaccident Control Room doses shall be maintained within the limits of GDC 19. [25.1.21] [25.1.6] [25.1.39] [25.10.23] [25.3.23]

Special industrial safety requirements that impose design bases requirements on the RS coolers shall be specified. [25.2.8] [25.1.54] [25.5.6]

Design Feature - Refer to Section 14.1.28.

14.4 RECIRCULATION SPRAY RINGS AND NOZZLES

14.4.1 Basic Function

Requirement - In the event of an accidental release of high-energy fluids inside containment, the RS headers/nozzle shall support the RS System function to cool, depressurize and reduce the levels of airborne radioactive iodines in containment. [25.1.1] [25.1.2] [25.10.1]

Design Feature - The RS header/nozzle arrangement will spray the pH controlled containment sump water into the containment structure, thereby transferring the heat from the containment atmosphere/structure to the spray water, condensing the steam to reduce containment pressure and supporting the transfer of airborne radioactive iodine to the sump. [25.8.1] [25.8.2]

14.4.2 Performance

Requirement - The RS headers/nozzles in combination with other RS System components shall meet the requirements of the containment safety analyses to depressurize the containment within one hour following an accident and maintain the containment subatmospheric. [25.4.2] [25.4.4] [25.4.72] X

[25.4.139] [24.2.17]

[25.4.139]
[24.2.17]

Design Feature - In accordance with the containment safety analyses, [25.4.4] [25.4.2] [25.4.72] [24.2.1] to enhance containment heat and fission product removal, the RS rings and nozzles perform the following:

1. Produce droplet sizes such that the droplet thermal effectiveness is a minimum of 0.9. [25.4.46] [25.4 [illegible]] [25.8.8] [24.2.3] [25.4.30] [24.6.4]

2. Maximize spray coverage inside containment by:

   a. Locating the RS rings near the top of the dome. The IRS and ORS subsystems form an approximately 100-ft diameter, 360-degree ring at elevation 377 ft-10 in. and elevation 376 ft-10 in., respectively. [25.8.8] [25.8.1] [25.8.2] [25.4.29]

   b. Using two sizes of nozzles (1HH30100 and 1/2 B60) in the spray rings. [25.6.10] [24.2.6] The size and arrangement of these nozzles provide a spray pattern that maximizes coverage [25.4.62] [25.4.59] [25.4.31] [24.2.4] and supports droplet sizes in accordance with the calculations that generate droplet thermal effectiveness. [25.4.46] [24.2.3] [24.3.2]

It should be noted that spray nozzles are selected in various sizes to give the optimum combination of small spray particles for maximum heat transfer and larger particles for better coverage towards the center and sides of the containment.

14.4.3 Regulations, Codes, and Standards

Requirement - The RS headers/nozzles shall be designed, manufactured, examined, inspected, tested, and certified to generally recognized codes and standards or clearly stated quality requirements to assure a quality product in keeping with the required safety function. [25.1.11] [25.10.1] [25.1.34] [25.1.29] [25.5.5] [25.5.4] [25.1.25] [25.2.8] [25.1.54] [25.5.6]

Design Feature - The RS spray headers are essentially 8-inch piping. Refer to Section 14.1.3 for design features. The quality requirements for the RS nozzles were controlled by the requirements outlined in the specification. [25.6.10]

The RS spray headers/nozzles are located within the system safety-related boundary. [25.8.1] [25.8.2]

Design Feature - Refer to Section 14.1.17.

14.4.18 Redundancy, Diversity, and Separation

1. Requirement - The RS headers/nozzles shall not be shared among nuclear units unless it can be shown that such sharing will not significantly impair safety functions, including, in the event of an accident in one unit, an orderly shutdown, and cooldown of the remaining units. [25.1.15] [25.10.1]

   Design Feature - An ORS and an IRS ring are provided for each reactor unit. There are no cross-connects between the units and no dependence on the opposite unit's RS rings. [25.8.1] [25.8.2]

2. Requirement - Redundant RS headers/nozzles shall be provided to ensure that under accident conditions, the minimum performance criteria can be met, assuming single failure. [25.1.1] [25.1.37] [25.10.1]

   [25.4.139]

   Design Feature - The minimum performance criteria of the RS System will be met in the event of a loss of one train. [25.4.2] [25.8.1] [25.8.2] Refer to Sections 11.1.4, item 5 and 14.4.14 for further information.

3. Requirement - In the event of a loss of offsite power, redundant RS System components shall be powered by separate emergency power trains such that each RS System train is powered independently. [25.1.1] [25.10.1]

   Design Feature - The above requirement is not applicable to spray rings.

4. Requirement - Redundant RS headers/nozzles shall be separated to ensure availability in case of damage due to internally generated missiles, pipe whip, or jet impingement from high-energy line breaks. [25.1.14] [25.10.1]

   Design Feature - The RS rings are supported/restrained to ensure availability in the event of internally generated missiles, pipe whip, or jet impingement from high-energy line breaks. (See Section 14.4.5, item 2.) Separate cubicles cannot be provided for the RS rings since that would adversely affect the performance requirement to maximize spray coverage of containment atmosphere.

14.4.27 Materials, Processes, Parts, and Equipment

Requirement - Special materials, processes, parts, and equipment requirements necessary for RS spray header operation and maintenance shall be specified and controlled by plant procedures. [25.2.8] [25.1.56] [25.5.6]

Design Feature - Refer to Section 14.1.27.

14.4.28 Personnel Safety

Requirement - Onsite radiation exposure due to normal operation and maintenance of the RS spray headers shall be monitored and controlled such that operator doses are maintained ALARA. Postaccident Control Room doses shall be maintained within the limits of GDC 19. [25.1.21] [25.1.6] [25.1.39] [25.10.23] [25.3.23] Special Industrial Safety Requirements that impose design basis requirements on the RS spray headers shall be specified. [25.2.8] [25.1.56] [25.5.56]

Design Feature - Refer to Section 14.1.28.

14.5 CASING COOLING TANK

14.5.1 Basic Function

Requirement - In the event of an accidental release of high-energy fluids inside containment, the CCT shall support the RS System function to cool, depressurize and reduce the levels of airborne radioactive iodines in containment. [25.1.1] [25.1.2] [25.10.1]

[25.4.139]
[24.2.17]

Design Feature - To ensure that the ORS pumps perform in accordance with the containment safety analyses [25.4.2] [25.4.4] ~~[25.4.72]~~ and in accordance with the ORS pump NPSH analyses [25.4.2], a dedicated source of cold water is provided in the form of the casing cooling water supply from the CCT. To support the postaccident ORS pump NPSH requirements, cold water from the CCT is injected into the ORS pump suction to increase the NPSH available to the pumps. [25.8.1] [25.8.2] (It should be noted that proper operation of the ORS pumps also supports the airborne iodine removal function assigned to the RS System.)

[25.4.139]

14.5.2 Performance

Requirement - The CCT in combination with the other RS System components shall meet the requirements of the containment safety analyses to depressurize the containment within 1 hour following an accident and maintain the containment at subatmospheric conditions. [25.4.2] [25.4.4] ~~[25.4.72]~~

[25.4.139]
[24.2.17]

Design Feature - To ensure that the ORS pumps perform in accordance with the safety analyses [25.4.2] [25.4.4] [25.4.72] [24.2.1] and in accordance with the requirements of the ORS pump NPSH analyses [25.4.2], the CCT has the following design features. [25.4.139]

1. A useable volume of water of 113,904 gal. [25.4.110] The Technical Specifications ensure that a volume of 116,500 gal of water is present in the tank during Modes 1, 2, 3, and 4. [25.3.12] [25.3.11]

2. The temperature of the fluid stored in the CCT is maintained between 35°F and 50°F. [25.3.12] [25.3.11] (Note that the lower limit on the temperature is to prevent freezing of the CCT fluid.)

14.5.3 Regulations, Codes, and Standards

Requirement - The CCT shall be designed, manufactured, examined, inspected, tested, and certified to generally recognized codes and standards or clearly stated quality requirements to ensure a quality product in keeping with the required safety function. [25.1.11] [25.10.1] [25.1.34] [25.1.29] [25.5.4] [25.5.5] [25.1.25] [25.2.8] [25.1.54] [25.5.6]

Design Feature - The code used in design, purchase, and installation of the CCT and its supports (foundation) was the modified API Standard 650, "Welded Steel Tanks for Oil Storage." [25.6.3] [24.9.4] [24.9.3] [25.2.3]

The CCT is located within the system safety-related boundary.

14.5.4 Design Conditions

Requirement - The CCT shall be designed and purchased to be capable of withstanding the pressure, temperature, flow, and chemistry of the fluid passing through the system. [25.2.4] [25.2.8] [25.1.54] [25.5.6]

Design Feature [25.4.139] X

1. The ORS pump NPSH analysis [25.4.2] establishes a maximum allowable fluid temperature of 50°F. The tank is purchased to a design temperature of 150°F. [25.8.59]

2. In accordance with the NPSH analysis, the CCT is an atmospheric tank. The purchased tank is atmospheric and is designed to internal pressures due to liquid rising to the overflow nozzle [25.6.3] and to a design pressure of 0.5 psi vacuum. [25.8.59]

Design Feature

[25.4.139]

1. The CCT is required to be operable for 1 hour following the accident. [25.4.2] The CCT is located out in the yard [25.8.60], which is Zone Yard-1 of the EZD. [25.3.1] [24.14.130] Consequently, the tank is designed to ambient exterior conditions (i.e., pressure, temperature, humidity) and conservatively designed to a 0-inch corrosion allowance. [25.8.59] [24.13.11]

2. The CCT is not qualified to electromagnetic levels since none are identified for the areas in which the Units 1 and 2 CCTs are located.

3. The CCT is located in the yard and is above the external flood level. (Refer to Section 14.1.5.)

14.5.7 Interface Requirements

1. Requirement - Location of loads and forces imposed on structures due to CCT installation/operation shall be included in the associated building design. [25.2.8] [25.1.54] [25.5.6]

   Design Feature - The CCT is supported/anchored on the extension of the Casing Cooling Pumphouse foundation. The information, as discussed in Sections 14.5.5, 14.5.9, 14.5.10, and 14.5.28, is used as structural interface design criteria to ensure that the extended foundation of the Casing Cooling Pumphouse is designed to include the features addressed in the above sections.

2. Requirement - The boration level of the fluid in the CCT shall be the same as the RWST. [25.4.76]

   Design Feature - Initial fill and makeup water to the CCT is provided from the RWST via the use of a hose connection. [25.3.8] [24.5.3]

For system-level interface requirements/design features, see Section 14.1.7.

14.5.8 Material Requirements

Requirement - The CCT shall be fabricated with materials compatible with the fluids/gases carried in order to minimize corrosion or shall have a protective coating. [25.2.8] [25.1.54] [25.5.6]

The CCT shall be fabricated with materials compatible with the environment and physical interfaces to prevent electrolysis and other chemical reactions or shall have a protective coating. [25.2.8] [25.1.54] [25.5.6]

1. The casing cooling pumps are in Zone CCPH-1 (i.e., mild environment) of the Environmental Zone Description (EZD). [25.3.1] [25.8.61] [24.14.137] The casing cooling pumps are required to be operable for about one hour postaccident in accordance with the safety analyses. [25.4.139] [25.4.2] The casing cooling pumps and motors are purchased to be capable of operating in the normal operating and accident environmental conditions of the Casing Cooling Pumphouse. [25.6.8] [24.14.138] [24.13.11] (Refer to Section 14.1.6.)

2. The casing cooling pumps are not qualified to electromagnetic levels since none are identified for the areas where the pumps are located.

3. The casing cooling pumps are located in the Casing Cooling Pumphouse where internal flooding is not applicable. (Refer to section 14.1.6.)

14.6.7 Interface Requirements

1. Requirement - The EP System shall be provided with the casing cooling pump electrical motor data listed in the Design Feature below. These data are required to produce the following EP System calculations. [25.1.1] [25.1.35]

   a. Load calculation
   b. Cable-sizing calculation
   c. Short-circuit calculation
   d. Breaker-sizing and protection coordination calculation
   e. Voltage profile calculation

   Design Feature - The following information will be obtained from the casing cooling pump motor data sheets attached to the specification. [25.6.8] [24.14.139]

   a. Full-load current
   b. Rated voltage
   c. Locked-rotor current
   d. Power factor
   e. Motor efficiency
   f. Revolutions per minute (rpm)
   g. Horsepower (hp)
   h. Minimum starting voltage

2. Requirement - The EP System shall be provided with the following circuit-breaker data. [25.1.1] [25.1.35]

Design Feature - Refer to Section 14.1.28.

14.6 CASING COOLING PUMPS

14.6.1 Basic Function

Requirement - In the event of an accidental release of high-energy fluids inside containment, the casing cooling pumps shall support the RS System function to cool, depressurize and reduce the level of airborne radioactive iodines in containment. [25.1.1] [25.1.2] [25.10.1]

[25.4.139]
[24.2.17]

Design Feature - To ensure that the ORS pumps perform in accordance with the containment safety analyses [25.4.2] [25.4.4] [25.4.72] [24.2.1] and the ORS pump NPSH analyses [25.4.2], casing cooling pumps are provided to inject cold water (obtained from the CCT) into the ORS pump suction to increase the NPSH at the ORS pump suction. [25.8.1] [25.8.2] (It should be noted that proper operation of the ORS pumps also support the airborne iodine removal functions assigned to the RS System)

[25.4.139]

14.6.2 Performance

Requirement - The casing cooling pumps in combination with other RS System components shall meet the requirements of the containment safety analyses to depressurize the containment within one hour following an accident and maintain the containment at subatmospheric condition. [25.4.2] [25.4.4] [25.4.72]

[25.4.139]

Design Feature - To support ORS pump operation in accordance with the requirements of the safety analyses, the ORS pump NPSH requirements have to be met. The ORS pump NPSH analysis [25.4.2] assumes a flow of 800 gpm of casing cooling liquid injected into the ORS pump suction for one hour starting immediately after the accident. With the injection of the casing cooling liquid, the minimum available NPSH for the ORS pumps occurs at 650 seconds following an accident.

The casing cooling pumps that perform this injection function are purchased to be able to deliver 960 gpm against a head of 101 ft. [25.4.53] [25.4.57] [25.4.82] [25.4.83] Via a recirculation line, 160 gpm is diverted back to the CCT. [25.8.1] [25.8.2] A restriction orifice in the injection line maintains the flow at 800 gpm. [25.4.53] [25.4.57] Calculation [25.4.61] indicates that a flow of 800 gpm is reached at 610 seconds [24.6.2], following an accident. Drawings [25.8.1] [25.8.2] show that the casing cooling pumps are started up by a CDA signal.

encourage drainage, and floor surface conditions (roughness, paint requirements, etc.) to reduce contamination (if applicable). [25.11.1]

Design Feature - Proper drainage facilities and floor surface conditions are provided in the Casing Cooling Pumphouse where the casing cooling pumps are located. [25.6.29] [24.14.78] [25.8.106]

6. Requirement - In accordance with the requirements of the containment safety analyses, the casing cooling pumps shall be automatically activated when the containment pressure exceeds the value used in the containment analyses to initiate the RS System. [25.1.1] [25.4.2] [25.4.4] [25.4.72]

[25.4.139]
[24.2.17]

Design Feature - The RPS System design includes initiation of the casing cooling pumps by the CDA signal whose setpoint is based on the assumptions of the containment safety analyses. [25.8.1] [25.8.2] [25.4.2] [25.4.4] [25.4.72] [25.3.11] [25.3.12]

For system-level interface requirements/design features, see Section 14.1.7.

14.6.8 Material Requirements

Requirement - The casing cooling pumps shall be fabricated with materials compatible with the fluids/gases carried in order to minimize corrosion or shall have a protective coating. [25.2.8] [25.1.54] [25.5.6]

The casing cooling pumps shall be fabricated with materials compatible with the environment and physical interfaces to prevent electrolysis and other chemical reactions or shall have a protective coating. [25.2.8] [25.1.54] [25.5.6]

Design Feature - Selection of the casing cooling pumps' materials is based on the service conditions identified in Sections 14.6.4, the environmental conditions identified in Section 14.6.6, and casing cooling fluid chemistry addressed in Section 14.1.12.

The casing cooling pumps are made of austenitic stainless steel or materials of equal corrosion resistance. [25.6.8] [24.9.2]

Interfacing piping systems are fabricated of material similar to those of the pumps based on the galvanic scale, i.e., the pumps as well as the piping are made of stainless steel. (Refer to Section 14.1.8.)

14.6.16 Instrumentation and Control Requirements

Requirement - Instruments and controls, interlocks and permissives shall be provided to monitor and control casing cooling pump operation during normal and accident conditions and to maintain the system within prescribed operating ranges. [25.1.16] [25.10.1] [25.1.54] [25.5.6] [25.2.8]

Design Feature - Refer to Section 14.1.16.

14.6.17 Access, Administrative Control, and Security

Requirement - The access control, carding, and zoning of cubicles housing the casing cooling pumps shall be in accordance with the requirements identified in 10CFR73 and 10CFR50.54(p) and incorporated by Virginia Power in the plant-specific security program. [25.1.36] [25.3.37] [25.5.1] [25.5.5] [25.2.8] [25.1.54] [25.5.6] [25.1.56]

Design Feature - Refer to Section 14.1.17.

14.6.18 Redundancy, Diversity, and Separation

1. Requirement - The casing cooling pumps shall not be shared among nuclear units unless it can be shown that such sharing will not significantly impair safety functions, including, in the event of an accident in one unit, an orderly shutdown and cooldown of the remaining units. [25.1.15] [25.10.1]

   Design Feature - Two casing cooling pumps are provided for each reactor unit. There are no cross-connects between the units and no dependence on the opposite unit's casing cooling pumps. [25.8.1] [25.8.2]

2. Requirement - Redundant casing cooling pumps shall be provided to ensure that under accident conditions, the minimum performance criteria can be met, assuming single failure. [25.1.1] [25.1.37] [25.10.1]

   Design Feature - The minimum performance criteria of the RS System will be met in the event of a loss of one train (i.e., one ORS pump with its associated casing cooling pump and one IRS pump). [25.4.2] [25.4.139] [25.8.1] [25.8.2] Refer to Section 11.1.4, item 5 for further information.

3. Requirement - In the event of a loss of offsite power, redundant casing cooling pumps shall be powered by separate emergency power trains so that each RS System train is powered independently. [25.1.1] [25.10.1]

14.6.28 Personnel Safety

Requirement - Onsite radiation exposure due to normal operation and maintenance of the casing cooling pumps shall be monitored and controlled such that operator doses are maintained ALARA. Postaccident Control Room doses shall be maintained within the limits of GDC 19. [25.1.21] [25.1.6] [25.1.39] [25.10.23] [25.3.23] Special Industrial Safety Requirements that impose design bases requirements on the casing cooling pumps shall be specified. [25.2.8] [25.1.54] [25.5.6]

Design Feature - Refer to Section 14.1.28.

14.7 CCT RECIRCULATION PUMPS AND REFRIGERATION UNITS

14.7.1 Basic Function

Requirement - In the event of an accidental release of high-energy fluids inside containment, the RS System components shall support the RS System function to cool, depressurize and reduce the level of airborne radioactive iodine in containment. [25.1.1] [25.1.2] [25.10.1]

[25.4.139]
[24.2.17]

Design Feature - To ensure that the ORS pumps perform in accordance with the containment safety analyses [25.4.2] [25.4.4] [25.4.72] [24.2.1] and in accordance with the ORS pump NPSH analyses [25.4.2], the temperature of the fluid in the CCT has to be maintained within the allowable limits of the NPSH analysis.

[25.4.139]

To support the above function, CCT recirculation pumps are provided which, during normal plant operations, circulate the CCT fluid through casing cooling refrigeration units and keep the fluid at the required temperature. [25.8.1] [25.8.2]

14.7.2 Performance

Requirement - The RS System components shall meet the requirements of the containment safety analyses to depressurize the containment within 1 hour following an accident and maintain the containment at subatmospheric conditions. [25.4.2] [25.4.4] [25.4.72]

Design Feature - To support ORS pump operation in accordance with the requirements of the safety analyses, the ORS pump NPSH requirements have to be met. The ORS pump NPSH analysis [25.4.2] requires the casing cooling system fluid to be maintained at no greater than 50°F.

1. The refrigeration units are purchased to deliver 6 tons of refrigeration, and handle 26 gpm of borated fluid. With an inlet fluid temperature of 45°F, the outlet fluid temperature will be 39°F. [25.6.13] [24.9.13] [25.7.16] In accordance with references [25.4.35] [25.4.65] [25.7.17] [24.14.98] assuming worst ambient/inlet conditions, one 6-ton chiller will be able to reduce the CCT temperature by 0.9°F/day.

2. The CCT recirculation pumps will provide a maximum flow rate of 20 gpm to the chillers. [25.6.14] [24.14.46] [24.14.7] When the water in the CCT reaches 45°F, the pumps receive a signal from the CCT temperature indicators to automatically start up and circulate the tank fluid through the refrigeration units. Similarly, when the tank fluid temperature reaches 42°F, the pumps are automatically shut off. [25.3.47]

14.7.3 Regulations, Codes, and Standards

Requirement - The CCT recirculation pumps and refrigeration units shall be purchased to codes and standards commensurate with the quality requirements of their required functions. [25.1.54] [25.5.6] [25.2.8] [25.12.10]

Design Feature - The CCT recirculation pumps and refrigeration units are not a part of a radioactive fluid boundary (slight contamination of the CCT fluid will occur due to make up from the RWST, but the radioactivity level will not be significant), are not required for electric generation, nor do they have any postaccident mitigation function. Consequently, the equipment is purchased as non-safety-related [25.6.13] [25.6.14] [24.9.13] [24.14.46] and are not fabricated/installed to any specific industrial code or standard other than those listed in the purchase orders (e.g., the refrigeration units were purchased according to the National Electric Code [25.2.19] and the ANSI and ARI standards applicable at the time of purchase) to ensure an adequately reliable and quality product. The CCT recirculation pumps and refrigeration units are within the system non-safety-related boundary. [25.8.1] [25.8.2]

14.7.4 Design Conditions

Requirement - The CCT recirculation pumps and refrigeration units shall be designed and purchased to be capable of withstanding the pressure, temperature, flow, and chemistry of the fluid passing through the system. [25.2.4] [25.2.8] [25.1.54] [25.5.6]

Design Feature - In accordance with the requirements of the ORS pumps NPSH analyses [25.4.2] [25.4.139] and the Technical Specifications [25.3.11] [25.3.12], 14.8.2 Performance

Requirement - The RS System MOVs in combination with other RS System components shall meet the requirements of the containment safety analysis to depressurize the containment within 1 hour following an accident and to maintain the containment at subatmospheric conditions; and provide containment isolation capability in accordance with the acceptable limits defined in 10CFR50, Appendix J. [25.1.19] [25.1.53] [25.10.1] [25.5.4] [25.5.5] [25.4.2] [25.4.4] [25.4.72]

[25.4.139]

Design Feature - To support RS System operation in accordance with the assumptions of the safety analyses [25.4.2] [25.4.4] [25.4.72] [24.2.1] and the ORS pump NPSH analysis [25.4.2], the RS MOVs have the following design features.

[25.4.139]
[24.2.17]

1. The ORS pump suction isolation MOVs are normally in the open position. They receive an "open" signal after initiation of a CDA to ensure their open status and allow flow of containment sump water to the ORS pump suction. If closed, these MOVs will open within 30 seconds after receiving an open signal [25.6.19] [24.9.7]. This opening time is acceptable since the ORS pump start up is delayed by 210 seconds following a CDA. (See Section 14.2.2.) These MOVs also serve as containment isolation valves and can be isolated in case the associated ORS pump is being taken out of service, following an accident. [25.8.1] [25.8.2]

2. The ORS pump discharge isolation MOVs are normally in the open position. They receive an "open" signal upon initiation of a CDA signal to ensure their open status and allow flow containment sump water from the ORS pump discharge to the RS coolers. If closed, these MOVs will open within 30 seconds upon receiving an open signal. [25.6.19] [24.9.7] This opening time is acceptable since the ORS pump start up is delayed by 210 seconds following a CDA. (See Section 14.2.2.) These MOVs also serve as containment isolation valves and can be isolated in case the associated ORS pump is being taken out of service, following an accident. [25.8.1] [25.8.2]

3. Casing cooling pump discharge isolation MOVs (1-RS-MOV-101A, B; 2-RS-MOV-201A, B) are normally in the open position. They receive an "open" signal upon initiation of a CDA signal to ensure their open status and allow flow of water from the CCT to the ORS pump suction line. [25.8.1] [25.8.2] If closed, these MOVs will open within 30 seconds upon receiving an open

Design Feature - Proper drainage facilities and floor surface conditions provided in the Safeguards Building/valve pit area where the RS System MOVs are located. [25.6.29] [24.14.78] [25.8.66].

6. Requirement - In accordance with the requirements of the containment safety analyses, the RS System MOVs shall receive an "open" signal when the containment pressure exceeds the value used in the containment analyses to initiate the RS System. [25.1.1] [25.4.2] [25.4.4] [25.4.72]

[25.4.139]
[24.2.17]

Design Feature - The RPS System design includes initiation of an "open" signal to the RS System MOVs by the CDA signal, whose setpoint is based on the assumptions of the containment safety analyses [25.8.1] [25.8.2] [25.4.2] [25.4.4] [25.4.72] [25.3.11] [25.3.12]

For system-level interface requirements design features, see Section 14.1.7.

14.8.8 Material Requirements

Requirement - The RS System MOVs shall be fabricated with materials compatible with the fluids/gases carried in order to minimize corrosion or shall have a protective coating. [25.2.8] [25.1.54] [25.5.6]

The RS System MOVs shall be fabricated with materials compatible with the environment and physical interfaces to prevent electrolysis and other chemical reactions or shall have a protective coating. [25.2.8] [25.1.54] [25.5.6]

Design Feature - Selection of the material for the RS and casing cooling MOVs are based on the service conditions identified in Sections 14.8.4, the environmental conditions identified in Section 14.8.6, and the sump water and casing cooling fluid chemistry, respectively, addressed in Section 14.1.12.

The RS MOVs are made of stainless steel which is corrosion resistive. [25.6.19] [24.9.7] [25.6.20] [24.9.8] [24.9.9] [25.6.22] [25.6.34] [24.9.14] Valve seating, valve stem materials, etc., are made of corrosion/erosion resistive material also. [25.6.19] [24.9.7] [25.6.20] [24.9.8] [24.9.9] [25.6.34] [24.9.14]

Interfacing piping systems are fabricated of material similar to those of the RS System MOVs based on the galvanic scale, i.e., the MOVs as well as the piping are made of stainless steel. (Refer to Section 14.1.8.)

Direct physical interfaces between dissimilar metals are prevented via the NAPS piping and mechanical equipment installation specification. [25.6.32]

14.8.17 Access, Administrative Control, and Security

Requirement - The access control, carding, and zoning of cubicles housing the RS System MOVs shall be in accordance with the requirements identified in 10CFR73 and 10CFR50.54(p) and incorporated by Virginia Power in the plant-specific security program. [25.1.36] [25.3.37] [25.5.4] [25.5.5] [25.2.8] [25.5.6] [25.1.54] [25.1.56]

Design Feature - Refer to Se on 14.1.17.

14.8.18 Redundancy, Diversity, and Separation

1. Requirement - Safety-related RS System MOVs shall not be shared among nucle   s unless it can be shown that such sharing will not significantly   safety functions, including, in the event of an accident in the unit, an orderly shutdown and cooldown of the remaining units. [25.1.15] [25.10.1]

   Design Feature - Two ORS pumps disc rge and suction MOVs and two pairs of casing cooling pump discharge MOVs are provided for each reactor unit. There are no cross-connects between the units and no dependence on the opposite unit's RS MOVs. [25.8.1] [25.8.2]

2. Requirement - Redundant RS System MOVs shall be provided to ensure that under accident conditions, the minimum performance criteria can be met, assuming single failure. [25.1.30] [25.1.26] [25.1.1] [25.1.37] [25.10.1]

   Design Feature - The minimum performance criteria of the RS System will be met in the event of a loss of one train. [25.4.2] [25.4.139] [25.4.5] [25.8.1] [25.8.2] (Refer to Section 11.1.4, item 5, for further information.)

   The RS MOVs also provide containment isolation function. (Refer to Section 14.1.18, item 4 for details).

3. Requirement - In the event of a loss of power, redundant RS System MOVs important to safety shall be powered by separate emergency power trains such that each RS System train is powered independently. [25.1.1] [25.10.1]

   Design Feature - The RS MOVs are powered in accordance to the above requirement. (See Section 14.8.13.)

4. Requirement - Redundant RS System MOVs shall be separated to ensure availability in case of damage due to internally generated

(e.g., paint, thermal insulation, etc.) shall, to the extent practical, prohibit the creation of sump debris and corresponding sump screen blockage following an accident. [25.12.2]

Design Basis Implication

a. Qualified paint is used inside containment (to the extent practical) to minimize the potential of sump screen blockage by dislodged paint chips. [25.6.15] [24.11.3] Surfaces painted with unqualified paint are covered with a stainless steel screen to retain the dislodged paint chips. [24.11.4]

b. Thermal insulation used inside containment resists, to a degree, shredding and transport to the sump screens. [25.6.16] [24.11.5] It should be noted that all exposed insulation in containment is jacketed with stainless steel or silicone rubber impregnated fiberglass cloth encased in a stainless steel mesh and is designed to withstand a following a LOCA environment. [25.5.34]

Subsequent to SG replacement at NAPS-1 (and the use of fiberglass insulation) head loss due to post LOCA debris accumulation in the emergency sump was analyzed and factored into the NAPS-1 LOCA analysis [25.4.138] [25.4.139] [25.12.11]

3. Constraint - To ensure heat transfer coefficients in accordance with the assumptions of the containment safety analyses [25.4.2] [25.4.4] [25.4.72], the RS coolers shall be maintained clean, empty, and dry, thus preventing fouling of the tubes.

Design Basis Implication - Normally closed MOVs are provided to ensure that the RS coolers are isolated from the SW System. [25.8.105] Vents and drains are provided to drain out the water and blow the tubes dry with compressed air after periodic tests of the heat exchanger SW inlet and outlet valves. [25.8.1] [25.8.2] [25.5.12] In accordance with Virginia Power's commitment [25.5.26], the RS coolers are inspected weekly for service water leakage. The vents and drains addressed above support this inspection.

CHAPTER 16 TABLE OF CONTENTS

1. SWEC Calculation 11715-ES-150, Rev. 2, "Loctic Input (Containment Integrity)," August 2, 1979.

2. SWEC Calculation 14938.37-US(B)-259, Rev. 0, "LOCA Analysis for Revir Technical Specification on Containment Air Partial Pressure," May 28, 1987.

3. SWEC Calculation 14938.37-US(B)-260, Rev. 0, "Main Steam Line Break Analysis," May 28, 1987.

4. SWEC Calculation 14938.44-UR(B)-012-0 "North Anna LOCA Dose in Control Room," April 22, 1988.

5. SWEC Calculation 11715-RP-A101-0, "DBA Thyroid Dose at E.B., N.O.R. and LPZ," August 30, 1972.

6. SWEC Calculation 14237.02-UR(B)-001-1, "Determination of Accident Analysis Results after Accounting for 2% Safety Margin for Instrument Error," March 21, 1984.

7. SWEC Calculation, 14799.02-UR(B)-001-0, "Update of UFSAR Accident Dose To Account for 2% Instrument Error," March 22, 1984.

8. Virginia Power Calculation SM-429, "Shutdown Reactivity Following a LOCA," October 19, 1986.

9. SWEC Calculation 02072.2010-US(B)-274-0, "Containment LOCA Analysis with New Steam Generators", June 17, 1992

16.1 ORS AND IRS PUMP NPSH ANALYSIS (UNIT 2)

Calculation Number, Revision, and Date - SWEC Calculation 14938.37-US(B)-259, Rev. 0, May 28, 1987.

Calculation Title - "LOCA Analysis For Revised Technical Specifications on Containment Air Partial Pressure."

Purpose - To document the pressure/temperature transients resulting from a LOCA in the reactor containment and the available NPSH for the RS and LHSI pumps. (It should be noted that the NAPS Units 1 and 2 LOCA analysis (Calculation 14938.37-US(B)-259-0) includes the RS and LHSI pump NPSH analyses. The RS NPSH portion of the LOCA analysis is summarized in this section.)

Assumptions - Except as noted all assumptions/data are the same as used in the LOCA analyses.

1. Initial pressure (psia):
   a. Total — 9.75 - 10.59
   b. Air partial — 8.9
2. Initial temperature (°F):
   a. Dry bulb — 120
   b. Dewpoint — 96 - 120
3. Initial service water temp (°F) — 35-97
4. RWST temperature — 40 - 50
5. Containment free volume ($10^6 ft^3$) — 1.825 1.916 (include 5% margin) X
6. Spray thermal effectiveness (5) — 100
7. LHSI switchover to sump (gallons removed from RWST) — 315,252
8. Containment effective diameter (ft) — 121.1
9. Difference between average floor elevation and centerline of first-stage impeller:
   a. Inside RS (ft) — 6.9
   b. Outside RS (ft) — 6.8
   c. LHSI (ft) — 6.2
10. Recirculation phase suction friction loss:
    a. Inside RS — 1.1 ft at 3300 gpm
    b. Outside RS — 4.9 ft at 3640 gpm (2840 gpm from sump and 800 gpm from casing cooling system)
    c. LHSI — 9.2 ft at 4030 gpm/pump (minimum ESF)

Assumptions - The following assumptions were used in the calculation.

1. Worst ambient conditions, 93°F with 7 1/2 mph wind.
2. Negligible resistance to heat transfer through metal tank wall to water.
3. Heat gain due to solar is one-third of total steady state heat gain.
4. Tank contains 120,000 gallons of water at 45°F.
5. At steady state all energy in air space goes into the water.

Inputs - The following sources were used as inputs to the calculation.

1. NAS-191, Rev 3, Field Fabricated Tanks.
2. NAS-54, Rev 4, Thermal Insulation.
3. Pittsburgh Corning Information for Foam Glass Insulation.
4. P.O. NA-531/1531 CCT Chiller.

Methodology - The calculation is performed in three parts. Assuming worst ambient conditions, the following calculations are done in Btu/hr.

1. Heat gain through tank wall to the water.
2. Heat gain through tank roof and wall to the tank air space.
3. Solar heat gain which is assumed to be one-third of the sum of items (a) and (b) above.

The total tank heat gain is developed by adding the above three contributions. The sensible heat removal rate is determined by subtracting the tank heat gain from the as-purchased chiller capacity.

Conclusion - The maximum steady state heat gain by the CCT is 33,549 Btu/hr. The as-purchased chiller is capable of removing 0.92°F per day of sensible heat under worst ambient conditions.

Insert 16·16

Insert 16·17

INSERT 16.16

16.16 ORS AND IRS PUMP NPSH ANALYSIS (UNIT 1)

Calculation Number, Revision and Date - SWEC Calculation 02072.2010-US(B)-274, Rev 0, June 17, 1992.

Calculation Title - "Containment LOCA Analysis With New Steam Generator."

Purpose - To document the pressure/temperature transients resulting from a LOC[illegible] in the reactor containment and the available NPSH for the RS and LHSI pumps. (It should be noted that the NAPS Unit 1 LOCA analysis (Calculation 02072.2010-US(B)-274-0) includes the RS and LHSI pump NPSH analyses. The RS NPSH portion of the LOCA analysis is summarized in this section.)

Assumptions - Except as noted all assumptions/data are the same as used in the LOCA analyses.

1. Initial pressure (psia):

a.	Total	9.75 - 10.59
b.	Air partial	8.9

2. Initial temperature (°F):

a.	Dry bulb	120
b.	Dewpoint	120

3.	Initial service water temp (°F)	35-99
4.	RWST temperature	50
5.	Containment free volume ($10^6 ft^3$)	1.916 (include 5% margin)
6.	Spray thermal effectiveness (5)	10[illegible]
7.	LHSI switchover to sump (gallons removed from RWST)	341,377
8.	Containment effective diameter (ft)	121.1

9. Difference between average floor elevation and centerline of first-stage impeller:

a.	Inside RS (ft)	6.9
b.	Outside RS (ft)	6.8
c.	LHSI (ft)	6.2

10. Recirculation phase suction friction loss:

a.	Inside RS	1.1 ft at 3300 gpm

01287.B07

	b.	Outside RS	4.9 ft at 3640 gpm (2840 gpm from sump and 800 gpm from casing cooling system)
	c.	LHSI	9.2 ft at 4030 gpm/pump (minimum ESF)
11.	Break effluent flashing model		Pressure flash - the break effluent expands at constant enthalpy to the containment total pressure. The standard vapor component goes to the containment atmosphere. This assumption conservatively neglects the evaporative cooling effect that the liquid component most certainly will realize.
12.	Mixing of steam with safety injection in the cold legs (cold leg breaks)		Complete mixing at the break between the break effluent (either steam or two phase) and the ECCS injection spillage (liquid). Thus, some or all of the steam is quenched, resulting in more mass and energy release to the floor and less to the atmosphere.
13.	Sensible energy transfer to spillage		If the spillage temperature is less than the dewpoint temperature, it is heated up to the dewpoint temperature before it is added to the floor. The energy in the containment atmosphere is decremented by the amount of heat transferred to the spillage. Thus, more energy is added to the floor and less to the atmosphere.

Inputs - The following sources were used as inputs to the calculation.

1. SWEC Calculation 14938.37-US(B)-249, Rev. 0, dated 5/28/87.

2. Memorandum from K.L. Basshore to R.C. Carroll, dated 9/10/91, "Confirmatory Containment Analysis for Steam Generator Replacement - North Anna Unit 1.

3. Letter from D.R. Beynon (W) to R.C. Carrol (Virginia Power), "Purchase Order BNT370788 - North Anna Unit 1 New Mass and Energy Release Data in support of Steam Generator Replacement, "VRA-92-082, May 28, 1992, transmits report titled "LOCA Mass and Energy Release Analisis Report - North Anna Unit 1, "report supplemented with addendum titled, "Addendum, LOCA Mass and Enegy Release, Low Head SI-4150 gpm," in letter VRA-92-091, June 16, 1992.

01287.B07

4. Virginia Power letter from R.K. Bayer to D.E. McLellan, dated 5/8/92, NP-1486-SO7-041, "Containment Analysis Case Documentation Steam Generator Replacement Project North Anna Power Station - Unit 1, IR-4714/NP-1486.

5. SWEC Calculation 02072.1610-US(B)-273, Rev. 0, dated 5/27/92.

6. Virginia Power Report "Analysis & System Modification for RS/LHSI Pump NPSH," dated September 1977 - Figure 2.2-3 (minimum required NPSH to avoid cavitation vs. flow rate for LHSI pump test).

7. Hydraulic Institute Standards, 14th editor, 1983 - Figure 70 (NPSH Reduction for Pumps Handling Hydrocarbon Liquids and High Temperature Water).

8. W topical report WCAP-10325 March 1979 version, "Westinghouse LOCA Mass and Energy Release Model for Containment Design," dated April 1979.

Methodology - The LOCTIC computer program is used to calculate both the pressure/temperature transient resulting from a LOCA in the containment as well as the available NPSH for the RS and LHSI pumps. However, when calculating the available NPSH, several of the input variables are changed (as shown in the assumptions section) in order to produce a conservative analysis. Sensitivity studies are performed on the initial temperature, pressure, dewpoint, break type and RWST, containment and SW temperature. All these variables influence the calculated NPSH of the RS and LHSI pumps.

Results and Conclusions - Different breaks were analyzed to develop the worst case scenario resulting in minimum NPSH available at the RS pumps. The analysis determined that the limiting case for the RS pumps' NPSH was the hot leg DER with normal safeguards (i.e., both trains available), minimum SW temperature and maximum RWST temperature. The minimum calculated NPSH available to the IRS pumps was 10.4 ft at 610 sec. after the LOCA. The minimum calculated NPSH available to the ORS pumps was 15.3 ft at 670 sec after the LOCA.

16.17 IMPACT ON RS PUMP NPSH DUE TO SUMP DEBRIS (UNIT 1 ONLY)

Calculation Number, Revision and Date - SWEC Calculation 02072.1610-US(B)-273, Rev 0, May 27, 1992.

Calculation Title - "Head Loss Across Emergency Sump Screen Due to Insulation Debris Caused by LOCA Events."

Purpose - The jet forces of a postulated high energy line break can generate significant debris which, in turn, may have an impact on the performance of all pumps that draw water from the containment emergency sump. In accordance with Regulatory Guideline 1.82, Rev. 1, the purpose of this analysis is to identify the break that potentially generates the largest quantity of debris, determine that portion of the debris that reaches the containment emergency sump screens and calculate the impact of this debris on the performance of the pumps in question. The debris causes an additional pressure drop across the screens which lessens the available NPSH to the RS amd LHSI pumps.

Assumptions

1. Transport of debris on the floor starts at approx. 220 seconds after the accident or just prior to RS pump start.

2a. 30% of the total transportable debris load (fine, suspended debris) is on the fine screens at 1800 seconds.

b. 40% of the total transportable debris load (fluffy fragments) accumulate at the bottom of the fine and coarse screens at 1800 sec.

c. 30% of the total transportable debris load (large floating fragments) is too large to transport to the screens.

3. LOCA pipe break jet impingement on the steam generator nozzle crossover pipe to the reactor cooling pump produces the largest insulation debris load.

4. Quench spray held up in containment = 10%

5. Weight of water at elevated temperatures is approx. 8 lbm/gal

6. NUREG/CR-0897 suggests using 7L/D's for the destruction distance. However, the included angle must be assumed. Since the jet can only impact 1/2 of the pipe, the included angle is 1/2 of 360° or 180°.

7. Insulation thickness for steam generator & crossover leg piping = 4 in.

8. Crossover leg piping O.D. = 36.57 in.

9. The screen area is defined as being the area just in front of the screen. The actual open area of the screens has no bearing on this analysis because the head loss equation for debris is based on the area just in front of the debris.

01287.B07

Inputs

1. SWEC Calculation 14938.37-US(B)-259-0, "LOCA Analysis for Revised Tech Spec on Containment Air Partial Pressures," May 28, 1987.

2. NCRODP-38 North Anna System Description.

3. SWEC Drawing 11715-FP-9B-12.

4. NUREG/CR-0897, Rev. 1, October 1985.

5. Handbook of Engineering Mechanics by W. Flugge, McGraw-Hill Book Co., 1st Edition, 1962.

6. SWEC Drawing 11715-FC-16A-15.

7. SWEC Drawing 11715-FS-14AA-2.

8. SWEC Calculation 12050-ES-220-0, "Containment Floor Volume with Respect to Elevation," April 9, 1980.

9. SWEC Calculation 13075-US-240-0, "Estimated Minimum Containment Sump Volume for Recirculation Spray Pump Startup," May 20, 1981.

10. U.S. Nuclear Regulatory Commission "Regulatory Guide 1.82", November 1985.

11. SWEC Drawing 11715-FP-14D-10.

12. SWEC Drawing 11715-FV-1S-3.

13. SWEC Drawing 11715-FC-16F-9.

14. SWEC Calculation 02072.0210-US(B)-270-0.

15. Letters Transco Products Inc./SWEC, from E.L. Avery to D. McLellen, "Primary System Insulation Material," dated 3/24/92 and 4/9/92.

16. NUREG/CR-2791, SAND82-7067, September 1982.

17. LOCTIC Run #R0844C02, Job 9106, Run 2/06/92, Insulation Debris Study (Max. ESF with 1-LHSI pump failure).

18. Walkdown Report - Steam Generator Fiberglass Insulation Debris Evaluation, dated 2/14/92.

19. SWEC Calc 01039.0532-US(B)-90, Rev. 0, dated 1/24/91 (Title: Validation of LOCTIC results using blowdown option)

Methodology

1. The pipe break in the primary or second piping system (closest to containment sump) that produces the largest amount of insulation debris is identified.

2. The total amount of transportable debris caused by the identified pipe break is in accordance with NUREG 0897, Rev. 1.

3. The net area of the fine screens is calculated.

4. Two limiting LOCTIC cases that affects the head loss is analyzed:

   a. Max. Flow Case - To maximize screen head loss the following pumps are used:

      1) 2 RSS (IRS) + 2 RSS (ORS) + 2 Quench spray + 1 LHSI

   b. Min. Flow Case - NPSH limiting case.

      1) 1 RSS (IRS) + 1 RSS (ORS) + 1 Quench spray + 1 LHSI

5. LOCTIC Max. Flow Case using method 4a is developed.

6. The amount of water on the containment floor due to the identified pipe break and assuming 10% water holdup is calculated.

7. The velocity at the fine screens with respect to time for both cases is calculated.

8. The containment water elevation with respect to time for both cases is developed.

9. The debris thickness on the fine screens with respect to time for both cases is developed.

10. The head loss due to the debris on the fine screen with respect to time for both cases is developed.

11. Results for both cases are plotted:

    a. Containment water level vs time
    b. Velocity vs time
    c. Head loss vs time

Results and Conclusions - Head loss (due to the debris blockage) vs time of the emergency sump fine screens is shown graphically.

The results, are used in conjunction with the NPSH analysis for the ORS, IRS and LHSI pumps to assure that NPSH margin exists after the available NPSH is reduced by an amount equal to the debris head loss.

01287.B07

17.1.4 RS Coolers

1. Statement of Maintenance Requirement - The RS coolers shall be maintained dry. The RS coolers shall be vented and drained (to remove the fluid from both the shell and tube sides of the cooler) and the inlet and outlet service water valves maintained isolated.

2. Frequency of Maintenance - At least once per week.

3. Reason - To preclude fouling of the heat transfer surfaces (i.e., the tubes) to ensure RS cooler operation in accordance with the assumptions of the safety analysis. [25.4.2] [25.4.4] [25.4.72] [25.4.139] [24.2.17]

4. Design Feature - Vents, drains and isolation valves are provided to maintain the coolers dry. Refer to Section 14.1.21, item 4 for further detail.

5. Reference - [25.5.26] [25.8.1] [25.8.2] [25.5.12]

17.1.5 RS Coolers

1. Statement of Maintenance Requirement - RS coolers preventative maintenance procedures shall include disassembly and inspection of RS coolers to verify absence of biological growth, and chemical cleaning (if necessary) based on the inspection. Procedures shall be available to return the system line up to its normal standby configuration upon completion of maintenance and testing.

2. Frequency of Maintenance - Every 12 months.

3. Reason - To verify cleanliness which will preclude fouling of the tubes and ensure operation of the RS coolers in accordance with the assumptions of the safety analysis. [25.4.4] [25.4.2] [25.4.78] [25.4.139] 72 [24.2.17]

4. Design Feature - Not applicable.

5. Reference - [25.5.19] [25.5.12]

17.1.6 RS Coolers

1. Statement of Maintenance Requirement - The RS cooler diaphragms shall be replaced after every five cycles (i.e., five containment pressurizations).

2. Frequency of Maintenance - After every 5 containment pressurizations.

Table 19.1-1. Technical Specification Surveillance Requirements

1. Design Basis Surveillance Requirement - CCT fluid volume

   a. System or Component - CCT

   b. Source - TS 4.6.2.2.2 [25.3.11] [25.3.12]

   c. Surveillance Required - CCT shall be demonstrated operable.

      Reason or Basis - To ensure availability of casing cooling fluid to support post-LOCA ORS pump NPSH requirements in accordance with the containment and NPSH analyses. [25.4.2] [25.4.4] [25.4.72] [25.4.139] [24.2.17]

   d. Surveillance Parameter - The CCT shall have a minimum fluid volume of 116,500 gallons.

      Reason or Basis To ensure that the required volume of 113,904 gallons (equivalent to 96000 gallons of useable water) is available in the CCT with sufficient margin. [25.4.2] [25.4.110] [25.4.139]

   e. Surveillance Frequency - At least once per 7 days.

      Reason or Basis - Technical basis not known, imposed by the NRC. [25.3.11] [25.3.12]

   f. Surveillance Conditions - Mode 1, 2, 3, or 4.

      Reason or Basis - Imposed by NRC (Condition IV events (LOCA, MSLB, FWLB, REA) during which the RS System is required to operate are credible only during Modes 1, 2, 3, 4.) [25.3.11] [25.3.12]

   g. Acceptance Criteria - The CCT volume shall be greater than 116,500 gallons.

      Reason or Basis - Refer to item d and Table 6.1-1, item 6 and Section 12.5.

Table 19.1-1. Technical Specification Surveillance Requirements

3. Design Basis Surveillance Requirement - CCT fluid temperature

   a. System or Component - CCT

   b. Source - TS 4.6.2.2.2 [25.3.11] [25.3.12]

   c. Surveillance Required - CCT shall be demonstrated operable.

      Reason or Basis - To ensure that the CCT fluid temperature is above freezing but within the temperature requirements established in the ORS pump NPSH analyses (to support ORS pump NPSH requirement). [25.4.2] [25.4.4] [25.4.72] [25.4.159] [24.2.1

   d. Surveillance Parameter - The CCT fluid temperature shall be between 35°F and 50°F.

      Reason or Basis - Refer to item c.

   e. Surveillance Frequency - At least once per 24 hours.

      Reason or Basis - Technical basis not known, imposed by the NRC. [25.3.11] [25.3.12]

   f. Surveillance Conditions - Mode 1, 2, 3, or 4.

      Reason or Basis - Imposed by NRC (Condition IV events (LOCA, MSLB, FWLB, REA) during which the RS System is required to operate is credible only during Modes 1, 2, 3, 4). [25.3.11] [25.3.12]

   g. Acceptance Criteria - The CCT fluid temperature shall be at least 35°F but no greater than 50°F.

      Reason or Basis - Refer to item d.

Table 19.1-1. Technical Specification Surveillance Requirements

5. Design Basis Surveillance Requirement - ORS pump discharge pressure

a. System or Component - ORS pumps

b. Source - TS 4.6.2.2.1 [25.3.11] [25.3.12]

c. Surveillance Required - ORS subsystem shall be tested on recirculation flow to demonstrate its operability under accident conditions.

Reason or Basis - To ensure ORS pump operability in accordance with the assumptions of the safety analyses, [25.4.2] [25.4.4] [25.4.72] [25.4.139] [24.2.17] without requiring full flow testing which would involve using the dike and wetting the internal surfaces of the RS cooler tubes. [25.5.22] [25.1.10] [25.10.1] [25.1.20]

d. Surveillance Parameter - Each ORS pump shall develop a discharge pressure greater than or equal to 115 psig.

Reason or Basis - [24.3.5] [24.3.4]

e. Surveillance Frequency - In accordance with the Inservice Inspection Program. [25.3.52] [25.3.53]

Reason or Basis - ASME XI requirement imposed by NRC. Refer to Section 13.1.1, item 4.

f. Surveillance Conditions - Mode 1, 2, 3, or 4.

Reason or Basis - Technical basis not known, imposed by NRC. [25.3.11] [25.3.12]

g. Acceptance Criteria - Each ORS pump discharge pressure shall be at least 115 psig.

Reason or Basis - Refer to items c and d.

Table 19.1-1. Technical Specification Surveillance Requirements

6. Design Basis Surveillance Requirement - Automatic initiation of RS pumps

a. System or Component - RS pumps

b. Source - TS 4.6.2.2.1 [25.3.11] [25.3.12]

c. Surveillance Required - Each RS pump shall be demonstrated to start automatically on a containment "high-high" pressure signal.

Reason or Basis - To ensure containment depressurization in accordance with the containment safety analyses in the event of a HELB inside containment. [25.4.2] [25.4.4] [25.4.72] [25.4.139] [24.2.17] [25.1.10] [25.10.1] [25.1.20]

d. Surveillance Parameter - Each ORS and IRS pump shall start (automatically) 210 sec and 195 secs, respectively, after the receipt of a containment "high-high" pressure signal.

Reason or Basis - Refer to item c.

e. Surveillance Frequency - At least once per 18 months.

Reason or Basis - Technical basis not known, imposed by the NRC. [25.3.11] [25.3.12]

f. Surveillance Conditions - Mode 5 or 6.

Reason or Basis - Technical basis not known, imposed by the NRC. [25.3.11] [25.3.12]

g. Acceptance Criteria - The ORS and IRS pump shall start within 195 ± 9.75 secs (IRS) and 210 ± 21 secs (ORS) of receipt of a containment "high-high" pressure signal.

Reason or Basis - Refer to item d. [24.11.8] [25.4.111]

Table 19.1-1. Technical Specification Surveillance Requirements

7. Design Basis Surveillance Requirement - RS System valve alignment

a. System or Component - RS System valves (including valves in the casing cooling subsystem)

b. Source - TS 4.6.2.2.1 [25.3.11] [25.3.12]

c. Surveillance Required - Four independent RS subsystems shall be demonstrated operable.

Reason or Basis - To ensure that the RS subsystem operates in accordance with the assumptions of the containment safety analyses. [25.4.2] [25.4.4] [25.4.72] [25.4.139] [24.2.17]

d. Surveillance Parameter - Each valve in the RS flow path that is not locked, sealed, or otherwise secured in position, shall be in its correct position.

Reason or Basis - Refer to item c.

e. Surveillance Frequency - At least once per 31 days.

Reason or Basis - Technical basis not known, imposed by the NRC. [25.3.11] [25.3.12]

f. Surveillance Conditions - Mode 1, 2, 3, or 4.

Reason or Basis - Imposed by NRC (Condition IV events (i.e., LOCA, MSLB, FWLB, REA) during which the RS System is required to operate are credible only during Modes 1, 2, 3, and 4.) [25.3.11] [25.3.12]

g. Acceptance Criteria - No RS valve shall be out of its required position.

Reason or Basis - Refer to item d.

Table 19.1-1. Technical Specification Surveillance Requirements

8. Design Basis Surveillance Requirement - Automatic actuation of RS System MOVs

a. System or Component - RS System MOVs (including MOVs in the casing cooling subsystem)

b. Source - TS 4.6.2.2.1 [25.3.11] [25.3.12]

c. Surveillance Required - Each RS System automatic valve shall be demonstrated to actuate to its correct position (refer to Section 3.3.4) on a containment "high-high" pressure signal.

[25.4.139]
[24.2.17]

Reason or Basis - To ensure RS System operation in accordance with the assumptions of the containment safety analyses. [25.4.2] [25.4.4] [25.4.72] [25.1.20] [25.10.1] [25.1.10]

d. Surveillance Parameter - Each RS System automatic valve in the system flow path shall actuate to its correct position.

Reason or Basis - Refer to item c.

e. Surveillance Frequency - At least once per 18 months.

Reason or Basis - Technical basis not known, imposed by the NRC. [25.3.11] [25.3.12]

f. Surveillance Conditions - Mode 5 or 6.

Reason or Basis - Technical basis not known, imposed by the NRC. [25.3.11] [25.3.12]

g. Acceptance Criteria - Each automatic valve in the RS System flow path shall actuate to its correct position.

Reason or Basis - Refer to item d.

Table 19.1-1. Technical Specification Surveillance Requirements

12. Design Basis Surveillance Requirement - Casing cooling pump discharge pressure

a. System or Component - Casing cooling pumps

b. Source - TS 4.6.2.2.1 [25.3.11] [25.3.12]

c. Surveillance Required - The casing cooling pumps shall be tested on recirculation flow to demonstrate their operability under accident conditions.

[25.4.139] [24.2.17]

Reason or Basis - To ensure casing cooling operability in accordance with the assumptions of the plant safety and ORS pump NPSH analyses [25.4.4] [25.4.2] [25.4.72] without requiring full flow testing which could impact the dry and ready conditions (to prevent fouling) of the RS System. (Refer to Section 3.3.1.) [25.1.10] [25.10.1] [25.1.20]

d. Surveillance Parameter - Each casing cooling pump shall develop a discharge pressure greater than or equal to 46 psig (Unit 2) or 58 psig (Unit 1).

Reason or Basis - [24.3.6] [24.3.4]

e. Surveillance Frequency - In accordance with the Inservice Inspection Program. [25.3.52] [25.3.53]

Reason or Basis - Imposed by NRC. Refer to Section 13.1.1, item 4.

f. Surveillance Conditions - Mode 1, 2, 3, or 4.

Reason or Basis - Technical basis not known, imposed by the NRC. [25.3.11] [25.3.12]

g. Acceptance Criteria - The casing cooling pump discharge pressure shall be at least 46 psig (for Unit 2) and 58 psig for (Unit 1).

Reason or Basis - Refer to items c and d.

Table 19.1-1. Technical Specification Surveillance Requirements

13. Design Basis Surveillance Requirement - Automatic initiation of casing cooling pumps

a. System or Component - Casing cooling pumps

b. Source - TS 4.6.2.2.1 [25.3.11] [25.3.12]

c. Surveillance Required - Each casing cooling pump shall be demonstrated to start automatically on a containment "high-high" pressure signal.

[25.4.139] [24.2.17]

Reason or Basis - To ensure availability of casing cooling fluid to support post-LOCA ORS pump NPSH requirements in accordance with the containment and NPSH analyses. [25.4.2] [25.4.4] [25.4.72] [25.1.10] [25.1.20] [25.10.1]

d. Surveillance Parameter - Each casing cooling pump shall start automatically upon receipt of a containment "high-high" pressure signal.

Reason or Basis - Refer to item c.

e. Surveillance Frequency - At least once per 18 months.

Reason or Basis - Technical basis not known, imposed by the NRC. [25.3.11] [25.3.12]

f. Surveillance Conditions - Mode 5 or 6.

Reason or Basis - Technical basis not known, imposed by the NRC. [25.3.11] [25.3.12]

g. Acceptance Criteria - The casing cooling pumps shall start automatically upon receipt of a containment "high-high" pressure signal.

Reason or Basis - Refer to item d.

Table 19.1-1. Technical Specification Surveillance Requirements

14. Design Basis Surveillance Requirement - Unobstructed flow through RS nozzles

a. System or Component - RS header/nozzle

b. Source - TS 4.6.2.2.1 [25.3.11] [25.3.12]

c. Surveillance Required - Each RS System header and spray nozzle shall be demonstrated to be in an unobstructed state.

[25.4.139]
[24.2.17]

Reason or Basis - To ensure RS System operation in accordance with the assumptions of the containment safety analyses. [25.4.2] [25.4.4] [25.4.72] X

d. Surveillance Parameter - Each spray header and spray nozzle shall be visually surveyed for flow indication using an air or smoke flow test.

Reason or Basis - Refer to item c. (Full flow testing of the nozzles using water would result in the need for containment cleanup. Consequently, smoke or air is used for testing purposes.)

e. Surveillance Frequency - At least once per 5 years.

Reason or Basis - Technical basis not known, imposed by the NRC. [25.3.11] [25.3.12]

f. Surveillance Conditions - Mode 5 or 6.

Reason or Basis - Introduction of smoke into the containment via the RS nozzles would adversely impact plant operations during Modes 1, 2, 3, and 4.

g. Acceptance Criteria - The RS headers and nozzles shall not be obstructed.

Reason or Basis - Refer to item c. Since the nozzles are not flow tested with water, nozzle erosion is not expected. Consequently, the thermal effectiveness of the water droplets will remain as designed and a demonstration of unobstructed nozzles is all that is required.

Table 19.1-1. Technical Specification Surveillance Requirements

16. Design Basis Surveillance Requirement - CDA signal interlock with the RS System

a. System or Component - RS System (RS and casing cooling pump and MOVs)

b. Source - TS 4.3.2.1.2 [25.3.11] [25.3.12]

c. Surveillance Required - The logic of the CDA signal interlock with the RS System initiation shall be demonstrated operable.

[25.4.139]
[24.2.17]

Reason or Basis - To ensure operation of the RS System in accordance with the assumptions of the containment safety analyses. [25.4.2] [25.4.4] [25.4.72] [25.1.10] [25.1.20] [25.10.1]

d. Surveillance Parameter - RS System actuation (i.e., startup of RS pump timers, casing cooling pumps, and opening of RS MOVs) in response to a manually initiated CDA signal.

Reason or Basis - Refer to item c.

e. Surveillance Frequency - At least once every 18 months.

Reason or Basis - Technical basis not known, imposed by the NRC. [25.3.11] [25.3.12]

f. Surveillance Conditions - Mode 5 or 6.

Reason or Basis - Technical basis not known, imposed by the NRC. [25.3.11] [25.3.12]

g. Acceptance Criteria - The total interlock function (i.e., initiation of the RS System in response to a CDA signal) shall be demonstrated operable.

Reason or Basis - Refer to item d.

Table 19.2-1. Licensing Basis Surveillance Requirements

4. Design Basis Surveillance Requirement - Freon leak from CCT refrigeration unit

   a. System or Component - CCT refrigeration units

   b. Source - [25.5.33]

   c. Surveillance Required - CCT refrigeration units shall be demonstrated operable.

      Reason or Basis - To ensure proper operation of the CCT refrigeration units so that the CCT fluid temperature is maintained within the requirement of the ORS pump NPSH analyses. [25.4.2] [25.4.139] X

   d. Surveillance Parameter - The CCT refrigeration units shall be inspected to verify absence of freon leaks.

      Reason or Basis - Refer to item c.

   e. Surveillance Frequency - None identified in reference identifying the requirement. [25.5.33]

      Reason or Basis - See above.

   f. Surveillance Conditions - Modes 1, 2, 3, and 4.

      Reason or Basis - Condition IV events (LOCA, MSLB, FWLB, REA) during which RS System operation is required are credible only during Modes 1, 2, 3, and 4.

   g. Acceptance Criteria - Verification of the absence of freon leaks.

      Reason or Basis - Refer to item d.

Table 19.2-1. Licensing Basis Surveillance Requirements

5. Design Basis Surveillance Requirement - Service water leakage into RS coolers

   a. System or Component - RS coolers

   b. Source - [25.5.26]

   c. Surveillance Required - The RS coolers shall be maintained dry.

      [25.4.139] [24.2.17]

      Reason or Basis - To prevent fouling of the RS cooler tubes and ensure that the RS cooler heat transfer coefficients are in accordance with the assumptions of the safety analyses. [25.4.4] [25.4.2] [25.4.72]

   d. Surveillance Parameter - The RS Coolers shall be inspected to determine if service water is leaking into the RS coolers.

      Reason or Basis - Refer to item c.

   e. Surveillance Frequency - At least once per week.

      Reason or Basis - [25.5.26]

   f. Surveillance Conditions - Modes 1, 2, 3, 4, 5, and 6.

      Reason or Basis - RS cooler tube fouling has to be prevented regardless of plant operational mode since the tubes have to be chemically cleaned to remove fouling when it occurs.

   g. Acceptance Criteria - Verification of the lack of moisture collection when the vents/drains associated with the RS coolers are opened.

      Reason or Basis - Refer to item d.

Table 19.2-1. Licensing Basis Surveillance Requirements

6. Design Basis Surveillance Requirement - Casing cooling pump bearing lubrication

   a. System or Component - Casing cooling pumps

   b. Source - [25.5.35]

   c. Surveillance Required - The casing cooling pumps shall be demonstrated operable.

      Reason or Basis - To ensure proper operation of the casing cooling pumps. [25.4.4] [25.4.72] [25.4.2] [25.4.139] [24.2.17] X

   d. Surveillance Parameter - The bearing lubrication of the casing cooling pumps shall be checked.

      Reason or Basis - To prevent damage to the casing cooling pumps.

   e. Surveillance Frequency - At least once per year.

      Reason or Basis - [25.5.35]

   f. Surveillance Conditions - Mode 1, 2, 3, or 4.

      Reason or Basis - To ensure proper operation of the casing cooling pumps in the event of a HELB in containment (which can only occur if the plant is in Mode 1, 2, 3, or 4).

   g. Acceptance Criteria - Quantity of bearing lubrication for the casing cooling pumps shall be in accordance with vendor requirements.

      Reason or Basis - Refer to item d.

Table 19.2-1. Licensing Basis Surveillance Requirements

7. Design Basis Surveillance Requirement - Dry testing of IRS pumps

a. System or Component - IRS pumps

b. Source - [25.5.9] [25.5.10] [25.5.36]

c. Surveillance Required - The IRS pumps shall be dry tested (start and stop) to demonstrate operability under accident conditions.

[25.4.139] [24.2.17]

Reason or Basis - To ensure IRS pump operability in accordance with the assumptions of the safety analyses [25.4.4] [25.4.2] [25.4.72], without performing flow tests which would require plant shutdown and installation of a dike in the containment sump.

d. Surveillance Parameter - Each IRS pump shall demonstrate staring capability.

Reason or Basis - Refer to item c.

e. Surveillance Frequency - Once every 3 months as documented in the In-Service Inspection Program. [25.3.52] [25.3.53]

Reason or Basis - To provide continuing assurance of mechanical reliability, the testing frequency of these pumps were increased from 1 month to 3 months by the NRC. [25.5.9]

f. Surveillance Conditions - Modes 1, 2, 3, 4

Reason or Basis - Technical bases not known, imposed by NRC.

g. Acceptance Criteria - Each IRS pump shall be demonstrated to startup.

Reason or Basis - Refer to item c.

Table 19.4-1. Preoperational and Startup Testing Requirements

1. Design Basis Surveillance Requirement - Full flow testing of the RS System

   a. System or Component - IRS and ORS Pumps, RS coolers, RS headers (all nozzles plugged) and associated valves/piping/fittings

   b. Source - [25.3.8] [25.10.15] [24.3.4]

   c. Surveillance Required - The RS System as a whole shall be demonstrated operable prior to initial plant startup.

      [25.4.139]
      [24.2.17]

      Reason or Basis - To ensure that the RS System operates in accordance with the flow assumption of the safety analyses. [25.4.2] [25.4.4] [25.4.72] X

   d. Surveillance Parameter - Preoperational testing was conducted over a throttled range of flow conditions (using the permanently installed spray header drain lines connected to the containment sump, blind flanges, pipe plugs in the spray nozzles and the portable dike surrounding the sump) to verify proper operation of the RS System as a whole. (Refer to Section 3.3.1.)

      Reason or Basis - Refer to item c.

   e. Surveillance Frequency - Once in plant life.

      Reason or Basis - Preoperational testing.

   f. Surveillance Conditions - Prior to initial plant startup.

      Reason or Basis - [25.5.13] Preoperational testing.

   g. Acceptance Criteria - Verify RS System operation/flow in accordance with the safety analyses.

      Reason or Basis - Refer to item d.

Table 19.4-1. Preoperational and Startup Testing Requirements

2. Design Basis Surveillance Requirement - Long-term operability of deep well, long shafted pumps

a. System or Component - ORS pumps

b. Source - [25.5.13] [25.10.6]

c. Surveillance Required - The ORS pumps shall be demonstrated operable (in the long term) prior to initial plant startup.

[25.4.139]
[24.2.17]

Reason or Basis - To ensure that the ORS pumps (which are deep well, long shafted pumps) are capable of operating (in the long term) in accordance with the assumptions of the safety analyses. [25.4.2] [25.4.4] [25.4.72] X

d. Surveillance Parameter - Preoperational testing was conducted for a 6-day extended run and a 450-hour test run with inspections between runs and upon completion of testing.

Reason or Basis - Refer to item c.

e. Surveillance Frequency - Once in plant life.

Reason or Basis - Preoperational testing.

f. Surveillance Conditions - Prior to initial plant startup.

Reason or Basis - [25.5.13] Preoperational testing.

g. Acceptance Criteria - Verify continued trouble-free operation of the ORS pumps after 450 hours of operation.

Reason or Basis - Refer to item d.

Table 19.4-1. Preoperational and Startup Testing Requirements

4. Design Basis Surveillance Requirement - Unobstructed flow through RS nozzles

a. System or Component - RS headers/nozzles

b. Source - [25.3.8] [25.10.15] [24.3.4]

c. Surveillance Required - Each RS System header and spray nozzle shall be demonstrated to be in an unobstructed state prior to initial plant startup.

[25.4.139]
[24.2.17]

Reason or Basis - To ensure RS System operation in accordance with the assumptions of the safety analyses. [25.4.2] [25.4.4] [25.4.72]

d. Surveillance Parameter - Preoperational testing was conducted on the RS system spray nozzles using air to verify an unobstructed flow path. Comparison of flow from each nozzle was made to assure that free flow existed.

Reason or Basis - Refer to item c. Full flow testing of the nozzles during pre-operational testing of the system as a whole (see Table 19.4-1, item 1) is not performed since it would result in wetting of the entire containment and the subsequent need for containment cleanup.

e. Surveillance Frequency - Once in plant life. (For periodic testing of RS headers/nozzles, see Table 19.1-1, item 14.)

Reason or Basis - Preoperational testing.

f. Surveillance Conditions - Prior to initial plant startup.

Reason or Basis - Preoperational testing.

g. Acceptance Criteria - Verification that the RS nozzles are not obstructed.

Reason or Basis - Refer to item d.

21.0 DESIGN BASIS TECHNICAL SPECIFICATION REQUIREMENTS

This section identifies the Technical Specification (TS) requirements and the basis thereof for the RS System. The TS, as defined by 10CFR50.36, are an integral part of the Operating License (OL) and establish limiting conditions of operation and surveillance requirements for the system. This chapter's emphasis is on the basis of the limiting conditions of operation. Section 19.1 addresses the design basis surveillance requirements contained in the TS. Therefore, they are not repeated herein, but are merely referenced.

In theory, the TS requirements are established based on the design basis. Critical design features are normally included in the TS based on NRC requirements at the time of issuance of an OL and amendments thereto. However, once the requirements are incorporated into the TS, the requirements themselves become a licensing basis that must be complied with. Clearly, the design basis and the TS must be consistent in all respects. Since noncompliance with the TS represents a violation of the license, it is important that the basis for the TS be clearly identified and controlled.

In general, only safety-related systems are addressed in the TS. These requirements are normally stated in a concise manner in the TS; therefore, the TS number and the requirement are simply restated in the SDBD. The basis is summarized in a clear and concise manner in this chapter. The basis may make reference to other sections of the SDBD which provide the reason for the requirement.

21.1 AUTOMATIC ACTUATION OF RS SYSTEM

Requirement - Technical Specification 3.3.2.1

The Engineered Safety Feature (ESF) Actuation System instrumentation channels and interlocks shall be operable.

Applicability: Modes 1, 2, 3, 4

Basis - The operability of the ESF instrumentation systems and interlocks ensures that 1) the RS System will be initiated when the containment pressure exceeds its "high-high" setpoint, 2) the specified coincidence logic is maintained, 3) sufficient redundancy is maintained to permit a channel to be out of service for testing or maintenance, and 4) sufficient system functional capability is available for ESF purposes from diverse parameters.

The operability of the ESF System is required to provide the overall reliability, redundance and diversity assumed available in the facility and RS System design for the protection and mitigation of accident and transient conditions. The integrated operation of each of these systems is consistent with the assumptions used in the accident analysis. (Refer to Section 2.2.1, Item 1.) [25.4.2] [25.4.4] [25.4.72] [25.4.111]

[25.4.139] [24.2.17]

2. Two separate and independent outside recirculation spray pump casing cooling subsystems, each composed of a casing cooling pump, and flow path capable of transferring fluid from the CCT to the suction of the outside recirculation spray pumps.

3. One CCT shall be operable with:

   a. Contained borated water volume of at least 116,500 gallons.
   b. Between 2300 and 2400 ppm boron concentration.
   c. A solution temperature $\geq$ 35° and $\leq$ 50°F.

Applicability: Modes 1, 2, 3, 4.

[25.4.139] [24.2.17]

Basis - The operability of the containment spray systems (RS and QS) ensures that containment depressurization and subsequent return to subatmospheric pressure will occur within 1 hour following a LOCA. The pressure reduction and resultant termination of containment leakage are consistent with the assumptions used in the accident analyses. [25.4.2] [25.4.4] [25.4.72] (Refer to Section 2.2.1, item 1)

The Technical Specification ensures the operability of the RS System while allowing partial reduction of total system availability. The specification takes into account the component redundance and subsystem independence. Partial loss of the system capability will not impact the remaining subsystems' performance in meeting the design function.

The CCT is necessary to increase the available NPSH of the ORS pumps. The specification allows for normal maintenance and repair of the CCT during plant operations.

Surveillance: Refer to Table 19.1-1.

21.4 CONTAINMENT ISOLATION VALVES

Requirement - Technical Specification 3.6.3.1

The containment isolation valves shall be operable.

Applicability: Modes 1, 2, 3, 4

Basis - The operability of the containment isolation valves ensures that the containment atmosphere will be isolated from the outside environment in the event of a release of radioactive material to the containment atmosphere or pressurization of the containment. (Refer to Section 2.2.1, item 3.)

Inservice inspection of ASME Code Class 1, 2, and 3 components and inservice testing of ASME Code Class 1, 2, and 3 pumps and valves shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR 50, Section 50.55a(g), except where specific written relief has been granted by the Commission pursuant to 10 CFR 50, Section 50.55a(g)(6)(i).

Applicability: In accordance with the requirements of the ISI Program.

Basis - In accordance with the requirements of 10CFR50.55a(g) (which implements in-service inspection in accordance with ASME XI), this Technical Specification is provided to ensure that operations or age does not alter safety-related components to a point beyond the established design basis. [25.1.46] [25.3.45] [25.3.52] [25.3.53] [25.2.21] [25.2.22]

Surveillance: Refer to Table 19.1-1.

21.7 CONTAINMENT SUMP INSPECTION

Requirement - Technical Specification 4.5.2.d

A visual inspection of the containment sump to verify that the subsystem suction inlets are not restricted by debris and that the sump components (trash racks, screens, etc.) show no evidence of structural distress or corrosion.

Applicability: Mode 5

Basis - To ensure proper operation of the RS System and postaccident flow rates in accordance with the assumptions of the safety analyses. [25.5.34] [25.12.2] [25.4.2] [25.4.4] [25.4.72] [25.4.139] [24.2.17]

Surveillance: Refer to Table 19.1-1.

21.8 RS SYSTEM LEAKAGE OUTSIDE CONTAINMENT

Requirement - Technical Specification 6.8.4

A program to reduce leakage from those portions of systems outside containment that could contain highly radioactive fluids during a serious transient or accident to as low as practical levels. The systems include the recirculation spray, safety injection, chemical and volume control, gas stripper, and hydrogen recombiners. The program shall include the following:

1. Preventive maintenance and periodic visual inspection requirements, and

22.0 DESIGN BASIS OPERATIONAL REQUIREMENTS

This chapter identifies the operational requirements that are an integral part of the design basis. Design basis operational requirements are those actions which must be performed by human manipulation to assure that the requirements of the plant safety analyses and design basis are satisfied. The operational requirements may include required manipulations of systems, components, structures, or controls, as well as the performance or prohibition of specific actions.

Automatic control actions performed by the instrumentation and controls system are not deemed to be operational requirements; however, subsequent operator actions may qualify as operational requirements if they are required to satisfy the plant safety analysis or the design basis.

This chapter is limited to only those design features that require human involvement to assure conformance with the plant safety analysis or the design basis. Routine operational requirements for starting, operating, or shutting down systems or components are not included here. The requirements contained in this section must be implemented by station procedures; however, the procedures are not referenced herein. The design basis specifies the requirements and the station procedures implement the requirement.

22.1 PLANT STARTUP OPERATION

This section identifies the design basis operational requirements for the RS System which are applicable during plant startup.

These requirements are as follows:

[25.4.139]
[24.2.17]

Operational Requirement - Prior to plant startup, the RS System shall be aligned for automatic initiation. [25.3.11] [25.3.12] [25.4.2] [25.4.4] [25.4.72]

Basis - In the event of an accident (LOCA, MSLB, FWLB, REA), to ensure automatic initiation and operation of the RS System in accordance with the assumptions of the safety analyses. (Refer to Sections 3.1, 3.3.1, and 11.1.)

22.2 NORMAL OPERATION

The section identifies the design basis operational requirements for the RS System which are applicable during normal plant operation. These requirements are as follows:

Operational Requirement - During normal operation, the RS System shall be aligned for automatic initiation and shall require no operator action other than general monitoring. [25.3.11] [25.3.12] [25.4.2] [25.4.4] [25.4.72]

Operational Requirement - RS System testing shall be conducted in accordance with the Inservice Inspection Test program, as identified in Chapter 19. Test procedures shall direct the operator to perform the required tests and to ensure minimum subsystem availability as defined in the Technical Specifications. Procedures shall also be available and used to verify component operability after maintenance. The RS System shall be tested as detailed in Sections 3.3.1 and 14.1.20.

Testing of RS System equipment shall require the operator to use the design features provided to test the equipment and perform surveillance (refer to Section 14.1.20 for design features).

Special operator action requirements for the RS System relative to testing are identified below:

1. Periodic recirculation flow testing of the IRS pumps shall require the operator to install the portable dike in the containment sump, remove the straight spool piece to the RS coolers and insert an elbow piece to allow water to flow back to the sump via the test line.

2. Periodic recirculation flow testing of the ORS pumps shall require the operator to fill the pump casing with water, isolate the main ORS system flow path and use the test line to recirculate the fluid flow back to the pump casing.

3. Testing the RS System as a whole shall require the operator to insert plugs in the spray nozzle sockets, use spool pieces to connect the RS header to permanently installed spray header drain lines to the sump and install the portable dike in the sump.

After test completion, the operator shall remove the spool pieces inserted for testing and align the system in accordance with the assumptions of the safety analyses.

Basis - To ensure that operations or age does not alter component reliability and performance to a point beyond the established design basis. [25.1.9] [25.1.10] [25.1.11] [25.10.1] [25.1.46] [25.3.45] [25.3.52] [25.3.53] [25.2.21] [25.2.22] [25.3.11] [25.3.12] (For further information, refer to Section 14.1.20.); and to ensure that after test completion, the RS System is once again aligned in accordance with the requirements of the safety analyses. [25.4.2] [25.4.4] [25.4.72] [25.4.139] [24.2.17] X

Table 23.1-3. Design Basis Setpoints, Safety Analysis Setpoint Parameters, RS System

	Parameter	Basis/Reason	Reference
1.	RS Pump Startup Delay Timers (IRS and ORS)	The RS Pump timers' setpoints are selected to delay the startup of the RS pumps but to remain within the assumptions of the safety analyses which require full flow recirculation spray within 300 seconds after a CDA signal. This delay allows for accumulation of water (from the ruptured line and QS System operation) to provide adequate containment sump fluid level to minimize cavitation of the pumps at startup.	[25.4.2] [25.4.4] [25.4.72] [25.4.111] [25.4.139] [24.2.17]
2.	Casing Cooling Tank Level	The CCT low level setpoint ensures that the volume of cool, borated water available to be injected into the ORS pump suction is in accordance with the assumptions of the ORS pump NPSH analysis. Cold water injection is required to support the postaccident ORS pump NPSH requirements.	[25.4.110] [25.4.2] [25.4.139]
3.	Casing Cooling Tank Temperature	The CCT temperature setpoints associated with initiating the recirculation/cooling of the CCT fluid ensure that the temperature of the water in the CCT is maintained within the temperature requirements of the ORS pump NPSH analyses. (Following an accident, cold water injection into the ORS pump suctions will increase the available NPSH to the ORS pumps by cooling the containment sump water.) Refer to Table 23.3-1.	[25.4.2] [25.4.139]
4.	Restart Delay Timers	Additional timer setpoints are established for loading the RS pumps onto the EP System following a loss-of-offsite power. The delay times are necessary to avoid overloading the EDG.	[25.4.130] [24.11.10] [25.8.27] [25.8.28] [25.8.30] [25.8.31] [25.0.11] [25.8.12] [25.8.14] [25.8.15]

Table 23.3-1. Design Basis Setpoints, Non-Safety-Related Parameter Setpoints, RS System

	Parameter	Basis/Reason	Reference
1.	Casing Cooling Tank Temperature Sensor Interlock with CCT Cooling Loop	The CCT temperature sensor setpoints ensure that the temperature of the fluid in the CCT is maintained between 42°F and 45°F which is well within the assumptions of the current ORS pump NPSH analyses. (A "high" CCT fluid temperature signal initiates startup of a CCT recirculation pump. The corresponding CCT refrigeration unit starts up on detection of flow through the unit. The CCT cooling loop is shutoff on the reverse logic.)	[25.4.2] [25.3.47] [25.4.139]
2.	Recirculation Spray Pump Valve Pit Level.	The recirculation spray pump valve pit high level setpoint provides indication to the operator of failure of components in the valve pit area resulting in the leakage of containment sump fluids into the valve pit located outside containment. This indication prompts the operator to perform isolation actions in accordance with plant procedures.	[24.23.2]
3.	ORS Pump Mechanical Seal Pressure Chamber Level	The level switch on the ORS Pump Mechanical Seal Pressure Chamber actuates an alarm if the level falls below the level set by the pump manufacturer. The seal water prevents containment sump water from leaking out of the ORS pump seals.	[25.6.1] [24.23.3]

24.0 OPEN ITEMS

This chapter identifies open items (as defined in Section 1.13), including errors and omissions that have been identified during the preparation of this SDBD. Each open item is noted in the body of the text with an open item number, rather than stating the full description of the open item within the text. The number used in the text [24.X.X] corresponds with the number in this chapter. The combination of the SDBD number and the open item establishes a unique number, e.g., SDBD-NAPS-AFW-24.6.2, that is used to monitor the status of the open item in the Design Basis Document Item Resolution System (DBD IRS).

24.2.1 Sections 2.2.1, 11.1.4, 14.1.1, 14.1.2, 14.2.2, 14.3.2, 14.4.2, 12.1, 12.2, 12.7, 12.8, 12.9, 12.10, 12.11, 14.5.2, 14.6.1, 14.7.1, 14.8.2, and 25.4; Tables 6.1-1 and 11.1-5, items 1, 2, 3, 4, 6. Information provided in MSLB safety analyses [25.4.72] appears to supersede the results of calculations [25.4.16] [25.4.17] [25.4.18] [25.4.19] [25.4.15]. The latter calculations, however, have not been voided.

24.2.2 Section 2.2.1. The statement that the consequences (i.e., temperature/pressures) of a LOCA or an MSLB envelop the consequences of all other accidents should be documented in the safety analyses.

24.2.3 Sections 11.1.4, 12.10, 14.1.2, 14.4.2, and 18.1.4; Tables 6.1-1 and 11.1-5, item 4. No calculation has been found documenting the bases of the assumption that the RS droplet thermal effectiveness at NAPS Units 1 and 2 is 0.9. Per calculation [25.4.34], the RS nozzle design and orientation appear to be based on Beaver Valley Unit 1. Also, the calculation done for Beaver Valley Unit 1, [25.4.46], calculates and documents droplet effectiveness between 0.99 to 0.8 for a set of nozzles similar in arrangement to NAPS Unit 1. Note that per [25.8.8], it appears that NAPS Unit 2 has a different configuration of nozzles than Unit 1. However, none of the existing safety analyses or system flow analyses address this difference. NAPS Units 1 and 2 should have a plant-specific calculation documenting RS droplet thermal effectiveness. It should be noted that, subsequent to BV1 having changed its nozzles, calculation [25.4.46] has been voided.

[25.4.139]

24.2.4 Sections 2.2.1, 12.11, 14.1.2, 11.1.4, and 14.4.2; Tables 11.1-5, item 5, and 6.1-1. The containment safety analysis [25.4.2] [25.4.72] and the site boundary dose analyses [25.4.3] [25.4.73] [25.4.74] assume 100-percent containment atmosphere spray coverage instead of 38 percent for QS and 74 percent for RS as documented in calculation [25.4.31]. It is possible that from a heat removal standpoint due to the high RS flow, sufficient mixing would occur (due to the temperature/pressure gradients caused) to support the 100-percent coverage assumption. This evaluation, however, has not been documented. The site boundary dose analyses may need to be revised to address actual spray coverage or to state why an 100 percent effective spray coverage is valid.

Insert 24.2

24.2.10 Sections 2.2.2, 4.3, 6.2, 13.2.1, 14.1.1, 14.1.2, 14.1.16, and 15.1.3. It is not clear why the NAPS Units 1 and 2 RG 1.97 Compliance Report [25.3.48] does not list instrumentation to monitor RS System containment isolation valve position as instrumentation required to meet RG 1.97 Type B variables.

24.3.1 Sections 3.1, 11.1.4, 14.1.2, item 2, 14.2.2, and 12.9; Tables 11.1-5, item 6, and 6.1-1. Calculation [25.4.81] calculates available sump water volume to support RS System operation (assuming that the IRS and ORS pumps start up at 195 seconds and 210 seconds, respectively, following a CDA) for a 6-in., single-ended LOCA (not the DBA LOCA) and an MSLB (original licensing period evaluation). The calculation documents that, assuming normal safeguards, a full RS System flow is not possible until 420 seconds following an MSLB. The current design basis LOCA and MSLB have not been analyzed for sump inventory. However, the current LOCA and MSLB analyses [25.4.2] [25.4.72] [25.4.139] as well as the calculation that calculates RS effective time [25.4.111], assume that a full RS flow is available 300 seconds following a CDA with no documented bases for that assumption.

24.3.2 Sections 3.1, 3.2, 12.10, and 14.4.2; Table 6.1-1. The number of nozzles as given in Sections 3.1 and 3.2 of this SDBD is based on drawings 11715-FP-14B-7 [25.8.8] and calculation 11715-185N. [25.4.62] The number does not agree with the number given in 11715-FM-091A. [25.8.1] Furthermore, the number of nozzles does not agree with the number given in 12050-FM-091 [25.8.2] and 12050-FP-14 [25.8.8] for Unit 2. The Unit 2 nozzle arrangement is not addressed in any available calculation.

24.3.3 Sections 3.1, 3.3.1 and 13.3.5. To ensure system operation in accordance with the requirements of the safety analyses, the RS System flow diagrams [25.8.1] [25.8.2] should be revised to indicate isolation valves 1-RS-146, 147 located in the RS and SI cross-connect as locked closed.

24.3.4 Sections 3.3.1, 19.1, and 19.4. All information related to test mode operation (including preoperational testing) was obtained from [25.10.15] and [25.3.8], which are not design basis documents. This information needs to be confirmed.

24.3.5 Sections 3.3.1, 19.1, and [24.3.4]. No calculation has been found documenting the acceptability of developing at least 115 psig for ORS pump testing.

24.3.6 Sections 3.3.1, 19.1, and [24.3.4]. A calculation is needed to document the acceptability of developing at least 58 psig for NAPS Unit 1 and 46 psig for Unit 2 casing cooling pump testing. (NAPS Units 1 and 2 Technical Specifications [25.3.11] [25.3.12] require different discharge pressures for casing cooling pump testing.)

24.3.7 Sections 3.3.1, 14.2.16, and [24.3.4]. No documentation was found identifying the bases for the limiting filling pressure to the seal head tank (to prevent overpressurization of the diaphragm) for the ORS pumps mechanical seals.

Insert 24.2

24.2.17 Sections 2.2, 3.1, 5.2, 7.1, 11.1.4, 12.1, 12.2, 12.4, 12.5, 12.7, 12.8, 12.9, 12.10, 12.11, 13.1, 13.2, 13.3, 14.1.1, 14.1.2, 14.1.7, 14.1.20, 14.2.2, 14.2.4, 14.2.7, 14.3.2, 14.3.4, 14.4.2, 14.5.1, 14.5.2, 14.6.1, 14.6.7, 14.7.1, 14.7.2, 14.8.2, 14.8.7, 17.1.4, 17.1.5, 21.1, 21.3, 21.7, 22.1, 22.5, Tables 6.1-1, 11.1-5, 12.0-1, 19.1-1 19.2-1, 19.4-1.

Calculation [25.4.72] should be revised to address SG replacement at NAPS-1 and to document why the calculation (as is) is still considered the bounding MSLB analysis for NAPS Units 1 and 2.

01287.B07

24.6.1 Sections 11.1.4, 12.1, and 12.2; Table 6.1-1. Currently, there is no calculation documenting maximum allowable RS flow based on the RS pumps NPSH requirements.

24.6.2 Sections 11.1.4, 14.6.2, and 12.3; Tables 6.1-1 and 11.1-5, Item 13. Per calculation [25.4.61], the casing cooling flow reaches 800 gpm at 610 seconds following a DBA.
[25.4.139]
However, the containment analyses [25.4.4] [25.4.2] [25.4.72] and the NPSH analysis [25.4.2] assume a constant casing cooling flow of 800 gpm after a CDA signal. Based on this assumption, the NPSH analysis establishes that the minimum available NPSH to the ORS pumps occurs at 650 seconds. This assumption may not be conservative since the lower flow rates between 0 to 610 seconds will reduce the available NPSH. The effect of the above on the minimum available NPSH, on the time at which it occurs and on the containment analyses should be evaluated.

24.6.3 Table 6.1-1. Upon issuance, the SDBD-NAPS-SW must be reviewed to ensure that the service water conditions/requirements on the tube side of the RS cooler are identified. The appropriate section of the SW SDBD then should be listed in Table 6.2-1.

24.6.4 Sections 11.1.4, 14.1.2, Item 5, and 14.4.2; Tables 6.1-1 and 11.1-5, Item 4. Per response to NRC Question 6.80, the analysis documenting drop size was performed by the vendor, Spraying Systems Company. A copy of that analysis could not be found and therefore cannot be referenced.

24.7.1 Section 7.1. No documentation was found confirming the existence of an inducer in the RS pumps (to support the NPSH issue).

24.7.2 Section 7.2 states that there have been no substantive changes in RS System design due to programmatic and regulatory issues. This is based on the documentation available for review at the time of writing this document. The validity of this statement has to be confirmed by Virginia Power.

24.7.3 Section 7.3 summarizes the significant design changes since the issuance of the operating license. However, this summary is very limited since none of the EWRs listed in Chapter 20 were available for review at the time of writing this document. It also should be noted that the completeness of the list of DCPs and EWRs in Chapter 20 must be confirmed by Virginia Power.

24.7.4 Section 7.3 discusses modifications made to the RS coolers to prevent overpressurization of the diaphragms during a containment Type A test or following an accident. No design documentation was found recording the actual modification.

24.7.5 Section 7.4. There is currently no documented commitment to increase the design margin on the RS cooler heat transfer coefficient (to ensure an adequate allowance for tube fouling) in the event of purchase of replacement RS coolers. This is a

Insert 24.8.1

recommendation based on previous plant experience on excessive tube fouling [25.7.36]

24.9.1 Sections 14.2 and 14.1; Table 9.1-1. T following "as-purchased" information related to the RS pumps is obtained from the specification [25.6.1], which is not a controlled document. When these data are confirmed, this open item may be closed.

1. IRS and ORS pump curves (Sections 14.2.1 and 14.2.2)
2. Codes and standards (Sections 9.1 and 14.2.3)
3. Fluid temperature (Section 14.2.4)
4. Fluid chemistry (Section 14.2.4)
5. IRS and ORS pump supports (See Section 14.2.7 item 3)
6. RS pumps are made of austenitic stainless steel. (See Section 14.2.8)
7. Voltage insulation rating (Section 14.1.13 item 2)
8. Seismic accelerations (Section 14.2.9, item 1)
9. IRS and ORS pump design pressure (Section 14.2.4)

24.9.2 Sections 14.6 and 14.1.13; Table 9.1-1. The following as-purchased information related to the casing cooling pumps is obtained from the specification [25.6.8], which is an uncontrolled document. When these data are confirmed, this open item may be closed.

1. Code and standards (Sections 14.6.3 and 9.1)
2. Design condition fluid temperature 40°F to 65°F (Section 14.6.4)
3. Design fluid chemistry pH - 5 to 8 (Section 14.6.4)
4. Casing cooling pump motor data (Section 14.3.7 item 3)
5. Casing cooling pumps are made of austenitic stainless steel (See Section 14.6.8)
6. Voltage insulation rating (Section 14.1.13 item 2)
7. Seismic accelerations (Section 14.6.9)

24.9.3 Sections 14.5.3 and 25.2; Table 9.1-1. The effective date of API 650 could not be determined from the original design documentation. Review of vendor documentation is required to determine effective date.

24.9.4 Section 14.5 and Table 9.1-1. The following as-purchased information related to the casing cooling tank is obtained from the specification [25.6.3], which is an uncontrolled document. When these data are confirmed, this open item may be closed.

1. Codes and standards (Sections 14.5.3 and 9.1)
2. Thermal insulation of 2 in. (Section 14.5.9)
3. Seismic accelerations (Section 14.5.9)

Insert 24.8.1

24.8.1 Sections 13.2.1 and 13.2.4; Tables 8.2-1, 8.2-4. The UFSAR Change Request included with DC-90-13-1 addresses a revision to Section 6.2.2.2 of the UFSAR to include a commitment to RG 1.82, NUREG 0847, and NUREG/CR-2791. This open item should be deleted after confirmation that the UFSAR has been revised to include this change.

01287.B07

specification [25.6.9] indicates that the "as-purchased" heat transfer coefficient is 3.8? x 10 Btu/hr-°F assuming a fouling resistance of 0. Reference [25.7.11] suggests that a fouling resistance of 0.0003 hr-°F-ft²/Btu is acceptable for heat exchangers using service water but not in continuous service. Since tube fouling is inevitable, an evaluation has to be made to establish the design margin on the heat exchanger after fouling has been addressed.

24.11.7 Table 11.1-5, Items 3, 4, 10, 11, and 13 address effect of RS System parameters on sump fluid temperature and the possible effect on SI performance relative to temperature of fluid used for emergency core cooling during the recirculation phase. The validity of this statement has to be confirmed by Westinghouse and the appropriate Westinghouse calculation (if any) referenced.

[25.4.139]

24.11.8 Sections 12.9, 14.2.2, and 19.1; Table 11.1-5, Item 6. The containment safety analyses [25.4.4] [25.4.2] [25.4.72] assume full RS spray availability at 300 secs following a CDA. Calculation [25.4.111], which calculates RS spray effective time, indicates that with an IRS pump start signal received at 195 secs following a CDA, full RS spray will be available (based on diesel sequencing, system fill time, etc.) at 255.3 secs. Similarly, with an ORS pump start signal received at 210 secs following a CDA, full RS spray will be available at 292 secs. Technical Specification 4.6.2.2.1 calls for an IRS pump delay within 195 ± 0.75 secs and an ORS pump delay within 210 ± 21 secs. Based on Calculation [25.4.111], an IRS and ORS tolerance margin of ±0.75 and ±21 secs, respectively, could result in a full RS spray as late as (255.3 + 0.75) 256.1 secs and a full ORS spray as late as (292 + 21) 313 secs following a CDA. As a result, the ORS spray effective time could exceed the assumptions of the safety analyses due to the allowable tolerance margin. The basis for defining these tolerances is not known but is assumed to have been supplied in the manufacturer's information. The allowable tolerances need to be reviewed/revised to ensure that the requirements of the safety analyses are met.

24.11.9 Section 14.1.2 and Table 11.1-5, Item 8. The [illegible] [25.10.9] states that the coarse mesh screen opening for the containment su[illegible] or Unit 2 is 0.615 in. and the fine mesh for both Units is 0.12 in. No drawing/specification was found confirming the accuracy of this information.

24.11.10 Sections 12.9, 14.1.2, 14.2.2, and 14.2.7, Item 10; Tables 11.1-5, Item 6, and 23.1-3. Calculation [25.4.111] which calculates RS effective time assumes loss of offsite power concurrent with the accident, but does not address a LOOP subsequent to the accident. It is expected that the available margin in the depressurization time documented in the containment analyses [25.4.2] [25.4.72] is sufficient to meet the design basis containment depressurization requirements for other LOOP scenarios. This evaluation, however, needs to be documented in the safety analyses. Note that such an evaluation was performed for Surry Units 1 and 2. [25.4.131] [25.7.43]

[25.4.139]

24.12.1 Sections 14.2.11, 12.1, and 12.2. UFSAR Section 6.2.2.3.2.1 should be revised to reflect the most current minimum NPSHA values for the RS pumps as documented in Calculation [25.4.2]. and [25.4.139]

24.12.2 Sections 12.6 and 14.1.12. Calculation [25.4.76] which establishes spray and sump pH does not evaluate the impact of a boration level of 2300 ppm (NAPS Technical Specifications allow a boration level range of between 2300 ppm and 2400 ppm) on the sump pH.

24.13.1 Section 13.1.1, item 3. Upon issuance, the PDBD must be reviewed to ensure that codes and standards applicable to non-system-specific components (e.g., level indicators, flow switches, etc.) performing safety-related/non-safety-related functions are provided and the appropriate section referenced in this SDBD.

24.13.2 Section 13.1.1, item 4. A review is necessary to ensure consistency between the approved exceptions to ASME XI testing as a result of Virginia Power to NRC correspondence versus what is currently listed in the NAPS ISI program relative to RS components.

24.13.3 Section 13.1.2, item 1. The NAPS Quality Assurance Program was unavailable for review at the time of writing of this document. The document should be included as a reference and added to Chapter 13.1.2, item 1.

24.13.4 Sections 14.1.5, 13.1.2, item 2, and 15.2.1, item 3. No documentation has been found which evaluates the effect of seiches on structures/systems/components. When found, such documentation should be referenced in this section.

24.13.5 Sections 13.1.2, 13.2.1, 14.1.5, and 15.2.2, item 2. The NAPS Units 1 and 2 pipe rupture reports inside containment [25.3.54] [25.3.55] are not controlled documents. Virginia Power should confirm that the containment design related to protection of safety-related equipment from the dynamic effects of a HELB has been maintained.

24.13.6 Not used.

24.13.7 Sections 14.1.5, item 2, 13.1.2, 14.2.18, item 4, and 15.2.2, item 2. No documentation was found other than the UFSAR [25.10.45] relative to protection of safety-related components located within containment from internally generated missiles. When such documentation is found, it should be included as a reference in the SDBD.

24.13.8 Sections 14.1.5, item 2, 14.2.18, 13.1.2, and 15.2.2, item 2. No documentation was found other than the UFSAR [25.10.46] relative to protection of safety-related components located outside containment from pipe whip or jet impingement due to a HELB. When such documentation is found, it should be included as a reference in the SDBD.

25.1.61 NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants Resolution of Generic Technical Activity A-36," July 31, 1980.

insert 25.1

25.1.62 GL 89-10, "Safety Related Motor Operated Valves, Testing and Surveillance," June 28, 1989.

25.2 INDUSTRY CODES AND STANDARDS

25.2.1 ANSI N18.2, "Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants," 1973.

25.2.2 "Draft ASME Code for Pumps and Valves for Nuclear Power Plants," November 1968.

25.2.3 Modified API Standard 650, "Welded Steel Tanks for Oil Storage." [24.9.3]

25.2.4 ANSI B31.7.0, "Code for Nuclear Power Piping," 1969, Addenda through 1970.

25.2.5 "ASME Boiler and Pressure Vessel Code," Section III, 1968.

25.2.6 ANSI B31.1.0, "Power Piping Code," 1967.

25.2.7 ANS 5.1, "Decay Heat Power in Light Water Reactors," 1979.

25.2.8 ANSI N45.2.11, "Quality Assurance Requirements for the Design of Nuclear Power Plants," 1974.

25.2.9 IEEE Std. 308-1971, "Criteria for Class 1E Electric Systems for Nuclear Power Generating Stations," September 1971.

25.2.10 IEEE Std. 336-1971, "Installation, Inspection and Testing Requirements for Instrumentation and Electric Equipment During the Construction of Nuclear Power Generating Stations," May 1971.

25.2.11 IEEE Std. 279-1971, "Criteria for Protection Systems for Nuclear Power Generating Stations," May 1971.

25.2.12 USAS B31.1.0, "Code for Pressure Piping," 1967 Edition.

25.2.13 ANSI B16.5, "Steel Pipe Flanges, Flanged Valves and Fittings," 1973.

25.2.14 ANSI N271, "Containment Isolation Provisions for Fluid Systems," 1976.

Insert 25.1

25.1.63 RG 1.82, Rev. 1, "Water Sources for Long-term Recirculation Cooling Following a Loss-of-Coolant Accident," November 1985.

25.1.64 NUREG/CR-2791, "Methodology for Evaluation of Insulation Debris Effects," September 1982.

25.1.65 NUREG-0897, "Containment Emergency Sump Performance," October 1985.

25.1.66 GL 85-22, "Potential for Loss of Post-LOCA Recirculation Capability due to Insulation Debris Blockage," December 3, 1985.

01287.B07

25.4.124 11715/12050-E-14, Rev. A, "4160V Short Circuit Appendix R Evaluation," SWEC, May 23, 1977.

25.4.125 11715/12050-E-17, Rev. A, "Static Charger Calculation," SWEC (no date available).

25.4.126 EE-0057 Rev. 0, "DC System Equipment Sizing," Virginia Power, February 27, 1978.

25.4.127 12050-ES-225, Rev. 1, "Quench Spray Flow Delay Time to Sump," SWEC, April 20, 1981

25.4.128 12050-ES-226, Rev. 0, "Recirculation Spray Holdup Time Delay," SWEC, March 9, 1981

25.4.129 SM-429 (no revision number available), "Shutdown reactivity following a LOCA," Virginia Power, October 19, 1986.

25.4.130 EE-0027 (no revision number available), "Emergency Diesel Generator Load Sequencing," Virginia Power, February 12, 1989.

25.4.131 14937.65-US(B)-074, Rev. 2, "Sensitivity of Containment Depressurization and Subatmospheric Peak to IRS/ORS Timer Delay," SWEC, SPS, March 17, 1989.

25.4.132 ME-125 (no revision number available), "Maximum Differential Pressure Across Motor Operated Valves," Virginia Power, April 10, 1987.

25.4.133 ME-120 (no revision number available), "Maximum Differential Pressure Across Motor Operated Valves," Virginia Power, March 4, 1987.

25.4.134 13075.62-NP(B)-001-X2, Rev. 1, "Peak Spreading of Amplified Response Spectra for NAPS," SWEC, February 3, 1981.

25.4.135 13075.62-NP(B)-002-X2, Rev. 1, "Enveloping of Amplified Response Spectra for NAPS Units 1 & 2 Structures," SWEC, February 6, 1981.

25.4.136 13075.62-NP(B)-100-X2, Rev. 1, "Enveloping of Amplified Response Spectra for NAPS Units 1 & 2 Structures," SWEC, February 6, 1981.

25.4.137 11715-SEO-4433, Rev. 0, "Seismic Support for Cable Trays," SWEC, April 10, 1980.

Insert 25.4

INSERT 25.4

25.4.138 02072.1610-US(B)-273, Rev. 0, "Head Loss Across Emergency Sump Screens Due to Insulation Debris Caused by LOCA Events," May 27, 1992.

25.4.139 02072.2010-US(B)-274, Rev. 0, "Containment LOCA Analysis with New Steam Generator," June 17, 1992.

01287.B07

25.12 MISCELLANEOUS REFERENCES

25.12.1 E&DCR No. P-2152, "NA1, LHSI/RS Cross-Connect," January 17, 1978.

25.12.2 Alden Research Laboratories, "Hydraulic Model Studies of the Reactor Containment Building Sump," July 1977.

25.12.3 Notes of Conference, between AEC and SWEC, July 8, 1966.

25.12.4 EWR No 84-613.

25.12.5 DCP 78-06, "Casing Cooling Subsystem," Unit 1 (no date available).

25.12.6 DCP 81-S20A (through 81-S20K), "NUREG-0696 Short Term I&C Project Data Acquisition System," Units 1 and 2.

25.12.7 DCP No 81-046AS, Equipment Qualification of GEMS LVL XMTR, Unit 1.

25.12.8 DCP No. 81-046BS, Equipment Qualification of GEMS LVL XMTR, Unit 2.

25.12.9 SWEC Task Item 107, MOV ReRate Evaluation - GDC 17, NAPS Units 1 and 2, December 1984.

25.12.10 Telecon between Virginia Power (John Benton), and SWEC (T. Carson, N. Hanely, S. Ferguson), September 6, 1990, - Item 3.

25.12.11 DC 90-13-1, Steam Generator Replacement North Anna Unit 1.

NCRODP TRAINING MODULES CHANGES

This section presents a general description of the Main Steam System and is designed to provide the reader with a general understanding of system operation and interrelationships with other plant systems. This module is to be used for instructional purposes only and is not to be used to operate the system.

Purpose

The primary purpose of the Main Steam System is to direct dry, saturated steam from the steam generators to the high pressure (HP) turbine for the production of electricity. The Main Steam System provides the following additional functions, which are described in more detail later in this module:

1. removes heat from the Reactor Coolant System (RCS) via the decay heat release valve, safety valves, atmospheric dump valves, and/or steam dump valves;

2. supplies alternate supply steam to the Auxiliary Steam System;

3. supplies steam to the Gland Seal System;

4. reduces the adverse effects of a main steam line break accident by limiting and/or stopping blowdown of the steam generators; and

5. supplies steam to the steam driven auxiliary feedwater pump (Terry turbine).

Major Flow Path

INSERT C

Dry, saturated steam exits each of the three steam generators into a large seismically supported pipe (see Figure 23-1). ~~A venturi (flow restrictor) is provided in the main steam piping, just before it exits Containment, to limit the maximum steam flow during a steam line break accident. The venturi also provides connections for the measurement of steam line flow. A trip valve (TV) is provided downstream of the venturi, just outside Containment, to isolate the~~

INSERT C

A flow restrictor (venturi) is provided at two locations, one within the steam generator outlet nozzle (Unit 1 only) and one in the main steam piping (Units 1 & 2), just before it exits Containment. These flow restrictors are provided to limit the maximum steam flow during a steam line break accident. The main steam piping flow restrictor also provides connections for the measurement of steam line flow. A trip valve (TV) is provided downstream of the flow restrictor, just outside Containment, to isolate the main steam header, either manually or automatically, when abnormal conditions exist. A non-return valve (NRV), located downstream of the trip valve, prevents main steam header, either manually or automatically, when abnormal conditions exist. A non-return valve (NRV), located downstream of the trip valve, prevents blowdown of the unaffected steam generators in the event of a steam line break upstream of the NRV. A bypass valve is provided around each trip valve and each non-return valve to permit pressure equalization across the valves and to minimize "perking" of the NRV.

Safety valves, an atmospheric dump valve, and a connection to the decay heat release valve are provided just upstream of the trip valve. These valves, in conjunction with the main turbine and the Steam Dump Control Subsystem, provide for energy removal from the RCS. Connections to the steam generator blowdown tank and main condenser allow blowing moisture from the main steam headers during startup of the system. Steam traps are located throughout the system to drain moisture during normal operation.

The three main steam headers join in a large common distribution manifold. This manifold provides steam to the following loads:

1. HP turbine,

2. Auxiliary Steam System,

3. steam dump valves,

4. moisture separator reheaters (MSR), and

5. Gland Seal System.

Steam to the moisture separator reheaters is used to superheat the exhaust steam from the HP turbine prior to entering the low pressure (LP) turbine. The steam supply to the moisture separator reheaters is controlled by a reheat control valve and is normally fully open during operation. This steam is almost totally condensed in the moisture separator reheaters and is normally directed to the high pressure (1st point) feedwater heaters. During warmup of the moisture separator reheaters, this steam is exhausted to the main condenser.

INSERT D

Major Components

This subsection describes, in general terms, the major components associated with the Main Steam System. The components are discussed in the order that they are encountered in the flow path.

~~Flow restricting venturi. The flow restricting venturi is mounted inside the main steam piping and provides a gradual change in the internal diameter of the main steam piping. The venturi acts as a nozzle during a steam line break accident to limit the maximum steam flow experienced through the piping. Additionally, the venturi provides static pressure taps for the development of flow signals.~~

Main steam safety valves. Five main steam safety valves are provided for each steam generator. The valves are designed to protect the integrity of the main steam piping from overpressurization. The valves are basically spring-opposed relief valves.

Atmospheric dump valve. One atmospheric dump valve is provided for each steam header and is designed to relieve excessive RCS energy to the atmosphere. The pressure setpoint at which the valve opens is variable but is normally set to open at 1035 psig, a pressure below the main steam safety valve settings. The safety valves have experienced problems with reseating, once opened. For this reason, the atmospheric dump valve opens first.

Decay heat release valve. One common air-actuated decay heat release valve is provided for the three steam headers. The valve allows RCS energy to be relieved to the atmosphere at a variable rate, as determined by the control room operator.

Main steam trip valves. One air-actuated main steam trip valve is provided in each main steam line. The valves are designed to isolate the associated steam header in the event of a steam line break accident and are essentially air-operated check valves.

INSERT D

__Flow restrictors.__ Two flow restrictors (venturis), one within the steam outlet nozzle (Unit 1 only) and one inside the main steam piping (Units 1 & 2), provide a gradual change in the internal diameter of the main steam piping. The flow restrictors act as a nozzle during a steam line break accident to limit the maximum steam flow experienced through the piping. Additionally, the main steam piping flow restrictor provides static pressure taps for the development of flow signals.

This section presents a detailed description of the flow paths, major components, power supplies, and maintenance procedures associated with the Main Steam System and other information that ensures the safe and reliable operation of the system.

Three main steam headers are provided for each reactor unit. The components associated with Unit 1 are described in this module. Significant differences between Unit 1 and Unit 2 are noted where appropriate.

Flow Paths

INSERT E

Approximately 4×10^6 lbm/hr of dry, saturated steam exits each steam generator under a pressure of 885 psig, during normal full power operation. The steam generator 32-inch outer diameter main steam discharge piping runs horizontally through Containment, penetrates the polar crane wall, and is then directed vertically downward for a distance of approximately 41 feet. ~~A flow restricting venturi, 15 feet in length, is housed in the vertical run of piping and is designed to limit the maximum steam flow during a steam line break accident. Two static pressure taps are located on each side of the venturi and provide the necessary pressure signals to pressure transmitters PT-1474 and -1475 (see Figure 23-1). The main steam piping (header 1A) exits Containment horizontally, through penetration number 1_.~~

The main steam piping enters the Main Steam Valve House at the 285-foot elevation. The main header contains an electro-pneumatic main steam trip valve and a motor-operated non-return valve. Steam traps are located throughout the system to facilitate the removal of moisture in the Main Steam System. Each of the three headers have branches in the Main Steam Valve House as follows:

1. a 3-inch line, which provides steam to the steam-driven auxiliary feed pump;

2. a 1-1/2-inch line, which provides the following functions:

    a. allows nitrogen purging of the steam generators (3/4- inch line).

INSERT E

Two flow restrictors (venturis), one within the outlet nozzle of the steam generator (Unit 1 only) and one housed in the vertical run of main steam piping (Units 1 & 2), are designed to limit the maximum steam flow during a steam line break accident. Within the main steam piping flow restrictor, two static pressure taps are located on each side of the venturi and provide the necessary pressure signals to pressure transmitters PT-1474 and 1475 (see Figure 23-1). The main steam piping (header 1A) exits Containment horizontally, through penetration number 73.

INSERT F

with the MSRs is listed in Table 23-1. The major system valves are located in the Main Steam Valve House with the exception of the MSR reheat control valves and the steam dumps, which are located in the Turbine Building. A listing of component locations is provided in Table 23-2. Control of the major components is from the MCR. Alternate control of the atmospheric dump valve is provided at the Auxiliary Shutdown Panel.

~~Flow restricting venturi. A flow restricting venturi (see Figure 23-1) is installed in each main steam header to limit the steam flow rate and subsequent RCS cooldown during a main steam line break accident. Additionally the flow venturi is used for steam flow measurement. The design of the flow restrictor is based on the following:~~

1. minimizing the cooldown rate of the RCS during a steam line break accident,

2. reducing pipe whip during a steam line break accident by limiting steam flow.

~~3. providing a differential pressure for the measurement of steam flow.~~

3 ~~4~~. minimizing the pressure loss (head loss) across the restrictor while at the same time restricting steam flow to acceptable values,

4 ~~5~~. withstanding the designed number of pressure and thermal cycles experienced over the life of the plant, and

5 ~~6~~. maintaining flow restrictor integrity in the event of a double-ended shear of the main steam line immediately downstream of the restrictor.

~~The venturi has an overall length of 15 feet and a throat diameter of 16 inches. The inlet diameter to the tube is 30 inches, with an outlet diameter of 32 inches. The flow restricting venturi is housed in a vertical run of piping outside the containment polar crane wall.~~

INSERT G

INSERT F

Flow restrictors. Two flow restrictors (see Figure 23-1), one within each steam generator outlet nozzle (Unit 1 only) and one installed in each main steam header (Units 1 & 2) are designed to limit the steam flow rate and subsequent RCS cooldown during a main steam line break accident. The design of the flow restrictors is based on the following:

INSERT "~~X~~ G"

The main steam piping flow restrictor is comprised of a venturi that has an overall length of 15 feet and a throat diameter of 16 inches. The inlet diameter to the tube is 30 inches, with an outlet diameter of 32 inches. This flow restrictor is also designed to provide a differential pressure for the measurement of steam flow. The flow restrictor is housed in a vertical run of piping outside the containment polar crane wall. The main steam piping flow restrictor is common to both Units 1 & 2.

The steam generator outlet nozzle flow restrictor is comprised of seven venturis housed within a single forging. The seven venturis have a net flow area of 1.4 square feet. The flow restrictor is welded into the outlet nozzle of the steam generator. The steam generator outlet nozzle flow restrictor is common to Unit 1 only.

Steam generators. Each steam generator is a vertical shell, U-tube heat exchanger that is the heat transfer interface between the primary and secondary loops. The steam generators are divided into two major portions: the primary side and the secondary side. The primary side consists of a cast hemispherical chamber, internally partitioned by a divider plate which separates the chamber into inlet and outlet chambers. The primary and secondary sides are separated by the tube sheet and the U-tubes. The tube sheet is penetrated by ~~3388~~ 3592 (Unit 1) and 3388 (Unit 2) U-tubes which are welded to the tube sheet. The reactor coolant enters the primary side of the steam generator into the inlet chamber. The coolant then enters the U-tubes, flows up and through the tubes, exiting into the outlet chamber, where it returns to the RCP.

The secondary side of the steam generator contains feedwater, recirculating water (hot water drainage from the first and second stage separators), and steam. The boundaries for the secondary side consist of the upper and lower shell, the top of the tubesheet and the outside of the U-tubes. The internal structure of the secondary side consists of four distinct regions:

1. the downcomer - the circular area (annulus) between the tube bundle wrapper and the outer shell of the steam generator extending from the moisture separators to the tubesheet;

2. the evaporator - the area inside the tube wrapper extending from the tubesheet to the top of the tube bundle;

3. the riser - the transition area between the evaporator and the steam drum extending from the top of the tube bundle to the bottom of the moisture separators; and

4. the steam drum - the area inside the shell and top head extending from the bottom of the moisture separators to the main steam outlet nozzle.

During normal operations, feedwater enters the steam generator at the main feedwater nozzle which is located in the upper portion of the downcomer section. Feedwater flows into the downcomer via the main feed ring which is an annular-shaped, horizontal pipe with J-tube extensions. As the feedwater discharges from the feed ring, it mixes with and is preheated by the recirculating water from the steam drum. These two sources of water flow downward through the downcomer region, over the tubesheet, and vertically upward into the evaporator.

The preferred method is the use of the temporary piping arrangement (See Figure 38-60). The piping connects to a blank flange downstream of the LMC for 1-SI-TV-1842 (SI-232), passes through individual loop isolation valves, and connects to a blank flange upstream of RC-195. The pressurized water in the accumulator flows through the open LMC valve, the individual manual isolation valve, and RC-195, pressurizing the volume between the discs. Relief line isolation valve RC-77 must be shut to prevent accumulator water from flowing to the reactor side of the loop stop valve.

Prior to plant operations the loop stop valve body is vented, the pressurizer connections removed, and the blind flanges installed. During loop stop valve body pressurization with the accumulators, accumulator level must be maintained in the operating range, with accumulator pressure above 450 psig. Only (1) SI accumulator may be pressurized to > 350 psig when disc pressurization is in service. The PZR PORV's are only capable of relieving the pressure resulting from the inadvertent dumping of (1) SI accumulator when pressurized to > 350 psig. If 2 or more SI accumulators at > 350 psig should inadvertently dump the RCS could be overpressurized and brittle fracture could occur.

__Steam generators__. The steam generator (see Figure 38-19, 61, 62 and 67) is a vertical, pressurized-water, Class A vessel. The vessel is constructed with carbon steel and is located in the Containment Building, extending from the 259-foot level to the 323-foot level. The steam generator weighs ~~662,000~~ 702,000 (Unit 1) and 662,000 (Unit 2) pounds (dry) and is 812 inches high. The vessel shell consists of a lower head and shell, a transition cone and a steam drum, with an upper head. The upper steam drum has an outside diameter of 175.75 inches and the lower shell has an outside diameter of ~~135~~ inches. The steam generator serves as the heat exchange boundary between the primary and secondary loops, and is divided into primary and secondary sides.

OK

CHANNEL HEAD (HEMISPHERICAL CHAMBER)

The PRIMARY SIDE of the steam generator is an integral part of the RCS (see Figure 38-70). The primary side is comprised of a hemispherical chamber, a tubesheet, U-tubes, and manways. The CHAMBER is made of carbon steel clad on the internal surfaces with austenitic stainless steel. The chamber is internally partitioned with a non-removable DIVIDER PLATE that is ~~clad with~~ Inconel. The divider plate separates the chamber into inlet and outlet chambers. The divider plate has a small hole drilled through it which allows both chambers to be drained prior to maintenance. The hole permits a small amount of reactor coolant to bypass the U-tubes during normal flow conditions. The bypass flow has a negligible effect on the heat transfer from the RCS to the secondary side fluid. The RCS hot leg penetrates the

CHANNEL HEAD

INSERT AA

inlet chamber at the 31-inch inlet nozzle, while the cold leg penetrates the chamber at its 31-inch outlet nozzle. The nozzles are clad with stainless steel for corrosion protection and ~~have~~ ~~carbon-steel junctions~~ to provide a means of welding to the RCS piping.

ARE PROVIDED WITH A SAFE END

The upper boundary of the chamber is formed by the TUBESHEET. The tubesheet is a flat disc forging 21 inches thick and clad with Inconel on the reactor coolant side for corrosion protection and compatibility with the U-tubes. The tubesheet forms the pressure boundary between the primary and secondary plants and is designed to withstand a 1600 psid primary-to-secondary dp, or a 670 psid secondary-to-primary dp. ~~The tubesheet has 6776 penetrations which provide access to the 3388 U-tubes and is welded to the chamber and the lower shell along its perimeter. In order to provide a large heat transfer area, the steam generator has a tube bundle which consists of 3388, vertically inverted, 0.875-inch (outer diameter), .050 inch wall thickness, U-TUBES.~~ Since these tubes also form part of the primary boundary, they are constructed of Inconel for increased tube integrity. The U-tubes are expanded at the end and welded onto the reactor coolant face of the tubesheet. The tubes are supported against lateral vibrations by ~~an~~ anti-vibration bars at the U-bend of the tubes. By reducing or limiting the lateral vibration, the likelihood of fatigue failure is reduced.

CHANNEL HEAD

The MANWAYS provide access to the ~~primary chamber~~ for maintenance and inspection. There are two manways, each of which is closed by a manway cover sealed with a stainless steel gasket retainer plate and a spiral-wound gasket.

The SECONDARY SIDE of the steam generator contains feedwater, recirculating water (hot water drainage from the first and second stage separators), and steam. The boundaries for the secondary side consist of the upper shell and the steam drum, the top of the tubesheet, and the outside of the U-tubes. The internal structure of the secondary side consists of four distinct regions:

1. the downcomer - the circular area (annulus) between the tube bundle wrapper and the outer shell of the steam generator extending from the separators to the tubesheet;

2. the evaporator - the area inside ~~the tube~~ wrapper extending from the tubesheet to the top of the tube bundle;

INSERT AA

The tubesheet is welded to the lower shell and channel head along its perimeter. There are 7184 penetrations in each steam generator tubesheet for Unit 1, and 6776 penetrations for each steam generator tubesheet for Unit 2. These penetrations provide access to the U-tubes. In order to provide a large heat transfer area, the steam generator has a tube bundle which consists of vertically inverted, 0.875-inch (outer diameter) U-TUBES. There are 3592 U-tubes for each Unit 1 steam generator and 3388 U-tubes for each Unit 2 steam generator.

3. the riser - the transition area between the evaporator and the steam drum extending from the top of the tube bundle to the bottom of the separators; and

4. the steam drum - the area inside the shell and top head extending from the bottom of the separators to the main steam outlet nozzle.

(UNIT 2 ONLY)

The steam generator secondary side is comprised of a shell, a feedwater inlet nozzle, a tube bundle wrapper, a downcomer flow resistance plate assembly, moisture separators, and a wet layup nozzle. The SHELL consists of a lower shell, a transition cone, a steam drum, and an upper head. The shell material is made of carbon steel, with the tubesheet welded to the shell along its perimeter at the lower end. The transition cone is welded to the upper end of the lower shell, with the steam drum and the upper head welded together above that. The shell has the following penetrations:

1. feedwater inlet,

2. bottom blowdown connections,

3. secondary side drain connection,

4. manways,

5. handholds,

6. steam nozzle,

7. instrumentation connections, and

8. wet layup nozzle.

The feedwater inlet nozzle supplies feedwater from the Main Feedwater System or the Auxiliary Feedwater System. The two bottom blowdown connections are attached on the secondary side above the tubesheet and connect to the Steam Generator Blowdown and Recovery System. This connection is used to control the water-solids concentration in the steam generator as well as for continuous sampling by the Secondary Sampling System. The drain connection is used to drain the secondary side of the steam generator during maintenance periods. It connects to the bottom blowdown lines or can be valved to the containment sump.

UNIT 2 STEA GENERATORS A PROVIDE WITH

INSERT B

There are two 16-inch manways which provide access to the steam generator. Each manway is sealed with Inconel gaskets to prevent leakage. The manways are located on the upper section of the shell on the steam drum. ~~There are also~~ two 6-inch handholds located above the tubesheet which allow viewing of the secondary side of the tube sheet. A 2.5-inch hole, machined through the shell and wrapper between the fifth and sixth support plates, provides an inspection port for the tube region. The steam nozzle penetrates the upper head at the top of the steam generator and connects to the Main Steam System. The vessel also has instrumentation penetrations for pressure and water level.

The wet layup nozzle connects the steam generator to the Steam Generator Wet Layup Circulation System. The wet layup system provides forced circulation inside the steam generator during cold shutdown wet layup to ensure proper mixing of water chemistry chemicals.

The feedwater enters the steam generator at the feedwater nozzle (see Figure 38-61) which connects to a perforated pipe known as the feed ring. The feed ring distributes the incoming feedwater into the downcomer region. The feedwater exits through the top of the inverted J-tubes, flowing downward into the downcomer. The J-tubes are not evenly distributed around the ring. There are more J-tubes that supply feedwater to the inlet chamber side of the downcomer than supply water to the outlet chamber side. This results in an increased circulation ratio, increasing the velocity across the tubesheet, enhancing the sludge removal by the blowdown system, and reducing hot side superheat. (See Figure 38-64, 65, 66 and 69).

The feedwater from the feed ring mixes with, and is preheated by, the hot drainage (recirculating water) from the separators. This secondary side water flows through the downcomer region by traveling downward over the inside wall of the shell and over the outside wall of the tube bundle wrapper. The TUBE BUNDLE WRAPPER is a steel cylinder that fully encloses the tube bundle, separating the downcomer from the evaporator region. The wrapper not only directs flow, but also provides additional heating of the incoming feedwater prior to its entry into the evaporator region. The preheating, in conjunction with the feedwater preheating by the recirculating water and the Main Feedwater System heaters, prevents thermal shock to the tubesheet and the U-tubes. The tube bundle wrapper is ~~welded~~ to the lower shell.

CONNECTED

INSERT "B"

Unit 1 steam generators are provided with four 6-inch handholes which are located approximately 14 inches above the tubesheet (two centered with the tubelane and two 90 degrees to the tubelane). Two additional 6-inch handholes are aligned with the tubelane above the flow distribution baffle, bringing the total number of handholes to six. Two 4-inch inspection ports are located on the transition cone lower shell at an elevation slightly above the top tube support plate and aligned with the tubelane. This location facilitates inspection of the top support plates and Row 1 tube u-bends. The u-bend region of the Row 1 tubes are directly observable through these inspection ports. The steam nozzle penetrates the upper head at the top of the steam generator and connects to the Main Steam System. Within the steam generator outlet nozzle is a flow restrictor comprised of seven venturis housed in a single forging that is welded to the inside of the nozzle. This flow restrictor reduces the rate of energy released to the Containment during a main steam line break.

PLATES CONNECTED

Tube support ~~baffles~~ are ~~welded~~ to the inner surface of the wrapper. These support ~~baffles~~ maintain correct U-tube alignment and spacing. A transition cone is welded to the upper end of the wrapper to provide a transition between the evaporator and the steam drum (the riser section). A sufficient gap is provided between the bottom of the wrapper and the tubesheet, allowing uniform introduction of the feedwater into the evaporator region.

OF THE UNIT 2 STEAM GENERATOR'S

Located in the downcomer region is a ring-shaped perforated plate, referred to as the DOWNCOMER FLOW RESISTANCE PLATES. The ring is segmented into twenty sections, each of which are welded to the wrapper via two perpendicular gusset plates per section. A 0.19 to .25-inch gap is maintained between the radial edge of each segment and the shell. The purpose of the plate is to reduce the flow induced stresses on some of the tubes by creating better flow characteristics in the downcomer annulus.

quatrofoil tube support hol[es]

At the bottom of the downcomer region, the secondary fluid flows inward over the secondary face of the tubesheet, then upward around the U-tubes through ~~3/4 inch flow holes drilled in the tube support plates~~. As it rises over the surface of the tubes in the tube bundle,

3,310

this water is heated at the rate of ~~3.167~~ X 10⁶ Btu/hr by the reactor coolant flowing inside the tubes. This heat transfer process results in boiling as the secondary side water reaches the saturation temperature corresponding to the steam generator pressure. The steam-water mixture which is formed continues to rise upward in the evaporator region. As it rises, the quality of the mixture increases. At the top of the evaporator region the mixture exits to the steam drum via the riser region. As the steam rises through the deck in the riser region, it passes through steam chimneys. The chimneys project 16 inches above and below the deck and are flared at the top and bottom. The chimneys prevent water draining off the deck from mixing with the rising steam.

For the ... and three 3/4 in ... holes f... Unit 2

The moisture-laden steam enters the steam drum where it undergoes a water-steam separation process in the SEPARATORS (see Figure 38-68). The first stage separator, at the top of the tube bundle wrapper, consists of three swirl vane assemblies, each of which is comprised of four flat blades welded to a central hub (see Figure 38-71). The blades are also welded to a swirl vane wrapper and make up the first stage in the steam separation process. As the wet steam enters the separator, the swirl vanes force the steam into a circular motion. The centrifugal action imparted to the mixture separates the steam from the water by forcing the denser water outward to the wall of the separator. The water collects on the wall of the separator and drains over the lip of an inner barrel, returning to the downcomer region, where

99.90 PERCENT STEAM WITH A MAXIMUM OF 0.10 PERCENT MOISTURE FOR UNIT 1 STEAM GENERATORS AND IS

it mixes with and preheats the incoming feedwater. The majority of the water is removed in this process, with the steam rising to the second stage separator through deflectors. The three deflectors are circular plates welded into a "wedding cake" arrangement 14 inches high. The plates are vertically separated by fins. The bottom two plates are doughnut-shaped; the top plate is solid. The deflectors provide additional change of direction for the steam leaving the swirl vane assemblies.

The second stage separator consists of two-tier, four-bank vane assemblies installed parallel to the direction of steam travel (see Figure 38-72). The directional changes in the surface contour of the vanes collect the water from the steam-water mixture. The water drains through eight 4-inch pipes welded into the bottom drain channel of the vane banks. Each drain pipe is approximately 137 inches long. The drain pipes run vertically down through the steam chimneys or through existing deck slots. Each pipe has a drain cup welded to the pipe exit to limit the entry of steam from below. Drain pipes which pass through the chimneys have ladder rungs welded to them, providing easy access to the feedwater pipe area. The water is directed into the downcomer. The steam that exits the second stage separator is 99.75 percent steam with a maximum of 0.25 percent moisture. FOR UNIT 2 STEAM GENERATORS

At the top of the steam generator, the 32-inch steam outlet nozzle is the connection from the steam generator to the Main Steam System. A reducing connection is welded to the nozzle to connect the 32-inch nozzle to the 28-inch steam piping. An internal flow restrictor is welded into the steam outlet nozzle on Unit 1.

<u>Pressurizer</u>. The pressurizer (see Figure 38-20) is a 1400 ft^3 vertical, cylindrical pressure vessel whose purpose is to maintain RCS pressure during steady state operation, limit the changes in pressure during transient operations, and prevent the RCS from exceeding the design plant pressure. The pressurizer consists of the pressurizer vessel, the spray line and valves, the surge line, power-operated relief valves, safety valves, and a pressurizer relief tank. The pressurizer is located in the Containment and extends from the 265-foot elevation to the 310-foot elevation.

During normal operations, the electrical load on the plant is relatively constant; however, load changes are possible. During decreases in load which cause increased reactor coolant temperature, the density of the coolant decreases. The coolant expansion causes an insurge into the pressurizer, causing pressure to increase. The spray system responds to the increase in pressure by injecting subcooled water into the upper portion of the pressurizer which is filled

3. secondary side leak testing, and

4. initiation of cold feedwater into the steam generator.

The surveillance requirements for inspection of the steam generator tubes ensure that the structural integrity of this portion of the RCS is maintained. The program of in-service inspection of steam generator tubes is based on a modification of Regulatory Guide 1.83, Revision 1. In-service inspection of steam generator tubing is essential in order to maintain surveillance of the conditions of the tubes in the event that there is evidence of mechanical damage or progressive degradation due to design, manufacturing errors, or in-service conditions that lead to corrosion. In-service inspection of steam generator tubing also provides a means of characterizing the nature and cause of any tube degradation so that corrective measures can be taken.

The plant is expected to be operated in a manner such that the secondary coolant is maintained within those chemistry limits found to result in negligible corrosion of the steam generator tubes. If the secondary coolant chemistry is not maintained within these limits, localized corrosion may likely result in stress corrosion cracking. The extent of cracking during plant operation is limited by the amount of allowed steam generator tube leakage between the primary coolant system and the secondary coolant system (500 gpd per steam generator). Cracks having primary-to-secondary leakage less than this limit during operation have an adequate margin of safety to withstand the loads imposed during normal operation and by postulated accidents. Operating plants have demonstrated that this primary-to-secondary leakage can readily be detected by radiation monitors during steam generator blowdown. Leakage in excess of this limit requires plant shutdown and an unscheduled inspection, during which the leaking tubes will be located and plugged.

North Anna has NRC approval for up to 18 percent steam generator tube ~~tub~~ plugging. This also increases the maximum FQ limit from 2.15 to 2.19. The LOCA safety analyses performed for these conditions were based on double-ended cold-leg guillotine breaks with discharge coefficients of 0.6 and 0.4. For the limiting case, a discharge coefficient of 0.4, the results were a peak cladding temperature of 2165°F, a maximum local cladding oxidation of 5.77 percent and a whole core metal-water reaction of less than 0.3 percent.

The non-LOCA safety analysis was performed to evaluate the effects of this tube plugging regarding reduced primary flow resulting in more sever pump coastdown

TABLE 38-1

PRINCIPAL DESIGN PARAMETERS OF THE REACTOR COOLANT SYSTEM

Design thermal power	2893 MWt 9.8738 x 10^9 Btu/hr
Design pressure	2485 psig
Design temperature (except pressurizer)	650°F
Coolant flow rate through core	105.1 x 10^6 lbm/hr
Average temperature, maximum	580.8 ~~586.8~~°F (Unit 1) 586.8°F (Unit 2)
Normal operating pressure	2235 psig
System volume	10,089 ft^3 (UNIT 1) 9957 ft^3 (UNIT 2)

TABLE 38-3
STEAM GENERATOR PARAMETERS

Number	three per unit
Type	vertical U-tube
Number of tubes	3388 (UNIT 2) 3592 (UNIT 1)
Tube outside diameter	0.875 in.
Nozzles and manways	
Primary inlet nozzle (1 ea), ID	29.5 in.
Primary outlet nozzle (2 ea), ID	31.0 in.
Steam nozzle (1 ea), ID	32.0 in.
Feedwater nozzle (1 ea), nominal	16.0 in.
Instruments taps (6 ea), nominal	1 in.
Primary manways (2 ea), ID	16 in.
Secondary manways (2 ea), ID	6 in.
Bottom blowdown (2 ea) nominal	2 in.

. - UNIT 1,
2ea - UNIT 2

Primary side design

Design pressure	2485 psig
Design temperature	650°F
Design thermal power (NSSS), total	3310×10^6 Btu/hr (UNIT 1) 3167×10^6 Btu/hr (UNIT 2)
Coolant flow	35×10^6 lbm/hr
Normal operating pressure	2250 psig (UNIT 1) 2235 psig (UNIT 2)

TABLE 38-3
STEAM GENERATOR PARAMETERS
(Continued)

Secondary side design	
Design pressure	1100 psig
Design temperature	650°F
Normal operating steam pressure, full load	879 psig (UNIT 1) 790 psig (UNIT 2)
Normal operating steam pressure, full load	529.2°F (UNIT 1) 519°F (UNIT 2)
Steam flow (each)	4.04 x 10^6 lbm/hr (UNIT 2) 4.25 x 10^6 lbm/hr (UNIT 1)
Dimensions	
Overall height, including support skirt	812 in.
Upper shell outside diameter	175.75 in.
Lower shell outside diameter	135 in.
Dry weight	702,000 lbs. (UNIT 1) 662,000 lbs. (UNIT 2)
Flooded weight (70°F water)	1,125,000 lbs. (UNIT 1) 1,094,800 lbs. (UNIT 2)
Operating weight	838,000 lbs. (UNIT 1) 801,000 lbs. (UNIT 2)
Maximum tube sheet differential pressure	1600 psid

LIST OF FIGURES
(Continued)

Figures

* TO BE UPDATED FOR UNIT 1 FROM THE WESTINGHOUSE TECHNICAL MANUAL FOR MODEL 51F STEAM GENERATORS

NUCLEAR CONTROL ROOM

OPERATOR DEVELOPMENT PROGRAM

NORTH ANNA POWER STATION

MODULE NCRODP-52

SAFETY INJECTION SYSTEM

Virginia Power
Nuclear Training

19.4% (24.9% for Unit 2)

2. RWST level is below ~~24.9 percent~~, and

3. Either MOV-1863A or B respectively has opened.

During the recirculation mode, the LHSI pumps take a suction on the containment sump. If the recirculation line isolation valves did not shut radioactive gases from the sump water would be released to the atmosphere through the RWST vent. The valves do not shut until minimal cooling flow is ensured by MOV-1863A or B opening.

The LHSI pumps take a suction on either the RWST or on the containment sump. During normal operations and the injection mode, the LHSI pumps are lined up to receive water from the RWST through motor-operated, isolation valves MOV-1862A and B. During the recirculation mode, these isolation valves are shut and the motor-operated, isolation valves MOV-1860A and B from the containment sump are opened. On receipt of a low-low RWST level, MOV-1860A and B will open automatically if a SI recirc mode signal is present and the respective LHSI pump recirculation valves have shut.

The LHSI pump discharge can be directed to the RCS cold legs, the HHSI pump suction, or the RCS hot legs. During normal plant operations and the injection mode, the discharge of the pumps is lined up to the RCS cold legs through normally open, pump discharge valves MOV-1864A and B and a pair of normally open, isolation valves MOV-1890C and D that are piped in parallel. The motor operators for MOV-1890 C and D are normally deenergized with their breakers locked open. On initiation of the recirculation mode, the discharge of the LHSI pumps continues to the RCS cold loops with some portion being directed to the suction of the HHSI pumps through normally shut, isolation valves MOV-1863A and B. This lineup ensures net positive suction head to the HHSI pumps, since water is no longer being provided to the HHSI pumps from the RWST. During the recirculation mode, the discharge of the LHSI pumps is periodically lined up to the RCS hot legs through normally shut, isolation valves MOV-1890A and B. On Unit 1, the outside recirculation pumps RS-P-2A and B can discharge to the LHSI pump discharge headers in the event of failure of one or both of the LHSI pumps. Each outside recirculation pump is normally isolated from the corresponding LHSI pump by a pair of series manual isolation valves. They are operated from outside the safeguards building with a T-handle wrench inserted into the associated remote valve operator (a recessed, square-shaped hole in a round, brass device).

LHSI pumps discharge flow indication is provided for each pump on Control Room vertical board Safeguards Panels. Flow transmitters FT-1945 and -1946 measure 0 to 500 inches of water differential pressure across FE-1945 and -1946 for LHSI pumps 1A and B and transmit a signal to 0 to 4000 gpm flow indicators FI-1945 and -1946.

Hydrostatic test pump. The hydrostatic test pump is controlled by START and STOP pushbuttons and a speed controller on Unit 1 Control Room vertical board Safeguards Panel, Train A. Green and red indication lights are on the handswitch. Depressing the START pushbutton shuts the hydrostatic test pump circuit breaker and energizes the solenoid valve that regulates air to the speed controller. The speed of the test pump is controlled by the hand indicator controller HIC-1947. The electric input is converted to a pneumatic signal, and there is an approximate time delay of 15 seconds between adjustment of the speed controller and actual speed change of the pump. Depressing the STOP pushbutton energizes time delay relays that open the pump circuit breaker and deenergize the solenoid valve after a specified time delay. By procedure, the pump is not stopped until 15 seconds have elapsed since the speed was reduced to minimum. This ensures that the pump will be at the slowest speed when it is started again.

Tank Instrumentation and Controls

Accumulators. Each accumulator has redundant level and pressure indications on the Control Room vertical board Safeguards Panels. Each pressure transmitter provides indication on a 0 to 800 psig pressure gage and inputs to the HIGH/LOW ACCUMULATOR TANK PRESSURE alarm (window 1J-D8). Each level transmitter provides indication on a 0 to 100 percent level gage and provides input to the HIGH/LOW ACCUMULATOR TANK LEVEL alarm (window 1J-D7). The level indication range for the accumulators is about a 12 inch span and equals 7216 gallons at 0 percent level to 7932 gallons at 100 percent level. The setpoints for the common pressure and level alarms are listed in Table 52-6.

Refueling water storage tank. The RWST is provided with four channels of level indication on Control Room vertical board Safeguards Panels. Level transmitters LT-QS100A, B, C, and D provide signals to level indicators LI-QS100A, B, C, and D. The level indication range for the RWST is 16,784 gallons at 0 percent level to 487,000 gallons at 99.1 percent level. Level transmitters LT-QS100C and D also provide a signal to a RWST LOW LEVEL alarm (window 1J-A2) at ~~28.3%~~. When two of the four channels have sensed a low-low RWST
22.8% (28.3% for Unit 2).

19.4% (24.9% for Unit 2)

level condition at ~~24.9~~ percent, a signal is initiated to automatically realign safety injection to the recirculation lineup and annunciate RWST in BYPASS MODE (1J-C2). The RWST has a fifth level transmitter LT-QS103 which actuates an RWST HIGH LEVEL alarm (window 1K-E1) at 98.5% and an RWST LEVEL BELOW NORMAL alarm (window 1K-E2) at 96.7%. These alarms inform the operator if the RWST level is not within the proper range as required by Technical Specifications. The alarm setpoints are listed in Table 52-6. Further details on RWST controls and indications are presented in the QS module (NCRODP-53).

Boron injection tank. Local BIT temperature indication is provided by temperature controllers TIC-1934A and B. The BIT strip heaters are energized at 141°F and deenergized at 151°F to maintain BIT temperature in the proper band. Either temperature controller can actuate the BIT HIGH/LOW TEMPERATURE alarm (window 1J-A7). The alarm setpoints are listed in Table 52-6.

BIT header pressure indication has been disabled as per EWR 90-027A. Pressure transmitter 1-SI-PT-1934 was isolated and removed, and its electrical output to the main control room indicator and alarm was disconnected.

BIT recirculation flow is sensed on the BIT outlet to the boric acid tank by FT-1934. Indication is provided locally, near the boron injection tank.

Valve Instrumentation and Controls

Accumulator isolation valves. The accumulator isolation valves MOV-1865A, B, and C are controlled by OPEN/CLOSE pushbuttons on Control Room vertical board Safeguards Panels. Open and shut lights on the pushbutton indicate the valve position. Each valve opens automatically when RCS pressure increases above 2010 psig, or on receipt of an SI signal. The valve can only be shut by using the CLOSE pushbutton and only if the SI signal is not present and RCS pressure is less than 2000 psig. Valve monitor #3 displays a red light when ALL of the isolation valves are fully open. An ACCUMULATOR OUTLET VALVE CLOSED alarm (window 1J-D6) is also actuated if any of the valves are not fully open.

Accumulator test line isolation valves. The accumulator test line isolation valves HCV-1850A, B, C, D, E, and F are controlled by OPEN/CLOSE pushbuttons on Control Room vertical board Safeguards Panels. Open and shut lights on the pushbutton indicate the valve

The HHSI pump suction valves MOV-1863A and B are controlled by OPEN/CLOSE pushbuttons on Control Room vertical board Safeguards Panels. These valves are normally shut valves from the LHSI pump discharge to the HHSI pump suction and open automatically on a low-low RWST level of ~~24.9 percent~~ 19.4% (24.9% for Unit 2) when RWST in BYPASS MODE (1J-C2) annunciates if the safety injection recirc. mode signal is present.

<u>HHSI pump discharge valves</u>. The HHSI pump recirculation isolation valve MOV-1373 is controlled by a CLOSE/OPEN handswitch on Control Room benchboard 1-1. The switch spring returns to the center position. This valve has no automatic operation and is positioned based on the need for HHSI pump recirculation.

The HHSI pump discharge charging line stop valves MOV-1289A and B are controlled by CLOSE/OPEN handswitches on Control Room Benchboard 1-1. These valves shut automatically on receipt of a SI signal and have blocking contacts that prevent valve opening when a SI signal is present.

The HHSI pump discharge to RCS hot leg isolation valves MOV-1869A and B are each controlled by a pushbutton handswitch and a keyswitch on Control Room vertical board Safeguards Panels. Each handswitch has two positions, ON and OFF, that energize or deenergize the valve control circuits. Each keyswitch has two positions, OPEN and CLOSE, to allow operation of its respective valve. Keyswitches are used to prevent inadvertent operation of the valve, as these valves allow HHSI pump discharge to go to the RCS hot legs. Valve monitors 2 and 1 are energized if MOV-1869A and B, respectively, are not fully shut.

The HHSI pump discharge to RCS cold leg isolation valve MOV-1836 is controlled by an ON/OFF pushbutton and an OPEN/CLOSE handswitch on Control Room vertical board Safeguards Panels, Train B. The pushbutton handswitch energizes or deenergizes the valve control circuit. The OPEN/CLOSE handswitch allows operation of the valve. Two handswitches are used to prevent inadvertent operation of the valve, as this valve allows the HHSI pump discharge to bypass the BIT. Valve monitor 1 is energized if MOV-1836 is not fully shut.

The remaining motor-operated valves associated with the HHSI pump suction, discharge and recirculation are controlled by CLOSE/OPEN handswitches on Control Room vertical board 1-1. Further information may be found in the CVCS module (NCRODP-41).

BIT isolation valves. The BIT isolation valves MOV-1867A, B, C, and D are controlled by OPEN/CLOSE pushbuttons on Control Room vertical panel Safeguards Panels. Green and red lights on the pushbuttons indicate the valve positions. These valves are normally shut but open on receipt of a SI signal or upon depression of the respective OPEN pushbutton. The valves may be shut by depressing the CLOSE pushbutton as long as the SI signal is not present.

BIT recirculation isolation valves. The BIT recirculation to the boric acid tank isolation valves TV-1884A, B, and C are controlled by OPEN/CLOSE pushbuttons on Control Room vertical board Safeguards Panels. These valves are open during normal operation and shut on receipt of an SI signal (Train A to TV-1884A & C and Train B to TV-1884B). The isolation valves operate identically to the accumulator vent line trip valve discussed above.

LHSI pump suction valves. The LHSI pump suction valves MOV-1860A and B from the containment sump are controlled by OPEN/CLOSE pushbuttons on Control Room vertical board Safeguards Panels. These valves are normally shut and open only if at least one of the recirculation valves for the same LHSI pump is shut. If this condition is met, the valves open on receipt of a low-low RWST level of ~~24.9~~ percent if the SI recirc. mode signal is present or open if the OPEN pushbutton is depressed. 19.4% (24.9% for Unit 2).

The LHSI pump suction valves MOV-1862A and B from the RWST are controlled by OPEN/CLOSE pushbuttons on Control Room vertical board Safeguards Panels. Green and red lights on the handswitches indicate the valve positions. MOV-1862A and B are electrically interlocked with MOV-1860A and B respectively. MOV-1862A and B cannot be opened unless MOV-1860A or B is fully shut. Also, once MOV-1860A or B has fully opened, MOV-1862A or B automatically shuts.

LHSI pump discharge valves. The LHSI pump discharge valves MOV-1863A and B to the HHSI pump suction were covered under the paragraph on HHSI pump suction valves, above.

The LHSI pump discharge valves MOV-1864A/B and -1890C/D to the RCS cold legs are controlled by OPEN/CLOSE pushbuttons on Control Room vertical board Safeguards Panels. Green and red lights on the handswitches indicate the valve positions. These valves are normally open and have no special control functions associated with them. The breakers for the motor operators on MOV-1890C and D are locked open during normal plant operations.

The LHSI pump discharge valves MOV-1890A and B to the RCS hot legs are each controlled by a pushbutton handswitch and a keyswitch on Control Room vertical board Safeguards Panels. Each handswitch has two positions, ON and OFF, that energize or deenergize the valve control circuits. Each keyswitch has two positions, OPEN and CLOSE, to allow operation of its respective valve. Keyswitches are used to prevent inadvertent operation of these valves, as these valves allow LHSI pump discharge to go to the RCS hot legs. Valve Monitor Light No. 1 is energized when MOV-1836, 1869B, and 1890B are fully closed. Valve Monitor Light No. 2 is energized when MOV-1869A and MOV-1890A are fully closed.

The LHSI pump recirculation valves MOV-1885A, B, C, and D are controlled by OPEN/CLOSE pushbuttons on Control Room vertical board Safeguards Panels. These valves are normally open but shut on receipt of a low-low RWST level of ~~24.9 percent~~ if a SI recirc. mode signal is present and MOV-1863A or B respectively has opened. 19.4% (24.9% for Unit 2) if the SI signal has been reset, the SI recirc. mode signal will remain locked-in. The automatic switchover performs the following:

1. Opens the HHSI pump suction valves from LHSI pumps.

2. Closes LHSI pump recirculation isolation valves.

3. Opens LHSI pump suction valves from the sump, and

4. Closes the LHSI pump suction valves from the RWST.

Upon completion of this lineup, water collected in the containment sump is returned to the RCS by the low head or low head/high head recirculation flow paths. Cooling of the sump water is provided by the RS System. The switchover is performed so that sufficient level remains in the RWST for proper operation of the QS System. Once the automatic switchover has occurred, the operator must reset the SI signal and depress the appropriate SI recirculation mode reset pushbuttons on Control Room vertical board 1-3 before the valves operated by the switchover can be repositioned manually.

Manual switchover is performed after the RWST level ~~drops below 29 percent~~ reaches 23% (29% for unit 2) using ES-1.3 (Transfer to cold leg recirc.). The operator resets SI, then verifies two HHSI and two LHSI pumps are operating and closes the charging seal injection to the reactor coolant pumps (isolation valve MOV-1370). The operator performs the same steps as automatic switchover, above, and then shuts the HHSI pump suction valves from the RWST, MOV-1115B and D.

Long-term passive failure protection. To establish the long-term passive failure protection, the high head flow path is separated using E-1 (Loss of reactor or secondary coolant). The lineup involves the following:

1. Open the alternate charging header to cold legs valve MOV-1836.

2. Close one of the two valves in the discharge of the HHSI pumps, so that only one pump discharges through the BIT to the cold legs.

3. Close one of the two valves in the discharge of the HHSI pumps, so that only one pump discharges through MOV-1836 to the cold legs.

ANSWER KEYS TO REVIEW EXERCISES
(Continued)

Instrumentation and Controls

Review Exercise - Part I

1. minimize loads on the emergency buses.

2. LHSI PUMP LOCKOUT OR OVERLOAD TRIP

3. 15 seconds

4. Technical Specifications

5. 28.3 percent (Unit 2) & 22.8% (Unit 1)

6. locally

7. on receipt of a low-low RWST level signal if a SI recirc. mode signal is present

8. a handswitch and a keyswitch

9. on Control Room benchboard 1-1

10. low-low RWST level

Review Exercise - Part II

11. b

12. b

13. b

TABLE 52-6
ALARMS

Window	Alarm	Setpoint
1J-D7	Acc high level	71%
1J-D7	Acc low level	55%
1J-D8	Acc high pressure	>667 psig
1J-D8	Acc low pressure	<559 psig
1J-D6	Acc outlet valve closed	NOTE 1
1J-A5	LHSI pump 1A lock out or overload trip	NOTE 2
1J-F5	LHSI pump 1B lock out or overload trip	NOTE 2
1J-A2	RWST low level	21.5% ~~28.5%~~ (28.3% unit 2)
1J-A7	BIT high temperature	>160°F
1J-A7	BIT low temperature	<136°F
1K-E1	RWST high level	>98.5%
1K-E2	RWST level below normal	<96.7%
1G-G5	Hydrostatic test pump stuffing box reservoir low level	NOTE 3
1C-HB	HHSI pump discharge header valves not fully closed	NOTE 4
1J-C2	RWST in Bypass Mode	NOTE 5

NOTE 1: MOV-1865A, B, and/or C not fully open

NOTE 2: Handswitch in PULL-TO-LOCK or breaker tripped on overload

NOTE 3: 1/2 gallon below normal level

NOTE 4: MOV-1869A, B, and/or MOV-1836 not fully shut

NOTE 5: RWST level channel in "Test"

NUCLEAR CONTROL ROOM

OPERATOR DEVELOPMENT PROGRAM

NORTH ANNA POWER STATION

MODULE NCRODP-53

QUENCH SPRAY SYSTEM

Virginia Power
Nuclear Training

refrigeration units and recirculating pumps then operate automatically to maintain RWST temperature between 40° and 43°F.

QS System Interrelationships

<u>Recirculation Spray Systems</u>. The RS System is similar to the QS System in that it provides spray to the containment atmosphere during a LOCA or SLB to obtain and maintain a subatmospheric pressure. The RS System recirculates water from the containment sump through recirculation spray coolers to RS System spray headers. The QS System provides spray from immediately after the LOCA or SLB until the RWST empties. The RS System provides for long-term Containment depressurization during and after the QS System has been secured. The QS System also provides cool water to the suction of the inside RS pumps to increase the available NPSH to the pumps. Module NCRODP-54 provides a detailed description of the Recirculation Spray System.

<u>Safety Injection System</u>. The RWST provides water to both the QS and SI Systems immediately after a LOCA or SLB. The SI injects water from the RWST into the Reactor Coolant System (RCS) until the RWST level drops to 22.8% (28.3% for Unit 2). ~~28.1 percent~~. The SI System is then manually realigned to use water collected in the containment sump as a continuing source of water. If the SI system manual realignment is not accomplished, an auto swap-over will occur when RWST level drops to 19.4% (24.9% for Unit 2). ~~21.6 percent~~. Water from the RWST is also provided to the hydrostatic test pump for refilling the SI accumulators. A connection on the RWST recirculation line receives low head safety injection pump recirculation and water from the SI accumulator test line. Module NCRODP-52 provides a detailed description of the Safety Injection System.

<u>Chilled Water System</u>. The Chilled Water System provides cooling water to the RWST coolers. These coolers are used to initially remove heat from the RWST water. Module NCRODP-51 provides a detailed description of the Chilled Water System.

<u>Heat Tracing System</u>. Heat tracing is provided on all of the external piping used between the CAT, RWST, and QS pump suctions. It is also provided on the RWST cooling piping. The Heat Tracing System prevents the water in the pipes from freezing or chemicals from coming out of solution during cold weather. Module NCRODP-22 provides a detailed description of the Heat Tracing System.

supplies and locations are detailed in Table 53-2 and 53-3. Relief valve setpoints are listed in Table 53-4.

QS System Major Components

The QS System consists of a refueling water storage tank (RWST), two QS pumps, two spray rings, two RWST recirculation pumps, two RWST coolers, two RWST refrigeration units, and a CAT recirculation pump. The primary purpose of the QS System is to return the containment atmosphere to subatmospheric conditions within the required time following a LOCA or SLB. This subsection describes the major QS System components and the flow paths used to achieve the purpose. A piping diagram of the QS System is presented as Figures 53-2 and 53-3.

Refueling water storage tank. There is one RWST, located in the Yard to the east of Containment. The RWST provides

1. water to the SI System and the QS System,

2. backup water to the operating charging pump when the CVCS VCT level drops below its low level setpoint, and

3. water for the refueling cavity.

The RWST is a vertical, cylindrical (62 feet high, 38 feet in diameter) tank with a usable capacity of 488,000 gallons. Technical Specifications require that the RWST contain between 466,200 and 487,000 gallons of ~~2000~~ 2300 to ~~2100~~ 2400 ppm borated water during normal plant operations. The proper boron concentration is maintained by CVCS. During RWST makeup, the boron concentration in the makeup from the boric acid blender is adjusted to ensure that the RWST chemistry is within the requirements of the Technical Specifications. The RWST is vented to the atmosphere through a vent pipe at the top of the tank. The tank overflows to the Yard if the tank is overfilled.

The water in the RWST is initially cooled by the RWST coolers during initial filling of the RWST. The RWST refrigeration units maintain the RWST water between 40° and 43°F. This temperature range ensures that sufficient cooling capacity is available for the QS and RS based on RWST temperature. When the RWST temperature (TE-QS100A/B) increases above 43°F, the pump starts. It stops when RWST temperature drops below 40°F. Auxiliary contacts in the slow speed breakers allow operation of the RWST refrigeration units when either RWST recirculation pump is running in slow speed.

<u>CAT recirculation pump</u>. The CAT recirculation pump is controlled by a START and STOP pushbutton on the Safeguards Panel, Train A. Green and red indication lights are provided internal to the control switch. During normal plant operations the CAT recirculation pump is not running and is isolated.

Tank Instrumentation and Controls

<u>Refueling Water Storage Tank</u>. The RWST is provided with four channels of level indication on the Safeguards Panels. Level transmitters LT-QS100A, B, C, and D (see Figure 53-2) provide signals to level indicators LI-QS100A, B, C, and D. The level indication range for the RWST is 16,970 gallons at 0 percent level to 487,000 gallons at 100 percent level. Level transmitters LT-QS100A and B also provide a signal to a RWST LOW LEVEL alarm (window 1J-A2). The alarm setpoint is listed in Table 53-6. When two of the four channels have sensed a low-low RWST level condition at ~~24.9 percent~~ 19.4% (24.9% for unit 2), a signal is generated to realign safety injection to the recirculation mode automatically (see module NCRODP-52 for further details on the Safety Injection System). Placing any of the four RWST channels in test or any channel sensing a low-low RWST level actuates a common RWST IN BYPASS MODE alarm (window 1J-C2) and an individual RWST IN BYPASS MODE CHANNEL I, II, III, or IV alarm (window 1N-E1, F1, G1, or H1 respectively).

Level transmitter LT-QS103 provides input to an RWST HIGH LEVEL alarm (window 1K-E1) and an RWST LEVEL BELOW NORMAL alarm (window 1K-E2). These alarms inform the operator if the RWST level is not within the proper range as required by technical specifications. The alarm setpoints are listed in Table 53-6.

RWST temperature indication is provided on the Safeguards Panels. Temperature element TE-QS100A and B provide signals to temperature indicators TI-QS100A and B, a common RWST HIGH TEMPERATURE alarm (window 1J-A1), and RWST recirculation pump control circuitry. RWST temperature is indicated on 0 to 150°F temperature gages. The alarm setpoint is listed in Table 53-6. Each temperature element sends a control signal to its respective RWST recirculation pump control circuit. When a temperature element senses

The scenario which produces the "zero level" period is not the bounding scenario for MSLB depressurization. A complete loss of one entire train (one QS pump, one IRS pump and one ORS pump) is the bounding scenario.

Steam line break. A main SLB in the Containment causes containment pressure to increase rapidly until QS spray is initiated. Peak containment pressure of 45 psig (worst case), which is below Containment design pressure, occurs within the first 60 seconds of the accident and is coincident to the time that the steam generator empties. Containment temperature, which peaks at ~~430°F~~ 357°F, does exceed the containment design temperature limit of 280°F but is quickly reduced below the design temperature within 100 seconds when QS is initiated.

QS System Component Design Bases

RWST design basis. The RWST contains NTCT 91-0065 [466,200 to 487,000] gallons of ~~2000~~ 2300 to ~~2100~~ 2400 ppm borated water. The RWST is considered to be a passive component because it does not require any power source or operator action to perform its function. Only one RWST is required because it is a passive component and because it is used to supply water only during the short term following a CDA. During a LOCA, RWST water is sprayed into Containment by the QS System and collects in the containment sump. The SI System injects RWST water into the RCS which is losing primary coolant to the containment sump through the rupture. The water level of the containment sump provides NPSH to the RS pumps and the low head safety injection (LHSI) pumps. Approximately 3 minutes after the CDA signal, the RS pumps start. They use the water in the containment sump as their source of water. When RWST level drops to ~~21.6 percent~~ 19.4% (24.9% for Unit 2), the SI System switches over to its recirculation lineup where the LHSI pumps also use the containment sumps as their source of water. The QS pumps continue to use the RWST water until the tank empties and the QS pumps start to cavitate. The RWST is designed to provide a sufficient volume of water until the RS pumps and LHSI pumps can operate using the containment sump as a source of water. Further design criteria may be found in Table 53-1.

The water in the RWST is maintained between 40° and 43°F. This low temperature ensures adequate cooling capacity is available for the QS System to depressurize Containment within the required 60 minutes. The lower temperature limit of 40°F for the RWST is based on maintaining the structural integrity of Containment. Containment pressure can drop as low as 7.7 psia if QS is initiated during normal plant operations with the following starting conditions:

3. provide the Emergency Core Cooling System (ECCS) with water for effective core cooling on a long-term basis after a LOCA.

The RS System uses coolers to transfer heat from the water in the containment sump to 23
the Service Water (SW) System. After the water level in the RWST drops below 18 percent, the Safety Injection (SI) System also uses the water in the containment sump as its source of water for recirculation phase core cooling.

RS System Flow Paths

The RS System is composed of two inside and outside RS Subsystems and a Casing Cooling Subsystem. Each of the RS Subsystems consists of an RS pump, a recirculation cooler, and an RS spray ring. Two of the RS Subsystems use RS pumps located within Containment. The other two RS Subsystems use RS pumps located outside Containment. The Casing Cooling Subsystem consists of a casing cooling tank, two casing cooling recirculation pumps, two refrigeration units, and two casing cooling pumps. The Casing Cooling Subsystem provides cool water to the outside RS pumps to increase their net positive suction head (NPSH). During normal plant operations, the RS System is in standby, lined up for automatic initiation as follows:

1. Casing cooling tank is maintained with greater than 110,000 gallons of 35° to 50°F borated water.

2. Casing cooling pumps are in standby with their discharge valves shut.

3. RS pumps are in standby with the outside RS pump suction and discharge valves open.

4. Recirculation coolers are drained with their inlet and outlet service water valves shut.

RS System Interrelationships

Quench Spray System. The QS System is similar to the RS System in that it provides spray to the containment atmosphere during a LOCA or SLB to regain a subatmospheric pressure. The QS System provides water from the RWST until the RWST empties; the QS System is then secured. Sodium hydroxide (NaOH) is added to the QS water to remove radioactive iodine from the containment atmosphere and to provide pH control of the sump water. The QS System also provides 150 gpm to each inside RS pump to increase the pumps' NPSH. Module NCRODP-53 provides a detailed description of the Quench Spray System.

Safety Injection System. The RWST, which is part of the QS System, also provides water to the SI System immediately after a LOCA or SLB. The SI System injects water from the RWST into the Reactor Coolant System (RCS) until the RWST level drops to less than 18 23 percent. The SI System is then realigned manually or automatically to use water collected in the containment sump as a continuing source of water. The RS System cools the water in the containment sump by transferring the heat to the Service Water System. The Unit 1 outside RS pumps and low head safety injection (LHSI) pump discharges can be cross-connected to provide a long term backup for the low head safety injection pumps. Module NCRODP-52 provides a detailed description of the Safety Injection System.

Reactor Protection System. The Reactor Protection System's (RPS) Engineered Safety Features (ESF) Actuation Subsystem generates the CDA initiation signal for the RS System. Module NCRODP-77 provides a detailed description of the ESF Actuation Subsystem.

Service Water System. The SW System provides the cooling water for the recirculation coolers. The normally shut inlet and outlet header isolation valves open automatically on receipt of a CDA signal. Module NCRODP-13 provides a detailed description of the Service Water System.

The Unit 2 steam generator accelerometers are located on the 0.5 - inch jacking screws of a special mounting plate on the inspection port blind flange. The Unit 1 steam generator accelerometers are located on special mounting bolts adjacent to the secondary handholes.

Sensor installation location. The accelerometers are located in the upper vessel region, on the lower guide tubes, and near the inlets to the steam generators. The 2-1 designation in the following list includes both the active and passive sensors that are designated as A and B. In other words, 2-1 includes channel 2-1A (active) and 2-1B (passive). Specifically, the sensors are in the following locations:

1. Lower vessel (channels 1-1 and 2-1). Two accelerometers (active, passive) on each unit are located in the lower vessel region, and are positioned on the in-core instrument guide tube as close to the reactor vessel as practical for inspection and replacement. The accelerometers are located not more than 3 feet from the vessel bottom, inside the primary shield wall as close to the insulation as possible on in-core instrument conduits No. 2 and No. 5. The conduit size is 1.5 inch. Mounting blocks are provided for installation.

2. Upper vessel (channels 1-2 and 2-2). Accelerometers are located in the upper vessel region to qualitatively monitor the control rod extension and vessel internals for vibration. The accelerometers are located on a special (Rockwell-provided) stud in the tension gage hole. The sensors are adjacent, with approximately 4 inches square required for each. Sensors are mounted in an Atomics International "J-box" bolted directly to the special mounting stud. Flexible cable and conduit runs up the control rod drive service structure. Conduit and cable are installed to accommodate removal from the reactor head for refueling by disconnecting at the preamp J-box.

3. Steam generator inlets (Channels 1-3 through 1-5 and 2-3 through 2-5) on steam generators A, B, and C. ~~The accelerometers are located on the 0.5-inch jacking screws of a special mounting plate on the inspection port blind flange.~~ Two accelerometers are used on each steam generator. J-boxes 4 inches by 4 inches are used. Tapped mounting studs were provided by Atomics International.

4. Auxiliary channels (channel 1-6 through 1-8 and 2-6 through 2-8.) A wired-in auxiliary channel and five spare slots are included.

OPERATIONAL SPARE PARTS LIST

NORTH ANNA UNIT 1 SGRP
OPERATIONAL SPARE PARTS LIST

	Name of Part or Assembly	Seller's Dwg. No.	Seller's Item No.	Operating Spares Qty
1.	Inspection Port Wrapper Closure			
	a. Inner Plate Assembly	W 9736D03	G01	1
	b. Outer Plate Assembly	W 3414C70	G01	1
	c. Lock Washer	W 3416C75	H01	12
	d. S.S. Regular Hex Nut (0.750-10UNC-2B)	W 70210GQ	61E	1
2.	Inspection Port Closure			
	a. Gasket	W 3413C98	H07	12
	b. Cover	W 9739D68	H01	0
	c. Washer	W 3416C33	H02	6
	d. Bolt	W 9739D10	H01	6
3.	Upper Handhole Wrapper Closure			
	a. Inner Plate Assembly	W 6138E99	G01	1
	b. Outer Plate Assembly	W 6139E03	G01	1
	c. Dual Lock Washer	W 3414C71	H01	36
	d. S.S. Regular Hex Nut (0.500-13UNC-2B)	W 70210GQ	61L	2
4.	Handhole Closure			
	a. Gasket	W 3413C98	H06	36
	b. Cover	W 9739D69	H01	0
	c. Washer	W 3416C33	H02	18
	d. Bolt	W 9739D10	H01	18
5.	Primary Manway Closure			
	a. Gasket	W 3413C98	H04	12
	b. Insert	W 9739D12	H01	2
	c. Insert Screw	W 9739D79	H01	18
	d. Diaphragm	W 9740D10	H01	2
	e. Cover	W 9739D18	H01	0
	f. Stud			6
	g. Stud Nut			6
6.	Thread Lubricant Fel Pro N-2000 Hi-Purity Compound (1 lb. Can)			As Req'd.

FUNCTIONAL TESTING REQUIREMENTS

DCP 90-13-1
Functional Testing Requirements
and Acceptance Criteria

TEST	COMPONENT/SYSTEM	PROCEDURE OR ACCEPTANCE CRITERIA
Installation Testing		
Rotation Check	RCP - 1B	Correct Rotation
Electrical Insulation Checks	Breaker 15B3 Control Circuit	0-NAT-E-002
Control Circuit Continuity Check	Breaker 15B3 Control Circuit	AT-TEP-RT-9
Meggar Test Procedure	Breaker 15B3 Control Circuit	AT-TEP-RT-5
Current Loading Procedure	Breaker 15B3 Control Circuit	AT-TEP-RT-4
Relay Test Procedure	Breaker 15B3 Control Circuit	AT-TEP-RT-2
Operational Test Procedure	Breaker 15B3 Control Circuit	AT-TEP-RT-3
In-Service Current Readings	Breaker 15B3 Control Circuit	AT-TEP-RT-6
Secondary Hydrostatic Test	Main Steam: Lines 32"-SHP-1-601-Q2, 32"-SHP-2-601-Q2, 32"-SHP-3-601-Q2	WCAP 13453 (App. 4-35), Tech. Spec. 3/4.7.2 and ASME Section XI
	Feedwater: Lines 16"-WFPD-22-601C-Q2, 16"-WFPD-23-601C-Q2, 16"-WFPD-24-601C-Q2	WCAP 13453 (App. 4-35), Tech. Spec. 3/4.7.2 and ASME Section XI
	Chemical Feed: Lines 3/4"-CFPD-1-ICN9-Q2, 3/4"-CFPD-2-ICN9-Q2, 3/4"-CFPD-3-ICN9-Q2, from 16" feedwater line to the inboard containment isolation valve	WCAP 13453 (App. 4-35), Tech. Spec. 3/4.7.2 and ASME Section XI
	Level Instrumentation Piping: (flush instmt. lines during fill for hydro.): 1-FW-LT-1474 through 1477, 1-FW-LT-1484 through 1487 and 1-FW-LT-1494 through 1497	WCAP 13453 (App. 4-35), Tech. Spec. 3/4.7.2 and ASME Section XI
	Blowdown and Shell Drain Piping: Lines 1"-WGCB-6-601C-Q3-, 1"-WGCB-9-601C-Q3-, 1"-WGCB-12-601C-Q3-, 2-1/2"-WGCB-4-601C-Q3, 2-1/2"-WGCB-5-601C-Q3, 2-1/2"-WGCB-7-601C-Q3, 2-1/2"-WGCB-8-601C-Q3, 2-1/2"-WGCB-10-60K-Q3	WCAP 13453 (App. 4-35), Tech. Spec. 3/4.7.2 and ASME Section XI

DCP 90-13-1
Functional Testing Requirements
and Acceptance Criteria

TEST	COMPONENT/SYSTEM		PROCEDURE OR ACCEPTANCE CRITERIA
	Installation Testing (Con't)		
Secondary Hydrostatic Test (Cont.)	Sample Piping: Lines 2"-SS-220-601-Q3, 2"-SS-225-601-Q3, 2"-SS-227-601-Q3, 1"-SS-221-601-Q3, 1"-SS-226-601-Q3, 1"-SS-228-601-Q3 , from SG nozzle to first isolation valve;		WCAP 13453 (App. 4-35), Tech. Spec. 3/4.7.2 and ASME Section XI
	Wet Layup Piping: Lines 2"-SGD-4-601-Q2, 2"-SGD-6-601-Q2, 2"-SGD-8-601-Q2WCAP 13453 (App. 4-35) and Tech. Spec. 3/4.7.2 and ASME Section XI		WCAP 13453 (App. 4-35), Tech. Spec. 3/4.7.2 and ASME Section XI
	Steam Dome and Lower Assembly Joined by Girth Weld and Tube to Tubesheet Location: SGs 1-RC-E-1A-, 1B, 1C		WCAP 13453 (App. 4-35), Tech. Spec. 3/4.7.2 and ASME Section XI
Leak Test	Containment Purge System Flange		Proper Seal/No Leaks
	Pre-Operational Testing		
Calibration	S/G Level:	S/G A Narrow Range Channel I	1-ICP-FW-L-1474
		S/G A Narrow Range Channel II	1-ICP-FW-L-1475
		S/G A Narrow Range Channel III	1-ICP-FW-L-1476
		S/G A Wide Range	1-ICP-FW-L-1477
		S/G B Narrow Range Channel I	1-ICP-FW-L-1484
		S/G B Narrow Range Channel II	1-ICP-FW-L-1485
		S/G B Narrow Range Channel III	1-ICP-FW-L-1486
		S/G B Wide Range	1-ICP-FW-L-1487
		S/G C Narrow Range Channel I	1-ICP-FW-L-1494
		S/G C Narrow Range Channel II	1-ICP-FW-L-1495
		S/G C Narrow Range Channel III	1-ICP-FW-L-1496
		S/G C Wide Range	1-ICP-FW-L-1497
Instrumentation Calibration, Rescaling and Loop Checks	Loose Parts Monitoring:	VLPM Accelerometer Removal and Replacement.	IMP-C-MISC-06
	Steam Dump Control:	T-408 Steam Dump Control	ICP-P-1-T-408
	Tave and delta T loops:	Tavg Steam Dumps, Prz Level Power Mismatch and Rod Speed Control	ICP-P-1-T-403B

DCP 90-13-1
Functional Testing Requirements
and Acceptance Criteria

TEST	COMPONENT/SYSTEM		PROCEDURE OR ACCEPTANCE CRITERIA
	Pre-Operational Testing (Con't)		
Instrumentation Calibration, Rescaling and Loop Checks (Cont.)	Turbine Impulse Pressure:	1st Stage Pressure Channel III	ICP-MS-1-P-1446A
		1st Stage Pressure Channel IV	ICP-P-1-P-447
		1st Stage Pressure Control	ICP-P-1-T-446B
	Rod Speed and Direction: Calculation	ΔT/Tavg Rod Insertion Limit	ICP-P-1-T-409A
	Power Range Protection:	Power Range Protection Channel I	ICP-NI-1-N-41
		Power Range Protection Channel II	ICP-NI-1-N-42
		Power Range Protection Channel III	ICP-NI-1-N-43
		Power Range Protection Channel IV	ICP-NI-1-N-44
	RWST Level:	RWST Level Channel III	1-ICP-QS-L-100A
		RWST Level Channel IV	1-ICP-QS-L-100B
		RWST Level Channel I	1-ICP-QS-L-100C
		RWST Level Channgel II	1-ICP-QS-L-100D
	Steam Flow/Feed Flow:	S/G A FW Flow Prot Channel IV	1-ICP-MS-F-1475
		S/G A Steam Flow Channel IV	(Combined SF&FF)
		S/G A FW Flow Prot Channel III	1-ICP-MS-F-1474
		S/G A Steam Flow Channel III	(Combined SF&FF)
		S/G B FW Flow Prot Channel IV	1-ICP-MS-F-1485
		S/G B Steam Flow Channel IV	(Combined SF&FF)
		S/G B FW Flow Prot Channel III	1-ICP-MS-F-1484
		S/G B Steam Flow Channel III	(Combined SF&FF)
		S/G C FW Flow Prot Channel IV	1-ICP-MS-F-1495
		S/G C Steam Flow Channel IV	(Combined SF&FF)
		S/G C FW Flow Prot Channel III	1-ICP-MS-F-1494
		S/G C Steam Flow Channel III	(Combined SF&FF)
	Feed Flow:	FW Control Loop 1	ICP-P-1-F-1478
		FW Control Loop 2	ICP-P-1-F-1488
		FW Control Loop 3	ICP-P-1-F-1498

DCP 90-13-1
Functional Testing Requirements
and Acceptance Criteria

TEST	COMPONENT/SYSTEM		PROCEDURE OR ACCEPTANCE CRITERIA
	Pre-Operational Testing (Con't)		
Instrumentation Calibration, Rescaling and Loop Checks (Cont.)	Steam Pressure:	Steam Line A Steam Pressure Prot Channel II	1-ICP-MS-P-1474
		Steam Line A Steam Pressure Prot Channel III	1-ICP-MS-P-1475
		Steam Line A Steam Pressure Prot Channel IV	1-ICP-MS-P-1476
		Steam Line B Steam Pressure Prot Channel II	1-ICP-MS-P-1484
		Steam Line B Steam Pressure Prot Channel III	1-ICP-MS-P-1485
		Steam Line B Steam Pressure Prot Channel IV	1-ICP-MS-P-1486
		Steam Line C Steam Pressure Prot Channel II	1-ICP-MS-P-1494
		Steam Line C Steam Pressure Prot Channel III	1-ICP-MS-P-1495
		Steam Line C Steam Pressure Prot Channel IV	1-ICP-MS-P-1496
		Functional Check of Main Cond-nsor Steam Dump System	1-IMP-MS-T-408
Primary Hydrostatic Test	RCS: Lines 31"-RC-2-2501R-Q1, 31"-RC-5-2501R-Q1, 31"-RC-8-2501R-Q1, 29"-RC-1-2501R-Q1, 29"-RC-4-2501R-Q1, 29"-RC-7-2501R-Q1		1-PT-171.2
Flow Measurement	Reactor Coolant System		1-PT-27
	Post-Operational Testing		
SG Moisture Carry-over and Flow Calibration Test	Integrated Systems		1-ST-102
Calorimetric (Steam Output)	Integrated Systems		1-PT-24.1

REACTIVITY MANAGEMENT PROGRAM REVIEW

Reactivity Management Program Review

Per STD-GN-0001, a reactivity management review is required to be performed for all DCP's. Standard STD-GN-0001 identifies screening questions which must be addressed for each Design Change. For DC 90-13-1, the following screening questions were determined to be applicable.

Screening Questions

1. Will the DCP impact any of the following:

   D. Reactor Coolant System (RCS) T-avg and /or the instruments/equipment measuring T-avg?

   F. Control rod system and insertion limits?

   H. Safety analyses?

   I. RCS dilution rates?

   L. RCS Flow and/or the instruments/equipment measuring flow?

   M. Main Steam Flow and/or the instruments/equipment measuring flow?

   N. Calorimetric Power Monitoring?

The responses to these questions are included in the attached Memorandum, R. G. McAndrew to S. L. Wilkie, dated September 10, 1992 as well as in the supplemental sheet attached to this Appendix.

VIRGINIA POWER
NORTH CAROLINA POWER

Memorandum

To S.L.Wilkie

NORTH ANNA POWER STATION

From R.G. McAndrew

Sept. 10, 1992

REACTIVITY MANAGEMENT PROGRAM REVIEW FOR SGRP UNIT 1

Reference 1. Memorandum, S.L. Wilkie to R.G. McAndrew of Aug 14, 1992, same subject.

Reference 1 identified reactivity management screening questions from STD-GN-0001 marked with a "Yes" which require a response from reactor engineering. I have prepared responses to these screening questions for the reactivity management program portion of the DCP, see attached. Please contact me if there are any further questions.

R.G. McAndrew

R.G. McAndrew

cc:
R.M. Berryman IN-3SW
D. Dziadosz IN-3SW
C.B. LaRoe IN-3SW
K.L. Basehore IN-3SW
D.A. Heacock
C.T. Snow RGM for CTS
A.P. Main
R.M. Garver

RESPONSES TO REACTIVITY MANAGEMENT SCREENING QUESTIONS WITH "YES" RESPONSES FROM PROJECT ENGINEERING FOR SGRP

SCREENING QUESTION	RESPONSE
D.	Tave will be changed from 583.0° F to 580.8° F as part of this DCP. These changes have been included in the Nuclear Core Design & Licensing checklist used as design input for reactivity related analyses for North Anna 1 cycle 10. The impact on power range NI calibration and intermediate range NI trip setpoints will be automatically included in the design process established by NA&F.
F.	The Tave rod control program will be changed as part of this DCP. The reactivity impact of the rod control program change will be accounted for by NA&F as part of the reload design and analysis process for North Anna 1 cycle 10. Rod insertion limits will not be affected by this DCP.
H.	The replacement of the steam generators, including the Tave change and return to 100% power, is included in the Nuclear Core Design & licensing checklist which is used as design input for North Anna cycle 10 safety analyses. The Reload Safety Evaluation Document and Technical Report NE-883 Rev 1 (appendix 4-23 of the DCP) prepared by NA&F will address any safety analyses impacted by the SGRP.
I.	The impact of the increased RCS volume on events driven by RCS dilution rate was evaluated in Tech Report NE-883.
L.	RCS flowrate is expected to increase significantly after SGRP. The impact of the higher flowrate on reactivity was evaluated in Technical Report NE-883 and an RCS flowrate measurement will be performed in accordance with reload startup procedures. The flow increase has been evaluated and is within the capabilities of existing instrumentation to measure. RCS flowrate instrumentation will be rescaled prior to startup using the best available flowrate information.
M.	After SGR, main steam flow is expected to be within the capability of current instrumentation to measure. The increased main steam flow will impact the steam flow-based heat balance program, see item N. The

reactivity impact of the [illegible]ased main steam flow was addressed in Technical Report NE-883.

N. The steam flow-based heat balance will be affected by the SGRP. The steam flow and temperature changes are expected to be within the capability of the steam flow heat balance program to handle. Effects of RCS Tave changes on neutron monitoring and setpoints will be accounted for during the NA&F reload design and setpoint implementation process. The impact of the new SG insulation material on the heat balance program needs to be evaluated. The DCP indicates the new material is at least as effective as the existing material. The value for insulation losses is lumped with RCP heat input in the heat balance program and the current value was provided by Westinghouse. This concern has been raised as a comment in the 70% design review, and is currently being addressed by project engineering.

o The effect of insulation has been evaluated with regard to core power (heat balance) calculations. The net change due to improved insulation on the replacement steam generators is negligible.

SPECIFICATION 21809-M-001(Q), DECONTAMINATION SERVICES

TECHNICAL SPECIFICATION

FOR

DECONTAMINATION SERVICES

FOR

VIRGINIA ELECTRIC AND POWER COMPANY

NORTH ANNA POWER STATION UNIT 1

STEAM GENERATOR REPAIR PROJECT

BECHTEL
GAITHERSBURG, MARYLAND

NO.	DATE	REVISION	BY	CHKD	SUPV	QA	PE
0	05/29/92	Issued for Use Supersedes 20559-M-001 Rev. 2	EJL	MSB	MS	[illegible]	[illegible]

TABLE OF CONTENTS

ATTACHMENTS

0	N/R	N/R	N/R	N/R	N/R	N/R
REV. NO.	ELECT/CS	MECH	CIVIL	PLANT DESIGN	LICENSING/ NUCLEAR SYS.	CONSTRUCTION
COORDINATION SIGN-OFF						

TECHNICAL SPECIFICATION
FOR
DECONTAMINATION SERVICES

1.0 GENERAL

This Specification describes the requirements for mechanical decontamination (decon) of the reactor coolant system (RCS) pipe ends during the steam generator (SG) repair project at Virginia Power's North Anna Nuclear Power Station, Unit 1.

1.1 SCOPE

1.1.1 The scope of work under this Specification is applicable to the ends of the pipes remaining after removal of the three old steam generators. The piping involved are the reactor coolant system hot and cold legs. The decontamination will be performed after removal of the existing steam generators and/or RCS pipe sections.

1.1.2 To separate the steam generators from the RCS hot and cold legs, one of two methods outlined below may be employed on each SG: **

a) The two-cut method which results in the retention of both the hot and cold leg elbows located adjacent to the SG nozzles.

b) The three-cut method which results in the retention of the hot leg elbow and the removal of the cold leg elbow adjacent to the SG nozzle.

c) Deleted

The decontamination process shall be applied to the exposed ends of the remaining RCS piping regardless of the cut method used.

1.1.3 The decontamination process to be used shall be a proven technology, previously applied in an application similar to the work described herein.

1.1.4 The objective of the decontamination process is to reduce radiation dose rates, as measured in front of the open pipe end, to a value as low as reasonably achievable (ALARA). This will result in:

- a) Lower radiation exposure during subsequent work at the pipe ends
- b) Providing a clean metal surface at the pipe ends to facilitate machining and welding
- c) Minimizing airborne radioactivity, hot particle generation and thereby minimizing the need for respirator equipment during machining and welding
- d) Minimizing shielding requirements
- e) Minimizing impact on work activities in progress in adjacent areas

1.2 SUBMITTALS

1.2.1 Schedules

A time schedule for all engineering activities shall be submitted with the proposal. The schedule shall address activities associated with the following:

a) Preliminary Design (submit with the proposal)

b) Detailed Design (submitted no later than 30 days after award of contract)

c) Planning/Procedures (submitted no later than sixty days prior to mobilizing at the site).

d) Manufacturing and Purchasing

e) Construction/Implementation (Schedule to include rad-worker training, equipment set-up and mock-up training, and in-containment decontamination. The project schedule is included in the RFP for information).

1.2.2 Preliminary Design

The following engineering documents and information shall be submitted with the Proposal for the Owner's review:

a) Description of the decon process with all process parameters

b) Description of the decon equipment and internal and external pipe seals, including applicable drawings, diagrams, etc.

c) Description of previous experience and results, including the results of previously performed qualification tests. These tests shall include corrosion coupon evaluation to preclude IGA, IGSCC, pitting, cracking and general corrosion concerns.

d) Expected decon factor and basis of calculation

e) Type and volume of waste which will be generated, including the proposed method for waste handling and waste transfer from the decon equipment.

f) List of drawings and data required from Owner

g) Expected time to perform decontamination

1.2.3 Detailed Design

Based on an approved preliminary design concept, the Contractor shall complete the detailed design and submit the following information for the Owner's approval no later than 30 days after award of contract: **

a) A detailed description and drawings with dimensions of decon equipment and seals.

b) Estimated site support and services required from the Owner, such as;

- Space needed for storage outside containment
- Space needed for storage and operations inside containment
- Waste handling and waste transfer from decon system
- List of instruments and tools requiring calibration (the Owner will provide calibration services)

c) Qualification reports detailing the tests, demonstrations, and/or installations that qualify and test the decon equipment and seals on piping and material similar to that installed at the North Anna Plant.

d) Description of system design features and practices that mitigate the spread of contamination, including any previous test results that demonstrate:

- in-loop testing of seal integrity against movement due to pressure variations
- system features to prevent airborne contamination

e) Design features and safeguards to minimize the potential for excessive surface roughness and excessive metal removal

f) Pipe surface metal loss rates established during tests on similar base material and geometry

g) Expected changes in base material hardness after exposure to the decon process

h) Expected pipe surface roughness after decontamination

i) Deleted

j) Description of safeguards against hose rupture and hose separation, i.e., hose life and usage time based on tests and previous applications

1.2.4 Planning/Procedures

Based on the approved detailed design, the Contractor shall plan all construction activities in detail. The planning work shall consist of developing and submitting the following information to the Owner for approval no later than 60 days prior to site mobilization:

a) Work plan and inspection record.

b) Man-rem estimate for performance of the decontamination.

c) Manpower and shift-charts, including personnel qualification requirements and their on site experience with this type of decon process.

d) Training procedures and necessary mock-ups.

e) Time schedule for the procurement of equipment and material, installation, and performance of work.

1.2.5 Construction/Implementation

1.2.5.1 During the implementation of the decon, the contractor shall complete the final implementation/verification documents (completed procedures).

1.2.5.2 The following verification documents addressing the actual work performed at North Anna Unit 1 shall be submitted for Owner's plant records:

a) Test report documenting qualification and training performed on the Owner's mock-up.

b) Pipe surface metal loss due to decontamination

c) Base material hardness after decontamination

d) Pipe surface roughness after decontamination

e) Records of all consumable material used on the job site for the performance of this work, including written certification to material specification

1.2.6 The Contractor shall maintain copies of all engineering, personnel, and procedure qualification and verification documents at the job site, available for Owner's review. A final report documenting the work performed and the achieved results shall be presented to the Owner after completion of the decontamination work.

1.3 REFERENCE DOCUMENTS

The following reference documents are provided to assist the Contractor in these services:

a)	04803 D 933D245	29.00" Motor Operated 316 SST Gate Valve, Outline Drawing
b)	04809 J 618J842	29.00" Motor Operated Valve Arrangement Drawing
c)	11715-FP-9A-12	Reactor Coolant Piping, Sheet 1 General Layout Drawing
d)	11715-FP-9B-12	Reactor Coolant Piping, Sheet 2 General Layout Drawing
e)	11715-6.11-33A	31.0" x 40° Elbow (cold leg)
f)	11715-6.11-52A	31.0"/29.0" x 50° Elbow (hot leg)

1.4 QUALITY ASSURANCE REQUIREMENTS

1.4.1 Services performed shall be in accordance with the Bechtel Nuclear Quality Assurance Manual and the Owner approved Quality Assurance Program Plan. All work performed under this specification associated with the decontamination process shall be considered Quality Assurance Safety Related (QA SR).

1.4.2 The provisions of Title 10, Code of Federal Regulations, Part 21, entitled, "Reporting of Defects and Noncompliance", apply to the Contractor with respect to materials, equipment (and components thereof) and/or services provided under this Specification, and is designated as QA SR.

1.4.3 All work shall be performed in accordance with procedures that have been reviewed and accepted by the Owner.

1.4.4 All Contractor's instruments and tools requiring calibration shall be calibrated by the Owner prior to use.

1.4.5 Qualification testing of decon equipment and sealing devices on the Owner's mock-up facility shall be witnessed by the Owner.

1.5 IMPLEMENTATION

The following are the Contractor's work items on site, to be performed in compliance with approved methods and procedures, and according to the Work Package Time Schedule in force:

a) Perform on-site training in operating the decon system, including seals, using the Owner's mock-up of the SG channel head and coolant pipes.

) Perform installation, including equipment setup and installation of seals

2.0 CONDITIONS AND REQUIREMENTS FOR DECONTAMINATION

The work associated with each of the [six] ** pipe ends includes: **

a) Decontaminating the inside surfaces of the vertical cold leg pipe ends to a depth of 20 inches (three-cut method). The cold leg pipe ends have an inside diameter of 31 inches.

b) Decontaminating the inside surfaces of the hot leg elbows (two or three cut method) and the cold leg elbows (two cut method) to a depth of 20 inches. The hot leg elbows have an inside diameter of 31 inches at the open end and 29 inches at the other end (Fig. 1). The cold leg elbows have an inside diameter of 31 inches (Fig. 2).

c) Deleted

d) Decon performed under controlled vacuum inside the decon space between loop pipe seal and cover seal at the pipe end, to avoid spread of radioactive particles or fluids into the surrounding areas. If a continuous vacuum during operation cannot be assured, control of the proper function of the seal has to be provided.

e) Deleted

f) Designing, fabricating, and installing the seals into, and their removal from, the pipe ends to prevent uncontrolled spread of decon media.

g) Furnishing of all tools, equipment, and consumable materials necessary to perform the work, including, but not limited to the following:

1) Closed loop decon equipment with integrated cleaning for the used decon media to allow reuse and to keep waste generation at a minimum.

2) The radioactive waste generated shall be compatible with the Owner's waste handling system. The Owner will assume control of and dispose of all waste generated, including equipment which cannot be decontaminated for transport.

3) Double-acting seals inside the loop piping, covering an inside diameter tolerance of 1 inch caused by ovality or varying wall thickness.

Performance of a pressure test of the seal disk gasket at 7 psig (0.5 bars) to assure proper function of the seal to prevent entry of decon media behind the seal into the adjoining loop piping and to assure that the seal will not dislodge during the decon process.

**

4) Easy handling installation tool for the seals to attain predetermined installation depth. All installation activities shall be performed with auxiliary tools to reduce stay time and to keep personnel radiation exposure to a minimum. Connection and disconnection of the installation tool during installation and removal of the seals shall be done without personnel entry into the loop piping. All installation positions for the seals, for either the 2 or 3 -cut method, should be within the range of the installation tool.

5) Hoses shall be pressure tested at 1.5 times operating pressure.

6) Clamping devices and adapters to mount the seal box onto the pipe end. Devices shall cover all tolerances of the inside and outside pipe diameter that may occur.

7) Deleted

8) Instrumentation and control equipment to monitor the progress of decontamination.

9) All consumables required, including decon media, spare hoses, and spare parts.

h) Technically qualified and experienced personnel, including supervisors, to install and remove decontamination equipment and pipe seals.

i) Cleaning and maintaining cleanliness control during work activities.

j) Transfer of items to and from storage, lay-down, and work areas as required by this specification and as directed by the Owner.

3.0 MATERIALS

3.1 REACTOR COOLANT PIPING MATERIALS

The following lists the materials of construction for the existing hot and cold loop piping as well as for the hot leg isolation valves:

Nominal Size (Inches)	Item	Material of Construction
31 ID	RCS Cold Leg Piping	ASME SA-351, GR.CF8A
31 ID x 29 ID	RCS Hot Leg Reducing Elbows and Cold Leg Elbows	ASME SA-351, GR.CF8M
29 ID x 28 ID	Hot Leg Isolation Valve	ASME SA 351, GR.CF8M
7 ID	Hot Leg Cross-tie Piping	ASME SA-376, TP.304

3.2 MATERIAL REQUIREMENTS FOR DECONTAMINATION

3.2.1 All equipment, tools, and devices which will or may be in contact with the existing loop piping, shall be made of non-ferritic material.

3.2.2 For sealing materials, the following limitations and requirements shall be observed:

a) all metal parts shall be of stainless steel or aluminum

b) all rubber/elastomeric parts shall be specified as follows:

[Free halogen < 0.10% by weight
Free sulphur < 0.10% by weight
Free chloride < 50 ppm
Free fluoride < 50 ppm] **

3.2.3 Deleted

3.2.4 Consumables, such as markers, tapes, and protective covers, shall comply with the requirements of the Owner's Specification for Cleaning and Cleanliness Control.

3.2.5 Cleaning materials, such as solvents and rinse water, shall meet the requirements of the Owner's Specification for Cleaning and Cleanliness Control.

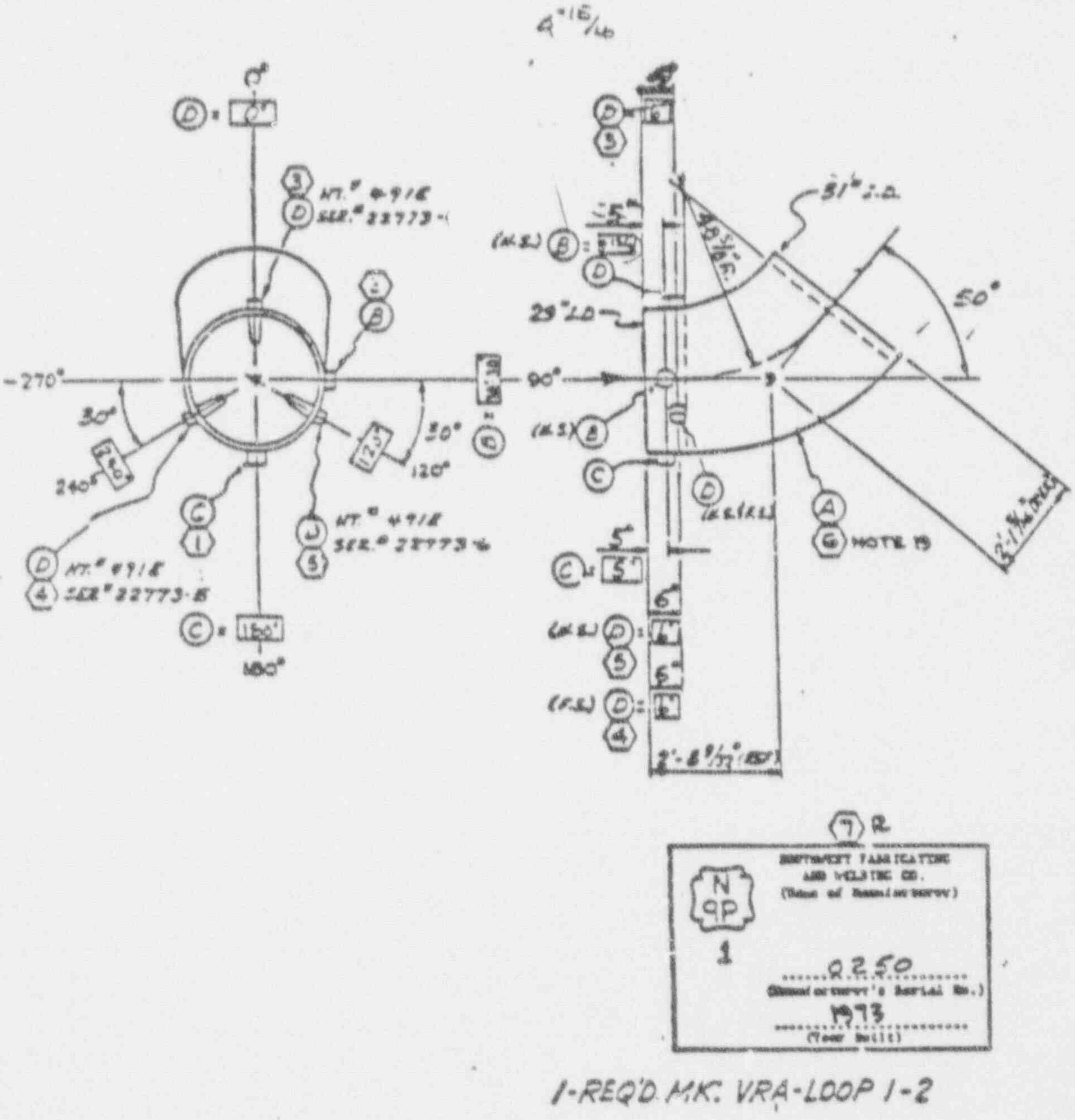


FIGURE 1

Hot Leg Elbow

RCS Pipe End Decontamination

Rev. 0

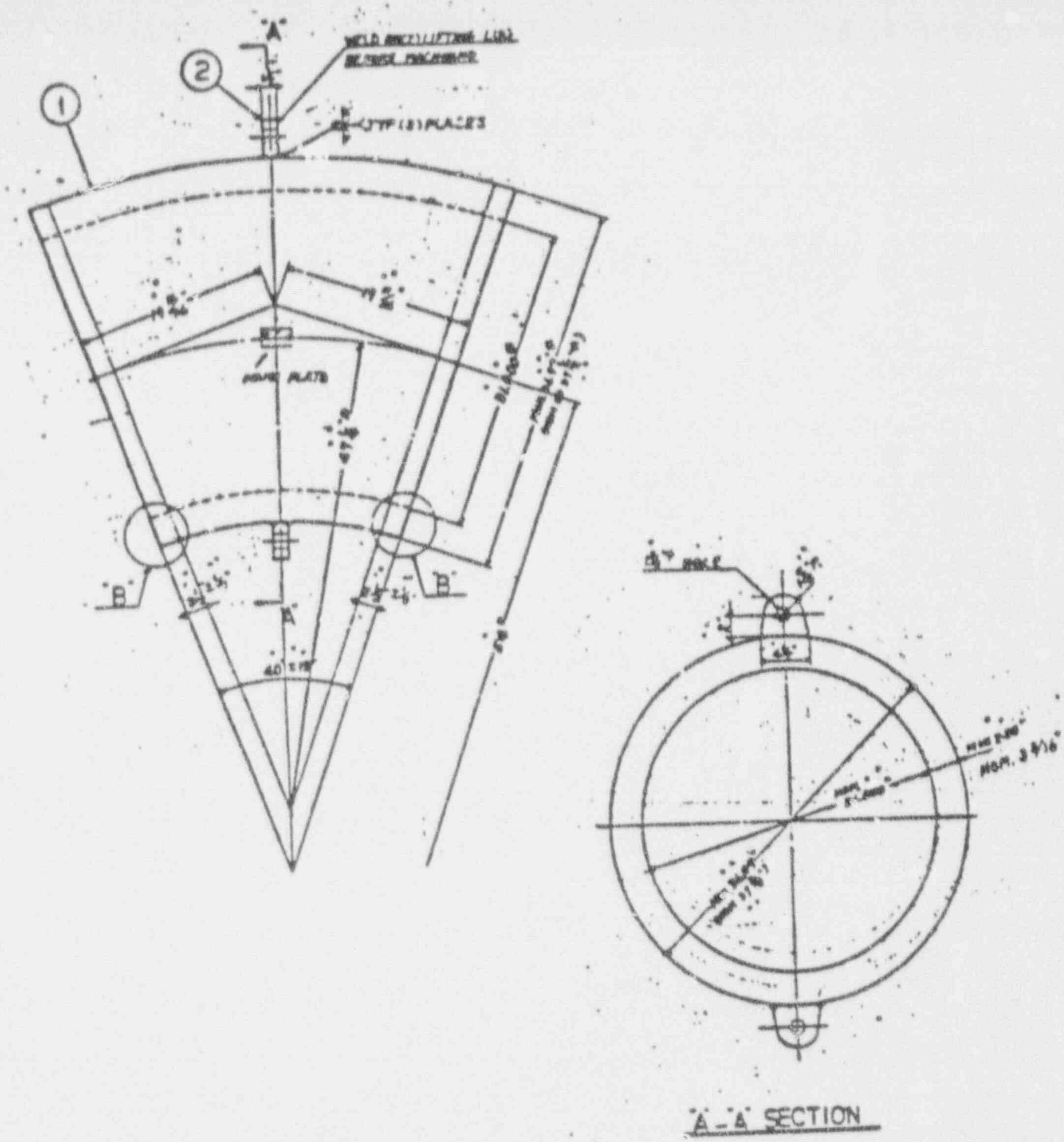


FIGURE 2

Cold Leg Elbow

RCS Pipe End Decontamination

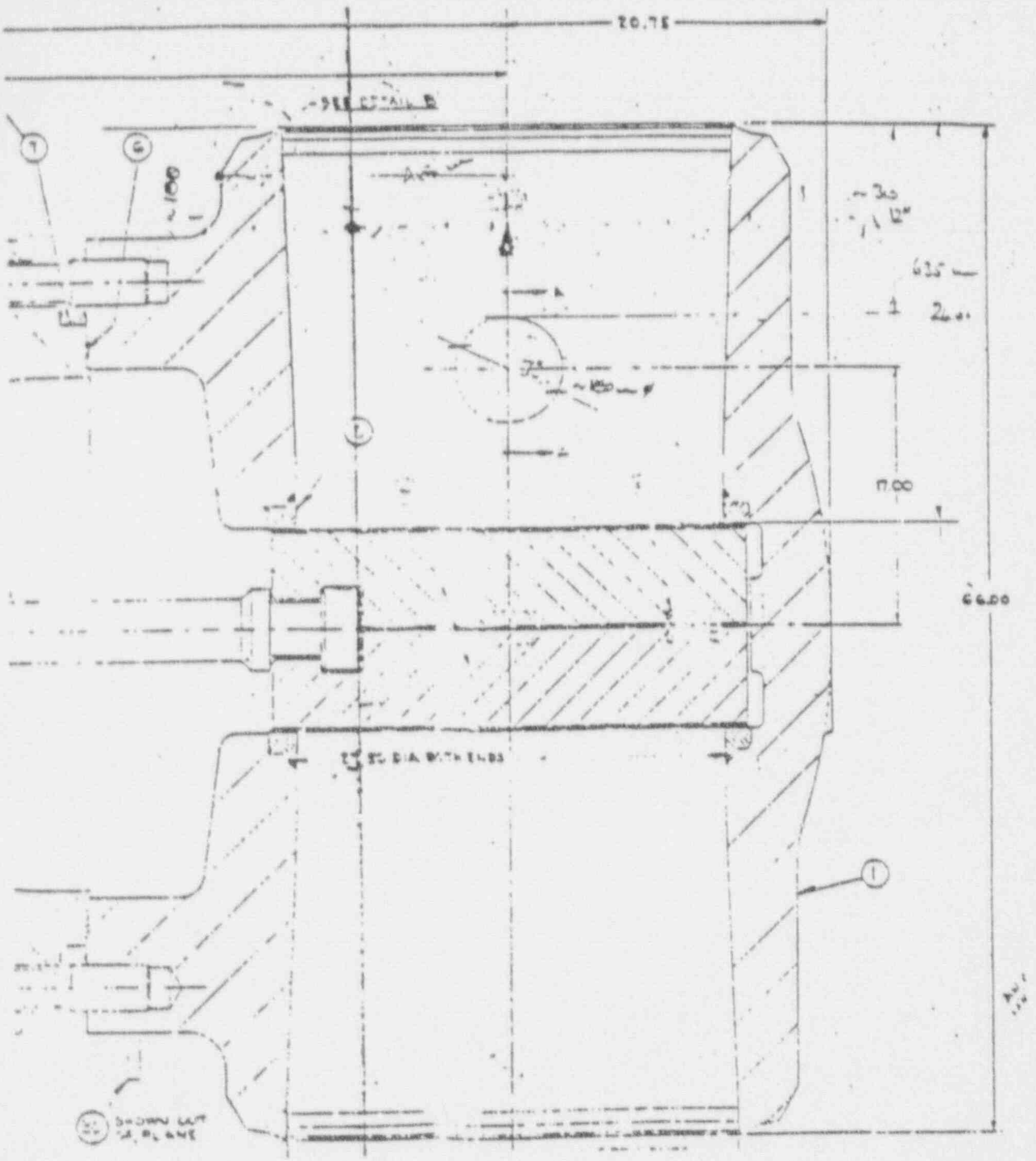


FIGURE 3

Hot Leg Isolation Valve

RCS Pipe End Decontamination

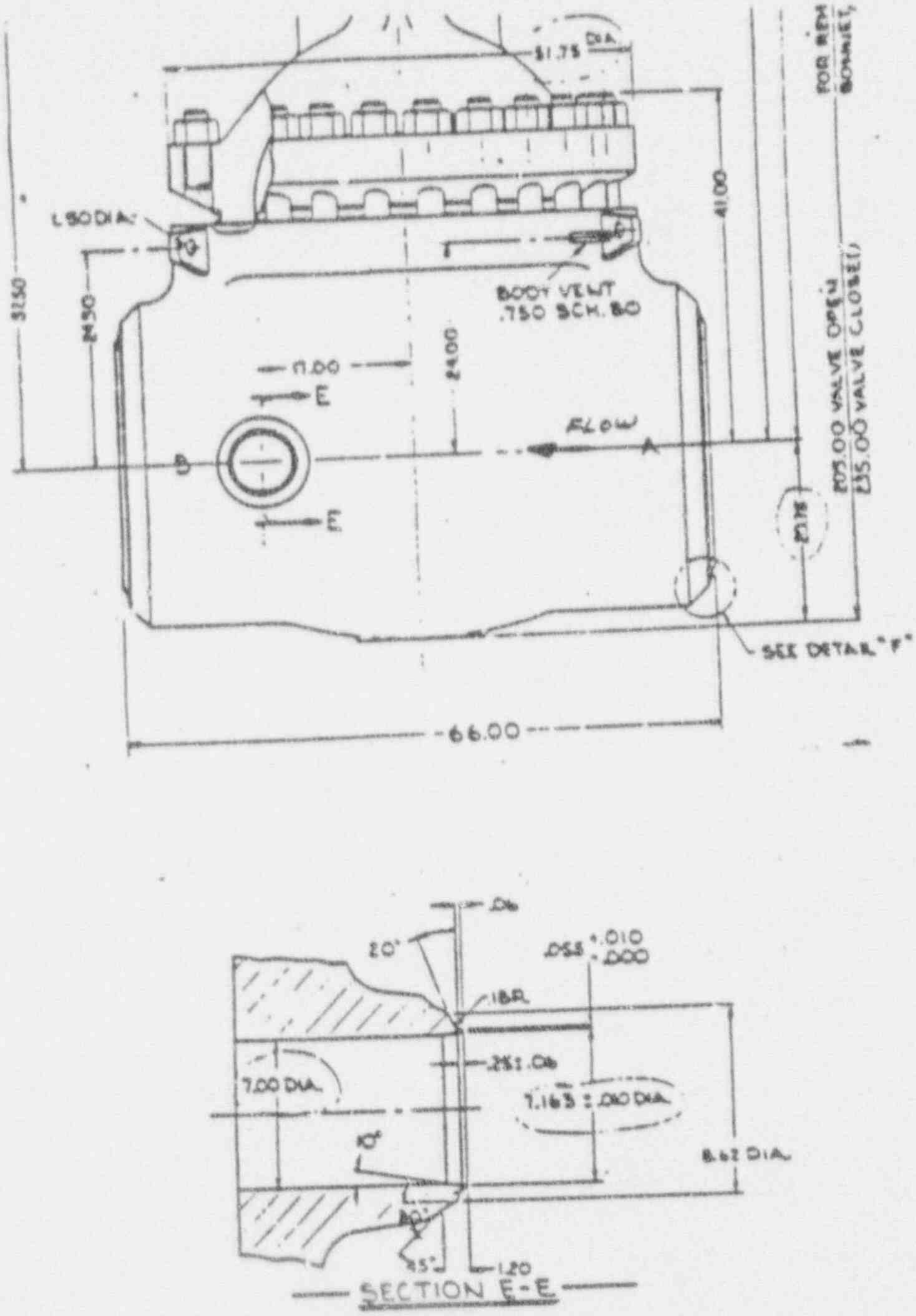


FIGURE 4

Cross Tie Penetration

RCS Pipe End Decontamination

PROCEDURE STD-FP-1991-5563, INSTALL FLOW RESTRICTOR

PER THE WESTINGHOUSE LETTER, "VIRGINIA POWER REPORT, 70% DRAFT CHANGE PACKAGE 90-13-1 FOR THE STEAM GENERATOR REPLACEMENT FOR NORTH ANNA UNIT 1," MR. N. J. LIPARULO TO MR. J. E. RICHARDSON (NRC), DATED SEPTEMBER 2, 1992, THIS ATTACHMENT HAS BEEN WITHHELD SINCE THERE IS INFORMATION IN THIS ATTACHMENT THAT COULD BE PROPPIETARY TO WESTINGHOUSE.

DELETED

REPORT FOR THE HAUL ROUTE INSPECTION AND EVALUATION
STEAM GENERATOR REPAIR PROJECT, NORTH ANNA UNIT 1

REPORT FOR
THE HAUL ROUTE INSPECTION AND EVALUATION
STEAM GENERATOR REPAIR PROJECT
NORTH ANNA UNIT 1

Prepared by R. O. Liatt

Reviewed by William J. Panamoz

Approved by K.L.Bault for D.E. McLellan

October 1991

STONE AND WEBSTER ENGINEERING CORPORATION

HAUL ROUTE INSPECTION AND EVALUATION
STEAM GENERATOR REPAIR PROJECT
NORTH ANNA UNIT 1

TABLE OF CONTENTS

HAUL ROUTE INSPECTION AND EVALUATION
STEAM GENERATOR REPAIR PROJECT
NORTH ANNA UNIT 1

1.0 BACKGROUND AND OBJECTIVE

The replacement of the Unit 1 steam generators requires that the existing, contaminated steam generators be transported from the containment equipment hatch platform to the Interim Old Steam Generator Storage Facility. Virginia Power has identified 2 possible routes. The routes, which are the subject of this report, extend westward from the Unit 1 equipment hatch beyond the "E Field" and security fence and continue westward south of the partially complete Radwaste Facility. Two options exist from this point, Option (1) proceeds southward, within the station security fence, up the slope to the storage facility, and Option (2) proceeds south and west of the switchyard and then eastward to the storage facility, both using existing roadways. A third option, Option (3), would use part of the Option (2) route to avoid a sharp turn and steep grade in the Option (1) route which is within the station security fence.

The objective of the haul route inspection and evaluation is to ensure that the grades and turning radii of the haul route can accommodate the transporter, that slopes and embankments will remain stable when subjected to the transporter wheel loads, that the right-of-way has sufficient strength to support the transporter wheel loads without excessive settlement, and that buried utilities and facilities can withstand the overburden pressures and remain within their allowable load limits.

The results and conclusions of this inspection are included in this report. Drawings are attached which depict the haul route options. The report concludes that the preferred haul route would be Option (2), the route which passes south and west of the switchyard and then eastward to the storage facility.

2.0 APPROACH

This effort consisted of three activities described below, (1) the survey of the proposed haul routes to the storage facility, (2) the inspection of the proposed haul route and (3) the evaluation of the route and the buried utilities and facilities.

2.1 Activity 1. Survey of Proposed Haul Routes and Transporter Turnaround Room

Stone & Webster surveyed all 3 options for the proposed haul route and prepared a drawing incorporating the survey data for the access road to the proposed steam generator storage facility. For use in the field, it was separated into 12 sheets, and it is included in Appendix A to this report.

This drawing includes the survey data along the entire route, including the existing paved roadways. Turnaround room for the steam generator transporter and its configuration relative to the storage facility (identified by station coordinates at each corner of the building) are included on this drawing. Spot elevations and contours were provided by the survey notes.

The survey identified haul route centerline coordinates, using the station grid. Stations were placed at 100 foot intervals. At each station several side shots perpendicular to the haul route were taken on each side of the centerline. The elevation and conditions at each point surveyed was recorded. The points surveyed from the haul route centerline station extended 25 feet in each direction resulting in a 50 feet wide right of way survey.

In addition, spot elevations, contours, surface condition, site drainage facilities, nearby structures, and buildings, and valve stations were also identified. The locations of other facilities which could pose a potential interference with the transporter were also identified during the survey. These included fences, fire hydrants, and overhead lines less than 25 feet above the haul route surface. At each section the edge of pavement, top of the slope at drainage ditches, ditch centerline and the ditch invert elevation was obtained. Inside the Protected Area, station drawing 13075-EY-8A was used as the basis to locate the haul route and identify all safety related and non-safety related buried utilities. The marked-up station drawing and Table A, which lists the buried utililties, are included in Appendix A to this report. These items are also shown on Sheet 1 of the survey drawing included in Appendix A.

The survey identified the radius at each curve as measured from the right of way centerline. This permitted an evaluation of the capability of the transporter to negotiate the curve. A minimum inside turning radius of 24 feet and a minimum outside turning radius of 45 feet, as provided by Virginia Power, were used for this evaluation.

The computerized drawing incorporating the information above and the final grading and surfacing of the haul route/access road was prepared from the survey notes using the Intergraph CAD system by S&W. This information will be provided to Virginia Power on a computer disk. The drawing includes the existing conditions at the site where the Old Steam Generator Storage Facility will be located. The drawing prepared from the survey notes was attached to the inspection plan described below and it was used in the walkdown inspection.

2.2 Activity 2. Walkdown Inspection of The Proposed Haul Route

Reference Station Drawings 1'715-FY-1B and 13075-EY-8A-1

An inspection plan (attached as Appendix A to this report) was prepared using the referenced station drawings and the available boring data. All buried utilities and facilities that are documented on station drawings were identified prior to the inspection and station coordinates were assigned to each item.

The inspection was supported by Stone & Webster surveyors. The ground surface elevation and the conditions at the surface were identified during the inspection, and the centerline coordinates of the proposed haul route were used as a reference during the inspection. Turning radii, grades, and lateral or overhead clearances were reviewed to verify the feasibility of the haul route. Outside the Protected Area, the drawing (included as 12 sheets in Appendix A) of the right-of-way was prepared by the surveyors and engineers identifying the width of the haul route, the turning radii, utility vaults, and other items on, under, or near the haul route.

All buried utilities and facilities within the Protected Area were identified and tabulated (included as Table A in Appendix A). The buried utilities outside the Protected Area which were located and evaluated as part of this effort are the security ductbank, the new service water pipe lines, the circulating water discharge tunnel built for Units 3 & 4, and the 12" diameter firemain.

The walkdown was documented in a walkdown report (attached as Appendix B to this report) and the results are summarized in Section 3.0.

2.3 Activity 3. Evaluation of Haul Route Right-Of-Way and Buried Utilities

The utilities and facilities buried beneath the haul route were evaluated using standard hand calculation methods of structural analysis for determining the stress below grade for a wheel or axle loading. The depth of the ductbanks and the abandoned Unit 3 & 4 circulating discharge tunnel was determined from station drawings. The depths of culverts and piping were determined with the aid of a scanner. At those locations where additional support is required to prevent overstressing the buried component, the specific component has been identified and the recommended means to prevent overloading is provided.

By inspection and observation, the haul route was evaluated with respect to the capability of the ground or pavement surface to support the wheel loads without settlements which could cause the transporter to become unstable. Virginia Power provided the proposed transporter wheel loadings, the limits on settlements under wheels, acceptable turning radii, and upper limits on the grade (percent).

Slopes and embankments were observed during the walkdown to permit making a judgement of their susceptibility to failure due to passage of the transporter. Where required, upgrades to the haul route consisting of increases to the width of the right-of-way and minor relocations of the haul route to avoid potential problem areas are identified in this report.

Whereas a detailed engineering study of the road design is outside the present scope of work, it would be prudent to perform a load test along the proposed route prior to transporting the old steam generators. The load test can be performed with a flat bed trailer loaded with concrete blocks such that the axle loading will approximate the axle loading of the steam generator transporter. Any problems with settlement or instability of the soils at the ground surface would be noted during the test. Areas where settlements appear to be excessive can be repaired prior to moving the steam generator.

3.0 WALKDOWN INSPECTION

The walkdown started at the Unit 1 containment. Using the drawings prepared by S&W and the yard drawing, included in Appendix A, the following items along the proposed haul routes were identified:

- culverts
- electrical ductbanks
- buried piping
- telephone and other communications cable

These items were observed at cable vaults or at locations where they penetrated the ground surface. They were followed to points where the transport vehicle will pass over them on the haul route. A scanner was used to obtain approximate depths of the underground items. However, it was not possible to obtain the depths of the fire protection piping near Station 20 of the Option (1) route. This is not a concern since the fire protection piping is usually buried a minimum of 2.5 to 3 feet below the ground surface.

In the Protected Area, fuel oil lines, fire protection piping, and electrical ductbanks were reviewed as shown on the yard drawing. Outside the Protected Area, electrical conduit, air lines, domestic water, fire protection piping and communications cabling were observed. Survey drawings were marked with depths of these items, when this was obtainable, and coordinates were also noted when these were deemed necessary.

No serious road damage along the haul route was found; i.e., potholes and serious bumps were not observed. Some cracking of the pavement and concrete at certain locations was observed, but none serious enough to impede the transporter. It appeared that soil at some locations may become soft or slippery when it is wet; however, gravel covers the haul route at most locations that are not paved, thereby providing a sound base for wheeled vehicles.

Slopes in the vicinity of the proposed routes also were observed. In general, these are sufficiently gentle (approximately 2.75:1, horizontal:vertical) and are at a sufficient distance (such that the transporter wheels will be greater than 10 feet from the top of the slope) from the centerline of the proposed routes that they should not be adversely affected by passage of the transporter if it travels along the centerline of the proposed routes. Section 4.3 provides additional details.

Overhead and side-to-side clearances were checked. For the Option (2) route, it was noted that at Station 48, the gate opening is limited to 20.5 feet and the minimum clearance below overhead wires is 19.5 feet near the edge of the road.

The walkdown also addressed the direct buried conduit located in the new steam generator unloading area. These conduit were observed to be about 2.5 to 3 feet deep and are sufficiently flexible that they should not be adversely affected by passage of the transporter.

The complete walkdown report is attached to this report in Appendix B.

4.0 HAUL ROUTE EVALUATION

The evaluations presented herein are based on Stone & Webster's field observations, experience, undocumented calculations, and on engineering judgement when necessary. In accordance with the scope of work, no documented engineering calculations were prepared for this effort.

4.1 Road

The evaluation in this section is based upon a transporter with the following characteristics:

Width	10	feet
Approximate length	110	feet
Inside turning radius	24	feet
Outside turning radius	45	feet
Number of axles	10	
Number of wheels per axle	8	
Wheel base	5	feet (approx)
Load per wheel	7	kips

This information was provided by Virginia Power.

The gross weight of the steam generator and transport vehicle is expected to be about 560 to 580 kips. This weight will be distributed over a minimum of 80 wheels resulting in an average wheel load of about 7 kips. This compares to a wheel load of about 12.5 kips for the site P&H crane, which, according to Virginia Power personnel, has a dead weight of 150 kips distributed over 12 wheels and has been moved over most of the proposed route several times without damaging underground utilities Therefore, the road should be capable of supporting the old steam generator transporter.

Within the Protected Area, the haul route will avoid critical underground items where possible. At the west end of the Protected Area, the route centerline is placed south of fire line D_1 and north of fuel oil line N_1. These are shown on drawing 13075-EY-8A-1, which is included in Appendix A.

There are a few locations where the road surface is cracked or damaged; however, even these locations should support the wheel load. Unpaved portions of the road should be capable of sustaining the wheel loads during dry weather.

A corrugated metal pipe protruding from the unpaved road is used as a valve housing at Station 13. The haul route centerline is very close to this valve housing. To avoid damaging the underlying piping and valves, the transporter should pass to the south of the centerline in this area. The road base in this area should be prepared in accordance with existing requirements for the North Anna site to provide a bearing surface that is comparable to that of the existing unpaved road. Also, the unpaved portion of the haul route near the storage facility should be treated in the same manner.

A sharp curve in the road with a steep grade, about 9%, and a centerline radius of 80 feet inside the fence from Stations 19 to 22 makes the Option (2) route outside the security fence area more favorable than Option (1). Although the curve radius exceeds the minimum required radius, presence of both a curve and a grade at the same location may cause potential maneuvering problems.

There is also a hairpin curve in the Option (2) route from Stations 31 to 36; however, this curve is on a flat surface with ample side-to-side clearance. There a few low branches, which require trimming, along this route near Station 45.

If it is necessary to avoid the hairpin turn in the vicinity of Station 19 in the Option (1) route and to minimize the path of travel of the transporter outside of the security fence, the Option (3) route could be used. The transporter could exit the security area at the gate at Station 19. It could then proceed up the hill on the Option (2) route for only about 200 feet and then reenter the Option (1) route at the gate near Station 21. Some grading outside the gate near Station 21 would be necessary and demolishing of a concrete invert may be required for this option.

4.2 Buried Utilities

Buried utilties which are 6 feet or more below the ground surface or which are buried at a depth sufficient for tornado missile protection are not deemed to be susceptible to damage from the steam generator transporter. Therefore, the evaluation of buried utilties included only those items within the 6 foot depth.

The following underground utilities were evaluated to determine their susceptibility to damage from the transporter wheel loads:

- The security ductbank near the Protected Area gate was evaluated. This ductbank is identified as electrical ductbank EDB_4. A loaded axle consisting of 8 wheels passing directly over this ductbank will cause bending in the reinforced concrete ductbank. When evaluated as a beam on an elastic foundation, the internal moment is only a fraction of the maximum moment allowed by the ACI

Code (ACI 318-89) even when a conservative load factor of 1.7 is applied to the load.

Other ductbanks in the Protected Area were also reviewed and were found to meet code requirements for the transporter loadings.

- o The transporter will pass over the North Anna Units 3 & 4 circulating water discharge tunnel after it passes through the gate when leaving the Protected Area. The roof of the tunnel was evaluated. This is a heavily reinforced concrete slab with a minimum thickness of 3 feet. The concrete meets the structural design requirements of the ACI code and its load capacity exceeds the applied load by a wide margin.

- o The various water piping including the service water piping, air piping, and direct buried electrical conduit, will be subjected to external soil pressure when the transporter passes over these items. These pressures diminish quickly with increased depth below the surface. The maximum vertical soil pressure on a pipe which is 18 inches deep is less than 10 psi for a wheel load of about 7 kips. Even when adjacent wheels are considered, these pressures should not adversely affect the buried utilities.

- o A number of culverts pass under the proposed haul route. These are constructed of large diameter corrugated metal pipe and many of these are close to the surface. Since these will be affected by an axle load from several wheels of the transporter, drainage culverts should be protected by laying flat plate over them to spread the load to the soil on either side of the culvert. The culverts requiring protection as shown in the attached Table 1.

4.3 Slope Stability at Sides of Haul Route

Slopes in the vicinity of the proposed routes also were inspected for stability. In general, these consist only of shallow ditches along the edges of the roadway, whose depths below the roadway are about 2 to 4 feet, with gentle side slopes, approximately 2.75:1 (horizontal to vertical). From about Station 37 to Station 47 along the Option (2) route, the roadway is approximately 2 to 5 feet above the surrounding area, with similar, gentle slopes, approximately 2.75:1. Assuming the transporter will be about 10 feet wide and will travel along the centerline of the proposed routes, the stability of these "slopes" will not be adversely affected by the transporter.

The ditches along the south side of the Option (1) route in the vicinity of Stations 18 to 19.25 and along the north side of the Option (2) route in the vicinity of Station 22 are somewhat steeper. Even so, these also are a sufficient distance away from the centerline of the roadway such that they will not be adversely affected by the transporter when it travels along the centerline of the proposed routes. However, it may not be possible for the transporter to negotiate the hairpin turn in the Option (1) route in the vicinity of Station 19 without approaching the southern edge of the roadway near this ditch. If this is the case for the proposed transporter, it may be necessary to backfill this ditch to ensure that the southern edge of the roadway does not suffer a slope failure due to the loading imposed by the transporter. This area, therefore, will require additional study when the transporter to be used for moving the steam generators is selected.

All other slopes along the proposed routes are a sufficient distance from the proposed path of travel that the loading to be imposed by the transporter will not adversely affect the stability of the slopes. This includes the approximately 20 feet deep cut that remains where the reactor buildings for Units 3 & 4 were to be constructed, which is about 50 feet north of the proposed route in the vicinity of Stations 6 to 11.

5.0 CONCLUSIONS

- o It is concluded that the Option (2) route (outside the security fence) should be used for transporting the old steam generators to the interim storage facility. This route will avoid the hairpin turn and steep grade from Stations 19 to 22 of the Option (1) route (inside the security fence).
- o An Option (3) route, as discussed in Section 4.1 above, could be selected in the event that the Option (2) route is not available. If this option is selected, it will be necessary to grade the area between the Option 1 and Option 2 roads and it may be necessary to demolish a concrete invert in this area.
- o Items shown in the attached Table 1 require protection and improvements as noted in the table.
- o To avoid damage to paved roadways, the old steam generator should be moved when the road surface is cool.

- The road base in the area of the Old Steam Generator Storage Facility and south of the valve housing at Station 13 should be prepared in accordance with existing requirements for the North Anna site to provide a bearing surface that is comparable to that of the existing unpaved roads.

- The slopes along the sides of the proposed routes generally are sufficiently gentle and at a sufficient distance from the centerline of the roadway that they will not be adversely affected by passage of the transporter. Additional studies are required, however, to demonstrate the stability of the slopes in areas where the transporter must approach within 10 feet of the top of the slope.

- A test should be performed using a trailer loaded to approximate the axle loading of the steam generator transporter. The trailer should be driven over the proposed haul route, especially the unpaved portions. This would reveal any problems with settlement or instability of the soils at the ground surface. The items described in Table 1 should be protected during the test.

TABLE 1

UNDERGROUND UTILITIES REQUIRING PROTECTION

ROUTE	DESCRIPTION	STATION	PROTECTION METHOD
All	Manhole	1	1 1/2" Steel Plate
All	24" Culvert	14	1 1/2" Steel Plate
Option (1)	24" Culvert	26	1 1/2" Steel Plate
Option (2)	21" Culvert	47.7	1 1/2" Steel Plate
All	10" Culvert	34.5	1 1/2" Steel Plate
All	12" Culvert	36.5	1 1/2" Steel Plate
All	18" Culvert	38	1 1/2" Steel Plate

APPENDIX A

INSPECTION PLAN

NORTH ANNA STEAM GENERATOR REPAIR HAUL ROUTE

A walkdown inspection of the North Anna steam generator repair haul route will be performed by a SWEC team consisting of:

- Structural engineers
- Geotechnical engineer
- Surveyors

Two routes defined by a previous survey will be inspected. This includes a primary route as well as an alternate route.

The following steps have been or will be performed as part of Activity 2 of the haul route inspection and evaluation, identified in the SWEC scope letter (NP1486.V01-081, NAS-20,264).

1. A right of way drawing from the Surveyor's data acquired during the previous survey was produced. This drawing is attached (12 sheets) and is a part of this plan.

2. Underground utilities within the protected area have been tabulated in Table A and marked on a plant site drawing, 13075-EY-8A-1. These give the following information:

   - Description
   - Elevation
   - Coordinates

   Table A and marked-up drawing 13075-EY-8A-1 are attached and are a part of this plan.

3. In the field, the location of each underground item will be identified by the survey crew. For those items below the surface of the ground, a scanner will be used to confirm the locations. Actual coordinates and elevations of the tops of these items will be noted on the attached table.

4. The haul route will be inspected for perturbations in the road surface and other potential hazards which may pose a danger to the haul vehicle. These potential hazards include but are not limited to the following:

   - potholes
   - soft spots
   - bumps

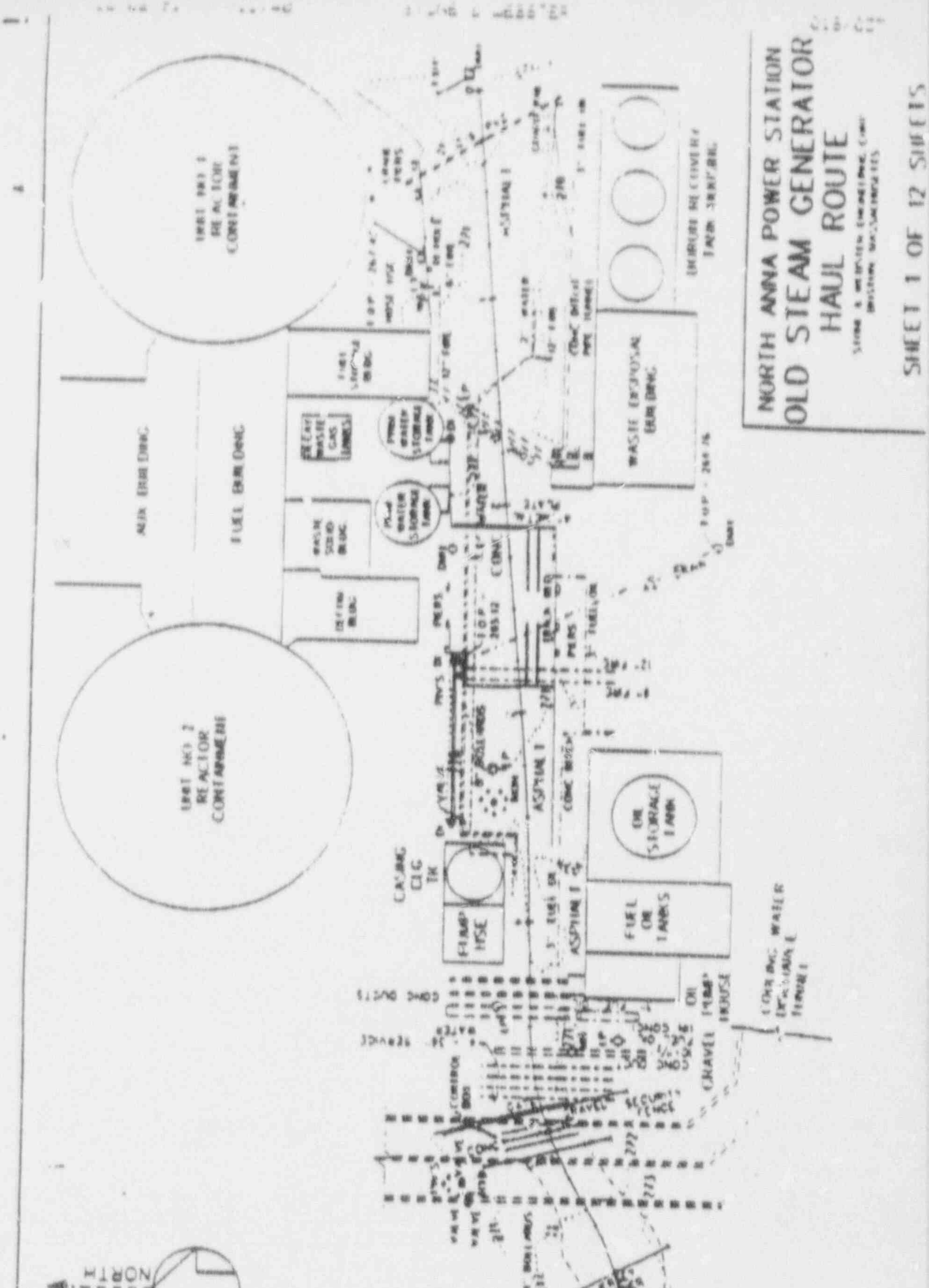
5. The sides of the proposed haul route shown on the attached drawing will be visually checked for slope stability.

6. During the walkdown, items not shown on the drawing or table will be noted for later evaluation.

7. The turns in the proposed route will be checked to determine if there are adequate clearances to accommodate the haul vehicle.

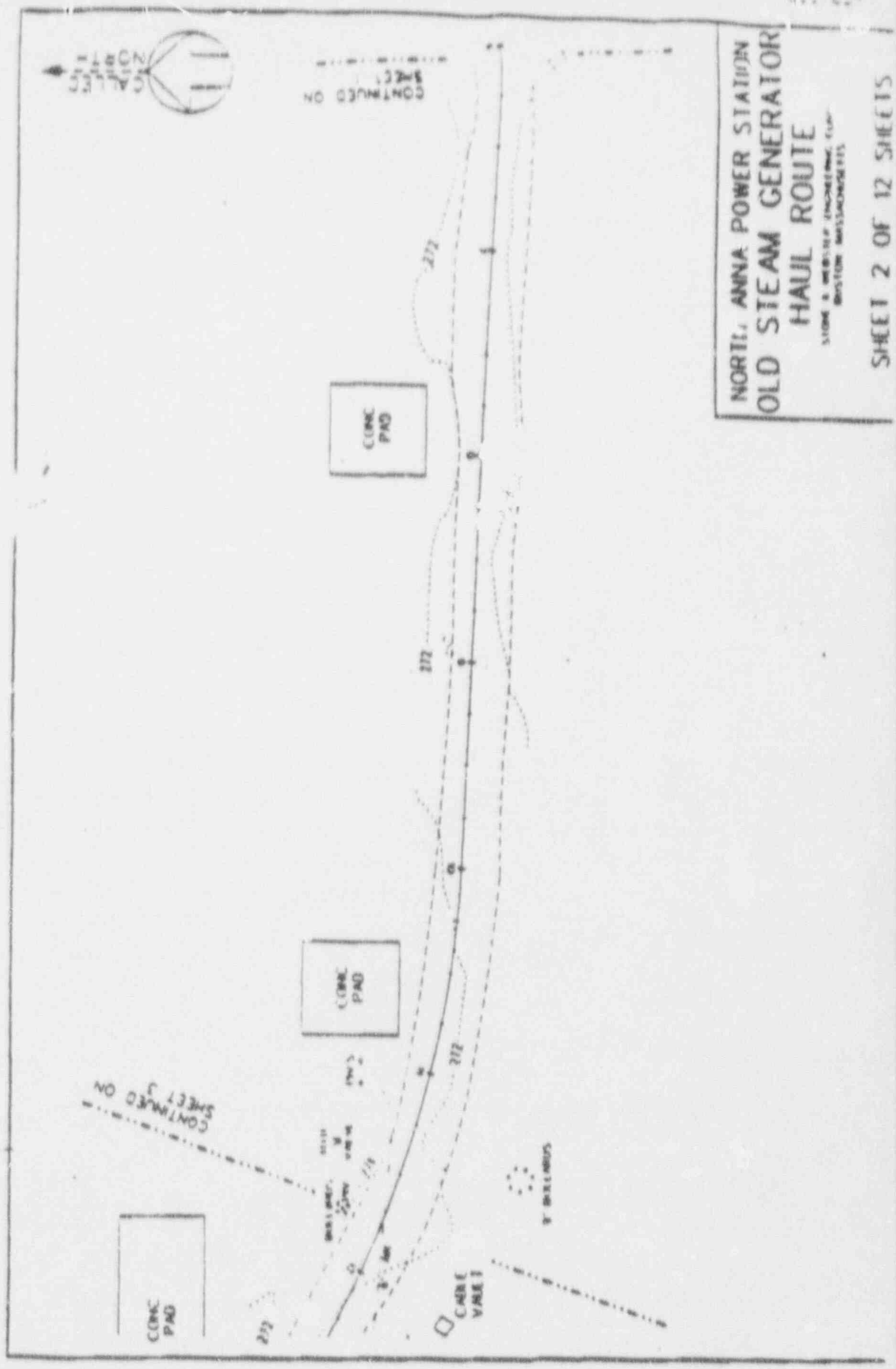
8. The overhead and side to side clearances will be checked to ensure that there is adequate clearance for the haul vehicle.

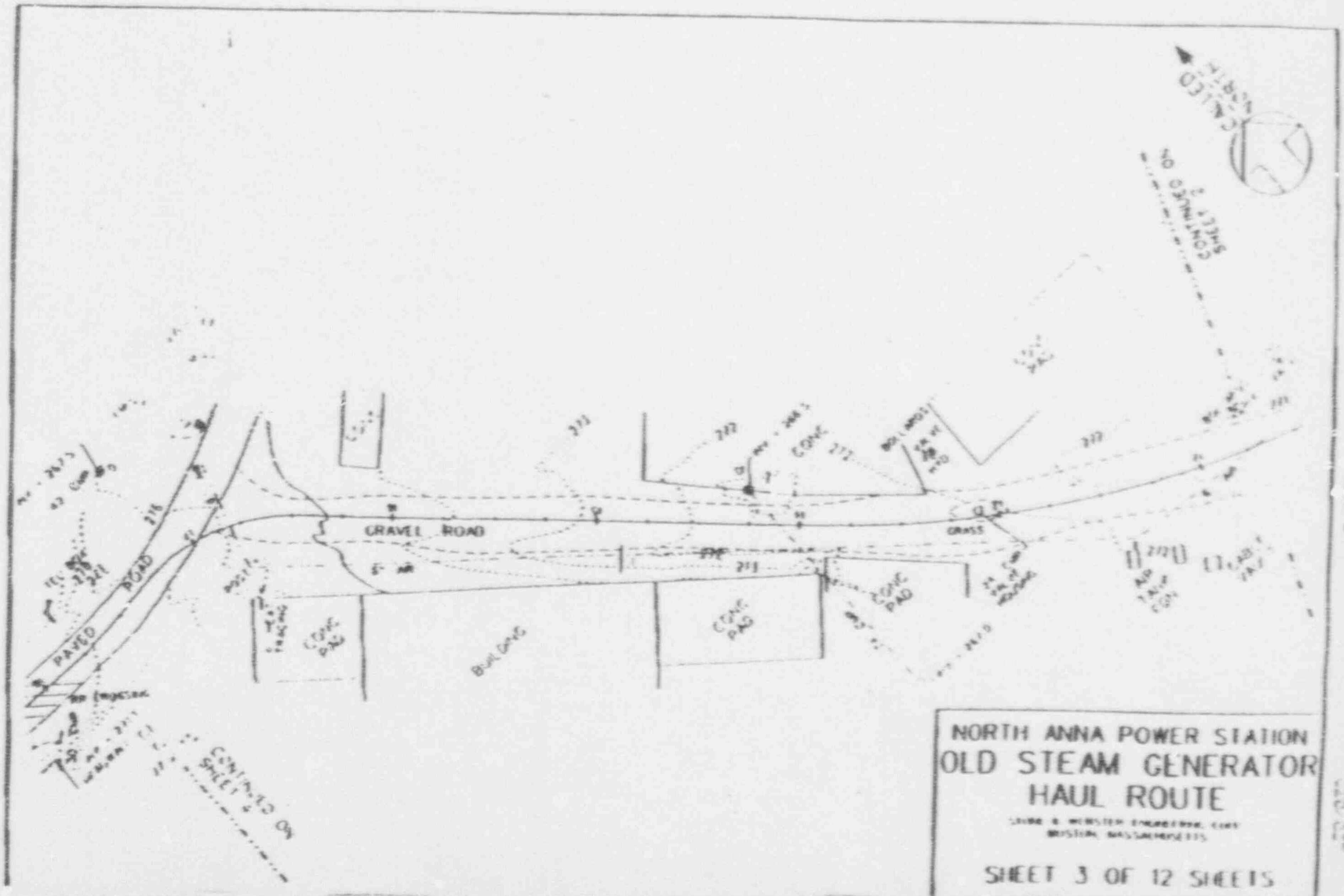
The results of the walkdown inspection will be documented in the letter report prepared in Activity 3.



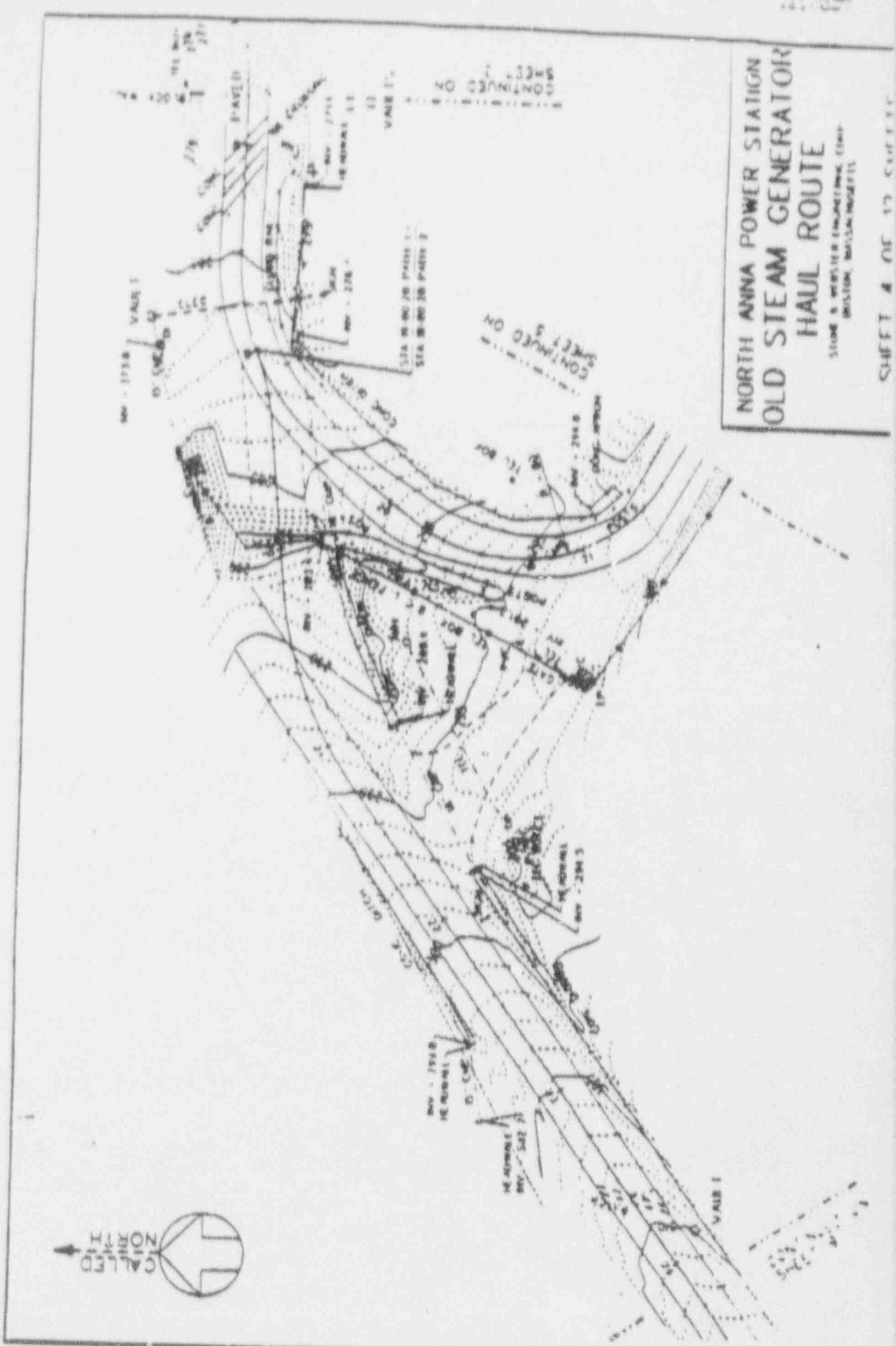
NORTH ANNA POWER STATION
OLD STEAM GENERATOR
HAUL ROUTE
SHEET 1 OF 12 SHEETS
UNIT NO 1 REACTOR CONTAINMENT
UNIT NO 2 REACTOR CONTAINMENT
AUX BUILDING
FUEL BUILDING
WASTE DISPOSAL BUILDING
OIL STORAGE TANK
FUEL OIL TANKS
CASING CLG TK
PUMP HSE
ASPHALT
CONC
GRAVEL
SECURITY FENCE
OIL PUMP HOUSE
COOLING WATER DISCHARGE TUNNEL
CALLED NORTH
CONTINUED ON SHEET 2

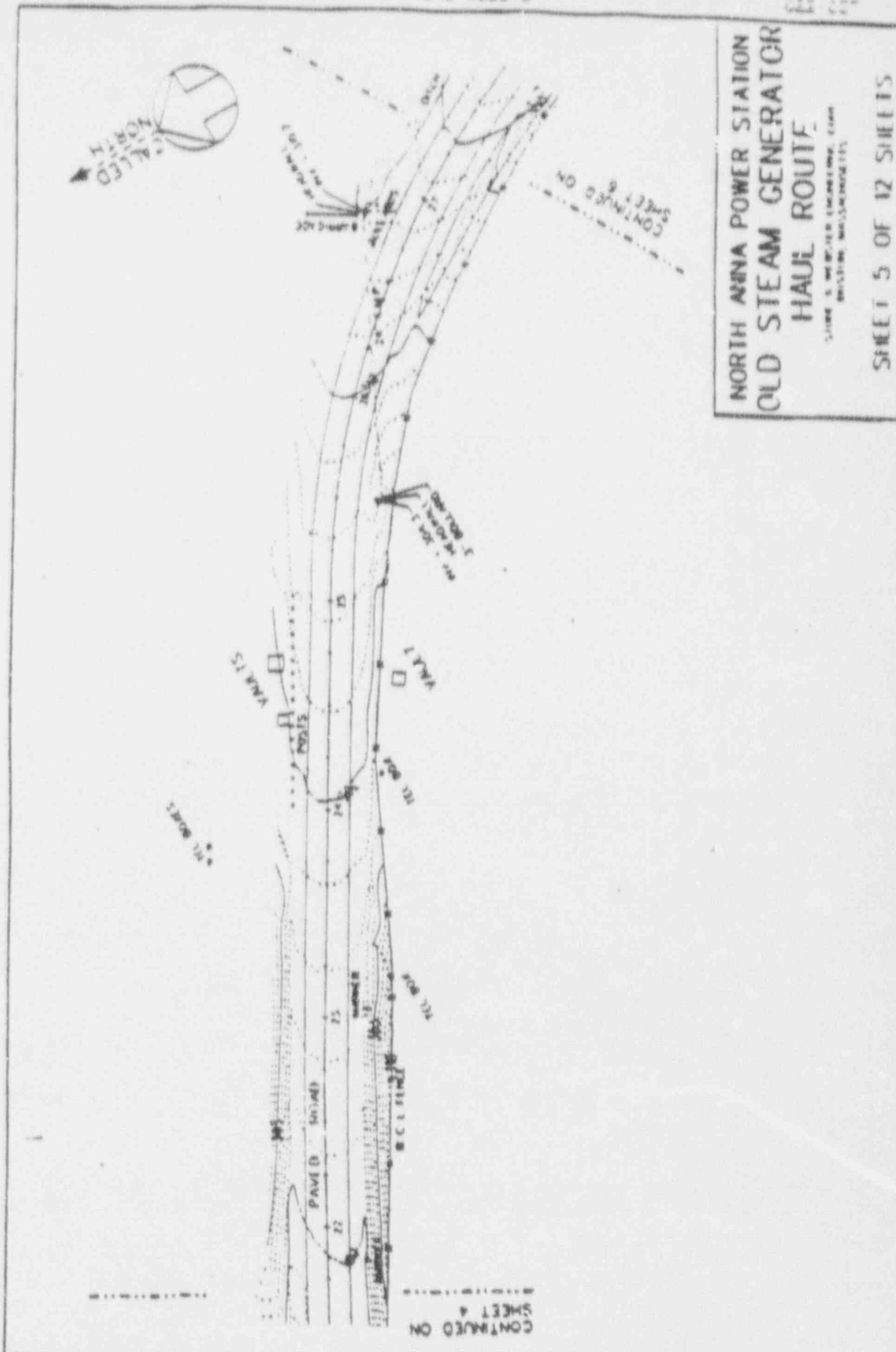
A-

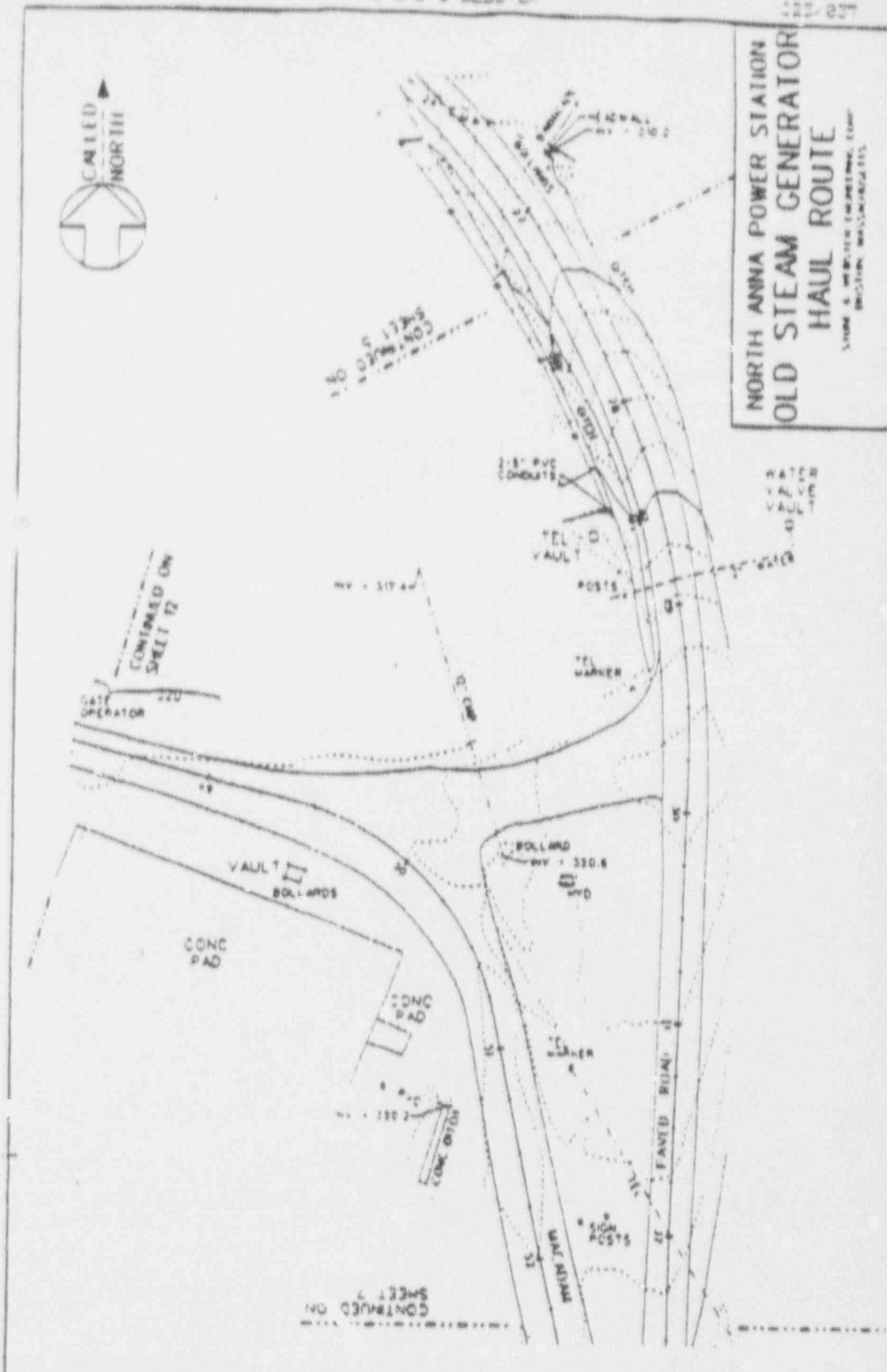

NORTH ANNA POWER STATION
OLD STEAM GENERATOR
HAUL ROUTE
STONE & WEBSTER ENGINEERING CORP.
BOSTON MASSACHUSETTS
SHEET 2 OF 12 SHEETS
CONTINUED ON SHEET 1
CONTINUED ON SHEET 3
CONC PAD
CONC PAD
CONC PAD
272
272
272
272
CABLE VAULT

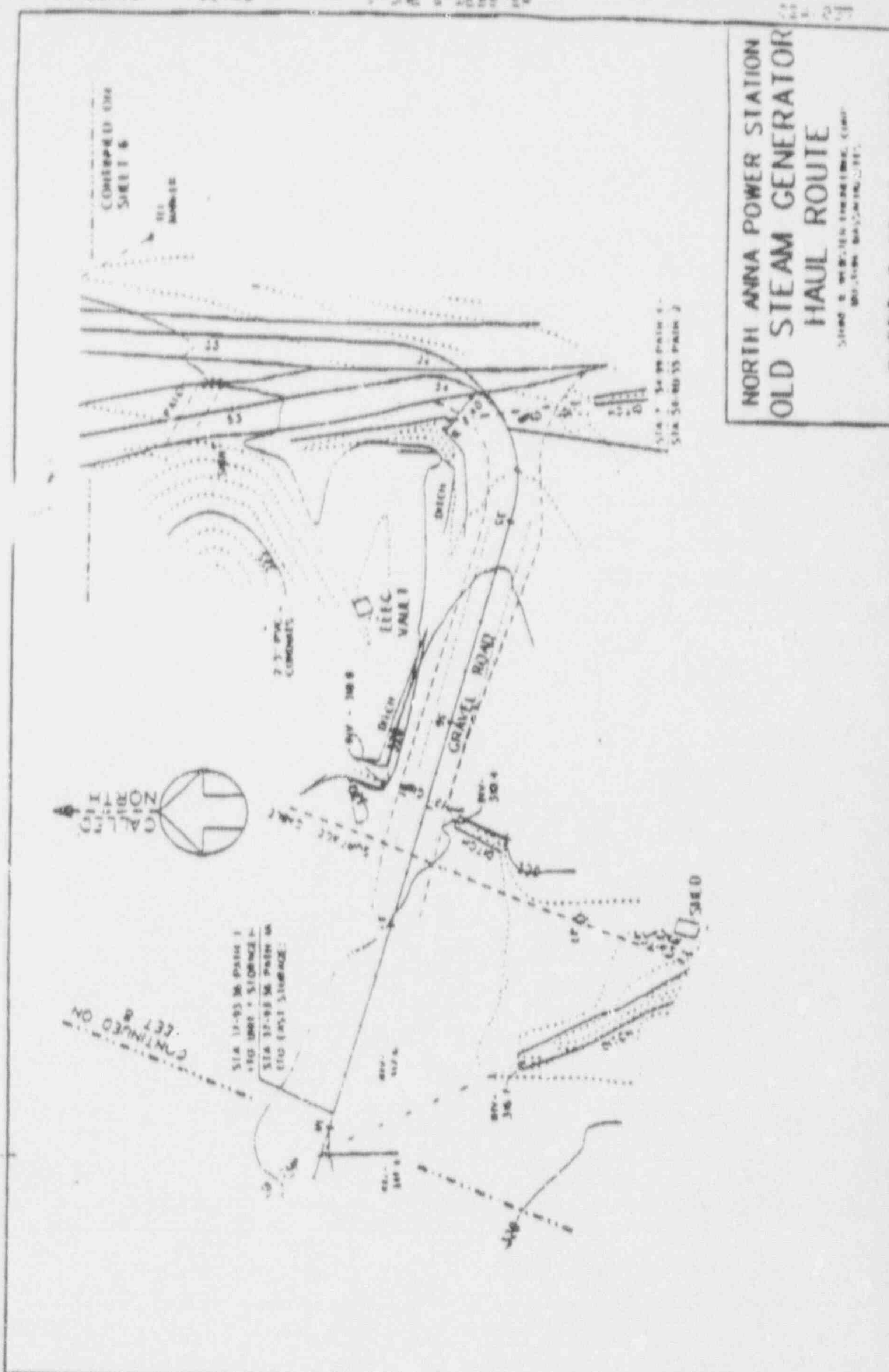


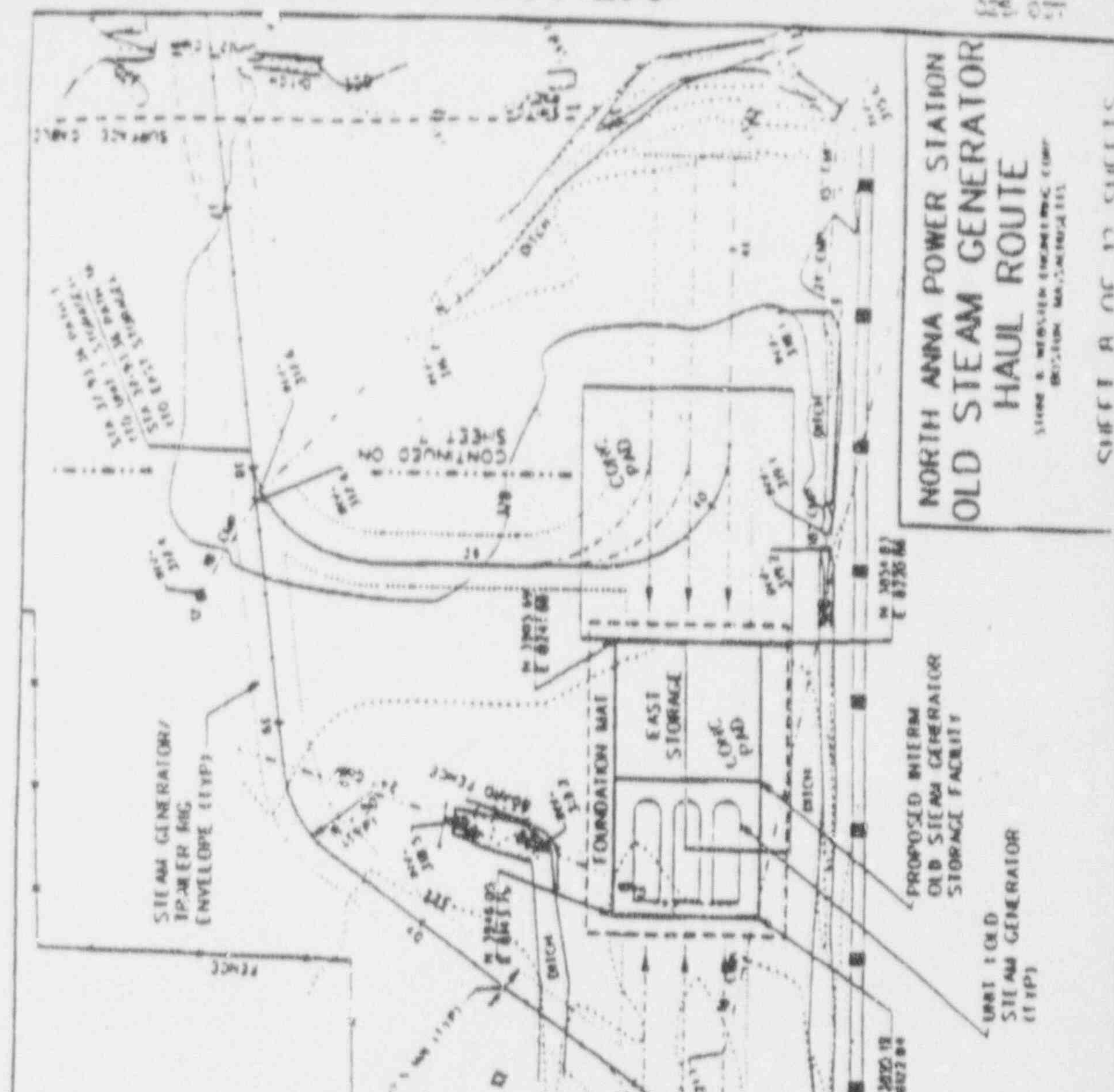
NORTH ANNA POWER STATION
OLD STEAM GENERATOR
HAUL ROUTE
SHEET 3 OF 12 SHEETS
PAVED ROAD
GRAVEL ROAD
CONC PAD
CONC PAD
CONC PAD
CONC
GRASS
CONTINUED ON SHEET 2
CONTINUED ON SHEET 4


NORTH ANNA POWER STATION
OLD STEAM GENERATOR
HAUL ROUTE
CONTINUED ON SHEET 3
CALLED NORTH

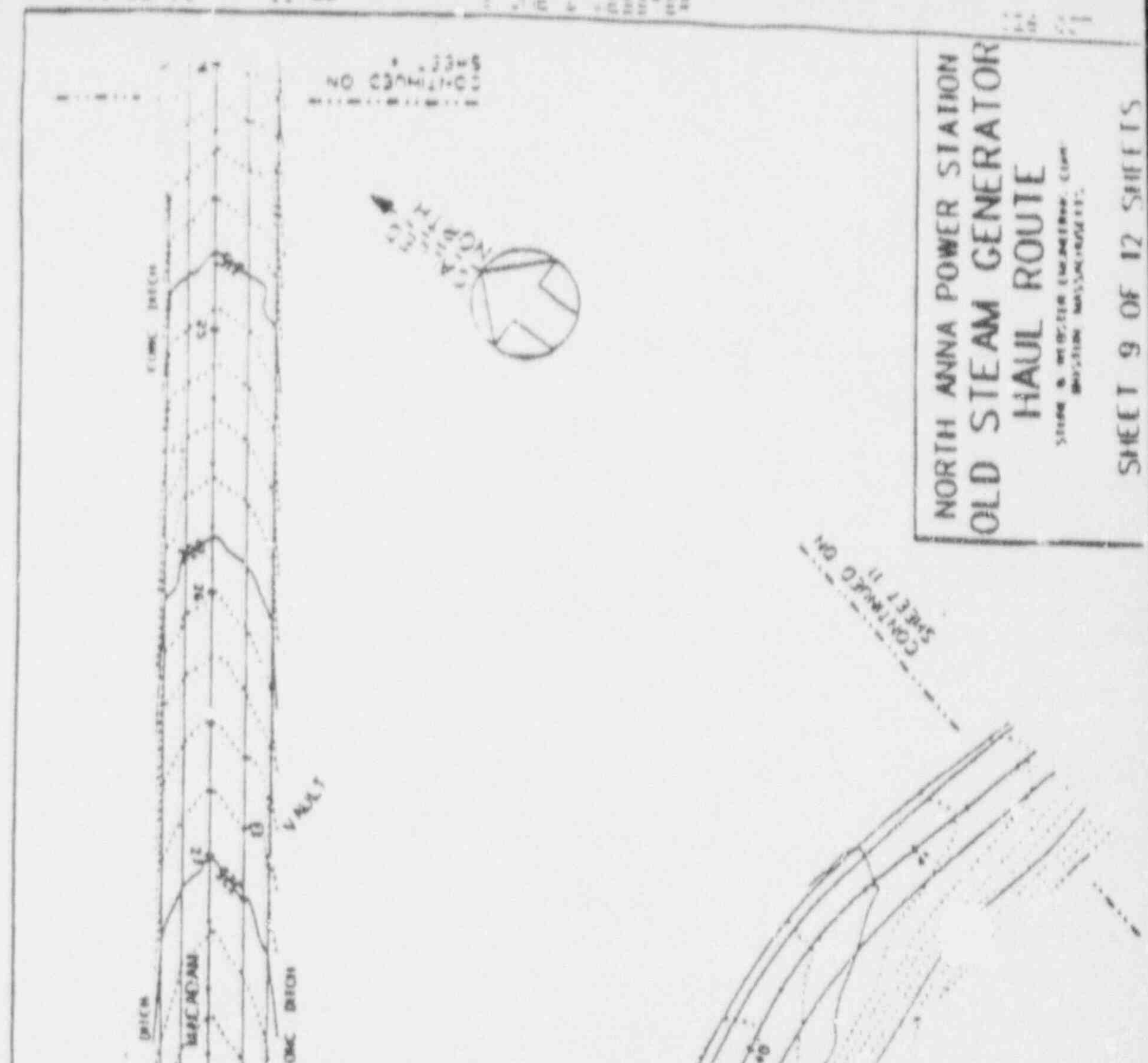

NORTH ANNA POWER STATION
OLD STEAM GENERATOR
HAUL ROUTE
SHEET 5 OF 12 SHEETS
CONTINUED ON SHEET 6
CONTINUED ON SHEET 4
CALLED NORTH
PAVED SHLD

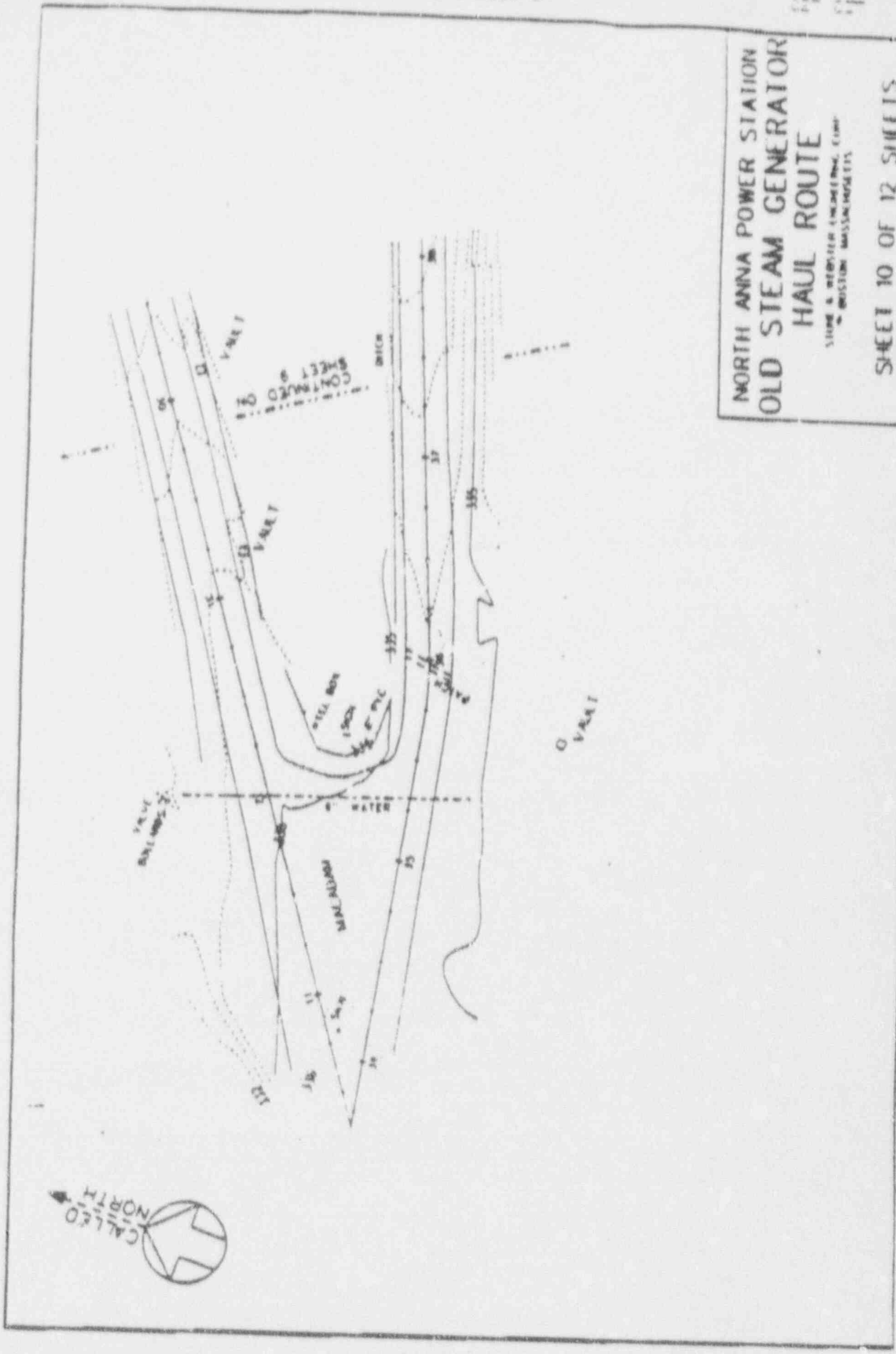

NORTH ANNA POWER STATION
OLD STEAM GENERATOR
HAUL ROUTE
CALLED NORTH
CONTINUED ON SHEET 5
CONTINUED ON SHEET 12
GATE OPERATOR
2-3" PVC CONDUITS
WATER VALVE VAULT
TEL VAULT
POSTS
TEL MARKER
BOLLARD
HYD
VAULT
BOLLARDS
CONC PAD
CONC PAD
CONC DITCH
PAVED ROAD
SIGN POSTS
CONTINUED ON SHEET 7

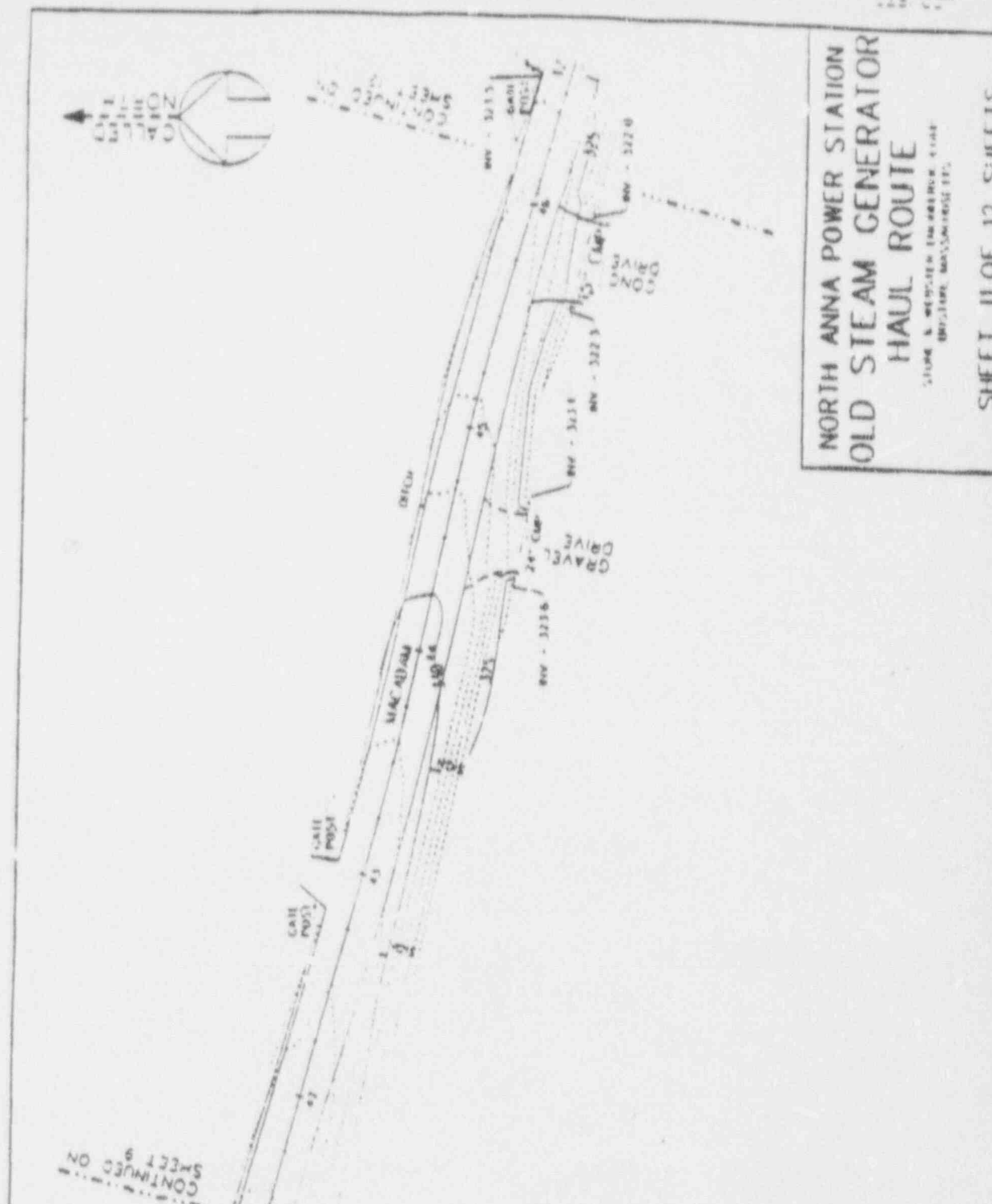

NORTH ANNA POWER STATION
OLD STEAM GENERATOR
HAUL ROUTE
CONTINUED ON SHEET 6
CALLED NORTH
ELEC VAULT
GRAVEL ROAD
DITCH
SHED
CONTINUED ON SHEET 8


NORTH ANNA POWER STATION
OLD STEAM GENERATOR
HAUL ROUTE
STEAM GENERATOR/
TRAILER RIG
ENVELOPE (TYP)
FOUNDATION MAT
EAST STORAGE
CONC. PAD
PROPOSED INTERIM
OLD STEAM GENERATOR
STORAGE FACILITY
UNIT 1 OLD
STEAM GENERATOR
(TYP)
CONTINUED ON SHEET 7
RAILROAD TRACKS
CALLED NORTH
FENCE
DITCH
GATE

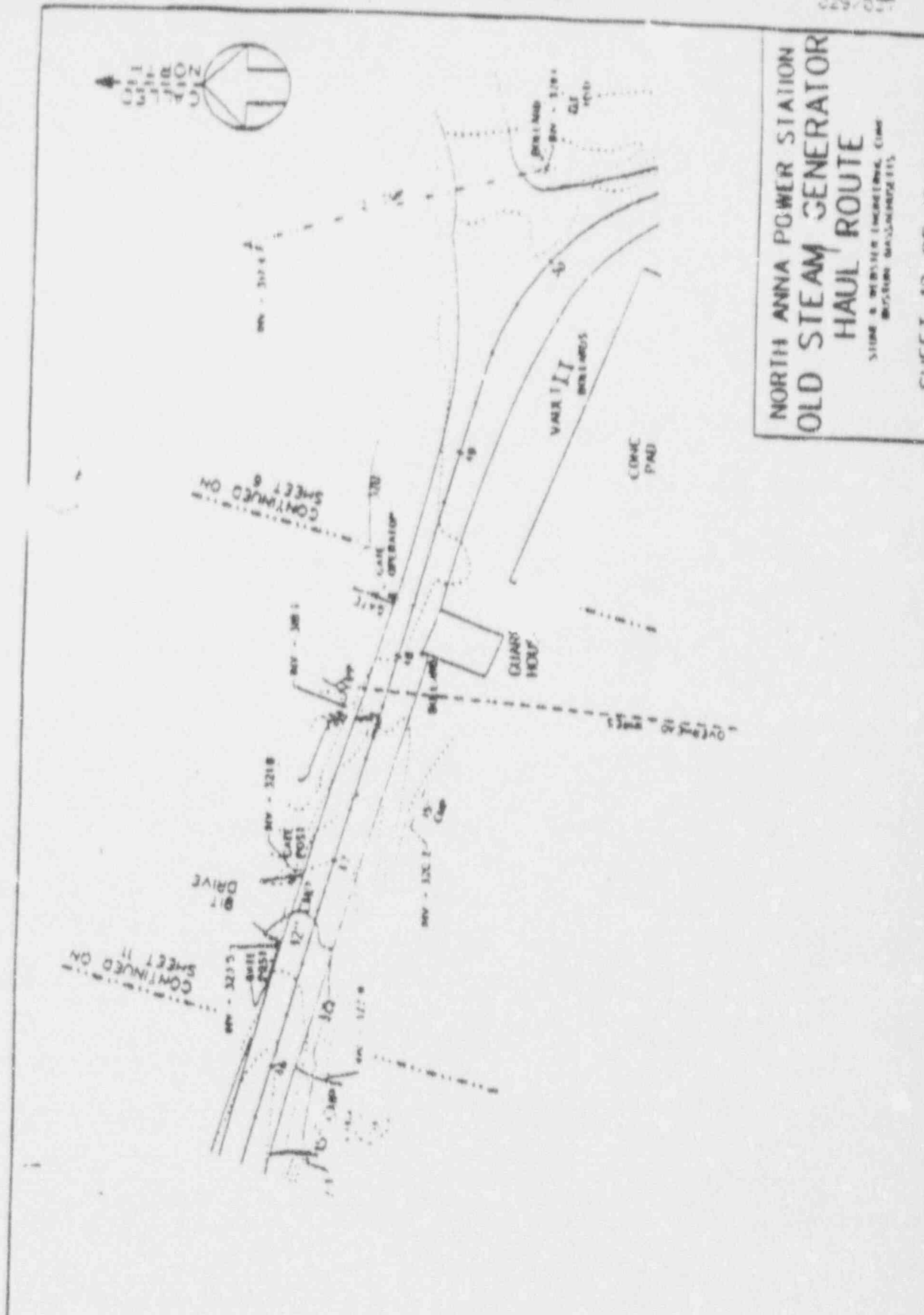
A-10


NORTH ANNA POWER STATION
OLD STEAM GENERATOR
HAUL ROUTE
SHEET 9 OF 12 SHEETS
CONTINUED ON SHEET 11
CONTINUED ON SHEET 10
VAULT
CONC DITCH


NORTH ANNA POWER STATION
OLD STEAM GENERATOR
HAUL ROUTE
SHEET 10 OF 12 SHEETS
CONTINUED ON SHEET 9
CALLED NORTH
MACADAM
WATER


NORTH ANNA POWER STATION
OLD STEAM GENERATOR
HAUL ROUTE
SHEET 11 OF 12 SHEETS
CALLED NORTH
CONTINUED ON SHEET 12
CONTINUED ON SHEET 9
CONC DRIVE
GRAVEL DRIVE
GATE POST
GATE POST
MACADAM
325
325

A-13


NORTH ANNA POWER STATION
OLD STEAM GENERATOR HAUL ROUTE
CONTINUED ON SHEET 6
CONTINUED ON SHEET 11
BIT DRIVE
CONC PAD

A-14

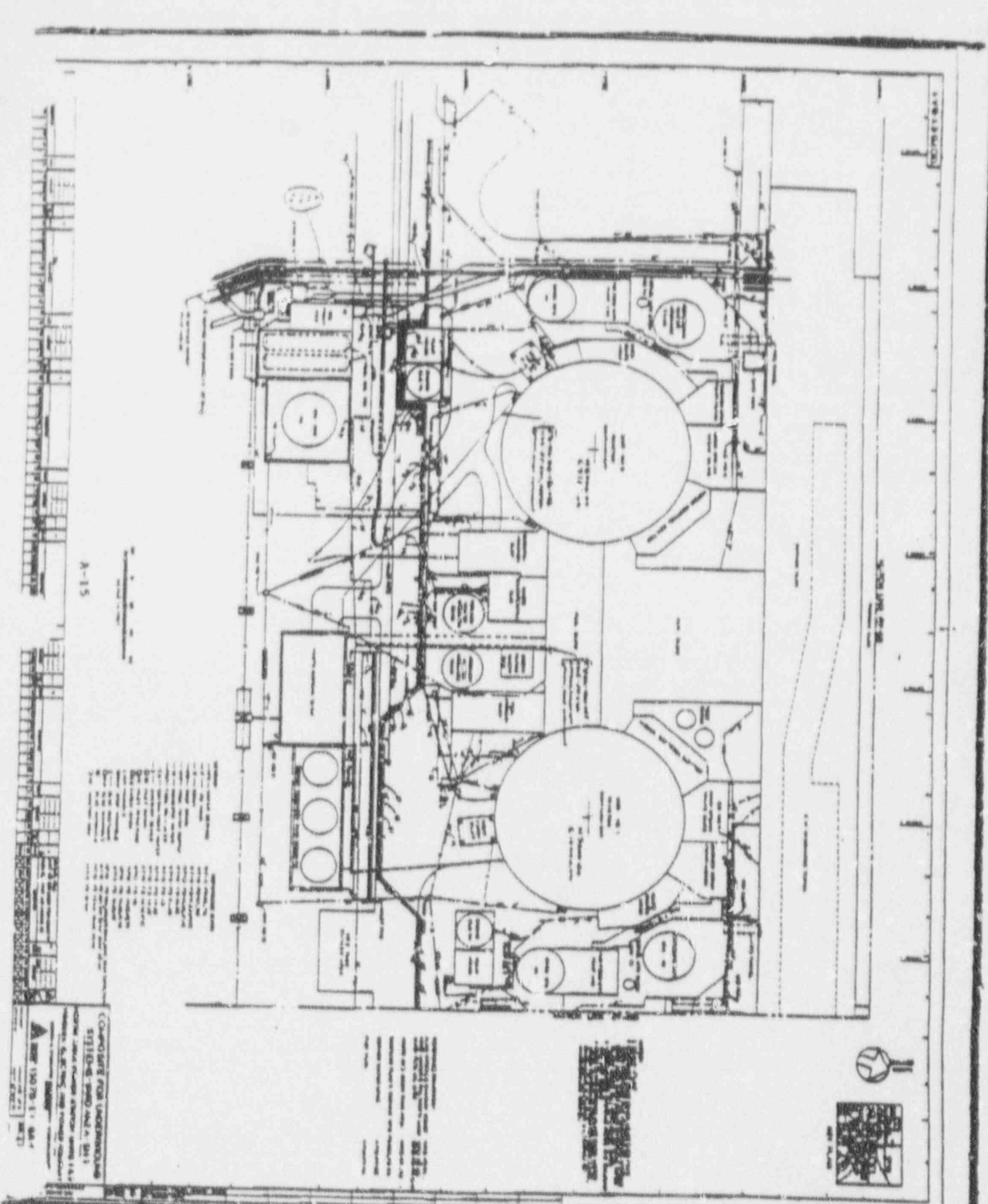


TABLE A

BURIED UTILITIES IN PROTECTED AREA

ITEM	DESCRIPTION	ELEVATION	COORDINATES	COMMENTS
A	21" Ø Storm Drain MH #3 to MH #13	@ MH #3 T.O.P. El 267.47' @ MH #13 T.O.P. El 266.36'	@ MH #3: *N4905.08', E9973.40' @ MH #13: N4886.50', E10066.90'	*/Approximate Location
B	6" Ø Fire Main - Runs E/W	T.O.P. El 261.93'	N4895.75'	
B_1	6" Ø Fire Main - Runs N/S	@ Hyd T.O.P. El 262.93'	E9950.16'	
C	12" Ø Fire Main - Runs E/W	East End T.O.P. El 261.59' West End T.O.P. El 264.97'	N4895.75'	
D	12" Ø Fire Main - Runs E/W	T.O.P. El 264.97'	N4855.91'	Notes: 1. Section of this pipe offsets under pipe tunnel T.O.P. El 259.47'. 2. Offsets down @ approx. E9800
D_1	12" Ø Fire Main - Runs SE/NW	T.O.P. El 264.97'	(SE) N4851.15' E9929.49') actual (NW) N4876.45' E9908.90')	
D_2	12" Ø Fire Main - Runs E/W	T.O.P. El 264.97'	N4882.91' (Note 1)	
D_3	12" Ø Fire Main - Runs E/W	T.O.P. El 259.55'	N4882.91' (Note 2)	
D_4	12" Ø Fire Main - Runs N/S	T.O.P. El 264.97'	E9696.00'	
D_5	12" Ø Fire Main - Runs E/N	T.O.P. El 264.97'	N4857.00') verified as N4855.9' E9687.9', N4856.4', E9637.1'	
E	4" Ø Domestic Water - Runs E/W	T.O.P. El 265.03'	N4852.33'	Notes: 1. Same as above except T.O.P. El 259.10' 2. Same as above
E_1	4" Ø Domestic Water - Runs SE/NW	T.O.P. El 265.03'	(SE) N4852.33' E9930.17' (NW) N4881.33' E9901.17'	
E_2	4" Ø Domestic Water - Runs E/W	T.O.P. El 265.03'	N4881.33' (Note 1)	
E_3	4" Ø Domestic Water - Runs E/W	T.O.P. El 265.20'	N4881.33' (Note 2)	
E_4	4" Ø Domestic Water - Runs N/S	T.O.P. El 259.03'	E9697.50'	
E_5	4" Ø Domestic Water - Runs E/W	T.O.P. El 265.03'	N4855.50'	

ITEM	DESCRIPTION	ELEVATION	COORDINATES	COMMENTS
F	2" Ø Domestic Water - Runs E/W	T.O.P. El 265.18'	N4851.08'	Notes: 1 Same as above except T.O.P. El 259.02'. 2. Same as above
F_1	2" Ø Domestic Water - Runs SE/NW	T.O.P. El 265.18'	(SE) N4851.08" E9931.42' (NW) N4880.08' E9902.42'	
F_2	2" Ø Domestic Water - Runs E/W	T.O.P. El 265.18'	N4880.08' (Note 1)	
F_3	2" Ø Domestic Water - Runs E/W	T.O.P. El 265.10'	N4880.08' (Note 2)	
F_4	2" Ø Domestic Water - Runs N/S	T.O.P. El 259.18'	E9698.50'	
F_5	2" Ø Domestic Water - Runs E/W	T.O.P. El 265.18'	N4854.50'	
G	3" Ø Fuel Oil Fill Line - Runs E/W	T.O.P. El 268.14'	N4835.00'	*Approximate location
G_1	3" Ø Fuel Oil Fill Line - Runs N/S	T.O.P. El 268.14'	*E9877.00'	
G_2	3" Ø Fuel Oil Fill Line - Runs E/W	T.O.P. El 268.14'	N4840.55'	
G_3	3" Ø Fuel Oil Fill Line - Runs N/S	T.O.P. El 266.14'	E9822.00'	
G_4	3" Ø Fuel Oil Fill Line - Runs E/W	T.O.P. El 266.14'	N4825.06'	
G_5	3" Ø Fuel Oil Fill Line - Runs N/S	T.O.P. El 266.14'	E9748.00'	
H	12" Ø Fire Main - Runs N/S	T.O.P. El 265.07'	E9776.50'	
I	6" Ø Fire Main - Runs N/S	T.O.P. El 264.56'	E9771.33'	
J	42" Ø Storm Drain CB #1 to MH #11 - Runs NW/SE	@ CB #1: T.O.P. El 265.12' @ MH #11: T.O.P. El 264.76'	@ CB #1: N4886.50' E9777.41' @ MH #11: N4763.00' E9837.99'	
K	12" Ø Storm Drain From MH #1 to 42" Ø Storm Dr - Runs SW	@ MH #1: T.O.P. El 264.21' @ 42" Ø T.O.P. El 263.34'	@ MH #1: *N4886.50' *E9829.49' @ Offset: *N4867.00' *E9790.07' @ 42" Ø *N4863.00' *E9788.82'	*Approximate location
L	36" Ø Service Water - Runs N/S	T.O.P. El 264.32'	E9587.00'	@ N4851.22' Pipe Offsets Down to T.O.P. El 258.16' @ N4861.00' (Typ for 4 Serv Wtr Lines)
L_1	36" Ø Service Water - Runs N/S	T.O.P. El 264.32'	E9582.00'	
L_2	36" Ø Service Water - Runs N/S	T.O.P. El 264.32'	E9577.00'	
L_3	36" Ø Service Water - Runs N/S	T.O.P. El 264.32'	E9575.00'	

ITEM	DESCRIPTION	ELEVATION	COORDINATES	COMMENTS
M	1 1/2" Ø Fuel Oil Lines 10-Pipes 5" O.S. Encased in Conc - Runs N/S	T.O.P. El 265.16' (Typ)	Pipes Run Between E9614.37' & E9610.62	Confirmed in Field
N	3" Ø Fuel Oil Return - Runs N/S	T.O.P. El 266.14'	E9747.00'	
N_1	3" Ø Fuel Oil Return - Runs N/S	T.O.P. El 266.14'	N4835.00'	
O	3" Ø Fuel Oil Fill Line - Runs N/S	T.O.P. El 266.14'	E9615.00'	*Approximate Location
O_1	3" Ø Fuel Oil Fill Line - Runs E/W	T.O.P. El 266.14'	*N4835.00'	
EDB_1	13" Wide Elect Duct - Runs N/S	T.O.D. El 265.50'	East Edge @ E9629.00'	
EDB_2	13" Wide Elect Duct - Runs N/S	T.O.D. El 265.50'	East Edge @ E9622.08'	
EDB_3	4'-9" Wide Elect Duct - Runs N/S	T.O.D. El 265.50'	East Edge @ E9615.87'	
EDB_4	3'-3" Wide Elect Duct - Runs SW/NE	T.O.D. Slopes H.P. @ No. End El 262.91' L.P. @ So. End El 256.91'	℄ of Duct Bank *No. End: N4896.75' E9912.25' *So. End: N4838.29' E9874.43'	*Approximate Location @ So. End Duct Bank Offsets Due South to Waste Disposal Bldg Expanding to 6'-10" Wide @ Waste Disposal Bldg
EDB_5	22" Wide Elect Duct - Runs N/S	T.O.D. El 268.25'	℄ of Duct Bank *E9879.93'	*Approximate Location Duct Runs North From Waste Disposal Bldg to Duct Bank EDB_4
EDB_6	2'-7" Wide Elect Duct With Hand Hole Enclosure HH-558 - Runs N/S	T.O.D.El 268.25' H.H. Cover El 271.00' Top of Conc Enclosure El 270.00'	℄ of Duct Bank E9594.18' N4829.00' E9594.18'	HH Enclosure is 4'0" x 4'6" x 4'0" DP

APPENDIX B

WALKDOWN REPORT

Station NAPS

J.O. No. 02072.0510

Task Haul Route Evaluation

X First Walkdown
__ Second Walkdown

Purpose of Walkdown

The purpose of this walkdown was to inspect a primary and alternate haul route for transporting the North Anna spent steam generators from the containment building to the interim storage facility.

Participants

Stone & Webster:	Boston Personnel - W. J. Pananos, P. J. Trudeau, R. D. Ciatto
	Richmond Personnel - C. Leggatt + survey party
Virginia Power	D. Price, C. Ranganath

Photographs were taken __ Yes X No

Documents used in walkdown

Survey sketches prepared by S & W
North Anna Yard Drawing 13075-EY-8A-1
Inspection Plan - North Anna Steam Generator Repair Haul Route

Summary of Walkdown

The walkdown started at the Unit 1 containment. Using the sketches prepared by S&W and the yard drawing, we identified the following items along the primary and alternate haul routes for the spent steam generators:

- culverts
- electrical ductbank
- buried piping
- telephone and other communications cable

These items were observed at cable vaults or at locations where they penetrated the ground surface. We followed them to points where the transport vehicle will pass over them on the haul route. A scanner was used to obtain approximate depths of the underground items. However, it was not possible to obtain the depths of some piping.

In the protected area, fuel oil lines, fire protection piping, and electrical ductbanks were reviewed as shown on the yard drawing. Outside the protected area, electrical conduit, air lines, domestic water, fire protection piping and communications cabling were observed. Survey drawings were marked with depths of these items, when this was obtainable, and coordinates were also noted when these were deemed necessary.

No serious road damage along the haul route was found i. e., potholes and serious bumps were not found. Some cracking of blacktop and concrete at certain locations was observed, but none serious enough to impede the transporter. It appeared that soil at some locations may become soft when it is wet, however, gravel covers the haul route at most locations which are not black topped thereby providing a sound base for wheeled vehicles.

The sides of the proposed route were checked for slope stability. No stability problems were determined.

Overhead and side to side clearances were checked. For the alternate route, the survey plot was marked to note that the gate opening is limited to 20.5 feet and the minimum clearance below overhead wires is 19.5 feet near the edge of the road in this area.

The walkdown also addressed the direct buried conduit located in the new steam generator unloading area.

Open Items and Required Follow-up

a. Design drawings of the concrete slab at the fuel cask area will be reviewed and the slab will be evaluated for transporter loading. (Note: Drawings of this slab have not been found. However, Virginia Power personnel identified the slab thickness as 2 feet.)

b. An electrical ductbank, EDB_4, which is noted on the table attached to the inspection plan, will be evaluated for the transporter wheel load since this buried ductbank is only about 3 feet below the surface.

c. The Unit 3 & 4 circulating water discharge tunnel outside the protected area will be evaluated for the transporter wheel load.

d. An effort will be made to obtain information on the buried fire protection piping at the hairpin turn at station 20.5 of the primary route. (Note: Drawings of this piping have not been found.)

e. Superelevations at various locations along the haul route will be prepared and they will be included in the final haul route drawing which will be attached to the final report.

f. The transporter turning and maneuvering requirements will be obtained from Virginia Power.

g. The survey will be extended to include the area up to the fence crossing the RR track near the site of the interim storage facility.

h. The survey will be extended to include contours and underground cable in the area between Station 21 of the primary route and Station 22 of the alternate route. This area may be used as a second alternate route in the event that it is necessary to avoid the hairpin turn and steep grade of the primary route and if the entire alternate route is not available.

Results and Recommendations

A hairpin curve with a steep grade, about 9%, in the primary route from stations 19 to 22 makes the alternate route more favorable. There is also a hairpin curve in the alternate route from stations 31 to 36; however, this curve is on a flat surface. It will be determined if there is adequate room for maneuvering the vehicle at this turn.

If the alternate route is not available, another option exists to avoid the hairpin turn of the primary route. The transporter could exit the primary route at the gate at Station 19. It could then proceed up the hill on the alternate route for only about 200 feet and then reent the primary route at the gate near Station 21. Some grading outside the gate near Station 21 would be necessary.

The gross weight of the steam generator and transport vehicle is expected to be about 560 to 580 kips. This weight is distributed over 80 wheels resulting in an average wheel load of about 7 kips. This compares to a wheel load of about 12.5 kips for the site P&H crane which has a dead weight of 150 kips distributed over 12 wheels. This crane has been moved over most of the primary route several times without damaging underground equipment.

It is expected that power cable and communications lines would not be damaged by heavy wheel loads. Culverts are of some concern because of their close proximity to the surface and their large diameter. Also, piping which is close to the ground surface is a concern due to potential damage and leaks. These items can be protected during the transport operation.

Engineering Design Budget Impacted
As a Result of Walkdown

Yes ___ No ✓

Responsible APE

Distribution: DE McLellan
Walkdown participants
Job Book File
Lead Engineer (of affected disciplines)
Task Engineer
Responsible APE

Preparer

Signature 9/20/91
Date

STONE & WEBSTER ENGINEERING CORPORATION

BOSTON MA
CHATTANOOGA TN
CHERRY HILL NJ
CHICAGO IL
DALLAS TX
DECATUR AL
DENVER CO
FT LAUDERDALE FL
HOUSTON TX

NEW YORK NY
OAK RIDGE TN
PORTLAND ME
PORTLAND OR
RICHLAND WA
RICHMOND VA
PLEASANTON CA
TAMPA FL
WASHINGTON DC

September 25, 1992

J.O.No. 02072.2610
NAS-20,969

Mr. M. U. Oettler
Manager - Steam Generator Replacement Project
Virginia Power
Innsbrook Technical Center
5000 Dominion Boulevard
Glen Allen, VA 23060
ATTN: Mr. R. K. Bayer

HAUL ROAD EVALUATION
STEAM GENERATOR REPAIR PROJECT
NORTH ANNA POWER STATION - UNIT 1

As requested by Mr. R. K. Bayer, Stone & Webster has reviewed the impact of wheel loads from the steam generator prime mover and an end drop of the steam generator off the hauler on the "Report For The Haul Route Inspection and Evaluation" and Safety Evaluation. These documents were transmitted respectively by NAS-20,586 dated December 24, 1991, and NAS-20,433 dated October 4, 1991. The information on the prime mover and end drop was recently telecopied to Stone & Webster and was not available during preparation of the report or safety evaluation.

The results of this review indicated that the conclusions of the report and the 10CFR50.59 safety evaluation for the haul route will not change. Both the report and safety evaluation will have to be revised to reflect these two new issues. Prior to revising these documents, Stone & Webster requests Virginia Power formally transmit the telecopied information.

If we can be of any further help in this matter, please contact us.

D. E. McLellan
Project Engineer

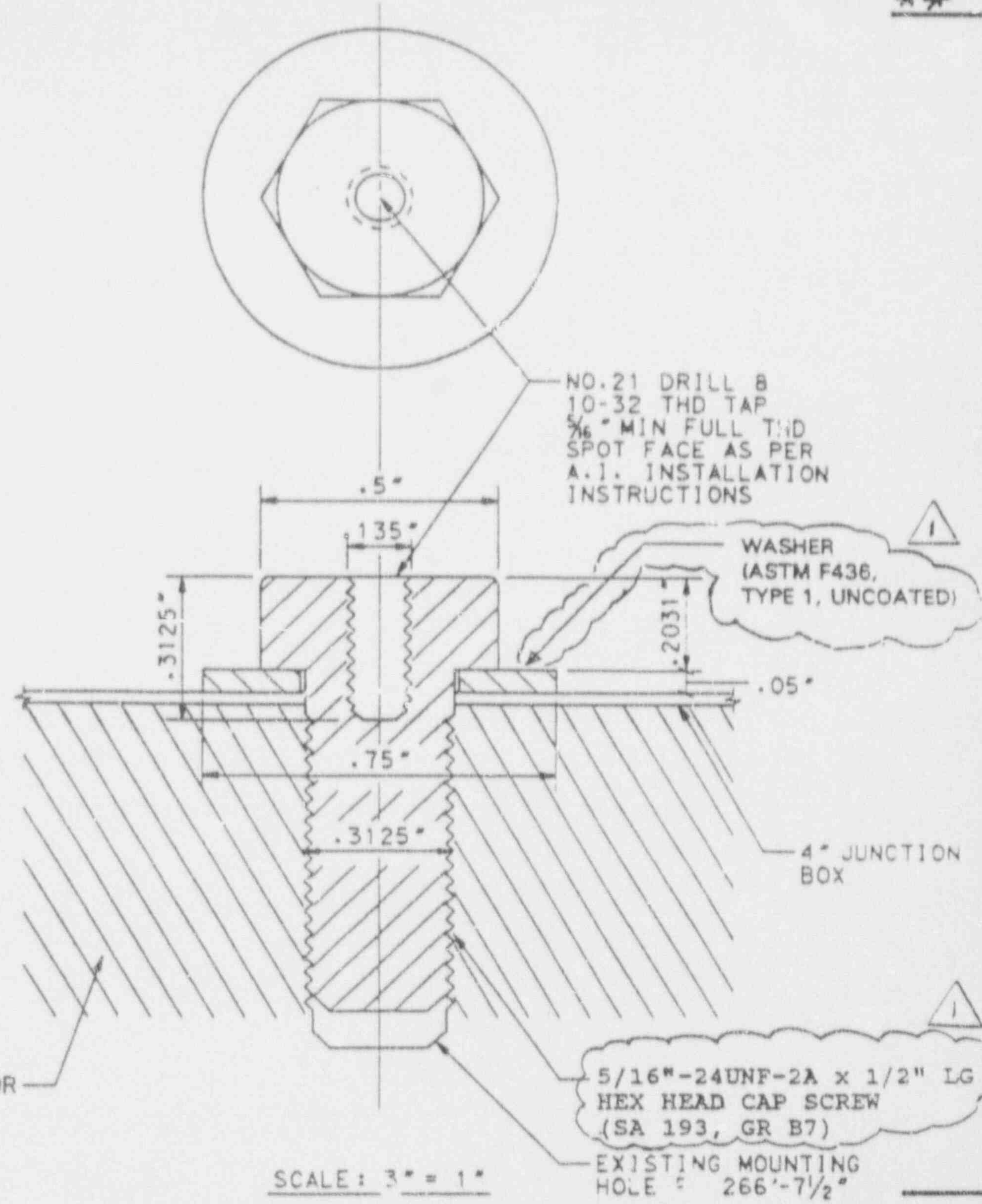
DEM:CM

WESTINGHOUSE SAFETY EVALUATION (SECL-90-113)

PER THE WESTINGHOUSE LETTER, "VIRGINIA POWER REPORT, 70% DRAFT CHANGE PACKAGE 90-13-1 FOR THE STEAM GENERATOR REPLACEMENT FOR NORTH ANNA UNIT 1," MR. N. J. LIPARULO TO MR. J. E. RICHARDSON (NRC), DATED SEPTEMBER 2, 1992, THIS ATTACHMENT HAS BEEN WITHHELD SINCE THERE IS INFORMATION IN THIS ATTACHMENT THAT COULD BE PROPRIETARY TO WESTINGHOUSE.

DCP DRAWINGS

**



COOPERATION SIGN-OFF

Rev. No.	ELECT/CS	MECH.	CIVIL	PLANT DESIGN	LICENSING/NUCLEAR SYS.	CONSTRUCTION
1	[signature]	NR	NR	NR	NR	NR

NON-SAFETY RELATED

** BECHTEL POWER COMPANY PROPRIETARY

BECHTEL POWER CORPORATION
GAITHERSBURG, MARYLAND

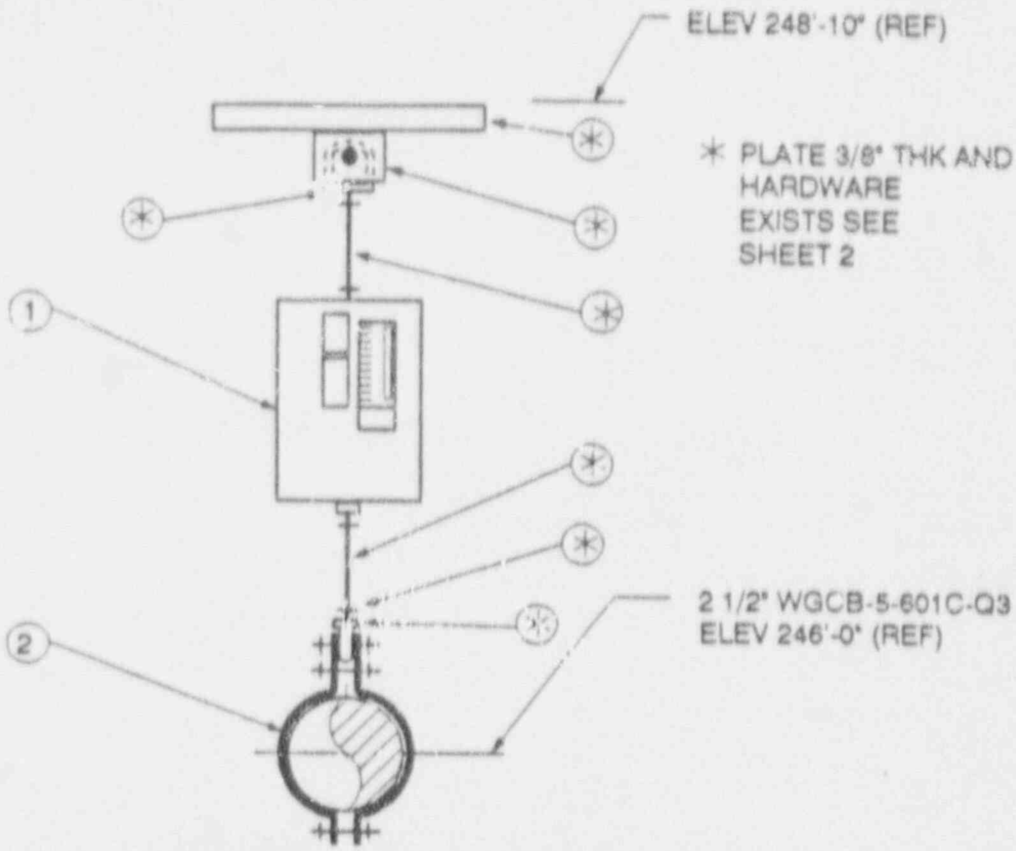
VIRGINIA ELECTRIC AND POWER COMPANY
NORTH ANNA UNIT 1 POWER STATION

STEAM GENERATOR REPAIR PROJECT
LPM - ACCELEROMETER
MOUNTING BOLT

NO.	DATE	REVISIONS	BY	CHK'D	DESIGN SUPV	ENG'R	APPR	APPR
1	7/20/92	REVISED AS NOTED	MOS	MSB	—	[initials]	[initials]	—
0	10/4/91	ISSUED FOR USE	MUF	[initials]	—	RFY	[initials]	—

JOB NO.	DRAWING NO.	REV
20559	N-9013-1-E-3500	1

REF DFN= DFN= 8½ x 11 "A" S



ELEV LOOKING NORTH WEST

NOTES:

1. FOR SUPPORT LOCATION SEE DWG. N9013-1-M-601
2. FOR GENERAL NOTES REFER TO DWG. N9013-1-M-700.01

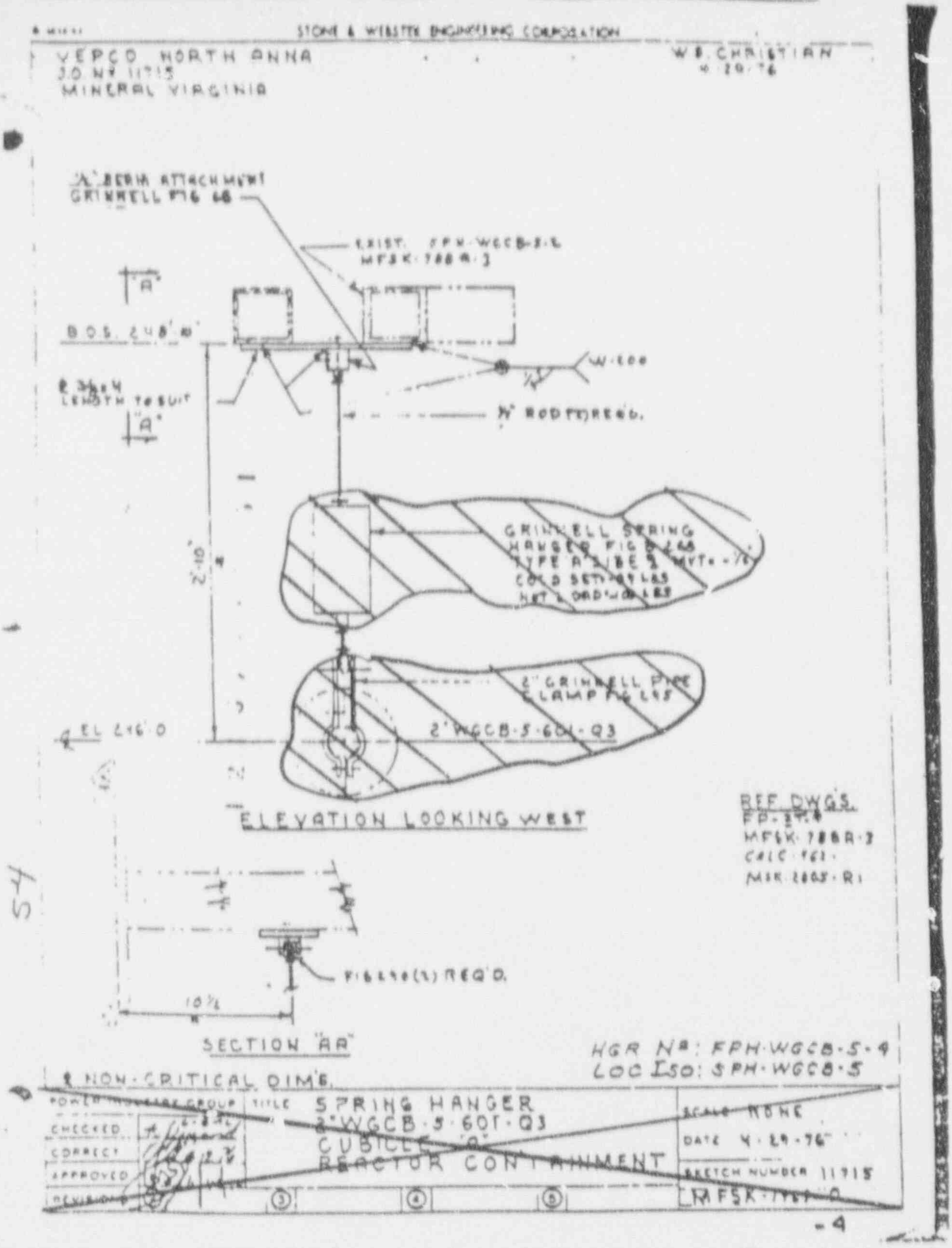
BILL OF MATERIAL

ITEM	QTY	DESCRIPTION	MAT'L
1	1	SIZE 00 GRINNELL FIG B-268 TYPE A VARIABLE SPRING	———
		HANGER HL = 47 # CL = 51 #	
2	1	2 1/2" DOUBLE BOLT PIPE CLAMP, GRINNELL FIG 295	———

ISSUED FOR CONSTRUCTION OF DC-90-13-1

SAFETY RELATED
DC 90-13-1

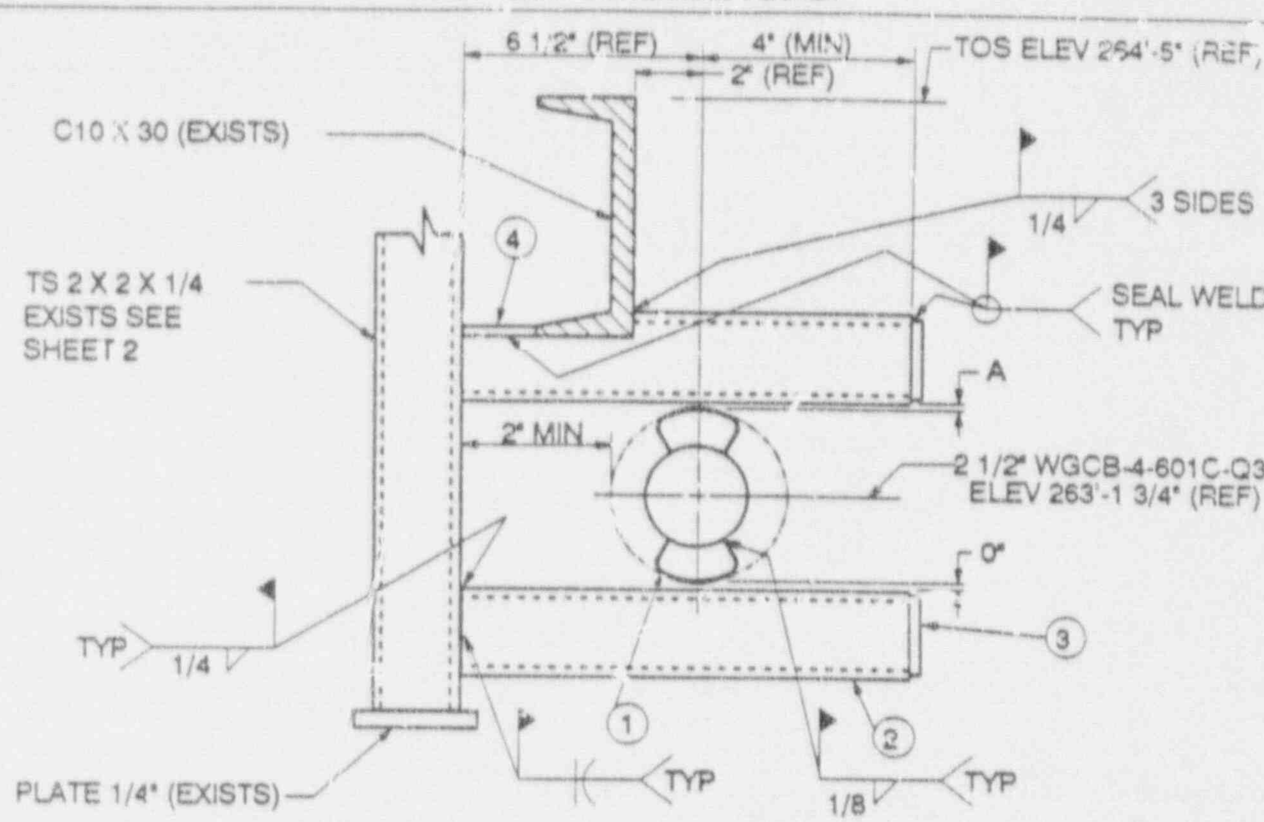
	BY	DATE	TITLE	
PREPARED	KCK	5/4/92	STEAM GENERATOR BLOWDOWN MODIFICATIONS CUBICLE A PIPE SUPPORT FPH-WGCB-5-4	CLIENT VIRGINIA POWER
CHECKED	RAB	5/4/92		PROJECT NORTH ANNA 1
APPROVED	DGDM	5/4/92		J.O. NO. 02072.0710
				ISSUE BY SWEC
				DATE
REVISIONS BY	0	0	0 0 0	NUMBER SHEET 1 OF 2
DATE				N9013-1-M-1MFSK1969-0



ISSUED FOR CONSTRUCTION OF DC-90-13-1

SAFETY RELATED
DC 90-13-1

	BY	DATE	TITLE			CLIENT	VIRGINIA POWER
			STEAM GENERATOR BLOWDOWN MODIFICATIONS			PROJECT	NORTH ANNA 1
PREPARED	DDM	4/17/92	CUBICLE A			J.O. NO.	02072.0710
CHECKED	RAB	4/17/92				ISSUE BY	SWEC
APPROVED	DDM	4/17/92	PIPE SUPPORT FPH-WGCB-5-4			DATE	
REVISIONS BY	0	0	0	0	0	NUMBER	SHEET 2 OF 2
DATE							N9013-1-M-1MFSK1969-0



ELEV LOOKING SOUTH EAST

NOTES:

1. FOR SUPPORT LOCATION SEE DWG. 1-M-601
2. FOR GENERAL NOTES REFER TO DWG N9013-1-M-700.01
3. FIELD TO REMOVE EXISTING PIPE ATTACHMENT AND TUBE STEEL FRAME
4. CLEARANCES 1/16" ≤ (A) ≤ 5/32", FIELD SHIM TO SUIT.

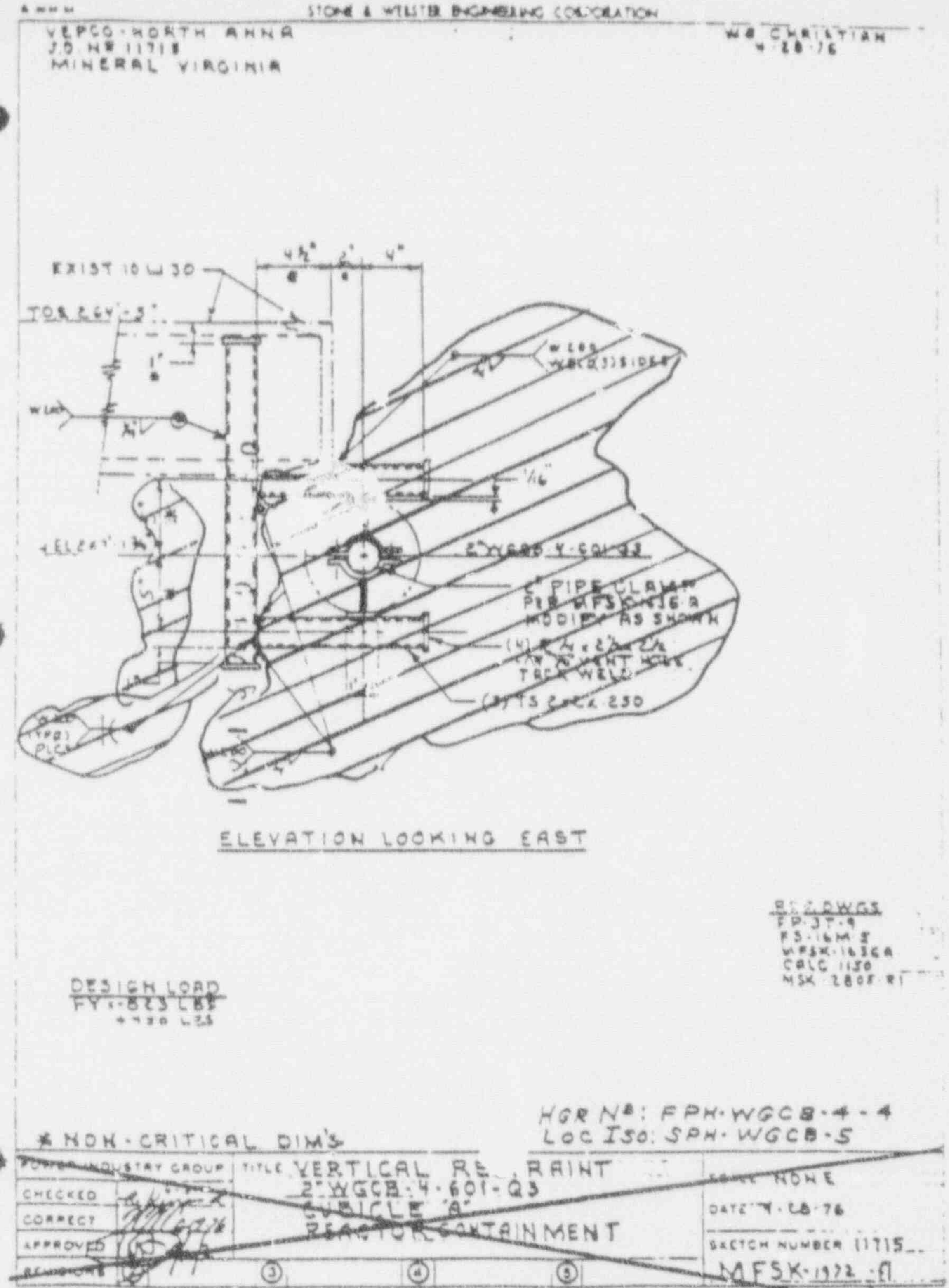
BILL OF MATERIAL

ITEM	QTY	DESCRIPTION	MAT'L
1	2	2 1/2" PROTECTION SADDLE FOR 2" INS, GRINNELL FIG 162	———
2	2	TS 2 X 2 X 1/4" (LENGTH TO SUIT BY FIELD)	A500 GR. B
3	2	STEEL STRIP 1/8" X 1 3/4" X 1 3/4" W/ 1/4" VENT HOLE	A606
4	1	STEEL STRIP 1/8" X 1 3/4" (LENGTH BY FIELD)	A606

ISSUED FOR CONSTRUCTION OF DC-90-13-1

SAFETY RELATED
DC 90-13-1

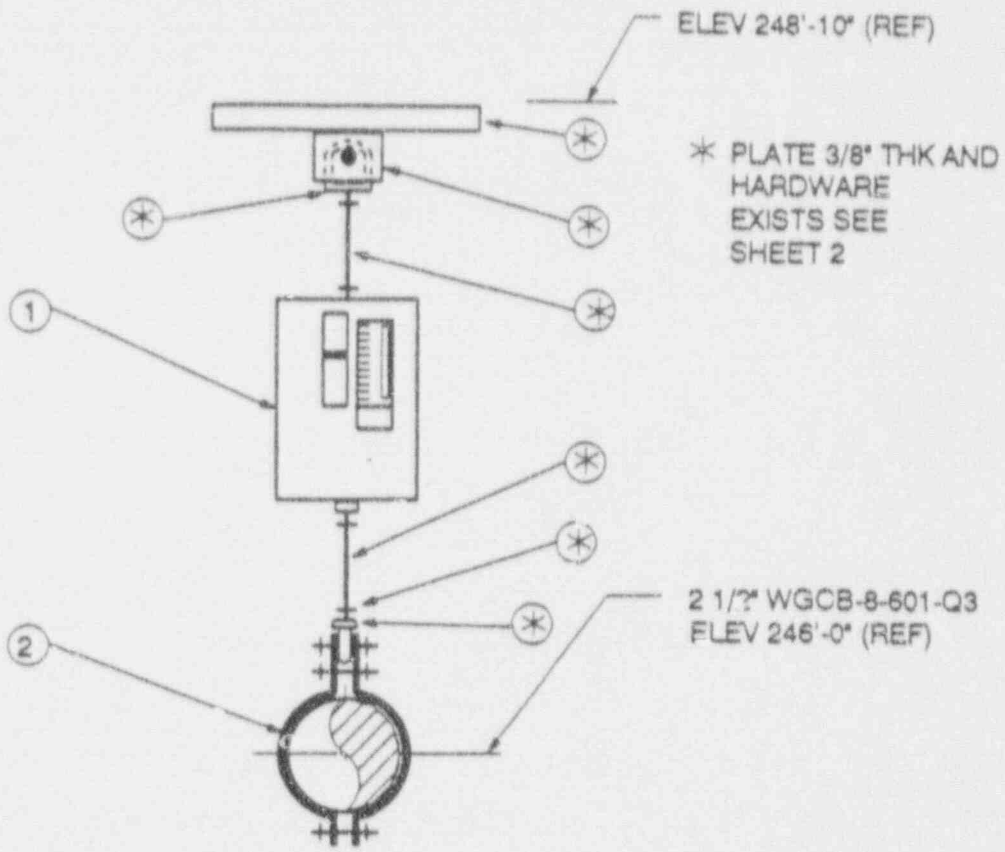
	BY	DATE	TITLE	
PREPARED	DOM	9/17/92	STEAM GENERATOR BLOWDOWN MODIFICATIONS CUBICLE A PIPE SUPPORT FPH-WGCB-4-4	CLIENT VIRGINIA POWER
CHECKED	DB	9/17/92		PROJECT NORTH ANNA 1
APPROVED		9/17/92		J.O. NO 02072.0710
				ISSUE BY SWEC
				DATE
REVISIONS BY				NUMBER SHEET 1 OF 2
DATE				N9013-1-M-1MFSK1972-1



ISSUED FOR CONSTRUCTION OF DC-90-13-1

SAFETY RELATED
DC 90-13-1

	BY	DATE	TITLE			CLIENT VIRGINIA POWER
PREPARED	DDA	9/17/92	STEAM GENERATOR BLOWDOWN MODIFICATIONS CUBICLE A PIPE SUPPORT FPH-WGCB-4-4			PROJECT NORTH ANNA 1
CHECKED	RAB	9/17/92				J.O. NO. 02072.0710
APPROVED		9/17/92				ISSUE BY SWEC
						DATE
REVISIONS BY	0	1	2	3	4	NUMBER SHEET 2 OF 2
DATE						N9013-1-M-1MFSK 1972-1



ELEV LOOKING SOUTH

NOTES:

1. FOR SUPP⌒RT LOCATION SEE DWG. N9013-1-M-602
2. FOR GENE⌐AL NOTES REFER TO DWG. N9013-1-M-700.01

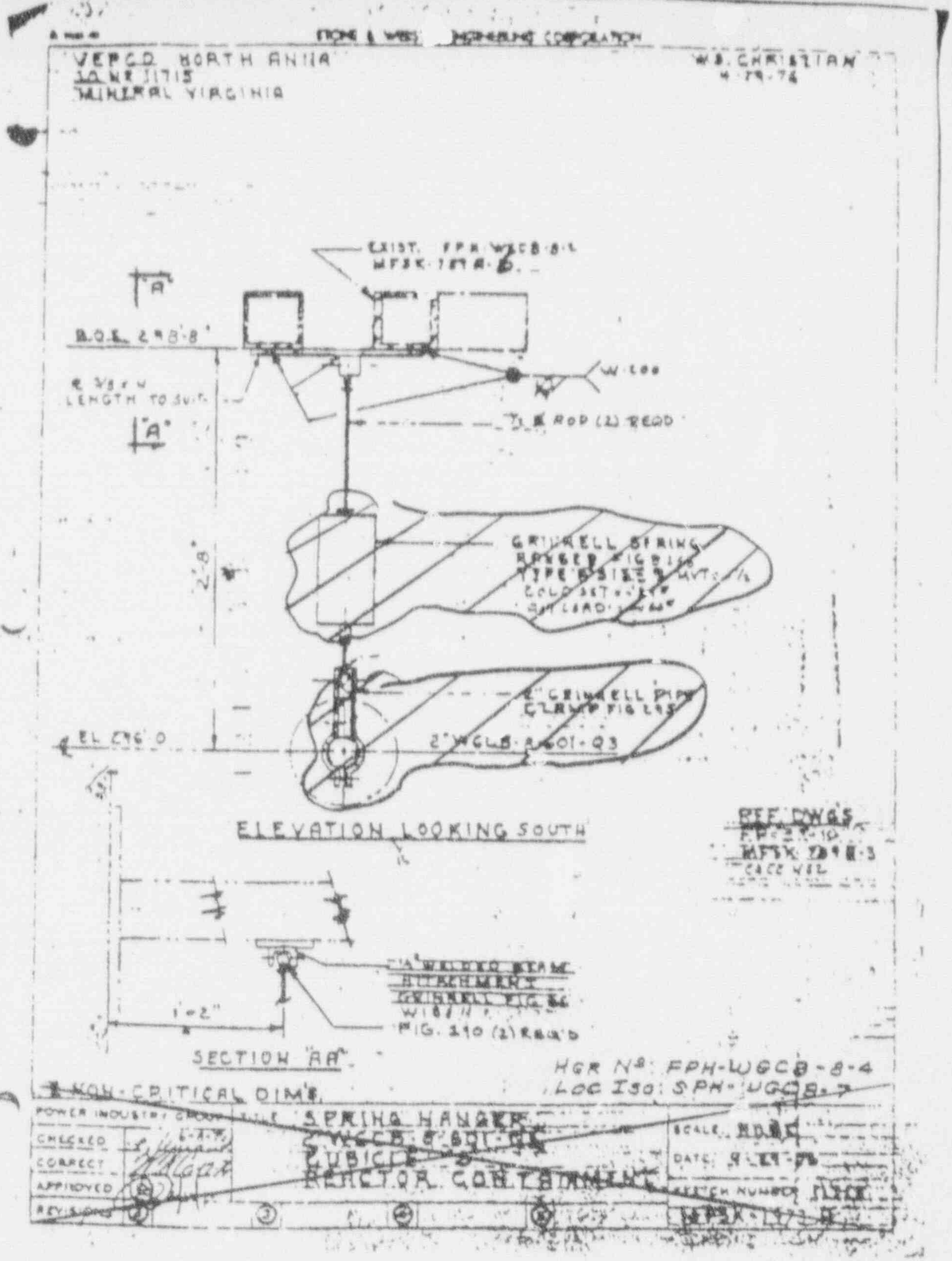
BILL OF MATERIAL

ITEM	QTY	DESCRIPTION	MAT'L
1	1	SIZE 00 GRINNELL FIG B-268 TYPE A VARIABLE SPRING	———
		HANGER HL = 43 # CL = 37 #	
2	1	2 1/2" DOUBLE BOLT PIPE CLAMP, GRINNELL FIG 295	———

SAFETY RELATED
DC 90-13-1

ISSUEL FOR REVIEW OF DC-90-13-1

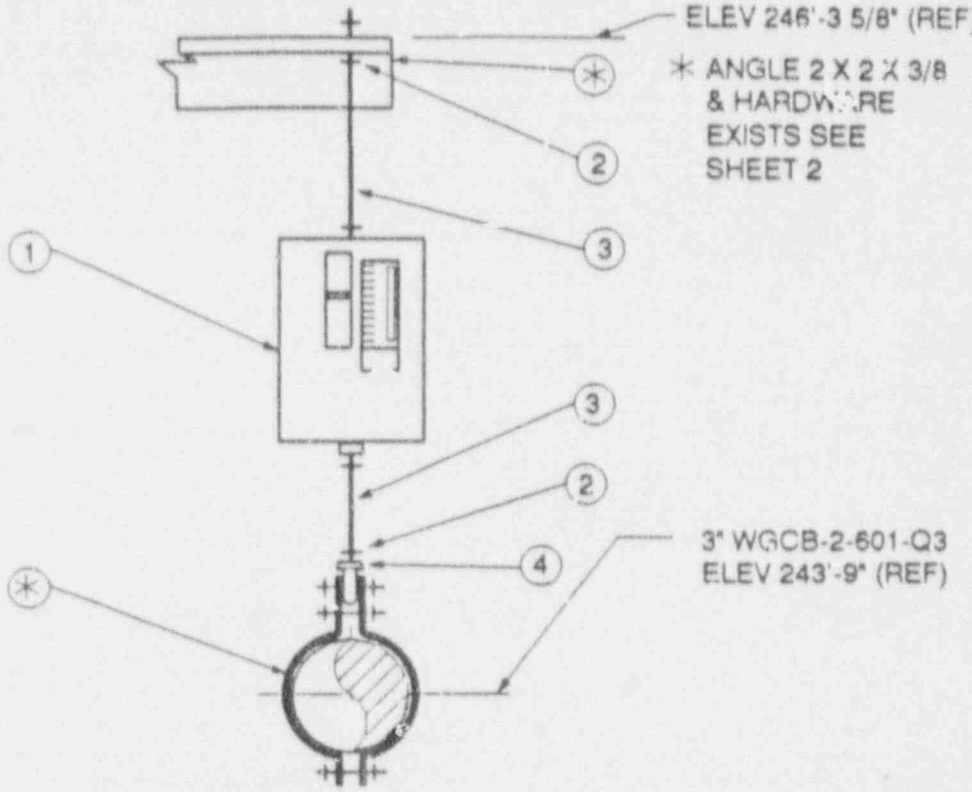
	BY	DATE	TITLE	
PREPARED	DCH	4/17/92	STEAM GENERATOR BLOWDOWN MODIFICATIONS CUBICLE B PIPE SUPPORT FPH-WGCB-8-4	CLIENT VIRGINIA POWER
CHECKED	RAB	4/17/92		PROJECT NORTH ANNA 1
APPROVED	GBM	4/17/92		J.O. NO. 02072.0710
				ISSUE BY SWEC
				DATE
REVISIONS BY	0	0	0 0 0	NUMBER SHEET 1 OF 2
DATE				N9013-1-M-1MFSK1-8-4 O



ISSUED FOR CONSTRUCTION OF DC-90-13-1

SAFETY RELATED
DC 90-13-1

	BY	DATE	TITLE	
PREPARED	DOK	4/17/92	STEAM GENERATOR BLOWDOWN MODIFICATIONS CUBICLE B PIPE SUPPORT FPH - WGCB-8-4	CLIENT VIRGINIA POWER
CHECKED	RAB	4/17/92		PROJECT NORTH ANNA 1
APPROVED	DGP	4/17/92		J.O. NO. 02072.0710
				ISSUE BY SWEC
				DATE
REVISIONS BY	0	0	0	NUMBER SHEET 2 OF 2
DATE				N9013-1-M-1MFSK1973-0



ELEV LOOKING WEST

NOTES:

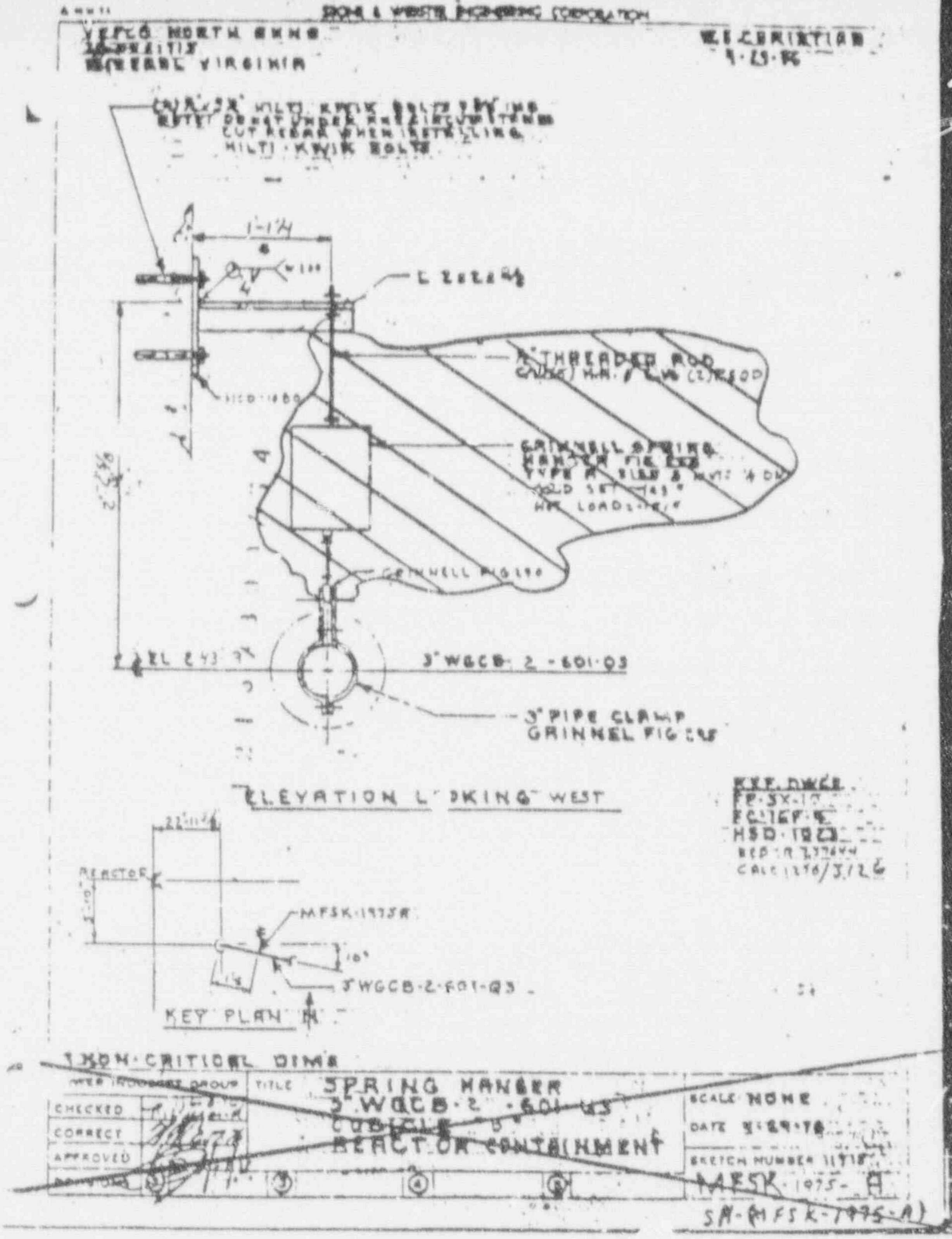
1. FOR SUPPORT LOCATION SEE DWG. N9013-1-M-602
2. FOR GENERAL NOTES REFER TO DWG. N9013-1-M-700.01

BILL OF MATERIAL			
ITEM	QTY	DESCRIPTION	MAT'L
1	1	SIZE 6 GRINNELL FIG B-268 TYPE A VARIABLE SPRING	———
		HANGER HL = 381 # CL = 373 #	
2	5	5/8" HEX NUT	A194 GR 2H
3	2	5/8" GRINNELL FIG 146 ALL THREAD ROD X 0'-9" LONG	———
4	1	5/8" GRINNELL FIG 290 WELDLESS EYENUT	———

SAFETY RELATED
DC 90-13-1

ISSUED FOR CONSTRUCTION OF DC-90-13-1

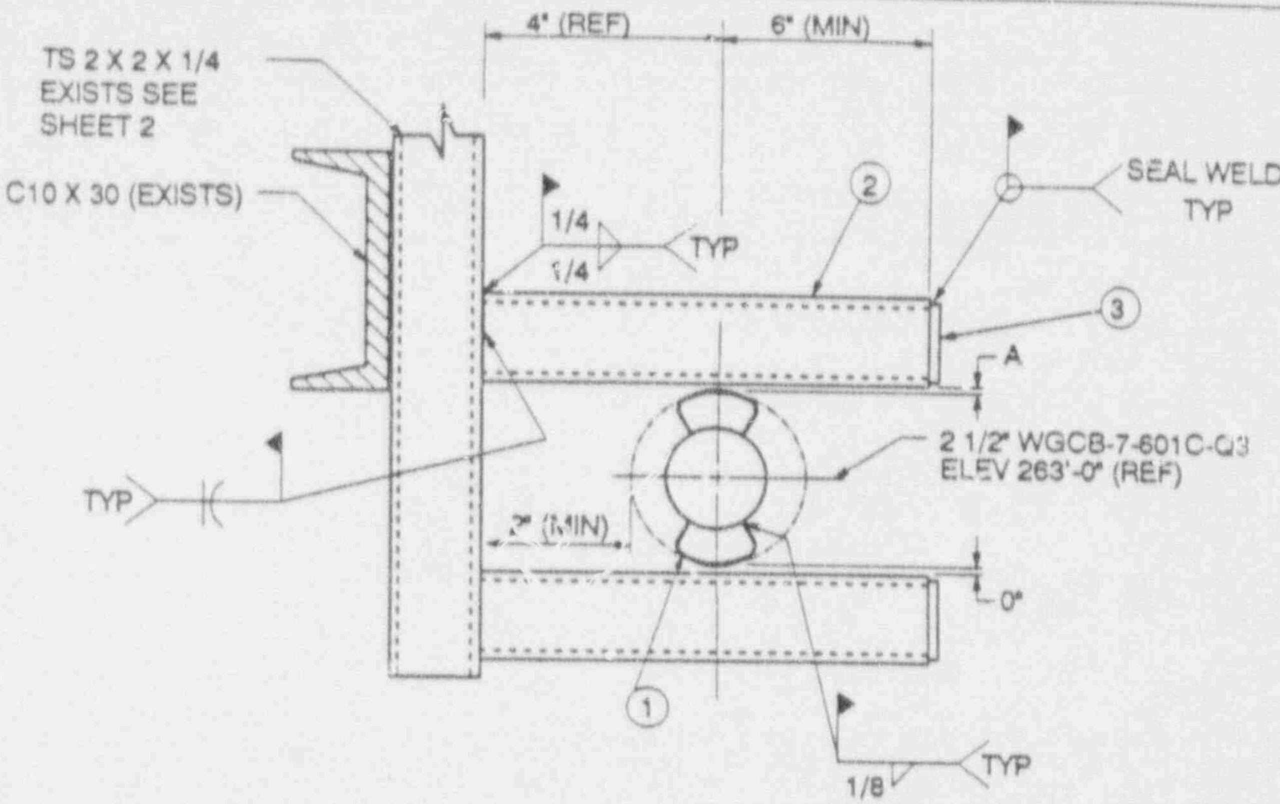
	BY	DATE	TITLE		
PREPARED	DDK	5/4/92	STEAM GENERATOR BLOWDOWN MODIFICATIONS CUBICLE B PIPE SUPPORT 1-WGCB-SH-MFSK-1975A	CLIENT	VIRGINIA POWER
CHECKED	RAB	5/ /92		PROJECT	NORTH ANNA 1
APPROVED	DEM	5/4/92		J.O. NO.	02072.0710
				ISSUE BY	SWEC
				DATE	
REVISIONS BY	0			NUMBER	SHEET 1 OF 2
DATE					N9013-1-M-1MFSK1975A 0



ISSUED FOR CONSTRUCTION OF DC-90-13-1

AFETY RELATED
DC 90-13-1

	BY	DATE	TITLE	
PREPARED	HRM	4/7/92	STEAM GENERATOR BLOWDOWN MODIFICATIONS CUBICLE B PIPE SUPPORT 1-WGCB-SH-MFSK-1975A	CLIENT VIRGINIA POWER
CHECKED	RDB	4/17/92		PROJECT NORTH ANNA 1
APPROVED	DSW	4/17/92		J.O. NO. 02072.0710
REVISIONS BY	1	2 3 4	5	ISSUE BY SWEC
DATE				DATE
				NUMBER SHEET 2 OF 2 N9013-1-M-1MFSK1975A-0



ELEV LOOKING NORTH

NOTES:

1. FOR SUPPORT LOCATION SEE DWG. N9013-1-M-602
2. FOR GENERAL NOTES REFER TO DWG. N9013-1-M-700.01
3. FIELD TO REMOVE EXISTING PIPE ATTACHMENT AND TUBE STEEL FRAME
4. CLEARANCES 1/16" $\leq$ (A) $\leq$ 5/32", FIELD SHIM TO SUIT

BILL OF MATERIAL

ITEM	QTY	DESCRIPTION	MAT'L
1	2	2 1/2" PROTECTION SADDLE FOR 2" INS, GRINNELL FIG 162	————
2	2	TS 2 X 2 X 1/4" (LENGTH TO SUIT BY FIELD)	A500 GR. B
3	2	STEEL STRIP 1/8" X 2 3/4" X 2 3/4" W/ 1/4" VENT HOLE	A606

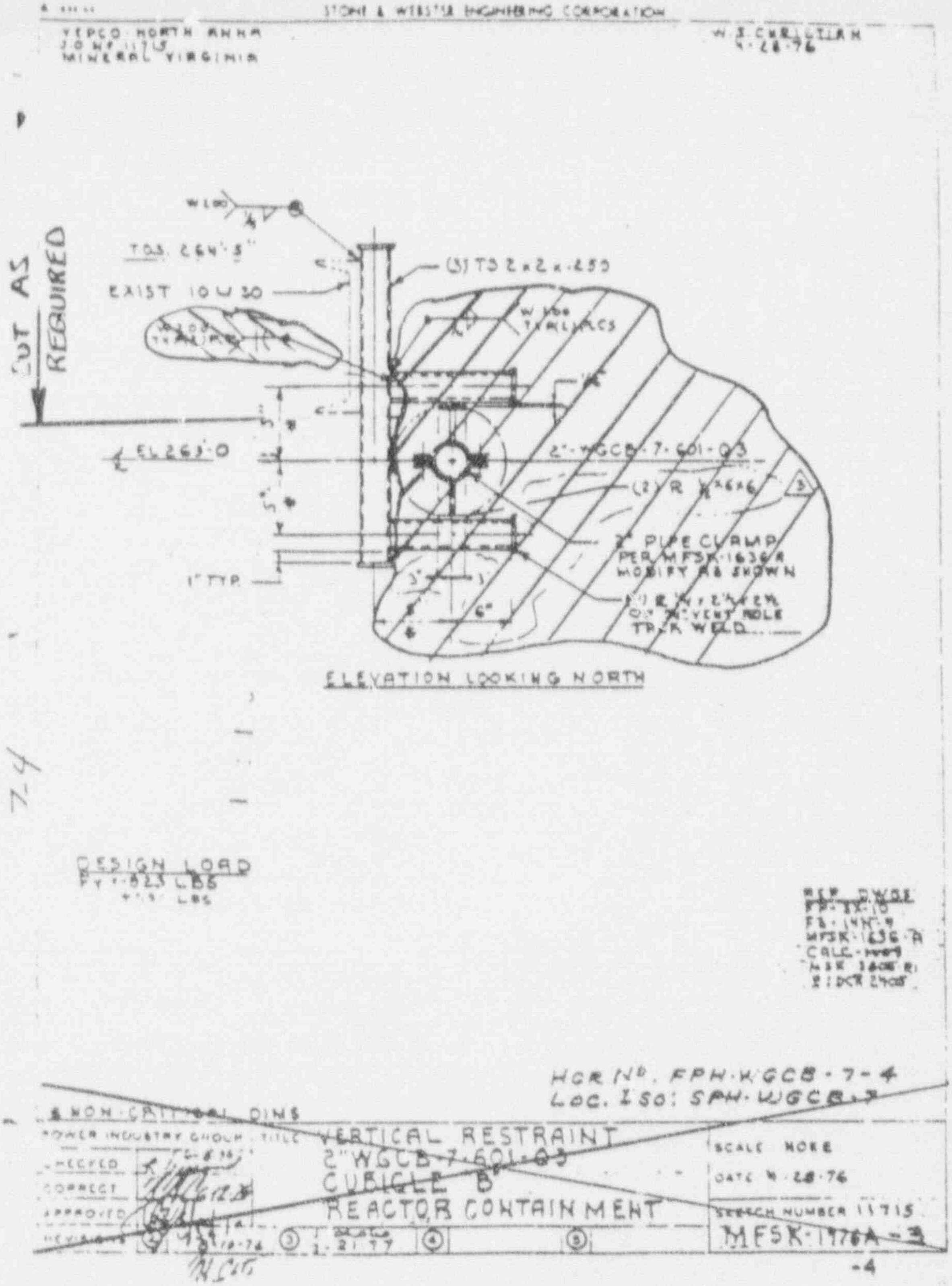
ISSUED FOR CONSTRUCTION OF DC-90-13-1

SAFETY RELATED
DC 90-13-1

	BY	DATE	TITLE	
PREPARED	DOM	9/17/92	STEAM GENERATOR BLOWDOWN MODIFICATIONS CUBICLE B PIPE SUPPORT FPH-WGCB-7-4	CLIENT VIRGINIA POWER
CHECKED	BB	9/17/92		PROJECT NORTH ANNA 1
APPROVED	[illegible]	9/17/92		J.O. NO. 02072.0710
				ISSUE BY SWEC
				DATE
REVISION BY	0	0	0 0 0	NUMBER SHEET 1 OF 2
DATE				N9013-1-M-1MFSK1976A- 1

STONE & WEBSTER ENGINEERING CORPORATION

VEPCO NORTH ANNA
J.O. NO 11715
MINERAL VIRGINIA

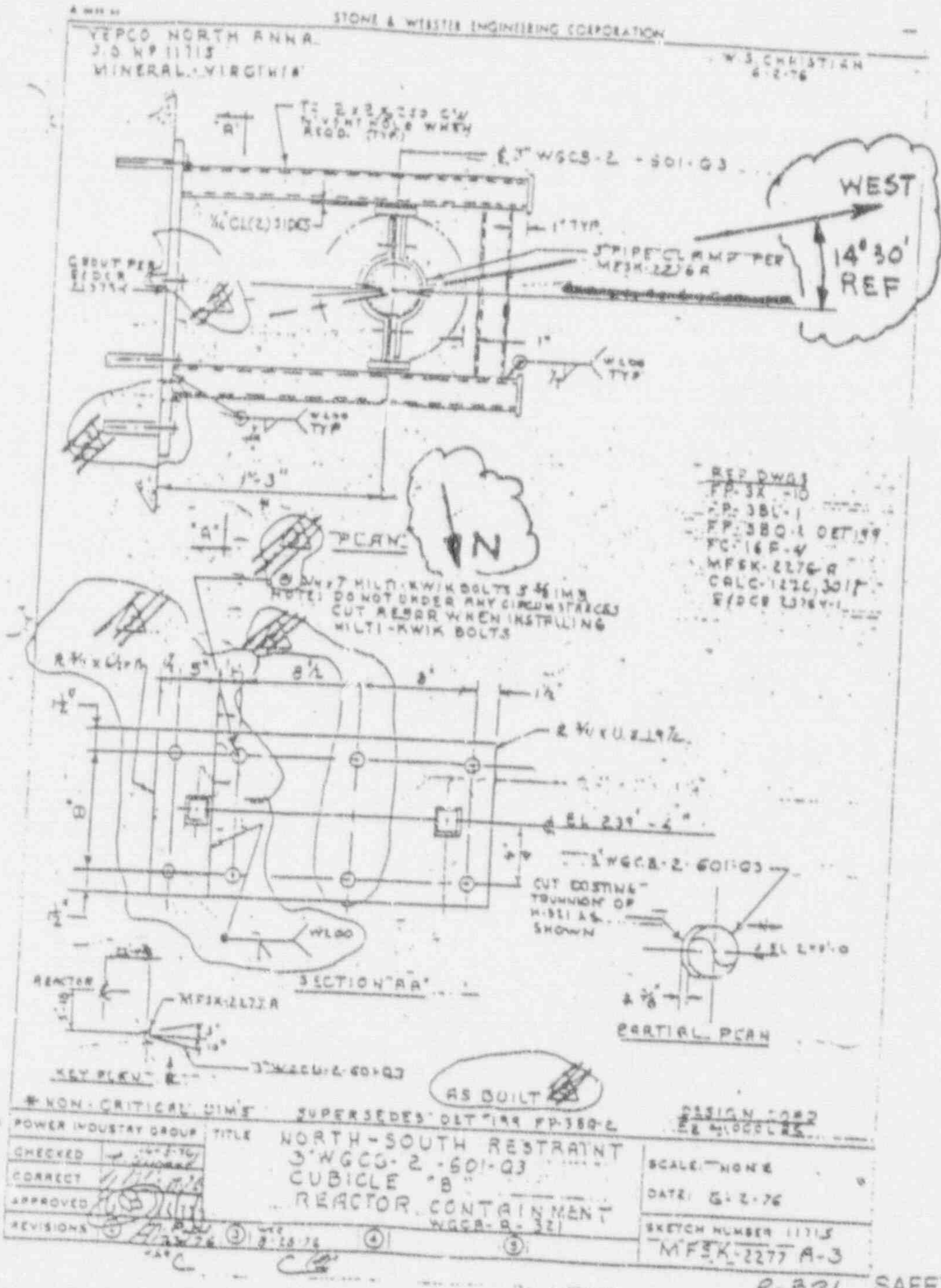


ISSUED FOR CONSTRUCTION OF DC-90-13-1

SAFETY RELATED
DC 90-13-1

	BY	DATE	TITLE	
PREPARED		9/17/92	STEAM GENERATOR BLOWDOWN MODIFICATIONS CUBICLE B PIPE SUPPORT FPH-WGCB-7-4	CLIENT VIRGINIA POWER
CHECKED	RAB	9/17/92		PROJECT NORTH ANNA 1
APPROVED		9/17/92		J.O. NO. 02072.0710
REVISIONS BY	①	② ③ ④ ⑤		ISSUE BY SWEC
DATE				DATE
				NUMBER SHEET 2 OF 2 N9013-1-M-1MFSK1976A-1

STONE & WEBSTER ENGINEERING CORPORATION



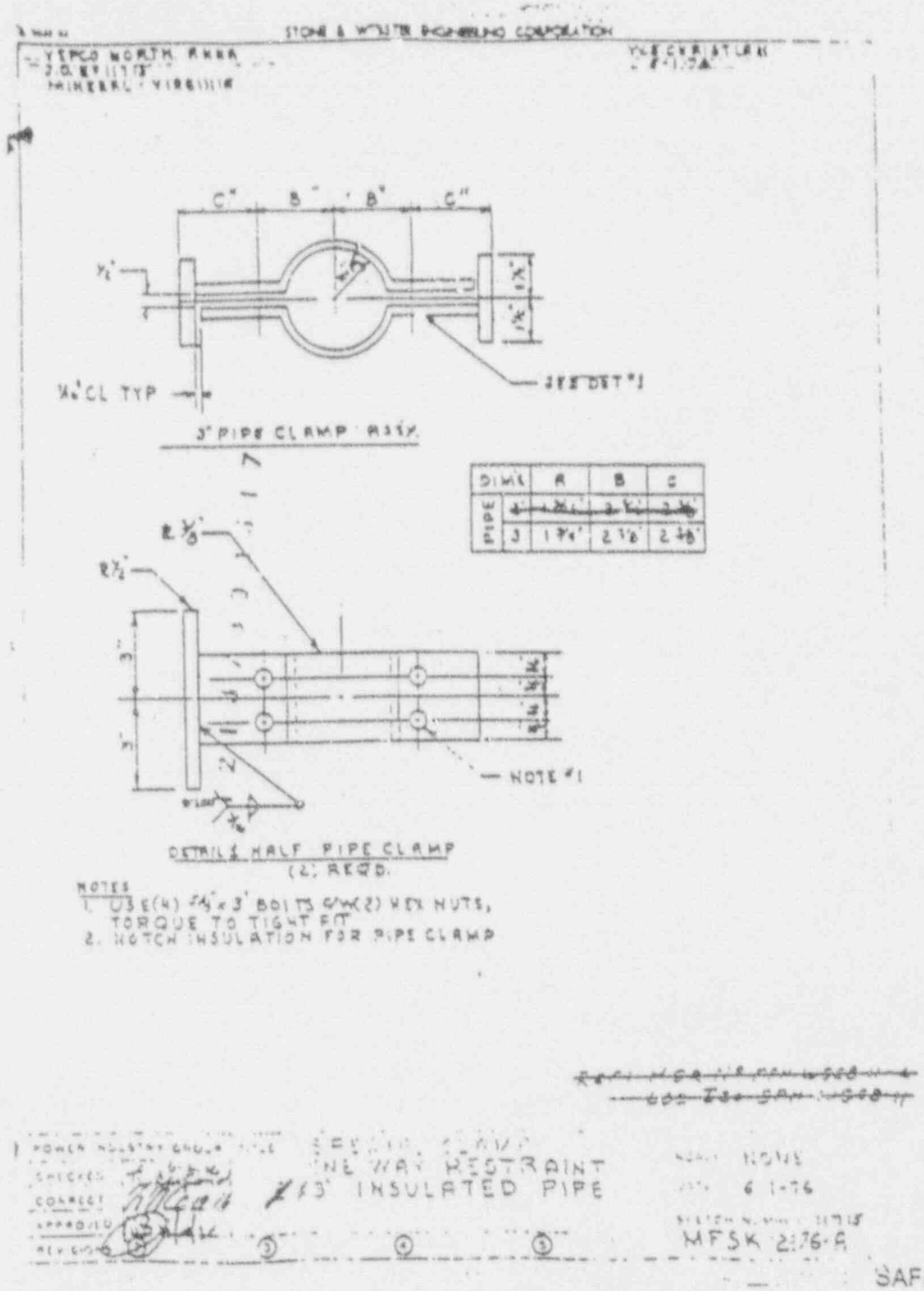
R-321

ISSUED FOR CONSTRUCTION OF DC-90-13-1

SAFETY RELATED
DC 90-13-1

	BY	DATE	TITLE	
PREPARED	DCA	4/17/92	STEAM GENERATOR BLOWDOWN MODIFICATIONS CUBICLE B PIPE SUPPORT 1-WGCB-R-321	CLIENT VIRGINIA POWER
CHECKED	RAB	4/17/92		PROJECT NORTH ANNA 1
APPROVED	DBm	4/17/92		J.O. NO. 02072.0710
				ISSUE BY SWEC
				DATE
REVISIONS BY	⓪	①	② ③ ④	NUMBER SHEET 1 OF 2
DATE				N9013-1-M-1MFSK2277A-0

STONE & WEBSTER ENGINEERING CORPORATION



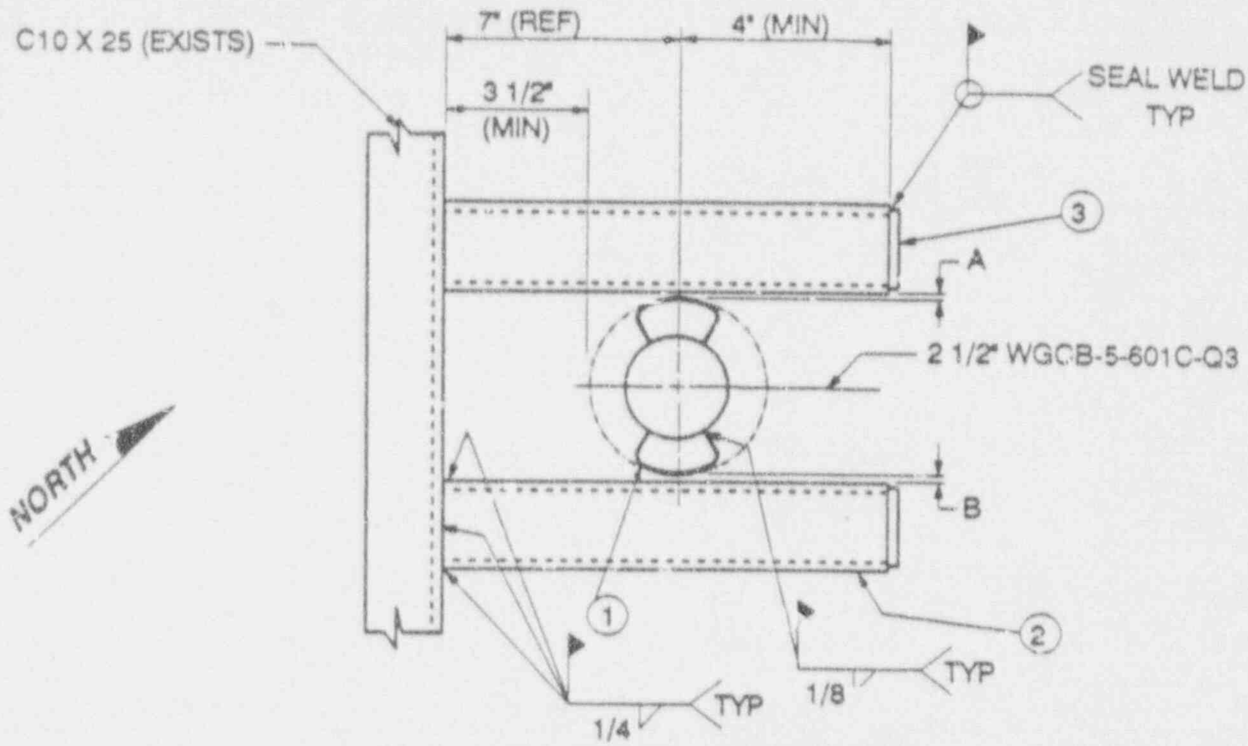
SAFETY RELATED

DC 90-13-1

ISSUED FOR CONSTRUCTION OF DC-90-13-1

	BY	DATE	TITLE	CLIENT	VIRGINIA POWER
PREPARED	DCM	4/17/92	STEAM GENERATOR BLOWDOWN	PROJECT	NORTH ANNA 1
CHECKED	RRB	4/17/92	MODIFICATIONS	J.O. NO.	02072.0710
APPROVED	DBM	4/17/92	CUBICLE B	ISSUE BY	SWEC
			PIPE SUPPORT 1-WGCB-R-321	DATE	

REVISIONS BY	0	1	2	3	4	NUMBER	SHEET 2 OF 2
DATE							N9013-1-M-1MFSK2277A-0



PLAN VIEW AT ELEV 261'-10" (REF)

NOTES:

1. FOR SUPPORT LOCATION SEE DWG. N9013-1-M-601
2. FOR GENERAL NOTES REFER TO DWG. N9013-1-M-700.01
3. FIELD TO REMOVE EXISTING PIPE ATTACHMENT AND ANGLE FRAME SEE SHEET 2
4. CLEARANCES 1/16" $\leq$ (A+B) $\leq$ 5/32", FIELD SHIM TO SUIT.

BILL OF MATERIAL

ITEM	QTY	DESCRIPTION	MAT'L
1	2	2 1/2" PROTECTION SADDLE FOR 2" INS, GRINNELL FIG 162	———
2	2	TS 3 X 3 X 1/4" (LENGTH TO SUIT BY FIELD)	A500 GR. B
3	2	STEEL STRIP 1/8" X 2 3/4" X 2 3/4" W/ 1/4" VENT HOLE	A606

SAFETY RELATED
DC 90-13-1

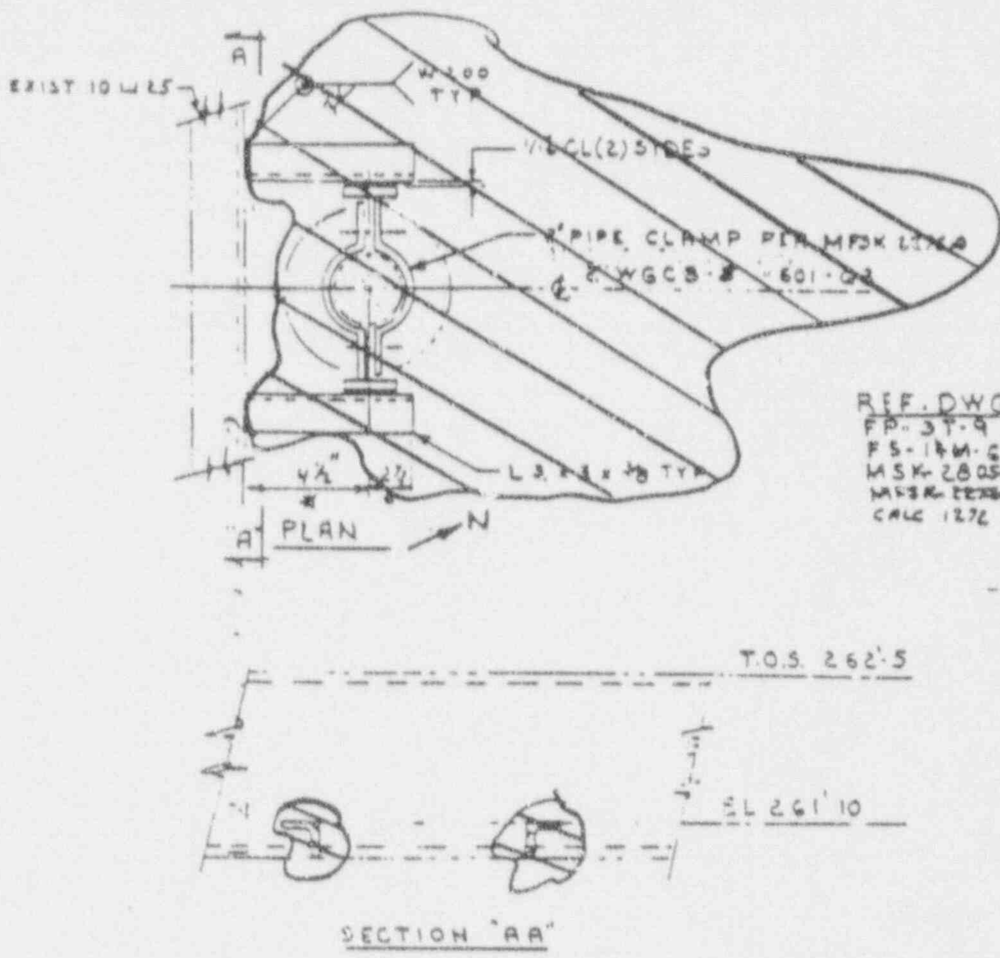
ISSUED FOR CONSTRUCTION OF DC-90-13-1

	BY	DATE	TITLE		
PREPARED	DOM	9/17/92	STEAM GENERATOR BLOWDOWN MODIFICATIONS CUBICLE A PIPE SUPPORT FPH-WGCB-5-5	CLIENT	VIRGINIA POWER
CHECKED	[illegible]	9/17/92		PROJECT	NORTH ANNA 1
APPROVED	[illegible]	9/17/92		J.O. NO.	02072.0710
				ISSUE BY	SWEC
				DATE	
REVISIONS BY	0	1 2 3 4		NUMBER	SHEET 1 OF 2 N9013-1-M-1MFSK2285A-1
DATE					

STONE & WEBSTER ENGINEERING CORPORATION

VEPCO NORTH ANNA
JO No 11715
MINERAL VIRGINIA

W. E. CHRISTIAN
6-2-76



REF. DWG'S
FP-3T-9
FS-14M-6
MSK-2805-R1
~~MFSK-2286B~~
CALC 1276

HGR I.D.: FPH-WGCB-5-5
LOC. ISO: SPH-WGCB-5

DESIGN LOAD
$F_x = \pm 730$ LB

* NON-CRITICAL DIMS.

TITLE	EAST & WEST RESTRAINT. 2"WGCB-5-601-Q3 CUBICLE "A" REACTOR CONTAINMENT
SCALE	NONE
DATE	6-2-76
SKETCH NUMBER	11715 MFSK-2285A

-5

5-5

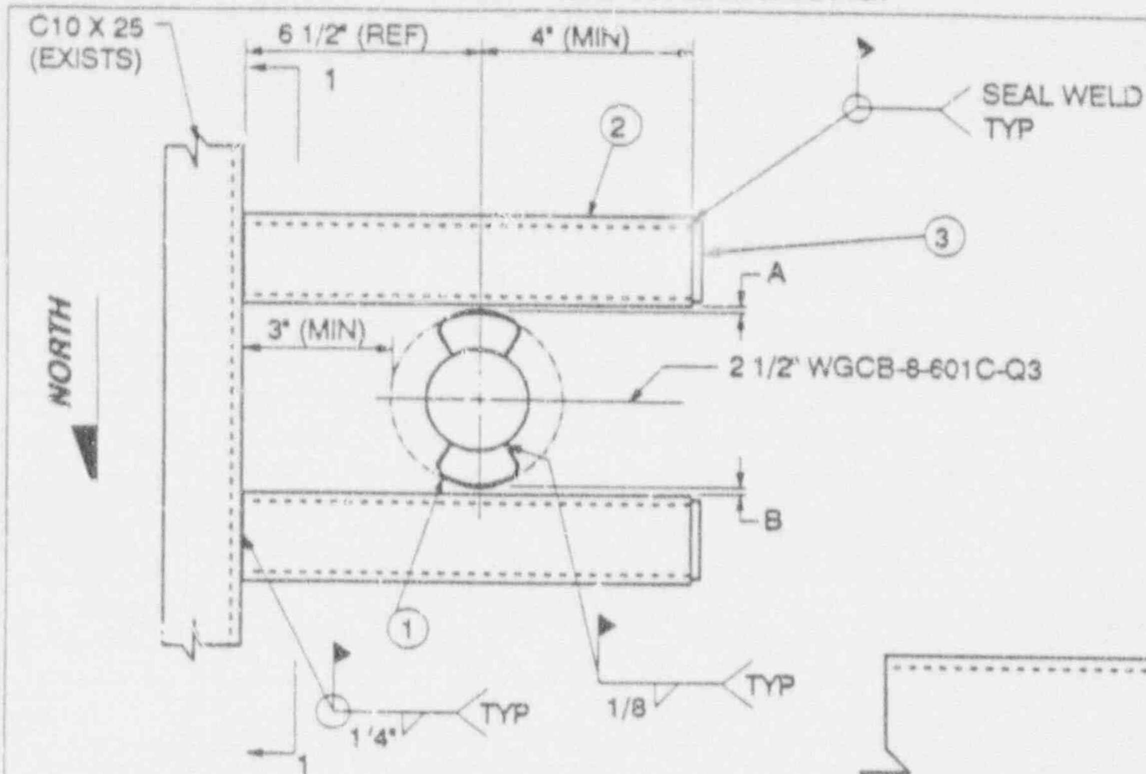
ISSUED FOR CONSTRUCTION OF DC-90-13-1

SAFETY RELATED
DC 90-13-1

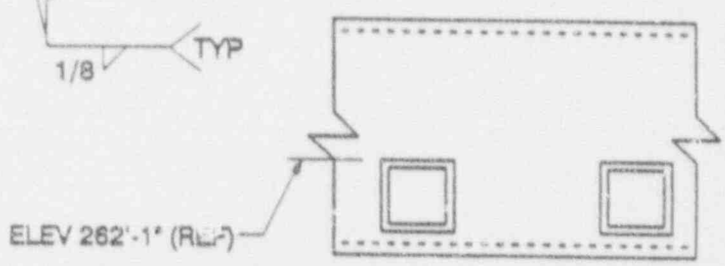
	BY	DATE
PREPARED	DOA	9/17/92
CHECKED	RAB	9/17/92
APPROVED	DSM	9/17/92
REVISIONS BY		
DATE		

TITLE
STEAM GENERATOR BLOWDOWN
MODIFICATIONS
CUBICLE A
PIPE SUPPORT FPH-WGCB-5-5

CLIENT	VIRGINIA POWER
PROJECT	NORTH ANNA 1
J.O. NO.	02072.0710
ISSUE BY	SWEC
DATE	
NUMBER	SHEET 2 OF 2 N9013-1-M-IMFSK2285A-1



PLAN VIEW



SECT 1-1

NOTES:

1. FOR SUPPORT LOCATION SEE DWG. N9013-1-M-602
2. FOR GENERAL NOTES REFER TO DWG. N9013-1-M-700.01
3. FIELD TO REMOVE EXISTING PIPE ATTACHMENT AND ANGLE FRAME SEE SHEET 2
4. CLEARANCES 1/16" ≤ (A+B) ≤ 5/32", FIELD SHIM TO SUIT.

BILL OF MATERIAL			
ITEM	QTY	DESCRIPTION	MAT'L
1	2	2 1/2" PROTECTION SADDLE FOR 2" INS, GRINNELL FIG 162	———
2	2	TS 3 X 3 X 1/4" (LENGTH TO SUIT BY FIELD)	A500 GR. B
3	2	STEEL STRIP 1/8" X 2 3/4" X 2 3/4" W/ 1/4" VENT HOLE	A606

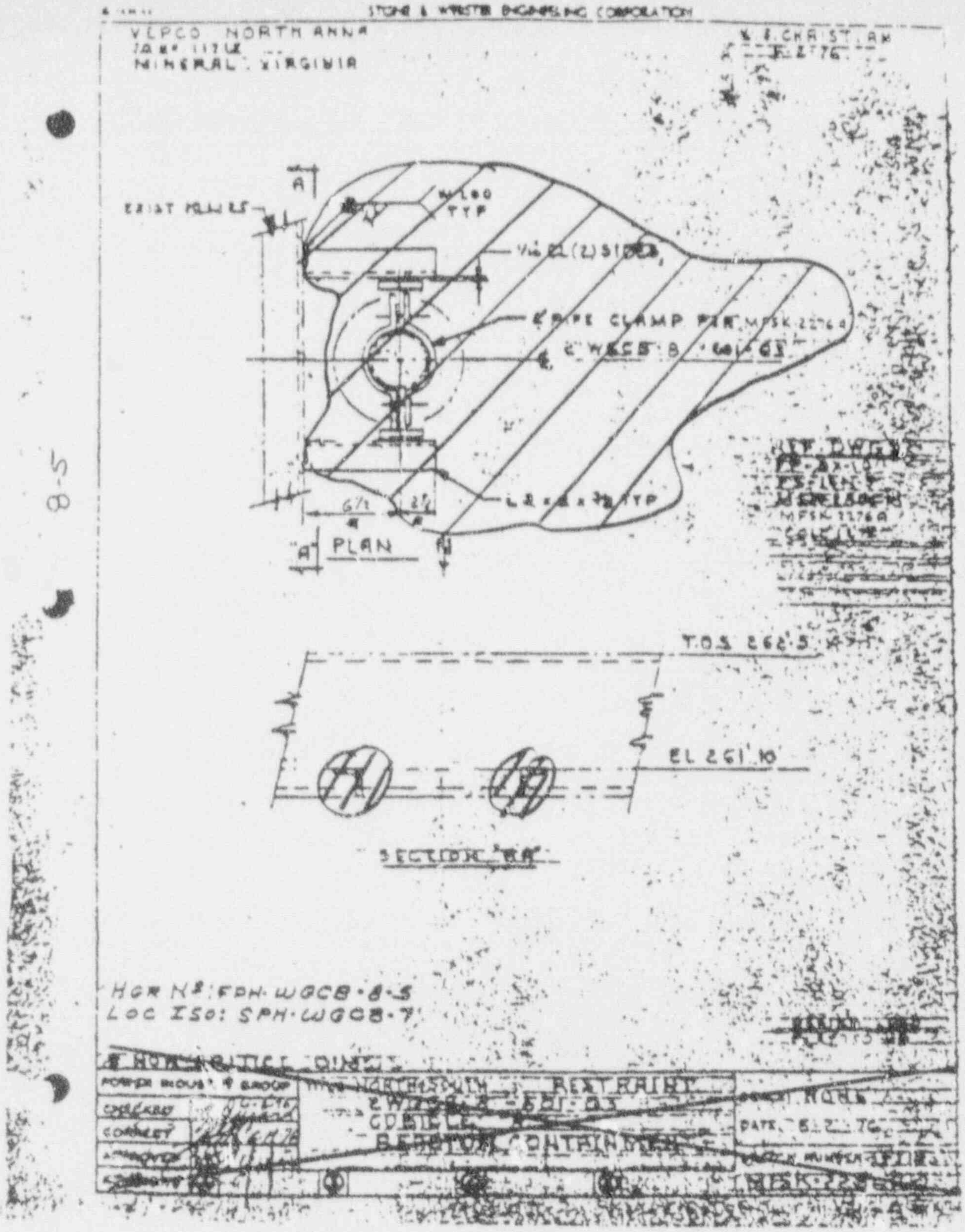
ISSUED FOR CONSTRUCTION OF DC-90-13-1

SAFETY RELATED
DC 90-13-1

	BY	DATE
PREPARED	DOM	9/17/92
CHECKED	BAB	9/17/92
APPROVED	DBM	9/17/92
REVISIONS BY		
DATE		

TITLE
STEAM GENERATOR BLOWDOWN
MODIFICATIONS
CUBICLE B
PIPE SUPPORT FPH-WGCB-8-5

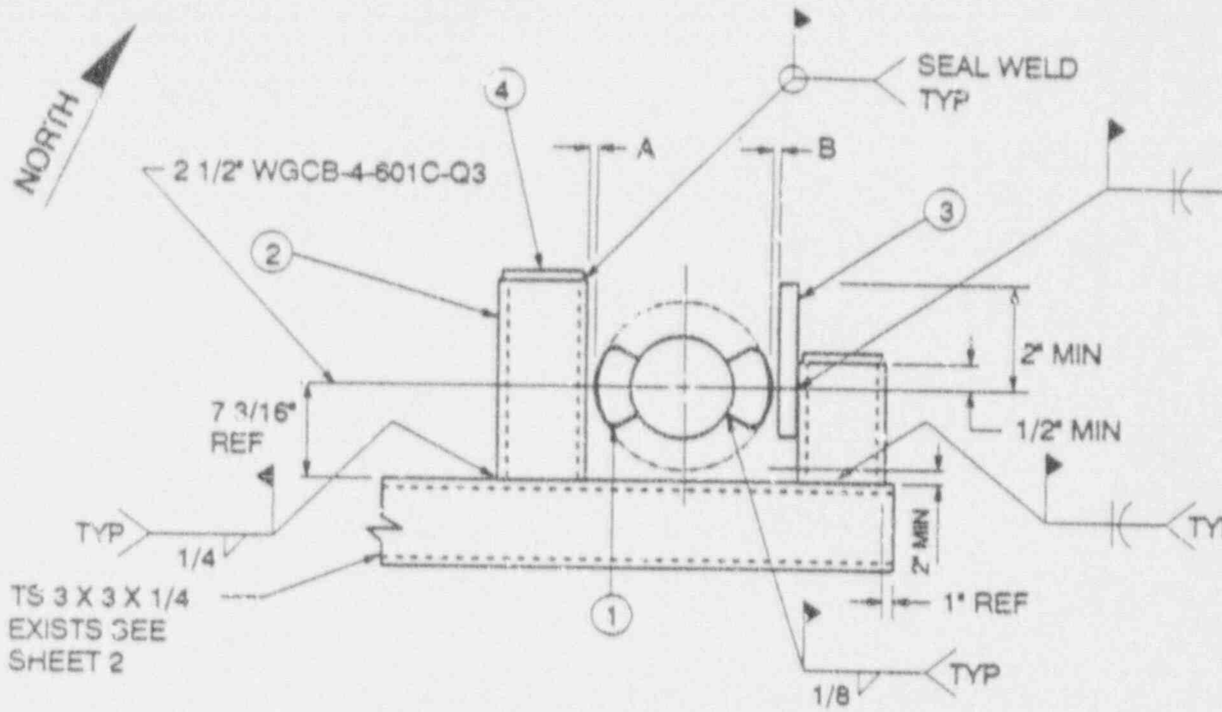
CLIENT VIRGINIA POWER
PROJECT NORTH ANNA 1
J.O. NO. 02072.0710
ISSUE BY SWEC
DATE
NUMBER SHEET 1 OF 2
N9013-1-M-1MFSK2286A-1



ISSUED FOR CONSTRUCTION OF DC-90-13-1

SAFETY RELATED
DC 90-13-1

	BY	DATE	TITLE			CLIENT	VIRGINIA POWER
PREPARED	DCH	9/17/92	STEAM GENERATOR BLOWDOWN MODIFICATIONS CUBICLE B PIPE SUPPORT FPH-WGCB-8-5			PROJECT	NORTH ANNA 1
CHECKED	RB	9/17/92				J.O. NO.	02072.0710
APPROVED	DCM	9/17/92				ISSUE BY	SWEC
REVISIONS BY	0	0	0	0	0	DATE	
DATE						NUMBER	SHEET 2 OF 2 N9013-1-M-1MFSK2286A-1



PLAN VIEW AT ELEV 256'-0" (REF)

NOTES:

1. FOR SUPPORT LOCATION SEE DWG. N9013-1-M-601
2. FOR GENERAL NOTES REFER TO DWG. N9013-1-M-700.01
3. FIELD TO REMOVE EXISTING PIPE ATTACHMENT AND ANGLE FRAME
4. CLEARANCES 1/16" ≤ (A+B) ≤ 5/32", FIELD SHIM TO SUIT.

BILL OF MATERIAL

ITEM	QTY	DESCRIPTION	MAT'L
1	2	2 1/2" PROTECTION SADDLE FOR 2" INS, GRINNELL FIG 162	———
2	2	TS 3 X 3 X 1/4" (LENGTH TO SUIT BY FIELD)	A500 GR. B
3	1	PLATE 1/2" X 4" (LENGTH TO SUIT BY FIELD)	A36
4	2	STEEL STRIP 1/8" X 2 3/4" X 2 3/4" W/ 1/4" VENT HOLE	A606

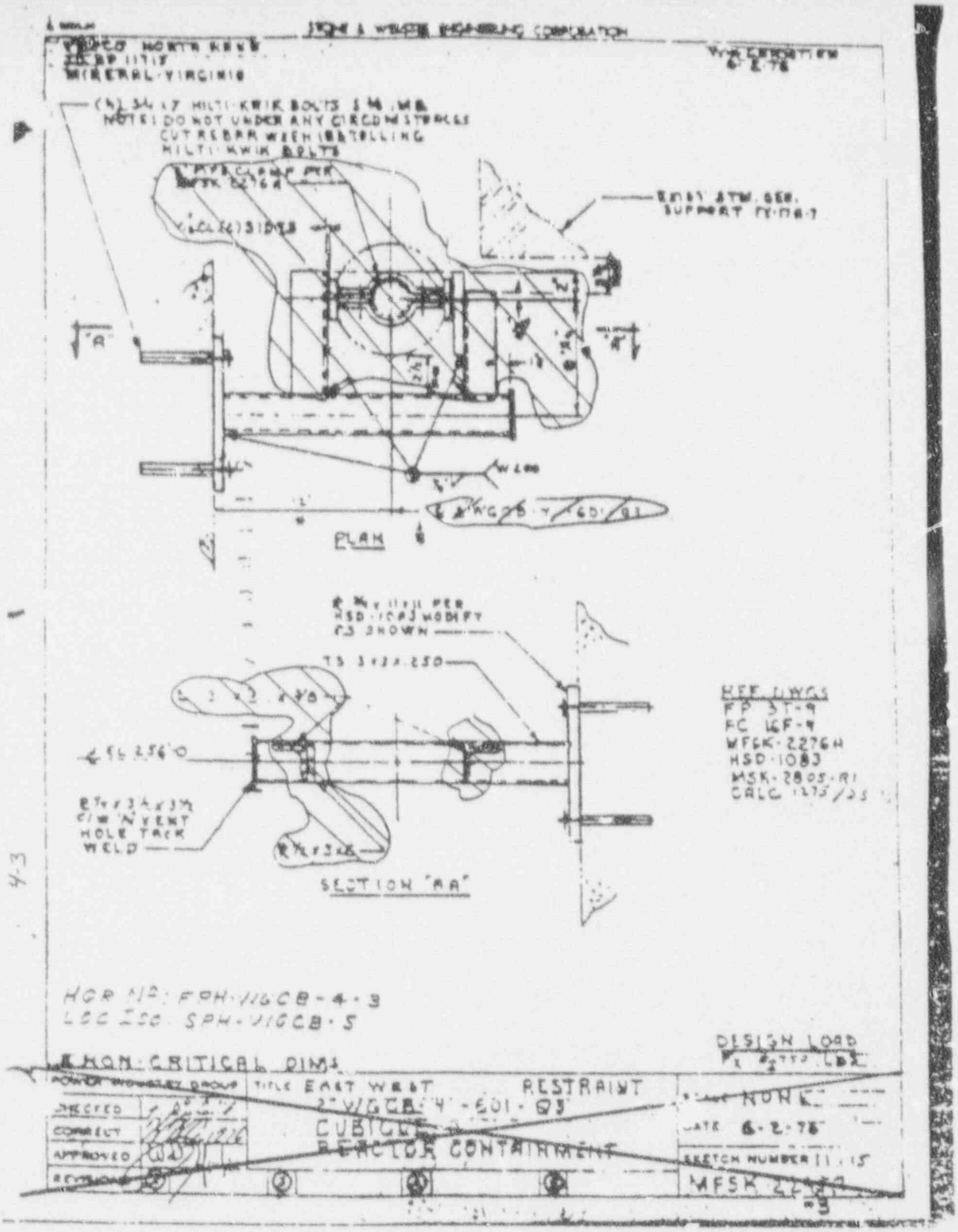
ISSUED FOR CONSTRUCTION OF DC-90-13-1

SAFETY RELATED
DC 90-13-1

	BY	DATE
PREPARED	DCM	9/17/92
CHECKED	RFB	9/17/92
APPROVED	WBP	9/17/92
REVISIONS BY	0	
DATE		

TITLE
STEAM GENERATOR BLOWDOWN
MODIFICATIONS
CUBICLE A
PIPE SUPPORT FPH-WGCB-4-3

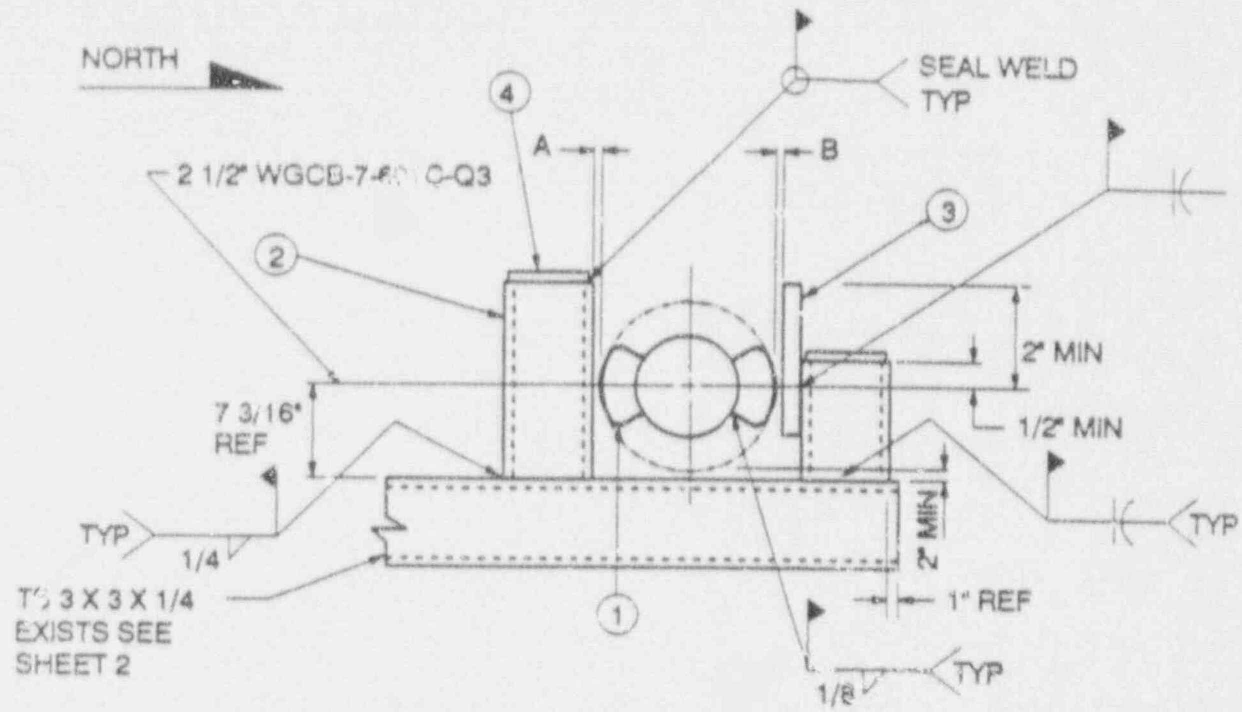
CLIENT	VIRGINIA POWER
PROJECT	NORTH ANNA 1
J.O. NO.	02072.0710
ISSUE BY	SWEC
DATE	
NUMBER	SHEET 1 OF 2
	N9013-1-M-1MFSK2295A-1



ISSUED FOR CONSTRUCTION OF DC-90-13-1

SAFETY RELATED
DC 90-13-1

	BY	DATE	TITLE		
PREPARED	WVM	9/17/92	STEAM GENERATOR BLOWDOWN MODIFICATIONS CUBICLE A PIPE SUPPORT FPH-WGCB-4-3	CLIENT	VIRGINIA POWER
CHECKED	RAB	9/17/92		PROJECT	NORTH ANNA 1
APPROVED	DSM	9/17/92		J.O. NO.	02072.0710
				ISSUE BY	SWEC
				DATE	
REVISIONS BY	0	0	0	NUMBER	SHEET 2 OF 2
DATE					N9013-1-M-1MFSK2295A-1



PLAN VIEW AT ELEV 256'-0" (REF)

NOTES:

1. FOR SUPPORT LOCATION SEE DWG. N9013-1-M-602
2. FOR GENERAL NOTES REFER TO DWG. N9013-1-M-700.01
3. FIELD TO REMOVE EXISTING PIPE ATTACHMENT AND ANGLE FRAME
4. CLEARANCES 1/16" ≤ (A+B) ≤ 5/32", FIELD SHIM TO SUIT.

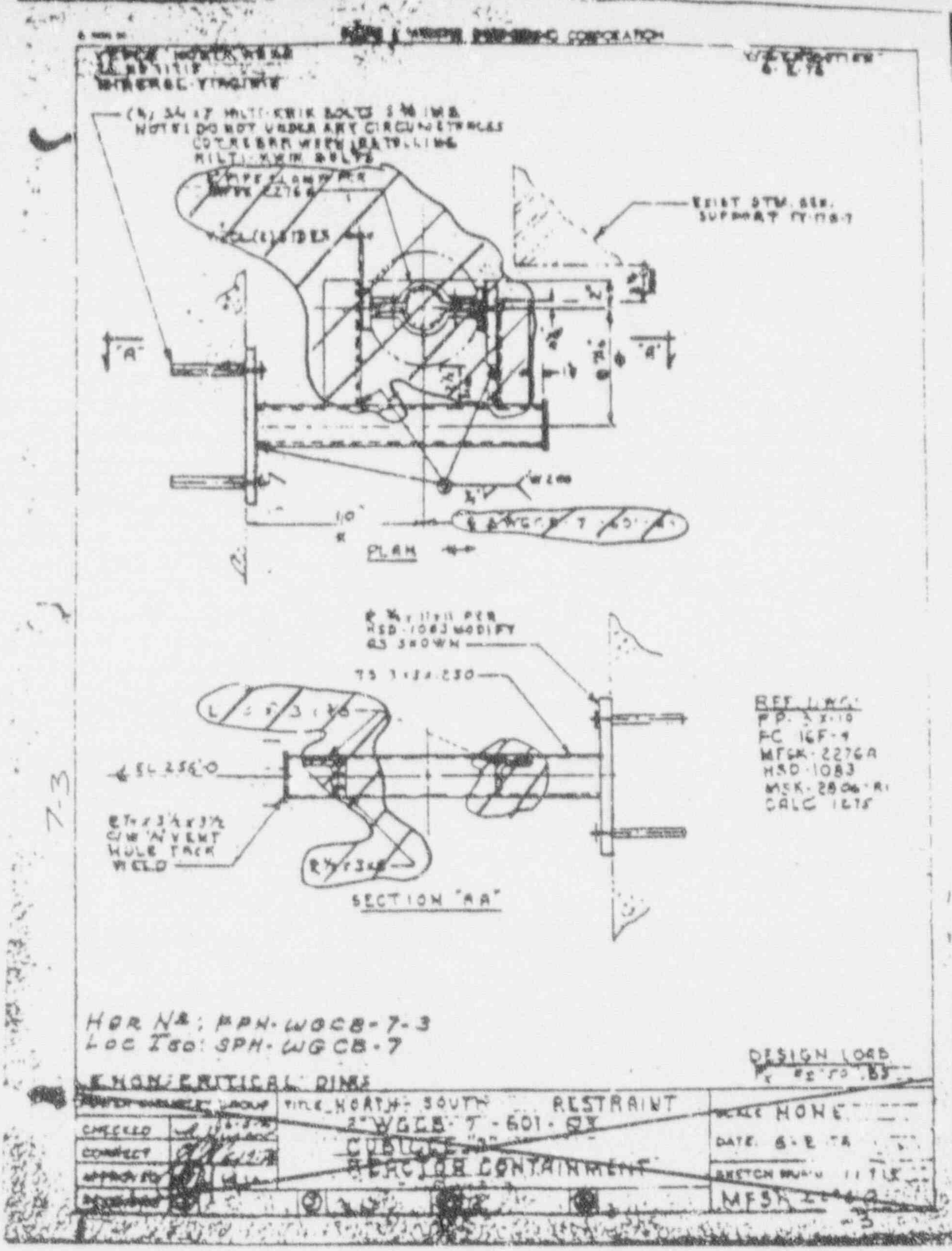
BILL OF MATERIAL

ITEM	QTY	DESCRIPTION	MAT'L
1	2	2 1/2" PROTECTION SADDLE FOR 2" INS, GRINNELL FIG 162	———
2	2	TS 3 X 3 X 1/4" (LENGTH TO SUIT BY FIELD)	A500 GR. B
3	1	PLATE 1/2" X 4" (LENGTH TO SUIT BY FIELD)	A36
4	2	STEEL STRIP 1/8" X 2 3/4" X 2 3/4" W/ 1/4" VENT HOLE	A606

ISSUED FOR CONSTRUCTION OF DC-90-13-1

SAFETY RELATED
DC 90-13-1

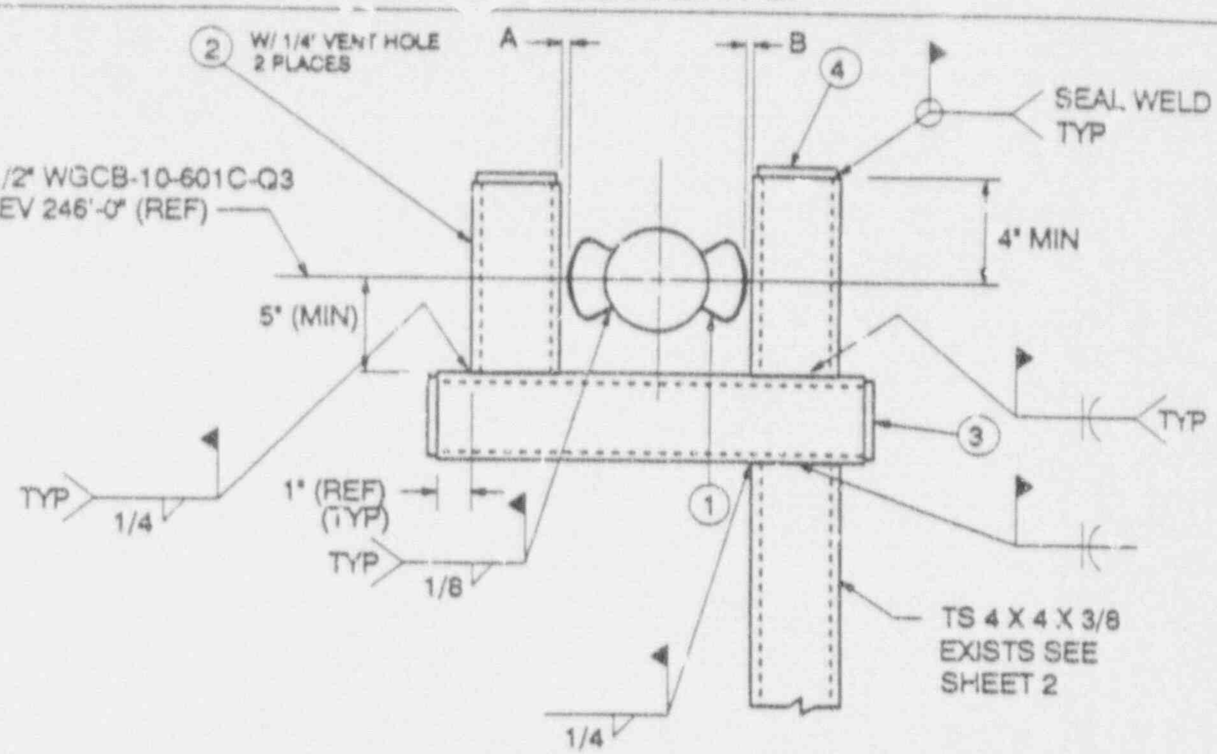
	BY	DATE	TITLE	
PREPARED	DOM	9/17/92	STEAM GENERATOR BLOWDOWN MODIFICATIONS CUBICLE B PIPE SUPPORT FPH-WGCB-7-3	CLIENT VIRGINIA POWER
CHECKED	[illegible]	9/17/92		PROJECT NORTH ANNA 1
APPROVED	[illegible]	9/17/92		J.O. NO. 02072.0710
				ISSUE BY SWEC
				DATE
REVISIONS BY	0	0	0 0 0	NUMBER SHEET 1 OF 2 N9013-1-M-1MFSK2296A-1
DATE				



ISSUED FOR CONSTRUCTION OF DC-90-13-1

SAFETY RELATED
DC 90-13-1

	BY	DATE	TITLE			CLIENT VIRGINIA POWER
			STEAM GENERATOR BLOWDOWN MODIFICATIONS			PROJECT NORTH ANNA 1
PREPARED	DCM	9/17/92	CUBICLE B			J.O. NO. 02072.0710
CHECKED	RAB	9/17/92				ISSUE BY SWEC
APPROVED		9/	PIPE SUPPORT FPH-WGCB-7-3			DATE
REVISIONS BY	0	1	2	3	4	NUMBER SHEET 2 OF 2
DATE						N9013-1-M-IMFSK2296A-1



ELEV LOOKING SOUTH-WEST

NOTES:

1. FOR SUPPORT LOCATION SEE DWG. N9013-1-M-603
2. FOR GENERAL NOTES REFER TO DWG. N9013-1-M-700.01
3. FIELD TO REMOVE EXISTING PIPE ATTACHMENT AND CUT BACK TUBE STEEL.
4. CLEARANCES 1/16" $\leq$ (A+B) $\leq$ 5/32", FIELD SHIM TO SUIT.

BILL OF MATERIAL

ITEM	QTY	DESCRIPTION	MAT'L
1	2	2 1/2" PROTECTION SADDLE FOR 2" INS, GRINNELL FIG 162	———
2	3	TS 4 X 4 X 1/4" (LENGTH TO SUIT BY FIELD)	A500 GR. B
3	2	STEEL STRIP 1/8" X 3 3/4" X 3 3/4" W/ 1/4" VENT HOLE	A606
4	2	STEEL STRIP 1/8" X 3 3/4" X 3 3/4"	A606

SAFETY RELATED
DC 90-13-1

ISSUED FOR CONSTRUCTION OF DC-90-13-1

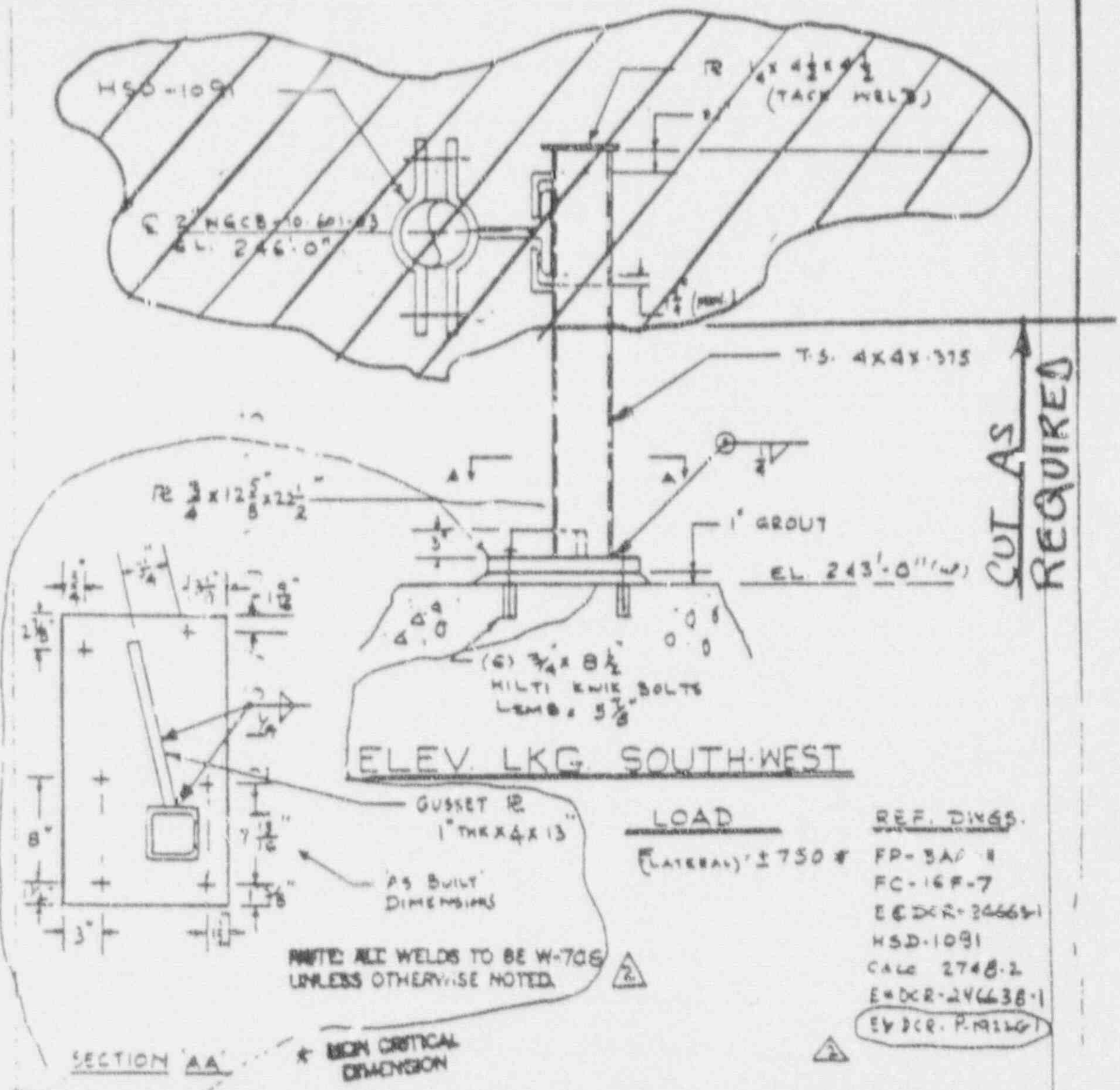
	BY	DATE	TITLE	
PREPARED	DCM	9/17/92	STEAM GENERATOR BLOWDOWN MODIFICATIONS CUBICLE C PIPE SUPPORT FPH-WGCB-10-1	CLIENT VIRGINIA POWER
CHECKED	[illegible]	9/17/92		PROJECT NORTH ANNA 1
APPROVED	[illegible]	9/17/92		J.O. NO. 02072.0710
				ISSUE BY SWEC
				DATE
REVISIONS BY	0	1	2 3 4	NUMBER SHEET 1 OF 2
DATE				N9013-1-M-1MFSK3040A-1

STONE & WEBSTER ENGINEERING CORPORATION

VEPCO NORTH ANNA
J.O. NO. 11715
MINERAL, VA.

APPEARS AS FPH-WGCB-10-1
ON LOC 130 SPH-WGCB-10

J. BHATIA
1-20-77



LOAD
F(LATERAL) = ±750 #

REF. DWGS.
FP-5A/ 4
FC-16F-7
E&DCR-24663-1
HSD-1091
CALC 2748.2
E&DCR-246638-1
E&DCR P-1922G-1

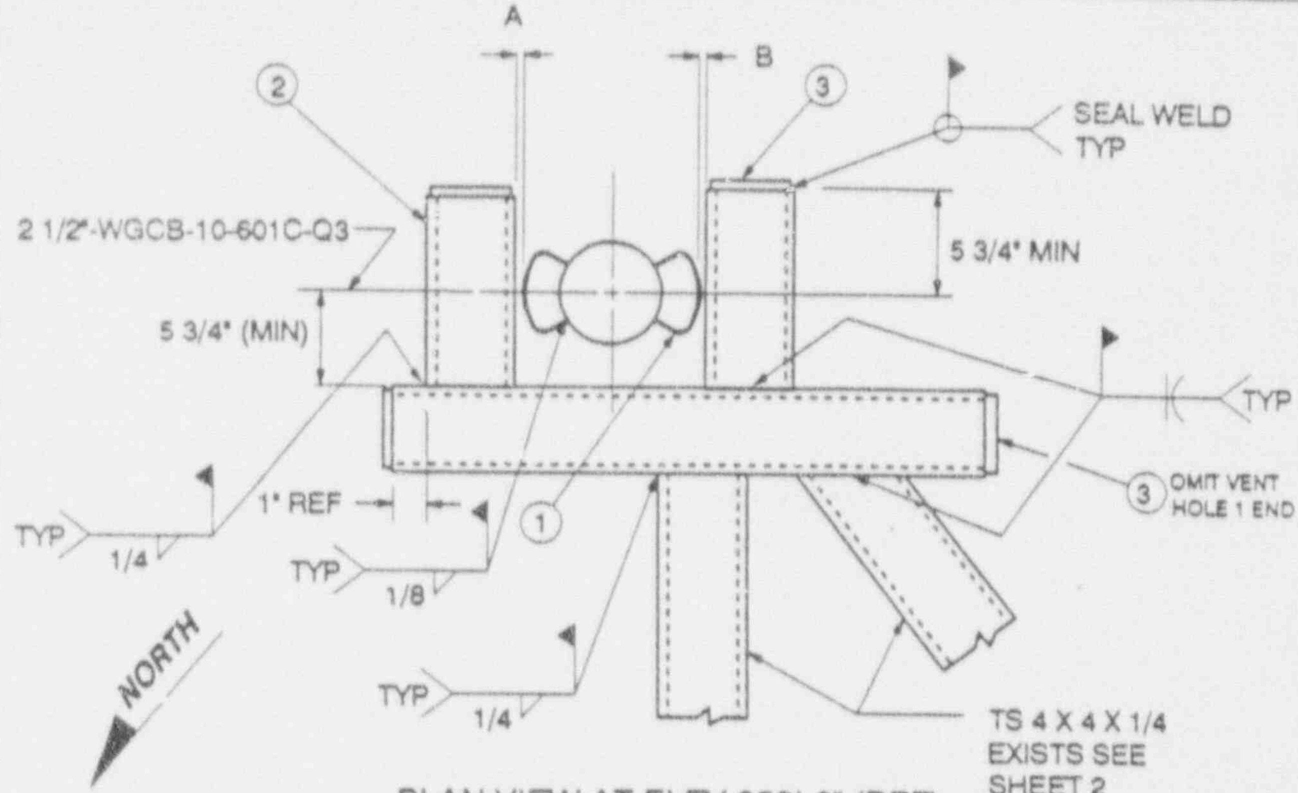
NOTE: MODIFY AS SHOWN PER E&DCR P-1922G EVALUATION PROCEDURE

POWER INDUSTRY GROUP	TITLE LATERAL RESTRAINT FOR 2" WGCB-10-601-Q3 REACT. CTMT. CUBICLE C	SCALE NONE
CHECKED CH 1-26-77		DATE 1-20-77
CORRECT		SKETCH NUMBER 11715
APPROVED		MFSK-3040A-2
REVISIONS ②	③ ④ ⑤	

ISSUED FOR CONSTRUCTION OF DC-90-13-1

SAFETY RELATED
DC 90-13-1

	BY	DATE	TITLE	
PREPARED		9/17/92	STEAM GENERATOR BLOWDOWN MODIFICATIONS CUBICLE C PIPE SUPPORT FPH-WGCB-10-1	CLIENT VIRGINIA POWER
CHECKED		9/17/92		PROJECT NORTH ANNA 1
APPROVED		9/17/92		J.O. NO. 02072.0710
				ISSUE BY SWEC
				DATE
REVISIONS BY	⓪	⓪ ⓪ ⓪ ⓪		NUMBER SHEET 2 OF 2
DATE				N9013-1-M-1 MESK3040A-1



PLAN VIEW AT ELEV 252'-6" (REF)

NOTES:

1. FOR SUPPORT LOCATION SEE DWG. N9013-1-M-603
2. FOR GENERAL NOTES REFER TO DWG. N9013-1-M-700.01
3. FIELD TO REMOVE EXISTING PIPE ATTACHMENT AND CUT BACK TUBE STEEL
4. CLEARANCES 1/16" $\leq$ (A+B) $\leq$ 5/32", FIELD SHIM TO SUIT.

BILL OF MATERIAL

ITEM	QTY	DESCRIPTION	MAT'L
1	2	2 1/2" PROTECTION SADDLE FOR 2" INS, GRINNELL FIG 162	—
2	3	TS 4 X 4 X 1/4" (LENGTH TO SUIT BY FIELD)	A500 GR. B
3	4	STEEL STRIP 1/8" X 3 3/4" X 3 3/4" W/ 1/4" VENT HOLE	A606

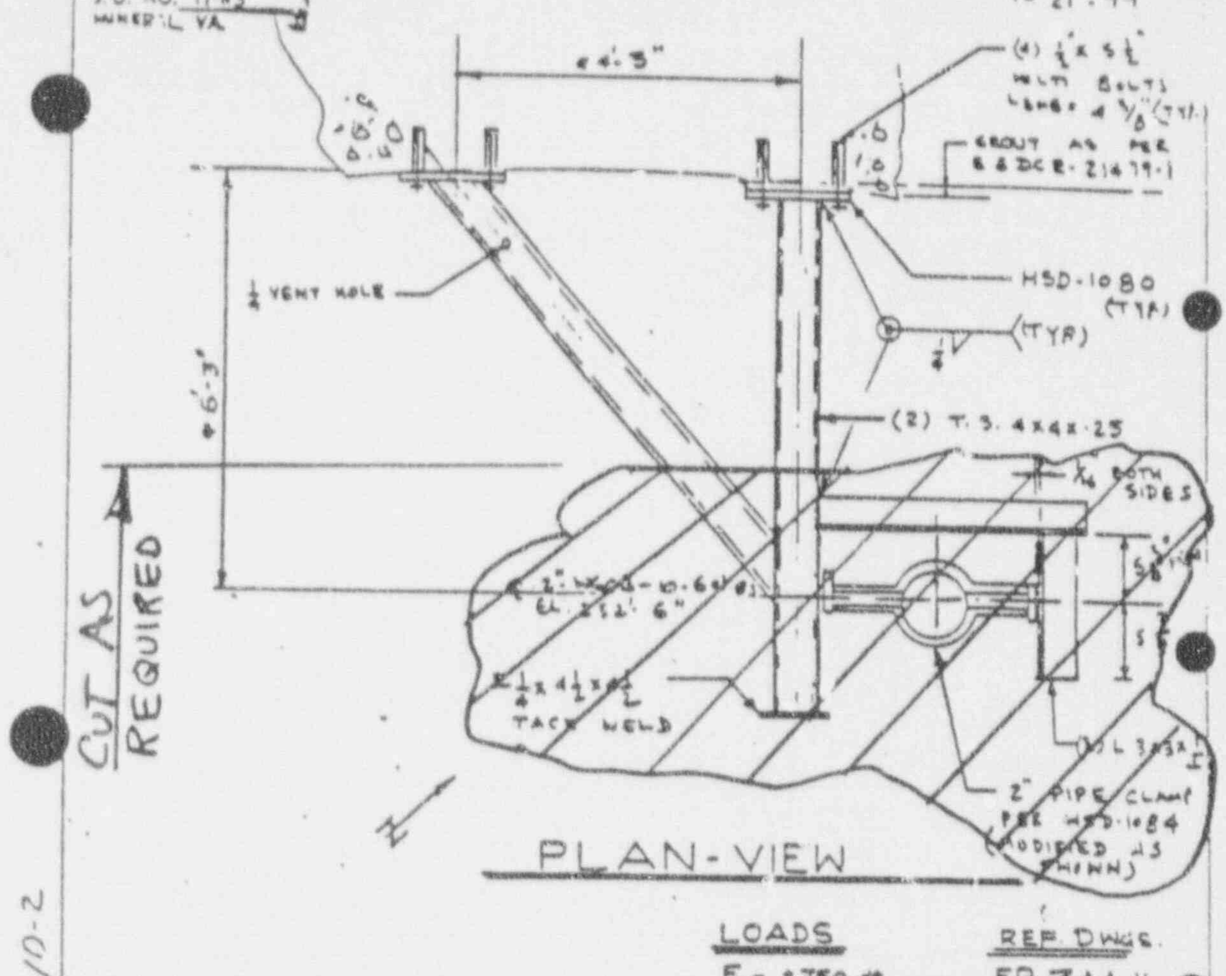
SAFETY RELATED
DC 90-13-1

ISSUED FOR CONSTRUCTION OF DC-90-13-1

	BY	DATE	TITLE	
PREPARED	DCM	9/17/92	STEAM GENERATOR BLOWDOWN MODIFICATIONS CUBICLE C PIPE SUPPORT FPH-WGCB-10-2	CLIENT VIRGINIA POWER
CHECKED	BJS	9/17/92		PROJECT NORTH ANNA 1
APPROVED	DBM	9/17/92		J.O. NO. 02072.0710
				ISSUE BY SWEC
				DATE
REVISIONS BY	0			NUMBER SHEET 1 OF 2
DATE				N9013-1-M-1MFSK3041A-1

VEPCO NORTH ANNA
J.O. NO. 11715
MINERAL VA.

J. BHATIA
1-21-77



LOADS
$F_x = 2750$ #

REF. DWGS.
FP-3AA-11
FC-16A-7
HSD-1091
E&DCR-24663-1
E&DCR-21479-1
CALC. 638
HSD-1080
E&DCR-24663C-1

NOTE: ALL WELDS TO BE W-200 A UNLESS OTHERWISE NOTED.

* NON CRITICAL DIMENSION

APPEARS AS FPH-WGCB-10-2 ON LOC 150 SPH-WGCB-10

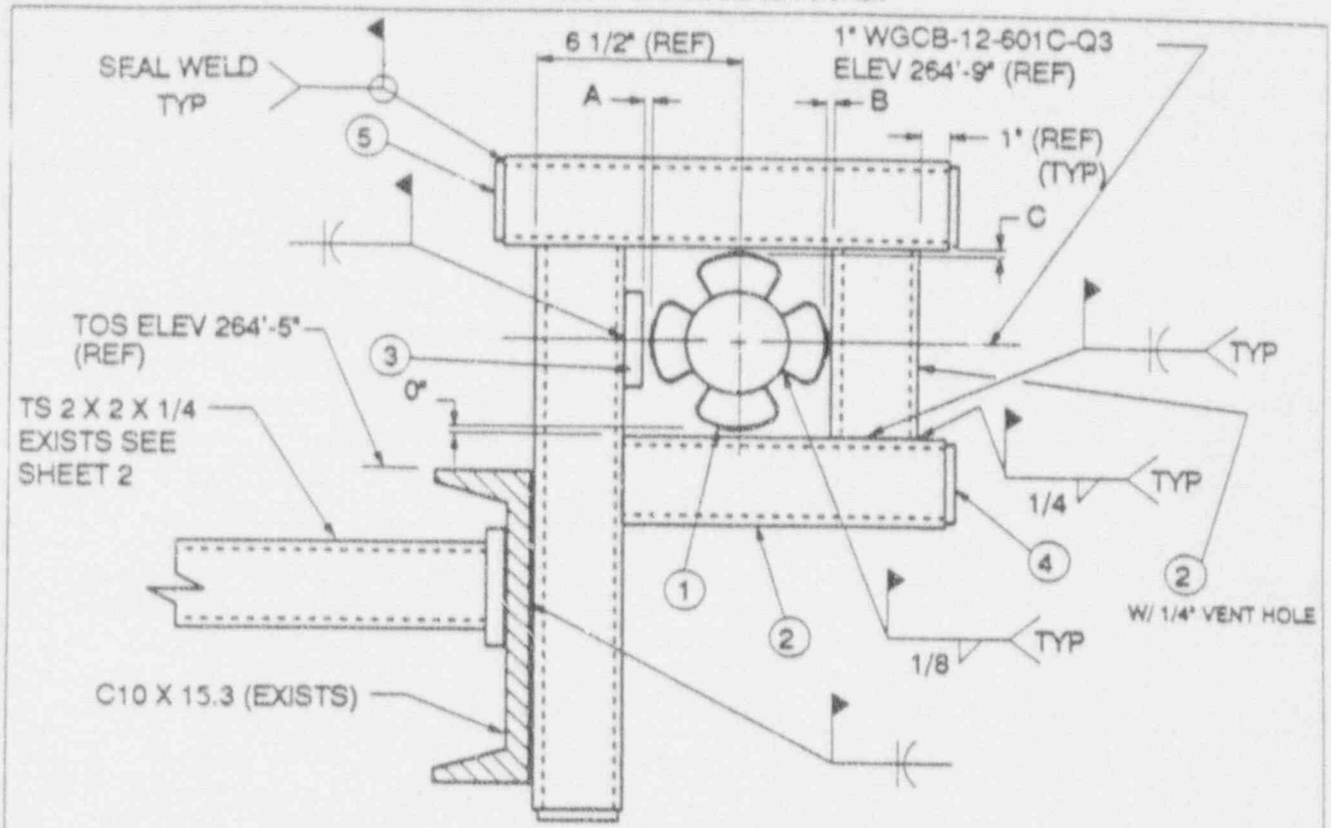
"DO NOT UNDER ANY CIRCUMSTANCES CUT REBAR WHEN INSTALLING MULTI KWIK BOLTS"

POWER INDUSTRY GROUP	TITLE ~~LATERAL RESTRAINT FOR~~	SCALE NONE
CHECKED	~~2-WGCB-10-601-Q3~~	DATE 1-21-77
CORRECT	~~REACT. CTMT. CUBICLE C~~	
APPROVED		SKETCH NUMBER 11715
REVISIONS (2) (3) (4) (5)		~~MFSK-3041A~~

ISSUED FOR CONSTRUCTION OF DC-90-13-1

SAFETY RELATED
DC 90-13-1

	BY	DATE	TITLE	
PREPARED	AOK	9/12/92	STEAM GENERATOR BLOWDOWN MODIFICATIONS	CLIENT VIRGINIA POWER
CHECKED	RRB	9/17/92	CUBICLE C	PROJECT NORTH ANNA 1
APPROVED		9/17/92	PIPE SUPPORT FPH-WGCB-10-2	J.O. NO. 02072.0710
REVISIONS BY				ISSUE BY SWEC
DATE				DATE
				NUMBER SHEET 2 OF 2 N9013-1-M-1MFSK3041A-1



ELEV LOOKING SOUTH-EAST

NOTES:

1. FOR SUPPORT LOCATION SEE DWG. N9013-1-M-603
2. FOR GENERAL NOTES REFER TO DWG. N9013-1-M-700.01
3. FIELD TO REMOVE EXISTING PIPE ATTACHMENT AND TS 4 X 4 X 1/4
4. CLEARANCES 1/16" ≤ (A+B), (C) ≤ 5/32", FIELD SHIM TO SUIT.

BILL OF MATERIAL

ITEM	QTY	DESCRIPTION	MAT'L
1	4	1" PROTECTION SADDLE FOR 2" INS, GRINNELL FIG 162	———
2	4	TS 3 X 3 X 1/4" (LENGTH TO SUIT BY FIELD)	A500 GR. B
3	1	PLATE 3" X 3" X (THICKNESS TO SUIT BY FIELD)	A36
4	3	STEEL STRIP 1/8" X 2 3/4" X 2 3/4" W/ 1/4" VENT HOLE	A606
5	1	STEEL STRIP 1/8" X 2 3/4" X 2 3/4"	A606

ISSUED FOR CONSTRUCTION OF DC-90-13-1

SAFETY RELATED
DC 90-13-1

	BY	DATE
PREPARED	DOM	9/17/92
CHECKED	RBB	9/17/92
APPROVED	KB	9/17/92
REVISIONS BY	0	
DATE		

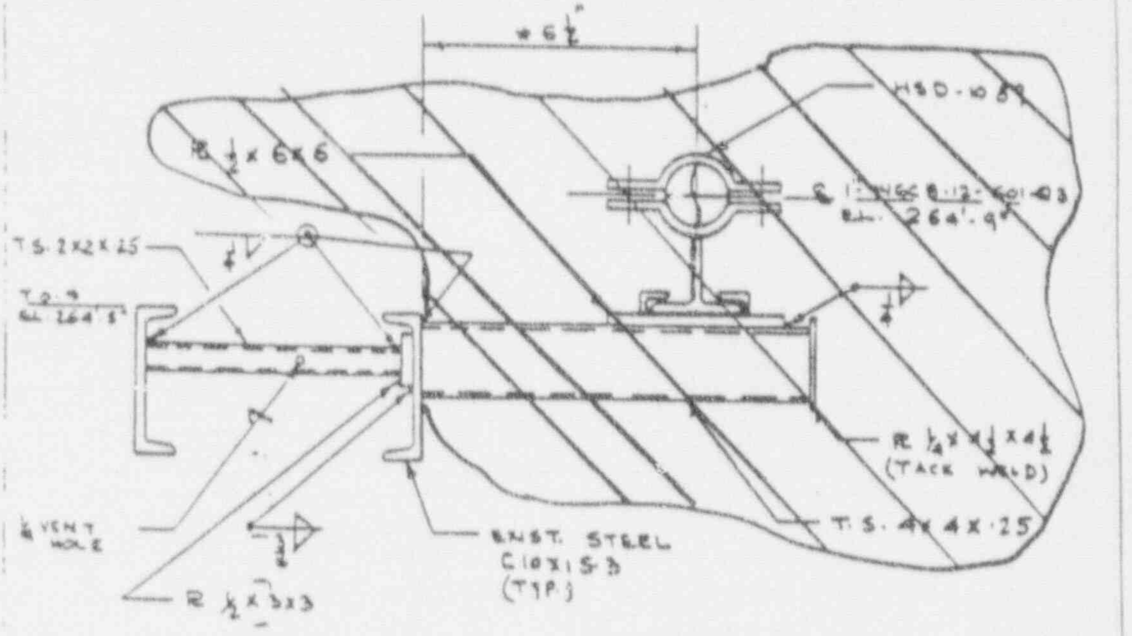
TITLE
STEAM GENERATOR BLOWDOWN
MODIFICATIONS
CUBICLE C
PIPE SUPPORT FPH-WGCB-12-1

CLIENT	VIRGINIA POWER
PROJECT	NORTH ANNA 1
J.O. NO.	02072.0710
ISSUE BY	SWEC
DATE	
NUMBER	SHEET 1 OF 2
	N9013-1-M-1MFSK3044A-1

STONE & WEBSTER ENGINEERING CORPORATION

VEPCO NORTH ANNA
J.O. NO. 11715
MINERAL, VA.

J. BHATIA
1-21-77



ELEV LKG SOUTH-EAST

APPEARS AS FPH-WGCB-12-1
ON LOC 130 SPH-WGCB-10

LOADS

$F_x = \pm 400$ #
$F_y = \pm 400$ #

REF. DWGS.

FP-3AA-11
FS-14P-7
HSD-1089
E&DCR-24663-1
CALC HSD-1-95
E&DCR-246638-1

NOTE: ALL WELDS TO BE W200 A UNLESS OTHERWISE NOTED.

* NON CRITICAL DIMENSION

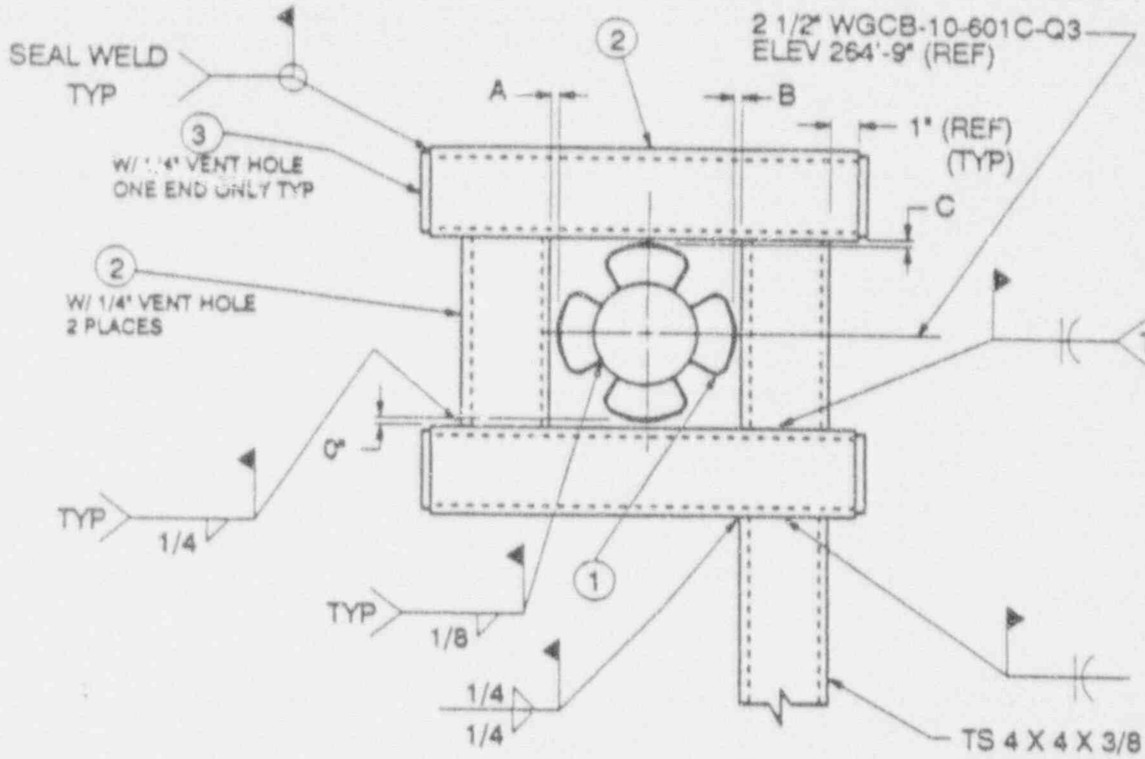
12-1

POWER INDUSTRY GROUP	TITLE LATERAL & VERTICAL RESTRAINT FOR 1" WGCB-12-601-Q3 REACT. CTMT., CUBICLE C		SCALE NONE
CHECKED	CAL 1-26-77		DATE: 1-21-77
CORRECT			SKETCH NUMBER 11715
APPROVED			MFSK-3044A-1
REVISIONS	② ③ ④ ⑤		

ISSUED FOR CONSTRUCTION OF DC-90-13-1

SAFETY RELATED
DC 90-13-1

	BY	DATE	TITLE		
PREPARED	[illegible]	9/17/92	STEAM GENERATOR BLOWDOWN MODIFICATIONS CUBICLE C	CLIENT	VIRGINIA POWER
CHECKED	RBP	9/17/92		PROJECT	NORTH ANNA 1
APPROVED	DEM	9/17/92	PIPE SUPPORT FPH-WGCB-12-1	J.O. NO.	02072.0710
REVISIONS BY	③		③ ③ ③ ③	ISSUE BY	SWEC
DATE				DATE	
				NUMBER	SHEET 2 OF 2
					N9013-1-M-1MFSK3044A-1



ELEV LOOKING NORTH-WEST

NOTES:

1. FOR SUPPORT LOCATION SEE DWG. N9013-1-M-603
2. FOR GENERAL NOTES REFER TO DWG. N9013-1-M-700.01
3. FIELD TO REMOVE EXISTING PIPE ATTACHMENT, CUT BACK EXISTING TS 4 X 4 X 3/8 AND REFABRICATE AS SHOWN
4. CLEARANCES 1/16" ≤ (A+B) , (C) ≤ 5/32", FIELD SHIM TO SUIT.

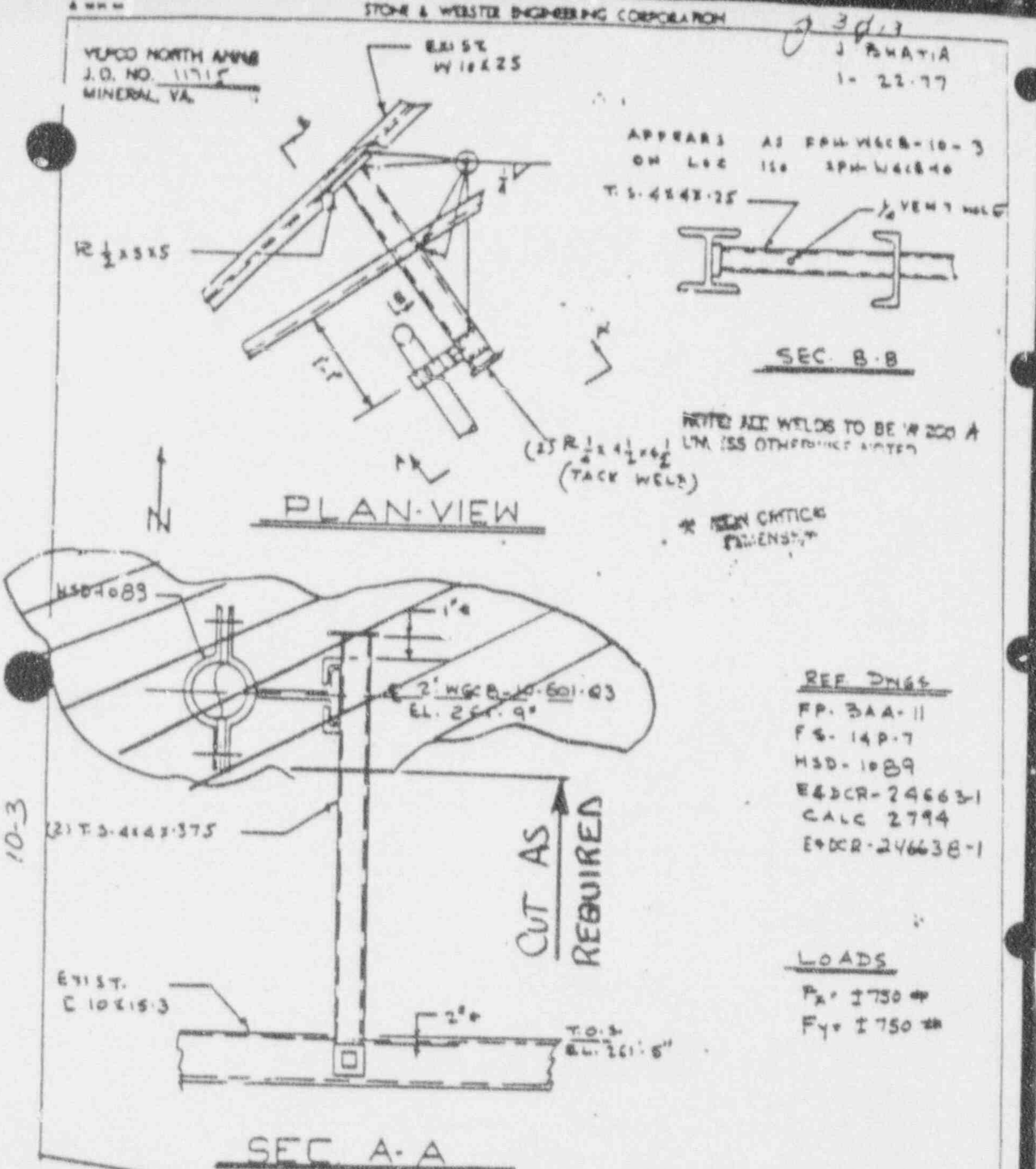
BILL OF MATERIAL

ITEM	QTY	DESCRIPTION	MAT'L
1	4	2 1/2" PROTECTION SADDLE FOR 2" INS, GRINNELL FIG 162	———
2	4	TS 4 X 4 X 1/4" (LENGTH TO SUIT BY FIELD)	A500 GR. B
3	4	STEEL STRIP 1/8" X 3 3/4" X 3 3/4"	A606

ISSUED FOR CONSTRUCTION OF DC-90-13-1

SAFETY RELATED
DC 90-13-1

	BY	DATE	TITLE	
PREPARED	DON	9/17/92	STEAM GENERATOR BLOWDOWN MODIFICATIONS CUBICLE C PIPE SUPPORT FPH-WGCB-10-3	CLIENT VIRGINIA POWER
CHECKED	BB	9/17/92		PROJECT NORTH ANNA 1
APPROVED	[illegible]	9/17/92		J.O. NO. 02072.0710
				①ISSUE BY SWEC
				DATE
REVISIONS BY	①			NUMBER SHEET 1 OF 2
DATE				N9013-1-M-1MFSK3017A-1



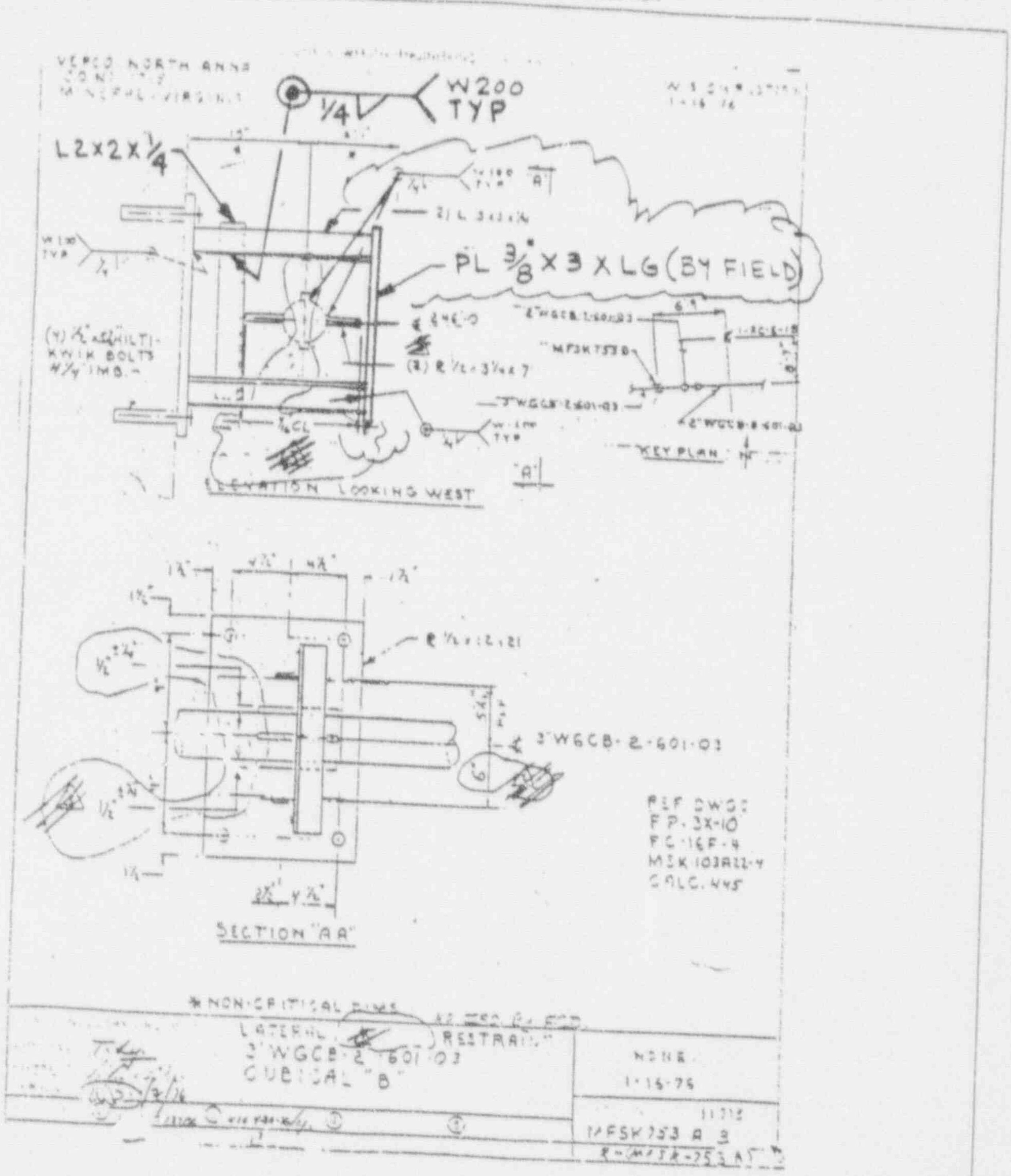
POWER INDUSTRY GROUP	TITLE ~~VERTICAL & LATERAL~~	
CHECKED Calc 1-26-77	RESTRAINT FOR 2"-WGCB-10-601-Q3	SCALE NONE
CORRECT	REACT. CTMT., CUBICLE-C	DATE 1-22-77
APPROVED		SKETCH NUMBER 11715

ISSUED FOR CONSTRUCTION OF DC-90-13-1

SAFETY RELATED
DC 90-13-1

	BY	DATE	TITLE	CLIENT VIRGINIA POWER
PREPARED		9/17/92	STEAM GENERATOR BLOWDOWN	PROJECT NORTH ANNA 1
CHECKED	RBB	9/17/92	MODIFICATIONS	J.O. NO. 02072.0710
APPROVED		9/17/92	CUBICLE C	ISSUE BY SWEC
			PIPE SUPPORT FPH-WGCB-10-3	DATE
REVISIONS BY	0	1 2 3 4		NUMBER SHEET 2 OF 2
DATE				N9013-1-M-1MFSK 3047A-1

STONE & WEBSTER ENGINEERING CORPORATION

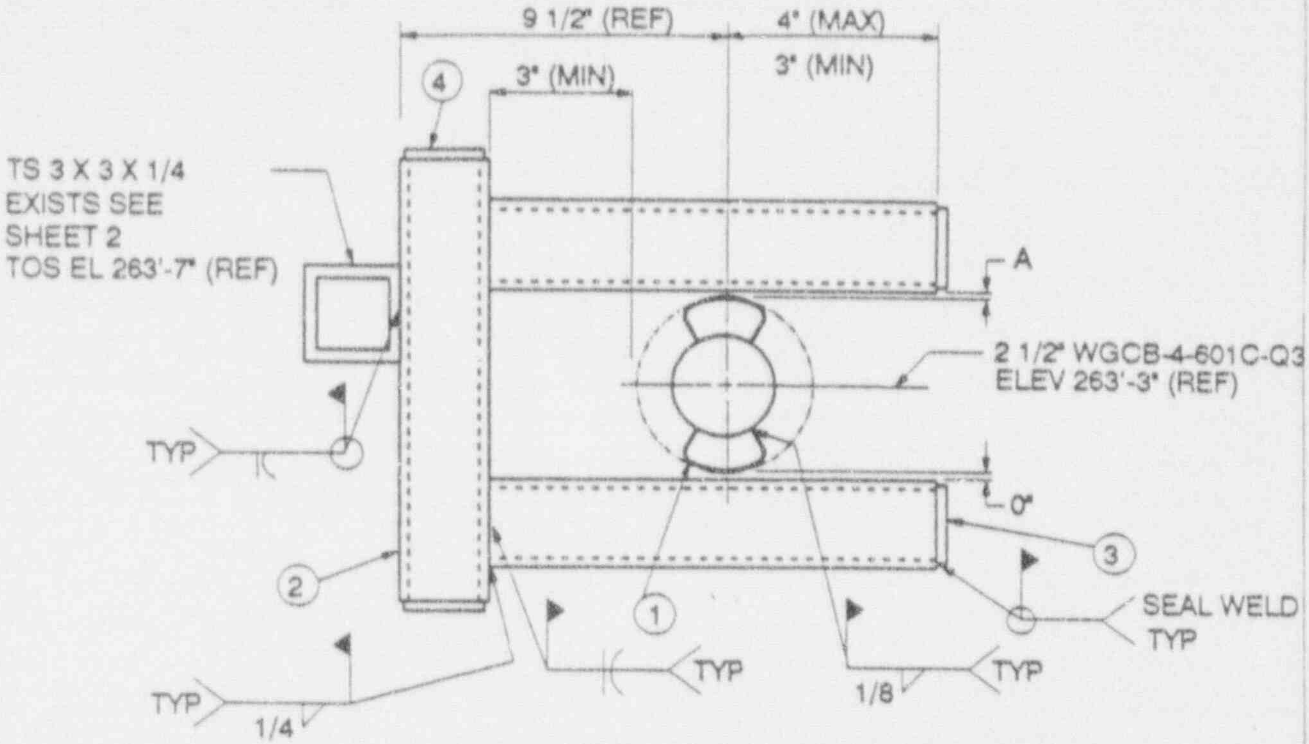


ISSUED FOR CONSTRUCTION OF DC-90-13-1

SAFETY RELATED
DC 90-13-1

	BY	DATE	TITLE			CLIENT	VIRGINIA POWER
PREPARED	DOM	4/17/92	STEAM GENERATOR BLOWDOWN MODIFICATIONS CUBICLE B			PROJECT	NORTH ANNA 1
CHECKED	RAB	4/17/92				J.O. NO.	02072.0710
APPROVED	DBM	4/17/92	PIPE SUPPORT 1-WGCB-R-MFSK-753A			①ISSUE BY	SWEC
REVISIONS BY	①	①	①	①	①	DATE	
DATE						NUMBER	SHEET 1 OF 1
							N9013-1-M-1MFSK753A-0

STONE & WEBSTER ENGINEERING CORPORATION



ELEV LOOKING NORTH WEST

NOTES:

1. FOR SUPPORT LOCATION SEE DWG. N9013-1-M-601
2. FOR GENERAL NOTES REFER TO DWG. N9013-1-M-700.01
3. FIELD TO REMOVE EXISTING PIPE ATTACHMENT AND ANGLE FRAME
4. CLEARANCES 1/16" ≤ (A) ≤ 5/32", FIELD SHIM TO SUIT.

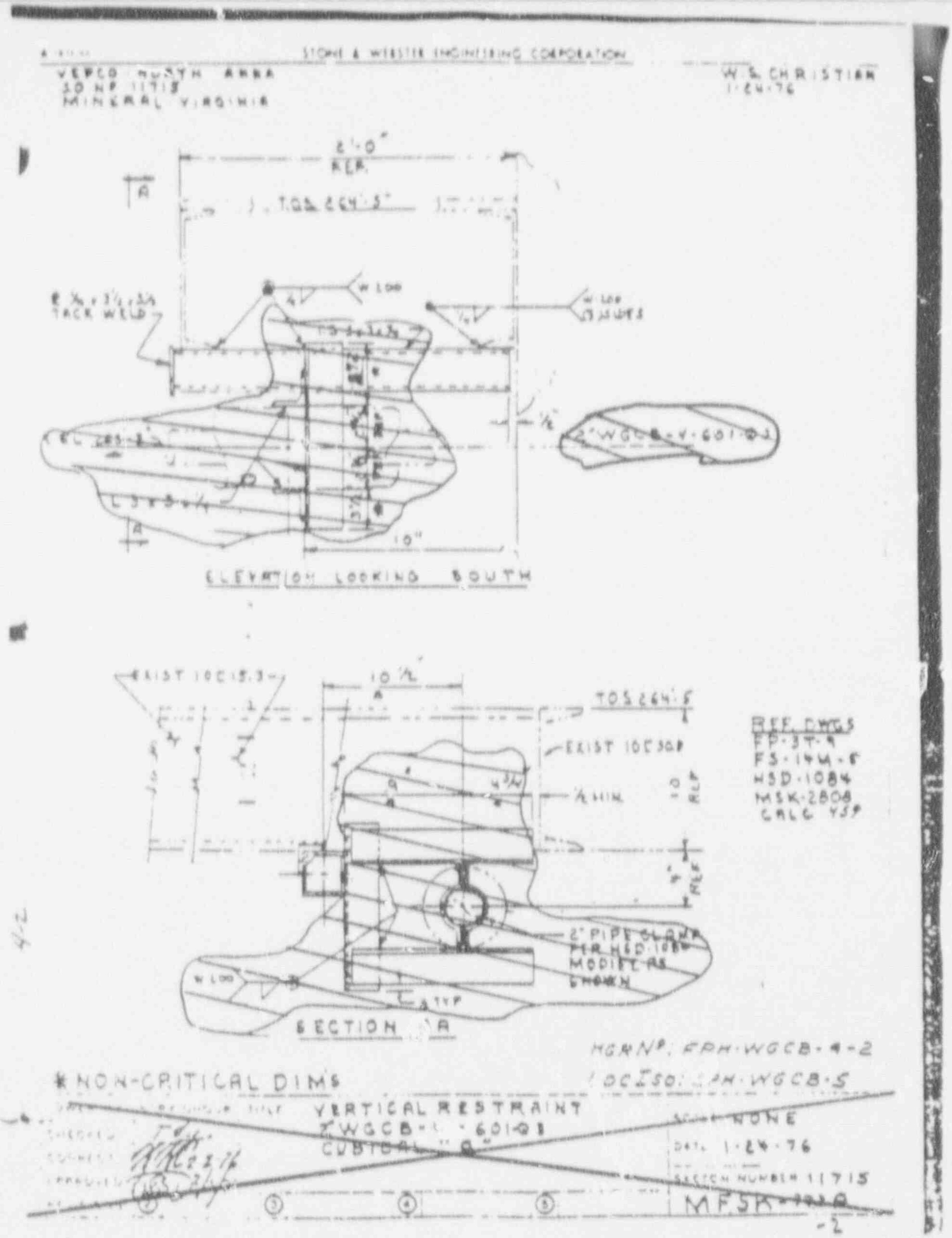
BILL OF MATERIAL

ITEM	QTY	DESCRIPTION	MAT'L
1	2	2 1/2" PROTECTION SADDLE FOR 2" INS, GRINNELL FIG 162	———
2	3	TS 3 X 3 X 1/4" (LENGTH TO SUIT BY FIELD)	A500 GR. B
3	3	STEEL STRIP 1/8" X 2 3/4" X 2 3/4" W/ 1/4" VENT HOLE	A606
4	1	STEEL STRIP 1/8" X 2 3/4" X 2 3/4"	A606

SAFETY RELATED
DC 90-13-1

ISSUED FOR CONSTRUCTION OF DC-90-13-1

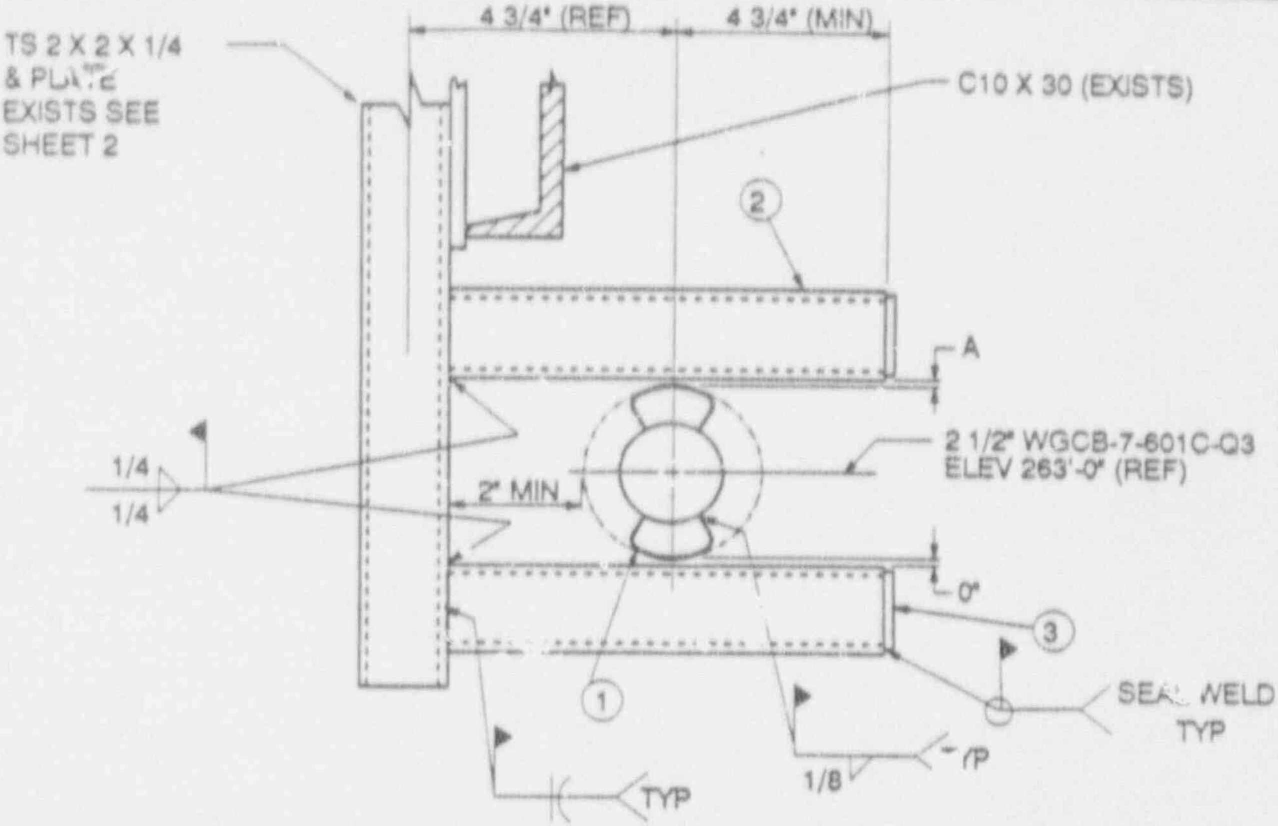
	BY	DATE	TITLE		
PREPARED	DOM	9/17/92	STEAM GENERATOR BLOWDOWN MODIFICATIONS CUBICLE A PIPE SUPPORT FPH-WGCB-4-2	CLIENT	VIRGINIA POWER
CHECKED	[illegible]	9/17/92		PROJECT	NORTH ANNA 1
APPROVED	[illegible]	9/17/92		J.O. NO.	02072.0710
				ISSUE BY	SWEC
				DATE	
REVISIONS BY	0			NUMBER	SHEET 1 OF 2 N9013-1-M-1MFSK773A-1
DATE					



ISSUED FOR CONSTRUCTION OF DC-90-13-1

SAFETY RELATED
DC 90-13-1

	BY	DATE	TITLE	
PREPARED	KCM	9/17/92	STEAM GENERATOR BLOWDOWN MODIFICATIONS CUBICLE A PIPE SUPPORT FPH-WGCB-4-2	CLIENT VIRGINIA POWER
CHECKED	RGB	9/17/92		PROJECT NORTH ANNA 1
APPROVED	BCM	9/17/92		J.O. NO. 02072.0710
				ISSUE BY SWEC
				DATE
REVISIONS BY	⓪	①	② ③ ④	NUMBER SHEET 2 OF 2
DATE				N9013-1-M-MFSK773A-1



ELEV LOOKING NORTH

NOTES:

1. FOR SUPPORT LOCATION SEE DWG. N9013-1-M-602
2. FOR GENERAL NOTES REFFR TO DWG. N9013-1-M-700.01
3. FIELD TO REMOVE EXISTING PIPE ATTACHMENT AND TUBE STEEL FRAME
4. CLEARANCES 1/16" ≤ (A) ≤ 5/32", FIELD SHIM TO SUIT.

BILL OF MATERIAL

ITEM	QTY	DESCRIPTION	MAT'L
1	2	2 1/2" PROTECTION SADDLE FOR 2" INS, GRINNELL FIG 162	———
2	2	TS 2 X 2 X 1/4" (LENGTH TO SUIT BY FIELD)	A500 GR. B
3	2	STEEL STRIP 1/8" X 1 3/4" X 1 3/4" W/ 1/4" VENT HOLE	A606

ISSUED FOR CONSTRUCTION OF DC-90-13-1

SAFETY RELATED
DC 90-13-1

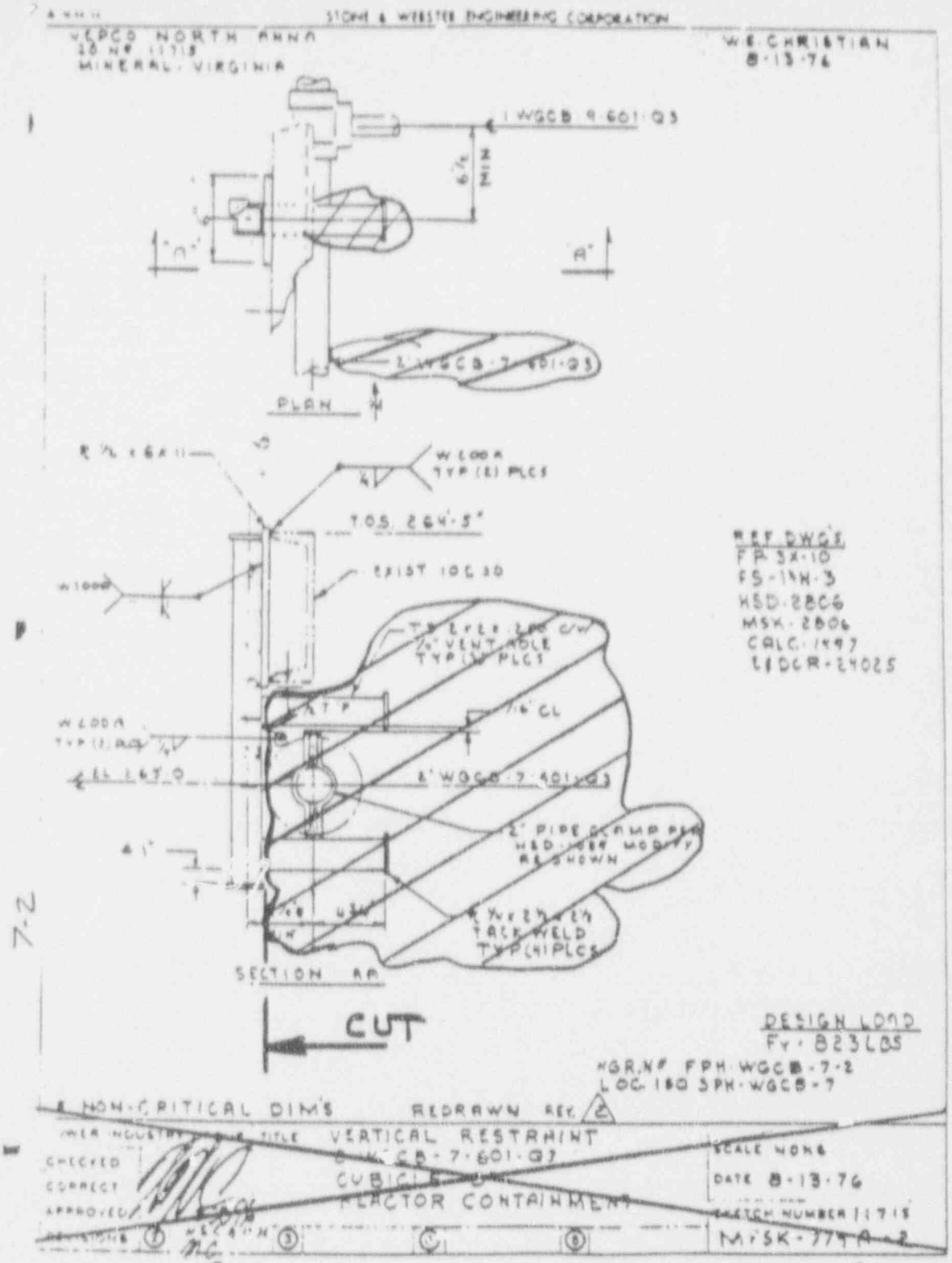
	BY	DATE
PREPARED	RSH	9-17-92
CHECKED	GB	9/17/92
APPROVED		9/17/92

TITLE
STEAM GENERATOR BLOWDOWN
MODIFICATIONS
CUBICLE B
PIPE SUPPORT FPH-WGCB-7-2

CLIENT VIRGINIA POWER
PROJECT NORTH ANNA 1
J.O. NO. 02072.0710
ISSUE BY SWEC
DATE

REVISIONS BY
DATE

NUMBER SHEET 1 OF 2
N9013-1-M-1MFSK774A-1

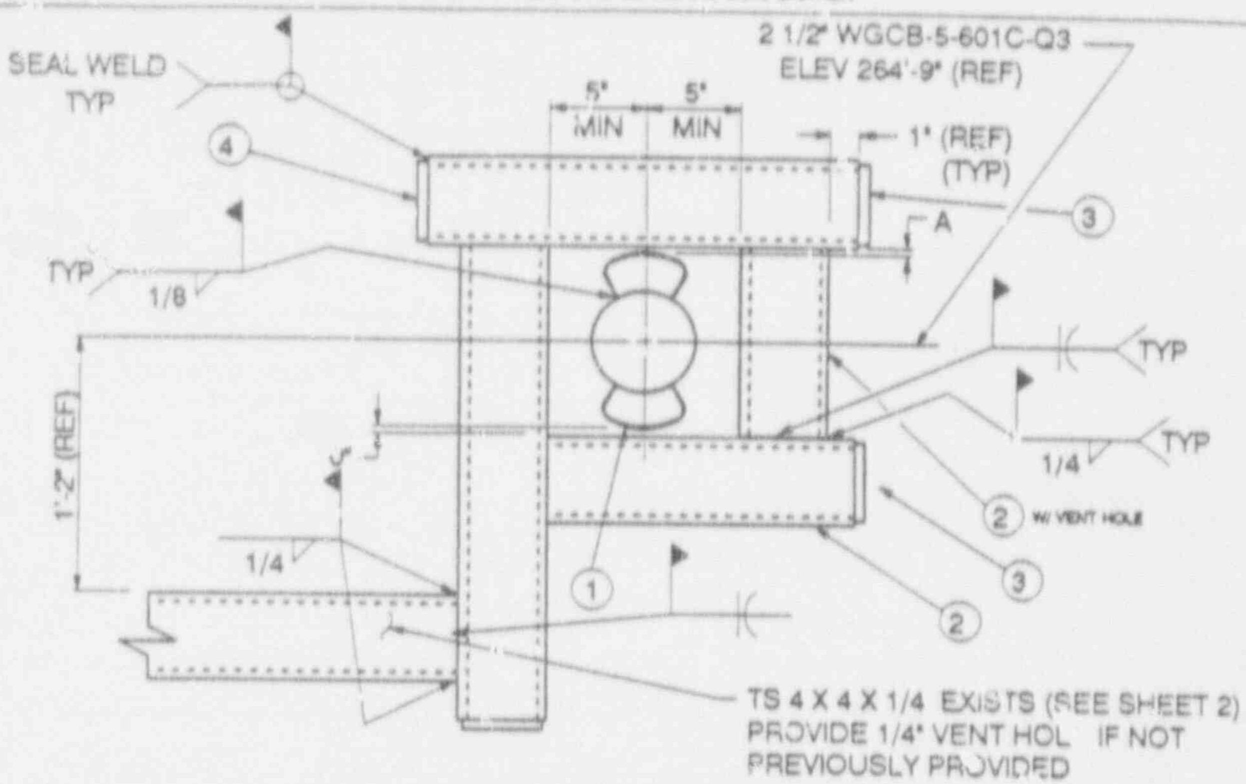


ISSUED FOR CONSTRUCTION OF DC-90-13-1

SAFETY RELATED
DC 90-13-1

	BY	DATE	TITLE			CLIENT VIRGINIA POWER
PREPARED	R.E.H.	9/13/92	STEAM GENERATOR BLOWDOWN MODIFICATIONS			PROJECT NORTH ANNA 1
CHECKED		9/17/92	CUBICLE B			J.O. NO. 020 2.0710
APPROVED		9/17/92	PIPE SUPPORT FPH-WGCB-7-2			ISSUE BY SWEC
						DATE
REVISIONS BY	0	0	0	0	0	NUMBER SHEET 2 OF 2
DATE						N9013-1-M-1MFSK774A-1

STONE & WEBSTER ENGINEERING CORPORATION



ELEV LOOKING NORTH EAST

NOTES:

1. FOR SUPPORT LOCATION SEE DWG. N9013-1-M-601
2. FOR GENERAL NOTES REFER TO DWG. N9013-1-M-700.01
3. FIELD TO REMOVE EXISTING PIPE ATTACHMENT AND ANGLE BOX FRAME, CUT BACK EXISTING TS 4 X 4 X 1/4 AND REFABRICATE AS SHOWN
4. CLEARANCES 1/16" ≤ (A) ≤ 5/32", FIELD SHIM TO SUIT.

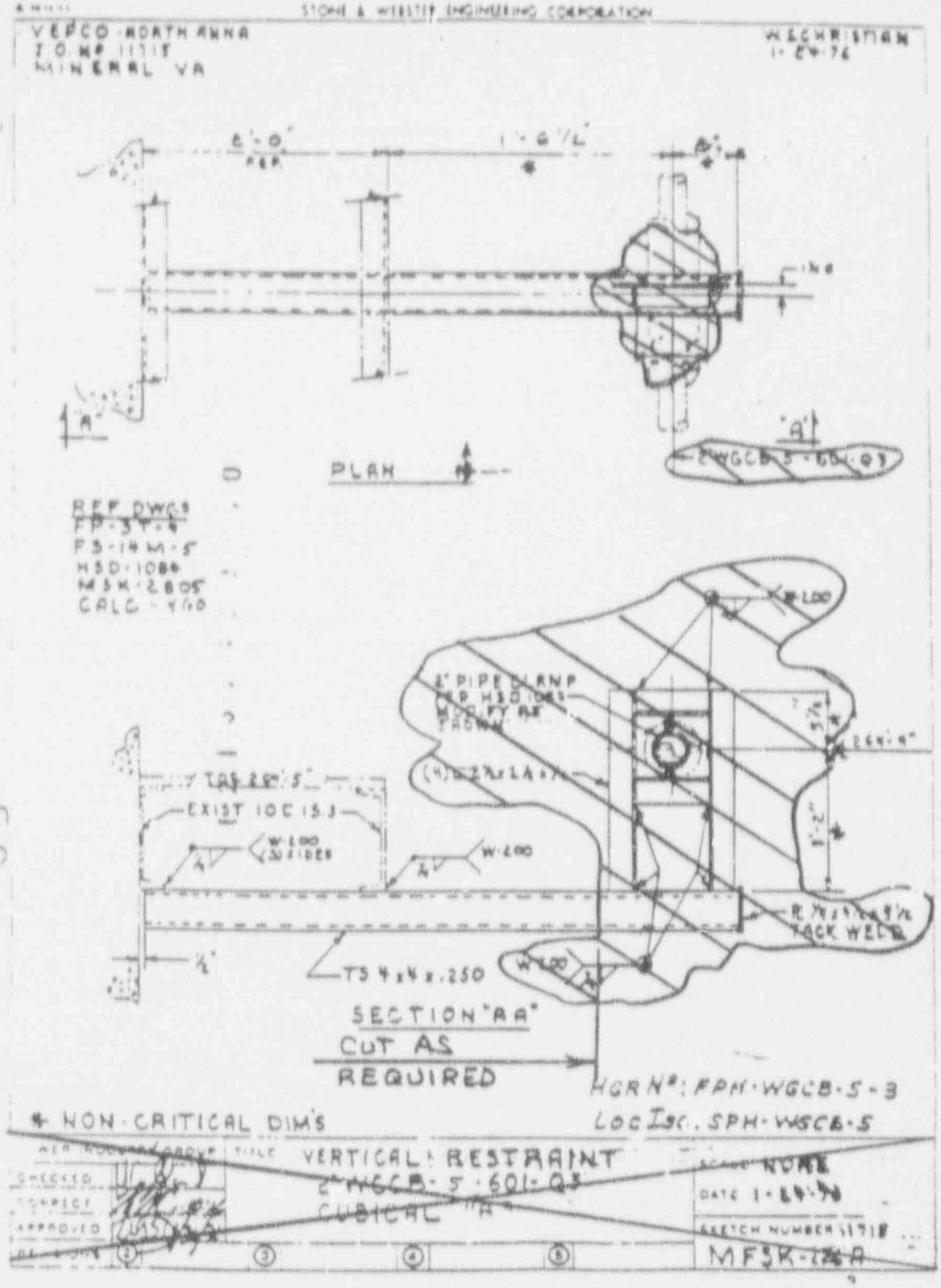
BILL OF MATERIAL

ITEM	QTY	DESCRIPTION	MAT'L
1	2	2 1/2" PROTECTION SADDLE FOR 2" INS, GRINNELL FIG 162	———
2	4	TS 4 X 4 X 1/4" (LENGTH TO SUIT BY FIELD)	A500 GR. B
3	3	STEEL STRIP 1/8" X 3 3/4" X 3 3/4" W/ 1/4" VENT HOLE	A606
4	1	STEEL STRIP 1/8" X 3 3/4" X 3 3/4"	A606

ISSUED FOR CONSTRUCTION CF DC-90-13-1

SAFETY RELATED
DC 90-13-1

	BY	DATE	TITLE	
PREPARED	DOM	9/17/92	STEAM GENERATOR BLOWDOWN MODIFICATIONS CUBICLE A PIPE SUPPORT FPH-WGCB-5-3	CLIENT VIRGINIA POWER
CHECKED	[illegible]	9/17/92		PROJECT NORTH ANNA 1
APPROVED	[illegible]	9/17/92		J.O. NO. 02072.0710
				ISSUE BY SWEC
				DATE
REVISIONS BY				NUMBER SHEET 1 OF 2
DATE				N9013-1-M-1MFSK776A-1

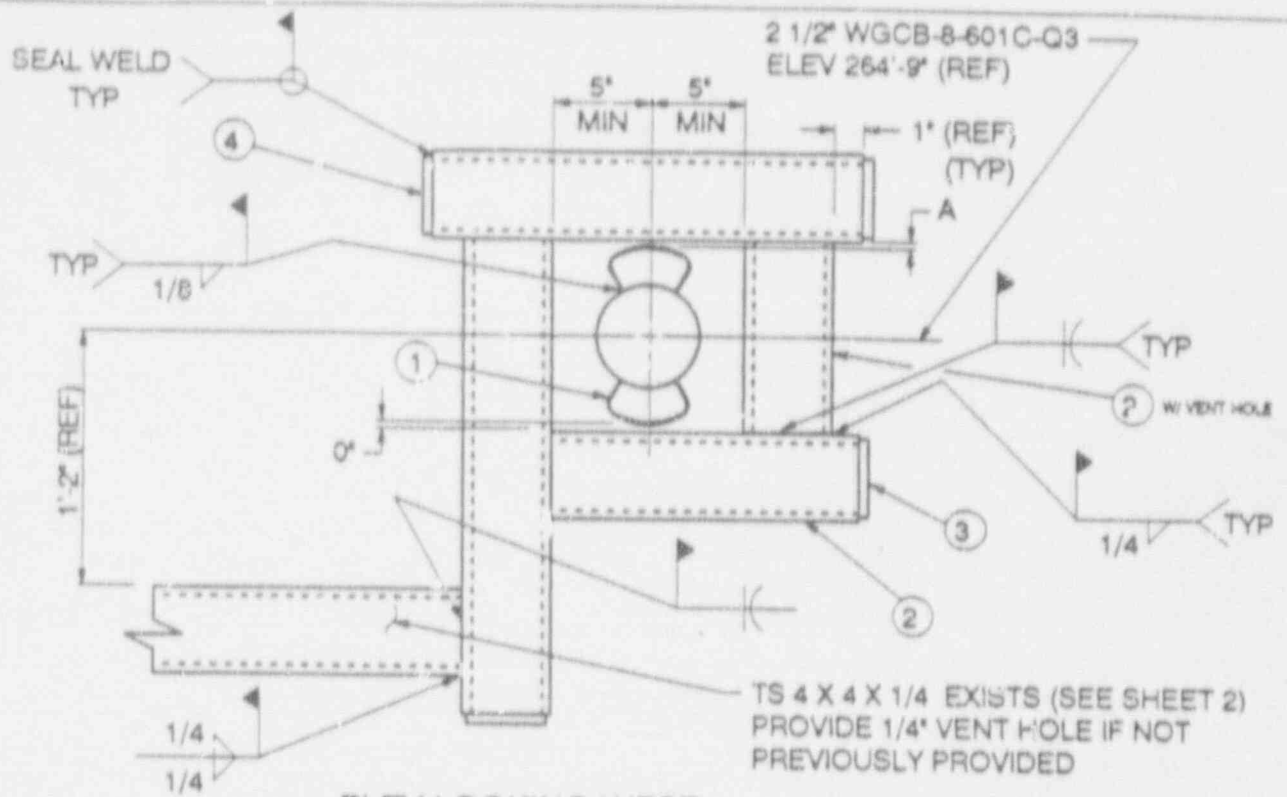


5-3

ISSUED FOR CONSTRUCTION OF DC-90-13-1

SAFETY RELATED
DC 90-13-1

	BY	DATE	TITLE	
PREPARED	DCM	9/17/92	STEAM GENERATOR BLOWDOWN MODIFICATIONS CUBICLE A PIPE SUPPORT FPH-WGCB-5-3	CLIENT VIRGINIA POWER
CHECKED	RAB	9/17/92		PROJECT NORTH ANNA 1
APPROVED		9/17/92		J.O. NO. 02072.0710
				ISSUE BY SWEC
				DATE
REVISIONS BY	⓪	① ② ③ ④		NUMBER SHEET 2 OF 2
DATE				N9013-1-M-IMFSK776A-1



ELEV LOOKING WEST

NOTES:

1. FOR SUPPORT LOCATION SEE DWG. N9013-1-M-602
2. FOR GENERAL NOTES REFER TO DWG. N9013-1-M-700.01
3. FIELD TO REMOVE EXISTING PIPE ATTACHMENT AND ANGLE BOX FRAME. CUT BACK EXISTING TS 4 X 4 X 1/4 AND REFABRICATE AS SHOWN
4. CLEARANCES 1/16" ≤ (A) ≤ 5/32", FIELD SHIM TO SUIT.

BILL OF MATERIAL

ITEM	QTY	DESCRIPTION	MAT'L
1	2	2 1/2" PROTECTION SADDLE FOR 2" INS, GRINNELL FIG 162	———
2	4	TS 4 X 4 X 1/4" (LENGTH TO SUIT BY FIELD)	A500 GR. B
3	3	STEEL STRIP 1/8" X 3 3/4" X 3 3/4" W/ 1/4" VENT HOLE	A606
4	1	STEEL STRIP 1/8" X 3 3/4" X 3 3/4"	A606

ISSUED FOR CONSTRUCTION OF DC-90-13-1

SAFETY RELATED
DC 90-13-1

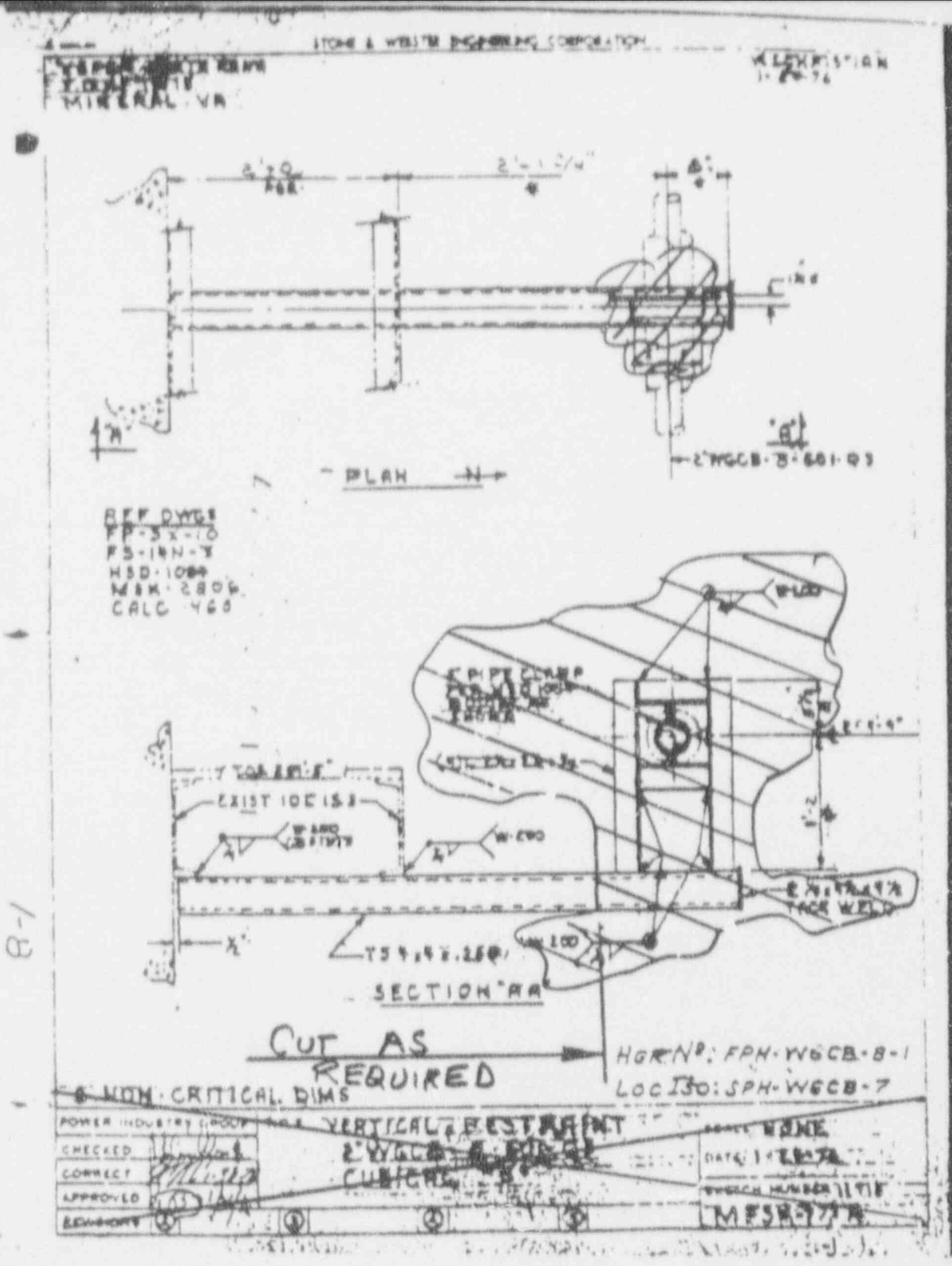
	BY	DATE
PREPARED	DCH	9/17/92
CHECKED	BRB	9/17/92
APPROVED	[illegible]	9/17/92

TITLE
STEAM GENERATOR BLOWDOWN
MODIFICATIONS
CUBICLE B
PIPE SUPPORT FPH-WGCB-8-1

CLIENT VIRGINIA POWER
PROJECT NORTH ANNA 1
J.O. NO. 02072.0710
ISSUE BY SWEC
DATE

REVISIONS BY ⓪ ① ② ③ ④
DATE

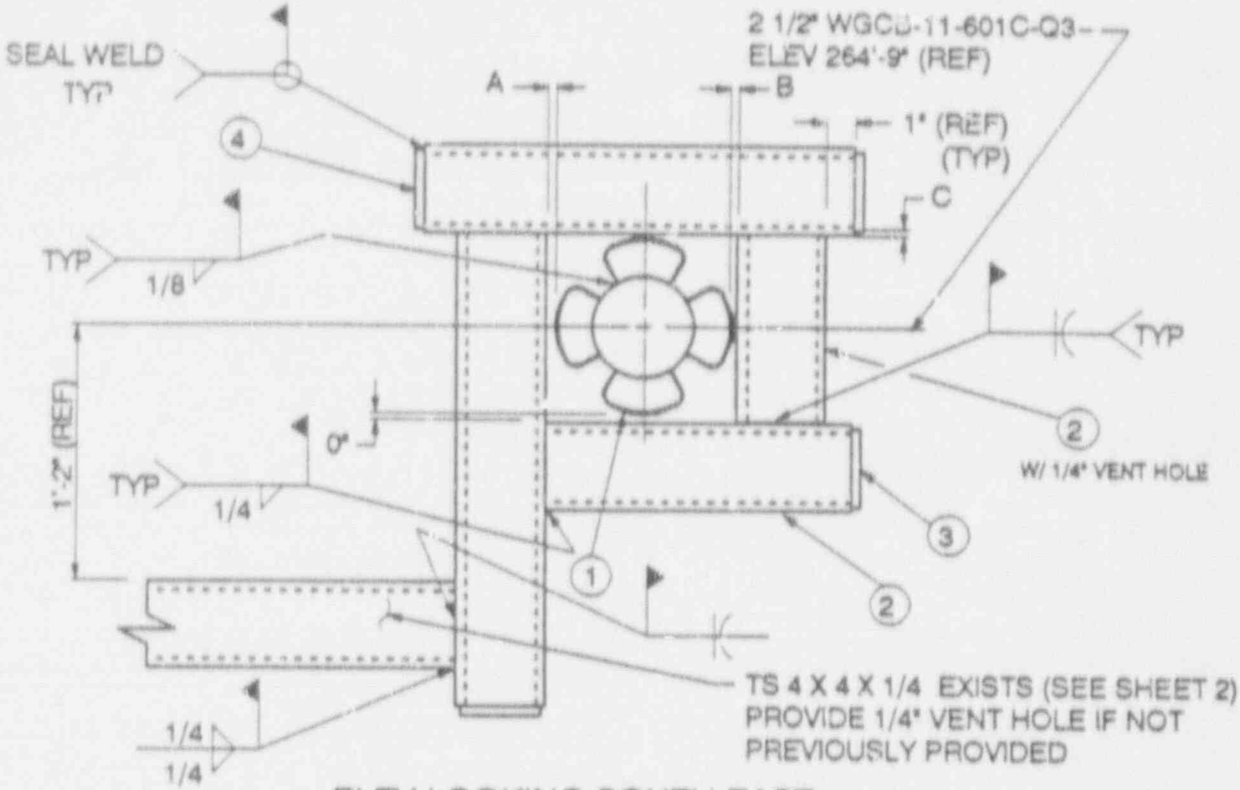
NUMBER SHEET 1 OF 2
N9013-1-M-1MFSK777A-1



ISSUED FOR CONSTRUCTION OF DC-90-13-1

SAFETY RELATED
DC 90-13-1

	BY	DATE	TITLE			CLIENT VIRGINIA POWER
PREPARED	KKK	9/17/92	STEAM GENERATOR BLOWDOWN MODIFICATIONS			PROJECT NORTH ANNA 1
CHECKED	RAB	9/17/92	CUBICLE B			J.O. NO. 02072.0710
APPROVED	[illegible]	9/17/92	PIPE SUPPORT FPH-WGCB-B-1			ISSUE BY SWEC
						DATE
REVISIONS BY	0	0	0	0	0	NUMBER SHEET 2 OF 2
DATE						N9013-1-M-IMFSK777A-1



ELEV LOOKING SOUTH EAST

NOTES:

1. FOR SUPPORT LOCATION SEE DWG. N9013-1-M-603
2. FOR GENERAL NOTES REFER TO DWG. N9013-1-M-700.01
3. FIELD TO REMOVE EXISTING PIPE ATTACHMENT AND ANGLE BOX FRAME, CUT BACK EXISTING TS 4 X 4 X 1/4 AND REFABRICATE AS SHOWN
4. CLEARANCES 1/16" ≤ (A+B) , (C) ≤ 5/32", FIELD SHIM TO SUIT.

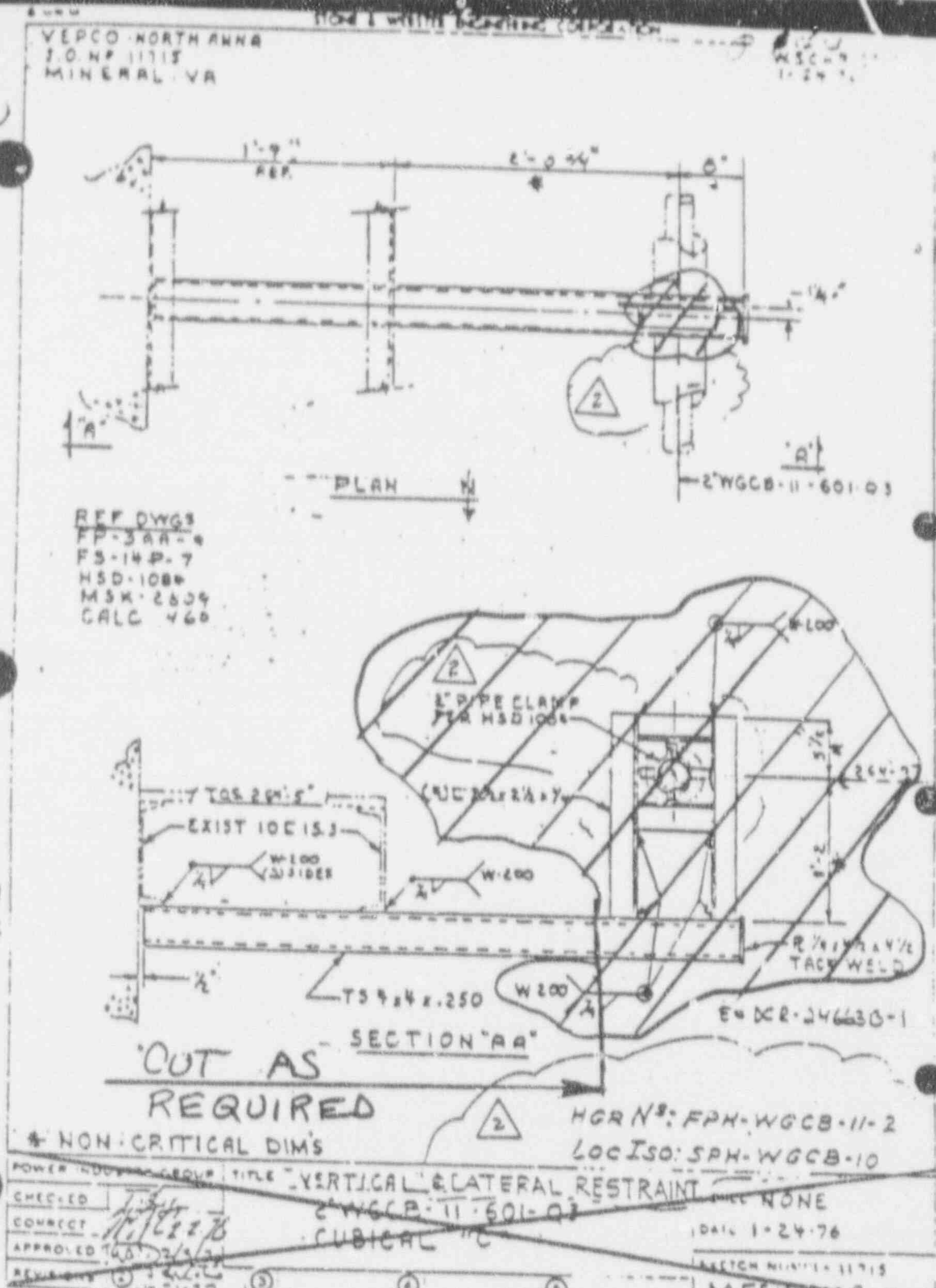
BILL OF MATERIAL

ITEM	QTY	DESCRIPTION	MAT'L
1	4	2 1/2" PROTECTION SADDLE FOR 2" INS, GRINNELL FIG 162	————
2	4	TS 4 X 4 X 1/4" (LENGTH TO SUIT BY FIELD)	A500 GR. B
3	3	STEEL STRIP 1/8" X 3 3/4" X 3 3/4" W/ 1/4" VENT HOLE	A606
4	1	STEEL STRIP 1/8" X 3 3/4" X 3 3/4"	A606

SAFETY RELATED
DC 90-13-1

ISSUED FOR CONSTRUCTION OF DC-90-13-1

	BY	DATE	TITLE	
PREPARED	DGH	9/17/92	STEAM GENERATOR BLOWDOWN	CLIENT VIRGINIA POWER
CHECKED	RG	9/17/92	MODIFICATIONS	PROJECT NORTH ANNA 1
APPROVED	[illegible]	9/17/92	CUBICLE C	J.O. NO. 02072.0710
			PIPE SUPPORT FPH-WGCB-11-2	ISSUE BY SWEC
REVISIONS BY	0			DATE
DATE				NUMBER SHEET 1 OF 2 N9013-1-M-1MFSK778A-1

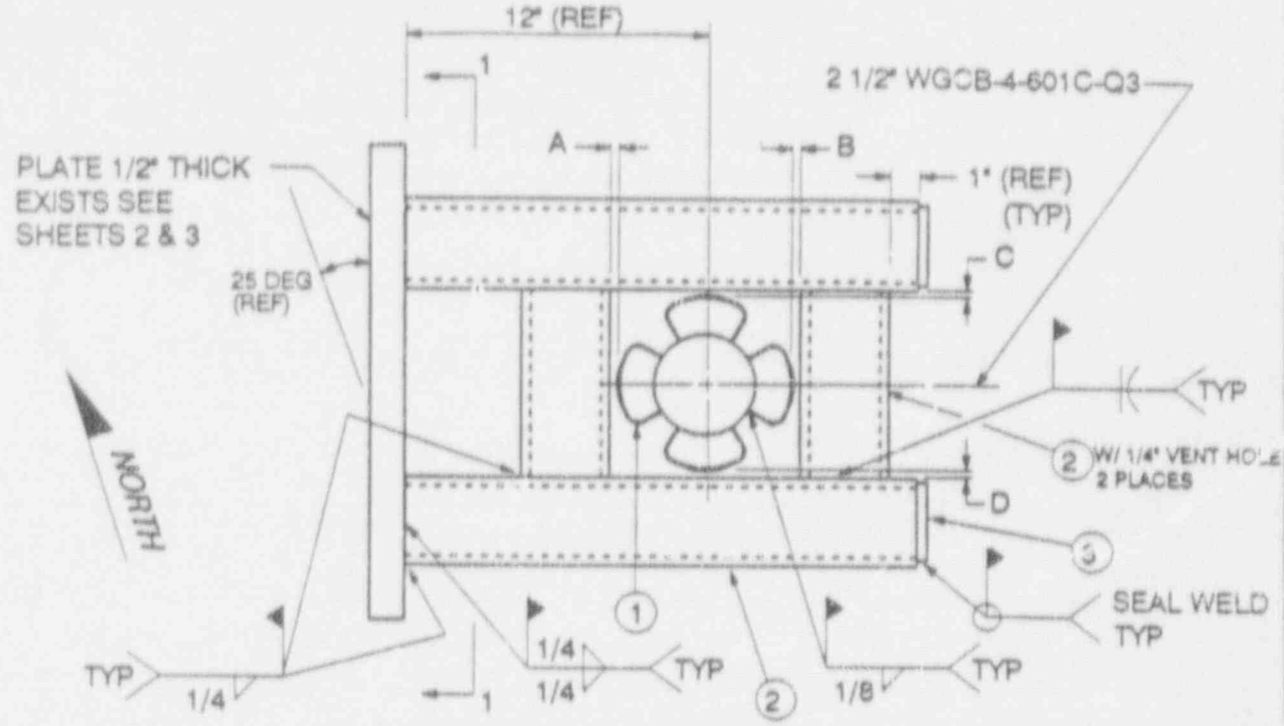


11-2

ISSUED FOR CONSTRUCTION OF DC-90-13-1

SAFETY RELATED
DC 90-13-1

	BY	DATE	TITLE			CLIENT VIRGINIA POWER
PREPARED	KCk	9/17/92	STEAM GENERATOR BLOWDOWN MODIFICATIONS CUBICLE C PIPE SUPPORT FPH-WGCB-11-2			PROJECT NORTH ANNA 1
CHECKED	RAB	9/17/92				J.O. NO. 02072.0710
APPROVED	DBD	9/17/92				ISSUE BY SWEC
REVISIONS BY	①	②	③	④	⑤	DATE
DATE						NUMBER SHEET 2 OF 2 N9013-1-M-IMFSK 778A-1



PLAN VIEW AT ELEV 252'-0" (REF)

NOTES:

1. FOR SUPPORT LOCATION SEE DWG. N9013-1-M-601
2. FOR GENERAL NOTES REFER TO DWG. N9013-1-M-700.01
3. FIELD TO REMOVE EXISTING PIPE ATTACHMENT AND ANGLE BOX FRAME
4. CLEARANCES 1/16" ≤ (A+B) , (C+D) ≤ 5/32", FIELD SHIM TO SUIT.

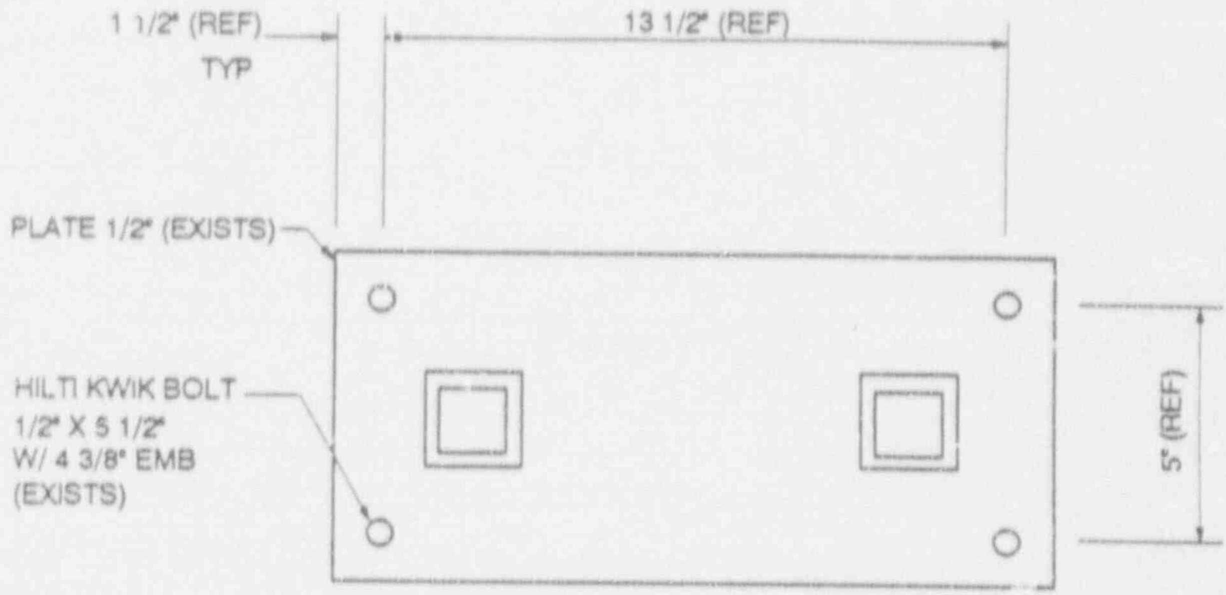
BILL OF MATERIAL

ITEM	QTY	DESCRIPTION	MAT'L
1	4	2 1/2" PROTECTION SADDLE FOR 2" INS, GRINNELL FIG 162	———
2	4	TS 3 X 3 X 1/4" (LENGTH TO SUIT BY FIELD)	A500 GR. B
3	2	STEEL STRIP 1/8" X 2 3/4" X 2 3/4" W/ 1/4" VENT HOLE	A506

ISSUED FOR CONSTRUCTION OF DC-90-13-1

SAFETY RELATED
DC 90-13-1

	BY	DATE	TITLE				
PREPARED	DOM	9/17/92	STEAM GENERATOR BLOWDOWN MODIFICATIONS CUBICLE A PIPE SUPPORT FPH-WGCB-4-1			CLIENT	VIRGINIA POWER
CHECKED	[illegible]	9/17/92				PROJECT	NORTH ANNA 1
APPROVED	[illegible]	9/17/92				J.O. NO.	020?.2.0710
						ISSUE BY	SWEC
						DATE	
REVISIONS BY	0	1	2	3	4	NUMBER	SHEET 1 OF 3
DATE							N9013-1-M-1MFSK780A-1

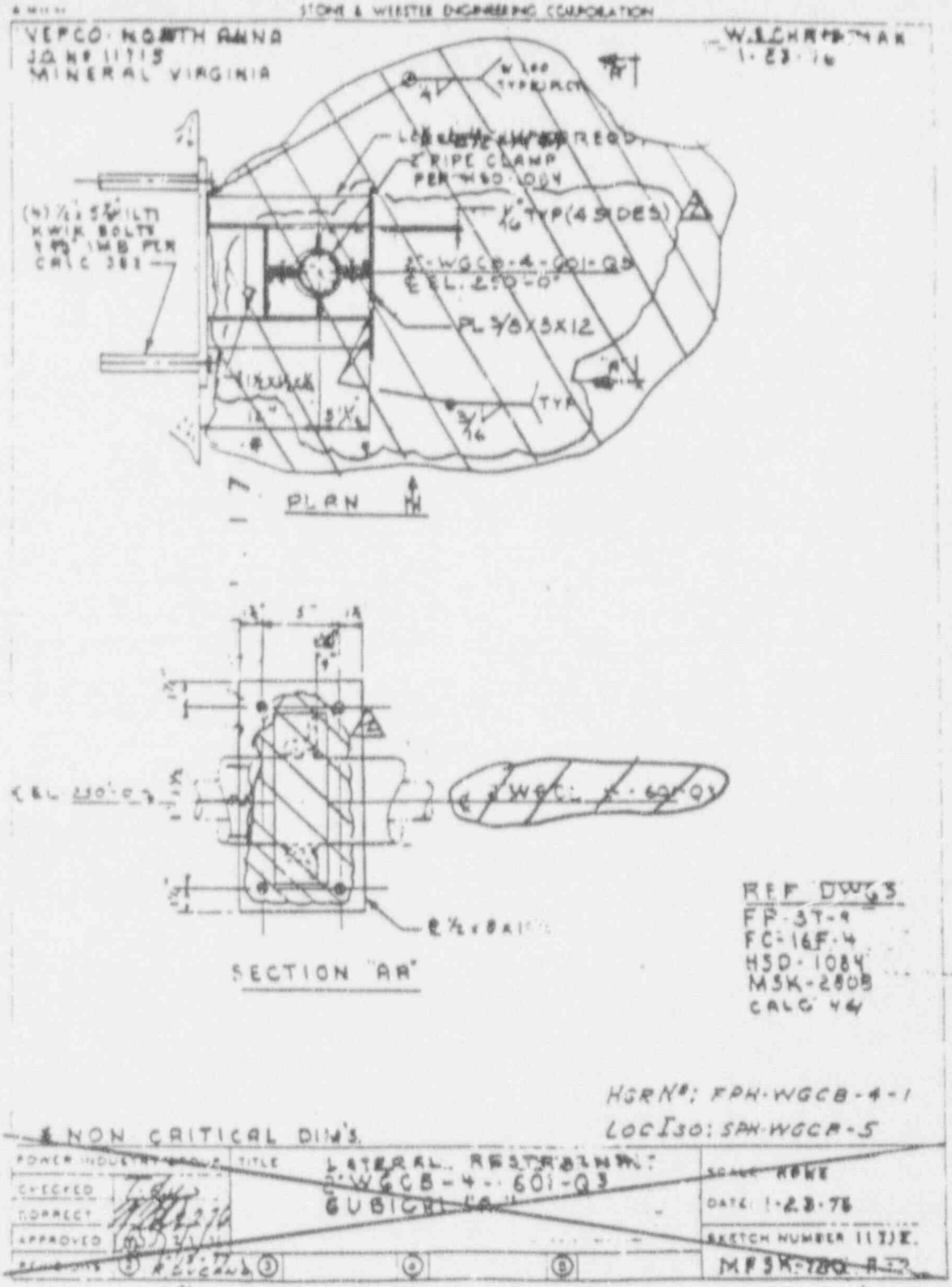


SECTION 1-1

ISSUED FOR CONSTRUCTION OF DC-90-13-1

SAFETY RELATED
DC 90-13-1

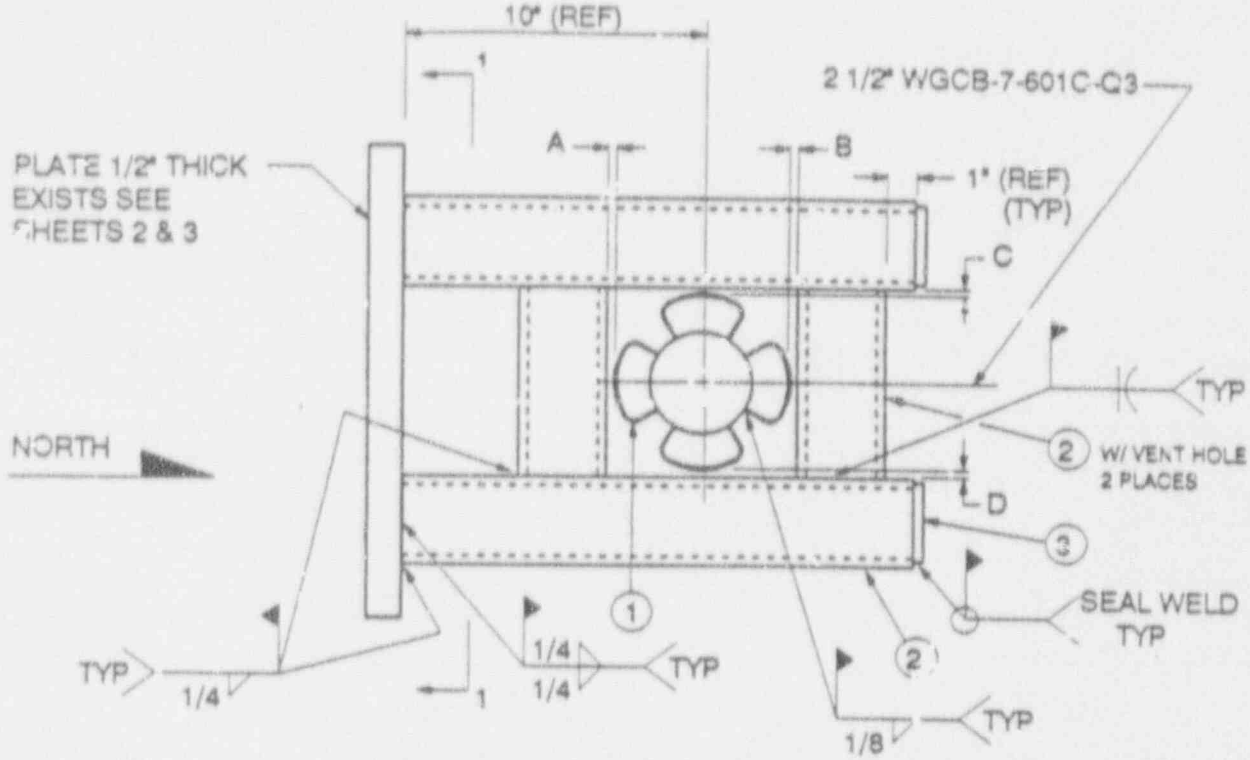
	BY	DATE	TITLE		
PREPARED	DOCY	9/17/92	STEAM GENERA[illegible]R BLOWDOWN	CLIENT	VIRGINIA POWER
CHECKED	[illegible]	9/17/92	MODIFICATIONS	PROJECT	NORTH ANNA 1
APPROVED	[illegible]	9/17/92	CUBICLE A	J.O. NO.	02072.0710
			PIPE SUPPORT FPH-WGCB-4-1	ISSUE BY	SWEC
				DATE	
REVISIONS BY	0	0	0	NUMBER	SHEET 2 OF 3
DATE					N9013-1-M-1MFSK780A-1



ISSUED FOR CONSTRUCTION OF DC-90-13-1

SAFETY RELATED
DC 90-13-1

	BY	DATE	TITLE	
PREPARED	DCM	9/17/92	STEAM GENERATOR BLOWDOWN MODIFICATIONS CUBICLE A PIPE SUPPORT FPH-WGCB-4-1	CLIENT VIRGINIA POWER
CHECKED	RAB	9/17/92		PROJECT NORTH ANNA 1
APPROVED	DCM	9/17/92		J.O. NO. 02072.0710
				ISSUE BY SWEC
				DATE
REVISIONS BY	0	0 0 0 0		NUMBER SHEET 3 OF 3
DATE				N9013-1-M-IMFSK 780A 1



PLAN VIEW AT ELEV 250'-0" (REF)

NOTES:

1. FOR SUPPORT LOCATION SEE DWG. N9013-1-M-602
2. FOR GENERAL NOTES REFER TO DWG. N9013-1-M-700.01
3. FIELD TO REMOVE EXISTING PIPE ATTACHMENT AND ANGLE BOX FRAME
4. CLEARANCES 1/16" ≤ (A+B) , (C+D) ≤ 5/32", FIELD SHIM TO SUIT

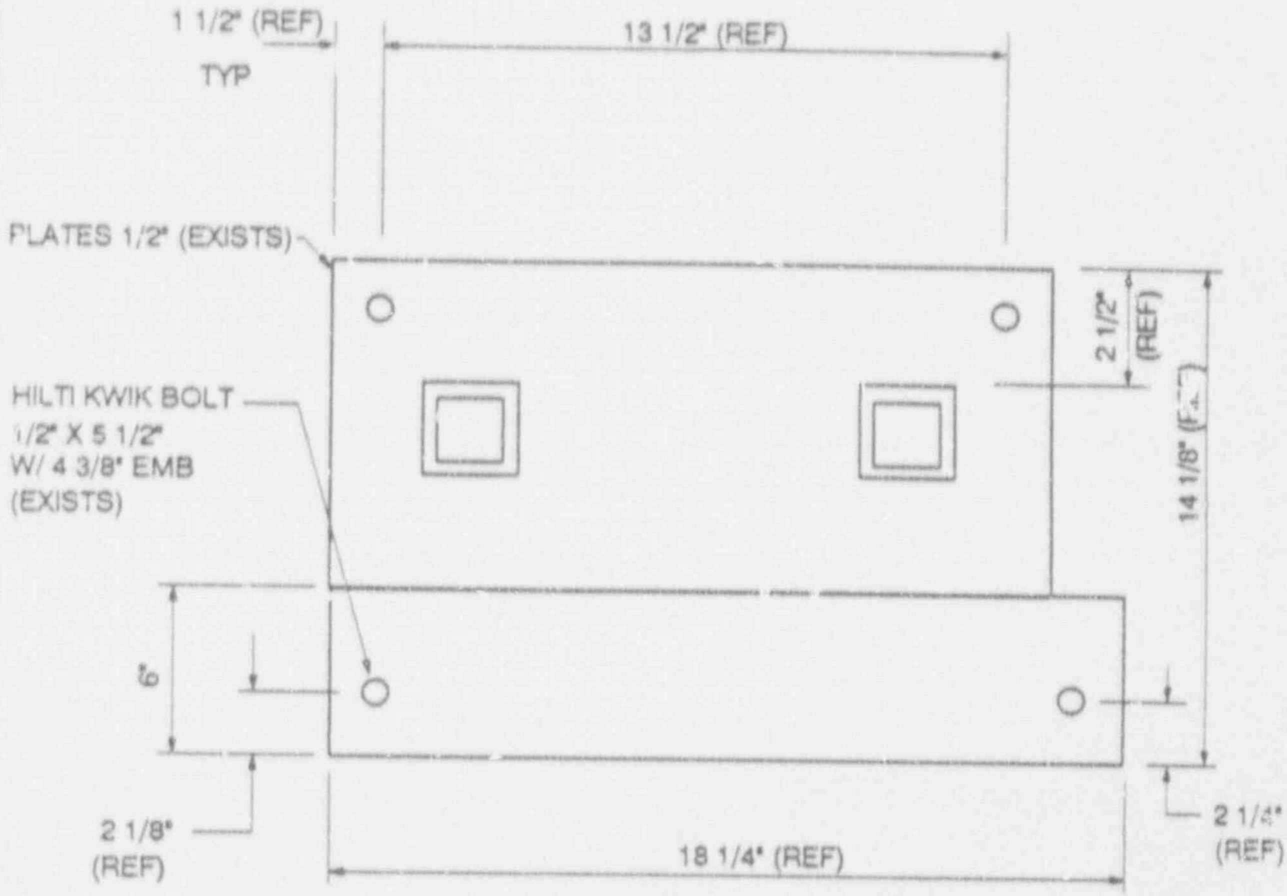
BILL OF MATERIAL

ITEM	QTY	DESCRIPTION	MAT'L
1	4	2 1/2" PROTECTION SADDLE FOR 2" INS, GRINNELL FIG 162	———
2	4	TS 3 X 3 X 1/4" (LENGTH TO SUIT BY FIELD)	A500 GR. B
3	2	STEEL STRIP 1/8" X 2 3/4" X 2 3/4" W/ 1/4" VENT HOLE	A606

SAFETY RELATED
DC 90-13-1

ISSUED FOR CONSTRUCTION OF DC-90-13-1

	BY	DATE	TITLE	CLIENT VIRGINIA POWER
PREPARED	DOM	9/17/92	STEAM GENERATOR BLOWDOWN	PROJECT NORTH ANNA 1
CHECKED		9/17/92	MODIFICATIONS	J.O. NO. 02072.0710
APPROVED		9/17/92	CUBICLE B	ISSUE BY SWEC
			PIPE SUPPORT FPH-WGCB-7-1	DATE
REVISIONS BY	0		1 2 3 4	NUMBER SHEET 1 OF 3
DATE				N9013-1-M-1MFSK781A-1

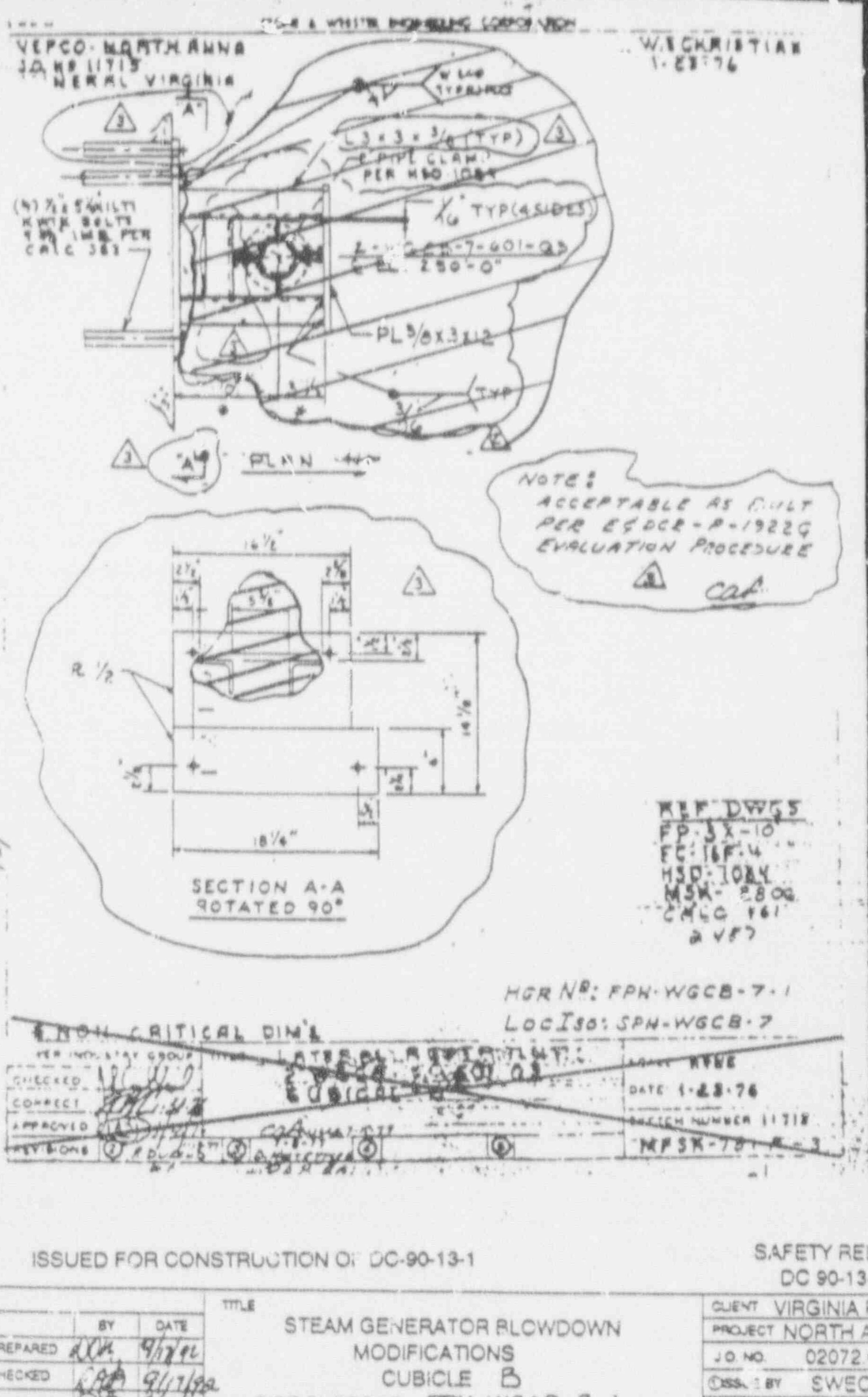


SECTION 1-1

ISSUED FOR CONSTRUCTION OF DC-90-13-1

SAFETY RELATED
DC 90-13-1

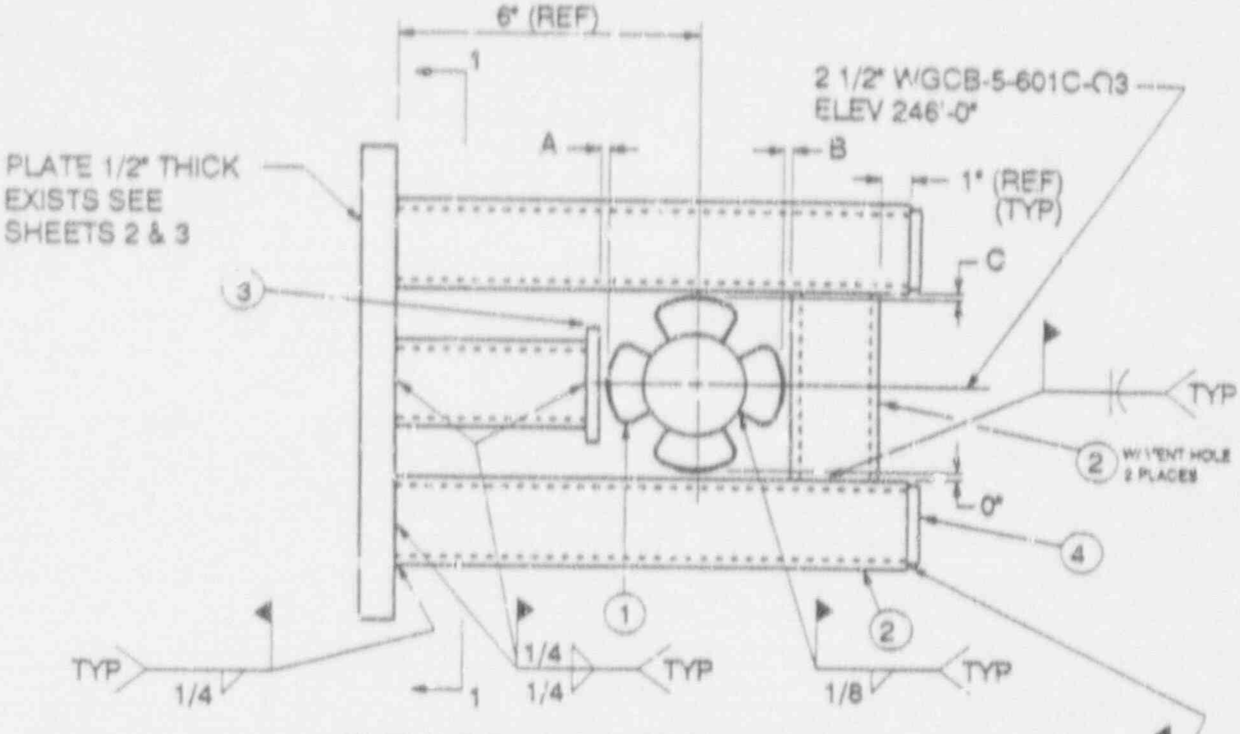
	BY	DATE	TITLE	
PREPARED	DOM	9/17/92	STEAM GENERATOR BLOWDOWN MODIFICATIONS CUBICLE B PIPE SUPPORT FPH-WGCB-7-1	CLIENT VIRGINIA POWER
CHECKED	[illegible]	9/17/92		PROJECT NORTH ANNA 1
APPROVED	[illegible]	9/17/92		J.O. NO. 02072.0710
				ISSUE BY SWEC
				DATE
REVISIONS BY	0			NUMBER SHEET 2 OF 3
DATE				N9013-1-M-1MFSK781A-1



ISSUED FOR CONSTRUCTION OF DC-90-13-1

SAFETY RELATED
DC 90-13-1

	BY	DATE	TITLE	CLIENT VIRGINIA POWER
PREPARED	RCH	9/17/92	STEAM GENERATOR BLOWDOWN MODIFICATIONS	PROJECT NORTH ANNA 1
CHECKED	CAA	9/17/92	CUBICLE B	J.O. NO. 02072.0710
APPROVED	CAA	9/17/92	PIPE SUPPORT FPH-WGCB-7-1	ISSUED BY SWEC
REVISIONS BY	0		0 0 0 0	DATE
DATE				NUMBER SHEET 3 OF 3 N9013-1-M-IMFSK78IA-1



ELEV LOOKING NORTH EAST

SEAL WELD
TYP

NOTES:

1. FOR SUPPORT LOCATION SEE DWG. N9013-1-M-601
2. FOR GENERAL NOTES REFER TO DWG. N9013-1-M-700.01
3. FIELD TO REMOVE EXISTING PIPE ATTACHMENT AND ANGLE BOX FRAME
4. CLEARANCES 1/16" ≤ (A+B) , (C) ≤ 5/32", FIELD SHIM TO SUIT.

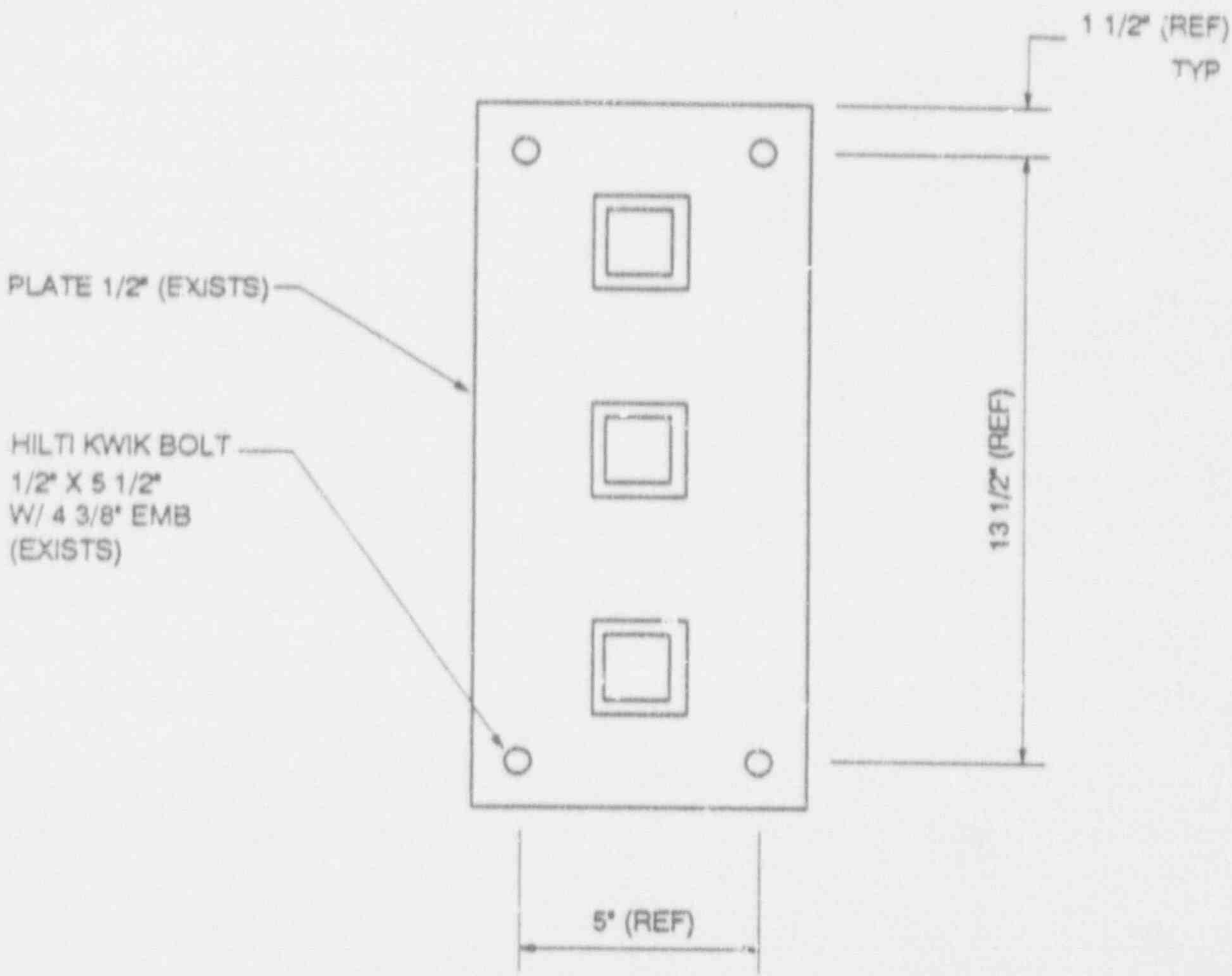
BILL OF MATERIAL

ITEM	QTY	DESCRIPTION	MAT'L
1	4	2 1/2" PROTECTION SADDLE FOR 2" INS, GRINNELL FIG 162	————
2	4	TS 2 X 2 X 1/4" (LENGTH TO SUIT BY FIELD)	A500 GR. B
3	1	PLATE 1/2" X 3" X 3"	A36
4	2	STEEL STRIP 1/8" X 1 3/4" X 1 3/4" W/ 1/4" VENT HOLE	A606

ISSUED FOR CONSTRUCTION OF DC-90-13-1

SAFETY RELATED
DC 90-13-1

	BY	DATE	TITLE	
PREPARED	Dom	9/17/92	STEAM GENERATOR BLOWDOWN MODIFICATIONS CUBICLE A PIPE SUPPORT FPH-WGCB-5-1	CLIENT VIRGINIA POWER
CHECKED	[illegible]	9/17/92		PROJECT NORTH ANNA 1
APPROVED	[illegible]	9/17/92		J.O. NO. 02072.0710
				ISSUE BY SWEC
				DATE
REVISIONS BY	0	1 2 3 4		NUMBER SHEET 1 OF 3 N9013-1-M-1MFSK783A-1
DATE				

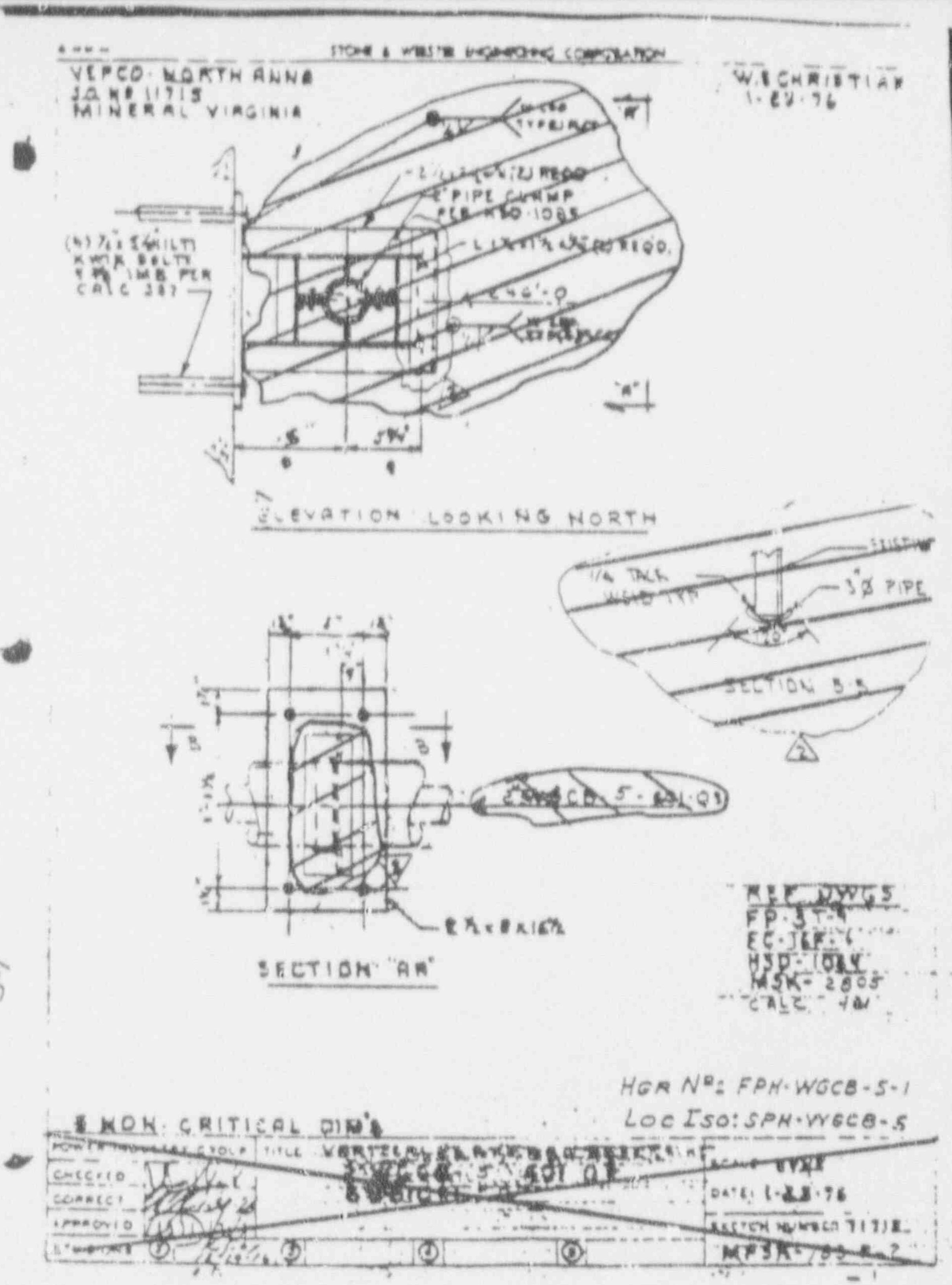


SECTION 1-1

ISSUED FOR CONSTRUCTION OF DC-90-13-1

SAFETY RELATED
DC 90-13-1

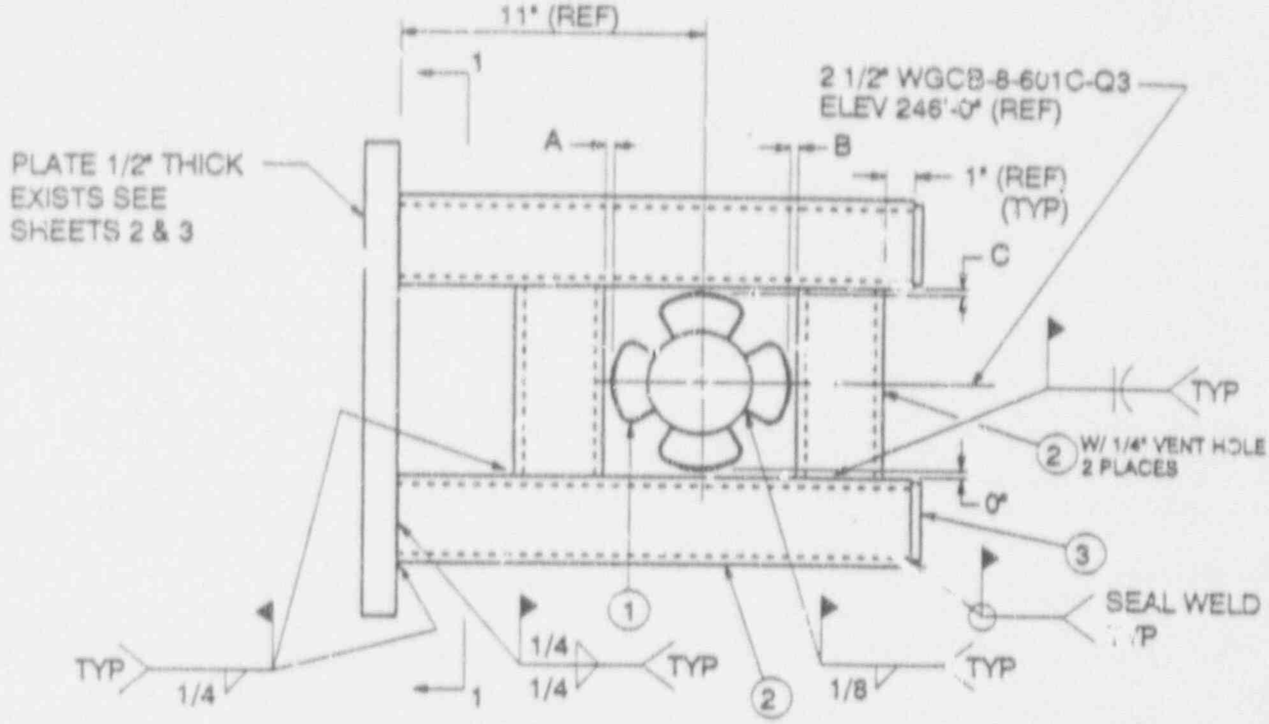
	BY	DATE	TITLE	
PREPARED	DCM	9/17/92	STEAM GENERATOR BLOWDOWN MODIFICATIONS CUBICLE A PIPE SUPPORT FPH-WGCB-5-1	CLIENT VIRGINIA POWER
CHECKED	[illegible]	9/17/92		PROJECT NORTH ANNA 1
APPROVED	[illegible]	9/17/92		J.O. NO. 02072.0710
				ISSUE BY SWEC
				DATE
REVISIONS BY	0			NUMBER SHEET 2 OF 3 N9013-1-M-1MFSK783A-1
DATE				



ISSUED FOR CONSTRUCTION OF DC-90-13-1

SAFETY RELATED
DC 90-13-1

	BY	DATE	TITLE	CLIENT VIRGINIA POWER
PREPARED	SCM	9/17/92	STEAM GENERATOR BLOWDOWN MODIFICATIONS	PROJECT NORTH ANNA 1
CHECKED	RAB	9/17/92	CUBICLE A	J.O. NO. 02072.0710
APPROVED	DDM	9/17/92	PIPE SUPPORT FPH-WGCB 5-1	ISSUE BY SWEC
REVISIONS BY	0	0 0 0 0		DATE
DATE				NUMBER SHEET 3 OF 3 N9013-1-M-IMFSK783A-1



ELEV LOOKING WEST

NOTES:

1. FOR SUPPORT LOCATION SEE DWG. N9013-1-M-602
2. FOR GENERAL NOTES REFER TO DWG. N9013-1-M-700.01
3. FIELD TO REMOVE EXISTING PIPE ATTACHMENT AND ANGLE BOX FRAME
4. CLEARANCES 1/16" $\leq$ (A+B) , (C) $\leq$ 5/32", FIELD SHIM TO SUIT.

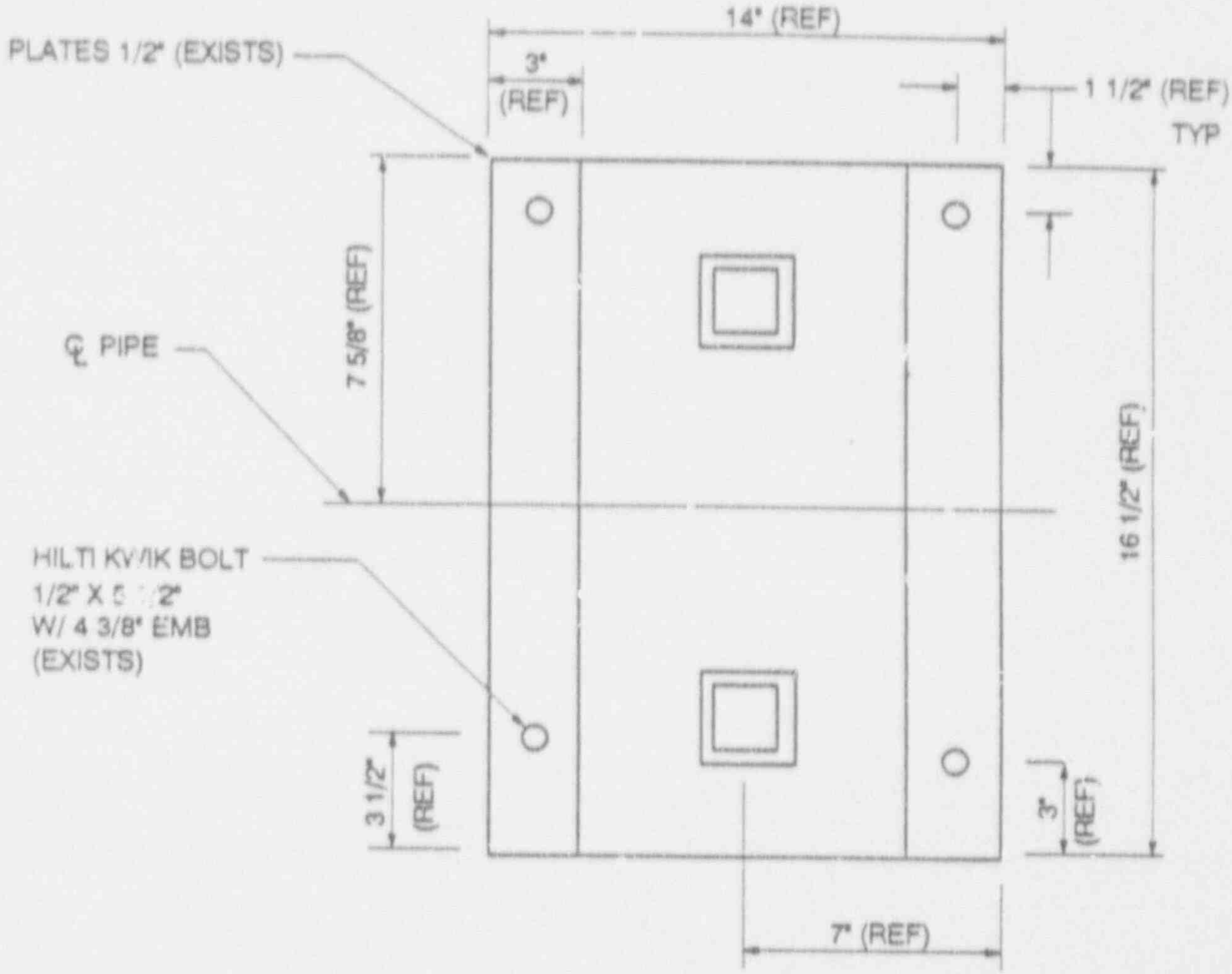
BILL OF MATERIAL

ITEM	QTY	DESCRIPTION	MAT'L
1	4	2 1/2" PROTECTION SADDLE FOR 2" INS, GRINNELL FIG 162	————
2	4	TS 3 X 3 X 1/4" (LENGTH TO SUIT BY FIELD)	A500 GR. B
3	2	STEEL STRIP 1/8" X 2 3/4" X 2 3/4" W/ 1/4" VENT HOLE	A606

SAFETY RELATED
DC 90-13-1

ISSUED FOR CONSTRUCTION OF DC-90-13-1

	BY	DATE	TITLE	
PREPARED	DOM	9/17/92	STEAM GENERATOR BLOWDOWN MODIFICATIONS CUBICLE B PIPE SUPPORT FPH-WGCB-8-3	CLIENT VIRGINIA POWER
CHECKED	[illegible]	9/17/92		PROJECT NORTH ANNA 1
APPROVED	[illegible]	9/17/92		J.O. NO. 02072.0710
				ISSUE BY SWEC
				DATE
REVISIONS BY	0	1 2 3 4		NUMBER SHEET 1 OF 3
DATE				N9013-1-M-1MFSK784A-1



SECTION 1-1

ISSUED FOR CONSTRUCTION OF DC-90-13-1

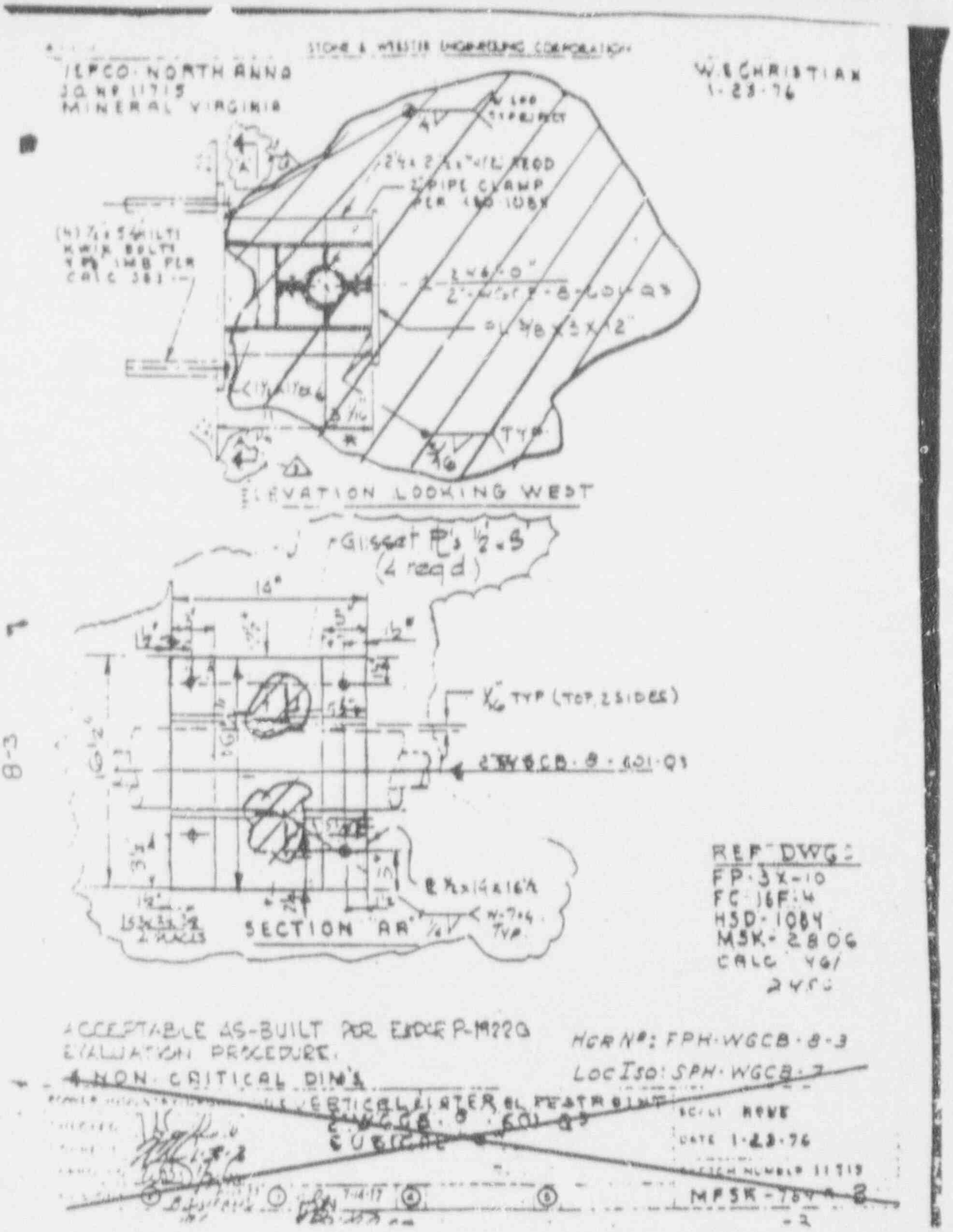
SAFETY RELATED
DC 90-13-1

	BY	DATE
PREPARED	DCM	9/2/92
CHECKED	RRB	9/7/92
APPROVED	[illegible]	9/11/92

TITLE
STEAM GENERATOR BLOWDOWN
MODIFICATIONS
CUBICLE B
PIPE SUPPORT FPH-WGCB-8-3

CLIENT	VIRGINIA POWER
PROJECT	NORTH ANNA 1
J.O. NO.	02072.0710
ISSUE BY	SWEC
DATE	
NUMBER	SHEET 2 OF 3
	N9013-1-M-1MFSK784A-1

REVISIONS BY	0	0	0	0	0
DATE					

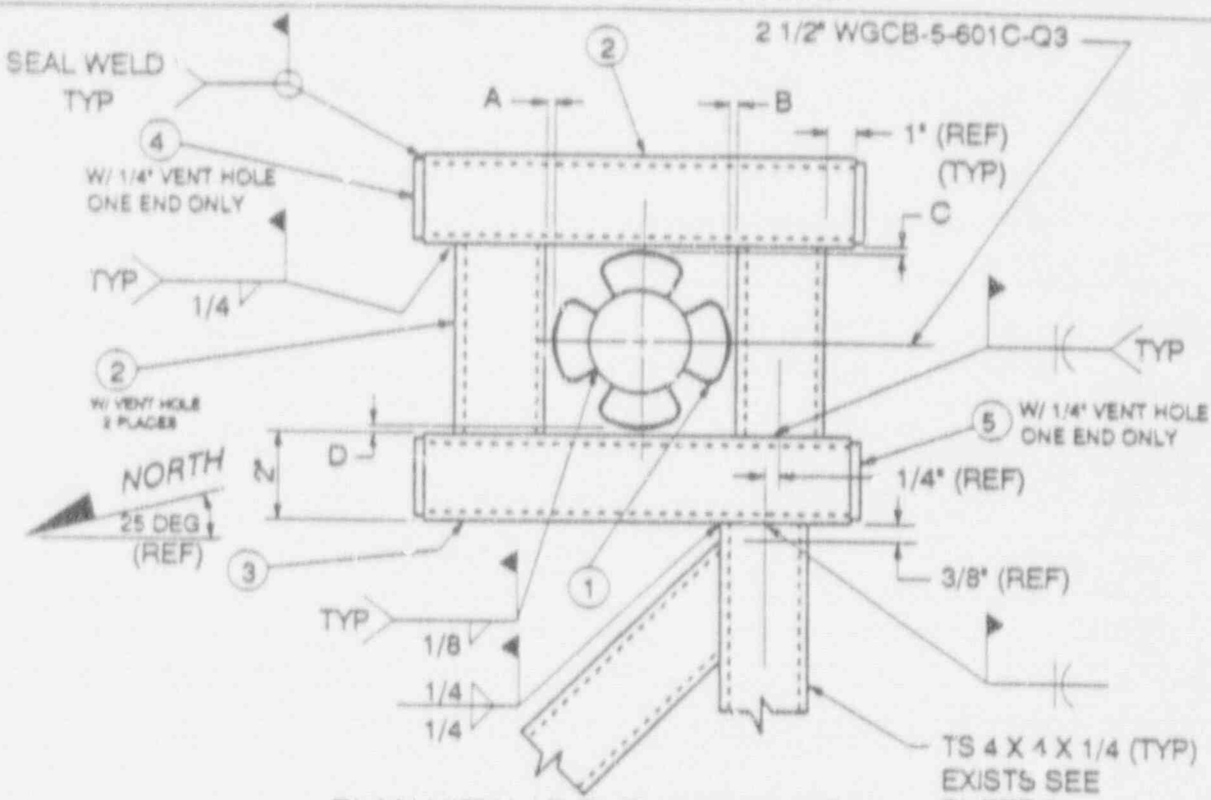


ISSUED FOR CONSTRUCTION OF DC-90-13-1

SAFETY RELATED
DC 90-13-1

	BY	DATE	TITLE	
PREPARED	[illegible]	7/11/92	STEAM GENERATOR BLOWDOWN MODIFICATIONS	CLIENT VIRGINIA POWER
CHECKED	[illegible]	9/17/92	CUBICLE B	PROJECT NORTH ANNA 1
APPROVED	[illegible]	9/17/92	PIPE SUPPORT FPH-WGCB-8-3	J.O. NO. 02072.0710
REVISIONS BY	0	0	0 0 0	ISSUE BY SWEC
DATE				DATE
				NUMBER SHEET 3 OF 3
				N9013-1-M-1MFSK784A-1

STONE & WEBSTER ENGINEERING CORPORATION



PLAN VIEW AT ELEV 249'-2" (REF)

NOTES:

1. FOR SUPPORT LOCATION SEE DWG. N9013-1-M-601
2. FOR GENERAL NOTES REFER TO DWG. N9013-1-M-700.01
3. FIELD TO REMOVE EXISTING PIPE ATTACHMENT AND ANGLE BOX FRAME, CUT BACK EXISTING TS 4 X 4 X 1/4 AND REFABRICATE AS SHOWN
4. CLEARANCES 1/16" $\leq$ (A+B) , (C+D) $\leq$ 5/32", FIELD SHIM TO SUIT

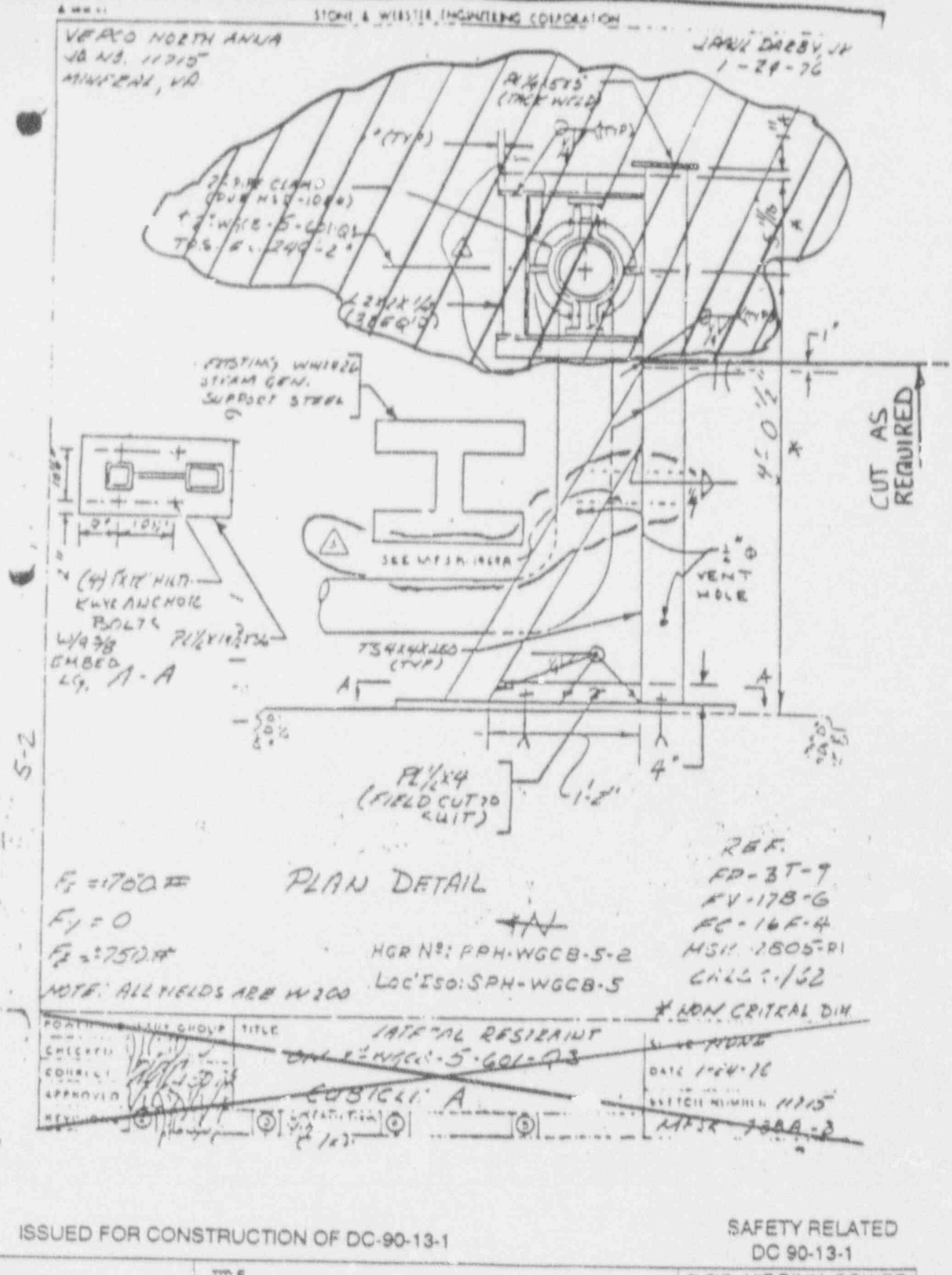
BILL OF MATERIAL

ITEM	QTY	DESCRIPTION	MAT'L
1	4	2 1/2" PROTECTION SADDLE FOR 2" INS, GRINNELL FIG 162	------
2	3	TS 4 X 4 X 1/4" (LENGTH TO SUIT BY FIELD)	A500 GR. B
3	1	TS 4 X 2 X 1/4" (LENGTH TO SUIT BY FIELD)	A500 GR. B
4	2	STEEL STRIP 1/8" X 3 3/4" X 3 3/4"	A606
5	2	STEEL STRIP 1/8" X 1 3/4" X 3 3/4"	A606

ISSUED FOR CONSTRUCTION OF DC-90-13-1

SAFETY RELATED
DC 90-13-1

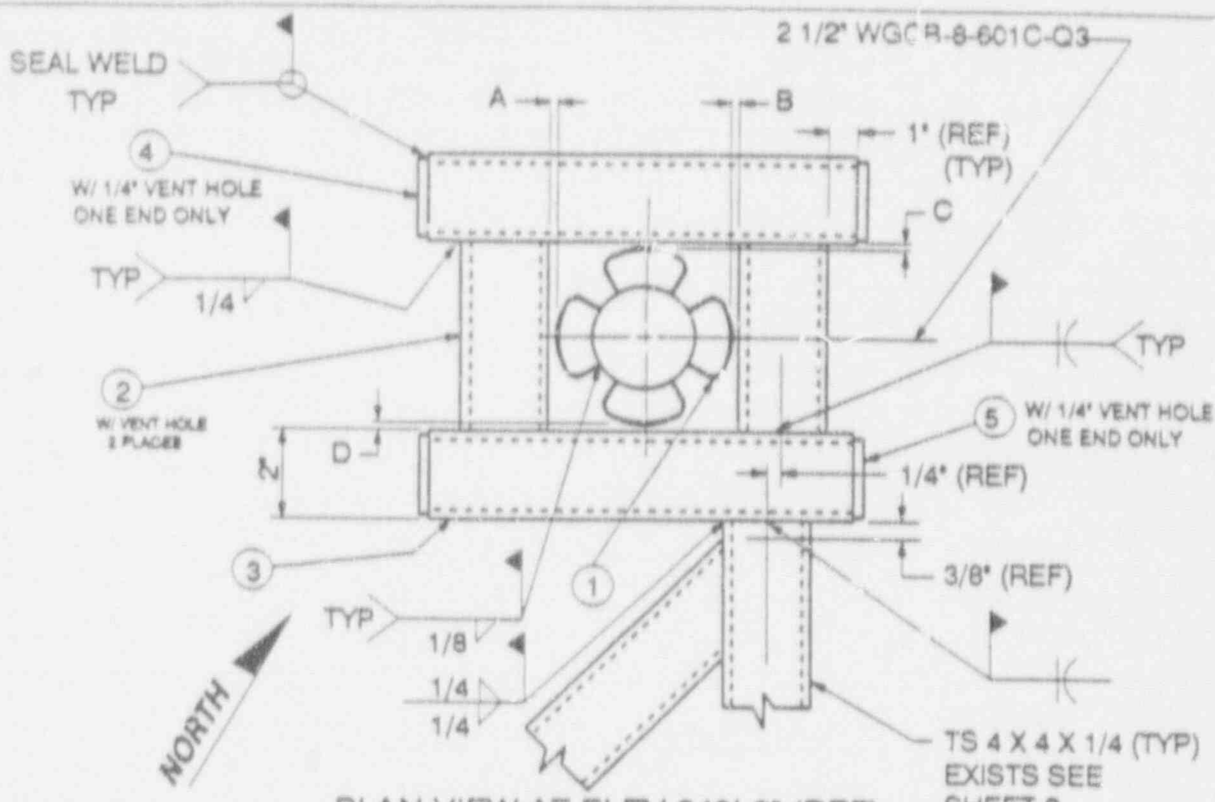
	BY	DATE	TITLE	
PREPARED	DCM	9/17/92	STEAM GENERATOR BLOWDOWN MODIFICATIONS CUBICLE A PIPE SUPPORT FPH-WGCB-5-2	CLIENT VIRGINIA POWER
CHECKED		9/17/92		PROJECT NORTH ANNA 1
APPROVED		9/17/92		J.O. NO. 02072.0710
				ISSUE BY SWEC
				DATE
REVISIONS BY	0			NUMBER SHEET 1 OF 2 N9013-1-M-1MFSK788A-I
DATE				



ISSUED FOR CONSTRUCTION OF DC-90-13-1

SAFETY RELATED
DC 90-13-1

	BY	DATE	TITLE	
PREPARED	RCK	9/17/92	STEAM GENERATOR BLOWDOWN MODIFICATIONS CUBICLE A PIPE SUPPORT FPH-WGCB-5-2	CLIENT VIRGINIA POWER
CHECKED	RAB	9/17/92		PROJECT NORTH ANNA 1
APPROVED		9/17/92		J.O. NO. 02072.0710
				ISSUE BY SWEC
REVISIONS BY				DATE
DATE				NUMBER SHEET 2 OF 2 N9013-1-M-IMFSK 7884-1



PLAN VIEW AT ELEV 249'-2" (REF)

NOTES:

1. FOR SUPPORT LOCATION SEE DWG. N9013-1-M-602
2. FOR GENERAL NOTES REFER TO DWG. N9013-1-M-700.01
3. FIELD TO REMOVE EXISTING PIPE ATTACHMENT AND ANGLE BOX FRAME, CUT BACK EXISTING TS 4 X 4 X 1/4 AND REFABRICATE AS SHOWN
4. CLEARANCES 1/16" ≤ (A+B) , (C+D) ≤ 5/32", FIELD SHIM TO SUIT.

BILL OF MATERIAL

ITEM	QTY	DESCRIPTION	MAT'L
1	4	2 1/2" PROTECTION SADDLE FOR 2" INS, GRINNELL FIG 162	———
2	3	TS 4 X 4 X 1/4" (LENGTH TO SUIT BY FIELD)	A500 GR. B
3	1	TS 4 X 2 X 1/4" (LENGTH TO SUIT BY FIELD)	A500 GR. B
4	2	STEEL STRIP 1/8" X 3 3/4" X 3 3/4"	A606
5	2	STEEL STRIP 1/8" X 1 3/4" X 3 3/4"	A606

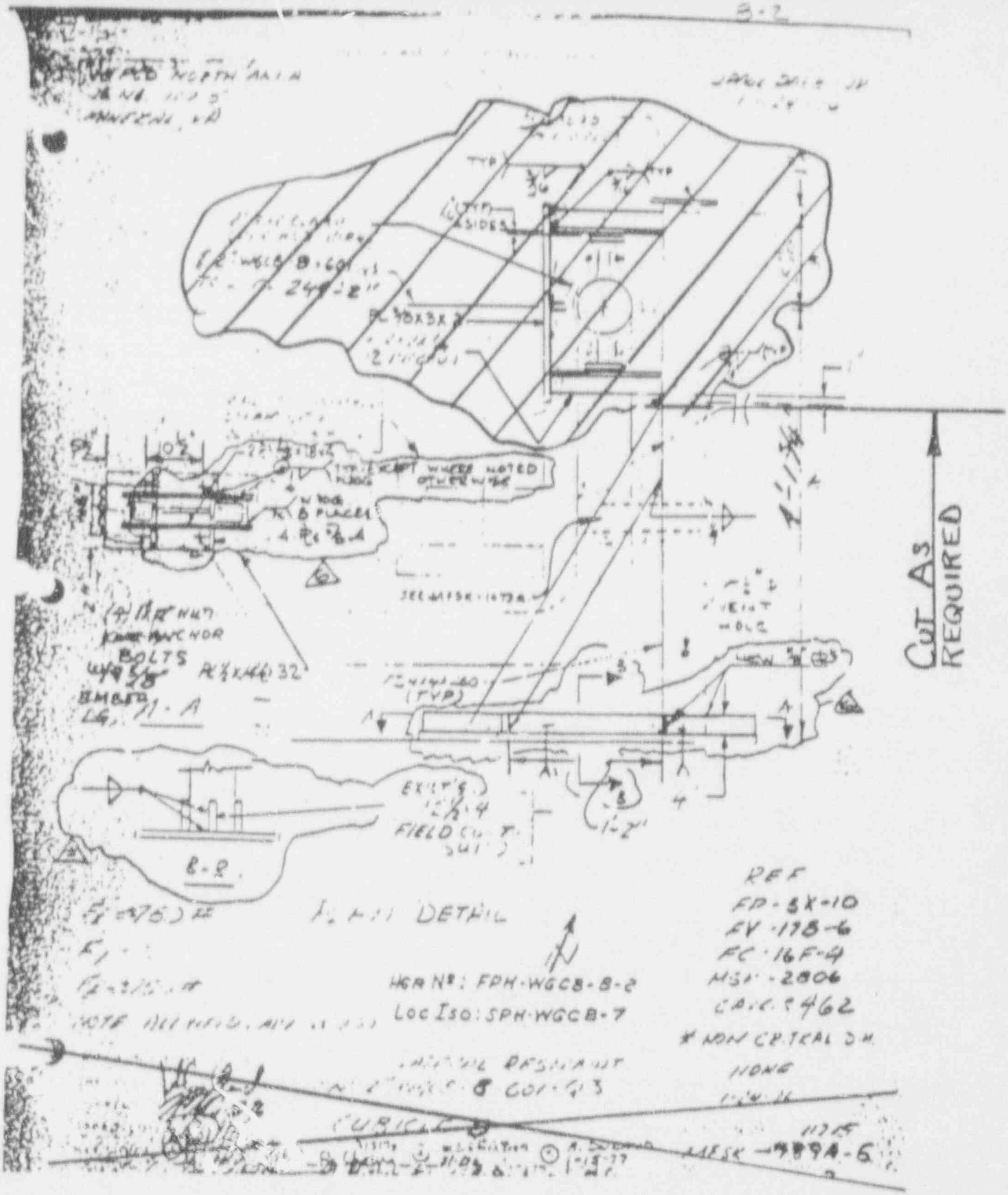
ISSUED FOR CONSTRUCTION OF DC-90-13-1

SAFETY RELATED
DC 90-13-1

	BY	DATE
PREPARED	DCW	9/17/92
CHECKED	[illegible]	9/17/92
APPROVED	[illegible]	9/17/92
REVISIONS BY		
DATE		

TITLE
STEAM GENERATOR BLOWDOWN
MODIFICATIONS
CUBICLE B
PIPE SUPPORT FPH-WGCB-8-2

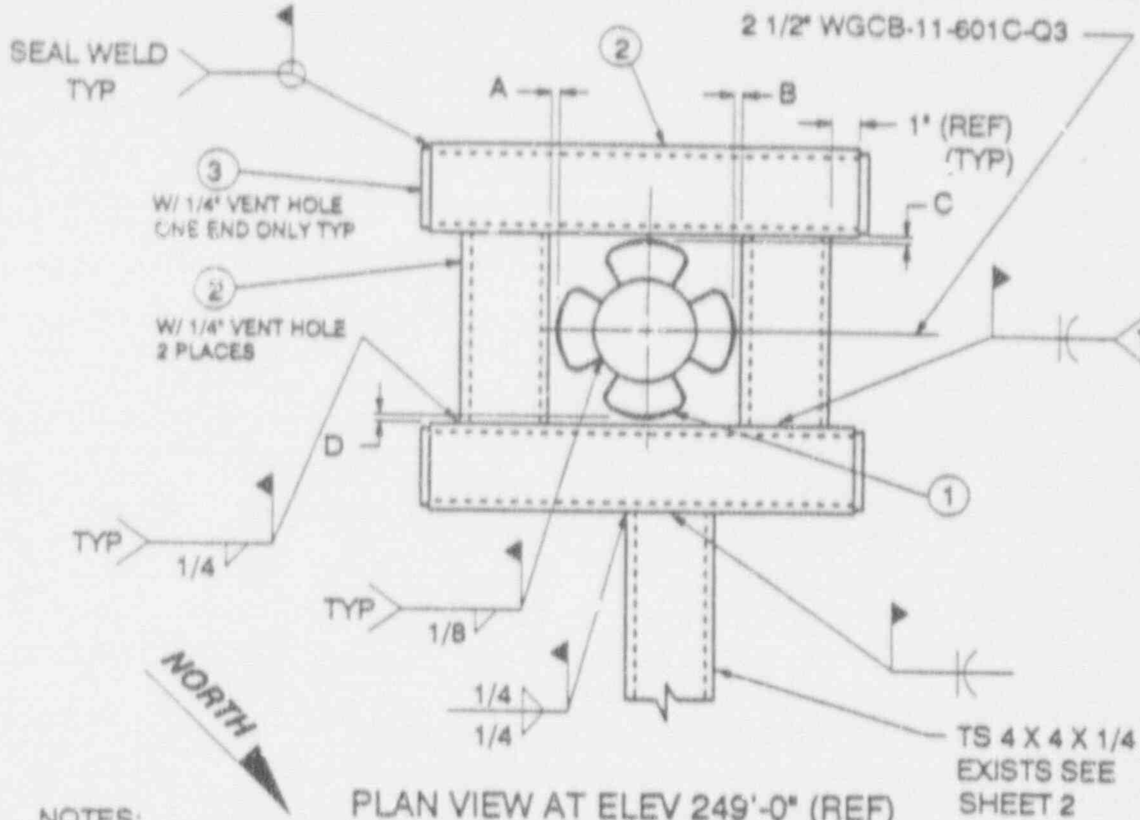
CLIENT	VIRGINIA POWER
PROJECT	NORTH ANNA 1
J.O. NO.	02072.0710
ISSUE BY	SWEC
DATE	
NUMBER	SHEET 1 OF 2 N9013-1-M-1MFSK789A-1



ISSUED FOR CONSTRUCTION OF DC-90-13-1

SAFETY RELATED
DC 90-13-1

	BY	DATE	TITLE			CLIENT VIRGINIA POWER
			STEAM GENERATOR BLOWDOWN			PROJECT NORTH ANNA 1
PREPARED	[illegible]	9/17/92	MODIFICATIONS			J.O. NO. 02072.0710
CHECKED	RAB	9/17/92	CUBICLE B			ISSUE BY SWEC
APPROVED	[illegible]	9/17/92	PIPE SUPPORT FPH-WGCB-B-2			DATE
REVISIONS BY	0	1	2	3	4	NUMBER SHEET 2 OF 2
DATE						N9013-1-M- IMFSK789A-1



PLAN VIEW AT ELEV 249'-0" (REF)

NOTES:

1. FOR SUPPORT LOCATION SEE DWG. N9013-1-M-603
2. FOR GENERAL NOTES REFER TO DWG. N9013-1-M-700.01
3. FIELD TO REMOVE EXISTING PIPE ATTACHMENT,
   CUT BACK EXISTING TS 4 X 4 X 1/4 AND REFABRICATE AS SHOWN
4. CLEARANCES 1/16" ≤ (A+B) , (C+D) ≤ 5/32", FIELD SHIM TO SUIT.

BILL OF MATERIAL

ITEM	QTY	DESCRIPTION	MAT'L
1	4	2 1/2" PROTECTION SADDLE FOR 2" INS, GRINNELL FIG 162	———
2	4	TS 4 X 4 X 1/4" (LENGTH TO SUIT BY FIELD)	A500 GR. B
3	4	STEEL STRIP 1/8" X [illegible] 3/4" X 3 3/4"	A606

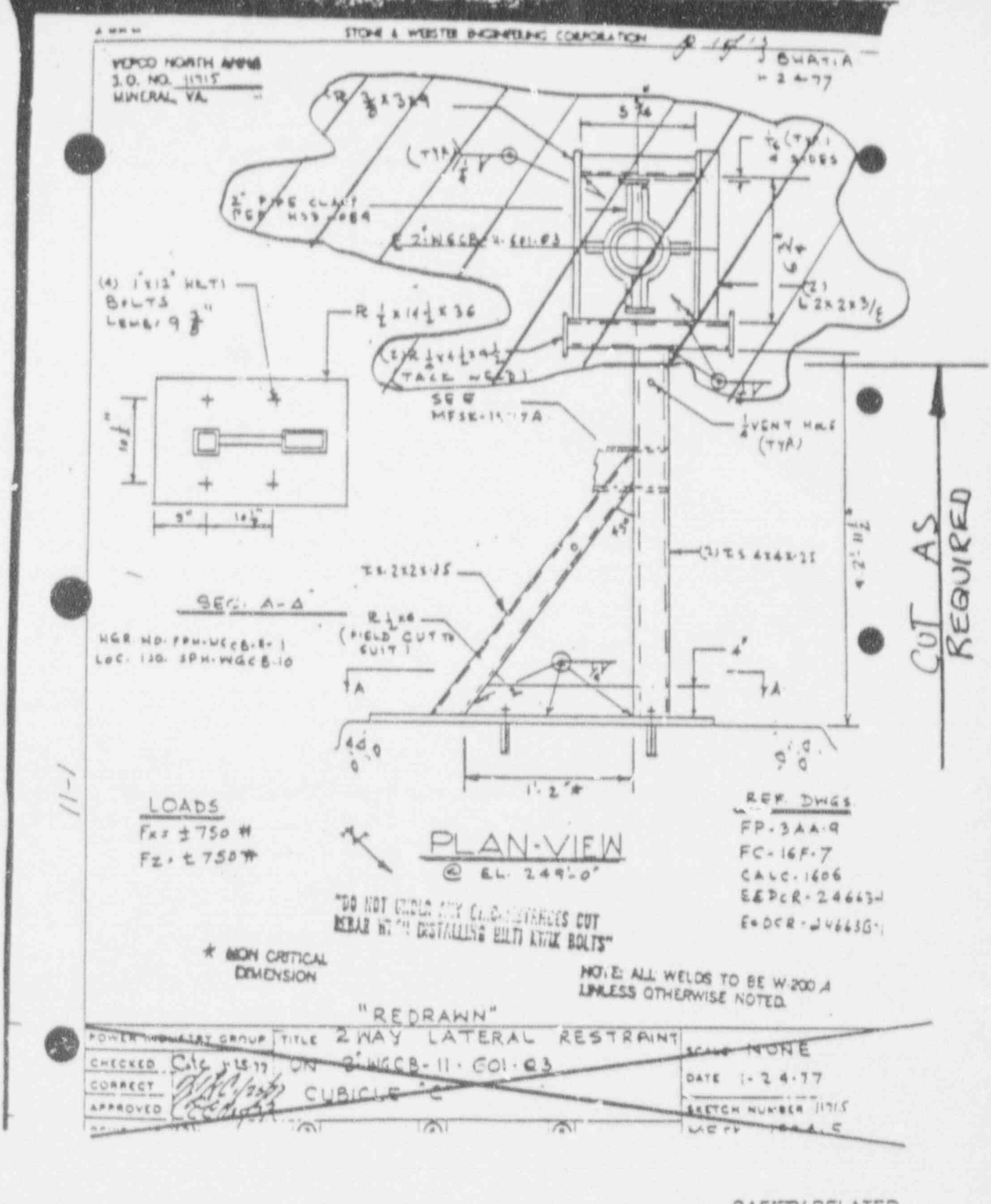
ISSUED FOR CONSTRUCTION OF DC-90-13-1

SAFETY RELATED
DC 90-13-1

	BY	DATE
PREPARED	DMW	9/17/92
CHECKED	[illegible]	9/17/92
APPROVED	[illegible]	9/17/92
REVISIONS BY	0	
DATE		

TITLE
STEAM GENERATOR BLOWDOWN
MODIFICATIONS
CUBICLE C
PIPE SUPPORT FPH-WGCB-11-1

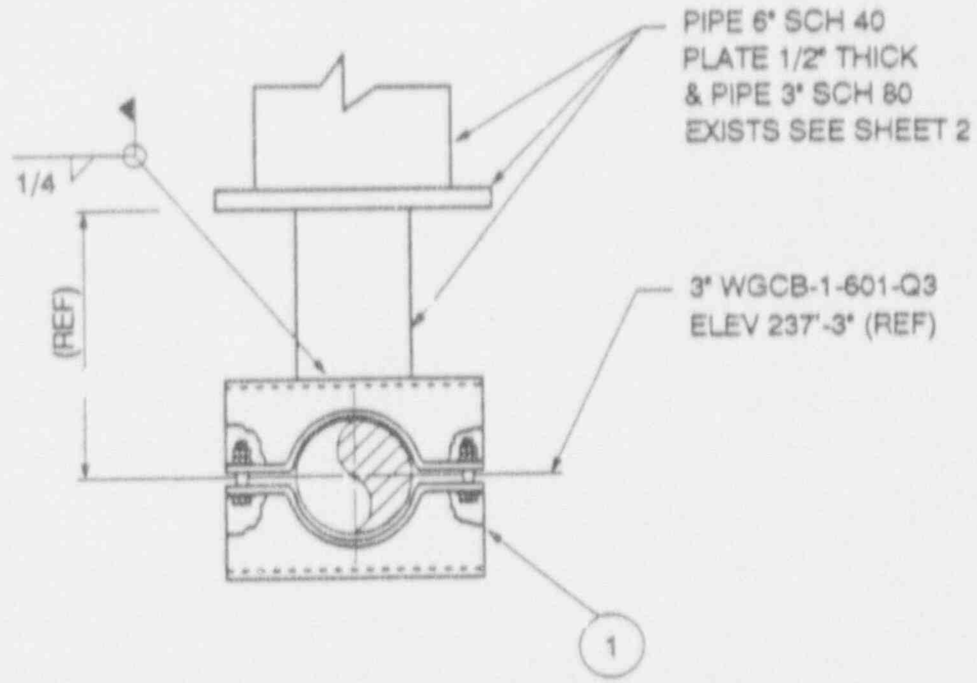
CLIENT	VIRGINIA POWER
PROJECT	NORTH ANNA 1
J.O. NO.	02072.0710
ISSUED BY	SWEC
DATE	
NUMBER	SHEET 1 OF 2
	N9013-1-M-1MFSK790A- 1



ISSUED FOR CONSTRUCTION OF DC-90-13-1

SAFETY RELATED
DC 90-13-1

	BY	DATE	TITLE	
PREPARED		9/17/92	STEAM GENERATOR BLOWDOWN MODIFICATIONS CUBICLE C PIPE SUPPORT FPH-WGCB-11-1	CLIENT VIRGINIA POWER
CHECKED		9/17/92		PROJECT NORTH ANNA 1
APPROVED		9/17/92		J.O. NO. 02072.0710
				ISSUE BY SWEC
				DATE
REVISIONS BY	0			NUMBER SHEET 2 OF 2
DATE				N9013-1-M-MFSK790A-1



ELEVATION LOOKING NORTH EAST

NOTES:

1. FOR SUPPORT LOCATION SEE DWG. N9013-1-M-601
2. FOR GENERAL NOTES REFER TO DWG. N9013-1-M-700.01
3. FIELD TO CUT & GRIND OFF EXISTING PIPE ATTACHMENT

BILL OF MATERIAL

ITEM	QTY	DESCRIPTION	MAT'L
1	1	3" CLAMP ANCHOR PER DRAWING N9013-1-M-700.07	————

SAFETY RELATED
DC 90-13-1

ISSUED FOR CONSTRUCTION OF DC-90-13-1

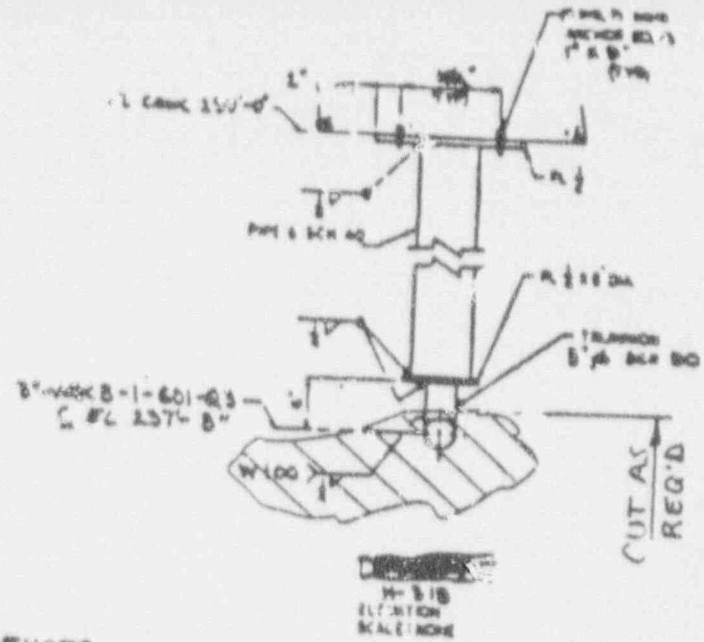
	BY	DATE
PREPARED	KCA	4/17/92
CHECKED	RGB	4/17/92
APPROVEL	[illegible]	4/17/92

TITLE
ST[illegible] GENERATOR BLOWDOWN
MODIFICATIONS
CUBICLE A
PIPE SUPPORT 1-WGCB-A-318

CLIENT	VIRGINIA POWER
PROJECT	NORTH ANNA 1
J.O. NO.	02072.0710
ISSUE BY	SWEC
DATE	
NUMBER	SHEET 1 OF 2

N9013-1-M-1PSSK103AAD.08-0

REVISIONS BY	0	0	0	0	0
DATE					



REFERENCES:
1) 11715-FP-8K
2) 11715-FP-16F
11715-ECI-103AAD
4) 11715-FP-5BM, DET 166
5) E&DCR: 23516
6) 11715-MSK-103A7 & 8

NOTES:
1) ALL WELDS TO BE W-100 UNLESS OTHERWISE SPECIFIED

TRANSPOSED INFORMATION CHECKED BY
PAUL WONG
Paul Wong 7-4-97
Signature Date

INFORMATION TRANSPOSED BY E. MICHEL
E. Michel 12-20-96
Signature Date

SAFETY RELATED

STEAM GENERATOR BLOWDOWN
REACT. CONTMT.
UNIT 1

SWEC J.O. NO. 14238.10 SUPPORT TYPE: ANC

3					
2					
1	ORIGINAL ISSUE		-	-	7-4-97
REV.	DESCRIPTION	CHKD	COMM	APP	DATE

SUPPORT NO. 1-46CB-A-318

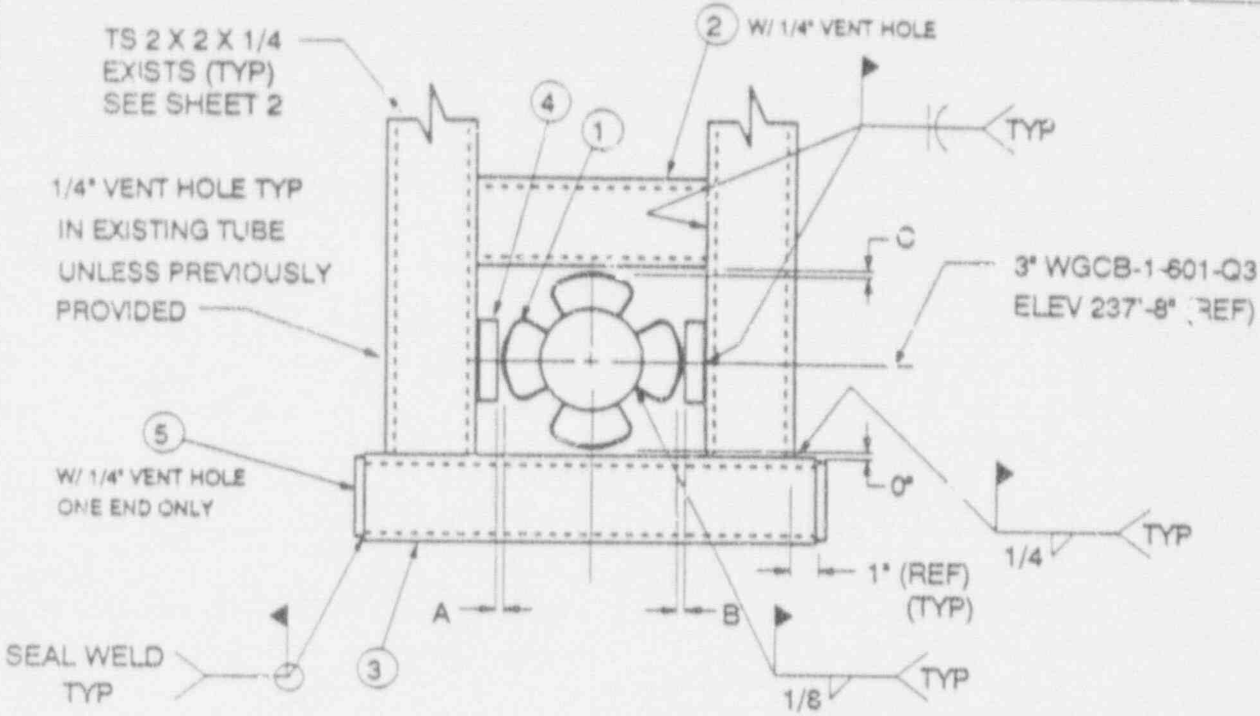
NORTH ANNA POWER STATION
VIRGINIA ELECTRIC AND POWER COMPANY

DWG NO. 11715-PSSK-103AAD.08 SHT 1 (of 1) REV 1

ISSUED FOR CONSTRUCTION OF DC-90-13-1

SAFETY RELATED
DC 90-13-1

	BY	DATE	TITLE	CLIENT	VIRGINIA POWER
PREPARED	DCM	4/17/92	STEAM GENERATOR BLOWDOWN MODIFICATIONS CUBICLE A PIPE SUPPORT 1-46CB-A-318	PROJECT	NORTH ANNA 1
CHECKED	RAB	4/17/92		J.O. NO	02072.0710
APPROVED		4/17/92		ISSUE BY	SWEC
REVISIONS BY	0			DATE	
DATE				NUMBER	SHEET 2 OF 2
					N9013-1-M-PSSK103AAD.08 0



ELEV LOOKING NORTH EAST

NOTES:

1. FOR SUPPORT LOCATION SEE DWG. N9013-1-M-601
2. FOR GENERAL NOTES REFER TO DWG. N9013-1-M-700.01
3. FIELD TO REMOVE EXISTING PIPE ATTACHMENT
4. CLEARANCES 1/16" ≤ (A+B) , (C) ≤ 5/32", FIELD SHIM TO SUIT

BILL OF MATERIAL

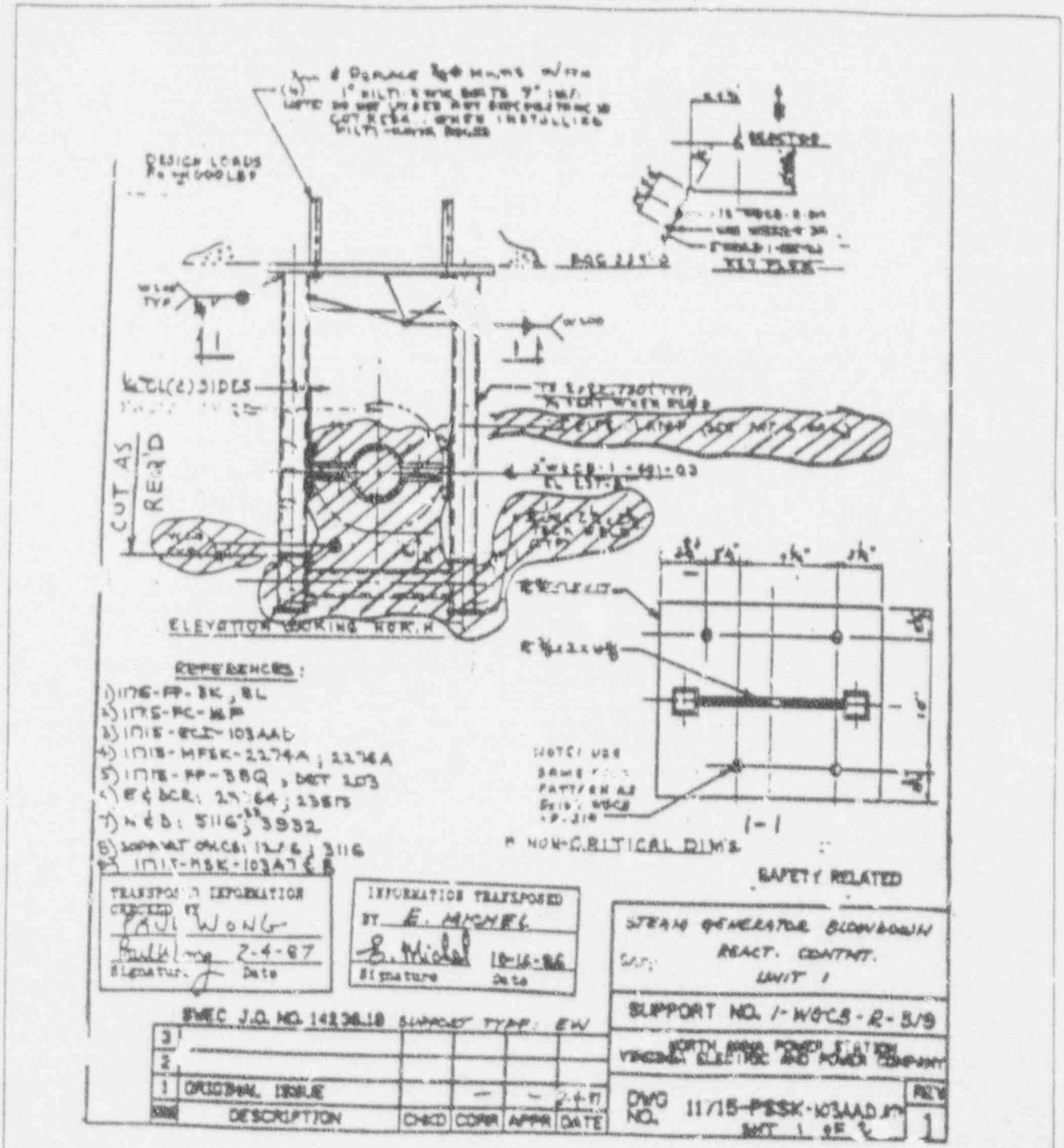
ITEM	QTY	DESCRIPTION	MAT'L
1	4	3" PROTECTION SADDLE FOR 3" INS, GRINNELL FIG 164	————
2	1	TS 2 X 2 X 1/4" W/VENT HOLE (LENGTH TO SUIT BY FIELD)	A500 GR. B
3	1	TS 2 X 2 X 1/4" (LENGTH TO SUIT BY FIELD)	A500 GR. B
4	2	PLATE 3 X 3 (THICKNESS TO SUIT)	A36
5	2	STEEL STRIP 1/8" X 3/4" X 1 3/4"	A606

ISSUED FOR CONSTRUCTION OF DC-90-13-1

SAFETY RELATED
DC 90-13-1

	BY	DATE	TITLE		
PREPARED	DON	9/17/92	STEAM GENERATOR BLOWDOWN MODIFICATIONS	CLIENT	VIRGINIA POWER
CHECKED	RB	9/17/92	CUBICLE A	PROJECT	NORTH ANNA 1
APPROVED	DM	9/17/92	PIPE SUPPORT 1-WGCB-R-319	J.O. NO.	02072.0710
				ISSUE BY	SWEC
				DATE	
REVISIONS BY	0			NUMBER	SHEET 1 OF 2
DATE					N9013-1-M-1PSSK103AAD 09-[illegible]

STONE & WEBSTER ENGINEERING CORPORATION



ISSUED FOR CONSTRUCTION OF DC-90-13-1

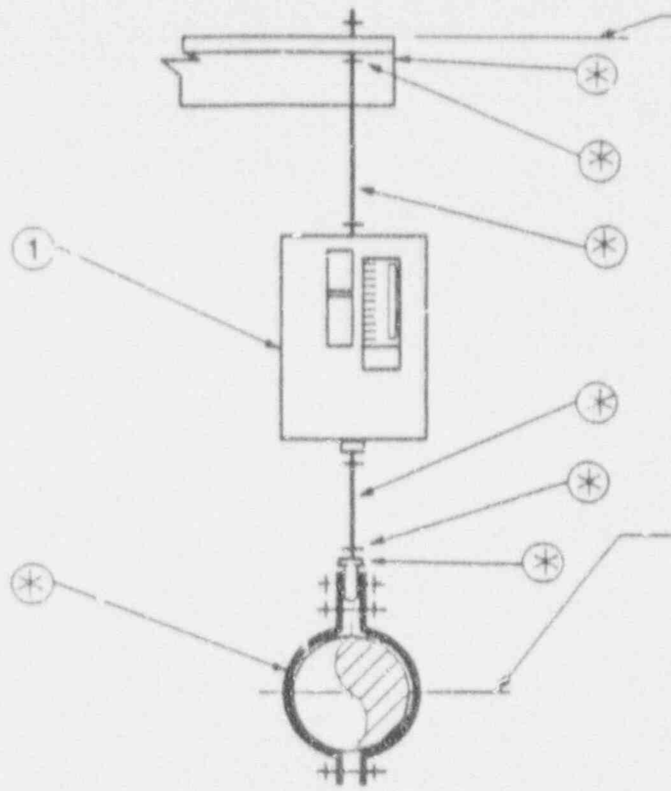
SAFETY RELATED
DC 90-13-1

	BY	DATE
PREPARED	DCM	9/17/92
CHECKED	RBP	9/17/92
APPROVED	DRM	9/17/92

TITLE
STEAM GENERATOR BLOWDOWN
MODIFICATIONS
CUBICLE A
PIPE SUPPORT 1-WGCB-R-319

CLIENT	VIRGINIA POWER
PROJECT	NORTH ANNA 1
J.O. NO.	02072.0710
ISSUE BY	SWEC
DATE	
NUMBER	SHEET 2 OF 2
	N9013-1-M-1PSSK103AAD.09-1

REVISIONS BY	⓪	①	②	③	④
DATE					



ELEV LOOKING SOUTH

NOTES:

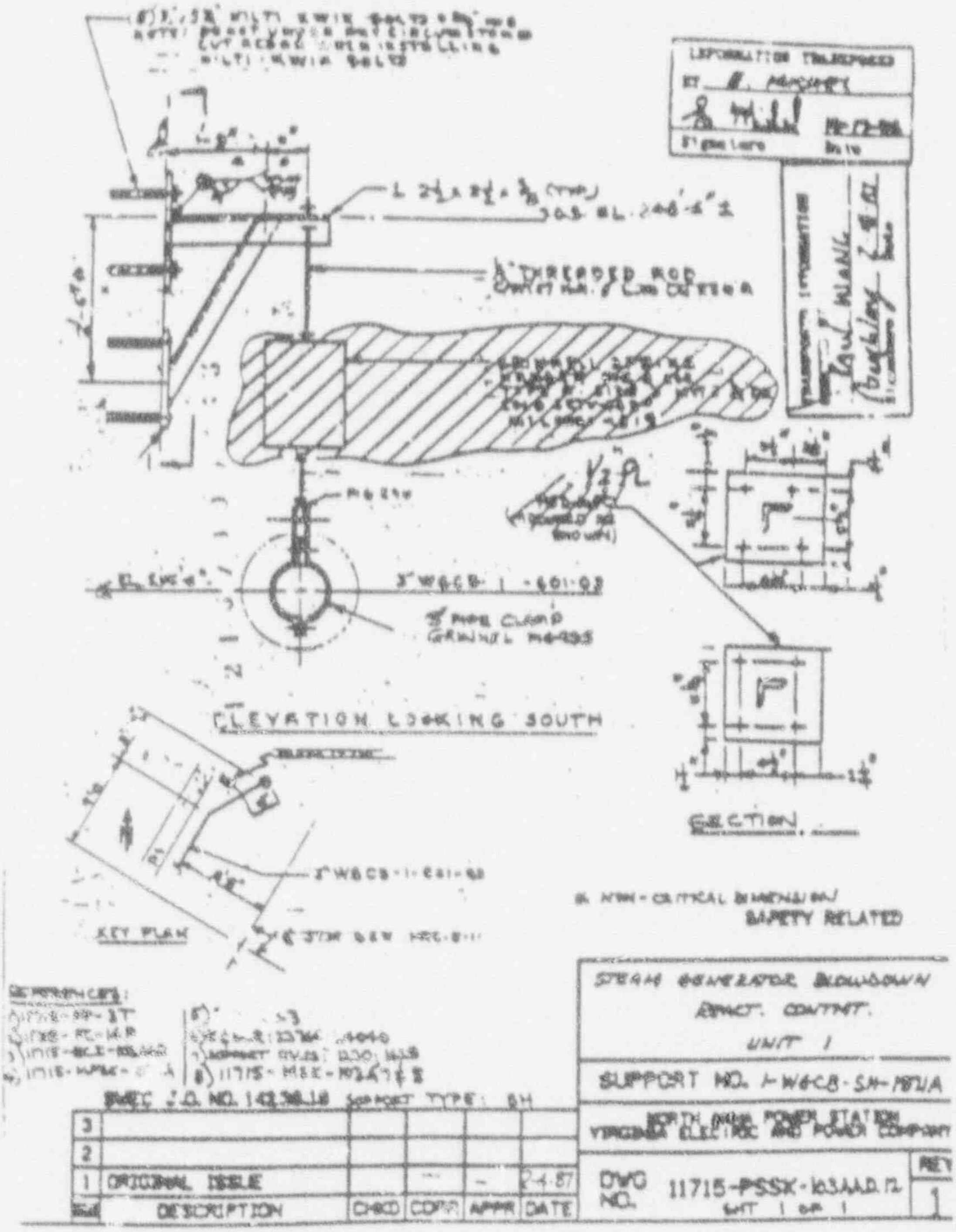
1. FOR SUPPORT LOCATION SEE DWG. N9013-1-M-601
2. FOR GENERAL NOTES REFER TO DWG. N9013-1-M-700.01

BILL OF MATERIAL			
ITEM	QTY	DESCRIPTION	MAT'L
1	1	SIZE 4 GRINNELL FIG B-268 TYPE A VARIABLE SPRING	——
		HANGER HL = 233 # CL = 227 #	

SAFETY RELATED
DC 90-13-1

ISSUED FOR CONSTRUCTION OF DC-90-13-1

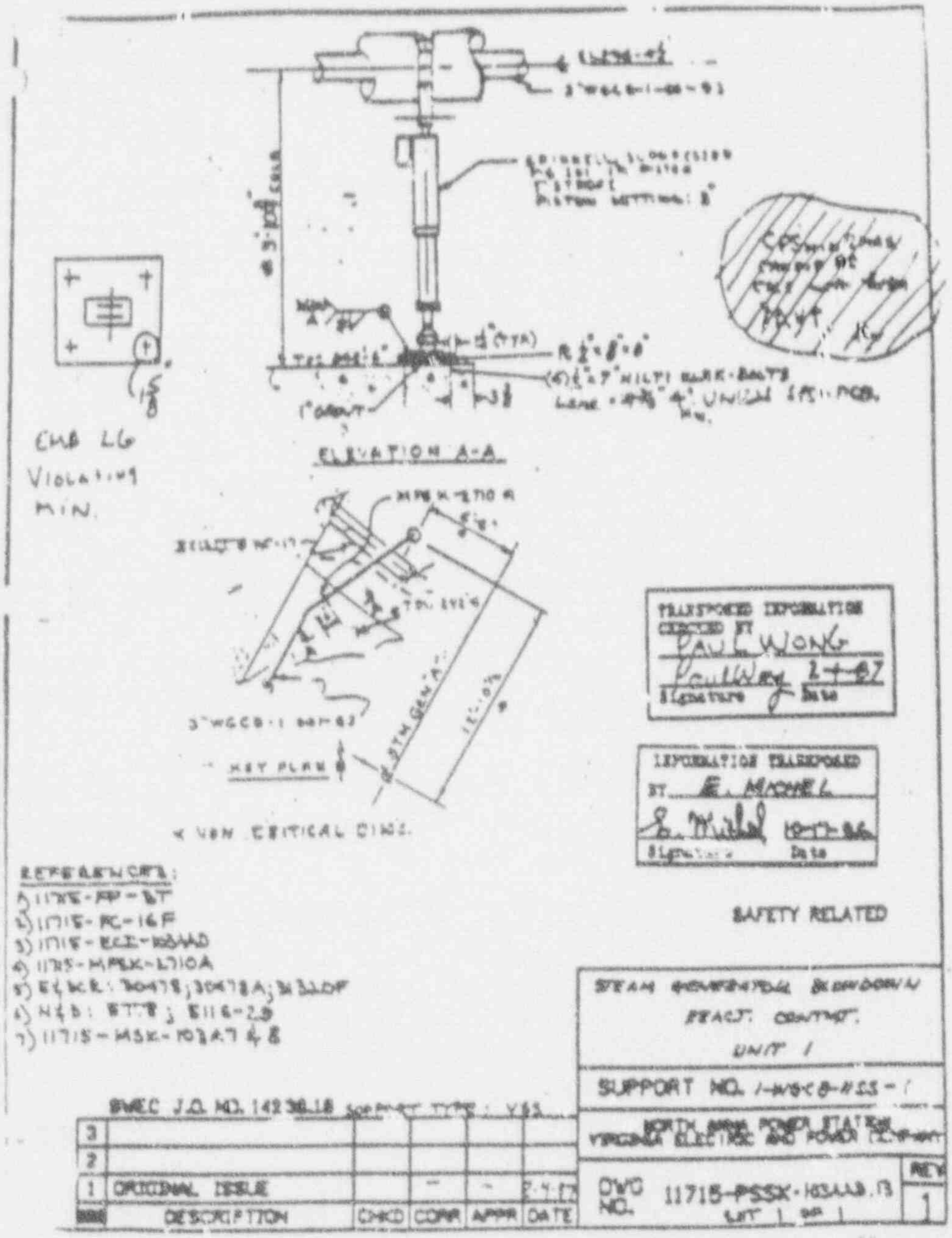
	BY	DATE	TITLE	
PREPARED	DOm	5/4/92	STEAM GENERATOR BLOWDOWN MODIFICATIONS CUBICLE A PIPE SUPPORT 1-WGCB-SH-MFSK-1971A	CLIENT VIRGINIA POWER
CHECKED	RAB	5/4/92		PROJECT NORTH ANNA 1
APPROVED	DS M	5/4/92		J.O. NO. 02072.0710
				ISSUE BY SWEC
				DATE
REVISIONS BY	0			NUMBER SHEET 1 OF 2
DATE				N9013-1-M-1PSSK103AAD.12-0



ISSUED FOR CONSTRUCTION OF DC-90-13-1

SAFETY RELATED
DC 90-13-1

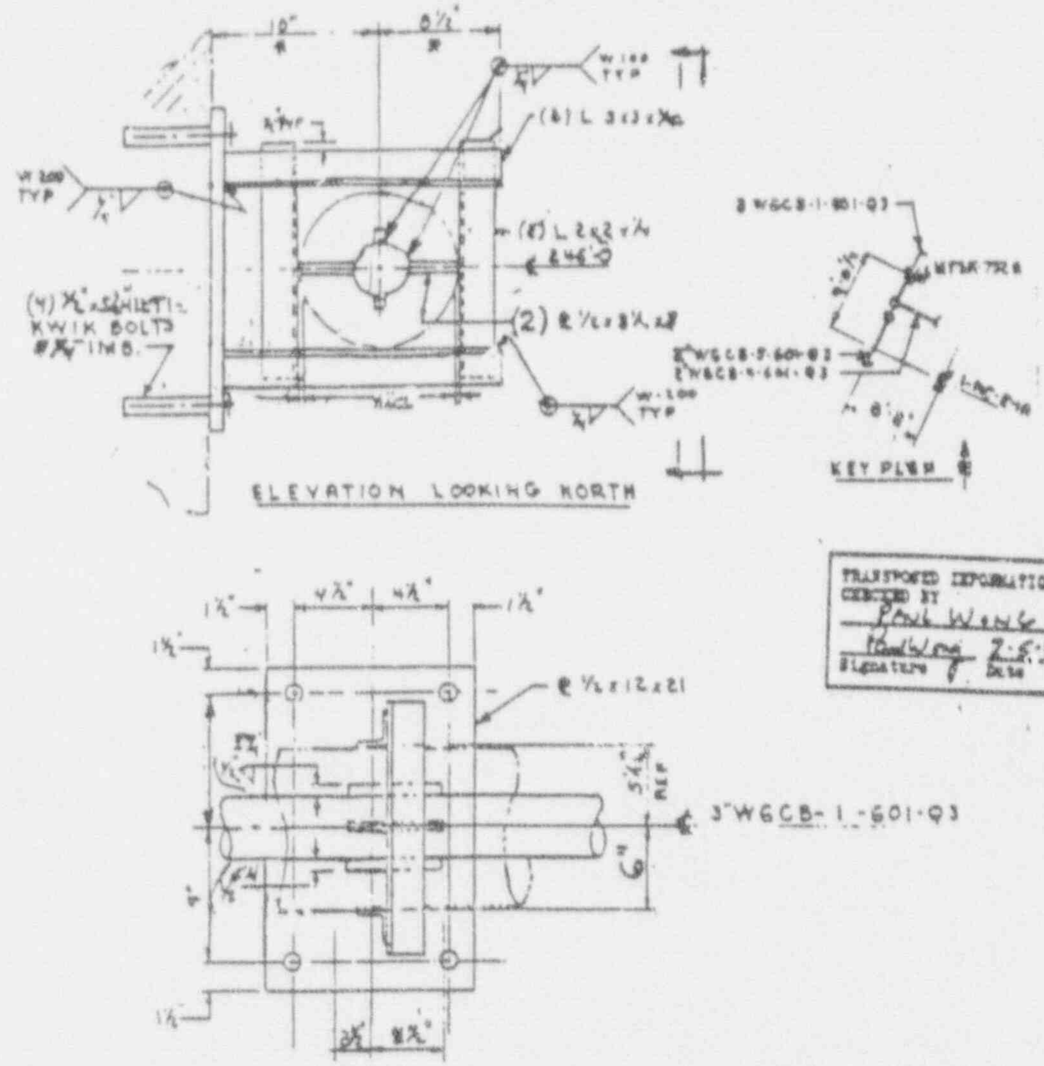
	BY	DATE	TITLE	
			STEAM GENERATOR BLOWDOWN MODIFICATIONS CUBICLE A PIPE SUPPORT 1-WGCB-SH-MFSK-1971A	CLIENT VIRGINIA POWER
				PROJECT NORTH ANNA 1
PREPARED	DCM	4/17/92		J.O. NO. 02072.0710
CHECKED	BRB	4/17/92		ISSUE BY SWEC
APPROVED	DSM	4/17/92		DATE
REVISIONS BY	0	1	2 3 4	NUMBER SHEET 2 OF 2
DATE				N9013-1-M-1PSSK103AAD.12-0



ISSUED FOR CONSTRUCTION OF DC-90-13-1

SAFETY RELATED
DC 90-13-1

	BY	DATE	TITLE	CLIENT VIRGINIA POWER
			STEAM GENERATOR BLOWDOWN MODIFICATIONS CUBICLE A	PROJECT NORTH ANNA 1
PREPARED	DOM	4/17/92		J.O. NO. 02072.0710
CHECKED	RBB	4/17/92		ISSUE BY SWEC
APPROVED	DBM	4/17/92	PIPE SUPPORT 1-WGCB-HSS-001	DATE
REVISIONS BY	⓪	① ② ③ ④		NUMBER SHEET 1 OF 1
DATE				N9013-1-M-1PSSK103AAD.13-0



TRANSPOSED INFORMATION
CHECKED BY Paul Wong
Paul Wong 7-5-87
Signature Date

REFERENCES:
1) 11715-FP-3T
2) 11715-FC-14F
3) 11715-BZZ-103AAD
4) 11715-MSK-103A18
5) SUPPORT CALC: 445 5118
6) 11715-MPSK-75ZA
7) N&D: 5118-37

INFORMATION TRANSPOSED
BY E. MICHEL
E. Michel 10-17-86
Signature Date

SAFETY RELATED

STEAM GENERATOR BLOWDOWN
REACT. CONTMT.
UNIT 1

SUPPORT NO. 1-WGCB-R-450

NORTH ANNA POWER STATION
VIRGINIA ELECTRIC AND POWER COMPANY

SWEC NO. 14238.18 SUPPORT TYPE: L.S.

NO.	DESCRIPTION	CHKD	CORR	APPR	DATE
3					
2					
1	ORIGINAL ISSUE		–	–	7-5-87

DWG NO. 11715-PSSK-103AAD.14 SHT 1 OF 1 REV 1

ISSUED FOR CONSTRUCTION OF DC-90-13-1

SAFETY RELATED
DC 90-13-1

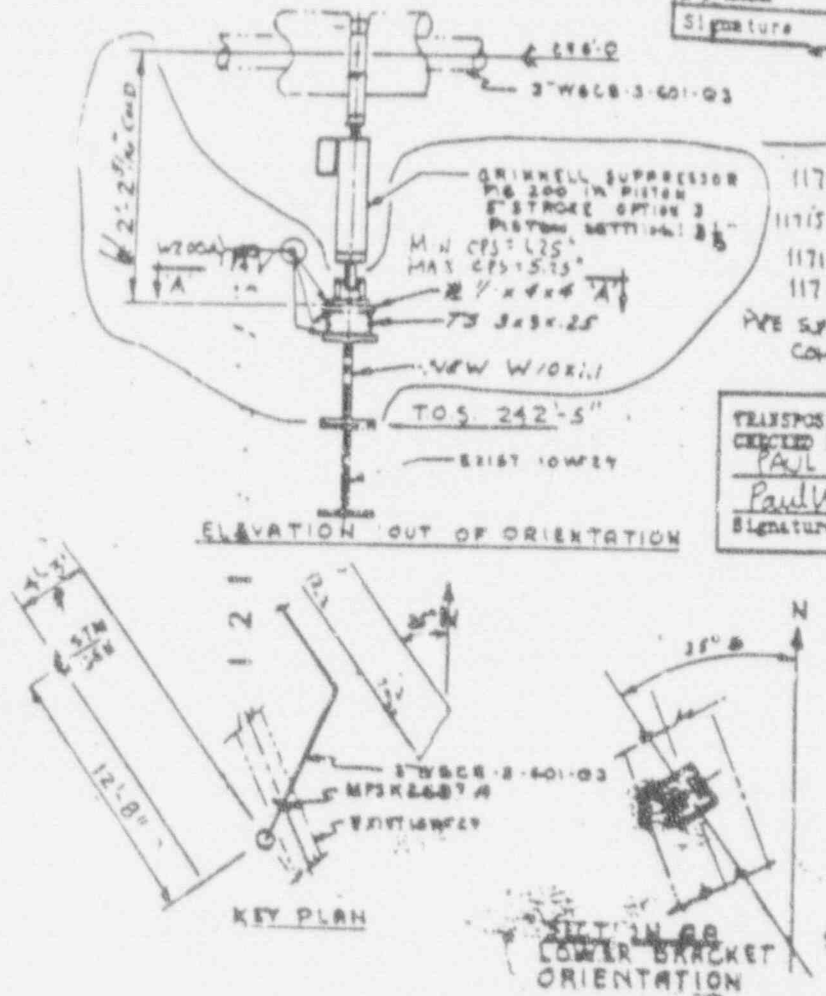
	BY	DATE
PREPARED	DOM	4/17/92
CHECKED	RAB	4/17/92
APPROVED	DJM	4/17/92

TITLE
STEAM GENERATOR BLOWDOWN
MODIFICATIONS
CUBICLE A
PIPE SUPPORT 1-WGCB-R-450

CLIENT	VIRGINIA POWER
PROJECT	NORTH ANNA 1
J.O. NO.	02072.0710
ISSUE BY	SWEC
DATE	
NUMBER	SHEET 1 OF 1
	N9013-1-M-1PSSK103AAD.14-0

REVISIONS BY	①	②	③	④	⑤
DATE					

INFORMATION TRANSPOSED
BY P. E. HULL
P E Hull 10-15-85
Signature Date



REF. DWG.
11715-FP-3AA,AB
11715-FM-119-3 (EBCR ZONE)
11715-MFSK-2687A
11715-MSK-1-3A25
PIPE SUPPORT CALC #5925
COMP # 5945-1

TRANSPOSED INFORMATION CHECKED BY
PAUL WONG
Paul Wong 1-26-87
Signature Date

SAFETY RELATED

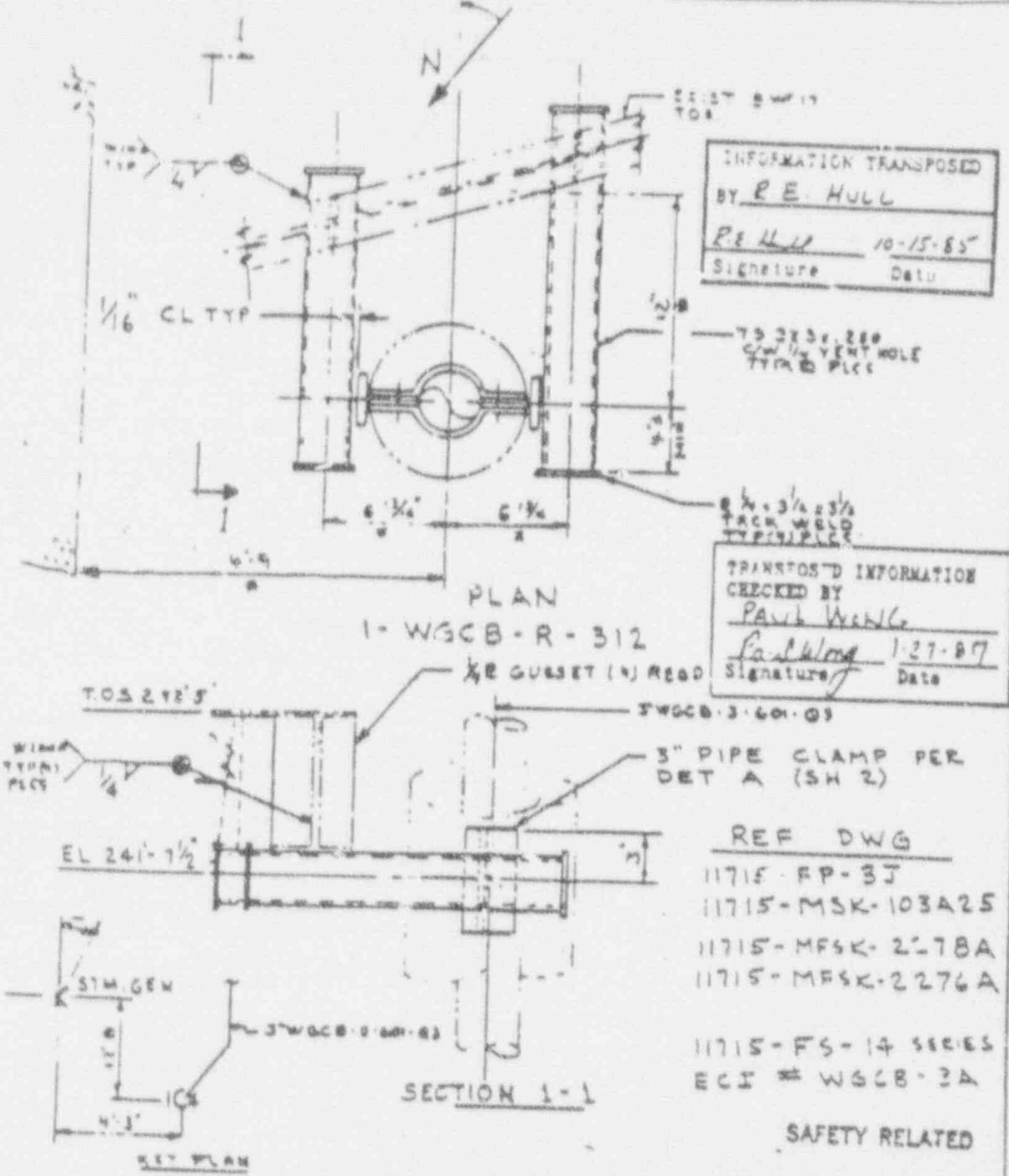
STM. GEN. BLOWDOWN SYS. REAC. CONTMT CUBICLE C UNIT 1
SUPT. NO. 1-BD-HSS-002
11715-PSSK-103BY.01 REV 1 SH 1 OF 1

ISSUED FOR CONSTRUCTION OF DC-90-13-1

SAFETY RELATED
DC 90-13-1

	BY	DATE	TITLE			CLIENT	VIRGINIA POWER
PREPARED	RDH	4/17/92	STEAM GENERATOR BLOWDOWN MODIFICATIONS CUBICLE C PIPE SUPPORT 1-BD-HSS-002			PROJECT	NORTH ANNA 1
CHECKED	RAB	4/17/92				J.O. NO.	02072.0710
APPROVED	DBA	4/17/92				ISSUE BY	SWEC
						DATE	
REVISIONS BY	⓪	②	①	①	④	NUMBER	SHEET 1 OF 1
DATE							N9013-1-M-1PSSK103BY.01-0

STONE & WEBSTER ENGINEERING CORPORATION



STM. GEN. BLOWDOWN SYS.
REAC. CONTMT - CUBICLE C
UNIT 1

SUPT. NO. 1 WGCB-R-312

11715-PSSK-13BY.02 REV 1
SH 1 OF 1

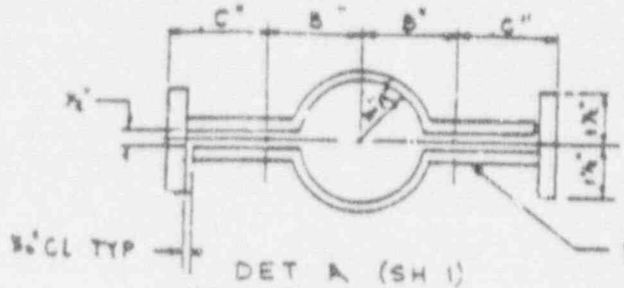
ISSUED FOR CONSTRUCTION OF DC-90-13-1

SAFETY RELATED
DC 90-13-1

	BY	DATE
PREPARED	DCW	4/17/92
CHECKED	RMB	4/17/92
APPROVED	WSM	4/17/92
REVISIONS BY		
DATE		

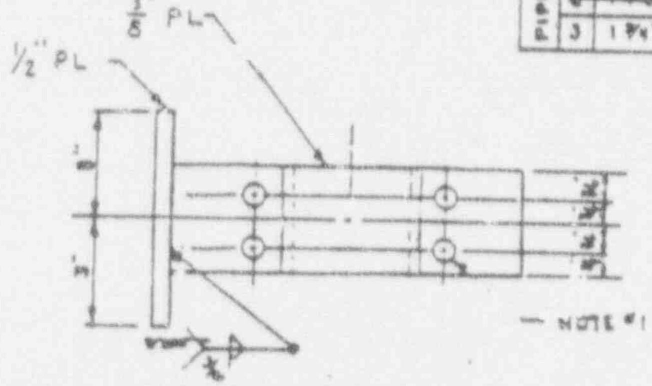
TITLE
STEAM GENERATOR BLOWDOWN
MODIFICATIONS
CUBICLE C
PIPE SUPPORT 1-WGCB-R-312

CLIENT	VIRGINIA POWER
PROJECT	NORTH ANNA 1
J.O. NO.	02072.0710
ISSUE BY	SWEC
DATE	
NUMBER	SHEET 1 OF 2
	N9013-1-M-1PSSK103BY.02-0



DET A (SH 1)
5" PIPE CLAMP ASSY.

DIML	A	B	C
~~W~~	~~8~~ ~~1 3/4~~	~~2 5/8~~	~~2 1/8~~
3	1 3/4"	2 1/8	2 3/8"



DET B HALF PIPE CLAMP
(2) REQ'D.

NOTES
1. USE (2) 3/4" x 3" BOLTS W/(2) HEX NUTS, TORQUE TO TIGHT FIT.
2. NOTCH INSULATION FOR PIPE CLAMP

SAFETY RELATED

STM. GEN. BLOWDOWN SYS.
REAC. CONTMT. - CUBICLE C
UNIT 1
SUPT. NO. 1-WGCB-R-312
11715-PSSK-103BY.2
SH 2 OF 2

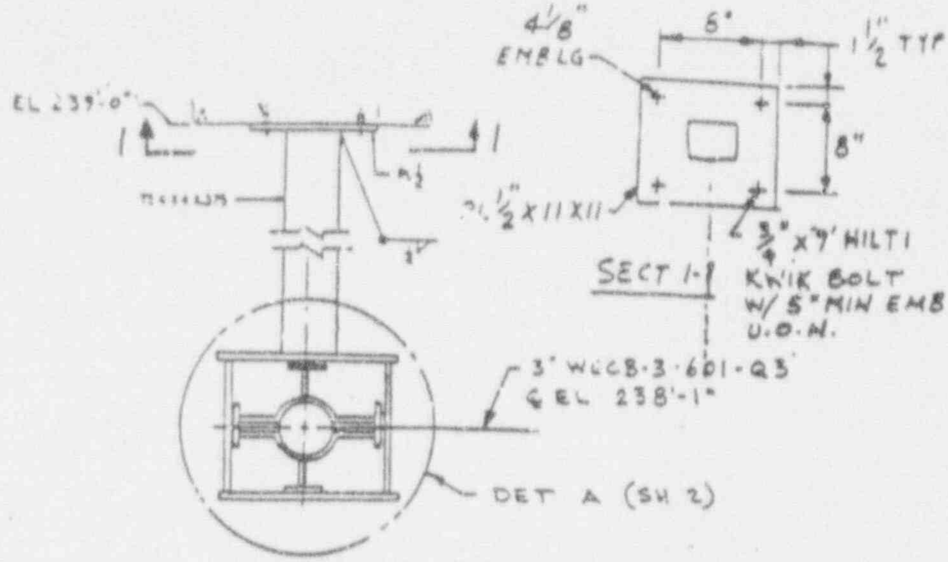
ISSUED FOR CONSTRUCTION OF DC-90-13-1

SAFETY RELATED
DC 90-13-1

	BY	DATE
PREPARED	DCA	4/17/92
CHECKED	RMB	4/17/92
APPROVED	DPM	4/17/92
REVISIONS BY	①	
DATE		

TITLE
STEAM GENERATOR BLOWDOWN
MODIFICATIONS
CUBICLE C
PIPE SUPPORT 1-WGCB-R-312

CLIENT	VIRGINIA POWER
PROJECT	NORTH ANNA 1
J.O. NO.	02072.0710
ISSUE BY	SWEC
DATE	
NUMBER	SHEET 2 OF 2
	N9013-1-M-1PSSK103BY.02-0



1-WGCB-R-304
LOOKING N-W

REF DWGS:
11715-FP-3J
11715-MSK-103A9
11715-MFSK-2840A
11715-MFSK-2276A
11715-FC-16 SERIES
ECI # WGCB-3A

TRANSFERRED INFORMATION CHECKED BY Paul Wong
Signature: Paul Wong Date: 1-27-87

INFORMATION TRANSFERRED BY R.E. HULL
Signature: R.E. Hull Date: 10-15-85

SAFETY RELATED

STM. GEN. BLOWDOWN SYS
REAC. CONTMT
UNIT 1

SUPPORT NO. 1-WGCB-R-304

NORTH ANNA POWER STATION
VIRGINIA ELECTRIC AND POWER COMPANY

SWEC J.O. NO. 14238.18

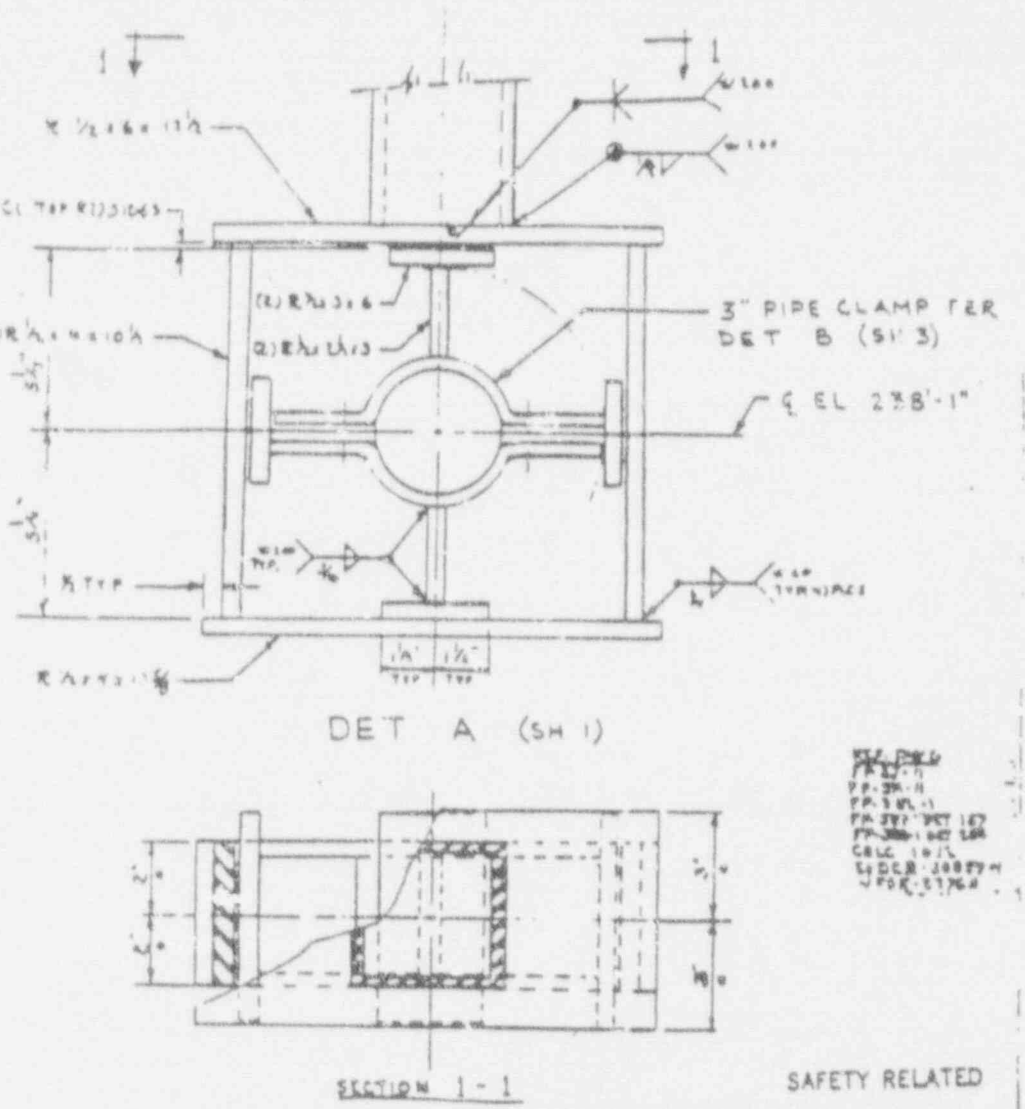
ISSUE	DESCRIPTION	CHKD	CORR	APPR	DATE
3					
2					
1	ORIGINAL ISSUE	—	—	—	1-27-87

DWG NO 11715-PSSK-103BY.3 SH 1 OF 3 — REV 1

SAFETY RELATED
DC 90-13-1

ISSUED FOR CONSTRUCTION OF DC-90-13-1

	BY	DATE	TITLE	
PREPARED	KDM	4/17/92	STEAM GENERATOR BLOWDOWN MODIFICATIONS CUBICLE C PIPE SUPPORT 1-WGCB-R-304	CLIENT VIRGINIA POWER
CHECKED	RAB	4/17/92		PROJECT NORTH ANNA 1
APPROVED		4/17/92		J.O. NO. 02072.0710
				ISSUE BY SWEC
				DATE
REVISIONS BY	①	②	③ ④ ⑤	NUMBER SHEET 1 OF 3
DATE				N9013-1-M-1PSSK103BY.03-0



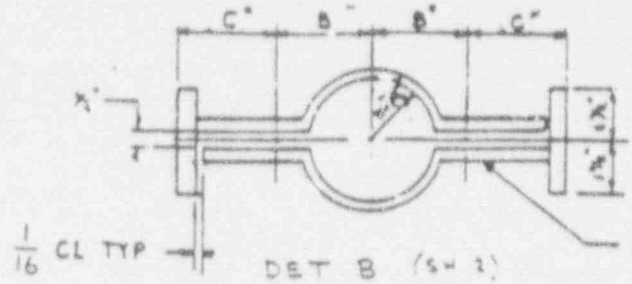
SAFETY RELATED

STM. GEN. BLOWDOWN SYS. REACTOR CONTAINMENT UNIT 1
SUPT. NO. 1-WGCB-R-304
11715-PSSK-103BY.03 REV 1 SH 2 OF 3

ISSUED FOR CONSTRUCTION OF DC-90-13-1

SAFETY RELATED
DC 90-13-1

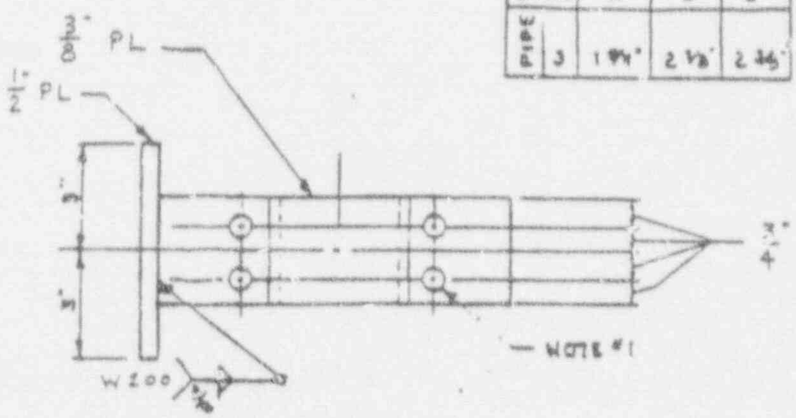
	BY	DATE	TITLE	
PREPARED	KDH	4/17/92	STEAM GENERATOR BLOWDOWN MODIFICATIONS CUBICLE C PIPE SUPPORT 1-WGCB-R-304	CLIENT VIRGINIA POWER
CHECKED	RRB	4/17/92		PROJECT NORTH ANNA 1
APPROVED		4/17/92		J.O. NO. 02072.0710
				ISSUE BY SWEC
				DATE
REVISIONS BY ①	②	③	④ ⑤	NUMBER SHEET 2 OF 3
DATE				N9013-1-M-1PSSK103BY.03-0



DET B (SH 2)

5" PIPE CLAMP ASSY.

DIMS		A	B	C
PIPE	3	1 3/4"	2 1/2"	2 3/4"



DET C HALF PIPE CLAMP
(2) REQ'D.

NOTES
1. USE (4) 5/8" x 3" BOLTS C/W (2) HEX NUTS, TORQUE TO TIGHT FIT
2. NOTCH INSULATION FOR PIPE CLAMP

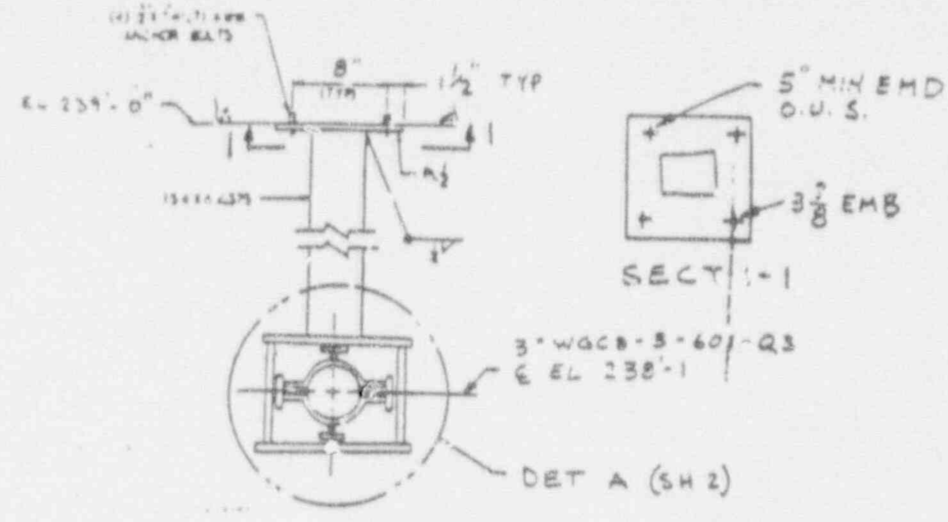
SAFETY RELATED

STM. GEN. BLOWDOWN SYS. REACTOR CONTAINMENT UNIT 1
SUPT. NO. 1-WGCB-R-304
11715-PSSK-103BY.3 REV 1 SH 3 OF 3

ISSUED FOR CONSTRUCTION OF DC-90-13

SAFETY RELATED
DC 90-13-1

	BY	DATE	TITLE	
PREPARED	[illegible]	4/17/92	STEAM GENERATOR BLOWDOWN MODIFICATIONS CUBICLE C PIPE SUPPORT 1-WGCB-R-304	CLIENT VIRGINIA POWER
CHECKED	[illegible]	4/17/92		PROJECT NORTH ANNA 1
APPROVED	[illegible]	4/17/92		J.O. NO. 02072.0710
				ISSUE BY SWEC
				DATE
REVISIONS BY	⓪	①	②	③
DATE				NUMBER SHEET 3 OF 3
				N9013-1-M-1PSSK103BY.03-0



1-WGCB-R-303
LOOKING N-W

REF DWG
11715-FP-3J
11715-MSK-103A9
11715-MFSK-2840A
11715-MFSK-2276A
11715-ECI-103BY
ECI # WGCB-3A

SAFETY RELATED

TRANS OSTD CALCULATION CHECKED BY PAUL WENG
Signature / Date

INFORMATION TRANSPOSED BY R.E. HULL
10-15-85
Signature / Date

STM. GEN. BLOWDOWN SYS.
REACTOR CONTAINMENT
UNIT 1

SUPPORT NO. 1-WGCB-R-303

NORTH ANNA POWER STATION
VIRGINIA ELECTRIC AND POWER COMPANY

SWEC J.O. NO. 14238.18

ISSUE	DESCRIPTION	CHKD	CORR	APPR	DATE
3					
2					
1	ORIGINAL ISSUE	—	—	—	2/14?

DWG NO. 11715-PSSK-103BY.4 SH 1 OF 3 — REV 1

ISSUED FOR CONSTRUCTION OF DC-90-13-1

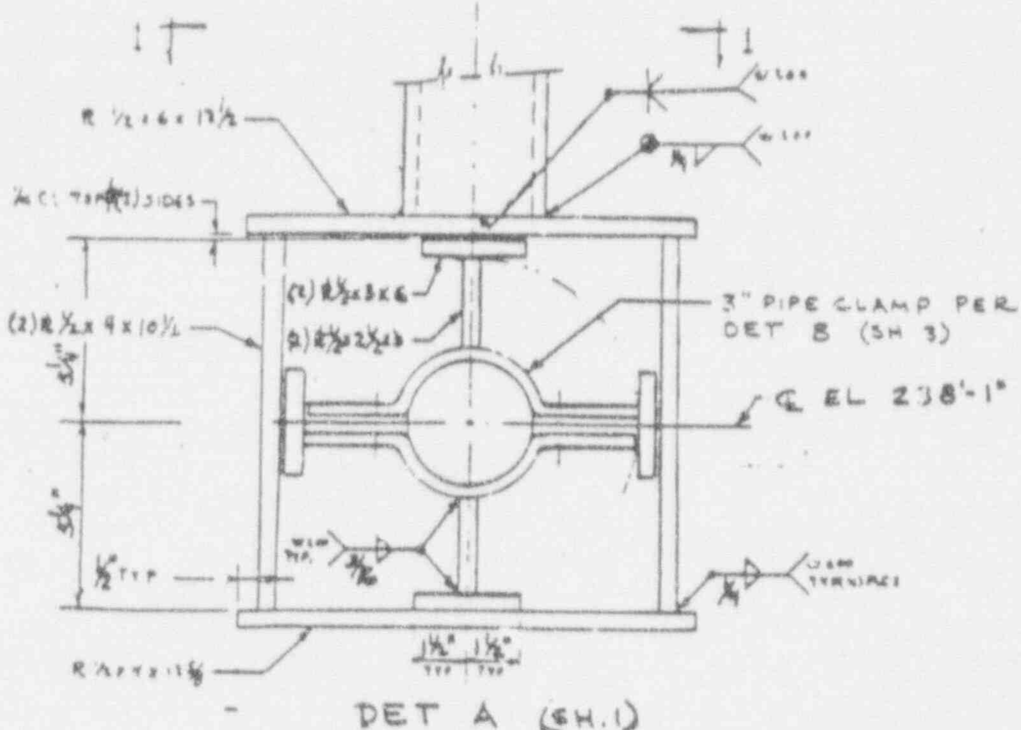
SAFETY RELATED
DC 90-13-1

	BY	DATE
PREPARED	DCM	4/17/92
CHECKED	RAB	4/17/92
APPROVED	DCM	4/17/92

TITLE
STEAM GENERATOR BLOWDOWN
MODIFICATIONS
CUBICLE C
PIPE SUPPORT 1-WGCB-R-303

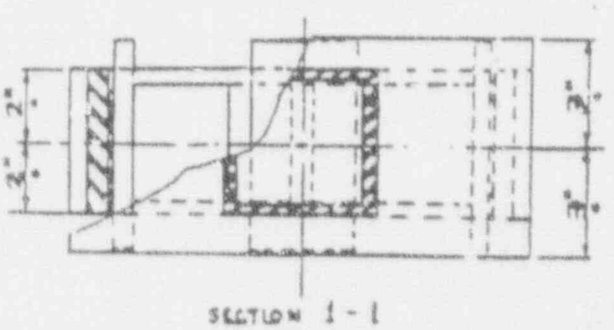
CLIENT	VIRGINIA POWER
PROJECT	NORTH ANNA 1
J.O. NO.	02072.0710
ISSUE BY	SWEC
DATE	
NUMBER	SHEET 1 OF 3
	N9013-1-M-1PSSK103BY.04-0

REVISIONS BY	①	②	③	④	⑤
DATE					



DET A (SH.1)

REF. DWGS
FP-3J,-3K,-3L
CALC 1612
E&DCR 30857
MFSK-2276A
FP-36M DET 167
FP-30Q DET 202



SECTION 1-1

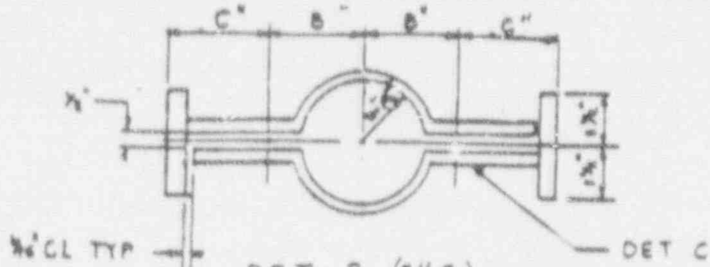
SAFETY RELATED

STM. GEN. BLOWDOWN SYS
REACTOR CONTMT
UNIT 1
SUPT. NO. 1-WGCB-R-303
11715-PSSK-103BY.04 SH 2 of 3 REV 1

ISSUED FOR CONSTRUCTION OF DC-90-13-1

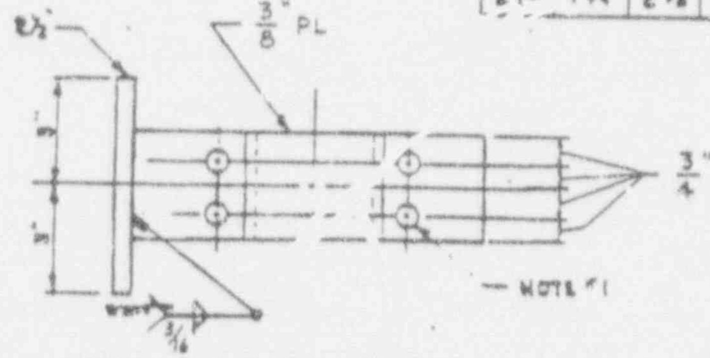
SAFETY RELATED
DC 90-13-1

	BY	DATE	TITLE	CLIENT VIRGINIA POWER
PREPARED	DOA	4/17/92	STEAM GENERATOR BLOWDOWN MODIFICATIONS CUBICLE C PIPE SUPPORT 1-WGCB-R-303	PROJECT NORTH ANNA 1
CHECKED	RAB	4/17/92		J.O. NO. 02072.0710
APPROVED	[illegible]	4/17/92		ISSUE BY SWEC
				DATE
REVISIONS BY	⓪	①	② ③ ④	NUMBER SHEET 2 OF 3
DATE				N9013-1-M-1PSSK103BY.04-0



DET B (SH 2)
5" PIPE CLAMP ASSY.

DIM'S	A	B	C
WB 5	1 1/4"	2 1/2"	2 3/8"



DET C HALF PIPE CLAMP
(2) REQ'D.

NOTES
1. USE (4) 5/8" x 3" BOLTS C/W (2) HEX NUTS, TORQUE TO TIGHT FIT
2. NOTCH INSULATION FOR PIPE CLAMP

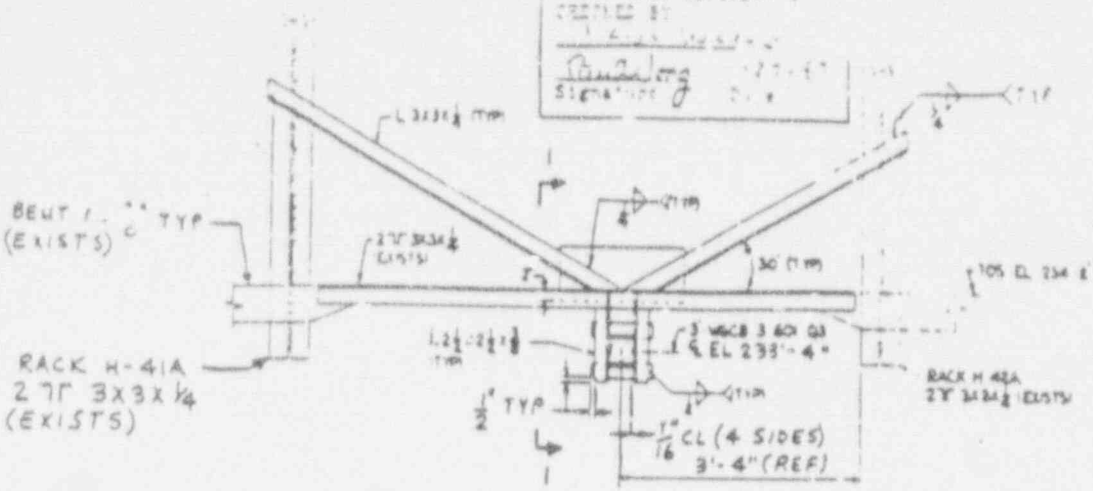
SAFETY RELATED

STM GEN BLOWDOWN SYS REACTOR CONTMT UNIT 1
SUPT. NO. 1-WGCB-R-303
11715-PSSK-103BY.04 SH 3 OF 3 REV 1

ISSUED FOR CONSTRUCTION OF DC-90-13-1

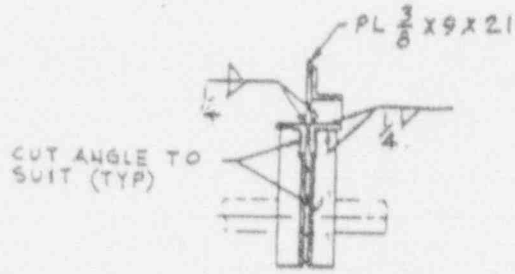
SAFETY RELATED
DC 90-13-1

	BY	DATE	TITLE			CLIENT VIRGINIA POWER
PREPARED	[illegible]	4/17/92	STEAM GENERATOR BLOWDOWN			PROJECT NORTH ANNA 1
CHECKED	[illegible]	4/17/92	MODIFICATIONS			J.O. NO. 02072.0710
APPROVED	[illegible]	4/17/92	CUBICLE C			ISSUE BY SWEC
			PIPE SUPPORT 1-WGCB-R-303			DATE
REVISIONS BY	①	①	①	①	①	NUMBER SHEET 3 OF 3
DATE						N9013-1-M-1PSSK103BY.04-0



1-WGCB-R-122
ELEVATION

REF DWG
11715-FP-3J
11715-MSK-103A7
11715-FS 14 SERIES



1-1

INFORMATION TRANSPOSED
BY R.E. HULL
R.E. Hull 10-15-85
Signature Date

NUCLEAR SAFETY RELATED
Q.A. CAT I

STM. GEN. BLOWDOWN SYS.
REAC. CONT MT
UNIT 1

SUPPORT NO. 1-WGCB-R-122

NORTH ANNA POWER STATION
VIRGINIA ELECTRIC AND POWER COMPANY

SWEC J.O. NO. 14238.18

ISSUE	DESCRIPTION	CHKD	CORR	APPR	DATE
3					
2					
1	ORIGINAL ISSUE	—	—	—	1-17-77

DWG NO. 11715-PSSK-103BY.5 SH 1 OF 1
REV 1

SAFETY RELATED
DC 90-13-1

ISSUED FOR CONSTRUCTION OF DC-90-13-1

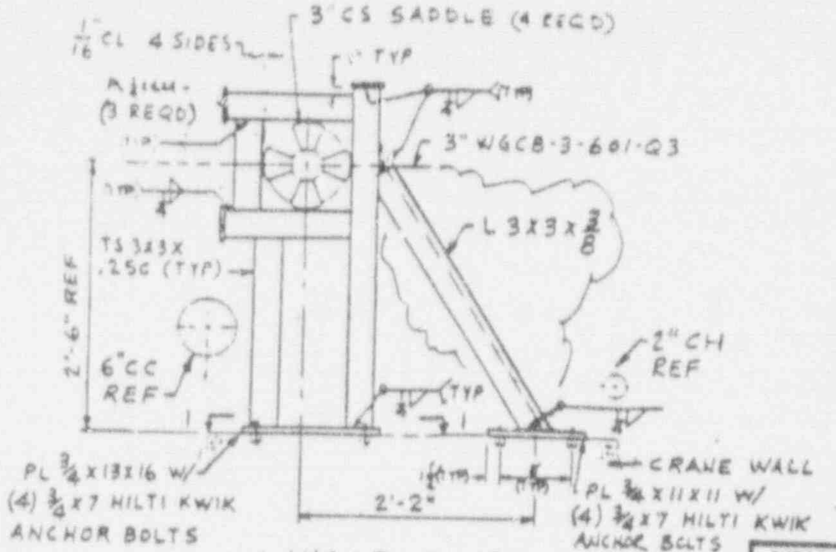
	BY	DATE
PREPARED	DCH	4/17/92
CHECKED	RRB	4/17/92
APPROVED	DAM	4/17/92
REVISIONS BY	①	
DATE		

TITLE
STEAM GENERATOR BLOWDOWN
MODIFICATIONS
CUBICLE C
PIPE SUPPORT 1-WGCB-R-122

CLIENT	VIRGINIA POWER
PROJECT	NORTH ANNA 1
J.O. NO.	02072.0710
①SSUE BY	SWEC
DATE	
NUMBER	SHEET 1 OF 1

N9013-1-M-1PSSK103BY.05-0

STONE & WEBSTER ENGINEERING CORPORATION

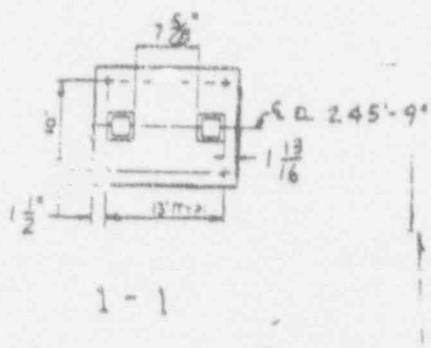


1-WGCB-R-121
PLAN

TRANSPOSED INFORMATION CHECKED BY PAUL WONG
Signature: Paul Wong — Date: 1-27-87

INFORMATION TRANSPOSED BY R.E. HULL
Signature: R.E. Hull — Date: 10-15-85

REF DWGS
11715-FP-3G
11715-MSK-103A7
11715-FC-16F



1-1

SAFETY RELATED

STM. GEN. BLOWDOWN SYS.
REAC. CONTAINMENT
UNIT 1

SUPPORT NO. 1-WGCB-R-121

NORTH ANNA POWER STATION
VIRGINIA ELECTRIC AND POWER COMPANY

SWEC J.O. NO. 14238.18

ISSUE	DESCRIPTION	CHKD	CORR	APPR	DATE
3					
2					
1	ORIGINAL ISSUE	-	-	-	1-27-87

DWG NO. 11715-PSSK-103BY.06 SH 1 of 1 — REV 1

SAFETY RELATED
DC 90-13-1

ISSUED FOR CONSTRUCTION OF DC-90-13-1

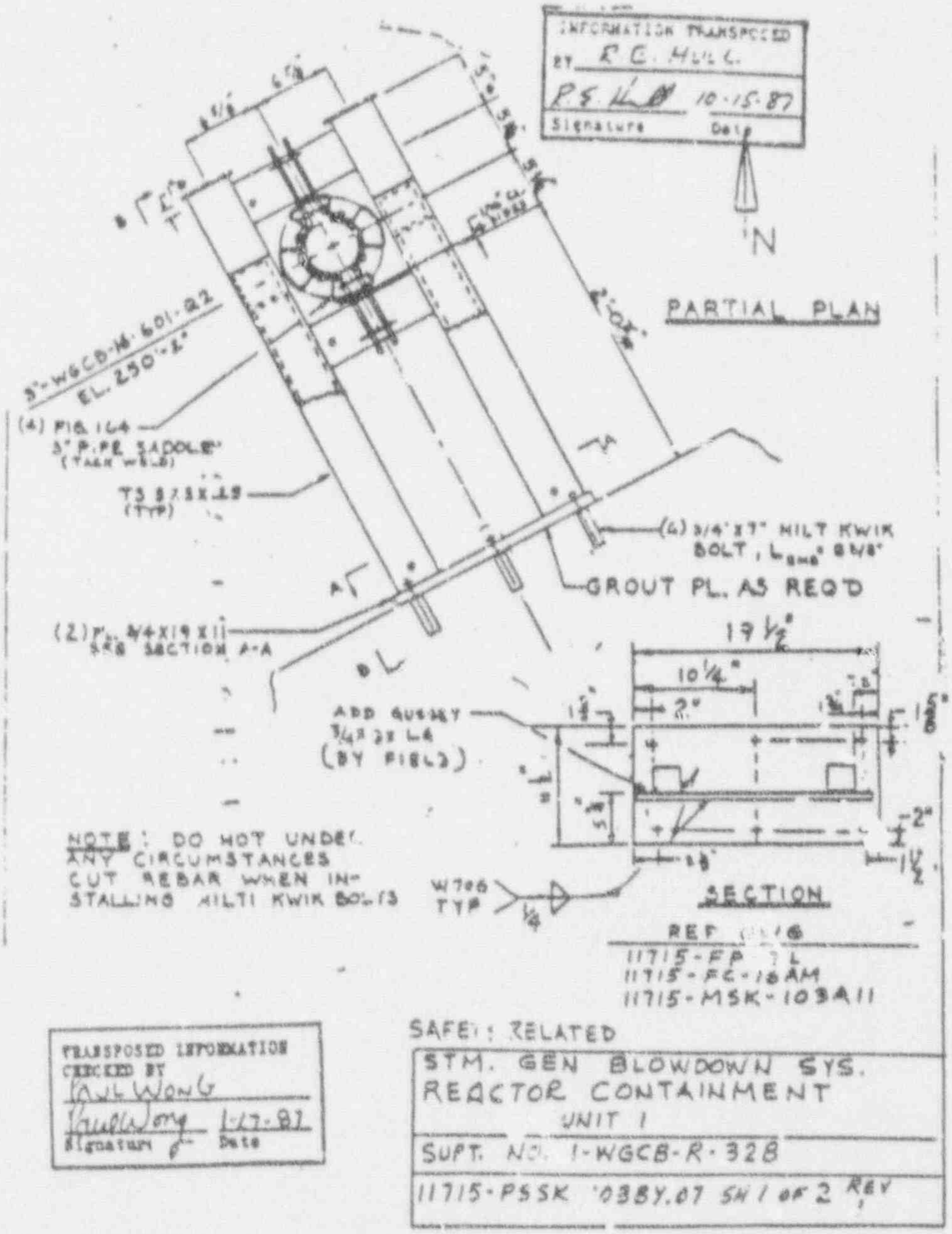
	BY	DATE
PREPARED	RDH	4/7/92
CHECKED	RHB	4/7/92
APPROVED		4/17/92
REVISIONS BY	①	
DATE		

TITLE: STEAM GENERATOR BLOWDOWN MODIFICATIONS CUBICLE C PIPE SUPPORT 1-WGCB-R-121

CLIENT	VIRGINIA POWER
PROJECT	NORTH ANNA 1
J.O. NO.	02072.0710
ISSUE BY	SWEC
DATE	
NUMBER	SHEET 1 OF 1

N9013-1-M-1PSSK103BY.06-0

STONE & WEBSTER ENGINEERING CORPORATION

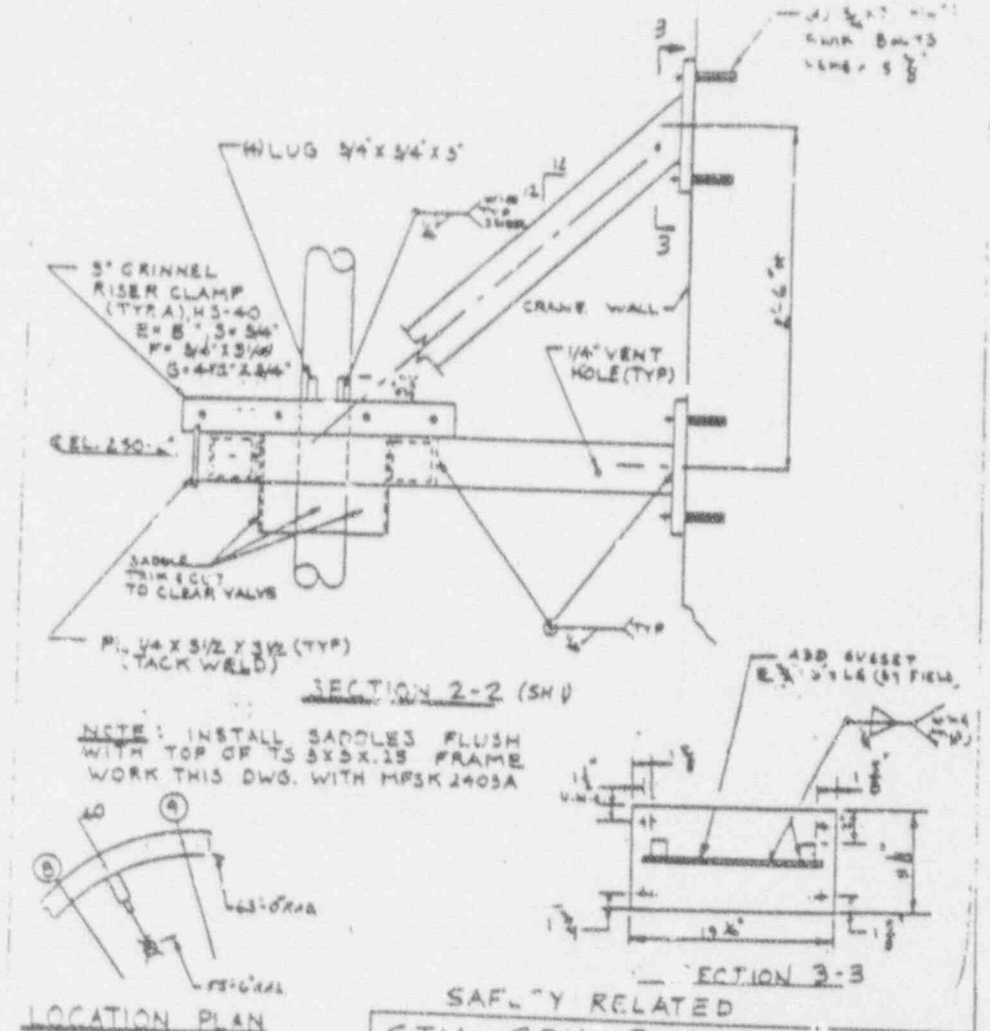


ISSUED FOR CONSTRUCTION OF DC-90-13-1

SAFETY RELATED
DC 90-13-1

	BY	DATE	TITLE	CLIENT	VIRGINIA POWER
PREPARED	DDA	4/17/92	STEAM GENERATOR BLOWDOWN MODIFICATIONS CUBICLE C PIPE SUPPORT 1-WGCB-R-328	PROJECT	NORTH ANNA 1
CHECKED	RAB	4/17/92		J.O. NO.	02072.0710
APPROVED	BSM	4/17/92		ISSUE BY	SWEC
REVISIONS BY	①	②	③	DATE	
DATE				NUMBER	SHEET 1 OF 2 N9013-1-M-1PSSK103BY.07-0

STONE & WEBSTER ENGINEERING CORPORATION



SAFETY RELATED

STM. GEN. BLOWDOWN SYS
REACTOR CONTAINMENT
UNIT 1

SUPT. NO. 1-WGCB-R-328

11715-PSSK-103BY.07 SH 2 OF 2 REV 1

ISSUED FOR CONSTRUCTION OF DC-90-13-1

SAFETY RELATED
DC 90-13-1

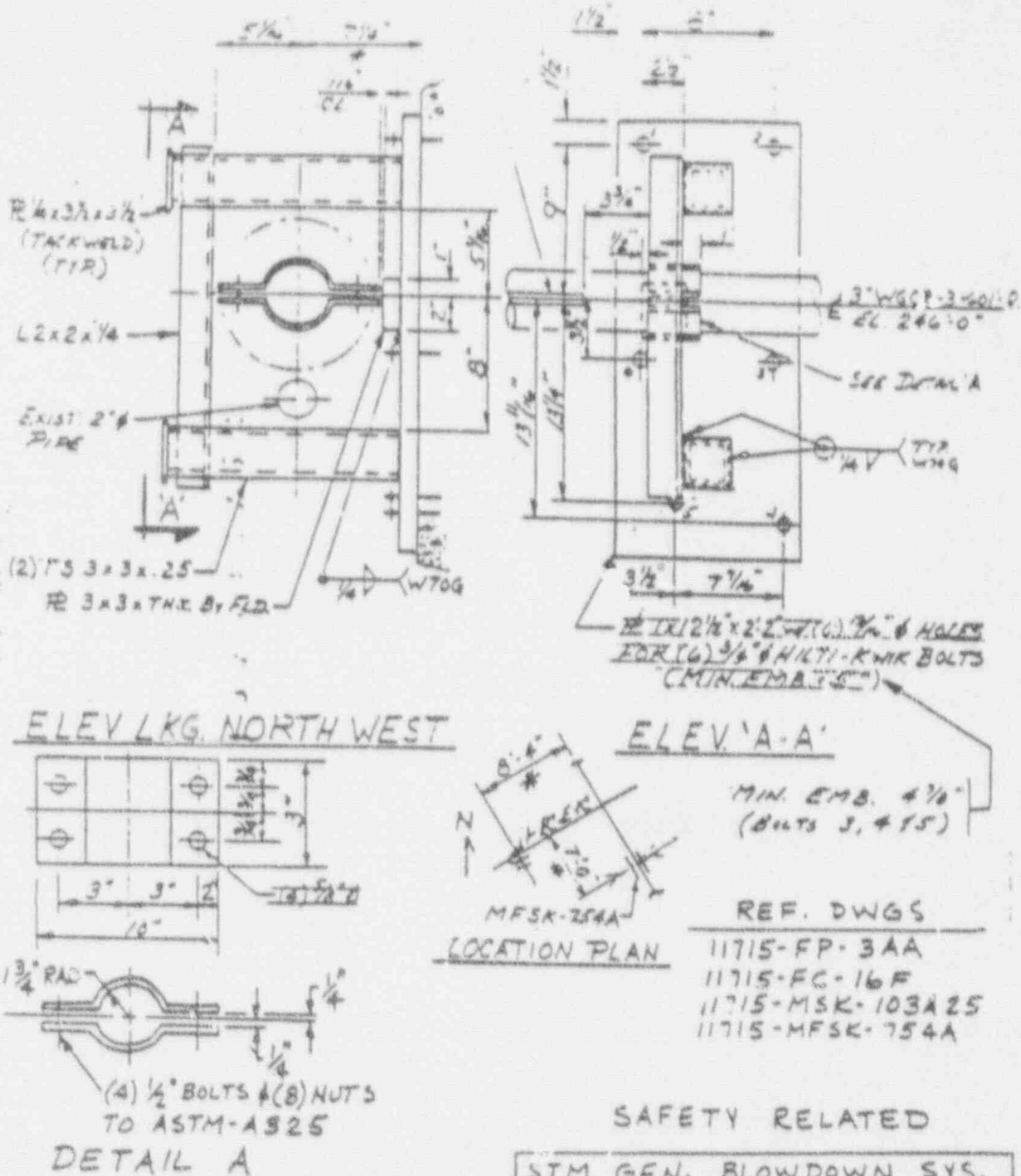
	BY	DATE
PREPARED	DCA	4/17/92
CHECKED	RAB	4/17/92
APPROVED	[illegible]	4/17/92

TITLE
STEAM GENERATOR BLOWDOWN
MODIFICATIONS
CUBICLE C
PIPE SUPPORT 1-WGCB-R-328

CLIENT	VIRGINIA POWER
PROJECT	NORTH ANNA 1
J.O. NO.	02072.0710
ISSUE BY	SWEC
DATE	
NUMBER	SHEET 2 OF 2
	N9013-1-M-1PSSK103BY.07-0

REVISIONS BY	①	②	③	④	⑤
DATE					

STONE & WEBSTER ENGINEERING CORPORATION



REF. DWGS
11715-FP-3AA
11715-FC-16F
11715-MSK-103A25
11715-MFSK-754A

SAFETY RELATED

TRANSPOSED INFORMATION CHECKED BY PAUL WONG	
Paul Wong	2-2-87
Signature	Date

INFORMATION TRANSPOSED BY R.E. HULL	
R.E. Hull	1-29-87
Signature	Date

STM. GEN. BLOWDOWN SYS.
REAC. CONTMT. - CUBICLE C
UNIT 1

SUPT. NO. 1-WGCB-R-MFSK-754A

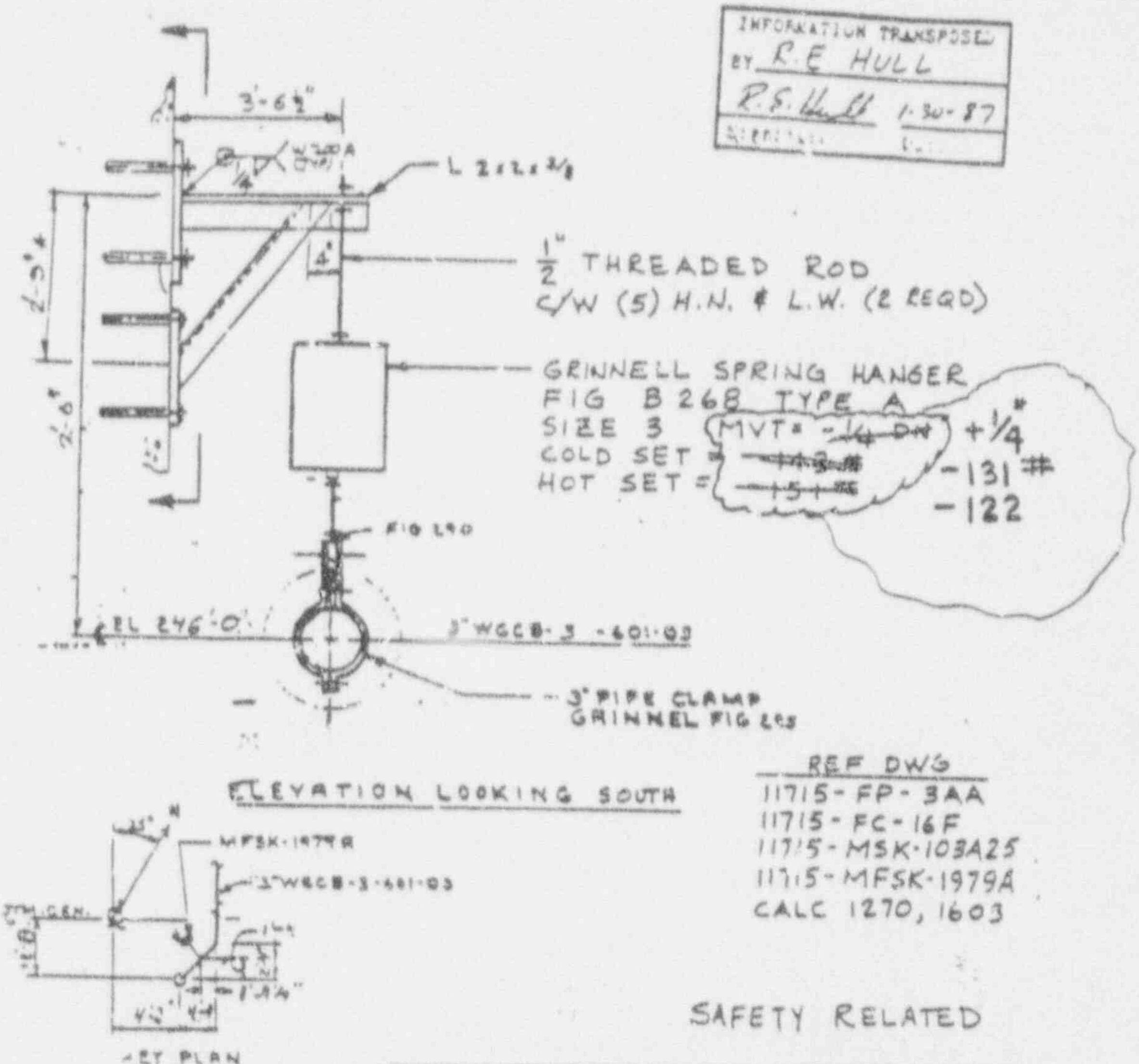
11715-PSSK-103BY.08 REV 1
SH 1 OF 1

ISSUED FOR CONSTRUCTION OF DC-90-13-1

SAFETY RELATED
DC 90-13-1

	BY	DATE	TITLE	
PREPARED		4/17/92	STEAM GENERATOR BLOWDOWN MODIFICATIONS CUBICLE C	CLIENT VIRGINIA POWER
CHECKED	RAB	4/17/92		PROJECT NORTH ANNA 1
APPROVED		4/17/92	PIPE SUPPORT 1-WGCB-R-MFSK-754A	J.O. NO. 02072.0710
				ISSUE BY SWEC
				DATE
REVISIONS BY				NUMBER SHEET 1 OF 1
DATE				N9013-1-M-1PSSK103BY.08-0

INFORMATION TRANSPOSED
BY R.E HULL
R.E. Hull 1-30-87



REF DWG
11715-FP-3AA
11715-FC-16F
11715-MSK-103A25
11715-MFSK-1979A
CALC 1270, 1603

SAFETY RELATED

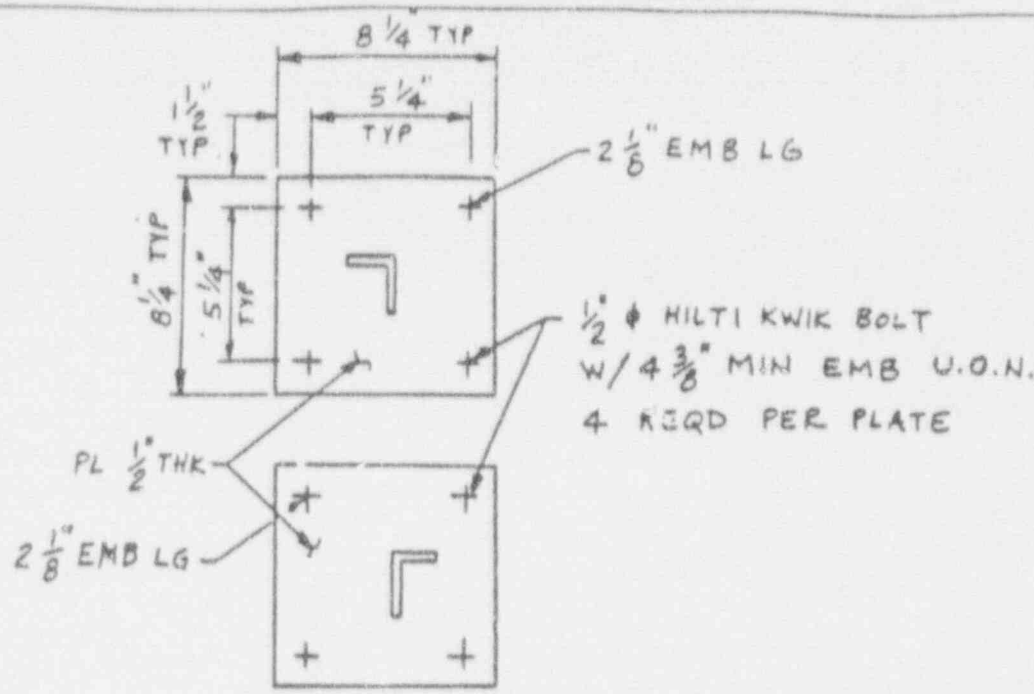
TRANSPOSED INFORMATION
CHECKED BY
PAUL WONG
Paul Wong 2-2-87
Signature Date

STM. GEN. BLOWDOWN SYS. REACTOR CONTAINMENT-CUBICLE C UNIT 1
SUPT NO. 1-WGCB-SH-MFSK-1979A
11715-PSSK-103BY.09 SH 1 OF 2 REV 1

ISSUED FOR CONSTRUCTION OF DC-90-13-1

SAFETY RELATED
DC 90-13-1

	BY	DATE	TITLE	CLIENT VIRGINIA POWER
PREPARED	DCH	4/17/92	STEAM GENERATOR BLOWDOWN	PROJECT NORTH ANNA 1
CHECKED	RAB	4/17/92	MODIFICATIONS	J.O. NO. 02072.0710
APPROVED	DBM	4/17/92	CUBICLE C	ISSUE BY SWEC
			PIPE SUPPORT 1-WGCB-SH-MFSK-1979A	DATE
REVISIONS BY	①	① ① ③ ③		NUMBER SHEET 1 OF 2
DATE				N9013-1-M-1PSSK103BY.09-0



SECT 1-1
(SH 1)

SAFETY RELATED

STM GEN BLOWDOWN SYS. REAC. CONTMT. - CUB C UNIT 1

SUPPORT NO. 1-WGCB-SH-MFSK-1979A

SWEC J.O. NO. 14238.18

3					
2					
1	ORIGINAL ISSUE		-	-	1-1-91
	DESCRIPTION	CHKD	CORR	APPR	DATE

NORTH ANNA POWER STATION
VIRGINIA ELECTRIC AND POWER COMPANY

DWG NO. 11715-PSSK-103BY.09 SH 2 of 2 — REV 1

SAFETY RELATED
DC 90-13-1

ISSUED FOR CONSTRUCTION OF DC-90-13-1

	BY	DATE
PREPARED	WCA	4/1/92
CHECKED	RAB	4/17/92
APPROVED	[illegible]	4/17/92

TITLE
STEAM GENERATOR BLOWDOWN
MODIFICATIONS
CUBICLE C
PIPE SUPPORT 1-WGCB-SH-MFSK-1979A

CLIENT	VIRGINIA POWER
PROJECT	NORTH ANNA 1
J.O. NO.	02072.0710
ISSUE BY	SWEC
DATE	
NUMBER	SHEET 2 OF 2
	N9013-1-M-1PSSK103BY.09-0

REVISIONS BY	①	②	③	④	⑤
DATE					

GENERAL NOTES:

1. ALL WORK SHALL BE PERFORMED IN ACCORDANCE WITH THE VIRGINIA POWER ACCIDENT PREVENTION MANUAL.

2. ALL WELDING SHALL BE PERFORMED IN ACCORDANCE WITH APPROVED STATION PROCEDURES.

3. ALL STRUCTURAL STEEL IS TO BE PREPARED AND PAINTED IN ACCORDANCE WITH SPECIFICATION NAS-3001.

4. PIPE SUPPORT LOCATION MAY DEVIATE FROM THAT SPECIFIED ON THE PIPING DRAWINGS BY ± 3" AXIALLY. THIS TOLERANCE DOES NOT ALLOW A SUPPORT TO BE RELOCATED TO THE OTHER SIDE OF ELBOWS, TEES, VALVES, ETC., OR ONTO PIPE WELDS.

5. SCALE = NONE

6. FIELD TO CUT AND TRIM PIPE PROTECTION SADDLES AS REQUIRED TO CLEAR PIPE WELDS AND ELBOW & VALVE WELDS.
   FOR PIPE PROTECTION SADDLE & WELD DETAIL SEE DRAWING N9013-1-M-700.16

7. IF THE TUBE STEEL FOR EXISTING SUPPORTS IS BEING TRIMMED AND THE TUBE END IS BEING CLOSED OFF, A VENT HOLE WILL BE REQUIRED IF NOT PREVIOUSLY PROVIDED IN THE REMAINING TUBE SECTION. IF POSSIBLE, VENT HOLES WILL BE LOCATED IN TUBE END CAPS. OTHER VENT HOLES WILL BE LOCATED ON THE BOTTOM OF HORIZONTAL TUBE MEMBERS AND ON THE SIDE OF VERTICAL TUBE MEMBERS.

ISSUED FOR CONSTRUCTION OF DC-90-13-1

SAFETY RELATED
DC 90-13-1

	BY	DATE	TITLE			CLIENT VIRGINIA POWER
PREPARED	R.E.H	9/17/92	STEAM GENERATOR BLOWDOWN MODIFICATIONS GENERAL NOTES			PROJECT NORTH ANNA 1
CHECKED	RAB	9/17/92				J.O. NO. 02072.0710
APPROVED	[illegible]	9/17/92				ISSUE BY SWEC
						DATE
REVISIONS BY	0	1	2	3	4	NUMBER SH 1 OF 1
DATE						N9013-1-M-700.01 - 1

1

IMAGE EVALUATION
TEST TARGET (MT-3)

150mm

6"

1

IMAGE EVALUATION
TEST TARGET (MT-3)

150mm

6"

1

IMAGE EVALUATION
TEST TARGET (MT-3)

150mm

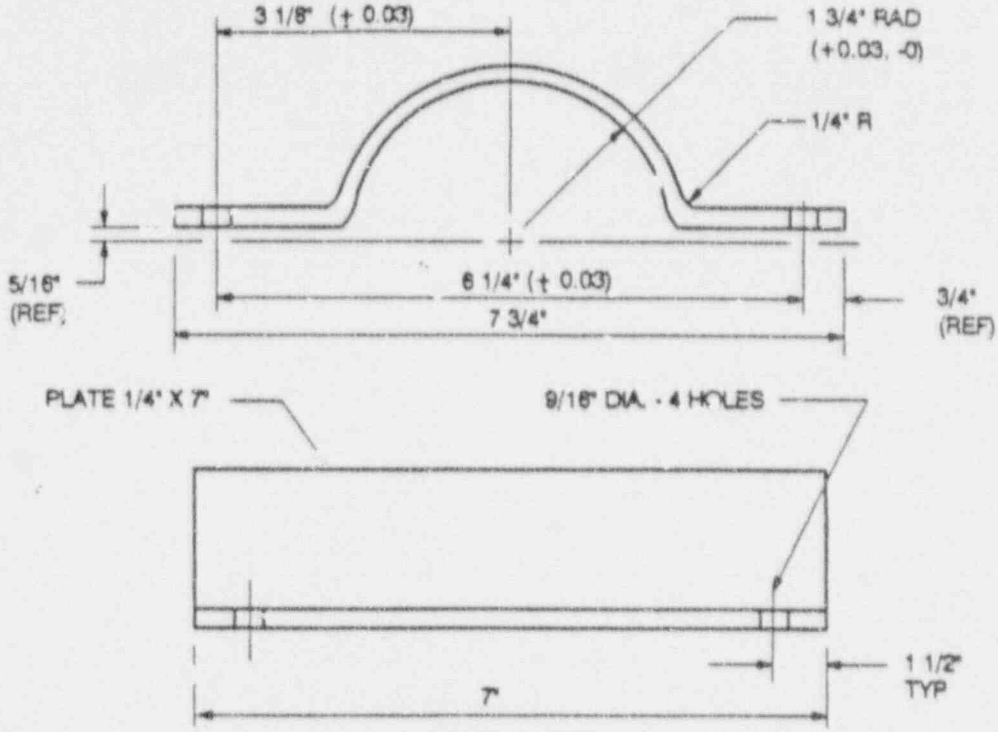
6"

1

IMAGE EVALUATION
TEST TARGET (MT-3)

150mm

6"



DETAIL ITEM 1

NOTES:

1. GAPS BETWEEN CLAMP HALVES ON OPPOSITE SIDES OF PIPE SHALL BE EQUAL WITHIN 1/8"
2. HAND TIGHTEN JAM NUTS (ITEM 5), MAINTAINING EQUAL GAP BETWEEN CLAMP HALVES. ROTATE TIGHTENING OF NUTS (ITEM 5) UNTIL A TORQUE OF 50 FT-LBS (+5,-0) IS ACHIEVED. PREVENT THE JAM NUTS FROM ROTATING AND TORQUE NUTS (ITEM 4) TO 55 FT-LBS (+5,-0).
3. INSIDE SURFACES OF CLAMP HALVES AND MATING SURFACES OF RUN PIPE ARE TO BE CLEAN, DRY, UNPAINTED, AND FREE OF GREASE, OIL, DEBRIS, ETC.
4. HOLES MAY BE MACHINED OR PUNCHED, BUT SHALL BE INDEXED FROM CLAMP HALF CENTER LINE.
5. WELD USING E-70XX ELECTRODES.
6. CLAMP ANCHOR TO BE INSTALLED AND BOLTS TORQUED PRIOR TO WELDING TO SUPPORT STRUCTURE.
7. REFER TO GENERAL NOTES ON DRAWING S9013-1-M-700.01

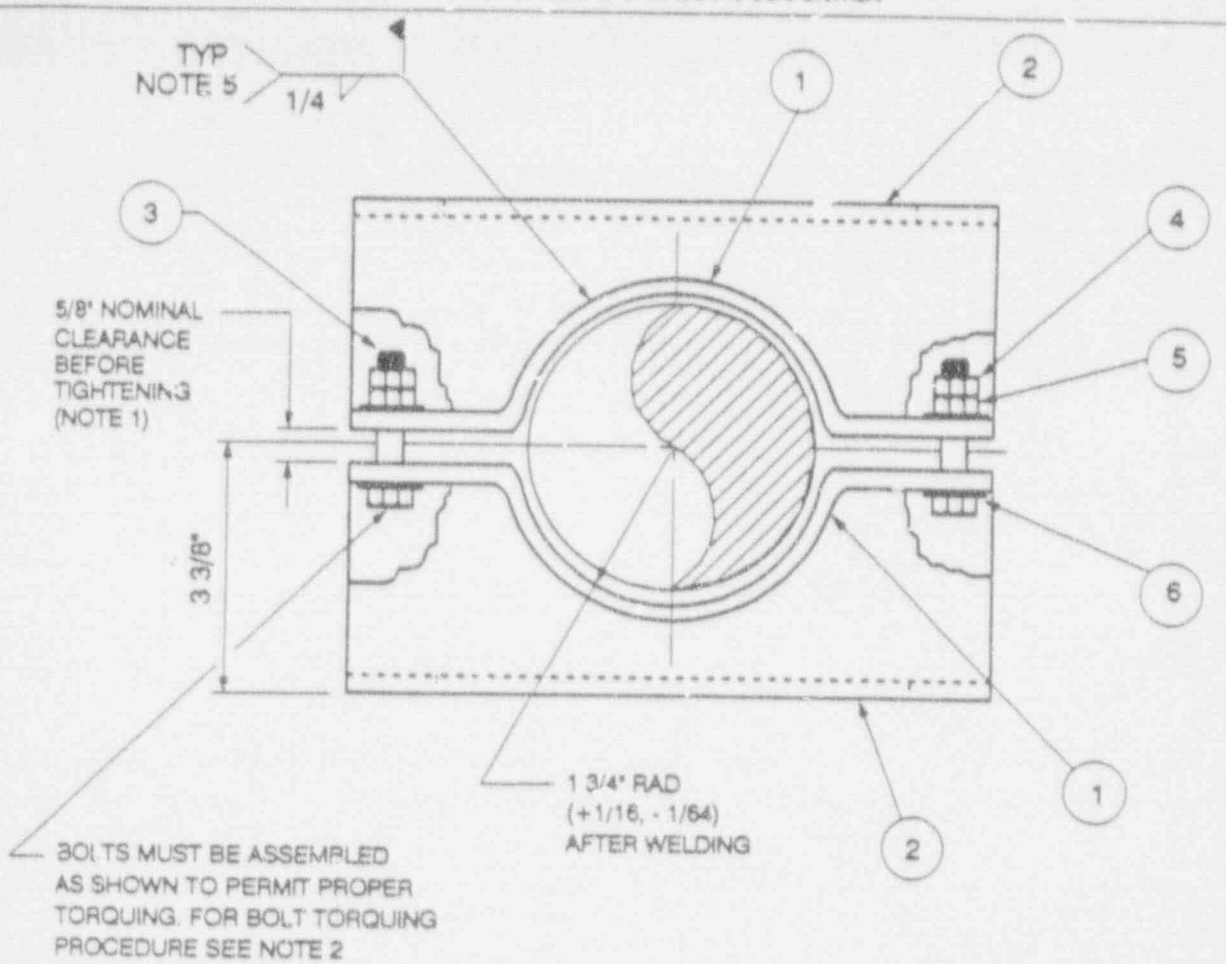
ISSUED FOR CONSTRUCTION OF DC-90-13-1

SAFETY RELATED
DC 90-13-1

	BY	DATE	TITLE
PREPARED	RDM	4/17/92	STEAM GENERATOR BLOWDOWN MODIFICATIONS 3" CLAMP ANCHOR ASSEMBLY
CHECKED	RAB	4/17/92	
APPROVED	[illegible]	4/17/92	

CLIENT	VIRGINIA POWER
PROJECT	NORTH ANNA 1
J.O. NO.	02072.0710
ISSUE BY	SWEC
DATE	
NUMBER	SHEET 1 OF 3
	N9013-1-M-700.07-0

REVISIONS BY	0	1	2	3	4
DATE					



SECTION 1-1

BILL OF MATERIAL

ITEM	QTY	DESCRIPTION	MAT'L
1	2	PLATE 1/4" X 7" (LENGTH TO SUIT BY FIELD)	A36
2	2	TS 6 X 6 X 3/8" (LENGTH TO SUIT BY FIELD)	A500 GR. B
3	4	1/2" BOLT X 2 1/2" LONG (13-UNC-2A)	A193 GR B7
4	4	1/2" HEAVY HEX NUT (13-UNC-2B)	A194 GR 2H
5	4	1/2" HEAVY HEX JAM NUT (13-UNC-2B)	A194 GR 2H
6	4	1/2" FLAT WASHERS	F-436

ISSUED FOR CONSTRUCTION OF DC-90-13-1

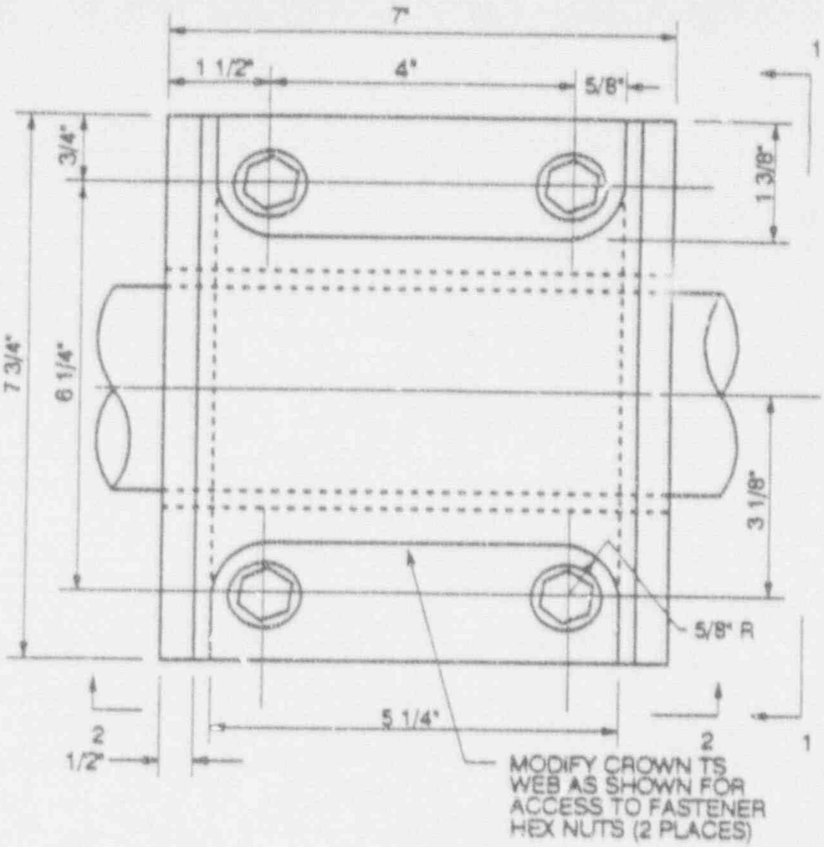
SAFETY RELATED
DC 90-13-1

	BY	DATE
PREPARED	DOA	4/17/92
CHECKED	RBB	4/17/92
APPROVED	DBM	4/17/92

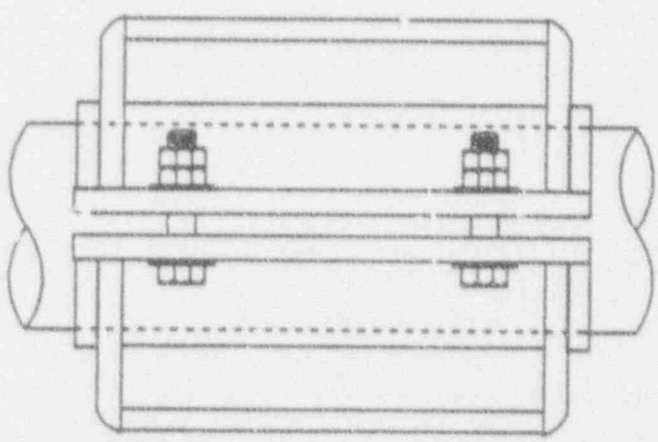
TITLE
STEAM GENERATOR BLOWDOWN MODIFICATIONS
3" CLAMP ANCHOR ASSEMBLY

CLIENT	VIRGINIA POWER
PROJECT	NORTH ANNA 1
J.O. NO.	02072.0710
ISSUE BY	SWEC
DATE	
NUMBER	SHEET 2 OF 7
	N9013-1-M-700.07-0

REVISIONS BY	0	1	2	3	4
DATE					



PLAN VIEW

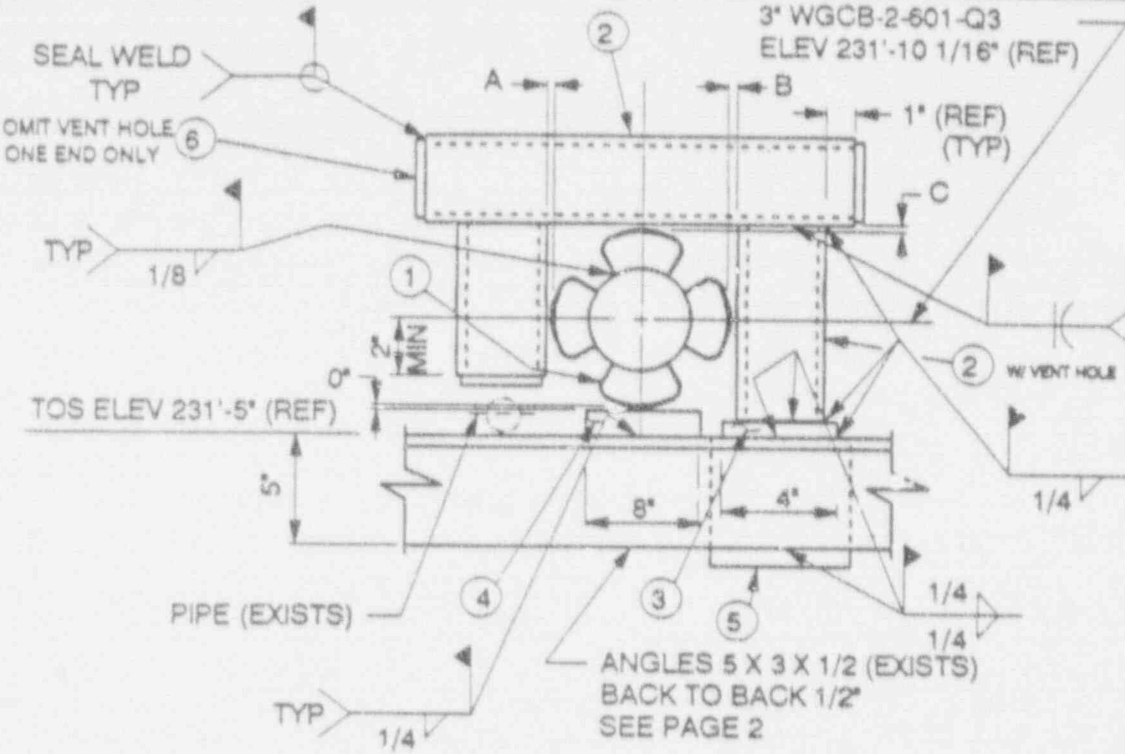


SECTION 2-2

ISSUED FOR CONSTRUCTION OF DC-90-13-1

SAFETY RELATED
DC 90-13-1

	BY	DATE	TITLE	
			STEAM GENERATOR BLOWDOWN MODIFICATIONS 3" CLAMP ANCHOR ASSEMBLY	CLIENT VIRGINIA POWER
PREPARED	RDM	4/17/92		PROJECT NORTH ANNA 1
CHECKED	RAB	4/17/92		J.O. NO. 02072.0710
APPROVED	[illegible]	4/17/92		ISSUE BY SWEC
				DATE
REVISIONS BY	0	0	0	NUMBER SHEET 3 OF 3
DATE				N9013-1-M-700.07-0



ELEV LOOKING NORTH EAST

NOTES:

1. FOR SUPPORT LOCATION SEE DWG. N9013-1-M-602
2. FOR GENERAL NOTES REFER TO DWG. N9013-1-M-700.01
3. FIELD TO REMOVE EXISTING PIPE ATTACHMENT.
4. CLEARANCES 1/16" ≤ (A+B) , (C) ≤ 5/32", FIELD SHIM TO SUIT.
5. THIS MODIFIED SUPPORT WAS PREVIOUSLY SHOWN AS DETAIL 26 ON DRAWING 11715-FP-3.

BILL OF MATERIAL

ITEM	QTY	DESCRIPTION	MAT'L
1	4	3" PROTECTION SADDLE FOR 3" INS, GRINNELL FIG 164	———
2	3	TS 3 X 3 X 1/4" (LENGTH TO SUIT BY FIELD)	A500 GR. B
3	1	PLATE 3/4" X 4" X 5"	A36
4	1	PLATE 8" X 5" (THICKNESS TO SUIT BY FIELD)	A36
5	1	PLATE 1/2" X 5" X 5 1/2"	A36
6	3	STEEL STRIP 1/8" X 2 3/4" X 2 3/4" W/ 1/4" VENT HOLE	A606

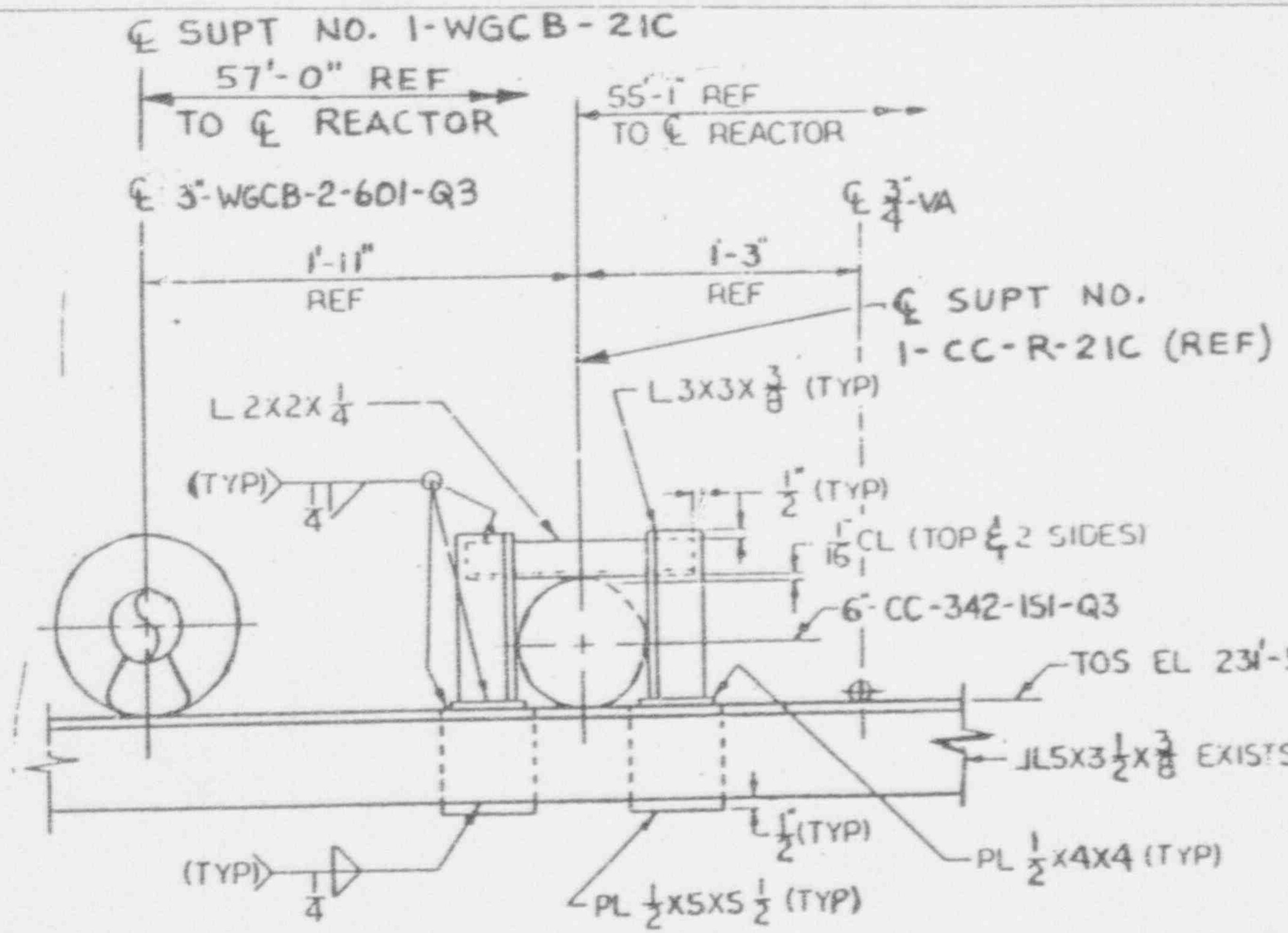
ISSUED FOR CONSTRUCTION OF DC-90-13-1

SAFETY RELATED
DC 90-13-1

	BY	DATE
PREPARED	R.E.A	9-17-92
CHECKED	[illegible]	9/17/92
APPROVED	[illegible]	9/17/92
REVISIONS BY		
DATE		

TITLE
STEAM GENERATOR BLOWDOWN
MODIFICATIONS
CUBICLE B
PIPE SUPPORT 1-WGCB-R-21C

CLIENT VIRGINIA POWER
PROJECT NORTH ANNA 1
J.O. NO. 02072.0710
ISSUE BY SWEC
DATE
NUMBER SHEET 1 OF 2
N9013-1-M-700.08-1



ELEVATION

FROM DET. 26, DWG. 11715-FP-3AT

ISSUED FOR CONSTRUCTION OF DC-90-13-1

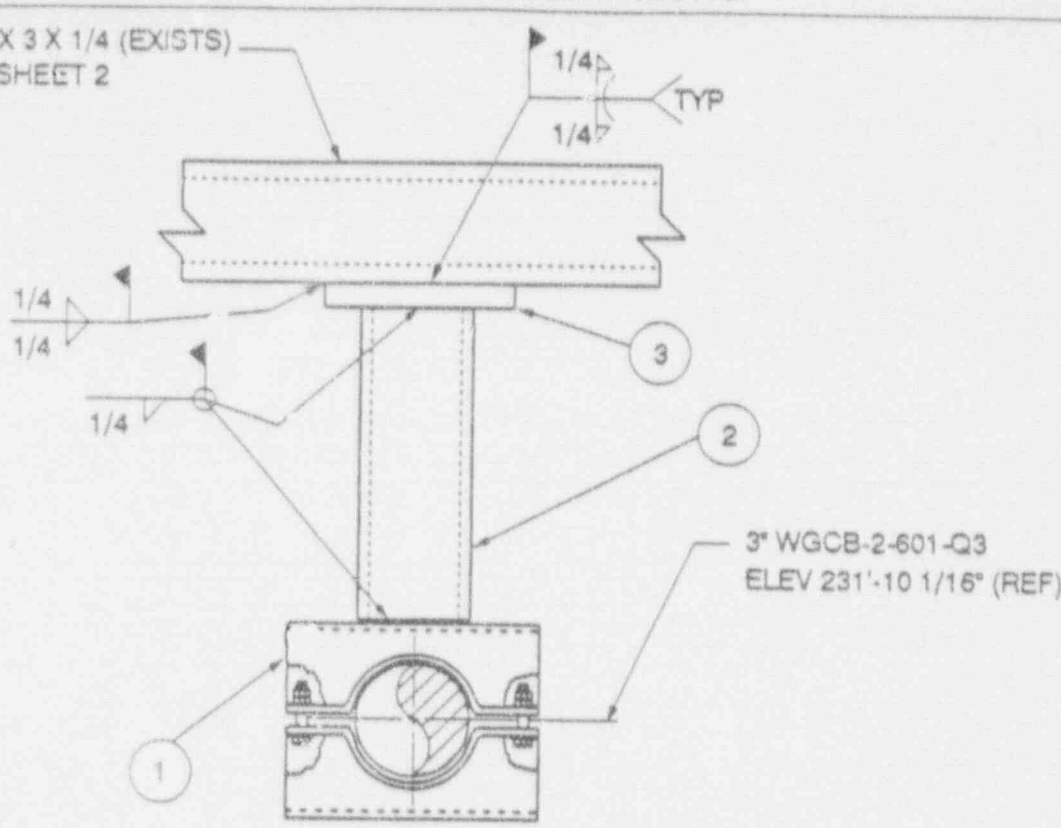
	BY	DATE
PREPARED	R.E.K.	9-17-92
CHECKED	RAB	9/17/92
APPROVED	[illegible]	9/17/92
REVISIONS BY		
DATE		

TITLE STEAM GENERATOR BLOWDOWN MODIFICATIONS
CUBICLE B
PIPE SUPPORT 1-WGCB-R-21C

SAFETY RELATED
DC 90-13-1

CLIENT	VIRGINIA POWER
PROJECT	NORTH ANNA 1
J.O. NO.	02072.0710
ISSUE BY	SWEC
DATE	
NUMBER	N9013-1-M-700.08-1 SHEET 2 OF 2

STONE & WEBSTER ENGINEERING CORPORATION



ELEVATION LOOKING NORTH

NOTES:

1. FOR SUPPORT LOCATION SEE DWG. N9013-1-M-602
2. FOR GENERAL NOTES REFER TO DWG. N9013-1-M-700.01
3. FIELD TO CUT & GRIND OFF EXISTING PIPE ATTACHMENT
4. THIS MODIFIED SUPPORT WAS PREVIOUSLY SHOWN AS DETAIL 23 ON DRAWING 11715-FP-3AT

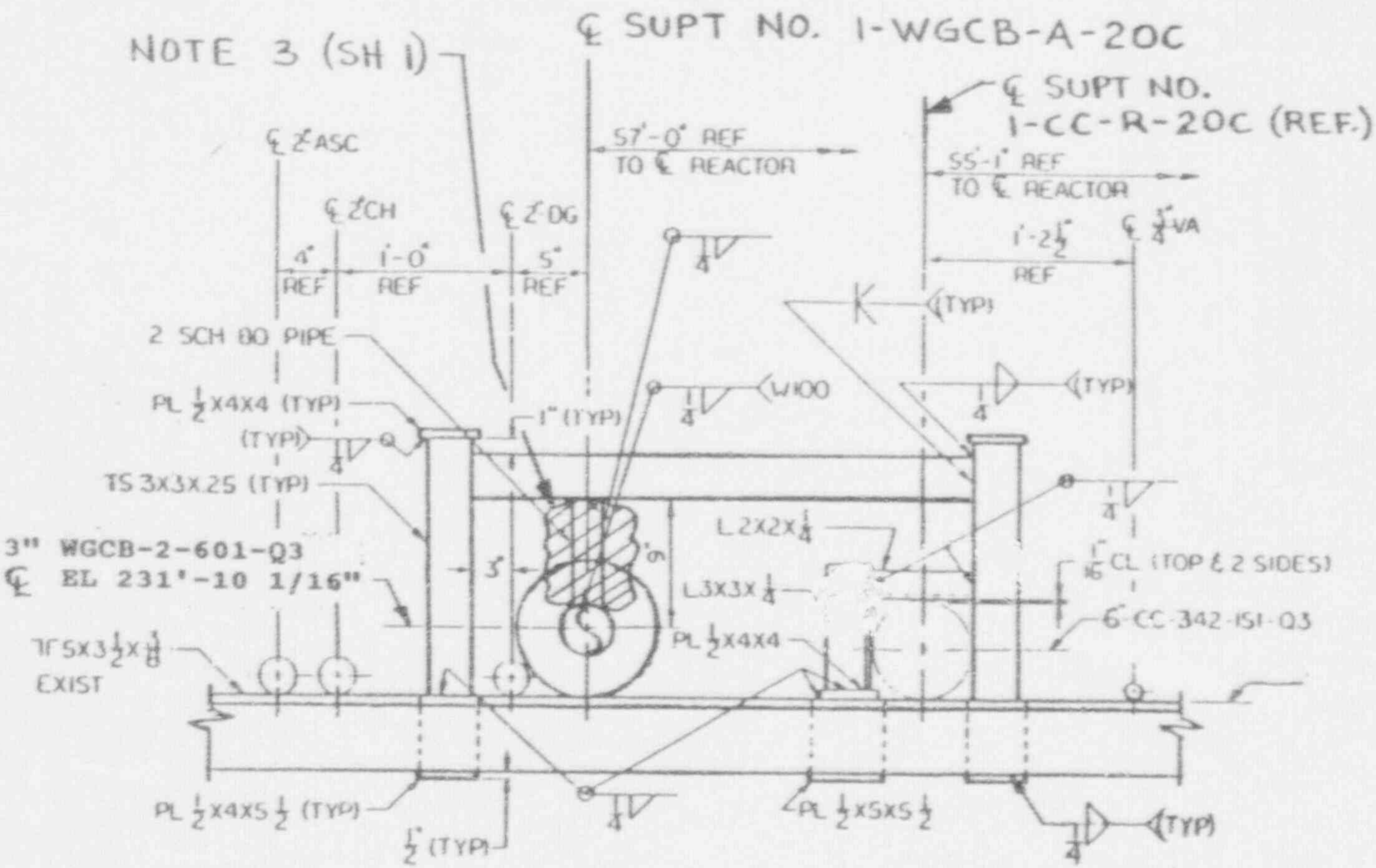
BILL OF MATERIAL

ITEM	QTY	DESCRIPTION	MAT'L
1	1	3" CLAMP ANCHOR PER DRAWING N9013-1-M-700.07	———
2	1	TS 3 X 3 X 1/4 (FIELD CUT TO SUIT)	A500 GR B
3	1	PLATE 1/2 X 4 X 4	A36

ISSUED FOR CONSTRUCTION OF DC-90-13-1

SAFETY RELATED
DC 90-13-1

	BY	DATE	TITLE	CLIENT VIRGINIA POWER
PREPARED	R.E.H	4-17-92	STEAM GENERATOR BLOWDOWN MODIFICATIONS	PROJECT NORTH ANNA 1
CHECKED	RAB	4/17/92	CUBICLE B	J.O. NO. 02072.0710
APPROVED	[illegible]	4/17/92	PIPE SUPPORT 1-WGCB-A-20C	ISSUE BY SWEC
				DATE
REVISIONS BY	0	0	0 0 0	NUMBER SHEET 1 OF 2
DATE				N9013-1-M-700.09-0



ELEVATION

FROM DET. 23, DWG. 11715-FP-3AT

ISSUED FOR CONSTRUCTION OF DC-90-13-1

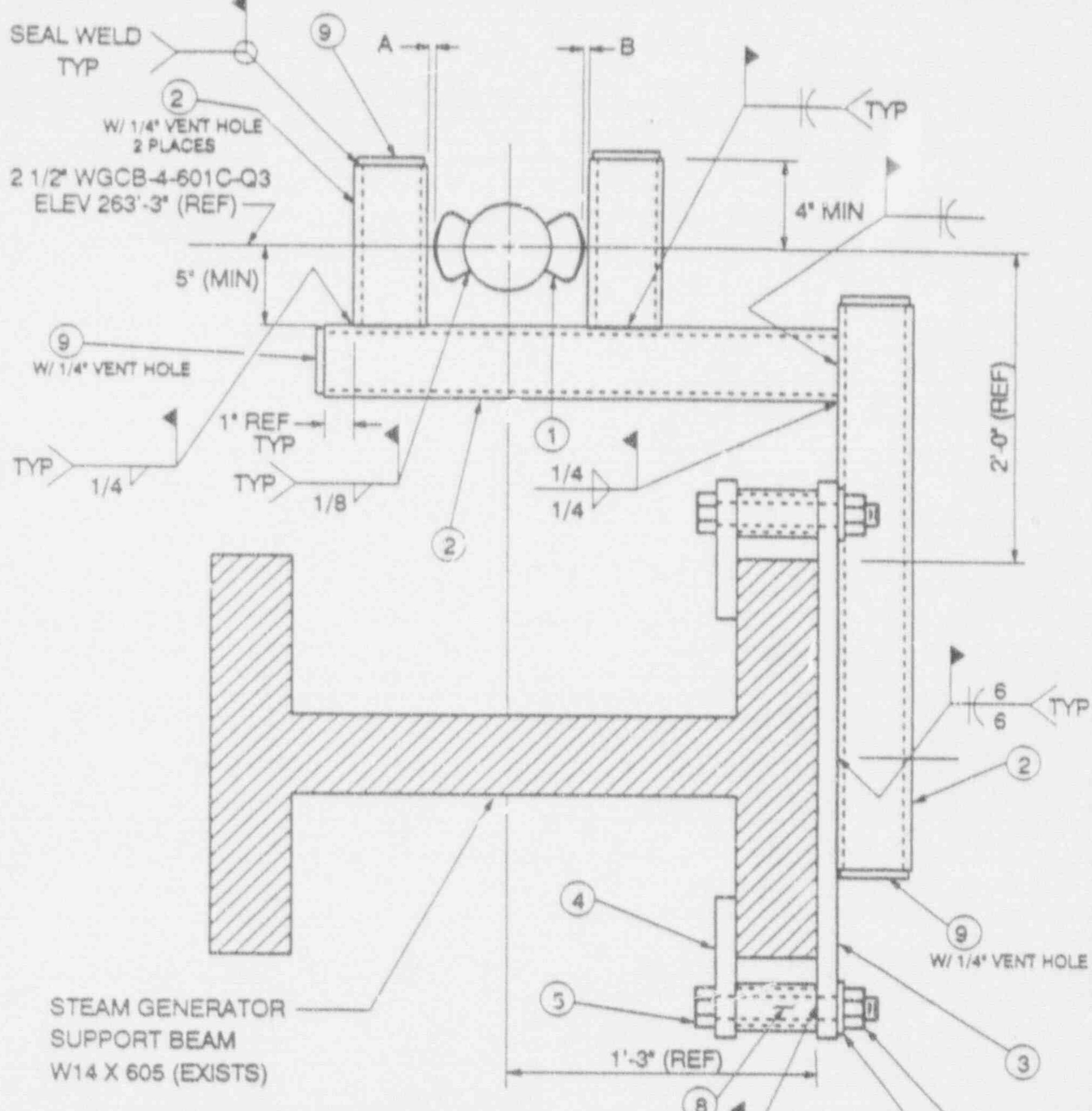
SAFETY RELATED
DC 90-13-1

STONE & WEBSTER ENGINEERING CORPORATION

	BY	DATE
PREPARED	R.E.K.	4-17-92
CHECKED	R.A.G.	4/17/92
APPROVED		4/17/92
REVISIONS BY		
DATE		

TITLE: STEAM GENERATOR BLOWDOWN MODIFICATIONS
CUBICLE B
PIPE SUPPORT 1-WGCB-A-20C

CLIENT: VIRGINIA POWER
PROJECT: NORTH ANNA 1
J.O. NO. 02072.0710
ISSUE BY: SWEC
DATE
NUMBER: SHEET 2 OF 2
N9013-1-M-700.09-0

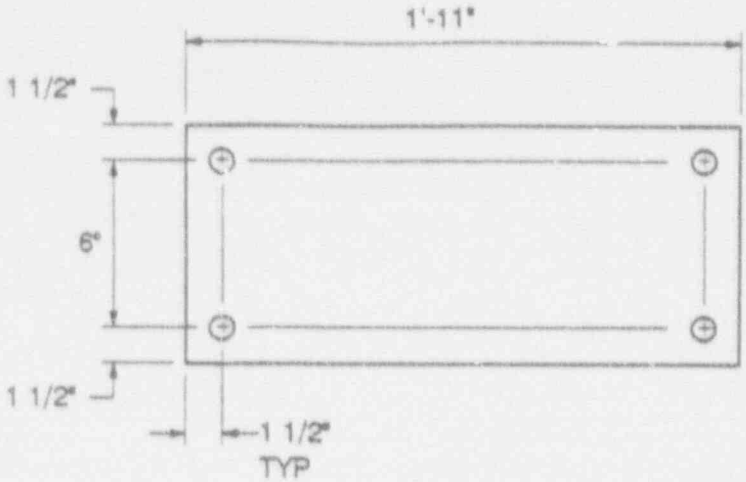


ELEV LOOKING NORTH-WEST

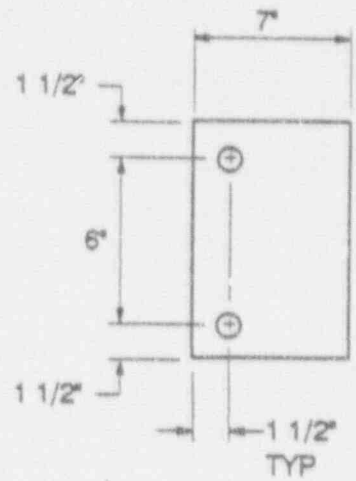
ISSUED FOR CONSTRUCTION OF DC-90-13-1

SAFETY RELATED
DC 90-13-1

	BY	DATE	TITLE		
			STEAM GENERATOR BLOWDOWN MODIFICATIONS CUBICLE A PIPE SUPPORT 1-WGCB-PH-4.1	CLIENT	VIRGINIA POWER
PREPARED	R.E.H	9/17/92		PROJECT	NORTH ANNA 1
CHECKED	[illegible]	9/17/92		J.O. NO.	02072.0710
APPROVED	[illegible]	9/17/92		ISSUE BY	SWEC
				DATE	
REVISIONS BY	0			NUMBER	SHEET 1 OF 2
DATE					N9013-1-M-700.10-1



DETAIL A



DETAIL B

NOTES:

1. FOR SUPPORT LOCATION SEE DWG. N9013-1-M-601
2. FOR GENERAL NOTES REFER TO DWG. N9013-1-M-700.01
3. TORQUE BOLTS (ITEM 5) TO 55 $^{+5}_{-0}$ FT-LB
4. CLEARANCES 1/16" $\leq$ (A+B) $\leq$ 5/32", FIELD SHIM TO SUIT.

BILL OF MATERIAL

ITEM	QTY	DESCRIPTION	MAT'L
1	2	2 1/2" PROTECTION SADDLE FOR 2" INS, GRINNELL FIG 162	————
2	4	TS 3 X 3 X 1/4" (LENGTH TO SUIT BY FIELD)	A500 GR. B
3	1	PLATE 3/4 X 9 X 23" (SEE DETAIL A)	A36
4	2	PLATE 3/4 X 9 X 7" (SEE DETAIL B)	A36
5	4	3/4"-10 UNC-2A X 7" LG	A325
6	4	3/4" HEX NUT (SEE NOTE 3)	A194 GR. 2H
7	4	3/4" FLAT STEEL WASHER	F436
8	4	TS 2 X 2 X 1/4" X 0'-4 1/8" LG	A500 GR. B
9	5	STEEL STRIP 1/8" X 2 3/4" X 2 3/4"	A606

SAFETY RELATED
DC 90-13-1

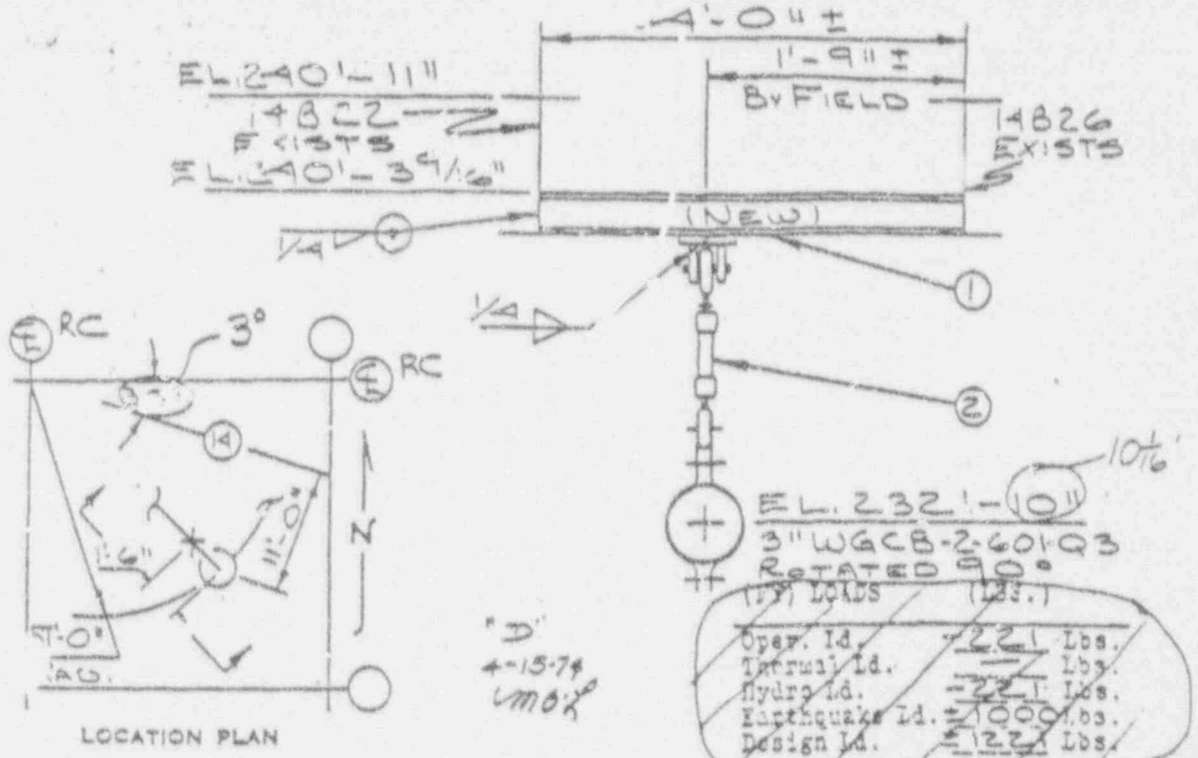
ISSUED FOR CONSTRUCTION OF DC-90-13-1

	BY	DATE	TITLE	
PREPARED	R.E.A.	9-17-92	STEAM GENERATOR BLOWDOWN MODIFICATIONS CUBICLE A PIPE SUPPORT 1-WGCB-PH-4.1	CLIENT VIRGINIA POWER
CHECKED	RAB	9/17/92		PROJECT NORTH ANNA 1
APPROVED	DBS	9/17/92		J.O. NO. 02072.0710
				ISSUE BY SWEC
				DATE
REVISIONS BY	0	0	0 0 0	NUMBER SHEET 2 OF 2
DATE				N9013-1-M-700.10-1

STONE & WEBSTER ENGINEERING CORPORATION

CUSTOMER SOUTHWEST FAB. & WELDING CO.
ORDER OR CONT. No. 0281-2
JOB NAME NORTH ANNA PWR. STA. UNIT # 1

REV.	1	2	3	4	5
ENG / DATE					



(1) LOADS	(LBS.)
Oper. Ld.	-221 Lbs.
Thermal Ld.	— Lbs.
Hydro Ld.	-221 Lbs.
Earthquake Ld.	±1000 Lbs.
Design Ld.	±1221 Lbs.

ITEM NO.	MATERIALS AND OPERATIONS	QUAN.	SHIP.
	SWAY STRUT ASSEMBLY CONSISTING OF:	ONE	
1	6" Wide Flange @ 15.5#/Ft., 3'-11" Lg., TW-61#	1	
2	Fig. 211 #1 Sway Strut Assembly, 3 1/2" O.D. Pipe,	1	
	Carbon Steel, W-5'-4 1/16", Load-1221#, (New Design)		
	Hanger Assembly Sk.&Eng.	1	
	Bundle& Tag	1	
	MK:# WGCB-R-131		
	Keeler & Long Epoxy White Primer # 7107		

FOR MATERIALS AND OPERATIONS SEE SKETCH NO. ______ REV. ______

REF. DRW'G. NOS. PIPE 11715-FP-3F-_ STEEL 11715-FS-11C-_ MARK NO. WGCB-R-131 SKETCH NO. 6147 REV. ____

11715 - GRINNELL SK. NO. 6147

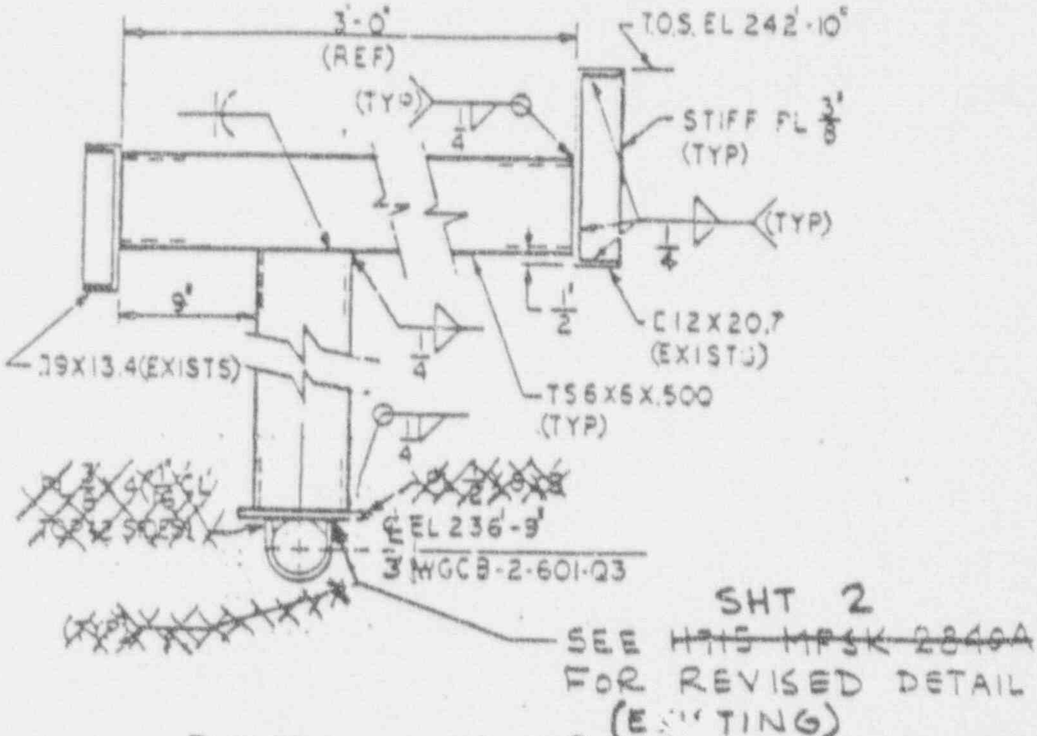
ISSUED FOR CONSTRUCTION OF DC-90-13-1

SAFETY RELATED
DC 90-13-1

	BY	DATE
PREPARED	DCM	4/17/92
CHECKED	RAB	4/17/92
APPROVED	VBM	4/17/92
REVISIONS BY	①	② ③ ④ ⑤
DATE		

TITLE
STEAM GENERATOR BLOWDOWN
MODIFICATIONS
CUBICLE B
PIPE SUPPORT 1-WGCB-R-131

CLIENT VIRGINIA POWER
PROJECT NORTH ANNA 1
J.O. NO. 02072.0710
① ISSUE BY SWEC
DATE
NUMBER SHEET 1 OF 1
N9013-1-M-700.11-0



DETAIL 202
H-323
ELEV. LKG S-EAST

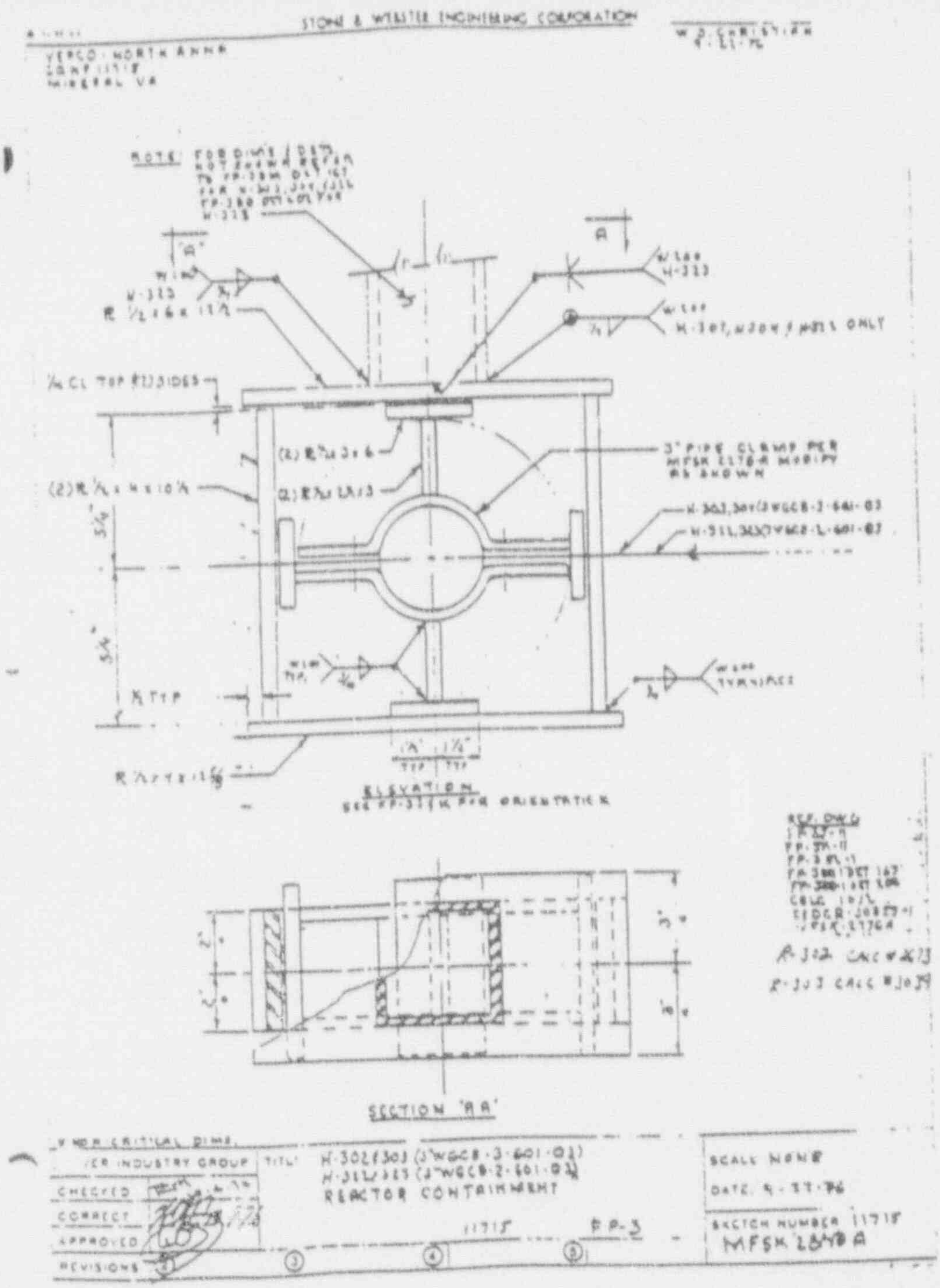
FROM DRAWING 11715-FP-3BQ

ISSUED FOR CONSTRUCTION OF DC-90-13-1

SAFETY RELATED
DC 90-13-1

	BY	DATE	TITLE		
PREPARED	RCK	4/17/92	STEAM GENERATOR BLOWDOWN MODIFICATIONS	CLIENT	VIRGINIA POWER
CHECKED	RAB	4/17/92	CUBICLE B	PROJECT	NORTH ANNA 1
APPROVED	[illegible]	4/17/92	PIPE SUPPORT 1-WGCB-R-323	J.O. NO.	02072.0710
				①ISSUE BY	SWEC
				DATE	
REVISIONS BY	①	②	③ ④ ⑤	NUMBER	SHEET 1 OF 3
DATE					N9013-1-M-700.12-0

STONE & WEBSTER ENGINEERING CORPORATION



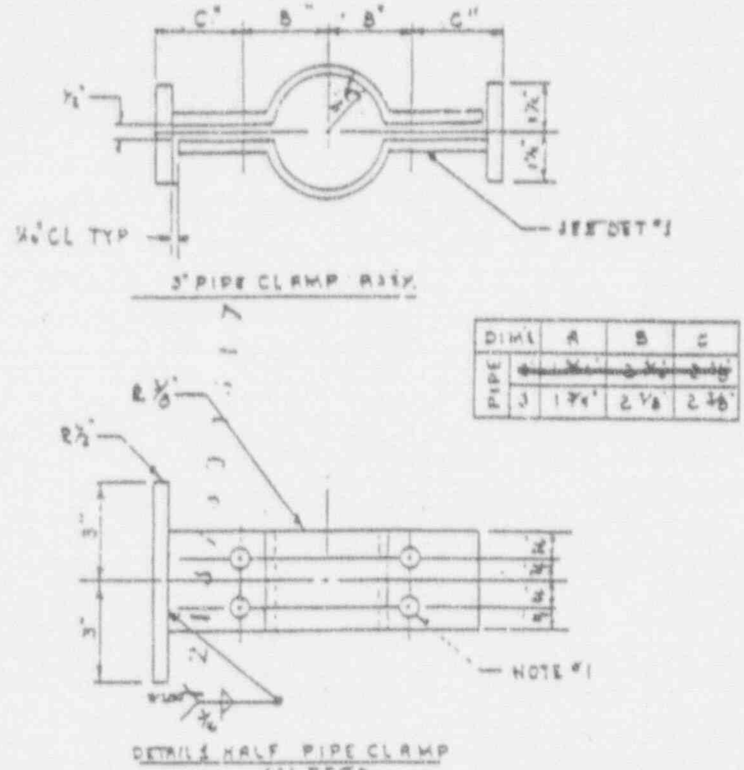
SAFETY RELATED
DC 90-13-1

ISSUED FOR CONSTRUCTION OF DC-90-13-1

	BY	DATE	TITLE	
PREPARED	DOA	4/17/92	STEAM GENERATOR BLOWDOWN MODIFICATIONS CUBICLE B PIPE SUPPORT 1-WGCB-R-323	CLIENT VIRGINIA POWER
CHECKED	RAB	4/17/92		PROJECT NORTH ANNA 1
APPROVED		4/17/92		J.O. NO. 02072.0710
				①ISSUE BY SWEC
				DATE
REVISIONS BY	①	② ③ ④ ⑤		NUMBER SHEET 2 OF 3
DATE				N9013-1-M-700.12-0

STONE & WEBSTER ENGINEERING CORPORATION

VEPCO NORTH ANNA
MINERAL - VIRGINIA



DIM	A	B	C
PIPE 3	1 7/8"	2 1/8"	2 3/8"

NOTES
1. USE (4) 5/8" x 3" BOLTS C/W (2) HEX NUTS, TORQUE TO TIGHT FIT
2. NOTCH INSULATION FOR PIPE CLAMP

ONE WAY RESTRAINT
2 & 3" INSULATED PIPE

SCALE NONE
DATE 6-1-76
MFSK 2176-A

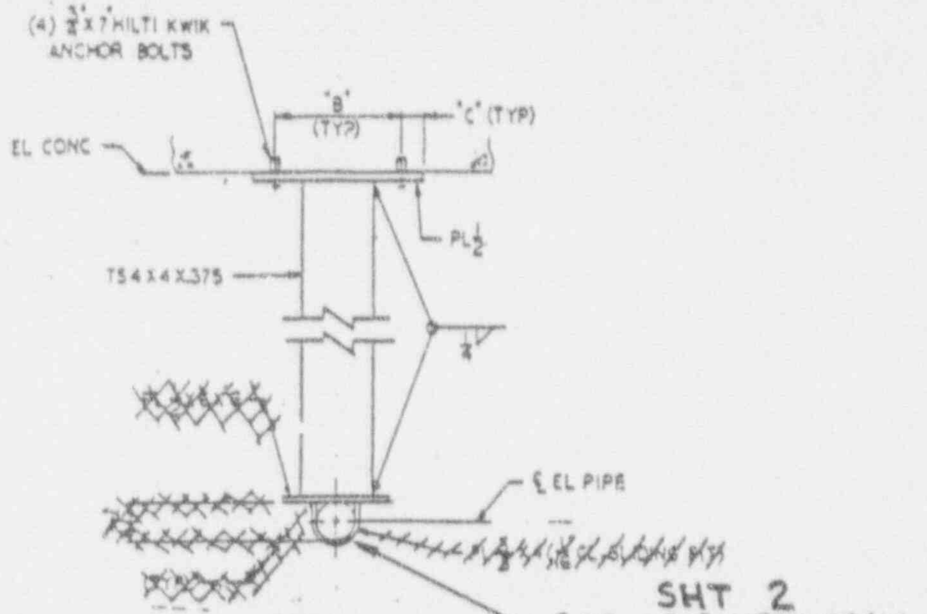
ISSUED FOR CONSTRUCTION OF DC-90-13-1

SAFETY RELATED
DC 90-13-1

	BY	DATE
PREPARED	DCH	4/17/92
CHECKED	RAB	4/17/92
APPROVED		4/17/92
REVISIONS BY	⓪	
DATE		

TITLE
STEAM GENERATOR BLOWDOWN
MODIFICATIONS
CUBICLE B
PIPE SUPPORT 1-WGCB-R-323

CLIENT VIRGINIA POWER
PROJECT NORTH ANNA 1
J.O. NO. 02072.0710
ISSUE BY SWEC
DATE
NUMBER SHEET 3 OF 3
N9013-1-M-700.12-0



DETAIL 167
H-NO SEE TABLE
ELEVATION
SCALE: NONE

TABLE FOR DETAIL 167 (G-3)							
CODE NO	LINE DESIGNATION	ORIENTATION	℄ EL PIPE	EL CONC	"A"	"B"	"C"
H-269	3-CC-203-151-Q3	LOOKING W	229-0	229-6	3 9/16	8"	1 1/2
H-272	3-CC-205-151-Q3	N-E	229-0	229-6	↓	↓	↓
H-300	4-CC-23-151-Q3	N-W	238-1'	230'-0	4 9/16"	10 1/2"	2"
H-301	3-CC-185-1501-Q3				3 9/16"	8"	1 1/2"
H-302	3-CC-186-1501-Q3						
H-303	3-WGCB-3-601-Q3						
H-304	3-WGCB-3-601-Q3		↓	↓			
H-322	3-WGCB-2-601-Q3	↓	236-9	237-9	↓	↓	↓

FROM DRAWING 11715-FP-3BM-1

ISSUED FOR CONSTRUCTION OF DC-90-13-1

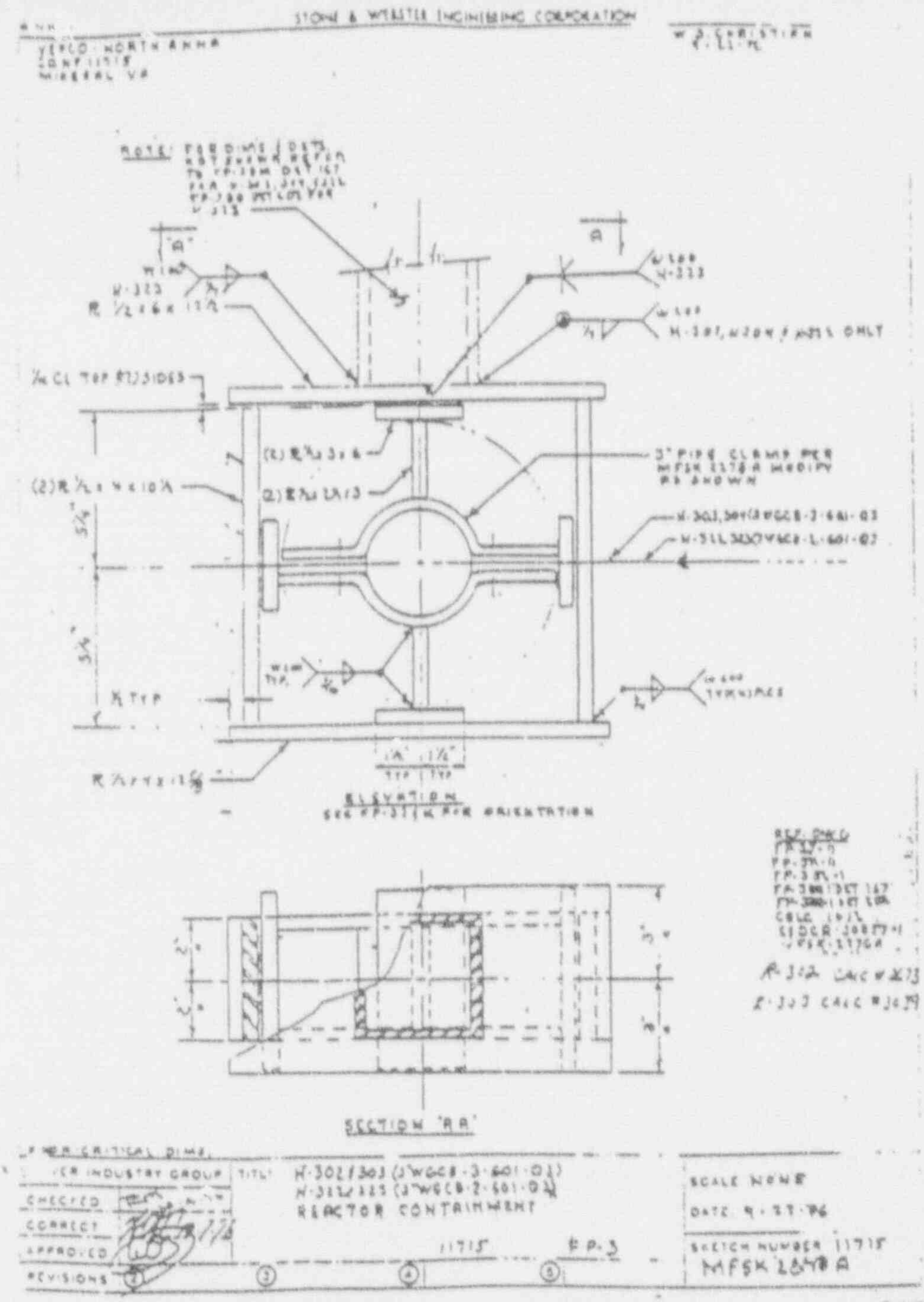
SAFETY RELATED
DC 90-13-1

	BY	DATE
PREPARED	[illegible]	4/17/92
CHECKED	RAB	4/17/92
APPROVED	[illegible]	4/17/92

TITLE
STEAM GENERATOR BLOWDOWN MODIFICATIONS
CUBICLE B
PIPE SUPPORT 1-WGCB-R-322

CLIENT VIRGINIA POWER
PROJECT NORTH ANNA 1
J.O. NO. 02072.0710
ISSUE BY SWEC
DATE
NUMBER SHEET 1 OF 3
9013-1-M-700.13-0

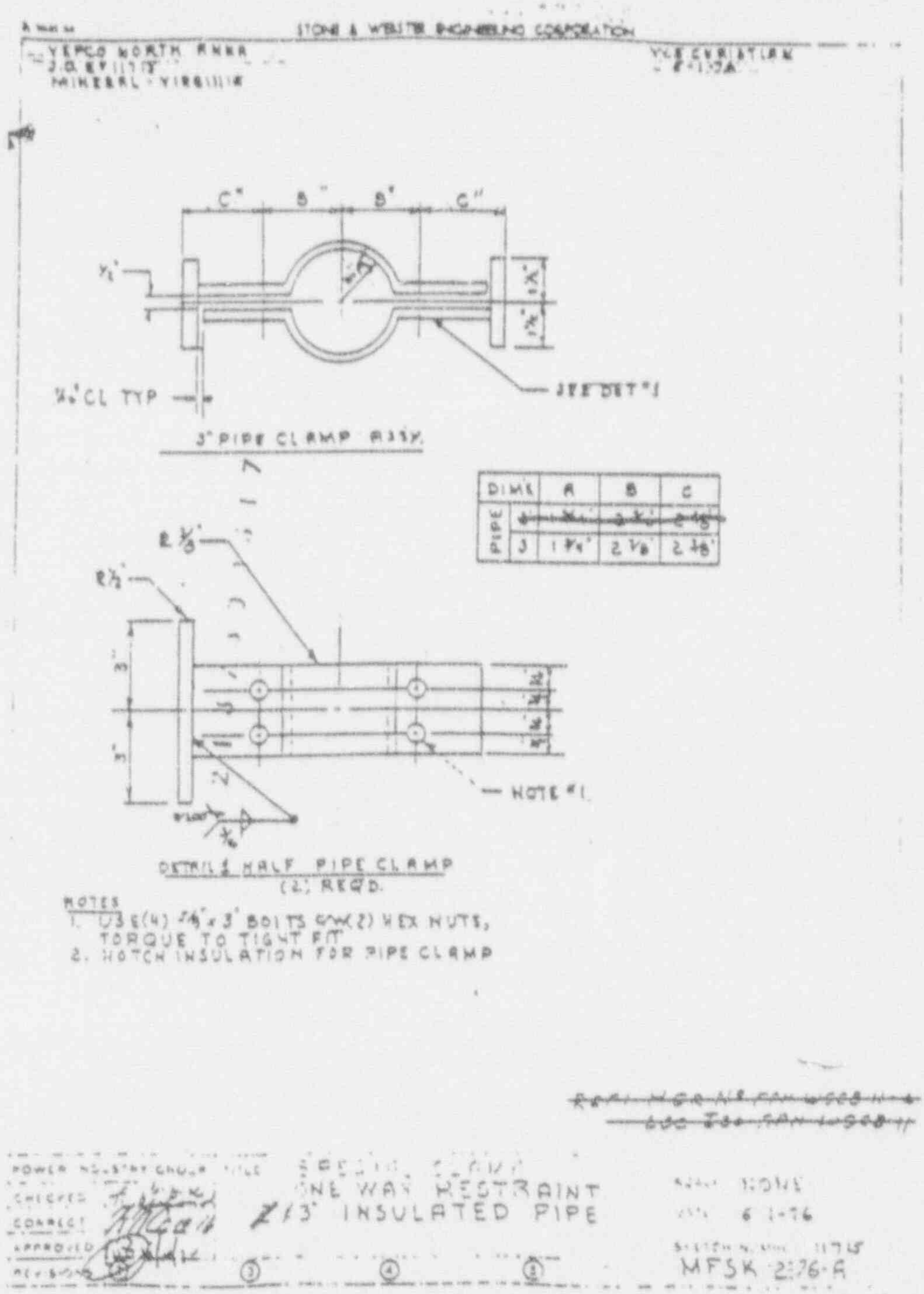
REVISIONS BY	①	②	③	④	⑤
DATE					



SAFETY RELATED
DC 90-13-1

ISSUED FOR CONSTRUCTION OF DC-90-13-1

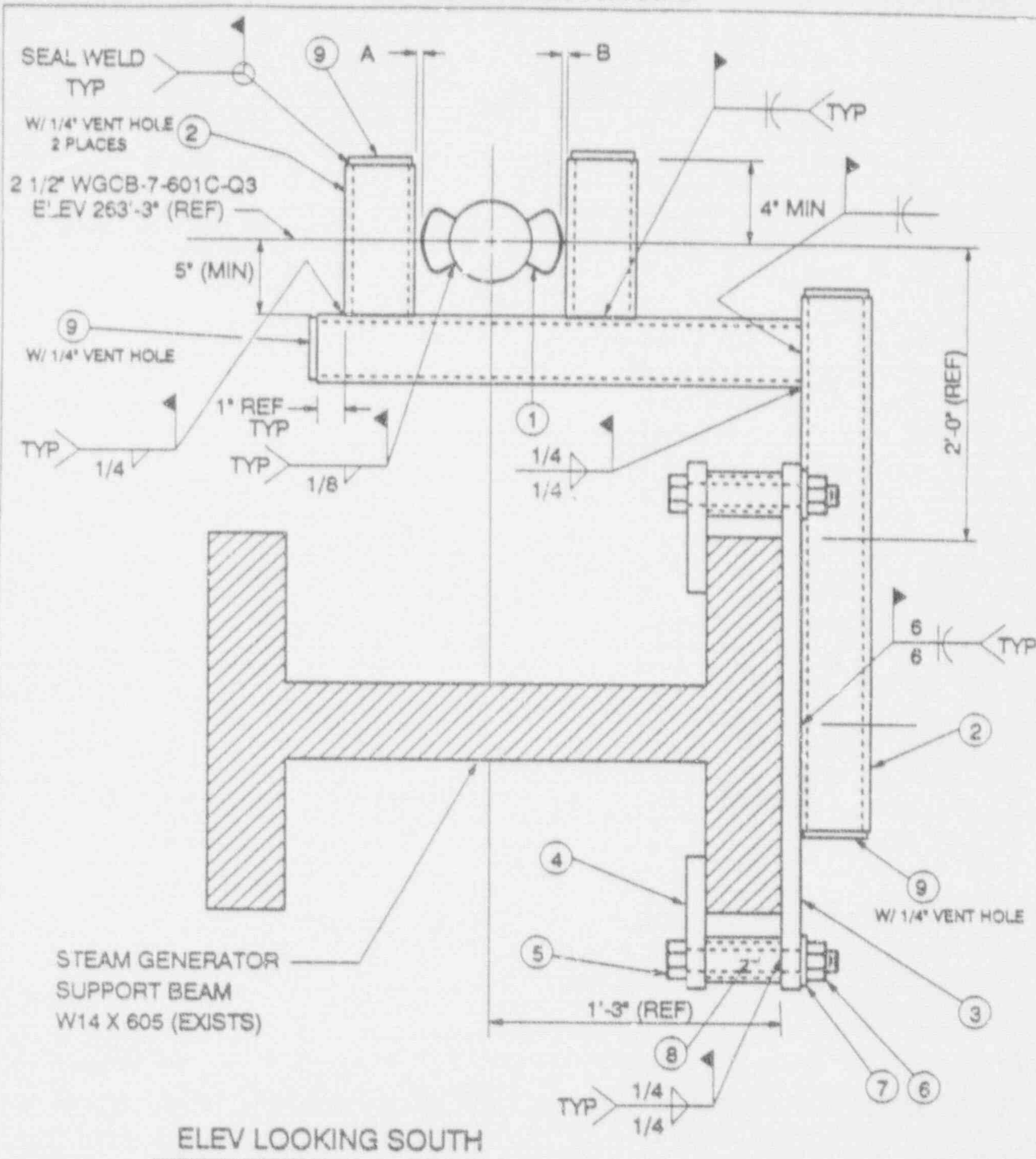
	BY	DATE	TITLE	CLIENT VIRGINIA POWER
			STEAM GENERATOR BLOWDOWN MODIFICATIONS	PROJECT NORTH ANNA 1
PREPARED	DCA	4/17/92	CUBICLE B	J.O. NO. 02072.0710
CHECKED	RAP	4/17/92		ISSUE BY SWEC
APPROVED	DBM	4/17/92	PIPE SUPPORT 1-WGCB-R-322	DATE
REVISIONS BY	⓪	①	② ③ ④	NUMBER SHEET 2 OF 3
DATE				N9013-1-M-700.13-0



ISSUED FOR CONSTRUCTION OF DC-90-13-1

SAFETY RELATED
DC 90-13-1

	BY	DATE	TITLE	
PREPARED	DDA	4/17/92	STEAM GENERATOR BLOWDOWN MODIFICATIONS CUBICLE B PIPE SUPPORT 1-WGCB-R-322	CLIENT VIRGINIA POWER
CHECKED	RAB	4/17/92		PROJECT NORTH ANNA 1
APPROVED	DBM	4/17/92		J.O. NO. 02072.0710
				ISSUE BY SWEC
				DATE
REVISIONS BY	① ② ③ ④ ⑤			NUMBER SHEET 3 OF 3
DATE				N9013-1-M-700.13-0

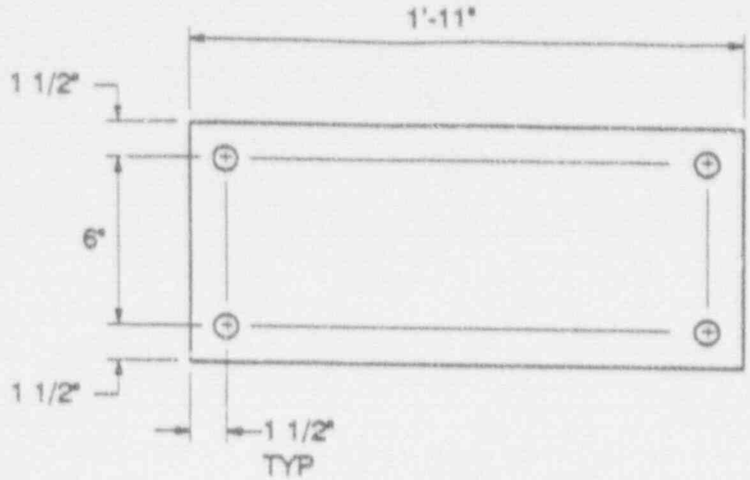


ELEV LOOKING SOUTH

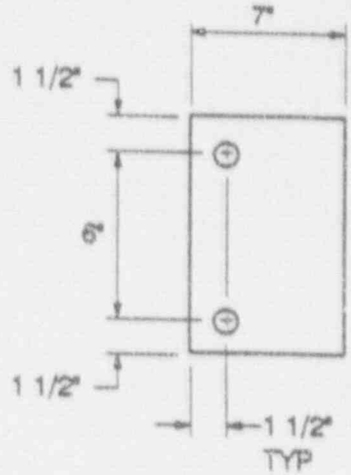
ISSUED FOR CONSTRUCTION OF DC-90-13-1

SAFETY RELATED
DC 90-13-1

	BY	DATE	TITLE	
PREPARED	REH.	9-17-92	STEAM GENERATOR BLOWDOWN MODIFICATIONS CUBICLE B PIPE SUPPORT 1-WGCB-PH-7.1	CLIENT VIRGINIA POWER
CHECKED		9/17/92		PROJECT NORTH ANNA 1
APPROVED		9/17/92		J.O. NO. 02072.0710
				ISSUE BY SWEC
				DATE
REVISIONS BY				NUMBER SHEET 1 OF 2
DATE				N9013-1-M-700.14-1



DETAIL A



DETAIL B

NOTES:

1. FOR SUPPORT LOCATION SEE DWG. N9013-1-M-601
2. FOR GENERAL NOTES REFER TO DWG N9013-1-M-700.01
3. TORQUE BOLTS (ITEM 5) TO 55 $^{+5}_{-0}$ FT-LB
4. CLEARANCES 1/16" $\leq$ (A+B) $\leq$ 5/32", FIELD SHIM TO SUIT.

BILL OF MATERIAL

ITEM	QTY	DESCRIPTION	MAT'L
1	2	2 1/2" PROTECTION SADDLE FOR 2" INS, GRINNELL FIG 162	———
2	4	TS 3 X 3 X 1/4" (LENGTH TO SUIT BY FIELD)	A500 GR. B
3	1	PLATE 3/4 X 9 X 23" (SEE DETAIL A)	A36
4	2	PLATE 3/4 X 9 X 7" (SEE DETAIL B)	A36
5	4	3/4"-10 UNC-2A X 7" LG	A325
6	4	3/4" HEX NUT (SEE NOTE 3)	A194 GR. 2H
7	4	3/4" FLAT STEEL WASHER	F436
8	4	TS 2 X 2 X 1/4" X 0'-4 1/8" LG	A500 GR. B
9	5	STEEL STRIP 1/8" X 2 3/4" X 2 3/4"	A606

ISSUED FOR CONSTRUCTION OF DC-90-13-1

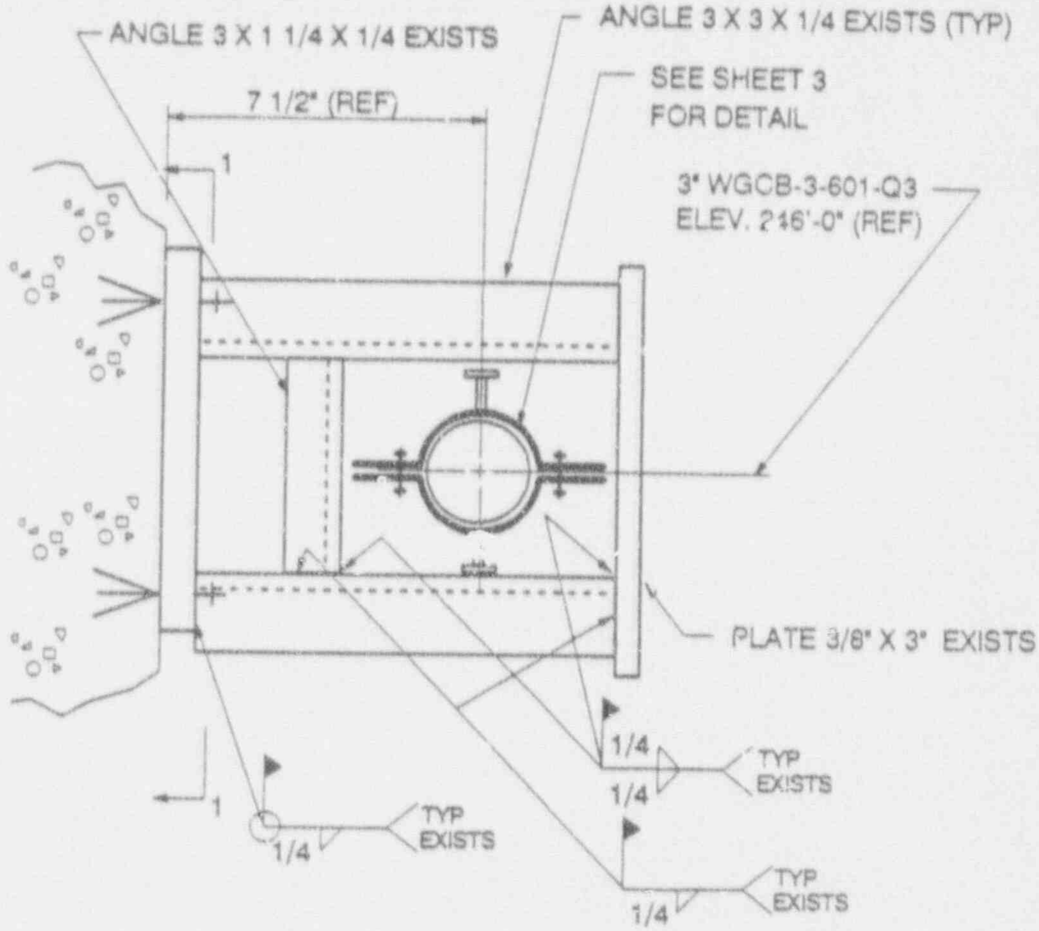
SAFETY RELATED
DC 90-13-1

	BY	DATE
PREPARED	ZEA	9-17-92
CHECKED	GAB	9/17/92
APPROVED	[illegible]	9/17/92

REVISIONS BY ① ② ③ ④ ⑤
DATE

TITLE
STEAM GENERATOR BLOWDOWN MODIFICATIONS
CUBICLE B
PIPE SUPPORT 1-WGCB-PH-7.1

CLIENT VIRGINIA POWER
PROJECT NORTH ANNA 1
J.O. NO. 02072.0710
①ISSUE BY SWEC
DATE
NUMBER SHEET 2 OF 2
N9013-1-M-700.14-1



ELEV. LOOKING SOUTH EAST

NOTES:

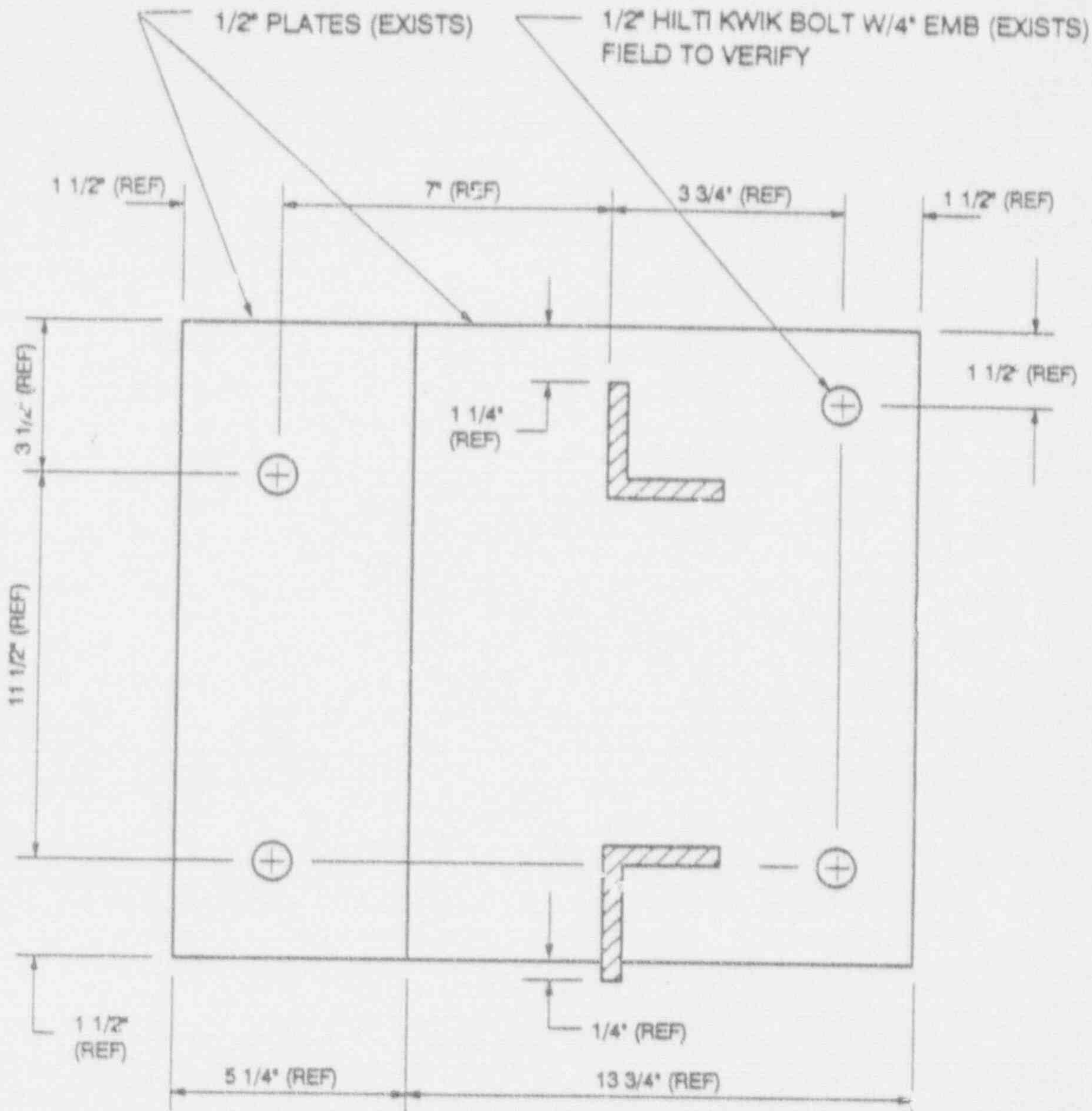
1. FOR SUPPORT LOCATION SEE DWG. N9013-1-M-603
2. FOR GENERAL NOTES REFER TO DWG. N9013-1-M-700.01
3. THIS SUPPORT, FPH-WGCB-11-3, WAS ORIGINALLY ON LINE NO. 2" WGCB-11-601-Q3. THE MARK NO. WILL REMAIN THE SAME FOR THE PURPOSES OF TRACEABILITY.

ISSUED FOR CONSTRUCTION OF DC-90-13-1

SAFETY RELATED
DC 90-13-1

	BY	DATE	TITLE	
			STEAM GENERATOR BLOWDOWN MODIFICATIONS CUBICLE C PIPE SUPPORT FPH-WGCB-11-3	CLIENT VIRGINIA POWER
PREPARED	DDR	4/17/92		PROJECT NORTH ANNA 1
CHECKED	RAB	4/17/92		J.O. NO. 02072.0710
APPROVED	DBM	4/17/92		ISSUE BY SWEC
REVISIONS BY	0			DATE
DATE				NUMBER SHEET 1 OF 3 N9013-1-M-700.15-0

STONE & WEBSTER ENGINEERING CORPORATION



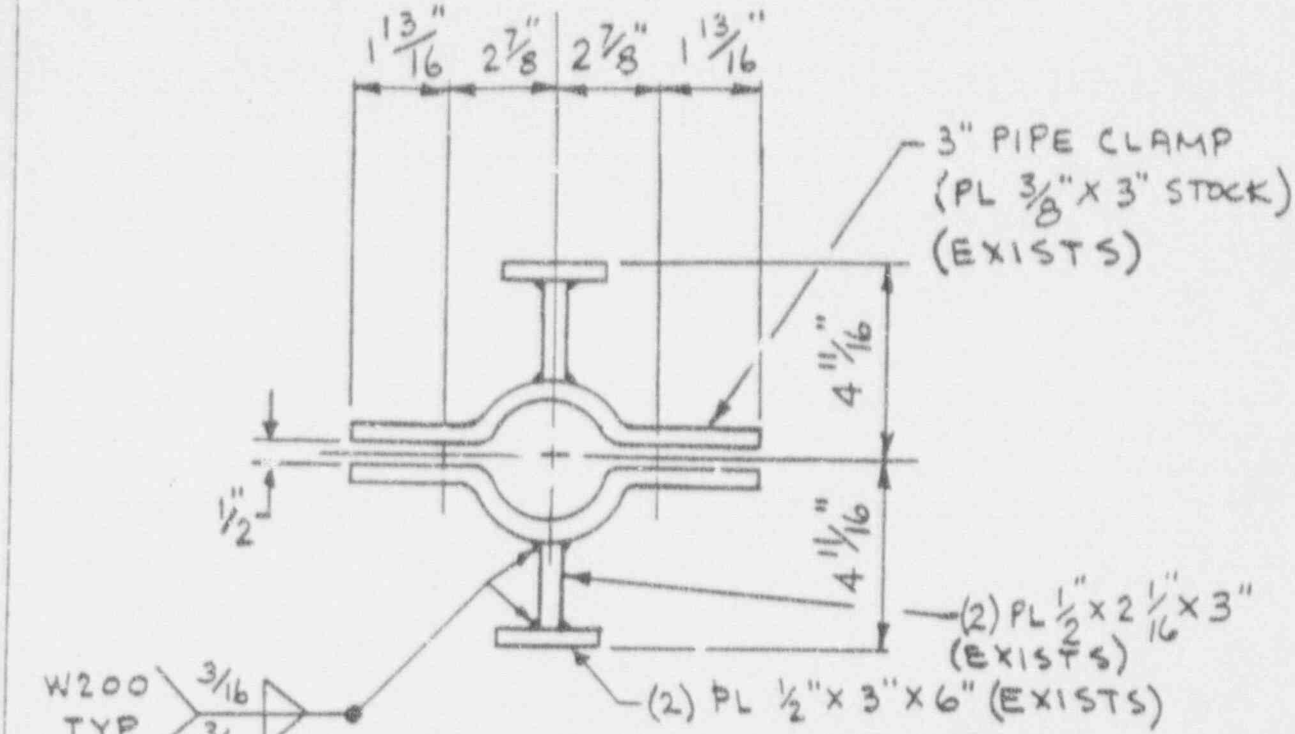
SECTION 1-1

ISSUED FOR CONSTRUCTION OF DC-90-13-1

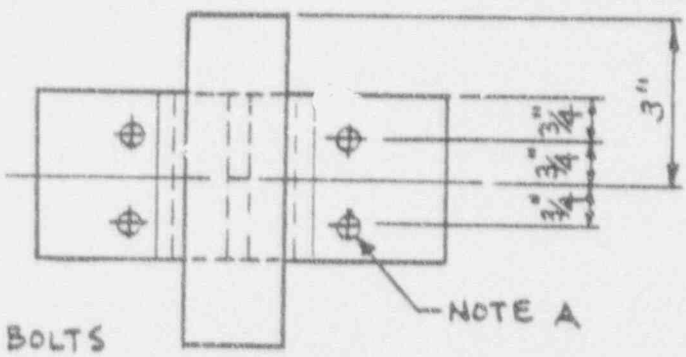
SAFETY RELATED
DC 90-13-1

	BY	DATE	TITLE	
PREPARED	DCK	4/7/92	STEAM GENERATOR BLOWDOWN MODIFICATIONS CUBICLE C PIPE SUPPORT FPH-WGCB-11-3	CLIENT VIRGINIA POWER
CHECKED	RAB	4/7/92		PROJECT NORTH ANNA 1
APPROVED	DSM	4/7/92		J.O. NO. 02072.0710
				ISSUE BY SWEC
				DATE

REVISIONS BY	0	0	0	0	0	NUMBER SHEET 2 OF 3
DATE						N9013-1-M-700.15-0



MODIFIED 3" PIPE CLAMP ASSY (EXISTS)



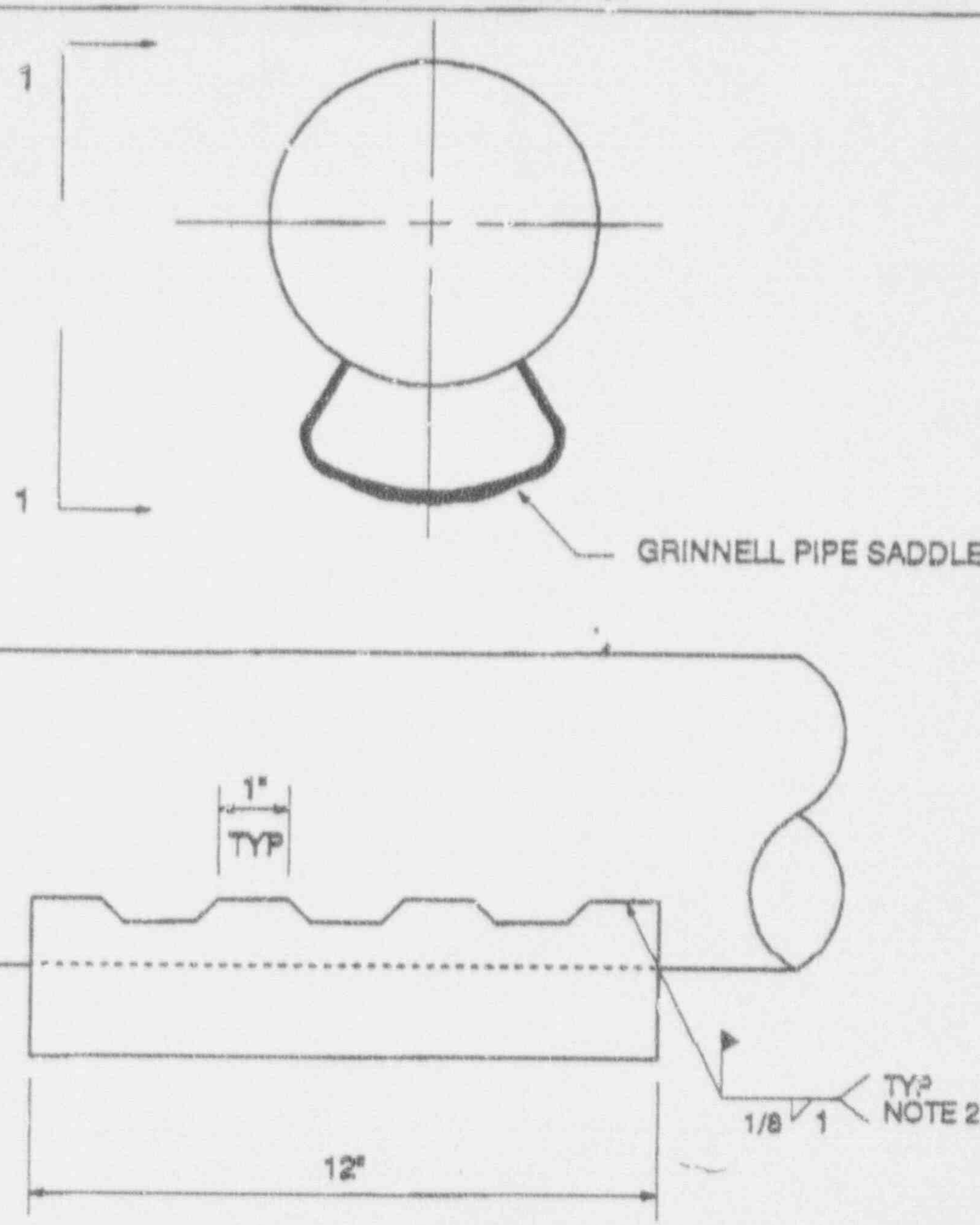
HALF PIPE CLAMP
2 REQ'D

NOTES:
A. USE (4) 5/8" X 3" BOLTS W/ 2 HEX NUTS, TORQUE TO TIGHT FIT
B. NOTCH INSULATION FOR PIPE CLAMP.

ISSUED FOR CONSTRUCTION OF DC-90-13-1

SAFETY RELATED
DC 90-13-1

	BY	DATE	TITLE	
PREPARED	KDG	4/17/92	STEAM GENERATOR BLOWDOWN MODIFICATIONS CUBICLE C	CLIENT VIRGINIA POWER
CHECKED	RAB	4/17/92		PROJECT NORTH ANNA 1
APPROVED	DSM	4/17/92	PIPE SUPPORT 1-FPH-WGCB-11-3	J.O. NO. 02072.0710
				ⓘISSUE BY SWEC
				DATE
REVISIONS BY	⓪	①	② ③ ④	NUMBER SHEET 3 OF 3 N9013-1-M-700.15-0
DATE				



SECT 1-1

NOTES:

1. FOR GENERAL NOTES REFER TO DWG. N9013-1-M-700.01
2. FIELD TO WELD AT LEAST TWO (2) LOCATIONS ON EACH SIDE

ISSUED FOR CONSTRUCTION OF DC-90-13-1

SAFETY RELATED
DC 90-13-1

	BY	DATE	TITLE	
PREPARED	[illegible]	5/7/92	STEAM GENERATOR BLOWDOWN MODIFICATIONS GRINNELL PIPE SADDLE DETAILS 1" & 2 1/2" FIG 182 AND 3" FIG 164	CLIENT VIRGINIA POWER
CHECKED	RAB	5/7/92		PROJECT NORTH ANNA 1
APPROVED	[illegible]	5/7/92		J.O. NO. 02072.0710
				ISSUE BY SWEC
				DATE
REVISIONS BY	0	1	2	NUMBER SHEET 1 OF 1
DATE				N9013-1-M-700.16-0

NDE REQUIREMENTS

Bechtel

P.O. Box 638
Mineral Virginia 23117

May 11, 1992
PR-BV2001-BP019

Virginia Power
Steam Generator Replacement Project
North Anna Power Station Unit No. 1
Post Office Box 635
Mineral, Virginia 23117

Attention: Mr. M. W. Gettler
Project Manager

Dear Mr. Gettler:

Subject: Steam Generator Replacement Project
North Anna Power Station Unit No. 1
Bechtel Job Number 21809
Welding and Nondestructive Examination Requirements

The purpose of this letter is to clarify the welding and nondestructive examination requirements applicable to the North Anna Unit No. 1 steam generator replacement.

In accordance with the contract document, the welding, nondestructive examination (NDE) and testing of replacement piping will be performed in accordance with Virginia Power Specifications NAS-1009, Revision 14.

Specification NAS-1009 invokes the ASME Boiler and Pressure Vessel Code, ~~Section III, Sub~~section XI, 1983 Edition with Summer 83 Addendum for the repair/replacement of piping and components designated Q1, Q2, or Q3.

As permitted by ASME Section XI, NAS-1009 invokes the original construction code (with certain exceptions noted below) for the repair/replacement of Q1, Q2, and Q3 piping and components as follows.

Paragraph 1.5 requires that fabrication, installation, and testing of Q1, Q2, and Q3 piping be in accordance with the original construction Code (ANSI B31.7 - 1969 Ed. with Addenda through 1970 and Code Cases 78, 81, 83R, and 115).

Paragraph 2.7 requires that welding and NDE be in accordance with ANSI B31.7 unless noted otherwise.

> Paragraph 4.1.1 specifies that all NDE for piping designated "Q" shall meet the acceptance standards of ANSI B31.7, paragraph 1-736 except that radiographic examination of Q1 and Q2 piping welds shall be performed in accordance with the Virginia Power NDE Manual (which is based on the ASME Section III Code) and the acceptance standard shall be in accordance with ASME Section III, Division 1, Subsection NB or NC, as applicable.

Therefore, for the steam generator replacement project, welding, preheat, post-weld-heat-treatment, specified NDE and weld acceptance of Q1, Q2, and Q3 piping will be in accordance with the ANSI B31.7 Code, 1969 Edition including 1970 Addendum, except that radiographic examination procedures and acceptance standards of Q1 and Q2 piping will be in accordance with the Virginia Power NDE Manual and the ASME Section III Code, 1986 Edition. Welding and NDE of the steam generator girth welds ,although not specifically addressed by Specification NAS-1009, will be in accordance with the ASME Section III Code, 1986 Edition. This requirement is based on the existing generators being designed and fabricated in accordance with the ASME Code and the new generator lower assemblies being fabricated in accordance with the ASME Code, 1986 Edition, Section III.

The steam generator repair/replacement will not be ASME Code stamped as stamping is not required by ASME Section XI.

Please provide your concurrence to the above by May 15, 1992. If any further information is required, please contact me at (703) 894-8055, or Erv Geiger at (703) 894-8037.

Sincerely,

BECHTEL POWER CORPORATION

FOR

Richard L. Miller
Project Manager

RLM/ELG:jll

cc: R. K. Bayer
L. F. Bennett
G. D. Brown
A. J. Bryan
W. J. Fidorowicz
E. L. Geiger
T. A. McKearney
M. A. Pitre
D. I. Price
B. P. Reilly
File 0260
File 0190
File 0609

Memorandum

[redacted]

VIRGINIA POWER
NORTH CAROLINA POWER

To: R.K. Bayer

Innsbrook Technical Center

From: E.W. Throckmorton

May 21, 1992

Steam Generator Replacement Project
Welding and Nondestructive Examination Requirements

I have reviewed the code requirements and code dates stated in Mr. Richard Miller's letter dated May 11, 1992, against NAS-1009 Rev. 14 and the NDE Manual and concur with one exception. Specification NAS-1009 invokes the ASME Boiler and Pressure Vessel Code, Section XI, 1983 Edition with Summer 83 Addendum for the repair/replacement of piping and components designated Q1, Q2, or Q3, not Section III, Subsection XI.

If any further information is required, please contact me at 2125.

EW Throckmorton

E.W. Throckmorton
ISI/NDE Supervisor

STEAM GENE ATOR REPLACEMENT
NORTH ANNA - UNIT 1
<u>FIELD INSPECTION OF WELDS</u>

	Reactor Coolant Nozzle Safe End to Pipe	Closure Girth Weld	Main Steam Nozzle to Pipe	Main Steam Pipe to Pipe	Feed Water Nozzle to Pipe	Feed Water Pipe to Pipe	All Other Shell Welds	Chemical Feed	Moisture Separator Welds	Blowdown 2 1/2" Pipe	Blowdown 1" Pipe
Pre-weld	LP ----	MT ----	MT ----	---- ----	LP ----	---- ----	MT*** ----	---- ----	---- ----	---- ----	---- ----
In-Process	RT	RT	----	----	----	----	----	----	----	----	----
Final	RT ----	RT ----	RT UT**	RT UT**	RT UT**	RT* MT* UT**	MT*** ----	MT or LP	VT ----	LP or MT	MT or LP
After PWHT	----	RT	RT	----	RT	----	----	----	----	----	----
After Hydro	VT ----	VT MT	VT MT	VT ----	VT MT	VT ----	---- MT	---- ----	---- ----	---- ----	---- ----
PSI Sect. XI Baseline	UT LP	---- ----	UT MT	UT MT	UT MT	UT MT	---- ----	---- ----	---- ----	---- ----	---- ----

* ...ition to B31.7, Engineering Technical Bulletin MA-1, dated April 12, 1990, invokes the following rejection criteria for chrome-moly welds not requiring volumetric e...minations:

1) Elongated indications greater than one inch in length.
2) Any group of indications in a line that have an aggregate length greater than three inches except when the distance between the successive indications exceeds four inches.

**These UT exams performed prior to PWHT to insure satisfactory exam at PSI Section XI Baseline.
***Flow Limiter Only

VENDOR TECHNICAL MANUAL CHANGE REQUEST

ATTACHMENT 2

(Page 1 of 1)

VENDOR TECHNICAL MANUAL CHANGE REQUEST - 723467(APR 91)

VIRGINIA POWER

Vendor Technical Manual Change Request

MTM 26

VTM Number	Revision Number	VTMCR Number
SEE BELOW		

Vendor Name	Vendor Document Number	Revision
WESTINGHOUSE		

Title	Vendor Code
MODEL 51 STEAM GENERATOR	

New Manual: ☐ New Equipment ☒ Existing Equipment | Manual Change: ☐ Existing Equipment ☐ Other

Other (Explain)

THE FOLLOWING MANUALS ARE AFFECTED AS DESCRIBED BELOW: WS-27, 28, 31, 32, 33, 34, 35, 36, 40, 41, 12, 43, 44, 45.

MANUAL WS-38 IS TO BE DELETED

Change Requested

THE EXISTING STEAM GENERATOR TECHNICAL MANUAL NEEDS TO BE REVISED TO STATE THAT IT NO LONGER APPLIES TO UNIT 1, YET IS STILL APPLICABLE FOR UNIT 2.

Requested By (Name)	Date	Department	Phone Number
MARK BARTH	9/18/92	BECHTEL	301-417-5950
Supervisor Approval (Signature)	**Date**	**Department**	**Phone Number**
M.D. [illegible]	9/18/92	BECHTEL	301-417-5141

Change Summary

Prepared By Vendor Information Coordinator (Signature)	Date
Procurement Engineering Supervisor Approval (Signature)	**Date**

Keys: VTM-Vendor Technical Manual; VTMCR-Vendor Technical Manual Change Request

Form No. 723467(Apr 91)

ATTACHMENT 2

(Page 1 of 1)

VENDOR TECHNICAL MANUAL CHANGE REQUEST - 723467(APR 91)

VIRGINIA POWER

Vendor Technical Manual Change Request

MTM 26

VTM Number	Revision Number	VTMCR Number

Vendor Name	Vendor Document Number	Revision
WESTINGHOUSE	P.O. # BNT 273343	

Title	Vendor Code
MODEL 51F STEAM GENERATOR	

New Manual	Manual Change
☒ New Equipment ☐ Existing Equipment	☐ Existing Equipment ☐ Other

Other (Explain)

Change Requested

THE NEW TECHNICAL MANUAL FOR THE UNIT 1 STEAM GENERATORS NEEDS TO BE ENTERED INTO THE SYSTEM.

Requested By (Name)	Date	Department	Phone Number
MARK BARTH	9/18/92	BECHTEL	301-417-5950
Supervisor Approval (Signature)	**Date**	**Department**	**Phone Number**
M.A. Smith	9/18/92	BECHTEL	301-417-5141

Change Summary

Prepared By Vendor Information Coordinator (Signature)	Date
Procurement Engineering Supervisor Approval (Signature)	**Date**

Keys: VTM-Vendor Technical Manual; VTMCR-Vendor Technical Manual Change Request

Form No. 723467(Apr 91)

ATTACHMENT 2

(Page 1 of 1)

VENDOR TECHNICAL MANUAL CHANGE REQUEST - 723467(APR 91)

VIRGINIA POWER

Vendor Technical Manual Change Request

MTM 28

VTM Number	Revision Number	VTMCR Number	
Vendor Name: TRANSCO	Vendor Document Number: P.O. # CNT 390946		Revision
Title: THERMAL INSULATION FOR SGs			Vendor Code
New Manual: ☒ New Equipment ☐ Existing Equipment	Manual Change: ☐ Existing Equipment ☐ Other		

Other (Explain)

Change Requested

THE NEW TECHNICAL MANUAL FOR THE UNIT 1 STEAM GENERATOR THERMAL INSULATION NEEDS TO BE ENTERED INTO THE SYSTEM.

Requested By (Name)	Date	Department	Phone Number
MARK BARTH	9/18/92	BECHTEL	301-417-5950
Supervisor Approval (Signature)	Date	Department	Phone Number
[signature]	9/18/92	BECHTEL	301-417-5141

Change Summary

Prepared By Vendor Information Coordinator (Signature)	Date
Procurement Engineering Supervisor Approval (Signature)	Date

Keys: VTM-Vendor Technical Manual; VTMCR-Vendor Technical Manual Change Request

Form No. 723467(Apr 91)

TECHNICAL REPORT NE-883 REV. 1, SAFETY ANALYSES AND EVALUATIONS SUPPORTING NORTH ANNA 1 OPERATION FOLLOWING STEAM GENERATOR REPLACEMENT

NAF TECHNICAL REPORT NE-883 REV. 1

SAFETY ANALYSES AND EVALUATIONS SUPPORTING NORTH ANNA 1 OPERATION FOLLOWING STEAM GENERATOR REPLACEMENT

NUCLEAR ANALYSIS AND FUEL
NUCLEAR ENGINEERING SERVICES
VIRGINIA POWER

June, 1992

Prepared by:	John R. Harrell J. R. Harrell	6-22-92 Date
Reviewed by:	Gary L. Darden G. L. Darden	6/22/92 Date
Approved by:	Joseph O. Erb for K. L. Basehore	6/22/92 Date

QA Category: Nuclear Safety Related

Key Words: NAPSI RC RCDES SG TH SGTUBE CFR5059

TABLE OF CONTENTS

1.0 INTRODUCTION

The purpose of this Technical Report is to consolidate and summarize the safety analyses and evaluations supporting North Anna 1 operation following steam generator replacement (SGR). The report evaluates the impact of steam generator replacement, and related changes to setpoints or operating parameters, on accident analyses which demonstrate the safety of the unit under normal and accident conditions. This report is a principal reference for the SGR project Design Change Package (DCP) Engineering Review and Design (ER&D). It will also support the North Anna 1 Cycle 10 (N1C10) Reload Safety Evaluation (RSE).

In this report, each UFSAR Chapter 15 accident analysis applicable to North Anna 1 operation with replacement steam generators, beginning with Cycle 10 (N1C10), is described and referenced. In most cases, the analysis supporting North Anna 1 operation following SGR is that which supported operation prior to the N1C9 restart. In some cases, however, the N1C9 restart reanalysis (extended steam generator tube plugging) is retained as the licensing basis analysis. The selection of the licensing basis analysis to be applicable to North Anna 1 was based on (a) the need for, and availability of, analysis margin relative to actual operating conditions, and (b) specific analysis considerations, such as improvements in modelling or the inclusion of evaluations for current engineering issues.

Several issues related to setpoints and procedures which fall under the purview of NAF were also considered in this report. These include (a) confirmation of Reactor Protection System (RPS), Emergency Safety

Features Actuation System (ESFAS), and Technical Specifications setpoints, (b) assessment of the increased flow rate associated with SGR, (c) evaluation of SGR impact on Emergency Operating Procedures (EOP), and (d) confirmation of boration and dilution nomographs for the operator curve book.

The following approach is taken for presenting the analyses and evaluations supporting North Anna Unit 1 operation following steam generator replacement. The report first presents a discussion of the Virginia Power reload design and safety analysis methodology (Section 1.1). The impact of the proposed change (steam generator replacement) on design inputs for which NAF is responsible is then presented in Section 2.1; this section attempts to define the full scope of impact of the proposed change. The applicable analysis for each UFSAR Chapter 15 transient analysis is then described and the analysis reference cited (Section 2.2). The revised LOCA mass and energy, and containment response analyses are also presented in this section. The Chapter 15 summary is followed by a discussion of the reload thermal/hydraulics analyses and evaluations which support the Chapter 15 accidents. The remaining sections address several additional safety considerations for which NAF is the responsible organization.

1.1 OVERVIEW OF VIRGINIA POWER RELOAD DESIGN AND SAFETY ANALYSIS METHODOLOGY

Topical Report VEP-FRD-42 Revision 1-A, "Reload Nuclear Design Methodology," (1) presents the methodology used by Virginia Power to perform nuclear reload design and safety analysis. The Topical Report provides (a) a description of how nuclear safety analysis parameters are determined and (b) a discussion of the use of calculated safety analysis parameters in performing "bounding analyses" to demonstrate the safe operation of the reload core. Since the approval of the Reload Nuclear Design Methodology Topical Report (1) by the NRC, the provisions of the report and the accompanying NRC Safety Evaluation Report permit Virginia Power to perform a technical safety evaluation to support operation with replacement steam generators by a key parameter evaluation, thereby substantially reducing the reanalysis effort. By the provisions of 10 CFR 50.59, the revised safety evaluations (or analyses) may be incorporated into the plant's licensing basis to support operation of North Anna Unit 1.

2.0 SAFETY ANALYSIS

2.1 SUMMARY OF SAFETY ANALYSIS INPUT CHANGES

2.1.1 Introduction

The purpose of this section is to discuss the impact of steam generator replacement on design inputs to analyses and evaluations which support North Anna 1 operation. The revised set of safety analyses developed in the sections to follow (which include the design input changes described in this section) will support operation of North Anna Unit 1 following steam generator replacement. Except where otherwise noted, the analyses presented herein are technically applicable to Unit 2 but are not necessarily the analysis of record for Unit 2. (This statement is presented for information only, since this report is not designed to be a vehicle for implementing these analyses for Unit 2.)

2.1.2 Impact of Steam Generator Performance on Safety Analysis

Reference (2) contains a scoping evaluation of the steam generator design changes and their expected impact on accident analyses. This evaluation examined the effect of the repaired steam generators on the non-LOCA safety analyses outlined in Chapter 15 of the UFSAR, including the effect on the mass and energy releases inside containment following a steamline rupture. This evaluation concluded that the acceptance criteria are expected to be met and the conclusions of the UFSAR to remain valid.

Nuclear Analysis and Fuel has performed a study to evaluate the performance of the replacement steam generators under postulated accident conditions (3). For purposes of accident analysis, the following characteristics of the replacement steam generators (Westinghouse Model 51F) were considered sufficiently different from the existing steam generator design (Model 51) to warrant explicit modelling in the Reference (3) analysis:

1. Increased number of tubes (3592 vs. 3388) (3),(2)

2. Increased secondary side total water mass (103,796 lbm vs. 102,000) (3),(4),(5)

3. Increased heat transfer area (54,500 ft^2 vs 51,000 ft^2) (3),(2)

4. Reduced tube thermal conductivity (3),(6)

5. Reduced tube material density and specific heat (3),(6)

6. Increased primary side volume (1134 ft^3 vs 1090 ft^3) (3),(2)

The Reference (3) calculation further quantified the impact of the repaired steam generators on existing safety analyses. Two transients (Loss of Load and Excessive Load Increase) were analyzed to compare the heat transfer characteristics of the existing and replacement steam generators under transient conditions. These accidents were selected because (a) they are representative of limiting events which cause primary side heatup or cooldown; and (b) they permit a straight-forward evaluation of steam generator effects in isolation, because these accident analyses are not complicated by actions of mitigating systems, such as safety injection and auxiliary feedwater.

With equal RCS flow rates, the two steam generator designs exhibited virtually equivalent behavior during the RCS heatup (Loss of Load) and RCS cooldown (excessive load increase) transients (3). The analysis examined seven transient parameters (nuclear power, core heat flux, pressurizer pressure, hot leg temperature, hot-to-cold leg temperature difference, steam pressure, and pressurizer liquid volume), and demonstrated that there is an insignificant difference in transient behavior between the Model 51 and 51F steam generator designs. The Reference (3) evaluation supports the general conclusion that the Model 51F steam generator may be considered a "replacement" component for the Model 51 steam generator from a safety analysis perspective.

At the end of this section, Figures 2.1-1 through 2.1-8 present graphs of nuclear power, pressurizer pressure, steam pressure, and hot leg temperature during loss of load and excessive load increase transients. These graphs permit comparison of the behavior of important transient parameters for both steam generator designs. Note that the initial steam pressure in the excessive load increase transient is different for the two steam generator designs. As a result, the two transient curves do not coincide. However, the pressure differential between these two curves is essentially constant, indicating that the transient behavior of the two steam generator designs is comparable.

2.1.3 Steam Generator Design and Operating Conditions

The following thermal/hydraulic design operating conditions are applicable to North Anna Unit 1 operation following steam generator replacement:

1. pressurizer pressure = 2250 psia

2. RCS average temperature between 580.8°F and 586.8°F (7)

3. minimum measured RCS total flow rate = 284,000 gpm

4. core thermal power = 2893 MWth

These values are consistent with those specified in the North Anna 1 Technical Specifications as applicable following steam generator replacement.

In most cases, the safety analyses applicable to North Anna 1 operation following steam generator replacement revert to those which were applicable prior to the extended Steam Generator Tube Plugging (SGTP) analysis effort. The DNBR penalties applied against retained DNBR margin to account for the DNBR effect of reduced flow due to extended SGTP (8),(9),(10),(11) may be removed following steam generator replacement, since the Technical Specification minimum measured RCS total flow rate of 284,000 gpm is expected to be met with margin. Note that the minimum measured flow rate assumed in the North Anna Statistical DNBR Evaluation Methodology implementation analysis (12),(13),(14),(15),(16) was 289,200 gpm; a previous Technical Specification revision from this RCS total flow rate to 284,000 gpm was justified by a simple DNBR margin assessment in Reference (17).

The operating conditions listed above have been appropriately utilized in both statistical DNBR and non-statistical and non-DNB accident analyses and evaluations. The RCS will be operated at an average temperature in accordance with the current Tavg program:

$$(T_{avg} = 547.0°F + P^*(T_{nom}-547))$$

where P is the fraction of rated thermal power, and Tnom is a temperature between 580.8°F and 586.8°F, as required by the safety analyses supporting current North Anna 1 and 2 operation (7). Reference (7) provides sensitivity information which demonstrates that operation at an average RCS temperature as low as 580.8°F does not significantly impact the results of either the LOCA or non-LOCA analyses. Unless otherwise noted, accident analyses were performed to support an average RCS temperature of 586.8°F.

According to the Westinghouse steam generator bid package (18), the expected total RCS flow rate with the replacement steam generators is 306,600 gpm, or 102,200 gpm/loop (0% SGTP). A steam generator design total RCS flow rate of 278,400 gpm, or 92,800 gpm/loop, was assumed. With a 2% flow calorimetric measurement error, this translates into a 284,000 gpm Technical Specification minimum measured RCS flow rate. The steam pressure, which varies as a function of power level and operating Tavg, is expected to be approximately 900 psia (Tavg = 586.8°F) or 850 psia (Tavg = 580.8°F) at 100% power (0% SGTP) (18). Each replacement steam generator will result in a small RCS volume increase of 44 ft^3 (from 1090 ft^3/SG to 1134 ft^3/SG) (2). Further information on steam generator thermal/hydraulic design data may be obtained from the Westinghouse Steam Generator Thermal/Hydraulic Design Data report (4).

The replacement steam generator design operating conditions had the potential to cause the results of containment analyses to become more limiting due to increased blowdown energy during a large break LOCA or main steamline break accident. Referrence (113) documents the Westinghouse calculation of LOCA mass and energy releases, which serves as a design input reference for the Stone and Webster containment analysis (115). Section 2.2.29 of this report summarizes the results of these analyses and demonstrates acceptable containment analysis results to support North Anna 1 operation following steam generator replacement.

2.1.4 Technical Specifications

North Anna 1 Technical Specification Figure 2.1-1, Table 2.2-1, and Table 3.2-1 were previously modified to reflect reactor protection system setpoints (OTΔT, low flow, and high flux reactor trips) and reactor coolant system flow rates applicable under conditions of extended SGTP (8),(9). Turbine runback and rod withdrawal block setpoints were reset by subtracting .03 from the proposed K1 and K4 values. These specifications will revert to the values based on the above design conditions (e.g., 284,000 gpm) following steam generator replacement.

North Anna 1 reactor protection system setpoints and minimum measured RCS flow rate should be verified in plant instrumentation and procedures to be consistent with the values in the Technical Specifications. The accident analyses and evaluations in this report assume all RPS/ESFAS setpoints other than those indicated above remain at the values indicated in Technical Specifications.

FIGURE 2.1-1

Loss of Load Transient
Normalized Nuclear Power

Solid Line = Model 51F   Dashed Line = Model 51

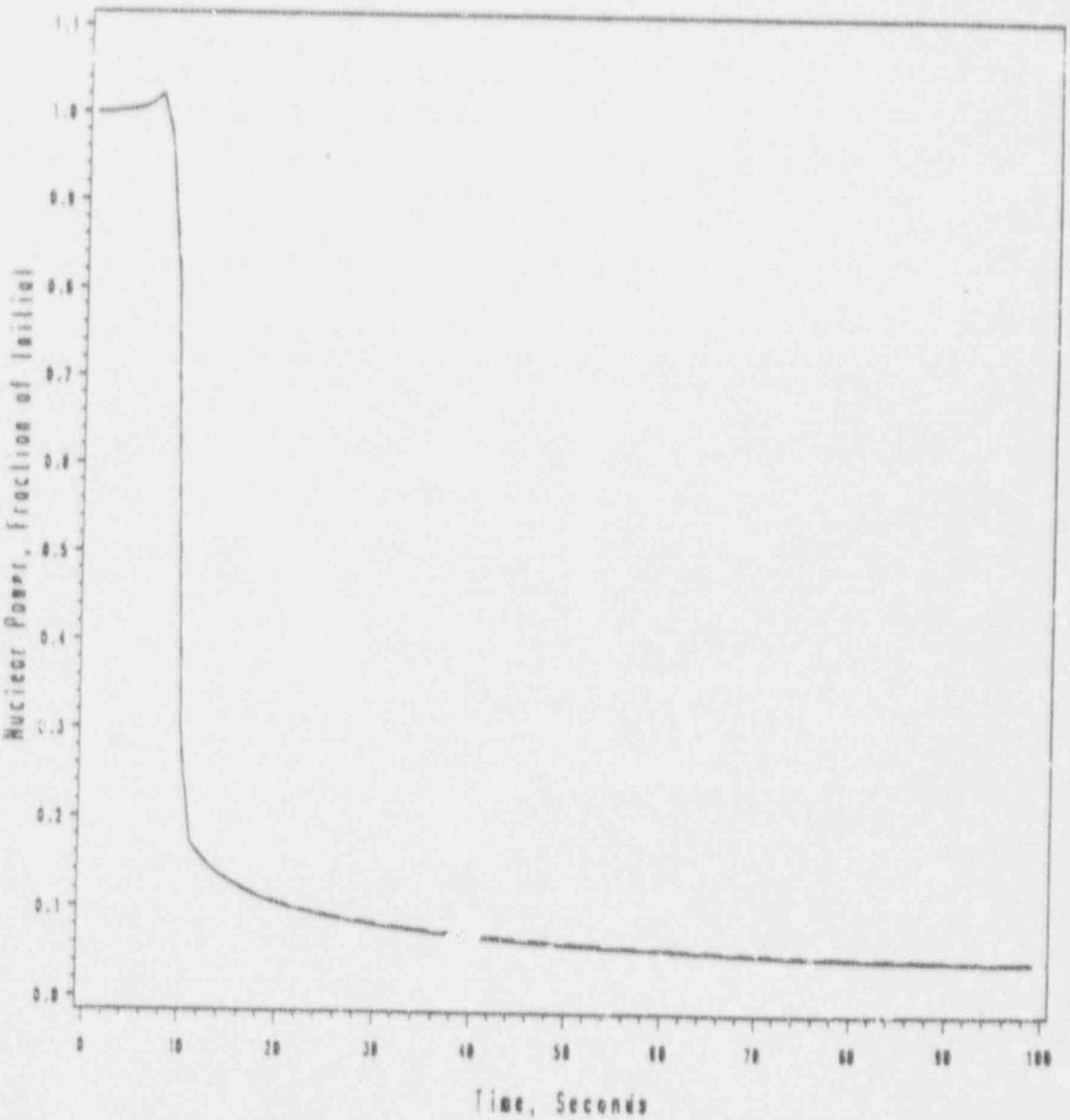


FIGURE 2.1-2

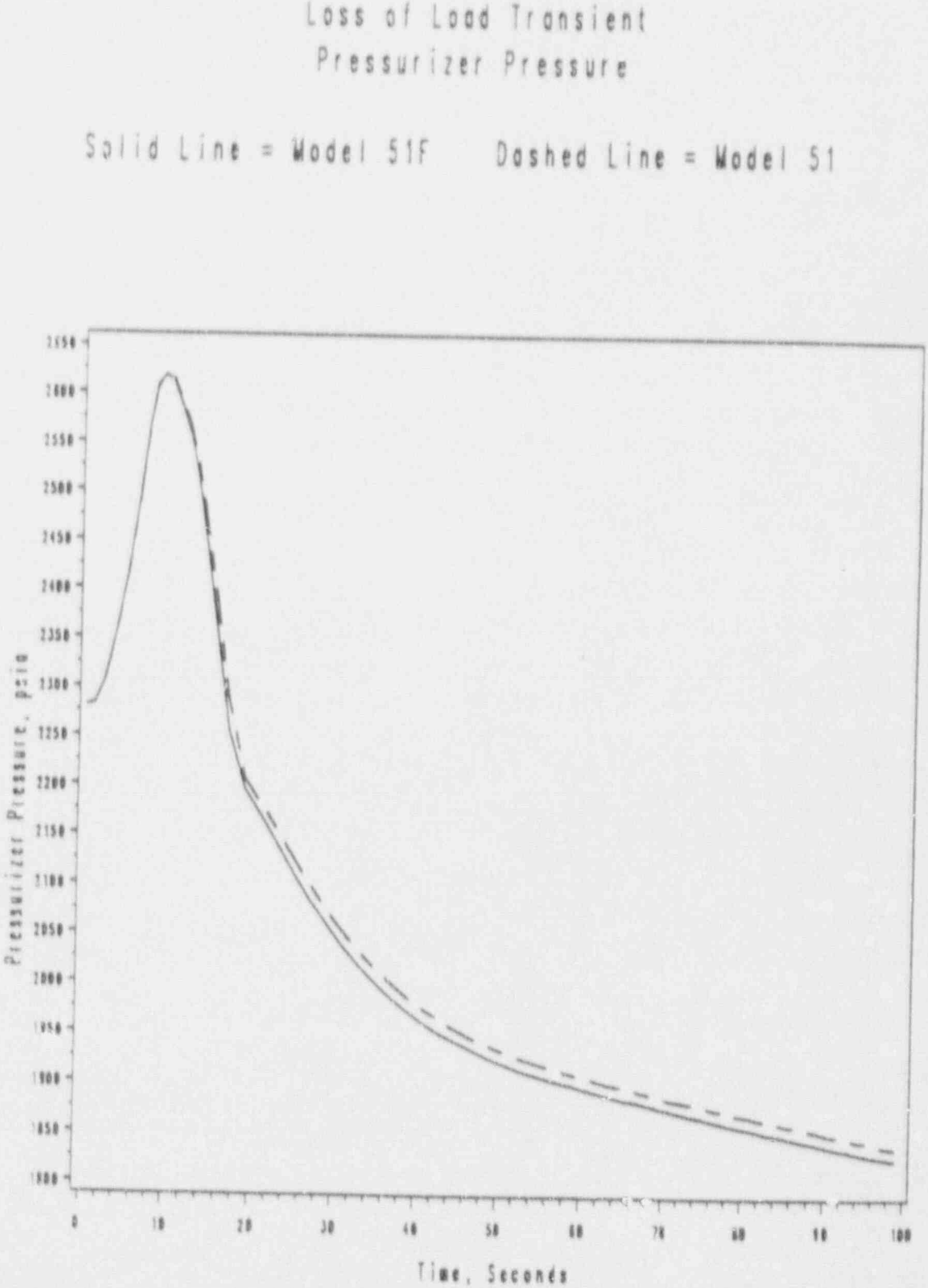

Loss of Load Transient
Pressurizer Pressure
Solid Line = Model 51F
Dashed Line = Model 51
Pressurizer Pressure, psia
Time, Seconds

FIGURE 2.1-3

Excessive Load Increase Transient
Steam Pressure

Solid Line = Model 51F   Dashed Line = Model 51

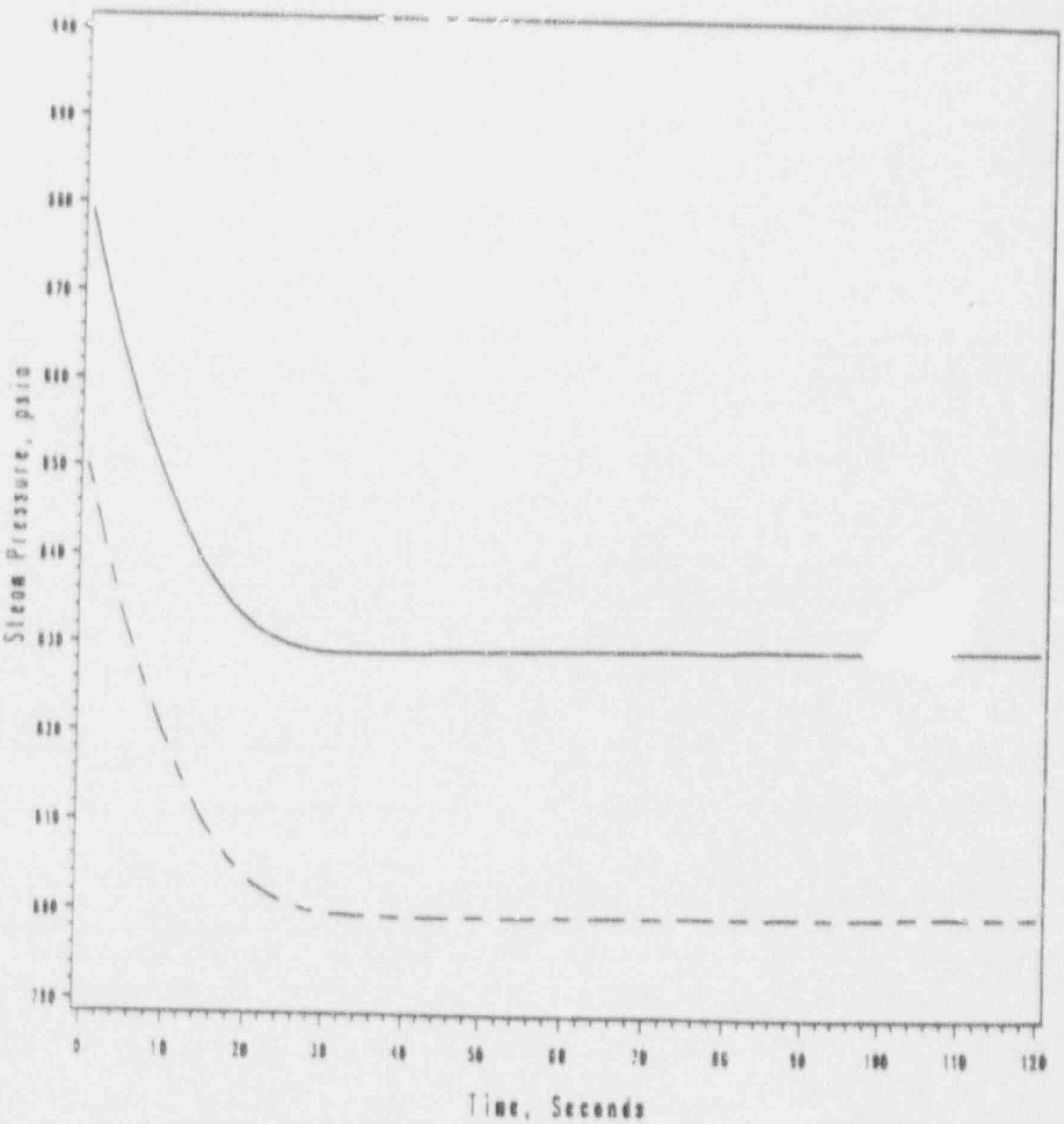


FIGURE 2.1-4

Excessive Load Increase Transient
Hot Leg Temperature

Solid Line = Model 51F   Dashed Line = Model 51

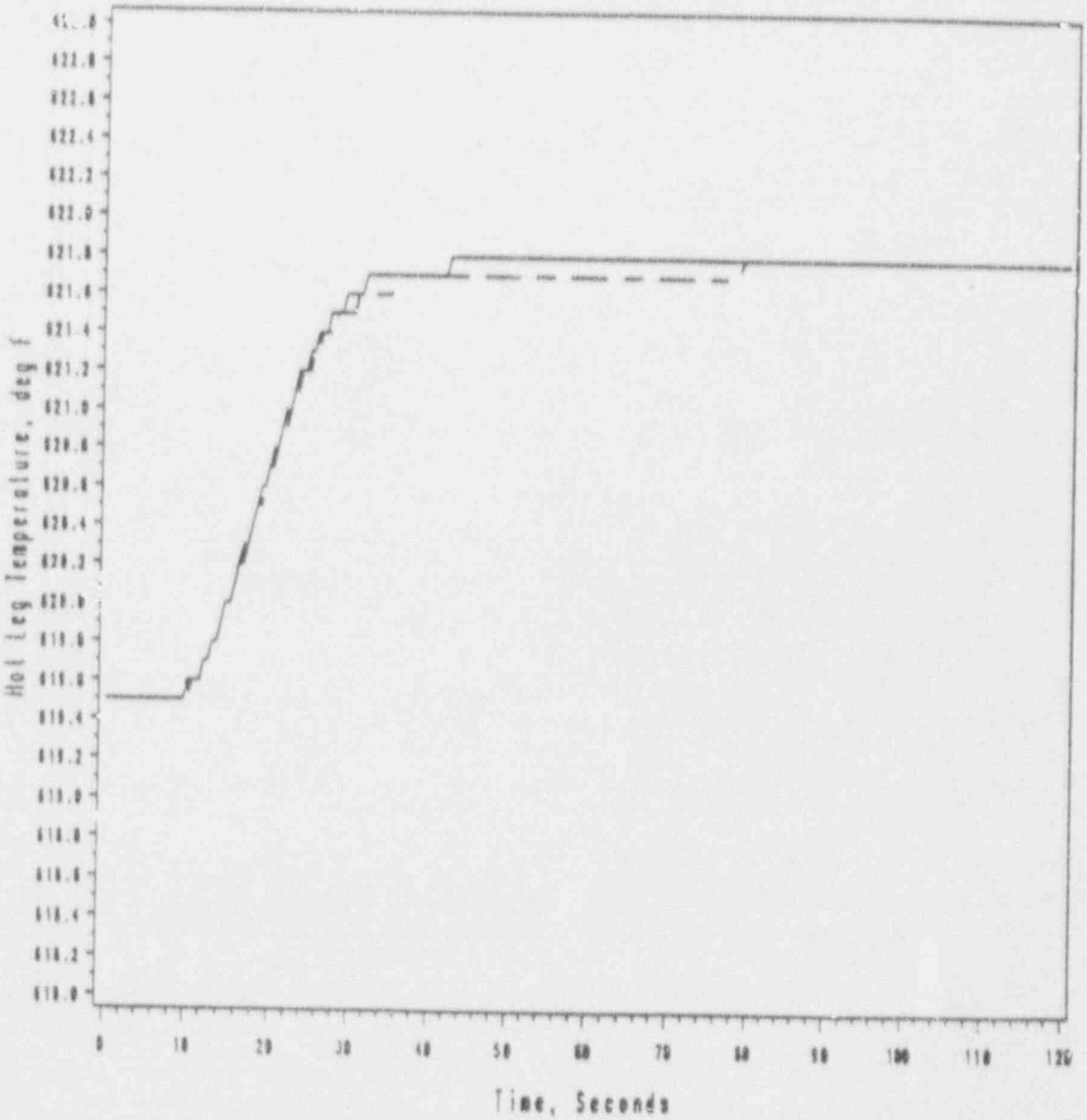


FIGURE 2.1-5

Excessive Load Increase Transient
Normalized Nuclear Power

Solid Line = Model 51F Dashed Line = Model 51

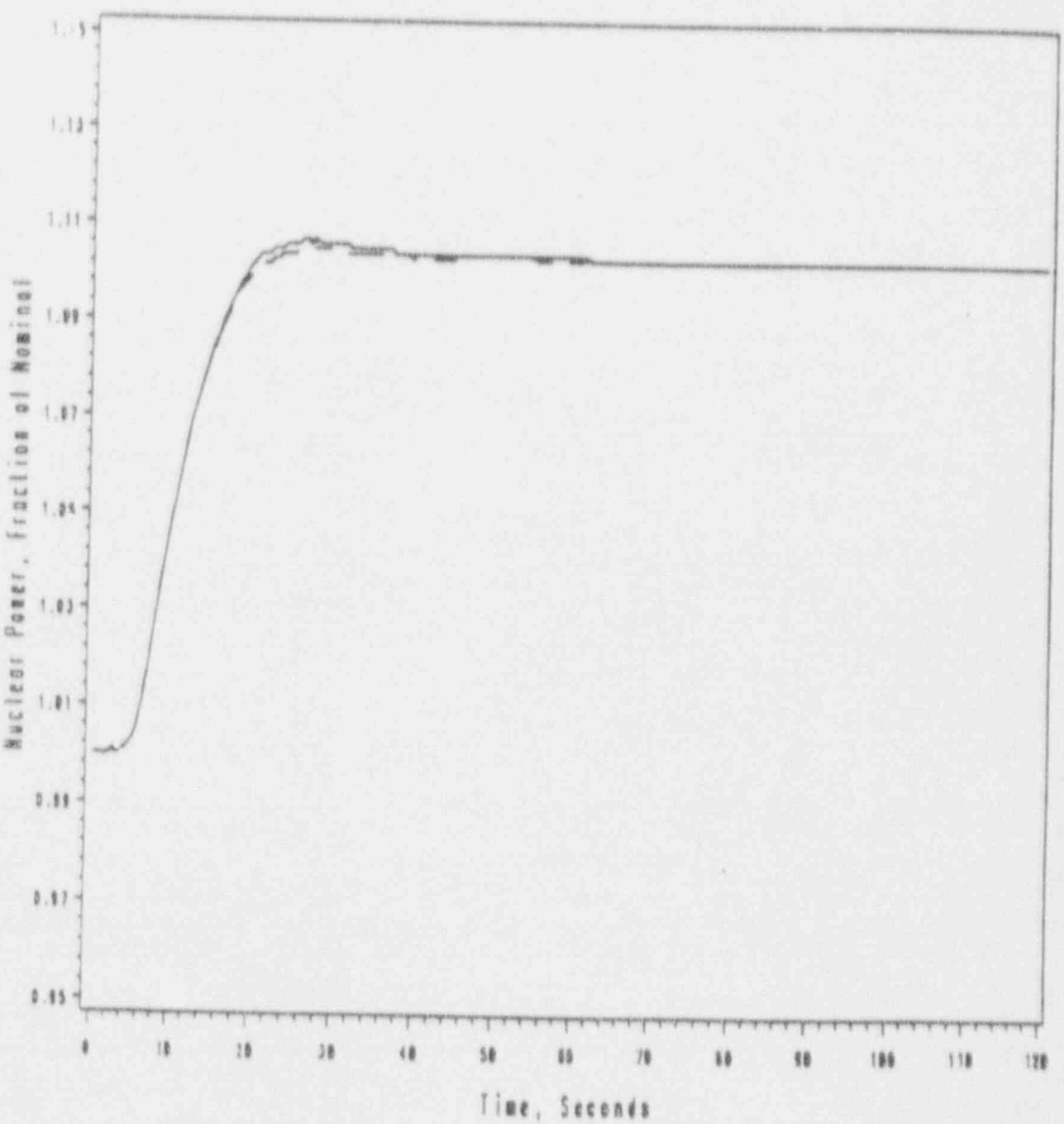
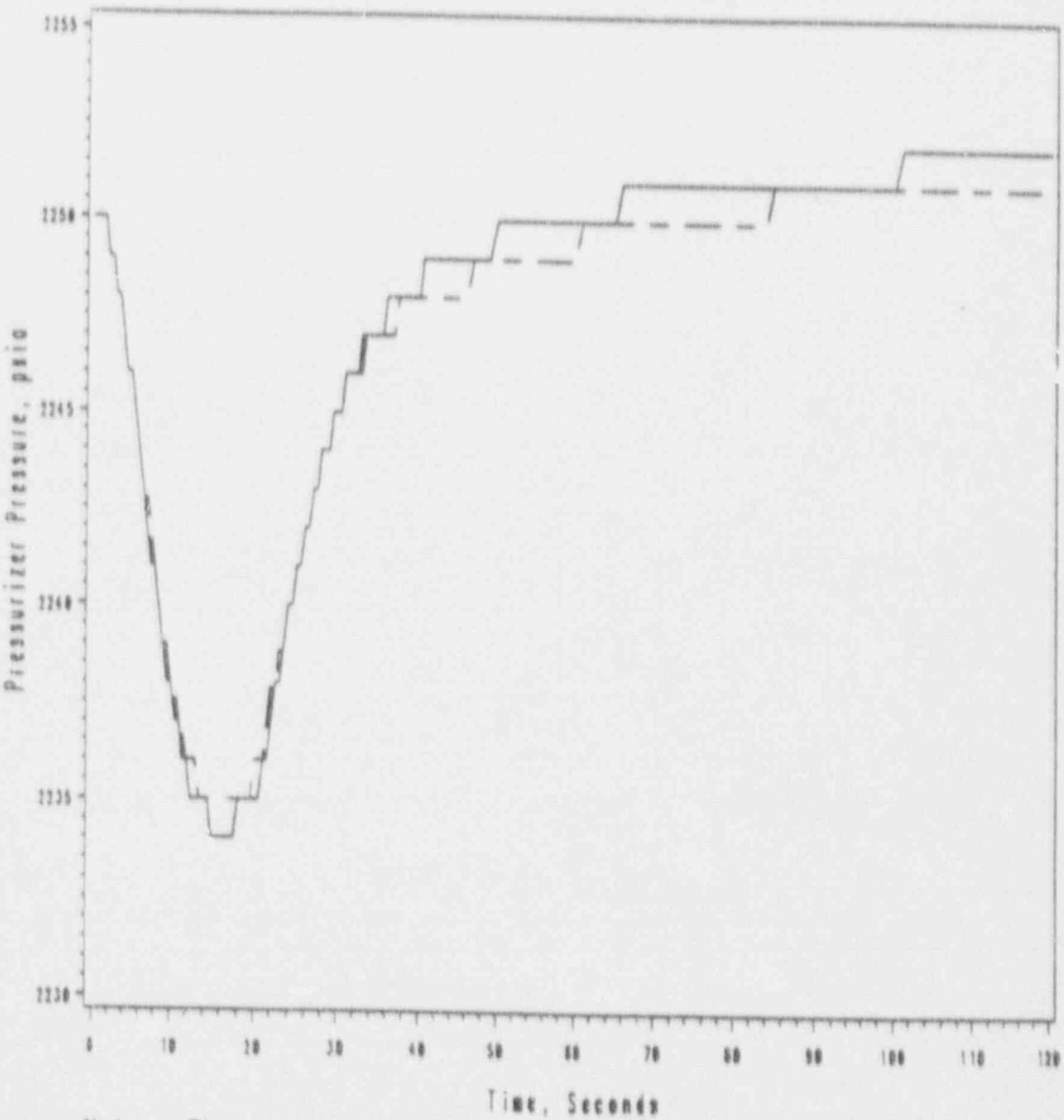


FIGURE 2.1-6

Excessive Load Increase Transient
Pressurizer Pressure

Solid Line = Model 51F Dashed Line = Model 51



Note: The stepwise nature of the graph is a consequence of automatic timestep selection for RETRAN and the extremely narrow pressure range presented on the y-axis.

FIGURE 2.1-7

Loss of Load Transient
Steam Pressure

Solid Line = Model 51F   Dashed Line = Model 51

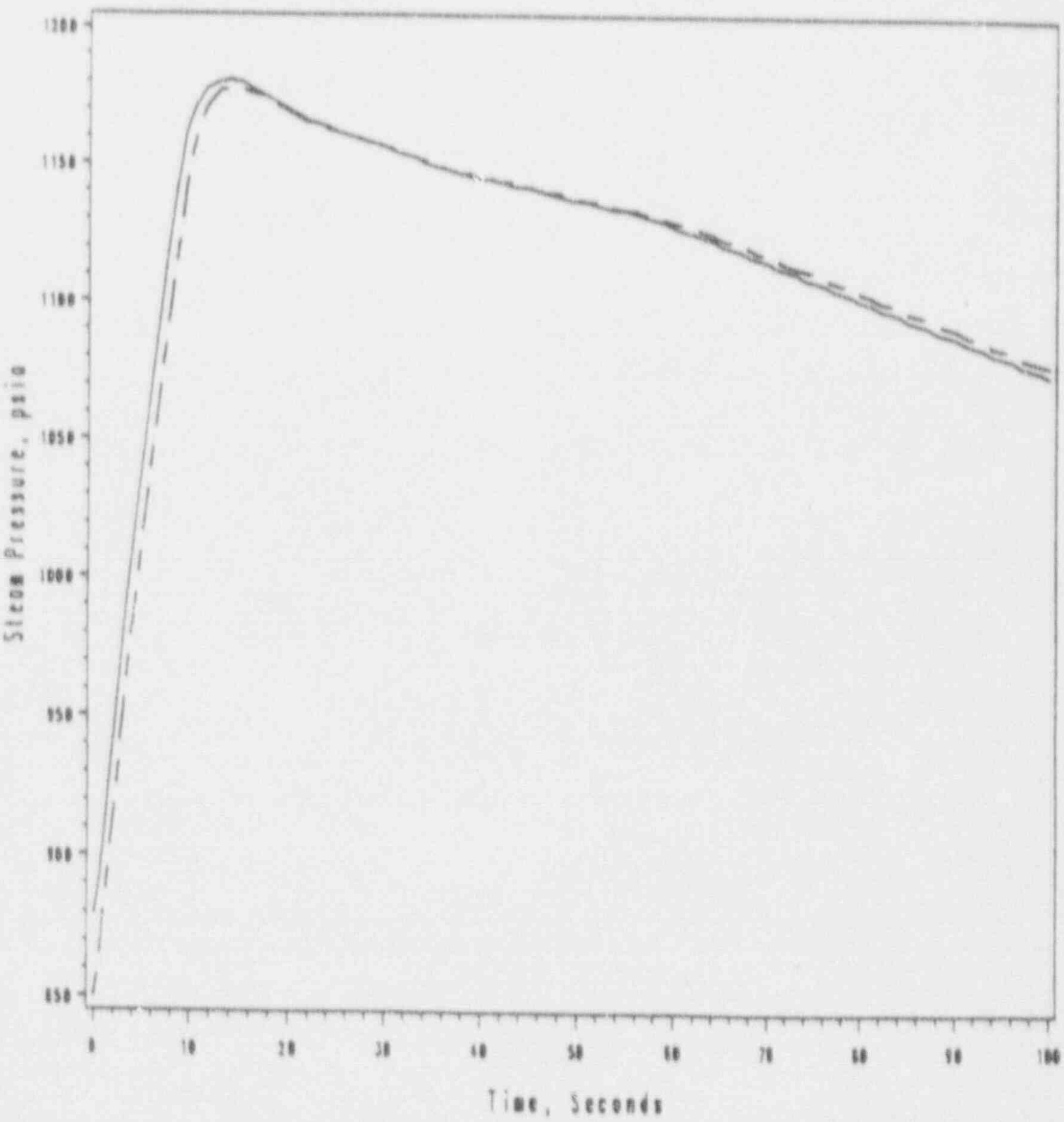
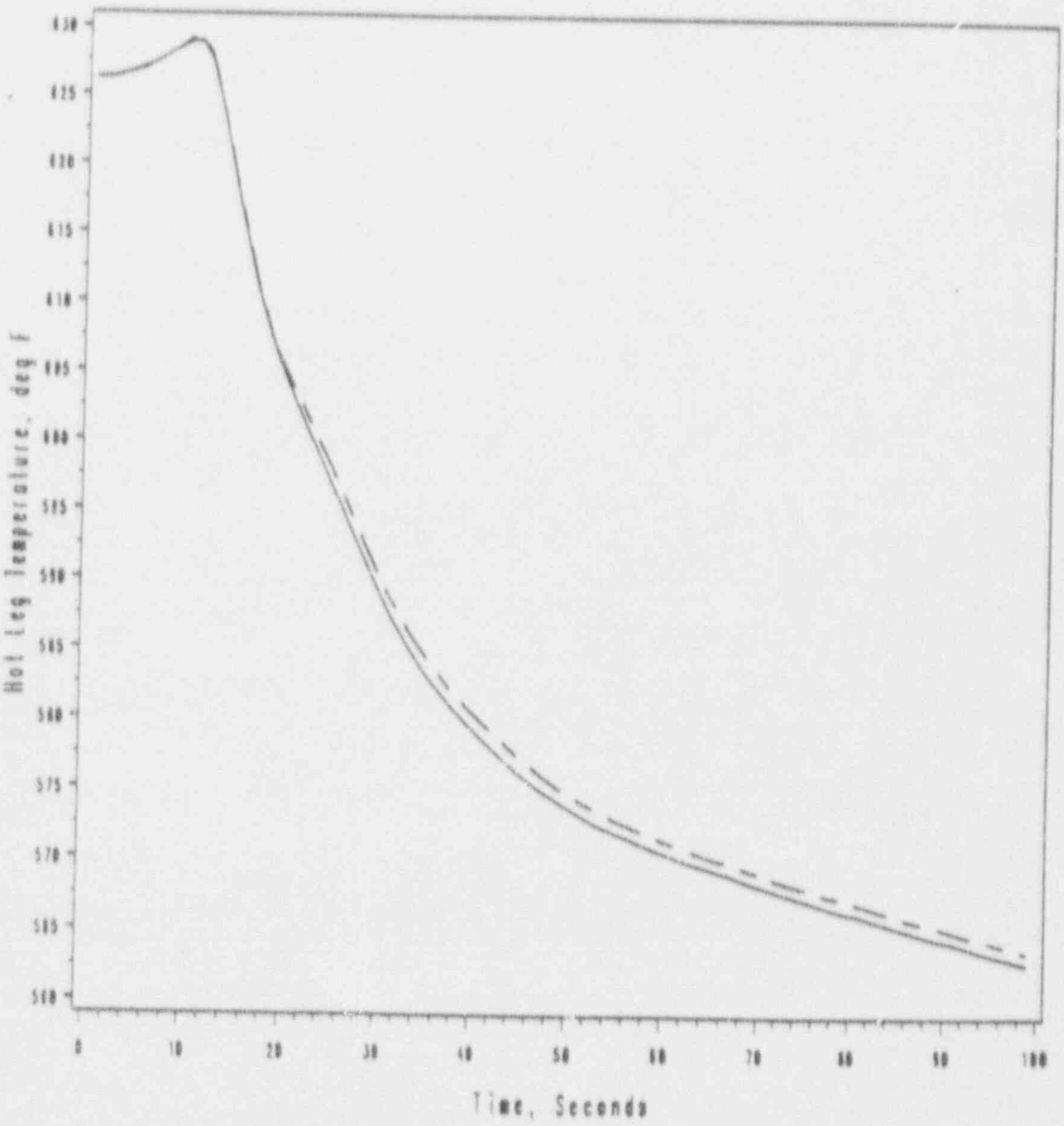


FIGURE 2.1-8

Loss of Load Transient
Hot Leg Temperature

Solid Line = Model 51F   Dashed Line = Model 51



2.2 SUMMARY OF ACCIDENT EVALUATIONS

This section discusses the UFSAR Chapter 15 accidents for which evaluations have been performed to support North Anna 1 operation following steam generator replacement. For each accident, a description of the transient, how the analysis is impacted by steam generator replacement, and the analysis applicable to North Anna 1 operation following steam generator replacement is provided.

A number of transients were explicitly reanalyzed to support N1C9 operation with extended steam generator tube plugging (SGTP) (10), (21),(22). These reanalyzed transients include the Chemical and Volume Control System (CVCS) Malfunction at Power (Boron Dilution at Power) (23), the Loss of External Load (24), the Loss of Normal Feedwater (including station blackout cases) (25), the Rod Bank Withdrawal at Power (26), the Complete Loss of Flow (27), the Locked Rotor and Sheared Shaft Events (28), and the Small (29) and Large Break (29),(30),(31),(32),(33),(34) Loss of Coolant Accidents. Where it is appropriate, extended SGTP transient analyses have been retained to support operation with the replacement steam generators.

2.2.1 Accidental Depressurization of of the RCS (UFSAR Section 15.2.12).

The most severe core conditions resulting from an accidental depressurization of the reactor coolant system are associated with the inadvertent opening of a pressurizer safety valve. Initially, the event results in a rapidly decreasing reactor coolant system pressure until this pressure reaches a value corresponding to the hot-leg saturation pressure. At that time, the pressure decrease is slowed considerably. The pressure continues to decrease, however, throughout the transient. The effect of the reduced pressure would be to decrease the neutron flux via the moderator density feedback, but the reactor control system (if in the automatic mode) functions to maintain the power essentially constant throughout the initial stage of the transient. The average coolant temperature decreases slowly, but the pressurizer level increases until reactor trip.

The existing licensing basis analysis of the Accidental Depressurization of the RCS was performed by Westinghouse as part of the core uprating project (35),(36),(37). This analysis is performed to ensure that the minimum DNBR remains above the DNBR design limit throughout the transient. Because the Reference (3) analysis demonstrated that the replacement steam generators' transient behavior is essentially identical to that of the current steam generator design in the analysis conditions (0% SGTP), and because this accident is insignificantly impacted by SGTP, it may be concluded that steam generator replacement does not adversely impact the results of the current main steam system depressurization licensing basis analysis. Analysis criteria will continue to be met following steam generator replacement provided the Technical Specification minimum measured RCS flow rate requirement (total RCS flow rate ≥284,000 gpm) is met. Because this requirement is expected to be met with margin, the current licensing basis analysis remains valid for North Anna 1 following steam generator replacement.

2.2.2 Accidental Depressurization of the Main Steam System (UFSAR Section 15.2.13).

The most severe core conditions resulting from an accidental depressurization of the main steam system are associated with an inadvertent opening of a single steam dump, relief, or safety valve. The analyses performed in consideration of the more severe main steam pipe rupture are presented in UFSAR Section 15.4.2.1 (Main Steamline Break Accident).

The steam release as a consequence of this accident results in an initial increase in steam flow that decreases as the steam pressure falls. The energy removal from the reactor coolant system causes a reduction in coolant temperature and pressure. In the presence of a negative moderator temperature coefficient (MTC), the cooldown results in a reduction of core shutdown margin.

The current analysis is performed to demonstrate that, in the presence of a stuck rod cluster control assembly and a single failure in the engineered safety features, there will be no departure from nucleate boiling (DNB) in the core for a steam release equivalent to the opening (with failure to close) of the largest of any single steam dump, relief, or safety valve.

The current licensing basis analysis (UFSAR Section 15.2.13) of the Accidental Depressurization of the Main Steam System was performed as part of the boron injection tank (BIT) boric acid concentration reduction project with the RETRAN (38),(39),(40),(41) system transient analysis code and the COBRA (42) detailed core thermal/hydraulics analysis code. The RETRAN models assume that the steam generator is in an unplugged (0% SGTP) condition, since this accident is insignificantly impacted by SGTP. The boric acid reduction transient analysis remains the licensing basis analysis supporting North Anna 1 following steam generator replacement.

Because the Reference (3) analysis demonstrated that the replacement steam generators' transient behavior is essentially identical to that of the current steam generator design in the analysis conditions (0% SGTP), and because this accident is insignificantly impacted by SGTP, it may be concluded that steam generator replacement does not adversely impact the results of the current main steam system depressurization licensing basis analysis.

2.2.3 Complete Loss of Flow Reanalysis (UFSAR Section 15.3.4).

As a part of the extended steam generator tube plugging analysis effort, the Loss of Flow Accident (LOFA) was reanalyzed (27) for SGTP levels up to an ag of 40%. The analysis was submitted to the NRC in Reference and was evaluated for a reduction in minimum measured RCS flow rate to 268,500 gpm in Reference (9). A statistical treatment of key analysis uncertainties was utilized in accordance with North Anna implementation (13) of the methodology described in References (14),(16), and (15).

Prior to the extended SGTP analysis, the most recent reanalysis of the transient was for the implementation of the Vantage 5H fuel product at North Anna (43). A summary of the Reference (27) analysis, which will remain the licensing basis analysis for North Anna 1 operation following steam generator replacement, is presented in the following sections. The conclusion of applicability of the Reference (27) analysis is based on the independence of steam generator steady state or transient performance and the LOFA analysis results. Furthermore, the Reference (3) analysis supports the conclusion that the Model 51F steam generators may be considered a replacement component for the purposes of safety analysis.

2.2.3.1 Accident Description

A complete loss of forced reactor coolant flow may result from a simultaneous loss of electrical power to all three reactor coolant pumps. If the reactor is at power at the time of the accident, the immediate effect of a LOFA is a rapid increase in the coolant temperature. This increase could result in DNB with subsequent fuel damage if the reactor is not promptly tripped. Reactor protection is provided by either the pump underfrequency or undervoltage trip function.

2.2.3.2 Method of Analysis

The LOFA was reanalyzed (27) with the RETRAN (38),(39),(40),(41) system transient analysis code and the RETRAN extended SGTP single loop model (44). All assumptions were consistent with or conservative with respect to (a) those in the previously approved analysis (43),(15),(16) and (b) the proposed plant configuration. The RETRAN code provided transient pressures, core inlet temperatures, heat fluxes and core flows which were used as input to a detailed thermal/hydraulic statepoint analysis performed with the COBRA (42) code. The WRB-1 correlation (45),(46) was used. Both the underfrequency and the undervoltage trip events were analyzed. A +6 pcm/°F Moderator Temperature Coefficient was conservatively assumed although the actual full power MTC will be zero or negative. Delay times of 0.6 second and 1.2 seconds were assumed for the underfrequency and undervoltage trips respectively.

2.2.3.3 Results and Conclusions

The underfrequency trip LOFA was found to be the most limiting event. Transient DNBR's remained above the statistical DNBR design limit throughout the transient. The LOFA analysis results will remain bounding following steam generator replacement provided the Technical

Specification minimum measured RCS flow rate is met. Because this requirement is expected to be met with margin, the extended SGTP LOFA analysis remains valid for North Anna 1 following steam generator replacement.

2.2.4 CVCS Malfunction (Boron Dilution) (UFSAR Section 15.2.4).

Prior to the implementation of the primary grade water flow path lockout provisions in Technical Specification 3.1.1.3.2, the boron dilution event analysis was evaluated on a reload basis to ensure that adequate time exists for operator response to correct an inadvertent boron dilution. With the approval (47) of the 2300 ppm boron Technical Specification change submittal (48), the primary grade water flow path lockout provisions permitted the elimination of the operator response time criterion from the Boron Dilution event analysis basis. To quote the Reference (47) Safety Evaluation Report (SER), "For the boron dilution transient, the presently specified NA-1& TS 3.1.1.3.2 precludes the possibility of an unplanned boron dilution by specifying that the primary grade water flow control valve be locked closed during operations in Modes 3, 4, 5, and 6 except during planned boron dilution or makeup activities. The current Standard Review Plan (SRP) acceptance criteria are met through this presently specified NA-1&2 TS 3.1.1.3.2."

Because no operator response time criteria are presently applicable to North Anna in Modes 3, 4, 5, and 6, steam generator replacement does not affect the results of the current licensing basis analysis as reflected in UFSAR Section 15.2.4. If the operator response time criteria were still part of the North Anna analysis basis, steam generator replacement would provide an analysis benefit since each replacement steam generator has approximately 44 ft^3 more primary side volume in the unplugged condition than the current steam generator design.

The consequences of boron dilution event at power (in terms of approach to the design DNBR limit) are identical to those of a slow rod withdrawal, for which an appropriate analysis exists as described in Section 2.2.17. Because the acceptance criterion for the Boron Dilution at Power (for loss of shutdown margin concerns) is the demonstration of adequate time for operator response to mitigate such a dilution, the Boron Dilution at Power event was reanalyzed for extended SGTP concerns (23). However, because the Reference (23) analysis assumed a very conservative level of STGP (40%), the licensing basis analysis to be applicable following Unit 1 steam generator replacement will be the Reference (49) analysis which assumes 25% steam generator tube plugging. This analysis was the licensing basis analysis prior to the extended SGTP analysis effort.

To complete the description of the boron dilution analysis basis, a description of the boron dilution at power event and its analysis is presented in the following sections.

2.2.4.1 Accident Description

Reactivity can be added to the core by feeding primary-grade water into the reactor coolant system via the reactor makeup portion of the chemical and volume control system. Boron dilution is a manual operation under strict administrative controls with procedures calling for a limit on the rate and duration of dilution. A boric acid blend system is provided to permit the operator to match the boron concentration of reactor coolant makeup water during normal charging to that in the reactor coolant system. The chemical and volume control system is designed to limit, even under postulated failure modes, the potential rate of dilution to a value which, after indication through alarms and instrumentation, provides the operator sufficient time to correct the situation in a safe and orderly manner.

The opening of the primary water makeup control valve creates a dilution flow path to the reactor coolant system. Inadvertent dilution from this source can be readily terminated by closing the control valve. For makeup water to be added to the reactor coolant system at pressure, at least one charging pump must be running in addition to a primary grade water transfer pump. The maximum dilution rate at pressure is, therefore, limited to the charging pump flow rate of 165 gpm.

Two separate operations are required to dilute: (a) The operator must switch from the automatic makeup mode to the dilute mode. (b) The start button must be depressed. Omitting either step prevents dilution.

Information on the status of the reactor coolant makeup is continuously available to the operator. Lights are provided on the control board to indicate the operating condition of the pumps in the chemical and volume volume control system. Alarms are actuated to warn the operator if boric acid or demineralized water flow rates deviate from preset values as a result of system malfunction.

2.2.4.2 Method of Analysis

The boron dilution at power event was analyzed (49),(50),(51) with the RETRAN (38),(39),(40),(41) system transient analysis code and a RETRAN single loop model. The Reference (49) analysis is valid for SGTP levels up to 25% in either the Model 51 or Model 51F replacement steam generators. This analysis will remain the licensing basis analysis supporting North Anna 1 operation following steam generator replacement.

The analysis developed an RSAC (52) limit curve of boron concentration versus isothermal temperature coefficient (ITC, defined as the sum of the moderator and doppler temperature coefficients) which is the locus of all critical boron concentrations (as a function of ITC) which render a constant 15 minutes to loss of shutdown margin following positive operator indication of a dilution in progress. The analysis assumed a hot full power differential boron worth of -11.5 pcm/ppm which is verified on a reload basis to be bounding. A dilution flow rate of 165 gpm was assumed. Credit is taken for the proper functioning of the overtemperature ΔT trip function. The 15 minute operator response time criterion criterion appears in the Standard Review Plan (NUREG-0800) (53) and in the NRC safety evaluation report (51) in response to the NRC submittal (50).

2.2.4.3 Results and Conclusions

Because of the procedures involved in the dilution process, and the administrative blocking of the primary grade water flow path, an inadvertent dilution in Modes 3, 4, 5, and 6 is not considered credible.

Nevertheless, numerous alarms and indications are available to alert the operator to any unintentional dilution of boron in the reactor coolant. For credible boron dilution events, such as the case of boron dilution at power, the maximum reactivity addition rate is slow enough to allow the operator sufficient time to determine the cause of the addition and to take corrective action before shutdown margin is lost. The licensing basis analysis (49) will remain bounding for North Anna 1 operation following steam generator replacement.

2.2.5 Excessive Load Increase (UFSAR Section 15.2.11).

An excessive load increase is defined as a rapid increase in the steam flow that causes a power mismatch between the reactor core power and the steam generator load demand. The reactor control system is designed to accommodate a 10% step load increase or a 5%/minute ramp load increase in the range of 15 to 100% of full power. Any loading rate in excess of these values may cause a reactor trip actuated by the reactor protection system. The event is analyzed to demonstrate that DNBR limits are met.

An earlier excessive load increase analysis utilized the LOFTRAN (54) system transient analysis code and the Westinghouse Improved Thermal Design Procedure. Four cases were included in the original analysis (55) (manual and automatic rod control, beginning and end of life). The end of life cases were demonstrated to be limiting, since the RCS cooldown due to load increase has the maximum impact on core power at end of life.

The two limiting UFSAR excessive load increase cases defined in the Reference (55) analysis were previously reanalyzed (56) in the analysis effort which supported the North Anna implementation (13) of the Statistical DNBR Evaluation Methodology (14),(16),(15). The RETRAN (40),(41) system transient analysis code, the COBRA (42) detailed thermal/hydraulics analysis code, and the WRB-1 critical heat flux (CHF) correlation (45) were utilized in these reanalyses. An additional excessive load increase calculation was performed in Reference (3) to examine the difference in system transient behavior between the current steam generator design and the replacement steam generators. This calculation also utilized the RETRAN (40),(41) system transient analysis code.

In both the automatic and manual reactor control end-of-life cases, the DNBR remains well above the DNBR limit (56). The DNBR analysis results with the replacement steam generators explicitly modelled would be bounded by those of the current licensing basis analysis, since the Reference (3) calculation demonstrates that the system transient behavior with the replacement steam generators is insignificantly different from that of the current steam generator design.

2.2.6 Excessive Heat Removal (Feedwater Malfunction)
(UFSAR Section 15.2.10).

Reductions in feedwater temperature or additions of excessive feedwater are means of increasing core power above full power. Such transients are attenuated by the thermal capacity of the secondary plant and of the reactor coolant system. The reactor protection system (high flux, overtemperature delta-T (OTΔT), and overpower delta-T (OPΔT) trips) prevent any power increase that could lead to a DNBR less than the limit value.

The feedwater malfunction transient was reanalyzed (57) in the analysis effort which supported the North Anna implementation (13) of the Statistical DNBR Evaluation Methodology (14),(16),(15). An evaluation of this analysis was performed in Reference (58) to quantify the impact of modifications made to the North Anna steam generator downcomer in response to observed vibrational wear to the steam generator tubes. More recently, the feedwater malfunction was reanalyzed in consideration of the potential for increased feedwater delivery to multiple steam generators (59). The RETRAN (40),(41) system transient analysis code, the COBRA (42) detailed thermal/hydraulics analysis code, and the WRB-1 critical heat flux (CHF) correlation (45) were utilized in the reanalyses.

Reference (3) has demonstrated that the transient heat transfer behavior of the replacement steam generators is insignificantly different from that of the current steam generator design. Therefore, analysis results with explicit modelling of the geometry and operating conditions of the replacement steam generators would be bounded by those of the current licensing basis analysis.

2.2.7 Loss of Normal Feedwater (incl. Station Blackout) Reanalysis (UFSAR Section 15.2.8/15.2.9).

The Loss of Normal Feedwater accident, including an "Appendix R" loss of offsite power case, was reanalyzed for extended SGTP concerns (25). This analysis will remain the licensing basis analysis for North Anna 1 operation following steam generator replacement. The conclusion of analysis validity is based on the Reference (3) analysis which demonstrates that system transient behavior with the replacement steam generators is insignificantly different from that with the current steam generator design. The inclusion of 40% steam generator tube plugging in the analysis provides an additional measure of conservatism for operation following steam generator replacement.

The Model 51F steam generators also have an increased secondary water mass at hot full power (HFP) relative to the existing steam generator model; this unquantified conservatism in the analysis further substantiates the conclusion that the Reference (25) analysis remains applicable to North Anna 1 operation following steam generator replacement. A summary of the Reference (25) analysis is presented in the following sections.

2.2.7.1 Accident Description

A loss of normal feedwater (due to pump failures, valve malfunctions, or loss of offsite AC power) results in a reduction in the capability of the secondary system to remove the heat generated in the reactor core.

If the reactor were not tripped during the accident, core damage could occur from a loss of heat sink. If an alternative supply of feedwater were not supplied to the plant, residual heat following reactor trip would heat the primary system water to the point where water relief from the pressurizer occurs. Significant loss of water from the reactor coolant system could conceivably lead to core damage. Since the plant is tripped well before the steam generator heat transfer capability is reduced, the primary system variables never approach a DNB condition.

The following provide the necessary protection against a loss of normal feedwater:

1. Reactor trip on low-low water level in any steam generator, or on water level below the AMSAC setpoint in two steam generators after a time delay, providing permissive C-20 is satisfied.
2. Reactor trip on low feedwater flow signal in any steam generator. (This signal is actually a steam/feedwater flow mismatch in coincidence with low water level.)
3. Two motor-driven auxiliary feedwater pumps (capable of delivering at least 340 gpm each) that are started on
   a. Low-low level in any steam generator,
   b. Trip of all main feedwater pumps,
   c. Any safety injection signal,
   d. Loss of offsite power,
   e. Manual actuation,
   f. AMSAC actuation.
4. One turbine-driven auxiliary feedwater pump (capable of delivering at least 700 gpm) which is started on the same signals as the motor-driven pumps.

The loss of normal feedwater accident analysis must demonstrate that the auxiliary feedwater system is capable of removing the stored and residual heat following a loss of normal feedwater, and preventing either overpressurization of the reactor coolant system or loss of water from the reactor coolant system. The results of the accident analysis establish the minimum flow requirement for the auxiliary feedwater pumps. An additional loss of normal feedwater case (the Appendix R case) is considered to demonstrate that the turbine-driven auxiliary feedwater pump is capable of providing adequate cooling to prevent RCS overpressurization or loss of RCS inventory through the pressurizer safety or relief valves.

2.2.7.2 Method of Analysis

The complete loss of normal feedwater event was analyzed (25) with the RETRAN (38),(39),(40),(41) system transient analysis code and the RETRAN extended SGTP single loop model (60). The Reference (60) model assumed a uniform steam generator tube plugging level of 40%. All assumptions were consistent with or conservative with respect to those in the previously approved analysis (61),(62),(63),(64) with the exception of the heat transfer coefficient modelling during the time between loss of feedwater and initiation of auxiliary feedwater. However, the manner in which the heat transfer coefficients used in this analysis were derived and utilized is very conservative with respect to actual plant conditions. Analysis details may be obtained from Reference (25).

2.2.7.3 Results and Conclusions

The loss of normal feedwater event analyses demonstrate that the auxiliary feedwater systems deliver sufficient feedwater to prevent the relief of reactor coolant water through the pressurizer relief or safety valves and to prevent system overpressurization. For the cases with and without offsite power available, and for the "Appendix R" case, the feedwater flow rates required to provide adequate cooling were demonstrated to be well below actual deliverable pump flow rates. The results and conclusions of the Reference (25) reanalysis remain valid for both the Model 51 and 51F steam generator designs, as evidenced by by the Reference (3) evaluation, and for steam generator tube plugging levels up to 40%.

2.2.8 Loss of External Load Reanalysis (UFSAR Section 15.2.7).

The Loss of External Electrical Load event was reanalyzed for extended SGTP concerns (24). This analysis will remain the licensing basis analysis for North Anna 1 operation following steam generator replacement. The conclusion of analysis validity is based on the Reference (3) analysis which demonstrates that system transient behavior with the replacement steam generators is insignificantly different from that with the current steam generator design. The inclusion of 40% steam generator tube plugging in the analysis provides an additional measure of conservatism for operation following steam generator replacement. A summary of the Reference (24) analysis is presented in the following sections.

2.2.8.1 Accident Description

A loss of load event can result from loss of external electrical load or from a turbine trip. For either case, offsite power is available for the continued operation of plant components such as the reactor coolant pumps. The case in which all AC power is lost (station blackout) is analyzed in Section 15.2.9 of the UFSAR.

For a turbine trip, the reactor is tripped directly (unless below approximately 30% power) by a signal derived from the turbine autostop oil pressure and turbine stop valves. The automatic steam dump system accommodates the excess steam generation. Reactor coolant temperatures and pressure do not significantly increase if the steam dump system and pressurizer pressure control system are functioning properly. If the turbine condenser is not available, the excess steam generation is dumped to the atmosphere. Additionally, main feedwater flow is lost if the turbine condenser is not available. In this situation, feedwater flow is maintained by the auxiliary feeuwater system. For a loss of external electrical load without subsequent turbine trip, no direct reactor trip signal would be generated.

Should the steam dump valves fail to open, or should their capacity be exceeded following a large loss of load, the steam-generator safety valves may lift and the reactor may be tripped by the high pressurizer pressure si~nal, the high pressurizer water level signal, or the overtemperature ΔT signal. The steam generator shell-side pressure and reactor coolant temperatures will increase rapidly. The pressurizer safety valves and steam-generator safety valves are, however, sized to protect the reactor coolant system and steam generator against overpressure for all load losses without assuming the operation of the steam dump system, pressurizer spray, pressurizer steam-operated relief valves, automatic rod cluster control assembly control, or direct reactor trip resulting from turbine trip.

2.2.8.2 Method of Analy_is

The complete loss load event was reanalyzed (24) with the RETRAN (38),(39),(40),(41) system transient analysis code and the RETRAN extended SGTP single loop model (44). All assumptions were consistent with or conservative with respect to those in the previously approved analysis (55),(16),(15).

2.2.8.3 Results and Conclusions

The results of the Reference (24) analysis support the conclusion that a total loss of external electrical load or an overpressurization due to a feedwater malfunction (a) without a direct or immediate reactor trip, (b) with conservative PSV setpoint modelling, and (c) with average steam generator tube plugging up to 40% presents no hazard to the integrity of the reactor coolant system or the main steam system. Pressure-relieving devices incorporated in the two systems are adequate to keep the maximum pressures within the design limits. Furthermore, the integrity of the core is maintained by operation of the reactor protection system, i.e., the DNBR will be maintained above the limit value. Thus there will be no cladding damage and no release of fission products to the reactor coolant system. Based on the Reference (3) comparison of the transient behavior of the Model 51 and 51F steam generators, it may be concluded that the Reference (24) Loss of Load analysis results remain bounding for either steam generator design.

2.2.9 Control Rod Drop/Misalignment (UFSAR Section 15.2.3).

When operating at power, a single or multiple dropped control rod may result in a transient leading to reduced margins to fuel design limits and, in particular, to DNB limits. This would be a result of increased power distribution peaking factors with the inserted (dropped) rods and a possible "return to power" transient produced by feedback or automatic control. Depending on the control system, the "return to power" transient could result a power level in excess of the initial level.

Normally the plant is protected from exceeding DNB limits through a negative flux rate trip system. The system will sense the initial rapidly decreasing neutron flux (as a negative rate) and trip the reactor to end the event. For some events, however, the flux decrease rate may be insufficient for a trip. For these, it was originally thought that the peaking factors and transient powers would not result in limits being exceeded, because a limited overshoot above the initial power level was assumed to occur.

In 1986, a core uprate program was implemented for North Anna which required reanalyses of most of the UFSAR Chapter 15 accidents (36),(37). These reanalyses included a reanalysis of the dropped rod event using the methodology of Reference (65). Credit was taken in the dropped rod reanalysis for negative flux rate trip protection, which was justified for dropped control rod worths greater than 400 pcm for three-loop plants.

In 1990, Virginia Power acquired from Westinghouse the transient database and methodology information necessary to perform the dropped rod analyses of either Reference (65) or (66). Reference (66) describes a dropped rod analysis methodology developed by Westinghouse and funded by the Westinghouse Owner's Group (WOG). Th . methodology is an extension of the methodology of Reference (65) and eliminates the need to take credit for the negative flux rate trip for dropped rod worths greater than 400 pcm. Since the acquisition of this information, Virginia power has performed evaluations which show the applicability of the methodology, the correlations, and the transient database for analysis of the dropped rod event for North Anna Units 1 and 2 (67). A reload evaluation code called RDROP has been developed (68) and documented (69) permitting cycle-specific dropped rod evaluations to be performed under the new Westinghouse methodology.

Dropped rod limit lines have been developed for use in the RDRCP reload dropped rod evaluation code (68),(69). The limit lines were developed for minimum measured RCS flow rates of 275,300 gpm (70) and 284,000 gpm (71). Because a Technical Specification minimum measured flow requirement of 284,000 will be applicable following steam generator replacement, the Reference (71) dropped rod limit lines will be utilized in reload evaluations of the droppea rod event.

Because the Reference (3) analysis supports the conclusion that the replacement steam generators' transient be'avior is essentially identical to that of the current steam generator design, it may be concluded that steam generator replacement does not adversely impact the results of the currently applicable dropped rod licensing basis analysis.

2.2.10 Partial Loss of Flow (UFSAR Section 15.2.5).

A partial loss of coolant flow accident can result from a mechanical or electrical failure in a reactor coolant pump, from a fault in the power supply to the pump, or from inadvertent closure of a loop isolation valve. If the reactor is at power at the time of the accident, the immediate effect of loss of coolant flow is a rapid increase in the coolant temperature. This increase could result in DNB with subsequent fuel damage if the reactor is not tripped promptly.

The results of the partial loss of flow accident analysis are bounded by those of the complete loss of flow event. A loss of flow analysis applicable to North Anna operation following steam generator replacement is presented in Section 2.2.3.

2.2.11 Rod Bank Withdrawal at Power Reanalysis
(UFSAR Section 15.2.2).

The Rod Withdrawal at Power Accident (RWAP) was reanalyzed for extended SGTP concerns (26). This analysis will remain the licensing basis analysis supporting North Anna Unit 1 operation following steam generator replacement.

Steam generator performance is a minor contributor in the analysis of the system transient response to a rod withdrawal at power event. Furthermore, the Reference (3) analysis supports the conclusion that the replacement steam generators' transient behavior is essentially identical to that of the current steam generator design in the analysis conditions (0% SGTP); therefore, it may be concluded on the basis of the Reference (3) analysis that steam generator replacement does not adversely impact the results of the currently applicable rod withdrawal at power licensing basis analysis.

A description of the rod withdrawal at power analysis is presented in the following sections.

2.2.11.1 Accident Description

The uncontrolled rod cluster control assembly (RCCA) withdrawal at power is a postulated Condition II event initiated by operator action. The transient is characterized by an increase in core heat flux resulting in a mismatch between core power generation and power removal by the steam generator. This power mismatch, which persists until the steam generator pressure reaches the relief or safety valve setpoint, causes an increase in the primary coolant temperature. The transient would result in violation of the core thermal limits if not terminated by either manual or automatic action. The reactor protection system is designed to terminate the transient prior to exceeding core thermal limits.

The automatic features of the reactor protection system that prevent core damage due to a RWAP event include the following reactor trips of which 1, 2, 4, and 5 are taken credit for in the RWAP analysis.

1. Power range neutron flux instrumentation actuates a reactor trip if two out of four channels exceed an overpower setpoint.

2. Reactor trip is actuated if any two out of three ΔT channels exceed an overtemperature ΔT setpoint. This setpoint is automatically varied with axial power imbalance, coolant temperature, and pressure to protect against DNB.

3. Reactor trip is actuated if any two out of three ΔT channels exceed an overpower ΔT setpoint. This setpoint is automatically varied with axial power imbalance to ensure that the allowable heat generation rate (KW/ft) is not exceeded.

4. A high pressurizer pressure reactor trip actuated from any two out of three pressure channels is set at a fixed point. This trip pressure setpoint is less than the lift pressure setpoint for the pressurizer safety valves.

5. A high pressurizer water level reactor trip actuated from any two out of three leve. channels is set at a fixed point.

In addition to the above listed reactor trips, the following RCCA withdrawal blocks limit the consequences of the RWAP accident. However, credit is not taken for these blocks in the analysis.

1. High neutron flux (one out of four).

2. Overpower ΔT (two out of three).

3. Overtemperature ΔT (two out of three).

The RWAP analyis is performed to verify that the high flux and overtemperature ΔT trip setpoints and system time constants are adequate to ensure that the core avoids the onset of DNB in the event of a rod withdrawal at power accident.

2.2.11.2 Method of Analysis

The rod withdrawal at power event was reanalyzed (26) in the extended SGTP analysis effort (22),(8),(30),(9) with the RETRAN (38),(39),(40),(41) system transient analysis code and a RETRAN single loop model. The RETRAN code provided transient pressures, core inlet temperatures, heat fluxes and core flows which were used as input to a detailed thermal/hydraulic statepoint analysis using the COBRA (42) code. The WRB-1 correlation (45),(46) was used.

To fully evaluate the RWAP event, a wide range of initial plant conditions are analyzed to determine those which are most limiting. Permutations of the following conditions were analyzed:

1. Initial NSSS power levels of 100, 60, and 10% with minimum feedback for a wide range of reactivity insertion rates.

2. Initial NSSS power levels of 100, 60, and 10% with maximum feedback for a wide range of reactivity insertion rates.

It is assumed in the analysis that the steam dump and rod control systems provide no protection in the event of a RWAP event. However, credit is take for pressurizer PORV's and safety valves, steam generator atmospheric relief valves and safety valves, as well as pressurizer spray (full flow from both valves is assumed).

2.2.11.1 Results and Conclusions

The reanalysis of the rod withdrawal at power event demonstrated that the minimum DNBR will remain above the DNBR design limit under any postulated RWAP event. Analysis criteria will continue to be met following steam generator replacement provided the Technical Specification minimum measured RCS flow rate requirement (total RCS flow rate ≥284,000 gpm) is met. (The licensing basis RWAP analysis (26) conservatively assumes a minimum measured total RCS flow rate of 275,300 gpm.) Because this requirement is expected to be met with margin, the current licensing basis [illegible] is remains valid for North Anna 1 following steam generator replacement.

2.2.12 Rod Withdrawal from Subcritical (UFSAR Section 15.2.1).

A rod cluster control assembly withdrawal accident is defined as an uncontrolled addition of reactivity to the reactor core caused by withdrawal of the rod cluster control assemblies, thereby producing a power excursion. Potential causes of the event include malfunctions of the reactor control and control rod drive systems, and operator error. At cold shutdown, protection against the consequences of this accident are provided by the high source range count rate trip. At hot standby and hot shutdown, protection is provided by the source range trip. During startup and at power, protection is provided by the intermediate range or power range high neutron flux reactor trips, or by the intermediate and power range rod stops.

The neutron flux response to a continuous reactivity insertion is characterized by an exponential increase. Once the amount of reactivity inserted corresponds to the delayed neutron fraction of the core, the power increase is very rapid, and is terminated by reactivity feedback effect of the negative doppler coefficient. This self-limitation of the power burst is of primary importance since it limits the power to a tolerable level during the delay time required for automatic protective action.

The rod withdrawal from subcritical transient was most recently reanalyzed (72) in the analysis effort which supported relaxation of North Anna Reload Safety Analysis Checklist (RSAC) (52) parameters and the North Anna implementation (13) of the Statistical DNBR Evaluation Methodology

(14),(16),(15). The RETRAN (40),(41) system transient analysis code, the COBRA (42) detailed thermal/hydraulics analysis code, and the WRB-1 critical heat flux (CHF) correlation (45) were utilized in these reanalyses.

The Reference (72) analysis will remain the licensing basis analysis for North Anna 1 operation following steam generator replacement. Steam generator replacement does not significantly affect the results of this analysis because the most significant transient parameter variations which affect analysis results are a result of reactor core and reactor protection system performance. Analysis criteria will continue to be met following steam generator replacement provided the Technical Specification minimum measured RCS flow rate requirement (total RCS flow rate ≥284,000 gpm) is met. Because this requirement is expected to be met with margin, the Reference (72) analysis remains bounding for North Anna 1 operation following steam generator replacement.

2.2.13 Inactive Loop Startup (UFSAR Section 15.2.6).

The Inactive Loop Startup event was most recently reanalyzed (72) in the analysis effort which supported the North Anna implementation (13) of the Statistical DNBR Evaluation Methodology (14),(16),(15). However, the initial system configuration and conditions assumed in the analysis of the inactive loop startup event are precluded from occurrence by Technical Specifications. The probability of occurrence or consequences of the event (both the minimum DNBR and reactivity insertion due to boron dilution aspects) are, therefore, not affected by steam generator replacement.

2.2.14 Spurious Operation of the Safety Injection System (UFSAR Section 15.2.14).

Spurious safety injection system operation at power could be caused by operator error or a false electrical actuating signal. Following the actuation signal, the suction of the coolant charging pumps is diverted from the volume control tank to the refueling water storage tank. The valves isolating the boron injection tank from the charging pumps and the valves isolating the boron injection tank from the injection header then automatically open. The charging pumps then force boric acid solution from the boron injection tank through the header and injection line, and into the cold legs of each loop. The safety injection pumps also start automatically but provide no flow when the reactor coolant system is at normal pressure. The passive safety injection accumulators and the low-head system also provide no flow at normal RCS pressure.

The spurious safety injection transient was most recently reanalyzed by Westinghouse for the North Anna core uprating analysis effort (36),(37). Their analysis revealed that the transient does not challenge the integrity of the RCS, and that the transient DNBR remains above the initial value.

The Westinghouse core uprating reanalysis of the spurious safety injection transient will remain the licensing basis analysis for North Anna Unit 1 operation following steam generator replacement. The conclusion of analysis applicability is based on the consideration that the spurious safety injection accident is an RCS cooling event which is not affected by primary to secondary heat transfer effects. The slight increase in volume associated with the replacement of the Model 51 with Model 51F steam generators might provide some small analysis benefit in retarding the transient cooldown.

2.2.15 Minor Secondary Steam Pipe Breaks (UFSAR Section 15.3.2) (UFSAR Section 15.2.14).

Section 15.3.2.1 of the North Anna UFSAR presents the description and results of the most recent evaluation of minor secondary system pipe breaks. The section concludes that the analyses presented in Section 15.4.2.1 of the UFSAR (Main Steamline Break) demonstrate that the consequences of a minor secondary system pipe break are acceptable, since a DNBR less than the DNBR limit does not occur even for a more critical major secondary system pipe break. The evaluation of the main steamline break event for steam generator replacement concerns is presented in Section 2.2.22.

2.2.16 Misloaded Fuel Assembly (UFSAR Section 15.3.3)

Fuel and core loading errors, such as those that can arise from the inadvertent loading of one or more fuel assemblies into improper positions, loading a fuel rod during manufacture with one or more pellets of the wrong enrichment, or the loading of a full fuel assembly during manufacture with pellets of the wrong enrichment will lead to increased heat fluxes if the error results in placing fuel in core positions calling for fuel of lesser enrichment. Also included among possible core-loading errors is the inadvertent loading of one or more fuel assemblies requiring burnable poison rods into a new core without burnable poison rods.

The UFSAR Section 15.3.3 analysis of the misloaded fuel assembly accident indicates that the consequences of this accident are limited either by administrative controls or operational actions in response to the detection of a misloaded assembly. Steam generator replacement does not affect this accident analysis in any way. As such, no further evaluation is necessary to support operation with the replacement steam generators.

2.2.17 Single Rod Withdrawal at Power (UFSAR Section 15.3.7)

The Single Rod Withdrawal at Power event produces a system transient response which is similar to the uncontrolled control bank assembly withdrawal; that is, it results in an increase in core heat flux and a mismatch between core power generation and power removal by the steam generators. This power mismatch, which persists until the steam generator pressure reaches the relief or safety valve setpoint causes an increase in the primary coolant temperature. The transient would result in a violation of the core thermal limits if not terminated by either manual or automatic action.

No single electrical or mechanical failure in the Rod Control System could cause the accidental withdrawal of a single rod cluster control assembly from the inserted bank at full power operation. The operator could deliberately withdraw a single rod cluster control assembly in the control bank. In the extremely unlikely event of simultaneous electrical failures resulting in single rod cluster control assembly withdrawal, "rod deviation" and "rod control urgent failure" would be displayed on the plant annunciator, and the rod position indicators would indicate the relative positions of the assemblies in the bank. The urgent failure alarm also inhibits automatic rod motion in the group in which it occurs. Withdrawal of a single rod cluster control assembly by operator action, whether deliberate or by a combination of errors, would result in an activation of the same alarm and the same visual indications.

In the unlikely event of multiple failures that result in continuous withdrawal of a single rod cluster control assembly, it is not possible in all cases to provide assurance of automatic reactor trip so that the core safety limits are not violated. Withdrawal of a single rod cluster control assembly results in both positive reactivity insertion tending to increase core power, and an increase in local power density in the core area "covered" by the rod cluster control assembly. The resulting power distributions for a single rod withdrawal are evaluated for each reload core to ensure that less than 5% of the core has radial peaking factors in excess of the design radial factor assumed in the rod bank withdrawal at power transient and core thermal/hydraulics licensing basis analysis. The rod bank withdrawal at power transient analysis is performed with the RETRAN (40),(41) system transient analysis code, the COBRA (42) detailed thermal/hydraulics analysis code, and the WRB-1 critical heat flux (CHF) correlation (45).

Although the single rod withdrawal at power is a Condition III transient (Infrequent Fault), the transient analysis is performed with the Statistical DNBR Evaluation Methodology (14),(16),(15) as permitted by the Reference (15) NRC Safety Evaluation Report (SER). The design radial peaking factor ($F\Delta H$) against which reload safety analysis checklist evaluations are performed includes measurement uncertainty consistent with a deterministic treatment of uncertainties. This approach is conservative. Full utilization of the Statistical DNBR evaluation methodology (14) would provide additional analysis margin, since the methodology accommodates the radial peaking factor measurement uncertainty in the DNBR design limit in a less restrictive but appropriately conservative manner.

The results of the Single Rod Withdrawal at Power event analysis are evaluated on a reload basis in the reload thermal/hydraulics evaluation calculation. (See Section 2.3.1.) For each reload core, it is verified that less than 5% of the fuel rods have radial peaking factors in excess of 1.55. This limit is reflected in the RSAC (52).

The Reference (73) Rod Bank Withdrawal at Power analysis will serve as the base transient analysis for the Single Rod Withdrawal at Power supporting North Anna 1 operation following steam generator replacement. Analysis criteria will continue to be met following steam generator replacement provided the Technical Specification minimum measured RCS flow rate requirement (total RCS flow rate ≥284,000 gpm) is met. Because this requirement is expected to be met with margin, the Reference (73) analysis remains bounding for North Anna 1 operation following steam generator replacement.

2.2.18 Volume Control Tank Rupture (UFSAR Section 15.3.6)

The analysis of the volume control tank rupture is in no way affected by steam generator replacement. As such, the existing UFSAR transient analysis remains the licensing basis analysis supporting North Anna 1 operation following steam generator replacement.

2.2.19 Waste Gas Decay Tank Rupture (UFSAR Section 15.3.5)

The analysis of the waste gas decay tank rupture is in no way affected by steam generator replacement. As such, the existing UFSAR transient analysis remains the licensing basis analysis supporting North Anna 1 operation following steam generator replacement.

2.2.20 Fuel Handling Accident Outside Containment (UFSAR Section 15.4.5)

The analysis of fuel handling accidents outside containment is in no way affected by steam generator replacement. As such, the existing UFSAR transient analysis remains the licensing basis analysis supporting North Anna 1 operation following steam generator replacement.

2.2.21 Fuel Handling Accident Inside Containment
(UFSAR Section 15.4.7)

The analysis of fuel handling accidents inside containment is in no way affected by steam generator replacement. As such, the existing UFSAR transient analysis remains the licensing basis analysis supporting North Anna 1 operation following steam generator replacement.

2.2.22 Major Secondary System Pipe Ruptures (Main Steamline Break) (UFSAR Section 15.4.2.1).

The steam release arising from a rupture of a main steam pipe would result in an initial increase in steam flow, which decreases as the steam pressure falls. The energy removal from the reactor coolant system causes a reduction of coolant temperature and pressure. In the presence of a negative moderator temperature coefficient, the cooldown results in a reduction of core shutdown margin. If the most reactive rod cluster control assembly is assumed stuck in its fully withdrawn position after reactor trip, there is an increased possibility that the core will become critical and return to power. A return to power following a steam pipe rupture is a potential problem mainly because of the high power peaking factors that exist, assuming the most reactive rod cluster control assembly to be stuck in its fully withdrawn position. The core is ultimately shut down by the boric acid injection delivered by the safety injection system.

The main steamline break analysis is performed to demonstrate that there would be no core damage due to the onset of DNB, and that the energy release to containment does not cause failure of the containment structure.

A series of calculations (74),(75),(76) were performed to demonstrate that the consequences of a main steamline break event would not exceed the analysis criteria as presented above. The applicability of these analyses is assessed for each reload core; penalties against available retained DNBR margin resulting from the reload assessment are quantified and applied as necessary. Because the Reference (3) analysis demonstrates that system transient behavior with the replacement steam generators is insignificantly different from that with the current steam generator design, the existing licensing basis MSLB analyses will remain valid for North Anna Unit 1 following steam generator replacement.

The replacement steam generators have also been evaluated for their impact on radiological dose calculations for the main steam line break (77). The effect of the minor physical differences between the Model 51 and Model 51F steam generators are more than offset by conservatisms used in the dose calculation methodology. The existing dose calculations bound operation with the replacement steam generators.

2.2.23 Rupture of a Main Feedwater Pipe (Main Feedline Break) (UFSAR Section 15.4.2.2).

A major feedwater line rupture is defined as a break in a feedwater pipe large enough to prevent the addition of sufficient feedwater to the steam generators to maintain shell-side fluid inventory in the steam generators. If the break is postulated in a feedline between the check valve and the steam generator, fluid from the steam generator may also be discharged through the break. Further, a break in this location could preclude the subsequent addition of auxiliary feedwater to the affected steam generator. (A break upstream of the feedline check valve would affect the nuclear steam supply system only as a loss of feedwater; see Section 2.2.7 of this report.)

Depending upon the size of the break and the plant operating conditions at the time of the break, the break could cause either a reactor coolant system cooldown (by excessive energy discharge through the break) or a reactor coolant system heatup. Potential reactor coolant system cooldown resulting from a secondary pipe rupture is evaluated in UFSAR Section 15.4.2.1. (See Section 2.2.22 of this report.) Therefore, only the reactor coolant system heatup effects are evaluated for a feedline rupture.

The consequences of feedline break events upstream of the feedline check valve are bounded in severity by the consequences of the loss of normal feedwater event which has been evaluated in Section 2.2.7 of this report. As described in the following, the consequences of feedline break events for breaks downstream of the check valve have been evaluated to assess the impact of steam generator replacement.

The main feedline break event was most recently reanalyzed (78) as part of an effort to justify a reduced flow rate from the motor driven auxiliary feedwater pumps. Because the Reference (3) analysis demonstrates that system transient behavior with the replacement steam generators is insignificantly different from that with the current steam generator design, the existing licensing basis analysis will remain valid for North Anna Unit 1 following steam generator replacement.

2.2.24 Control Rod Ejection (UFSAR Section 15.4.6).

The control rod ejection transient is defined as the mechanical failure of a control rod mechanism pressure housing, resulting in the ejection of a rod cluster control assembly and drive shaft. The consequence of this mechanical failure is a rapid reactivity insertion together with an adverse core power distribution, possibly leading to localized fuel rod damage.

The analysis methodology for the control rod ejection accident is documented in Reference (79); the most recent analysis is documented in Reference (80). The analysis is performed in two stages; first, an average core nuclear power transient calculation, and then a hot-spot heat transfer calculation. The average core power calculation is performed using point neutron kinetics methods to determine the average power generation with time, including the various total core feedback effects, i.e., Doppler reactivity and moderator reactivity. Enthalpy and temperature transients in the hot spot are then determined by multiplying the average core energy generation by the hot channel factor and performing a fuel rod transient heat transfer calculation. The power distribution without feedback is pessimistically assumed to persist throughout the transient.

The Reference (80) Control Rod Ejection analysis will serve as the licensing basis analysis supporting North Anna 1 operation following steam generator replacement. Steam generator replacement does not significantly affect the results of this analysis because the most significant transient parameter variations which affect analysis results are a result of reactor core and reactor protection system performance. Analysis criteria will continue to be met following steam generator replacement provided the Technical Specification minimum measured RCS flow rate requirement (total RCS flow rate ≥284,000 gpm) is met. Because this requirement is expected to be met with margin, the Reference (80) analysis remains bounding for North Anna 1 operation following steam generator replacement.

2.2.25 Steam Generator Tube Rupture (UFSAR Section 15.4.3).

The steam generator tube rupture accident is defined as the complete severance of a single steam generator tube which occurs during power operation. Three cases are presented in the UFSAR. The first case assumes the concentration of fission products in the reactor coolant is equal to the activity associated with 1% failed fuel. The second and third cases assume initial activities equal to the Technical Specification limit of 1 µCi/gm, with an additional iodine activity source from either a pre-accident iodine spike or an iodine spike concurrent with the accident. The accident leads to an increase in contamination of the secondary system due to leakage of radioactive coolant from the reactor coolant system. In the event of a coincident loss of offsite power, or failure of the condenser dump system, discharge of activity to the atmosphere takes place via the steam generator safety and or power operated relief valves.

Steam generator replacement and its accompanying effect on RCS flow, primary to secondary heat transfer, secondary steam pressure and RCS loop resistance would have insignificant impact on the analysis results of the steam generator tube rupture transient. The Reference (3) analysis supports the conclusion that the replacement steam generators' transient behavior is essentially identical to that of the current steam generator design. Therefore, it may be concluded on the basis of the Reference (3) analysis that steam generator replacement does not adversely impact the results of the currently applicable steam generator tube rupture analysis.

The replacement steam generators have also been evaluated for their impact on radiological dose calculations for the steam generator tube rupture transient (77). The effect of the minor physical differences between the Model 51 and Model 51F steam generators are more than offset by conservatisms used in the dose calculation. The existing dose calculations bound operation with the replacement steam generators.

2.2.26 Locked Rotor/Sheared Shaft (UFSAR Section 15.4.4).

The Locked Rotor/Sheared shaft accident was reanalyzed for extended SGTP concerns (28). However, this analysis assumed a reduced minimum measured Technical Specification thermal design flow rate of 275,300 gpm. Because the locked rotor is a limiting DNB transient, and because reload design margin in the RSAC (52) FΔH limit for locked rotor is tied up in the reduced flow (275,300 gpm) assumption, it is preferable to revert to the analysis applicable prior to the extended SGTP analysis effort (22),(8),(30),(9), which is documented in Reference (81). This analysis was approved in Reference (82). A summary of the Reference (81) analysis is presented in the following.

The locked rotor event is defined as the instantaneous seizure of a reactor coolant pump rotor. Flow through the affected reactor coolant loop is rapidly reduced, leading to a reactor trip on low flow signal. Following reactor trip, heat stored in the fuel rods continues to be transferred to the coolant causing the coolant to expand. At the same time, heat transfer to the secondary side of the steam generators is reduced, first because the reduced flow results in a decreased tube-side film coefficient, and then because the reactor coolant in the tubes cools down while the shell-side temperature increases. Turbine steam flow is reduced to zero upon plant trip. The rapid expansion of the coolant in the reactor core, combined with the reduced heat transfer in the steam generators, causes an insurge into the pressurizer and a pressure increase throughout the reactor coolant system. The insurge into the pressurizer (a) compresses the steam volume, (b) actuates the automatic spray system,

(c) opens the power operated relief valves, and (d) opens the pressurizer safety valves. The two power operated relief valves are designed for reliable operation and would be expected to function properly during the accident. However, for conservatism, their pressure reducing effect and the pressure-reducing effect of the spray are not considered in the overpressure portion of the analysis.

The locked rotor/sheared shaft event is analyzed (81) with the RETRAN (38),(39),(40),(41) system transient analysis code and a two-loop RETRAN model. The Reference (81) analysis and the North Anna Unit 1 extended SGTP analysis (83) confirmed that the results of the sheared shaft event are bounded by those of the locked rotor event.

The single reactor coolant pump locked rotor incident is analyzed in two parts. First a peak pressure calculation is performed using conservative assumptions that tend to maximize the heat transfer from the fuel to the coolant. This calculation assumes that the fuel rods in the core do not experience departure from nuceate boiling (DNB). Second, the calculation is repeated assuming the limiting fuel rod in the core experiences departure from nucleate boiling; the fraction of rods which are predicted to experience DNB (as determined by a reload evaluation based on an FΔH rod census evaluation) is calculated to verify that the percentage of fuel rod failure does not exceed that which is assumed in the currently applicable offsite dose calculation (13%) (84).

The Reference (81),(82) Locked Rotor analysis will remain the licensing basis analysis for North Anna 1 operation following steam generator replacement. Steam generator replacement does not significantly affect the results of this analysis because the most significant transient parameter variations which affect analysis results are a result of reactor core and reactor protection system performance. Analysis criteria will continue to be met following steam generator replacement provided the Technical Specification minimum measured RCS flow rate requirement (total RCS flow rate ≥284,000 gpm) is met. Because this requirement is expected to be met with margin, the Reference (81) analysis remains bounding for North Anna 1 operation following steam generator replacement.

The replacement steam generators have also been evaluated for their impact on radiological dose calculations for the RCP locked rotor transient (.7). The effect of the minor physical differences between the Model 51 and Model 51F steam generators is a negligibly small benefit for the calculation of the doses. The ex'sting dose calculations bound operation with the replacement steam generators.

2.2.27 Small Break LOCA Reanalysis (UFSAR Section 15.3.1)

The Small Break Loss of Coolant Accident was reanalyzed for extended SGTP concerns using the methodology described in Reference (29). The Reference (29) analysis will serve as the small break LOCA licensing basis analysis for both North Anna units following Unit 1 steam generator replacement. This section evaluates the impact of the replacement steam generators upon the results of the existing analysis.

In the Reference (29) analysis, assumptions have been made which reflect operation with extended steam generator tube plugging (SGTP) in addition to changes in other key analysis inputs. The key input changes incorporated in this analysis are listed below.

1. Use of the NOTRUMP code and associated evaluation model (References 2d, 2e of T.S. 6.9.1.7.e; also References (85), (86))
2. Assumption of 35% uniform steam generator tube plugging, including a reduced RCS total flowrate of 264400 gpm
3. Peak Heat Flux Hot Channel Factor, F(Q), of 2.32 (unchanged from present value; provided for information)
4. Increased value of Normalized Hot Channel Factor, K(z) represented as a linear function between the points below. (to be implemented in the Core Operating Limits Report, presenting maximum FQ x K(z) which satisfies large and small break LOCA acceptance criteria)

   (height, K(z)): (0.0, 1.0) (6.0, 1.0) (12.0, 0.925)

5. Peak value for Enthalpy Hot Channel Factor, FNΔh, of 1.60 (not being implemented, since FNΔh is limited by LBLOCA)
6. Flow imbalance of 52 gpm between minimum and maximum flow high head safety injection branch lines (at 0 psig backpressure)
7. A full core of North Anna Improved Fuel (NAIF) (bounds operation with 17x17 Standard and NAIF mixed cores)

This analysis was implemented under the provisions of Technical Specification 6.9.1.7 (Core Operating Limits Report, COLR) via the Virginia Power 10 CFR 50.59 change process. Reference (87) documents the analysis approval.

The effects of operation with Model 51F replacement steam generators has been evaluated with respect to the Reference (29) analysis. This evaluation is described here. The following features of the replacement steam generators were considered for potential impact upon the existing small break analysis behavior.

1. number of tubes
2. total tube heat transfer area
3. tube overall heat transfer coefficient
4. other secondary side characteristics

As reported in References (4) and (88), both the total number of tubes and primary-to-secondary heat transfer area are greater for the Model 51F SG's. The increased number of tubes provide approximately 6% greater primary side flow area. This area increase results in a benefit for the small break LOCA transient since increased flow area reduces the resistance (K/A^2) to the flow of steam through the tubes. This increased flow of steam may serve to clear the loop seal earlier in the transient and thus aid in mitigating any core level decrease. This would lead to less of a core uncovery transient and serve to reduce the calculated PCT. The steam generators play a vital role as heat sinks in the small break LOCA transient, supplementing the energy removed from the RCS by the break itself and ECCS operation. The increased heat transfer area (54500 ft^2

vs. 51500 ft^2) will increase the ability of the SG's to remove heat from the primary during the transient. This increased heat transfer ability is expected to provide an additional PCT benefit for the small break transient.

One feature of the replacement steam generators which potentially degrades heat transfer is the use of thermally treated Inconel 690 tubing instead of mill-annealed Inconel 600 in the existing design. The resulting change in overall heat transfer coefficient for the Model 51F SG's has been determined using design data from the Westinghouse thermal and hydraulic design reports for the two SG designs (4),(88). The overall heat transfer coefficient consists of four separate terms as reported in References (4) and (88). These are a primary side film coefficient, thermal conductivity of the tube wall, the secondary side film coefficient and a tube fouling resistance term. Table 2.2.27-1 provides the values for each term for the Model 51 and 51F SG's, from Reference (4) and (88).

For this evaluation, zero tube fouling has been assumed, so that the effect of the other physical SG differences can be isolated. The fouling term is typically an assumed value, inferred from prior Westinghouse experience and empirical data obtained from actual plant operation with particular SG tube materials and design. Including the fouling factor in this assessment would not allow a meaningful comparison of the other factors, which can be more explicitly obtained.

The Table 2.2.27-1 data indicate that the Model 51F overall heat transfer coefficient is approximately 3% smaller than that for the existing SG's. This would potentially cause less heat transfer from the RCS to the SG's during the transient, per unit heat transfer area. However, since the Model 51F SG's have greater heat transfer area and primary side flow area, there would be no net heat transfer degradation. Considering all effects, it is expected that at a given level of SG tube plugging, the Model 51F SG's would provide better overall heat transfer during a small break LOCA event. It is therefore concluded that the existing small break LOCA transient analysis will remain bounding for operation with the replacement steam generators.

Table 2.2.27-1

Model 51 and 51F Steam Generator Heat Transfer Resistances at 100% Load and 0% Tube Plugging

Term	Model 51	Model 51F
Inside Tube Resistance*	0.0001674	0.0001804
Tube Wall Resistance	0.0004575	0.0004628
Outside Tube Resistance	0.0000777	0.0000814
Fouling Resistance	0.0	0.0
Total Heat Transfer Resistance	0.0007026	0.0007246
Overall Heat Transfer Coefficient ($Btu/hr-ft^2-F$)	1423.	1380.

* resistances are in units of $hr-ft^2-F/Btu$

2.2.28 LARGE BREAK LOCA (UFSAR Section 15.4.1).

This discussion presents the results of the 15% steam generator tube plugging reanalysis of the large break LO transient for North Anna Power Station (89). Analysis assumptions have been made which reflect operation with 15% steam generator tube plugging (SGTP) in addition to changes in other key analysis inputs. The Reference (89) analysis will serve as the large break LOCA licensing basis analysis for North Anna Unit 1 following steam generator replacement. This 15% SGTP analysis is the current analysis of record for North Anna Unit 2 (90). This section evaluates the impact of the replacement steam generators upon the results of this analysis.

In the Reference (89) analysis, assumptions have been made which reflect operation with extended steam generator tube plugging (SGTP) in addition to changes in other key analysis inputs. The key input changes incorporated in this analysis are listed below.

1. Use of the 1981 evaluation model with BART (References 2a, 2b of T.S. 6.9.1.7.c, also References (91) and (92))

2. Assumption of 15% uniform steam generator tube plugging

3. Peak Heat Flux Hot Channel Factor, FQ(z), of 2.19 (unchanged from present value; provided for information)

4. Peak value for Enthalpy Hot Channel Factor, FNΔh, of 1.55 (unchanged from pre:.nt value; provided for information)

5. A full core of North Anna Improved Fuel (NAIF) (bounds operation with 17x17 Standard and NAIF mixed cores)

This analysis was implemented under the provisions of Technical Specification 6.9.1.7 (Core Operating Limi.. Report, COLR) via the Virginia Power 10 CFR 50.59 change process. Reference (90) documents the analysis approval.

The effects of operation with Model 51F replacement steam generators has been evaluation with respect to the Reference (89) analysis. This evaluation is described here. The following features of the replacement steam generators were considered for potential impact upon the existing large break analysis behavior.

- number of tubes
- total tube heat transfer area

As reported in References (4) and (88), both the total number of tubes and primary-to-secondary heat transfer area are greater for the Model 51F SG's. The increased number of tubes provide approximately a 6% increase in primary side tube flow area. This area increase results in a benefit for the large break LOCA transient since increased flow area reduces the resistance (K/A^2) to the flow of steam through the tubes during the reflood portion of the transient. This will allow better reflooding to occur, which would provide a benefit in calculated PCT. This is the main effect for consideration of the impact upon the large break LOCA analysis from the replacement steam generators. Another less significant consideration is discussed below.

The steam generators do not play a significant role in the removal of heat from the primary system during a large break LOCA. Even though the heat transfer area for the Model 51F SG's increases, (54500 ft^2 vs. 51500 ft^2) this will not affect the large break LOCA results. The dominant effect of the increased heat transfer area, therefore, is the expected reflooding benefit described above. It is concluded that the existing large break LOCA analysis results will remain bounding for operation with the replacement s m generators.

2.2.28.1 Post-LOCA Recirculation Switchover Time

Following a LOCA, safety injection enters the core region through the cold leg. Assuming a cold leg break, borated coolant enters the core region from the intact cold leg, down the downcomer, and into the core. Steam exits through the hot leg, and excess safety injection water spills out the break. Although the water vapor exits the core, only a small fraction of the dissolved boron can be expected to be carried off in the steam; the concentration of the boron, therefore, increases over time in the region of the core. If the boron concentration reaches the solubility limit, boron will begin to precipitate out of solution, forming a sticky paste which can block coolant flow channels in the core. Such a condition may lead to inadequate cooling of the fuel.

If the break is in the hot leg or in the pressurizer, safety injection water will flow down the downcomer, up through the core, and out the break, thereby continuously replacing the boric acid solution in the core region. In such a situation, switchover to hot leg recirculation is not necessary. However, there is no unambiguous way to locate the pipe break from the control room, so simultaneous or alternating cold and hot leg injection is required at a specific time for all LOCA's.

Reference (93) documents the analysis which calculates the time required to reach the solubility limit under modelled conditions. The calculational methodology requires that an inventory be taken of all sources of boric acid solution which may end up in the sump after a LOCA. Potential sources are the Refueling Water Storage Tank (RWST), the Reactor

Coolant System (RCS), the Safety Injection (SI) Accumulators, and the Emergency Core Cooling System (ECCS) piping. For each source the maximum boron concentration and volume are used to calculate the total amount of boron that can potentially concentrate in the reactor vessel. For ease of calculation, the concentration of each component is assumed to be equal to the concentration of the component with the highest concentration.

The time dependent boron concentration in the core is calculated by calculating the mass of steam which exits the core region essentially pure or free of boron over a finite time period. The boron associated with the steam leaving the region stays in solution and the mass of steam displaced by the liquid changed to steam is replaced by sump water which has its own weight percent boron associated with it. Over each time period the addition of the boron from the sump water entering the core region increases the boron concentration in that region. Since pure steam leaves the core and is assumed to condense to liquid in the containment, the sump concentration is decreased over each time period.

It is assumed that the decay heat calculated at the beginning of the evaporation time interval is constant over the entire time interval; this is conservative, as actual decay heat (and hence, the actual evaporation rate) are decreasing over the time interval.

The limiting case for boron precipitation is a double ended break of the cold leg pipe. The calculation of boric acid concentration increase is started when the core is quenched at approximately 10 minutes after the break occurs. After the core is quenched, it is assumed that the RCS is stable at a constant low pressure (e.g., 20 psia), the liquid in the core region is relatively stagnant, and core decay heat is the primary energy source.

The solubility limit is 27.4 weight percent boron as H_3BO_3, the NRC requires that 4 weight percent be used for margin. Therefore the time to switchover to hot leg recirculation is the time required for the boron concentration to reach 23.4 weight percent.

The Reference (93) calculation assumed unplugged (0% SGTP) steam generators. The replacement steam generators each result in a modest 44 ft^3 increase in RCS volume per steam generator relative to the current steam generators in the unplugged condition. It has been determined that the increase in RCS volume due to the replacement steam generators does not significantly impact the calculated post-LOCA sump boron concentration. Therefore, the Reference (93) analysis remains valid. This conclusion is easily verified for non-zero levels of tube plugging.

It may be demonstrated that expected levels of steam generator tube plugging will not invalidate the results of the Reference (93) recirculation switchover time calculation. For a change in average SGTP from 0% to 30% SGTP, the estimate of RCS volume used as input to the recirculation switchover time calculation is reduced from 9380.4 ft^3 (93) to approximately 7411 ft^3 (23). This represents approximately 2% of the total inventory modelled as accumulating in the containment sump following a large break LOCA. The volume-weighted average concentration of boric acid in this total inventory is approximately 1.36 wt%, with or without steam generator tube plugging. The volume of water in the core region is not affected by extended SGTP. Because (a) the 2% reduction in modelled sump water volume is partly offset by the corresponding reduction in boric acid inventory, (b) the reduction in total sump inventory is small, and (c) the calculational methodology for the recirculation switchover time is extremely conservative, it may be concluded that the results of the recirculation switchover time calculation are insignificantly affected by steam generator tube plugging levels up to 30%.

It is concluded that the Reference (93) recirculation switchover time analysis remains valid for North Anna Unit 1 following steam generator replacement. Furthermore, expected levels of steam generator tube plugging will not invalidate the results of the Reference (93) recirculation switchover time calculation.

2.2.28.2 Post-LOCA Shutdown Reactivity

Reference (94) documents the currently applicable North Anna post-LOCA shutdown reactivity analysis. The impact of steam generator replacement on this analysis and its associated reload safety analysis checklist parameters (52) must be evaluated to ensure that the reload core design is conservatively predicted to remain shutdown following a large break LOCA.

In the event of a large break LOCA, several sources are expected to provide water which will eventually end up in the containment sump. At North Anna, these include the Refueling Water Storage Tank (RWST), the Chemical Addition Tank (CAT), the Reactor Coolant System (RCS), the Safety Injection Accumulators (SIA), the Safety Injection System (SIS) piping, and the Boron Injection Tank (BIT). A volume weighted average boron concentration may be calculated by summing the product of the water volume and the boron concentration of each component and dividing by the sum of the volumes. (Uniform mixing is assumed.) The sump concentration is minimized by using minimum volumes for those components with a large boron concentration and maximum volumes for those components with little or no boron. On a reload basis, the critical boron concentration for the reload core at conditions expected in the post-LOCA containment sump is calculated and compared against the predicted sump concentration as a function of the minimum hot full power critical boron concentration.

The Reference (94) analysis assumed unplugged (0% SGTP) steam generators. The replacement steam generators each result in a modest 44

ft^3 increase in steam generator volume. In the unplugged condition, it may be concluded that there is essentially no difference between the volumes of the steam generators, and the Reference (94) analysis remains valid. For non-zero levels of tube plugging, the Reference (94) calculations remain valid because the reduction in RCS volume associated with SGTP is accompanied by a corresponding reduction in boric acid inventory; and because the RCS volume only represents a fraction of the inventory accumulated in the containment sump, the change in sump boron concentration resulting from the reduction in RCS volume is insignificant.

2.2.29 Containment Analysis in Support of Steam Generator Replacement

The existing containment analyses form the basis for the current Technical Specification limits on operating containment conditions (T.S. Figure 3.6-1). These analyses, documented in References (19),(20),(117),(95), were submitted for NRC review and approval by Reference (96). The approved Technical Specification changes were issued by NRC in Reference (97). The analyses included mass and energy releases for the large break LOCA, calculated by Stone and Webster with the LOCTIC code (Reference (98)), and mass and energy releases for the main steamline break accident, calculated by Westinghouse with the LOFTRAN code (Reference (99)). The current LOCA mass and energy analysis basis, described in UFSAR Section 6.2, involves use of the LOCTIC code to generate mass and energy releases for the blowdown and reflood portion of the transient, while Westinghouse-generated data is modified in conjunction with LOCTIC calculations for the later portions of the event. These mass and energy releases were used by SWEC in the LOCTIC code to analyze the containment transient behavior, for a number of limiting cases and initial containment conditions. The resulting Technical Specifications limits on initial containment temperature and pressure represent those conditions which, if they exist at the time of the limiting accidents analyzed, allow the analysis results to meet all applicable acceptance criteria.

The acceptance criteria are grouped into two categories: 1) those which are related to containment integrity (peak pressure, peak temperature, containment depressurization) and 2) those which are related to maintaining adequate operating conditions for key safeguards pumps (i.e., net positive suction head (NPSH) analyses). The following statements were taken from Reference (97), in which the NRC states conclusions concerning the acceptability of the existing analysis results.

From Section 2.1, LOCA Analysis

" Based on the LOCA analysis, which considered a spectrum of break sizes and locations and different single failure scenarios, the peak containment pressure was found to be 44.1 psig. The length of time to subatmospheric conditions was found to be 3310 seconds. The maximum subatmospheric peak was found to be -0.02 psig. The NPSHA showed 5.8 ft and 2.5 ft margin for the outside and inside recirculation spray pumps, respectively and 0.1 ft margin for the LHSI pumps. "

From Section 4.0, Summary

" Based on the above, the results indicate that the containment design criteria are not violated at the initial conditions of 120°F containment temperature, 97°F service water temperature and the lower volume of RWST water. Accident consequences are not increased by the proposed TS changes. Instrument uncertainties have been considered in the safety analyses which provided further evidence that accident consequences are not increased by these changes.

The results of the analyses show that none of the containment design bases are violated. That is, the following inequalities remain valid:

- Peak pressure: 44.1 psig (LOCA), 44.9 psig (MSLB) < 45 psig
- Depressurization to subatmospheric: 3310 seconds < 3600 sec
- Maintain subatmospheric pressure: -0.02 psig < 0.0 psig

Therefore, the containment design criteria specified in the introduction are: (1) peak pressure < 45 psig, (2) depressurize to subatmospheric < 3600 seconds, and (3) maintain pressure subatmospheric for duration of the accident are not violated and a TS change governing containment air temperature and minimum RWST volume is technically acceptable. "

From the above, it is concluded that the reference to specific analysis margins only summarizes key results and that the magnitude of these margins was not itself an integral part of the basis upon which NRC relied in issuing the Reference (97) approval. This interpretation is further supported by the SER summary section in which acceptability of the changes proposed in Reference (96) is stated in terms of meeting the key containment design criteria limits. Based on this interpretation, it is concluded that these basic regulatory limits form the maximum bounds of acceptable evaluation results when applied in the context of a 10 CFR 50.59 assessment of the North Anna Unit 1 steam generator replacement. The details of that assessment and associated analyses are presented in the following sections.

2.2.29.1 Introduction

Preliminary assessments were performed to determine the effect of steam generator replacement upon results of the LOCA and MSLB containment analyses. For the MSLB, it was concluded that there would be sufficient margin in the existing analysis assumptions to offset any detrimental effects of the replcement SGs. A primary consideration which further supported this expected conclusion was the proposed inclusion of flow restrictors in each SG outlet nozzle. This greatly limits the feasible break size, producing analysis margin with respect to operational limits assumed in the existing analysis. More detail concerning the evaluation of SG replacement effects upon MSLB containment analysis is presented in Section 2.2.29.4.

Scoping calculations were performed to determine the effects of SG replacement upon the existing LOCA containment integrity and NPSH analyses. The results (reported in Reference (109)), indicated that while there was margin to the acceptance criterion for each key parameter, certain analysis margins were decreased with respect to that quoted by the NRC in their approval of the existing analysis (Reference (97)). The relevant results are as follows:

		Reference (109) Result	Reference (97) Value
Containment Peak Pressure		44.41 psig	44.10 psig
Subatmospheric Peak Pressure		-0.01 psig	-0.02 psig
Depressurization Time		3330 sec	3310 sec
Calculated NPSH Margin -	LHSI pump	0.1 ft	0.1 ft
	IRS pump	2.3 ft	2.5 ft
	ORS pump	5.8 ft	5.8 ft

These results indicate that margins, versus those quoted in the Reference (97) approval, were reduced for each key parameter except the NPSH for the LHSI and ORS pumps. However, since each parameter meets the required regulatory limit, this only represents a reduction in analysis margin. The margin of safety as defined in 10 CFR 50.59 is unchanged. The interpretation presented above of the discussion in Reference (97) allows flexibility when implementing the steam generator replacement under the provisions of 10 CFR 50.59, since analysis results need only meet the explicit regulatory limits.

This approach was not possible for the peak pressure parameter, upon which the Technical Specifications for containment leak rate testing are based. North Anna Unit 1 T.S. 3.6.1.2 specifies that leak rate testing be performed with the containment at a pressure Pa, greater than or equal to 44.1 psig. This is the calculated peak pressure from the existing LOCA containment analysis. This value could not be increased without obtaining a Technical Specifications change, which was inconsistent with the overall SG replacement project objective of implementation via 10 CFR 50.59. In order to satisfy this major project objective, it was concluded that revised analysis results for peak containment pressure must preserve the margin as quoted in Reference (97).

As noted above, the existing LOCA containment analysis employs mass and energy releases calculated with the LOCTIC computer code. North Anna UFSAR Appendix 6A describes the LOCTIC model for calculating mass and energy releases during each phase of the LOCA transient (blowdown, reflood and post-reflood). Benchmark comparisons with mass and energy releases calculated using the Westinghouse mass and energy model (Reference (110)) are also presented. There is reasonable agreement in the calculated results between the SWEC and Westinghouse models. The expected impact of calculated differences between the models upon key containment parameters (e.g., pressure) is also discussed in UFSAR Appendix 6A.

It was concluded that the best means to confirm the acceptability of analysis margins for key containment analysis parameters, while implementing the SG replacement via the 10 CFR 50.59 process, was to use the Westinghouse mass and energy release model to reanalyze the LOCA containment releases. There are two versions of the Westinghouse model, documented in References (111) and (112). Each of these models has been reviewed and approved by NRC for use on PWRs with dry containment designs, such as North Anna. These models incorporate features which provide more explicit modelling of key phenomena for improved calculation of mass and energy releases throughout all phases of the LOCA accident. The reanalyses are summarized in the following two sections.

[illegible]

2.2.29.2 Large Break LOCA Mass and Energy Release Analysis

[illegible] performed the LOCA containment mass and energy [illegible] for use in the assessment of North Anna Unit 1 [illegible] replacement steam generators. These data were used by [illegible] the containment integrity and net positive suction [illegible] for the large break LOCA event. The Westinghouse [illegible] documented in Reference (III). A summary of the analyses performed [illegible] here.

The containment receives mass and energy releases following a [illegible] rupture of the RCS in the large break LOCA event. [illegible] purposes, the LOCA transient is divided into four phases:

1. Blowdown - includes the period from accident initiation until the time that the RCS pressure reaches initial equilibrium with containment.

2. [illegible] - the period of time when the lower plenum is being [illegible] accumulator and safety injection water. For the mass and energy release analysis, any water remaining in the [illegible] and downcomer at the end of blowdown is assumed to be instantaneously transferred to the lower plenum. This allows a continuous release of mass and energy, conservatively ignoring the refill period.

3. Reflood - time period beginning when lower plenum water rises into the core and ending when the core is quenched.

4. Post-reflood (Froth) - the remaining period following reflood. For the limiting pump suction break, this is characterized by a two-phase mixture which rises into the steam generator tubes where it is superheated prior to release from the break. [illegible] becomes twophase after the broken loop steam generator [illegible]

The analyses were performed using two versions of the Westinghouse mass and energy release model. Analyses of the pump suction double-ended rupture (PSDER) used the March 1979 model described in Reference (112). The model documented in Reference (111) was used in the analysis of the hot leg double-ended rupture (HLDER) cases, for post-blowdown releases only. The Reference (111) model was used since it has better capabilities for modelling the long term releases which were required for the HLDER event. The Reference (111) model remains a valid analytical tool which has been reviewed and approved by the NRC, although it does not contain the improved features of the Reference (112) model (i.e., steam/SI mixing during reflood and 1979 decay heat). These features of the Reference (112) model reduce the calculated energy releases to containment.

Generic studies have been performed to determine the effect of break size and location upon the LOCA mass and energy releases. It has been determined that the double-ended guillotine break is most limiting because of larger mass flow rates during the blowdown phase of the transient. Break size has little effect upon releases during the reflood and post-reflood phases.

Three distinct locations in the RCS loop piping can be postulated for pipe rupture:

1. Hot Leg - between reactor vessel and steam generator
2. Cold Leg - between reactor coolant pump and reactor vessel
3. Pump Suction - between steam generator and reactor coolant pump

Previous generic and plant-specific analyses have determined that analyzing the pump suction and hot leg double-ended guillotine breaks will produce limiting conditions for the containment integrity and net positive suction head analyses. When using the Reference (111) model, the PSDER has typically exhibited the peak calculated containment pressure for North Anna, which occurs during the reflood phase. PSDER analysis performed with mass and energy data from the Reference (112) model has no such reflood peak pressure. This behavior is caused by the pressure-reducing effect of mixing between safety injection and steam in the reactor coolant loops which is modelled during reflood in the Reference (112) model. The HLDER case typically has the next most limiting peak containment pressure, which occurs during the blowdown transient phase. The HLDER thus becomes the limiting break for peak containment pressure. These conclusions have been confirmed for operation with the replacement steam generators. The breaks analyzed for steam generator replacement are the hot leg double-ended rupture, HLDER (9.17 ft^2 area) and the pump suction double-ended rupture, PSDER (10.48 ft^2 area). For each case, break releases have been calculated for the blowdown, reflood and post-reflood phases of the LOCA.

In addition to break size and location, the effects of various single failures have been evaluated. An inherent assumption in the mass and energy analyses is that offsite power is lost. This results in actuation of the emergency diesel generator to provide power to the various safeguards components. Two cases of mass and energy release are analyzed to incorporate single failure effects. A minimum safeguards case is analyzed, to model the postulated failure of one emergency diesel generator, which minimizes safety injection flow. For the maximum safeguards case, no failure is postulated to occur. This maximizes safety injection flow delivered to the core. In this manner, the mass and energy release analysis bounds the effects of all credible single failures.

Various conservative assumptions concerning plant-specific parameters were employed to ensure that the calculated releases maximize the energy release to containment. These items are summarized below.

- Maximum expected RCS operating temperature + 4.0°F uncertainty
- 3% RCS volume increase to account for expansion and uncertainty
- Maximum licensed core power + 2% uncertainty
- Conservative primary-secondary, RCS metal heat transfer coefficients
- Allowance in core stored energy for effect of fuel densification
- Core stored energy increased 15% for margin
- RCS operating pressure + 36 psi uncertainty
- 0% Steam Generator tube plugging (bounds non-zero asymmetric plugging for containment energy release effects)
- Conservative containment backpressure, equal to design value

For calculating the blowdown mass and energy releases, the SATAN-VI code is used. This is the same version of the code as that used in the 1981 ECCS evaluation model (91).

The WREFLOOD code is used for calculating mass and energy releases during the reflood transient. The version used is modified from that used in the Reference (91) ECCS evaluation model. Reference (112) describes the methodology for use of this model. The model assumes complete mixing (i.e., thermal equilibrium) of the safety injection water and steam in the RCS loops. The amount and processes modelled in the mixing have been validated with comparisons to test data.

The FROTH code (111) is used for calculating the mass and energy releases during the post-reflood phase of the LOCA transient. Reference (112) presents the methodology for use of this model. The mass and energy release rates, and other specific interface data, are used in conjunction with calculations performed by the Stone and Webster LOCTIC code until the time of containment depressurization. This interface data consists of the following items, which are used in the containment response calculation as described in Section 2.2.29.3.

Post-Reflood Interface Data (at time of SG equilibration)

- SG Equilibration Time
- SG Pressure
- SG Energy Content
- SG Heat Transfer Rate

The process of steam generator equilibration involves the removal of energy from the steam generators in stages. At the beginning of the post-reflood period, the reactor coolant system is assumed to be at the containment design pressure of 45 psig. The steam generators are defined to be in equilibrium with containment when they have depressurized to the containment pressure. For this analysis, a conservative containment pressure transient is used in the determination of equilibration pressure and time for both the broken loop and intact loop steam generators. This containment pressure response was based upon a LOCTIC calculation for a pump suction double-ended rupture with greater mass and energy release than that of the final analysis case. This approach produces a conservative selection of pressure to establish equilibration. Using this interface pressure to define the equilibration pressure, the transient calculation of steam generator equilibration is performed as described in Reference (112).

After SG equilibration, the mass and energy releases are primarily influenced by core decay heat. The decay heat energy used is based upon the 1979 ANSI/ANS Standard (114), with two sigma uncertainty applied, along with other conservative inputs which maximize the energy release.

The sources of mass and energy considered in the analysis include all sources within the reactor coolant system, accumulators and pumped safety injection, in accordance with guidelines presented in Section 6.2.1.3 of the Standard Review Plan. No zirc-water reaction heat was considered in the analysis since the clad temperature did not increase sufficiently for this heat source to be significant.

The analysis techniques described above were employed to perform the mass and energy release analysis. Indicated below are the base cases which constitute the steam generator replacement licensing basis analysis. These data yielded limiting results for containment response and NPSH analyses which confirmed the limiting break location and safeguards equipment configuration for each of the key parameters reported in the analysis results. These key parameters are:

depressurization time, subatmospheric peak pressure, peak pressure and NPSH (for LHSI, ORS and IRS pumps). Sensitivity results confirmed that the following combinations were limiting for the indicated parameters.

BASE CASES - SGR ANALYSIS OF RECORD	LIMITING PARAMETER
PSDER-min SI (4150 gpm LH/610 gpm HH)	Depress. Time, Subatm. Peak, LHSI pump NPSH
HLDER-min SI (3500 gpm LH/553 gpm HH)	Peak Pressure
HLDER-max SI (6000 gpm LH/850 gpm HH)	IRS, ORS pump NPSH

For each case, mass and energy releases were generated for each phase of the transient. The detailed results may be found in Reference (113). Section 2.2.29.3 describes the use of these data in performing the containment response and NPSH analyses.

2.2.29.3 Large Break LOCA Containment Integrity and NPSH Analysis

The mass and energy release data described in the previous section is used in performing the containment response analyses with the Stone and Webster LOCTIC (98) computer code. In accordance with previous sensitivities for limiting break size and location (20), these analyses were performed for the pump suction double-ended guillotine rupture (PSDER) and hot leg double-ended guillotine rupture (HLDER) breaks. Sensitivity cases were performed which confirmed the limiting break and single failure assumptions that were established in the existing Reference (20) analysis.

The LOCA analysis consists of a peak pressure analysis, a depressurization analysis and an analysis for the recirculation spray and low head safety injection (LHSI) pumps which confirms that adequate NPSH is available. As discussed in Section 2.2.29.2, the limiting break analyzed for peak containment pressure is the HLDER. The depressurization analysis was performed for the limiting break, which is a double-ended rupture at the reactor coolant pump suction (PSDER) with minimum engineered safety features (ESF). Similarly, the PSDER with minimum engineered safety features (ESF) gives the limiting NPSH for the low head safety injection pumps. For recirculation spray pump NPSH, the HLDER with maximum ESF produces the limiting conditions. The mass and energy data from each of these scenarios was analyzed with the LOCTIC computer code and the results are summarized herein.

The containment analyses employed the same assumptions concerning loss of offsite power and single failure as in the existing analysis (20). The loss of offsite power is assumed to occur simultaneously with the start of the accident. The single failures analyzed include a diesel generator failure (i.e. minimum ESF) and a quench spray pump failure.

2.2.29.3.1 Peak Pressure Analysis

A peak pressure analysis was performed to ensure that the existing peak calculated containment pressure of 44.10 psig in T.S. 3.6.1.2 was not exceeded for operation with the replacement steam generators. The maximum temperature is also compared against the design containment temperature in this analysis. As described in Section 2.2.29.2, use of the Reference

(112) mass and energy release model results in the HLDER break having the peak calculated containment pressure.

For the HLDER break case, the Reference (113) values of Westinghouse mass and energy release rates were used directly in LOCTIC with no modification. The results confirmed that the peak pressure occurs during blowdown, and decreases afterwards with no secondary pressure peaks produced.

The summary of key results from the depressurization analysis is presented in Table 2.2.29-3 As shown on the table, the peak calculated containment pressure from the revised analysis of 43.94 psig is less than the existing Reference (20) analysis value of 44.10 psig listed in T.S. 3.6.1.2. This result provides confirmation that adequate analysis margin exists to support the 10 CFR 50.59 assessment of operation with replacement steam generators.

2.2.29.3.2 Depressurization Analysis

The depressurization analysis is performed to show that the containment can conservatively be returned to subatmospheric conditions within one hour and remain subatmospheric thereafter. The limiting transient for depressurization is the PSDER and the limiting single failure is the loss of one emergency diesel-generator which results in the failure of one train of ESF to actuate (i.e. minimum ESF). However, the initial conditions must be specified differently for the depressurization analysis to conservatively determine the peak subatmospheric pressure. The Reference (20) sensitivity studies have confirmed the conservative containment initial conditions for depressurization analysis; these values were used in the present analysis.

The chronology for the limiting pressure transient is presented in Table 2.2.29-2. This table illustrates the significant aspects of the pressure response to a LOCA when using the Reference (112) mass and energy data in the LOCTIC code. There are two pressure peaks. The first peak occurs as a result of the blowdown from the accident. The pressure continues to decrease, in response to the containment and recirculation spray systems, and the condensation of steam within the coolant loops which is calculated by the Reference (112) model. The second pressure peak follows both LHSI pump recirculation transfer and termination of quench spray flow upon emptying emptying the RWST, after which time the containment pressure increases and approaches atmospheric for the last time.

The summary of key results from the depressurization analysis is presented in Table 2.2.29-3. As noted on the table, results of the revised analysis meet the applicable regulatory limits.

2.2.29.3.3 Recirculation Spray Pump NPSH Analysis

The LOCTIC computer code was used to calculate the net positive suction head available (NPSHA) for the inside and outside recirculation spray pumps. The NPSH analysis is performed to make certain that the NPSHA exceeds that required (NPSHR) for the flow rate assumed throughout the analysis.

The assumptions made for the depressurization analysis maximize the energy release to the containment atmosphere (minimize energy release to the sump) in order to overestimate the containment pressure. The assumptions made for NPSHA analyses of the recirculation spray pumps minimize the energy release to the containment atmosphere and maximize the energy release to the containment floor. Thus, the containment pressure is underestimated and the containment floor water vapor pressure is overestimated. Since containment pressure is a positive term in the NPSHA equation and the floor water vapor pressure is a negative term, a conservative calculation of NPSHA results.

For the revised analysis, these assumptions were implemented by use of the Westinghouse mass and energy data (113) in a LOCTIC analysis which employs the pressure flash modelling described in UFSAR Table 6.2-47. This modelling essentially assumes that the steam and liquid components of the break effluent are perfectly mixed, and that the liquid component becomes saturated at the containment pressure before falling to the containment sump. In this manner, the energy contained in the sump water is maximized, which is conservative for NPSH calculations. The Westinghouse data (113) were used for a sufficient transient duration (1500 seconds) to confirm that the limiting values of recirculation spray pump NPSH have occurred.

NPSHA sensitivity studies were performed for the original FSAR analysis. These studies were performed to determine the limiting case for NPSHA and included various break locations, service water temperatures, RWST temperatures, break sizes and initial conditions. These studies were not redone for the present analysis since the changes being modelled are relatively small and do not affect the sensitivities.

Table 2.2.29-3 summarizes the key results from the recirculation spray pump NPSH analysis. As shown on the table, the NPSHA for both the IRS and ORS pumps exceeds the minimum requirements.

2.2.29.3.4 Low Head Safety Injection Pump NPSH Analysis

An analysis of the NPSHA for the low head safety injection pumps has been performed to ensure that the pumps can deliver up to 4030 gpm following the worst case LOCA. The injection mode and the recirculation mode have been evaluated previously and it was found that the recirculation mode is limiting. The calculation of NPSHA in the recirculation mode considers the static head and suction line pressure drop, the vapor pressure of the liquid in the sump and the containment pressure. This calculation ensures that the NPSHA meets the pump requirements.

The calculation of NPSHA is as follows:

$$NPSHA = h_{con\ press} - h_{vap} + h_{stat} - h_{loss}$$

where

$h_{con\ press}$ = containment pressure, ft
h_{vap} = vapor pressure of the fluid in the sump, ft
h_{stat} = static head of the fluid in the sump, ft
h_{loss} = suction line losses, ft

The previous sensitivity studies (20) have not been reanalyzed for operation with the replacement steam generators because the changes from existing analyses are slight and therefore do not invalidate the sensitivities. These sensitivities show that the PSDER with minimum ESF is the worst case. Maximizing the sump water temperature minimizes the NPSH available. This is done by minimizing containment heat removed and maximizing the sump and RWST water temperatures.

As discussed in Section 2.2.29.3.3, the LHSI pump NPSH analysis also employed the Westinghouse mass and energy data (113) in a LOCTIC analysis which assumes the pressure flash modelling. The PSDER with minimum safeguards is the limiting case for LHSI NPSH. In the revised analysis, the interface with the mass and energy data is somewhat different since Westinghouse data is being used versus LOCTIC data as in the existing analysis. This change is briefly described here.

The Westinghouse Reference (112) model data is used for the blowdown, reflood, and a portion of the post-reflood phases of the transient analysis. This interval extends from the time of the break to the time at which the intact loop steam generators are in equilibrium with containment. The Westinghouse data includes the interface information at time of steam generator equilibrium (as listed in Section 2.2.29.2), which is used in LOCTIC to calculate energy removal in the depressurization phase of the transient. This calculation is essentially the same as that described in UFSAR Section 6.2.1.1.1.2, in which LOCTIC removes sufficient energy from the RCS and steam generators to maintain equilibrium with conditions in containment. This allows use of the improved Westinghouse modelling of blowdown and reflood energy releases, while incorporating the effects of containment conditions upon the releases in the longer-term depressurization phase of the transient.

To confirm the existance of adequate analysis margin in available NPSH for the LHSI pumps, it was necessary to reduce the RWST setpoints for LHSI recirculation transfer. The setpoints for both manual and automatic recirculation transfer were reduced by 5.5% from their existing values. This reduction was implemented in conjunction with a revised assumption for duration of the automatic LHSI recirculation transfer sequence. The RWST volume changes and reduced automatic transfer duration are documented in Reference (120). The total sequence duration was selected to be 240 seconds, which has been confirmed to exceed the summation of individual stroke time limits for the valves involved in the transfer sequence, as contained in Periodic Test procedure 1-PT-213.8, "Valve Inservice Inspection - Safety Injection System, Rev. 4." No change to 1-PT-213.8 is required since the stroke times in the PT are more restrictive (smaller) than the revised analysis values. This is acceptable, since the procedure requires that any valve which cannot meet its acceptance criterion be declared inoperable, and the required Technical Specification actions taken.

The net effect of these changes upon RWST analysis inputs was a slight benefit in total available RWST volume assumed at the completion of the automatic recirculation transfer. Table 2.2.29-1 presents the input values associated with the RWST volume and drawdown which were altered from their values in the Reference (95) analysis.

Reference (121) documents the effect upon NPSH analyses of insulation debris within the steam generator cubicle which is postulated to become dislodged following a LOCA event. The effects have been quantified in terms of penalties on NPSH for the LHSI, IRS and ORS pumps. The results from Reference (115) presented within this report include the effects of these penalties.

The documentation changes required to reflect the proposed setpoint values discussed in this section are provided in Appendix B.

Table 2.2.29-1

Summary of RWST Volume Inputs
For Operation With Replacement Steam Generators
North Anna Unit 1

RWST Parameter	Analysis (95)	Analysis (115)
Technical Specification Minimum Volume, gal	466,200	466,200
Safety Analysis Minimum Volume, gal	452,327	452,327
Total RWST Volume at Manual LHSI Recirculation Transfer Setpoint, gal	151,033 (28.3% span)	124,908 (22.8% span)
Total RWST Volume at Completion of Manual Recirculation Transfer (125 sec), gal	137,075	110,950
Total RWST Volume at Auto LHSI Recirculation Transfer Setpoint, gal	134,852 (24.9% span)	108,727 (19.4% span)
Total RWST Volume at Completion of Auto Recirculation Transfer (240 sec), gal	81,252	81,927

Table 2.2.29-2

Chronology for PSDER LOCA
For Operation With Replacement Steam Generators
North Anna Unit 1

Time (sec)	Event
0.0	Accident occurs
3.5	Containment depressurization actuation signal (30 psia)
18.4	First containment peak pressure occurs
20.8	End of blowdown; core reflooding begins
20.8	Safety injection pumps become effective
42.0	Accumulators empty
62.5	Quench spray subsystem and casing cooling effective
253.4	Core reflooding ends; post-reflood frothing begins
304.0	Recirculation spray system becomes effective
1595.8	Post-reflood frothing ends
3370.0	Containment pressure becomes subatmospheric
3460.0	Safety injection pumps switch to recirculation mode
5350.0	Quench spray flow stops, RWST is empty
6340.0	Subatmospheric peak containment pressure occurs

Table 2.2.29-3

Summary of Key Containment Analysis Results
For Operation With Replacement Steam Generators
North Anna Unit 1

Analysis Parameter	Existing Analysis (95)	SG Replacement Analysis (115)	Regulatory Limit
Peak Pressure, psig	44.10	43.94	45.00 (1)
Peak Temperature, °F	271.7	272.3	280.0
Depressurization Time, sec	3310.00	3370.00	3600.0
Subatmospheric Peak Pressure, psig	-0.02	-0.04	0.0
LHSI Pump NPSH Available, ft	13.5	14.1	13.1
IRS Pump NPSH Available, ft	11.9	10.4	9.4
ORS Pump NPSH Available, ft	16.8	15.3	11.0

Notes:

(1) Limit for implementing SGR via 10 CFR 50.59 (see Section 2.2.29.1)

2.2.29.4 Main Steam line Break Containment Analysis

The existing main steam line break (MSLB) containment analysis was included in the Technical Specification change package of Reference (96). The analysis consisted of mass and energy releases for a wide range of postulated break sizes and initial reactor power levels. Westinghouse performed the analysis using the LOFTRAN computer code (99). Containment transient response was calculated for each case by SWEC with the LOCTIC code (98). The impact upon these analyses of operation with replacement steam generators was assessed in Reference (100). That evaluation is summarized here.

The relevant inputs to the Reference (96) MSLB containment analyses were reviewed and compared to those same parameters for the replacement steam generators. Two parameters, inclusion of the steam outlet nozzle integral flow restrictor and the lower secondary side masses at power conditions, are expected to have an impact on the containment peak pressure and temperature analysis. The inclusion of the steam outlet nozzle integral flow restrictors significantly limits the feasible break size. The reduced break size limits the rate at which the mass and energy is removed from the secondary side of the steam generators. This, in turn, limits the maximum expected containment pressure and temperature. The analyses supporting Reference (96), documented in Reference (101), included a sensitivity to the maximum break size. The reduction of the break area to the area of the steam generator outlet nozzle flow restrictor showed a significant reduction in both the containment peak pressure and peak temperature.

The initial mass on the secondary side is lower for the replacement steam generators than it is for the current steam generators. This will be a benefit for the containment analysis since a lower initial mass cannot remove as much energy from the steam generator secondary. If this were accounted for in the containment analyses, it would lower the peak pressure and temperature slightly.

The replacement steam generators have a beneficial impact on the calc[illegible] of the containment peak pressures and temperatures. The smal[illegible] the steam nozzle integral flow restrictors limits the ra[illegible] mass and energy enter the containment. This has been shown to be a benefit in previous sensitivity studies In addition, there is a small reduction in the secondary steam generator mass that, although unquantified, provides an additional margin. The other parameters important to containment analysis for the limiting cases remain unchanged. Therefore, the current containment analyses for the main steam line break remain bounding.

2.2.30 LTOPS Mass Addition and Heat Addition Transients

Transient analysis calculations were performed (102) to determine the maximum pressure overshoot over the low temperature overpressure protection system (LTOPS) setpoints in the event of a design basis low temperature overpressurization event. These calculations included a sensitivity evalution to determine the impact of a reduction in RCS volume due to steam generator tube plugging. The calculations revealed that overpressurization becomes slightly more limiting with increased steam generator tube plugging. As such, steam generator replacement provides an analysis benefit since it results in an increase in RCS volume, and a decrease in LTOPS setpoint pressure overshoot.

The Reference (102) analysis conservatively assessed LTOPS setpoint pressure overshoot for SGTP levels from 0% to 40%. Furthermore, the Reference (3) analysis demonstrates that the Model 51F steam generators may be considered a replacement component for the Model 51 steam generators. As such, the Reference (102) analysis remains valid for North Anna 1 operation with replacement steam generators.

Because the North Anna Unit 1 heatup and cooldown curves and LTOPS setpoints expire at a burnup occurring shortly after steam generator replacement, it is expected that the proposed revised heatup and cooldown curves and LTOPS setpoints based on the Reference (102) analysis will be approved by the NRC prior to steam generator replacement.

2.3 DETAILED CORE THERMAL/HYDRAULICS ASSESSMENT

The purpose of this section is to outline the impact of steam generator replacement on the detailed core thermal/hydraulics (T/H) analyses and methods for North Anna Unit 1.

A North Anna implementation analysis (13) of the Virginia Power Statistical DNBR Evaluation Methodology (14) was submitted to the NRC in 1987 (16) and approved in 1989 (15). This analysis methodology replaced the Westinghouse Improved Thermal Design Procedure (ITDP). In the Statistical DNBR Evaluation Methodology, a series of nominal T/H statepoints are randomly varied to account for the impact of key T/H analysis parameter uncertainties. Each random statepoint is subjected to a detailed core T/H analysis to determine the statepoint's minimum departure from nucleate boiling ratio (DNBR). The DNBR results are subjected to a statistical evaluation to determine the DNBR standard deviation associated with each nominal statepoint resulting from the random variation of T/H analysis parameters. The standard deviations are then correlated as a function of a system parameter (e.g., RCS temperature or pressure) to permit the maximization of the standard deviation over the entire range of operating parameters. The nominal statepoints included several statepoints at both thermal design flow and at low flow conditions.

The North Anna Statistical DNBR Evaluation Methodology implementation analysis (13) revealed that 87% of the random variation in the DNBR standard deviation is described by a linear function of core inlet temperature that is maximized at low inlet temperatures. A low temperature statepoint based on an assessment of the postulated cooldown events was chosen to maximize the DNBR standard deviation.

The design average RCS temperature at full power for North Anna Unit 1 will remain between 580.8°F and 586.8°F following steam generator replacement as allowed by the Reference (13) Statistical DNBR Evaluation Methodology implementation analysis and the safety evaluation for reduced temperature operation (7). Because the implementation analysis (13) conservatively maximized the DNBR standard deviation by demonstrating that it is a strong function of inlet temperature, and because the RCS flow rate associated with steam generator replacement is expected to be well above the Technical Specification minimum measured RCS flow rate requirement (284,000 gpm), the Reference (13) implementation analysis remains valid following steam generator replacement. Ensuring that transient T/H analysis DNBR's remain above the statistical DNBR limit will continue to ensure that the onset of DNB will be avoided at a 95% probability/95% confidence level.

An evaluation of the applicability of existing T/H analysis methods and transient T/H analyses is performed for each reload core. An example of this evaluation is found in Reference (103). The following sections describe the relevant evaluations considered in the reload thermal/hydraulics evaluation as they relate to steam generator replacement.

2.3.1 RELOAD T/H EVALUATION CONSIDERATIONS

2.3.1.1 Single Rod Withdrawal

As described in Section 2.2.17, the results of the Single Rod Withdrawal at Power event analysis are evaluated on a reload basis in the reload thermal/hydraulics evaluation calculation. For each reload core, it is verified that less than 5% of the fuel rods have radial peaking factors in excess of 1.55.

The single rod withdrawal at power DNBR verification described above remains valid provided the Technical Specification minimum measured RCS flow rate (total RCS flow rate ≥284,000 gpm) and average RCS temperature requirements are met. Because this requirement is expected to be met with margin, steam generator replacement presents no safety considerations for this transient evaluation.

2.3.1.2 Statically Misaligned Rod

The peak radial power factor, FΔH, resulting from a rod out of position is calculated for each reload core. The reload T/H evaluation presents an allowable FΔH below which any calculated reload peak radial power factor will not result in a minimum DNBR below the design limit minimum DNBR. The allowable FΔH has been calculated for the design thermal hydraulic conditions which will bound Unit 1 operation following steam generator replacement.

A dropped bank FΔH limit is not evaluated for North Anna, since North Anna is a negative flux rate trip plant. The dropped bank FΔH parameter is only relevant for turbine runback plants.

The statically misaligned rod verification described above remains valid provided the Technical Specification minimum measured RCS flow rate (total RCS flow rate ≥284,000 gpm) and average RCS temperature requirements are met. Because these requirements are expected to be met with margin, steam generator replacement presents no safety considerations for this accident evaluation.

2.3.1.3 Bypass Flow Rate

The current assessment of the bypass flow rate presented in the reload thermal/hydraulics evaluation remains valid for North Anna operation following steam generator replacement. Steam generator replacement and its associated effect on total loop resistance and RCS flow rate will have an insignificant effect on the fraction of total vessel flow assumed unavailable for core heat removal.

2.3.1.4 Core Thermal Limits

The Technical Specifications present core thermal limits, overtemperature and overpower ΔT trip setpoints, and an F(ΔI) function applicable to a thermal design flow rate of 284,000 gpm. These parameters remain valid provided the Technical Specification minimum measured RCS flow rate requirement (total RCS flow rate ≥284,000 gpm) is met. Because this requirement is expected to be met with margin, steam generator replacement presents no safety considerations for these parameters.

The Technical Specifications also present core thermal limits, overtemperature and overpower ΔT trip setpoints, and an $F(\Delta I)$ function generated assuming a minimum measured flow rate of 275,300 gpm. These parameters will automatically expire following steam generator replacement; as such, they present no considerations for operation following steam generator replacement.

2.3.1.5 Axial Shape Verification

The Relaxed Power Distribution Control (RPDC) methodology (104) permits plant operation with a wide range of allowable axial power skewing by testing potential axial power profiles against the criteria of the DNBR limit and fuel centerline melt. In the case of the former, shapes may be tested against the axial offset envelope to verify DNBR protection for overtemperature ΔT events, and as Loss of Flow Accident (LOFA) preconditions to verify protection for all non-OTΔT events; in addition, the shapes are tested for fuel centerline melt by calculating their peak linear powers, and comparing those values against the linear power limit.

There are no direct effects of steam generator replacement on axial shape verification. Steam generator replacement indirectly affects the total RCS flow rate; this affects the coolant temperature rise across the core and, in turn, the degree of axial power shape skewing. These effects are explicitly modelled in the reload core design process.

The axial shape methodology described below remains valid provided the Technical Specification minimum measured RCS flow rate (total RCS flow rate ≥284,000 gpm) and average RCS temperature requirements are met. Because these requirements are expected to be met with margin, steam generator replacement presents no safety considerations for thermal/hydraulic axial shape verification.

2.3.1.5.1 Loss of Flow Analysis

The condition I (normal operation) axial power profiles are evaluated as potential Loss of Flow Accident (LOFA) preconditions. The LOFA is the limiting non-OTΔT DNB event; the OTΔT events are examined as described in Section 2.4.1.5.4.

For the loss of flow shape evaluation, each potential Condition I axial power profile is a potential LOFA precondition. Since an acceptable LOFA transient DNBR was obtained in the licensing basis analysis with a 1.55-cosine axial power profile, it is necessary to show that predicted reload RPDC shapes are conservatively bounded by the 1.55-cosine as a LOFA precondition. A comparison calculation is performed at the LOFA checkpoint to simulate the LOFA DNBR shape sensitivity at the time of minimum DNBR.

2.3.1.5.2 Locked Rotor Analysis

It is also necessary to consider the impact of axial power shapes which may be experienced by the reload core on locked rotor analysis results. Past analyses have shown that the shape DNBR sensitivity is the same at the LOFA checkpoint as it is at the locked rotor statepoint. In other words, when the 1.55-cosine shape is bounding the LOFA, it is generally found to be bounding for the locked rotor. Therefore, the LOFA shape verification results are typically utilized as a surrogate for an explicit locked rotor shape verification.

2.3.1.5.3 Overpower Calculations

Overpower calculations are performed for each RPDC axial power profile, both condition I and II, by converting the maximum FQ, which is non-dimensional, into a dimensional linear power, which has units of kW/ft. This linear power is then compared to the reload safety analysis checklist (RSAC) linear power limit, which is based on the linear heat rate required to meld the fuel at its centerline. The calculations include corrections for uncertainty, xenon distribution, grid locations, fuel densification, gamma heating, and fuel stack height reduction.

2.3.1.5.4 Axial Offset Envelope Verification

Section 2.3.1.5.1 discussed the DNBR evaluation of the RPDC shapes for the limiting non-OTΔT event, which is the LOFA. A separate evaluation is performed for the OTΔT class of DNBR events. The verification of the AO envelope ensures that adequate DNB protection is provided by the OTΔT function for all of the potential Condition I and II axial power profiles generated in the core design analysis.

The OTΔT setpoint is an allowable core temperature rise as a function of T_{avg} and pressure. Upon reaching the setpoint, which is based upon DNBR core thermal limits, a plant trip occurs, thus providing appropriate DNB protection. The limit lines are based upon a 1.55 cosine axial power profile. If the profile skews sharply upward or downward, however, the DNBR effect may be more severe than that of the 1.55 cosine. To account for this possibility, a penalty upon allowable ΔT, as a function of power asymmetry is assessed. The penalty is called f(ΔI), and takes the form of a percentage penalty on allowable ΔT. The f(ΔI) function is developed from the axial offset envelope, which is a boundary of constant pressure, flow, inlet temperature and DNBR. The present methodology prescribes that it be developed from the 1.55 cosine axial power shape and several sharply skewed power profiles.

2.3.1.6 Available Retained DNBR Margin and Penalties

The Statistical DNBR Evaluation Methodology (14) implementation analysis for North Anna Unit 1 (16),(15),(13) established a statistical DNBR limit (SDL) of 1.26. The difference between this value and the WRB-1 CHF correlation (45),(46) limit of 1.17 represents the impact of a combining key DNBR analysis parameter uncertainties in a statistical manner. Transient analyses are performed against a 1.46 design DNBR limit. The percentage difference between the design DNBR limit and the SDL represents generic retained margin, against which penalties may be assessed to account for the DNB effect of changes in plant operating conditions or analysis methodology. Analysis DNB margin is the percentage difference between the minimum transient analysis DNBR and the design DNBR limit. Although it is atypical, this margin has been occasionally utilized to facilitate the justification of changes in plant operating conditions.

The assessment of penalties against generic retained margin is typically accomplished through the use of a DNBR partial derivative. To facilitate the evaluation of proposed T/H methodology or operating parameter changes, the maximum DNBR partial derivative for the parameter in question is calculated by examining the marginal effect of changes in the parameter on the calculated DNBR over a range of operating conditions. The product of the percentage change in the parameter in question, and the bounding DNBR partial derivative with respect to the parameter, provides a penalty which may be assessed against available retained margin to account for the effect of the parameter change.

Each reload thermal/hydraulics evaluation consolidates and summarizes DNBR penalties applicable to the reload core in question. The penalty which has been applied to account for the reduction in minimum measured RCS flow rate to 275,300 gpm may be removed following steam generator replacement since the Technical Specification will revert to 284,000 gpm. Other penalties must be considered on a case-by-case basis in the reload thermal/hydraulics evaluation to determine if they remain applicable following steam generator replacement.

2.3.1.7 Main Steamline Break

A main steamline break generic thermal/hydraulics analysis has been performed and documented in Reference (105). This analysis allows the verification of MSLB transient thermal/hydraulics results by the performance of a screening check. The Reference (105) generic thermal/hydraulics analysis remains valid provided the Technical Specification minimum measured RCS flow rate (total RCS flow rate ≥284,000 gpm) and average RCS temperature requirements are met. Because this requirement is expected to be met with margin, the Reference (105) generic main steamline break analysis remains valid for North Anna 1 operation following steam generator replacement.

The discussion in Section 2.2.22 may be consulted for further information on the main steamline break transient analysis.

3.0 ADDITIONAL SAFETY AND OPERATIONAL CONSIDERATIONS

The preceding section has documented the impact of operation with replacement steam generators upon the NSSS and containment safety analyses. Steam generator replacement creates specific physical plant changes which may have additional safety-related effects on operations. Further, the analysis effects discussed previously have the potential for changing certain operational documentation and procedures. These additional safety and operational considerations which are within NSA scope are addressed below. These considerations were:

1. Confirmation of Reactor Protection System (RPS) setpoints, Emergency Safety Features Actuation System (ESFAS) setpoints, and Technical Specifications values.
2. Assessment of increased RCS flow rate
3. Evaluation of Steam Generator Replacement impact on Emergency Operating Procedures and boration/dilution nomographs.
4. Recommended changes to Periodic Test Procedures, Nuclear Control Room Operator Development Program (NCRODP) and other documents.

Each of these considerations is evaluated in the following sections. Appendix B presents the changes to items listed above for operation with replacement steam generators as a result of the analyses addressed in this report.

3.1 Confirmation of RPS and ESFAS Setpoints, and Technical Specifications Values

A review of the North Anna Unit 1 Technical Specifications and ESFAS setpoint documentation revealed that no direct T.S. changes are required to support operation following steam generator replacement. As noted in

Section 2.1, all T.S. changes made to accommodate North Anna 1 extended SGTP will revert to the values in place prior to the extended SGTP analysis effort. It should be verified that the following setpoints are appropriately reflected in North Anna Unit 1 hardware.

North Anna 1 Technical Specifications Figure 2.1-1, Table 2.2-1, and Table 3.2-1 were previously modified to reflect reactor protection system setpoints (OTΔT, low flux, and high flux reactor trips) and reactor coolant system flow rates applicable under conditions of extended SGTP (8),(9). Turbine runback and rod withdrawal block setpoints were reset by subtracting .03 from the proposed K1 and K4 values. These specifications will revert to the values based on the previous design conditions (284,000 gpm) following steam generator replacement. North Anna 1 reactor protection system setpoints and minimum measured RCS flow rate should be revised in plant instrumentation and procedures to be consistent with the values in the Technical Specifications. The large break LOCA analysis no longer requires two high head safety injection pumps to be operable when one low head safety injection pump is out of service. Appendix B presents the associated T.S. pages which were issued with Unit 1 license amendments 153 and 154 for extended SG tube plugging operation. The temporary changes for each affected T.S. have a note which indicates that the change is effective 'for the period of operation until steam generator replacement.'

The Tavg program and associated setpoints should be revised to be consistent with the desired full power operating Tavg between 586.8°F and 580.8°F. Reference (118) recommends that Unit 1 operate at an RCS Tavg but will revert to the prior analysis basis. "10 CFR 50.59" is an indication that the appropriate accident analysis needs to be incorporated into the North Anna analysis basis via the current steam generator replacement safety evaluation. These revisions to analysis basis will be reflected in the North Anna 1 Cycle 10 reload evaluation process for operation following steam generator replacement.

1. Accidental Depressurization of the RCS: Current Basis
2. Accidental Depressurization of the Main Steam System (Credible Steamline Break): Current Basis
3. Complete Loss of Flow: Current Basis (ext. SGTP analysis)
4. CVCS Malfunction (Boron Dilution): Prior Basis
5. Excessive Load Increase: Current Basis
6. Feedwater Malfunction: Current Basis
7. Loss of External Load: Current Basis (ext. SGTP analysis)
8. Loss of Normal Feedwater: Current Basis (ext. SGTP analysis)
9. Misaligned Control Rod/Bank and Dropped Rod: Current Basis
10. Partial Loss of Flow: Current Basis
11. Rod Bank Withdrawal at Power: Current Basis (ext. SGTP analysis)
12. Rod Bank Withdrawal from Subcritical: Current Basis
13. Startup of an Inactive Reactor Coolant Loop: Current Basis
14. Spurious Operation of the Safety Injection System: Current Basis
15. Minor Secondary Steam Pipe Breaks: Current Basis
16. Misloaded Fuel Assembly: Current Basis
17. Single Rod Withdrawal at Power: Current Basis
18. Volume Control Tank Rupture: Current Basis
19. Waste Gas Decay Tank Rupture: Current Basis
20. Fuel Handling Accident Outside Containment: Current Basis
21. Fuel Handling Accident Inside Containment: Current Basis
22. Major Secondary System Pipe Ruptures (Main Steamline Break): Current Basis
23. Rupture of a Main Feedwater Pipe (Main Feedline Break): Current Basis

24. Control Rod Ejection: Current Basis

25. Steam Generator Tube Rupture: Current Basis

26. Locked Rotor and Sheared Shaft: Current Basis

27. Small Break LOCA: Current Basis

28. Large Break LOCA: Prior Basis (Analysis reverts to the 15% SGTP North Anna Large Break LOCA reanalysis, which is currently applicable to North Anna Unit 2 and was previously applicable to North Anna Unit 1.)

29. Containment Analysis: 10 CFR 50.59

30. LTOPS Mass Addition and Heat Addition Transients: Current Basis

The replacement steam generators have been evaluated for their impact on radiological dose calculations for the main steamline break, steam generator tube rupture, and the RCP locked rotor transients. In each case, the existing dose calculations remain bounding for North Anna 1 operation with the replacement steam generators.

In keeping with standard practice, each reload core is evaluated to verify that thermal/hydraulic design criteria are not exceeded. The reload thermal/hydraulics evaluation considers each of the following:

1. Single Rod Withdrawal: This Condition III accident is evaluated in the reload T/H evaluation to verify that less than 5% of the fuel rods would experience departure from nucleate boiling (DNB) during this transient.

2. Statically Misaligned Rod: The reload T/H evaluation presents a comparison of the maximum radial power factor, $F\Delta H$, resulting from a rod out of position to the $F\Delta H$ which results in DNB at design operating conditions.

3. Bypass Flow Rate: The T/H evaluation presents an assessment of core bypass flow to verify that the assumed bypass flow rate remains bounding for the reload core design.

4. Core Thermal Limits: Each reload T/H evaluation verifies that the core thermal limits presented in the Technical Specifications remain applicable to the proposed core design.

5. Axial Shape Verification: Predicted skewed axial power profiles for each reload core are evaluated to ensure that they would not cause thermal/hydraulic criteria to be exceeded at steady state conditions or during any non-LOCA transient.

6. Available Retained DNBR Margin and Penalties: The reload T/H evaluation summarizes available retained DNBR margin and penalties applicable to the proposed reload core design. Retained DNBR margin is used to account for generic or cycle specific issues affecting the thermal/hydraulic design of the core.

7. Main Steamline Break: The main steamline break accident is assessed to ensure that the DNBR criterion continues to be met for the proposed reload core design.

In addition, the effects of SGR on post-LOCA recirculation switchover time, post-LOCA shutdown reactivity, and low temperature overpressure protection system setpoints (heat and mass addition transients) have been evaluated and determined not to be adversely impacted by SGR.

4.0 CONTAINMENT ANALYSIS IN SUPPORT OF STEAM GENERATOR REPLACEMENT

The existing containment analyses form the basis for the current Technical Specification limits on operating containment r nditions (T.S. Figure 3.6-1). The analyses (described in UFSAR Section 6.2) included mass and energy releases for the large break LOCA, calculated by Stone and Webster with the LOCTIC code, and mass and energy releases for the main steamline break accident, calculated by Westinghouse with the LOFTRAN code. These mass and energy releases were used in LOCTIC to analyze the containment transient behavior for a number of limiting cases and initial containment conditions. The resulting Technical Specifications limits on initial containment temperature and pressure represent those conditions which, if they exist at the time of the limiting accidents analyzed, allow the analysis results to meet all applicable acceptance criteria.

The acceptance criteria are grouped into two categories: 1) those which ar: related to containment integrity (peak pressure, peak temperature, containment depressurization) and 2) those which are related to maintaining adequate operating conditions for key safeguards pumps (i.e., net positive suction head (NPSH) analyses). The following statements were taken from the NRC Safety Evaluation Report for the containment analyses, in which the NRC states conclusions concerning the acceptability of the existing analysis results.

From Section 4.0, Summary

" The results of the analyses show that none of the containment design bases are violated. That is, the following inequalities remain valid:

- Peak pressure: 44.1 psig (LOCA), 44.9 psig (MSLB) < 45 psig
- Depressurization to subatmospheric: 3310 seconds < 3600 sec
- Maintain subatmospheric pressure: -0.02 psig < 0.0 psig "

These basic containment design acceptance criteria and the related NPSH requirements form the bounds of acceptable evaluation results for the evaluation of replacement steam generators. Operation with the replacement steam generators can be implemented via the 10 CFR 50.59 process if reanalysis results are within these limits. For peak calculated containment pressure, the acceptance limit is 44.1 psig, since Technical Specification 3.6.1.2 specifies that leak rate testing be performed with the containment at a pressure greater than or equal to this value.

It was concluded that the best means to confirm the acceptability of analysis margins for key containment analysis parameters, while implementing the SG replacement via the 10 CFR 50.59 process, was to use the NRC-approved Westinghouse mass and energy release model to reanalyze the LOCA containment releases. The Westinghouse model incorporates features which provide more explicit modelling of key phenomena for improved calculation of mass and energy releases throughout all phases of the LOCA transient. The sources of mass and energy considered in the analysis include all sources within the reactor coolant system, accumulators and pumped safety injection, in accordance with guidelines presented in Section 6.2.1.3 of the Standard Review Plan.

The mass and energy data were generated for break locations and assumed failures which were confirmed to be limiting in the existing UFSAR analyses. Mass and energy releases were calculated for the pump suction double-ended guillotine rupture (PSDER) and hot leg double-ended guillotine rupture (HLDER) breaks.

The containment response analysis consists of a peak pressure analysis, a depressurization analysis and an analysis for the recirculation spray and low head safety injection (LHSI) pumps which confirms that adequate NPSH is available. The limiting break for peak pressure is the HLDER. The PSDER with minimum engineered safety features is the limiting break for depressurization analysis and for the LHSI pumps NPSH analysis. For inside recirculation (IRS) and outside recirculation (ORS) pump NPSH analyses, the HLDER with maximum ESF produces the limiting conditions. The mass and energy data from each of these scenarios was analyzed with the LOCTIC computer code.

The peak calculated containment pressure from the revised analysis is 43.94 psig, which is less than the existing analysis value of 44.10 psig in the UFSAR analysis. This result provides confirmation that adequate analysis margin exists to support the 10 CFR 50.59 assessment of operation with replacement steam generators.

The depressurization analysis is performed to show that the containment can conservatively be returned to subatmospheric conditions within one hour and remain subtamospheric thereafter. The revised depressurization analysis results meet all applicable regulatory limits, i.e., the containment is depressurized in less than 1 hour and conditions remain subatmospheric thereafter.

The analysis of available NPSH was performed using the limiting break scenarios for the LHSI pumps, the IRS pumps and ORS pumps. The Westinghouse mass and energy data was used in conjunction with LOCTIC calculations to provide the energy releases and containment conditions for use in the analysis.

To obtain acceptable margin in available NPSH for the LHSI pumps, it was necessary to reduce the RWST setpoints for LHSI recirculation transfer. The setpoint for both manual and automatic recirculation transfer were reduced by 5.5% from their existing values. This provided increased NPSH for the LHSI pumps, and was implemented in conjunction with a revised assumption for duration of the automatic LHSI recirculation transfer sequence. The net effect was a slight benefit in total available RWST volume assumed at the completion of the automatic recirculation transfer.

The revised analysis results indicate that available NPSH exceeds the minimum required for the LHSI, IRS and ORS pumps throughout the LOCA transient. These results are conservative for all allowed combinations of operating conditions within current Technical Specification limits.

4.1 Main Steam Line Break Containment Analysis

The existing main steam line break (MSLB) containment analysis included mass and energy releases for a wide range of postulated break sizes and initial reactor power levels. Westinghouse performed the analysis using the LOFTRAN computer code. Containment transient response was calculated for each case by Stone and Webster with the LOCTIC code. The impact upon these analyses of operation with replacement steam generators has been assessed and is summarized here.

The inclusion of the steam outlet nozzle integral flow restrictors significantly limits the feasible break size, which limits the rate at which the mass and energy is removed from the secondary side of the steam generators, and reduces the maximum calculated containment pressure and temperature. Sensitivity analyses confirm that break area equal to the area of the steam generator outlet nozzle flow restrictor significantly reduces both the containment peak pressure and peak temperature.

The initial mass on the secondary side is less for the replacement steam generators than that assumed in the existing analyses. This will be a benefit for the containment analysis since a lower initial mass cannot remove as much energy from the steam generator secondary. If this were accounted for in the containment analyses, it would lower the peak pressure and temperature slightly.

The replacement steam generators have a beneficial impact on the calculation of the containment peak pressures and temperatures following a main steamline break. The smaller break area of the steam nozzle integral flow restrictors limits the rate at which mass and energy enter the containment. This has been shown to be a benefit in previous sensitivity studies. In addition, there is a small reduction in the secondary steam generator mass that, although unquantified, provides an additional margin. The other parameters important to containment analysis for the limiting cases remain unchanged. Therefore, the current containment analyses for the main steam line break remain bounding.

5.0 ADDITIONAL SAFETY AND OPERATIONAL CONSIDERATIONS

The preceding section has documented the impact of operation with replacement steam generators upon t[illegible] [illegible]SS and containment safety analyses. Steam generator replacement [illegible]ates specific physical plant changes which may have additional safety-related effects on operations. Further, the analysis effects discussed previously have the potential for changing certain operational documentation and procedures. These additional safety and operational considerations which are within NSA scope are addressed below. These considerations were:

1. Confirmation of Reactor Protection System (RPS) setpoints, Emergency Safety Features Actuation System (ESFAS) setpoints, and Technical Specifications values.
2. Assessment of increased RCS flow rate
3. Evaluation of Steam Generator Replacement impact on Emergency Operating Procedures and boration/dilution nomographs.
4. Recommended changes to Periodic Test Procedures, Nuclear Control Room Operator Development Program (NCRODP) and other documents.

Reference (1) has documented changes to each of the above items for analyses evaluated within the scope of that report. These considerations are summarized in the following sections.

5.1 Confirmation of RPS and ESFAS Setpoints, and Technical Specifications Values

A review of the North Anna Unit 1 Technical Specifications and ESFAS setpoint documentation revealed that no direct T. S. changes are required to support operation following steam generator replacement. As noted in Section 2.0, all T. S. changes made to accommodate North Anna 1 Cycle 9 extended SGTP will revert to the values in place prior to the extended SGTP analysis effort. It should be verified that the following setpoints are appropriately reflected in North Anna Unit 1 hardware.

North Anna 1 Technical Specifications Figure 2.1-1, Table 2.2-1, and Table 3.2-1 were previously modified to reflect reactor protection system ... nts (OTΔT, low flow, and high flux reactor trips) and reactor coolant system flow rates applicable under conditions of extended SGTP following North Anna 1 Cycle 9 restart. Turbine runback and rod withdrawal block setpoints were reset by subtracting .03 from the proposed K1 and K4 values. These Unit 1 Technical Sp fications will revert to the values consistent with design conditions (e.g., 284,000 gpm) following steam generator replacement. North Anna 1 reactor protection system setpoints and minimum measured RCS flow rate must be verified in plant hardware and procedures to be consistent with the Technical Specifications.

The Tavg program and associated setpoints should be revised to be consistent with the desired full power operating Tavg between 586.8°F and 580.8°F. Reference (1) documents a recommendation that Unit 1 operate at an RCS Tavg of 580.8°F following steam generator replacement. This equals the Unit 2 value.

The large break LOCA analysis no longer requires two high head safety injection pumps to be operable when one low head safety injection pump is out of service.

The containment reanalysis has assumed a reduction in the RWST setpoints for LHSI recirculation transfer. The setpoint for both manual and automatic recirculation transfer were reduced by 5.5% from their existing values. The revised setpoints are:

LHSI Recirculation Auto Transfer Setpoint = 19.4% span

LHSI Recirculation Manual Transfer Setpoint = 22.8% span

The UFSAR, Emergency Operating Procedure, and NCRODP changes necessary to reflect these revised setpoints are provided in Reference (1).

6.0 CONCLUSIONS

The Reference (1) Technical Report presented all UFSAR Chapter 15 accident analyses applicable to North Anna 1 operation following steam generator replacement. The appropriate analysis for each accident has been identified.

In consideration of UFSAR Chapter 15 accidents, it has been determined that steam generator replacement does not involve an unreviewed safety question as defined in 10 CFR 50.59. Specifically:

1. The probability of occurrence or the consequences of any accidents or malfunction of equipment important to safety previously evaluated in the safety analysis report is not increased by the steam generator replacement. Applicable design constraints were analyzed and none were found to be more limiting than those currently documented in the UFSAR, provided North Anna Unit 1 is operated in accordance with the Unit 1 Technical Specifications that will be applicable following steam generator replacement.

2. The possibility of an accident or malfunction of equipment of a different type than previously evaluated in the safety analysis report is not created. Steam generator replacement does not involve any alterations to the physical plant which would introduce any new or unique operational modes or accident precursors. With respect to accident analyses, the Model 51F steam generators may be considered a "replacement" component for the existing Model 51 steam generators. As such, their performance does not create conditions that would lead

to an accident or equipment malfunction beyond the scope of those already evaluated in the safety analysis report.

3. The margin of safety as defined in the basis for any Technical Specification is not reduced by the proposed change. The replacement steam generators have been demonstrated to insignificantly affect the transient system response during postulated UFSAR Chapter 15 accidents. Accident analyses for all UFSAR Chapter 15 transients have been performed which bound allowable operation in accordance with the North Anna 1 Technical Specifications that will be applicable following steam generator replacement. All accident analyses meet their respective acceptance criteria. It may, therefore, be concluded that steam generator replacement does not decrease the margin of safety as defined in the basis for any Technical Specification.

No Technical Specifications changes are required to support North Anna Unit 1 operation following steam generator replacement. The Technical Specification changes made to support North Anna 1 operation with extended steam generator tube plugging will automatically revert to values applicable prior to the extended SGTP effort. As described in Section 4.1, North Anna 1 reactor protection system and turbine runback setpoints, and the minimum measured RCS flow rate should be revised in plant hardware and procedures to be consistent with the values in the Technical Specifications and in supporting accident analyses. In addition, plant setpoints and EOP setpoints to reflect the reduced LHSI recirculation transfer setpoint should be revised in accordance with the design information in Reference (1).

These revisions to analysis basis will be reflected in the North Anna 1 Cycle 10 reload evaluation process for operation following steam generator replacement.

REFERENCES

(1) J. R. Harrell: "Safety Analyses and Evaluations Supporting North Anna 1 Operation Following Steam Generator Replacement," Technical Report NE-883, Revision 1, dated June, 1992.

Appendix B

Summary of Procedure and Setpoint Changes Associated with Accident Analyses

Technical Spe-ification Pages
from Amendment Numbers 153 and 154

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555

VIRGINIA ELECTRIC AND POWER COMPANY

OLD DOMINION ELECTRIC COOPERATIVE

DOCKET NO. 50-338

NORTH ANNA POWER STATION, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 153
License No. NPF-4

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Virginia Electric and Power Company, et al., (the licensee) dated January 28, 1992, as supplemented February 27, 1992, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, license condition 2.D.(1) to Facility Operating License NPF-4 is modified to read as follows:**

   (1) Maximum Power Level

   VEPCO is authorized to operate the North Anna Power Station, Unit No. 1, at reactor core power levels not in excess of 2893 megawatts (thermal).*

   *The maximum reactor power level shall be limited to 2748 megawatts (thermal) which is 95% of RATED THERMAL POWER in accordance with the licensee's submittal dated January 28, 1992 (Serial No. 92-042) for the period of operation until the steam generator replacement.

3. In addition, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.D.(2) of Facility Operating License No. NPF-4 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 153, are hereby incorporated in the license. VEPCO shall operate the facility in accordance with the Technical Specifications.

4. This license amendment is effective as of its date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Herbert N. Berkow

Herbert N. Berkow, Director
Project Directorate II-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachments:
1. Page 4 of License
2. Changes to the Technical Specifications

Date of Issuance:

**Page 4 is attached, for convenience, for the composite license to reflect this change.

(1) Maximum Power Level

VEPCO is authorized to operate the North Anna Power Station, Unit No. 1, at reactor core power levels not in excess of 2893 megawatts (thermal).*

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 153, are hereby incorporated in the license. VEPCO shall operate the facility in accordance with the Technical Specifications.

(3) Additional Conditions

The matters specified in the following conditions shall be completed to the satisfaction of the Commission within the stated time periods following the issuance of this amendment or within the operational restrictions indicated. The removal of these conditions shall be made by an amendment to the license supported by a favorable evaluation by the Commission:

c. Virginia Electric and Power Company shall not operate the reactor in operational modes 1 and 2 with less than three reactor coolant pumps in operation.

d. VEPCO may use two (2) fuel assemblies containing fuel rods clad with an advanced zirconium base alloy cladding material as described in the licensee's submittals dated February 20, 1987 and September 30, 1988.

e. If Virginia Electric and Power Company plans to remove or to make significant changes in the normal operation of equipment that controls the amount of radioactivity in effluents from the North Anna Station, the Commission shall be notified in writing regardless of whether the change affects the amount of radioactivity in the effluents.

---

*The maximum reactor power level shall be limited to 2748 megawatts (thermal) which is 95% of RATED THERMAL POWER in accordance with the licensee's submittal dated January 28, 1992 (Serial No. 92-042) for the period of operation until the steam generator replacement.

ATTACHMENT TO LICENSE AMENDMENT NO. 153

TO FACILITY OPERATING LICENSE NO. NPF-4

DOCKET NO. 50-338

Replace the following page of the Appendix "A" Technical Specifications with the enclosed page as indicated. The revised page is identified by amendment number and contains vertical lines indicating the area of change. The corresponding overleaf page is also provided to maintain document completeness.

Remove Page	Insert Page
3/4 5-3	3/4 5-3

EMERGENCY CORE COOLING SYSTEMS

ECCS SUBSYSTEMS - $T_{avg} \geq 350°F$

LIMITING CONDITION FOR OPERATION

3.5.2 Two independent ECCS subsystems shall be OPERABLE with each subsystem comprised of:

a. One OPERABLE centrifugal charging pump.

b. One OPERABLE low head safety injection pump.

c. An OPERABLE flow path capable of transferring fluid to the Reactor Coolant System when taking suction from the refueling water storage tank on a safety injection signal or from the containment sump when suction is transferred during the recirculation phase of operation or from the discharge of the outside recirculation spray pump.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

a. With one ECCS subsystem inoperable, restore the inoperable subsystem to OPERABLE status within 72 hours or be in HOT SHUTDOWN within the next 12 hours. *

b. In the event the ECCS is actuated and injects water into the Reactor Coolant System, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 90 days describing the circumstances of the actuation and the total accumulated actuation cycles to date.

c. The provisions of Specifications 3.0.4 are not applicable to 3.5.2.a and 3.5.2.b for one hour following heatup above 324°F or prior to cooldown below 324°F.

---

* Adherence to ACTION "a" shall require the following equipment OPERABILITY for the period of operation until steam generator replacement:

- With one low head safety injection pump inoperable, two centrifugal charging pumps (one in each subsystem) and their associated flow paths shall be OPERABLE or be in HOT STANDBY within the next 6 hours, and be in HOT SHUTDOWN within the next 6 hours.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555

VIRGINIA ELECTRIC AND POWER COMPANY

OLD DOMINION ELECTRIC COOPERATIVE

DOCKET NO. 50-338

NORTH ANNA POWER STATION, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 154
License No. NPF-4

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Virginia Electric and Power Company et al., (the licensee) dated January 8, 1992, as supplemented by letters dated January 31, February 10 and February 25, 1992, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.D.(2) of Facility Operating License No. NPF-4 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 154 , are hereby incorporated in the license. VEPCO shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Herbert N. Berkow

Herbert N. Berkow, Director
Project Directorate II-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: March 3, 1992

ATTACHMENT TO LICENSE AMENDMENT NO. 154

TO FACILITY OPERATING LICENSE NO. NPF-4

DOCKET NO. 50-338

Replace the following pages of the Appendix "A" Technical Specifications with the enclosed pages as indicated. The revised pages are identified by amendment number and contain vertical lines indicating the area of change. The corresponding overleaf pages are also provided to maintain document completeness.

Remove Pages	Insert Pages
2-1	2-1
- -	2-2a
2-6	2-6
2-9	2-9
2-10	2-10
3/4 2-15	3/4 2-15

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS

REACTOR CORE

2.1.1 The combination of THERMAL POWER, pressurizer pressure, and the highest operating loop coolant temperature (T_{avg}) shall not exceed the limits shown in Figures 2.1-1* for 3 loop operation and 2.1-2 and 2.1-3 for 2 loop operation.

APPLICABILITY: MODES 1 and 2.

ACTION:

Whenever the point defined by the cnmbination of the highest operating loop average temperature and THERMAL POWER has exceeded the appropriate pressurizer pressure line, be in HOT STANDBY within 1 hour.

REACTOR COOLANT SYSTEM PRESSURE

2.1.2 The Reactor Coolant System pressure shall not exceed 2735 psig.

APPLICABILITY: MODES 1, 2, 3, 4 and 5.

ACTION:

MODES 1 and 2

Whenever the Reactor Coolant System pressure has exceeded 2735 psig, be in HOT STANDBY with the Reactor Coolant System pressure within its limit within 1 hour.

MODES 3, 4 and 5

Whenever the Reactor Coolant System pressure has exceeded 2735 psig, reduce the Reactor Coolant System pressure to within its limit within 5 minutes.

---

* For the period of operation until steam generator replacement, the combination of THERMAL POWER, pressurizer pressure, and the highest operating loop coolant temperature (T_{avg}) shall not exceed the limits shown in Figure 2.1-1a.

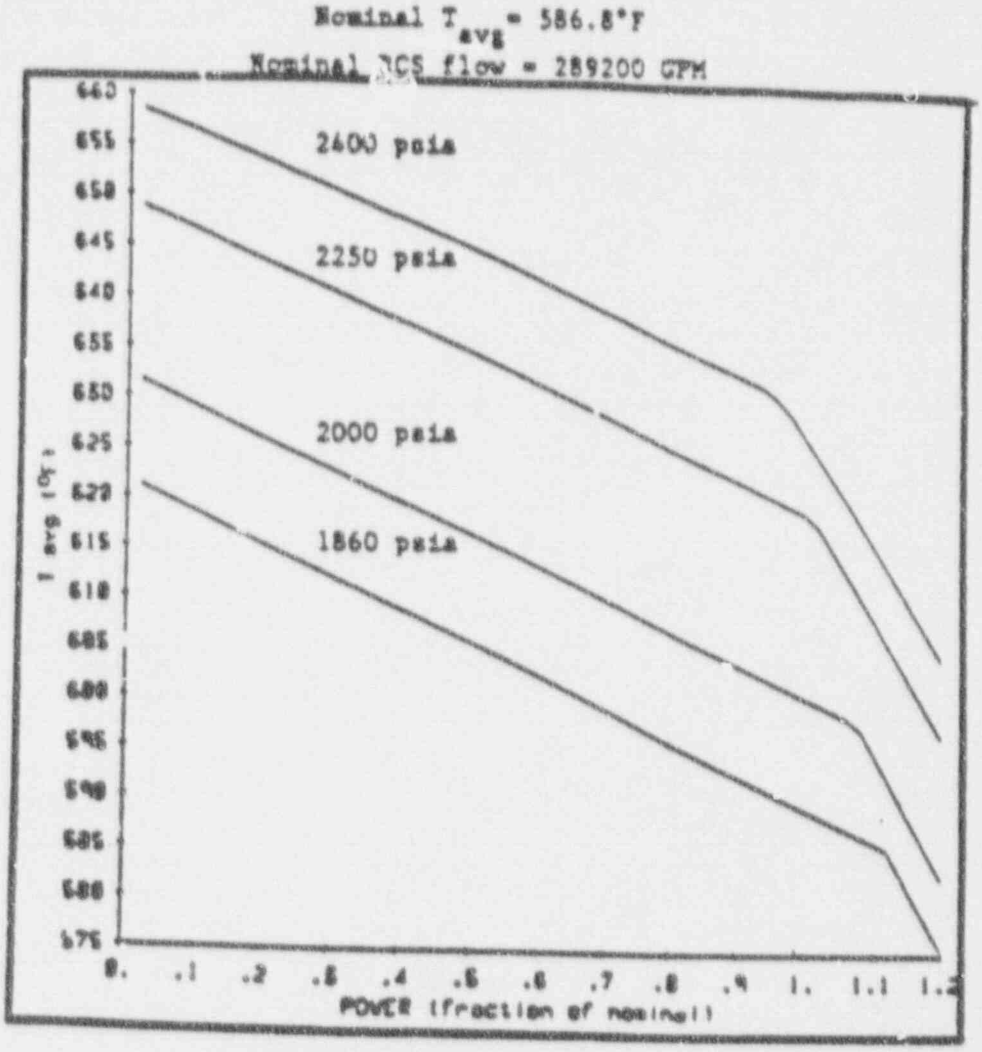


Figure 2.1-1 REACTOR CORE SAFETY LIMITS FOR THREE LOOP OPERATION

Nominal T_{avg} = 586.8°F

Nominal RCS flow = 268,500 GPM

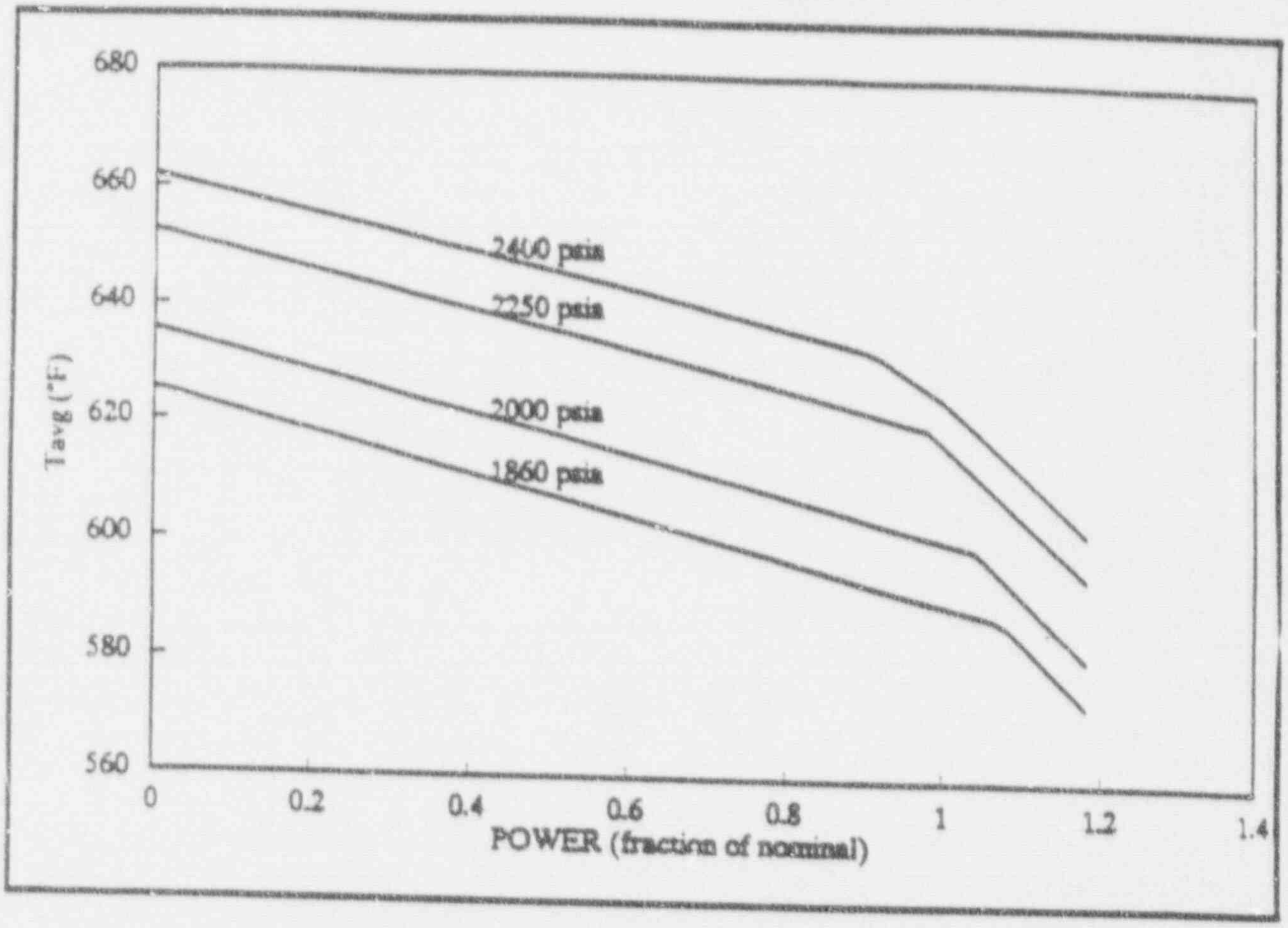


Figure 2.1-1a REACTOR CORE SAFETY LIMITS FOR THREE LOOP OPERATION FOR THE PERIOD OF OPERATION UNTIL STEAM GENERATOR REPLACEMENT

TABLE 2.2-1

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
1. Manual Reactor Trip	Not Applicable	Not Applicable
2. Power Range, Neutron Flux	Low Setpoint - $\leq$ 25% of RATED THERMAL POWER	Low Setpoint - $\leq$ 26% of RATED THERMAL POWER
	High Setpoint - $\leq$ 109%** of RATED THERMAL POWER	High Setpoint - $\leq$ 110%*** of RATED THERMAL POWER
3. Power Range, Neutron Flux, High Positive Rate	$\leq$ 5% of RATED THERMAL POWER with a time constant $\geq$ 2 seconds	$\leq$ 5.5% of RATED THERMAL POWER with a time constant $\geq$ 2 seconds
4. Power Range, Neutron Flux, High Negative Rate	$\leq$ 5% of RATED THERMAL POWER with a time constant $\geq$ 2 seconds	$\leq$ 5.5% of RATED THERMAL POWER with a time constant $\geq$ 2 seconds
5. Intermediate Range, Neutron Flux	$\leq$ 25% of RATED THERMAL POWER	$\leq$ 30% of RATED THERMAL POWER
6. Source Range, Neutron Flux	$\leq 10^5$ counts per second	$\leq 1.3 \times 10^5$ counts per second
7. Overtemperature ΔT	See Note 1	See Note 3
8. Overpower ΔT	See Note 2	See Note 3
9. Pressurizer Pressure--Low	$\geq$ 1870 psig	$\geq$ 1860 psig
10. Pressurizer Pressure--High	$\leq$ 2385 psig	$\leq$ 2395 psig
11. Pressurizer Water Level--High	$\leq$ 92% of instrument span	$\leq$ 93% of instrument span
12. Loss of Flow	$\geq$ 90% of design flow per loop *	$\geq$ 89% of design flow per loop *

* Design flow per loop is one-third of the minimum allowable Reactor Coolant System Total Flow Rate as specified in Table 3.2-1.

** The high trip setpoint for Power Range, Neutron Flux, shall be $\leq$ 103% RATED THERMAL POWER for the period of operation until steam generator replacement.

*** The allowable value for the high trip setpoint for Power Range, Neutron Flux, is required to be $\leq$ 104 RATED THERMAL POWER for the period of operation until steam generator replacement.

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

NOTATION (Continued)

Operation with 3 Loops	Operation with 2 Loops (no loops isolated)*	Operation with 2 Loops (1 loop isolated)*
K_1 = 1.264**	K_1 = ( )	K_1 = ( )
K_2 = 0.0220	K_2 = ( )	K_2 = ( )
K_3 = 0.001152	K_3 = ( )	K_3 = ( )

and $f_1(\Delta I)$ is a function of the indicated difference between top and bottom detectors of the power-range nuclear ion chambers; with gains to be selected based on measured instrument response during plant startup tests such that:

(i) for $q_t - q_b$ between -44 percent and +3 percent, $f_1(\Delta I) = 0$ (where q_t and q_b are percent RATED THERMAL POWER in the top and bottom halves of the core respectively, and $q_t + q_b$ is total THERMAL POWER in percent of RATED THERMAL POWER).

(ii) for each percent that the magnitude of $(q_t - q_b)$ exceeds -44 percent, the ΔT trip setpoint shall be automatically reduced by 1.67 percent of its value at RATED THERMAL POWER.

(iii) for each percent that the magnitude of $(q_t - q_b)$ exceeds +3 percent, the ΔT trip setpoint shall be automatically reduced by 2.00 percent of its value at RATED THERMAL POWER.

* Values dependent on NRC approval of ECCS evaluation for these operating conditions.

** The value for K_1 shall be equal to 1.132 for the period of operation until steam generator replacement.

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

NOTATION (Continued)

Note 2: Overpower $\Delta T \leq \Delta T_o \left[K_4 - K_5 \left(\frac{\tau_3 S}{1+\tau_3 S} \right) T - K_6 (T - T') - f_2(\Delta I) \right]$

Where:

ΔT_o	=	Indicated ΔT at RATED THERMAL POWER
T	=	Average temperature, °F
T'	=	Indicated T_{avg} at RATED THERMAL POWER ≤ 586.8°F
K_4	=	1.079*
K_5	=	0.02/°F for increasing average temperature
K_5	=	0 for decreasing average temperatures
K_6	=	0.00164 for $T > T'$; $K_6 = 0$ for $T \leq T'$
$\frac{\tau_3 S}{1+\tau_3 S}$	=	The function generated by the rate lag controller for T_{avg} dynamic compensation
τ_3	=	Time constant utilized in the rate lag controller for T_{avg} τ_3 = 10 secs.
S	=	Laplace transform operator (sec^{-1})
$f_2(\Delta I)$	=	0 for all ΔI

Note 3: The channel's maximum trip point shall not exceed its computed trip point by more than 2 percent span.

* The value for K_4 shall be equal to 1.016 for the period of operation until steam generator replacement.

TABLE 3.2-1

DNB PARAMETERS

PARAMETER	LIMITS 3 Loops in Operation	2 Loops in Operation ** & Loop Stop Valves Open	2 Loops in Operation ** & Isolated Loop Stop Valves Closed
Reactor Coolant System T_{avg}	≤ 591°F		
Pressurizer Pressure	≥ 2205 psig *		
Reactor Coolant System Total Flow Rate	≥ 284,000 gpm ***		

* Limit not applicable during either a THERMAL POWER ramp increase in excess of 5% RATED THERMAL POWER per minute or a THERMAL POWER step increase in excess of 10% RATED THERMAL POWER.

** Values dependent on NRC approval of ECCS evaluation for these conditions.

*** The value for the minimum allowable Reactor Coolant System Total Flow Rate is reduced to 268,500 gpm until steam generator replacement.

EOP Basis Document Changes

Number: 92-011

E O P B A S I S D O C U M E N T C H A N G E

TO: NORTH ANNA PROCEDURES
Attn: J. Daily

Wednesday , June 10 , 1992

FROM: J. O. Erb

Innsbrook Technical Center

Change to EOP Setpoint Document

The subject document page(s) I.1, I.3, I.4, I.5, I.6, AND I.7 were changed.

These changes revise setpoints I.1, I.3, I.4, and I.5 to make them consistent with Calculation EE-0093. Setpoints I.6 and I.7 are newly created setpoints that will need to be in effect following the SGRP on NAPS Unit 1.

No.	Old Value	New Value
I.1	26%	25% *
I.3	Graph	Revised Graph
I.4	29%	29% **
I.5	52%	46%
I.6	N/A	20%
I.7	N/A	23%

* This setpoint is not currently used in the NAPS EOPs

** The setpoint basis changed, not affecting the final setpoint value

The need for the new setpoints was determined during preparatory work for the NAPS Unit 1 SG replacement. During this work it was noted that the existing setpoints I.1, I.3, I.4, and I.5 were out of date and inconsistent with Calculation EE-0093.

Engineering Programs Effect: None, no new parameters being added to EOP instrumentation.

Implementation schedule: Incorporate in a timely manner. Changes are not urgent.

Attachmen to this form provide additional information. If you have any questions regarding this, please call.

Joseph O. Erb
J. O. Erb

Prepared by: W. M. Oppenheimer

Reviewed by: N. A. Smith

cc: L. C. Kidd -- Elec Eng, IN2W
J. G. Voissem -- NAPS Procedures
NSA File 24.1.1 -- IN3SW
Records Management Project File -- INGSW (Original)

North Anna

EOP Setpoint No: I.1

Revision: 06/10/92

Parameter: RWST LEVEL

Setpoint Value: 25%

Applicability: UNIT 1 & UNIT 2

Associated System/Component: RWST/ECCS

Description:

RWST level setpoint for transfer to containment sump recirculation. This value is used to verify that the automatic actions to realign the suction of the ECCS pumps from the RWST to the containment sump have been initiated.

Display Data:

Instrument	Units	Range	Division	Type	Location
LI-QS-200A	%	0-100%	2	METER	E2-CB-05
LI-QS-200B	%	0-100	2	METER	E2-CB-05
LI-QS-200C	FEET	0-11	4 INCHES	METER	E2-CB-05
LI-QS-200D	FEET	0-11	4 INCHES	METER	E2-CB-05
LI-QS-100A	%	0-100	2	METER	E1-CB-05
LI-QS-100B	%	0-100	2	METER	E1-CB-05
LI-QS-100C	%	0-100	2	METER	E1-CB-05
LI-QS-100D	%	0-100	2	METER	E1-CB-05

Key Assumptions:

None

Basis:

Reference 1 describes the ECCS design at North Anna where switchover to sump recirculation is manually initiated at 28.3% level and automatic initiation is started at 24.9% level.

The automatic switchover setpoint is rounded to 25% for use in the North Anna EOPs.

References:

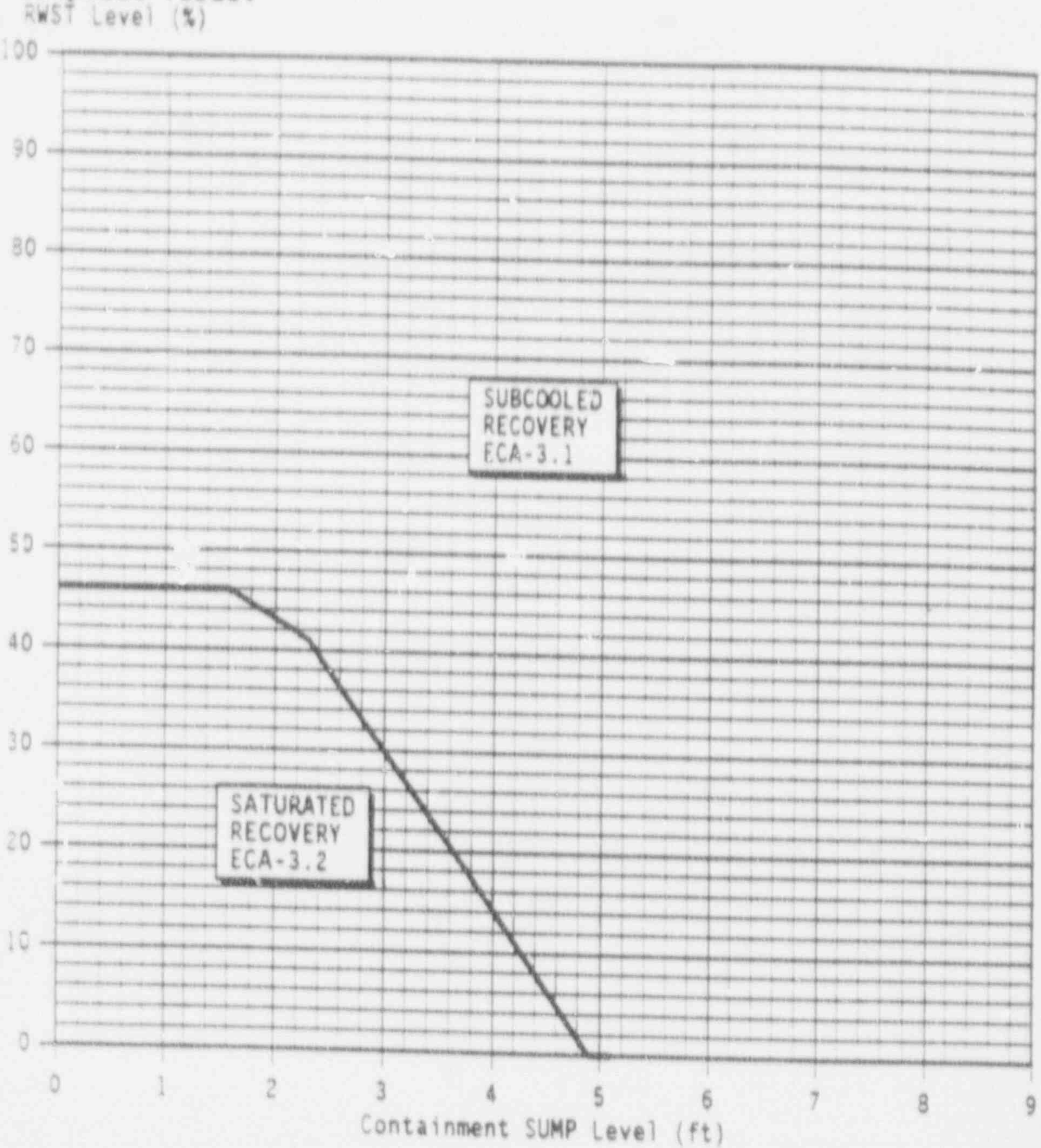
1) Calculation EE-0093, "Refueling Water Storage Tank Level Calibration", July 1989

North Anna **Page** 1

EOP Setpoint No: I.3 **Revision:** 6/10/92

Parameter: RWST LEVEL

Setpoint Value:



Applicability: UNIT 1 & UNIT 2

Associated System/Component: RWST/CONTAINMENT SUMP

EOP Setpoint No: I.3 **Revision:** 6/10/92

Description:

Combination of RWST level and containment sump level such that the total available water supply is equal to 50% of the pre-accident RWST water supply. This relationship is used as a transition criteria from a subcooled to a saturated recovery method following an SG tube rupture with loss of reactor coolant event.

Display Data:

Instrument	Units	Range	Division	Type	Location
LI-QS-200A	%	0-100%	2	METER	E2-CB-05
LI-QS-200B	%	0-100	2	METER	E2-CB-05
LI-QS-200C	FEET	0-11	4 INCHES	METER	E2-CB-05
LI-QS-200D	FEET	0-11	4 INCHES	METER	E2-CB-05
LI-QS-100A	%	0-100	2	METER	E1-CB-05
LI-QS-1[illegible]	%	0-100	2	METER	E1-CB-05
LI-QS-100C	%	0-100	2	METER	E1-CB-05
LI-QS-100D	%	0-100	2	METER	E1-CB-05

Key Assumptions:

None

Basis:

The transition criteria from the subcooled to saturated recovery methods is defined as:

$$V_{rwst} + \text{Max}\ (0,\ V_{sump} - V_{min}) = V_{12}$$

where Vrwst = water volume in RWST
Vsump = water volume in containment sump
Vmin = minimum sump water volume necessary to support NPSH requirements for the Recirculation Spray pumps.
V12 = average of the technical specification for the minimum average RWST water volume and the water volume corresponding to the RWST inaccessible inventory.

An indicated sump level corresponding to 1 ft 4 in was determined to be adequate to satisfy NPSH requirements of the recirculation spray pumps in Reference 1. Flow from the LHSI pumps is limited to less than the maximum flow capacity of one HHSI pump (560 gpm). For these conditions, the available NPSH at a sump level of 1 ft 4 in is expected to be adequate. From Reference 2 the containment sump level and volume are related as follows:

Table 1 - Sump Level vs. Volume

Level (ft)	Volume (cu ft)
1.0	223
1.6	569
2.3	3,740
>2.3	***

*** add 10,000 cu ft/ft for elevation above 2.3 ft

Interpolating this data:

Volume at 1 ft 4 in = 415.20 cu ft = 3,106 gal

Automatic switchover of the LHSI pumps occurs at approximately 25% of the RWST span (Reference 3). However, for the North Anna plant design, the containment quench spray pumps remain aligned to the RWST. From Reference 3, the inaccessible volume for the QS pumps is 3355 gallons.

The volume of water initially in the RWST is at least 466,200 gallons (Reference 3) so that

V12 = (466,200 + 3,355)/ 2 = 234,777.5 gals

Therefore,

Vrwst = 234,777.5
when Vsump < 3,106 gal (1 ft 4 in)

= 234,777.5 gal + 3,106 gal - Vsump
when 237,883.4 gal > Vsump >3,106 gal

= 0
when Vsump > 237,883.4

RWST indicated level is calculated as follows:
(Values from Reference 3)

Lrwst = (Vrwst - 16,788)/(466,200 - 16,788) x 94.7%

Table 2 presents the calculated RWST level as a function of containment sump level for use in the EOPs.

Table 2
RWST Level Criteria for Transition from Subcooled to Saturated Recovery

Indicated Sump Level (ft)	Sump Volume (gal)	RWST Volume (gal)	Indicated RWST Level (%)
0.0 - 1.33	0 - 3,106	234,777	46
1.6	4,256	233,627	46
2.3	27,977	209,906	41
4.9	221,095	16,788	0
5.1	237,883	0	<0

References:

1) EOP Setpoint No. M.9

2) Calculation SM-387 Rev 0, "RCS and Sump Liquid Inventories for Core Damage Assessment", March 1986

3) Calculation EE-0093, "Refueling Water Storage Tank Level Calibration Values", July 1989

North Anna

EOP Setpoint No: I.4 **Revision:** 06/10/92

Parameter: RWST LEVEL

Setpoint Value: 29%

Applicability: UNIT 1 & UNIT 2

Associated System/Component: RWST

Description:

RWST level for preparing for switchover to containment sump recirculation. This value is used to direct the operator to establish valve alignment for cold leg recirculation operation.

Display Data:

Instrument	Units	Range	Division	Type	Location
LI-QS-200A	%	0-100%	2	METER	E2-CB-05
LI-QS-200B	%	0-100	2	METER	E2-CB-05
LI-QS-200C	FEET	0-11	4 INCHES	METER	E2-CB-05
LI-QS-200D	FEET	0-11	4 INCHES	METER	E2-CB-05
LI-QS-100A	%	0-100	2	METER	E2-CB-05
LI-QS-100B	%	0-100	2	METER	E1-CB-05
LI-QS-100C	%	0-100	2	METER	E1-CB-05
LI-QS-100D	%	0-100	2	METER	E1-CB-05

Key Assumptions:

None

Basis:

This value is used to direct the operator to the procedure for switchover to sump recirculation. From Reference 1, the RWST level switchover setpoint is 28.3%. Rounding this value upward to the nearest 1/2 division gives the final EOP setpoint value:

$$L = 29\%$$

References:

1) Calculation EE-0093, "Refueling Water Storage Tank Level Calibration", July 1989

North Anna

Page 1

EOP Setpoint No: I.5

Revision: 06/10/92

Parameter: RWST LEVEL

Setpoint Value: 46%

Applicability: UNIT 1 & UNIT 2

Associated System/Component: RWST

Description:

The average of the technical specification for the minimum RWST water volume when full and the water volume corresponding to the RWST switchover setpoint, converted into RWST level units. This setpoint is used to determine if long term cooling will be available. If it is then a subcooled recovery procedure can be used.

Display Data:

Instrument	Units	Range	Division	Type	Location
LI-QS-200A	%	0-100%	2	METER	E2-CB-05
LI-QS-200B	%	0-100	2	METER	E2-CB-05
LI-QS-200C	FEET	0-11	4 INCHES	METER	E2-CB-05
LI-QS-200D	FEET	0-11	4 INCHES	METER	E2-CB-05
LI-QS-100A	%	0-100	2	METER	E2-CB-05
LI-QS-100B	%	0-100	2	METER	E1-CB-05
LI-QS-100C	%	0-100	2	METER	E1-CB-05
LI-QS-100D	%	0-100	2	METER	E1-CB-05

Key Assumptions:

None

Basis:

From Reference 1, this setpoint is calculated using the following steps:

1) Take the average of the RWST water volume when full and the water volume corresponding to the RWST switchover setpoint.

2) Convert the water volume calculated in step 1 into RWST level indication in plant specific units.

Automatic switchover of the LHSI pumps occurs at approximately 25% of the RWST span (Reference 2). However, for the North Anna plant design, the containment quench spray pumps remain aligned to the RWST and can continue to operate until RWST volume reaches 3355 gal. Therefore, the volume of water in the RWST which is unaccessible is only 3355 gallons (Reference 3). Also from Reference 3, the volume of water in the RWST when full (ie., at the TS minimum acceptable normal level) is 466,200 gallons. This value is equivalent to 94.7% level indication. The setpoint value

Basis: (Continued)

is calculated to be the average of these two:

$$Vavg = (466,200 + 3,355)/2$$

$$Vavg = 234,777 \text{ gallons}$$

Converting this into RWST level indication units (% of full tank) gives:

$$I = (234,777 - 16,788)/(466,200 - 16,788) * 94.7\% = 45.9\%$$

This value is rounded to the nearest 1/2 division to obtain the final EOP setpoint value:

$$I = 46\%$$

References:

1) Background Document to ECA-3.1, "SGTR with Loss of Reactor Coolant, Subcooled Recovery Desired", Revision 1A, July 1987, Pg. 70.

2) EOP Setpoint No. I.1

3) Calculation EE-0093, "Refueling Water Storage Tank Level Calibration", July 1989

North Anna

EOP Setpoint No: I.6

Revision: 06/10/92

Parameter: RWST LEVEL

Setpoint Value: 20%

Applicability: UNIT 1 POST SGR

Associated System/Component: RWST/ECCS

Description:

RWST level setpoint for transfer to containment sump recirculation. This value is used to verify that the automatic actions to realign the suction of the ECCS pumps from the RWST to the containment sump have been initiated.

Display Data:

Instrument	Units	Range	Division	Type	Location
LI-QS-200A	%	0-100%	2	METER	E2-CB-05
LI-QS-200B	%	0-100	2	METER	E2-CB-05
LI-QS-200C	FEET	0-11	4 INCHES	METER	E2-CB-05
LI-QS-200D	FEET	0-11	4 INCHES	METER	E2-CB-05
LI-QS-100A	%	0-100	2	METER	E2-CB-05
LI-QS-100B	%	0-100	2	METER	E1-CB-05
LI-QS-100C	%	0-100	2	METER	E1-CB-05
LI-QS-100D	%	0-100	2	METER	E1-CB-05
			2	METER	E1-CB-05

Key Assumptions:

None

Basis:

Reference 1 describes the ECCS design at North Anna where switchover to sump recirculation is manually initiated at 28.3% level and automatic initiation is started at 24.9% level.

Reference 2 describes reduction of these setpoints by 5.5% for steam generator replacement considerations.

The setpoints therefore reduce to 22.8 (automatic) and 19.4% (manual).

The automatic switchover setpoint is rounded up to 20% for use in the North Anna EOPs.

References:

1) Calculation EE-0093, "Refueling Water Storage Tank Level Calibration", July 1989

2) Calculation SM-471 Addendum B, June 1992

North Anna | **Page** 1

EOP Setpoint No: I.7 | **Revision:** 06/10/92

Parameter: RWST LEVEL

Setpoint Value: 23%

Applicability: UNIT 1 POST SGR

Associated System/Component: RWST

Description:

RWST level for preparing for switchover to containment sump recirculation. This value is used to direct the operator to establish valve alignment for cold leg recirculation operation. The value is applicable to Unit 1 after steam generator replacement.

Display Data:

Instrument	Units	Range	Division	Type	Location
LI-QS-200A	%	0-100%	2	METER	E2-CB-05
LI-QS-200B	%	0-100	2	METER	E2-CB-05
LI-QS-200C	FEET	0-11	4 INCHES	METER	E2-CB-05
LI-QS-200D	FEET	0-11	4 INCHES	METER	E2-CB-05
LI-QS-100A	%	0-100	2	METER	E1-CB-05
LI-QS-100B	%	0-100	2	METER	E1-CB-05
LI-QS-100C	%	0-100	2	METER	E1-CB-05
LI-QS-100D	%	0-100	2	METER	E1-CB-05

Key Assumptions:

None

Basis:

This value is used to direct the operator to the procedure for switchover to sump recirculation. From Reference 1, the RWST level switchover setpoint is 28.3%.

Reference 2 describes reduction of these setpoints by 5.5% for steam generator replacement conridrerations.

The setpoint therefore reduces to 22.8. this value is rounded up to the neares 1/2 division for use in the North Anna EOPs.

L=23%

References:

1) Calculation EE-0093, "Refueling Water Storage Tank Level Calibration", July 1989

2) Calculation SM-471 Addendum B, June 1992

Changes to Nuclear Control Room Operator Development Program Documents

This section has been incorporated in its entirety in NCRODP Training Modules Changes provided as Appendix 4-10 to DCP 9013.

A complete copy of this document including the proposed NCRODP changes from this section is available from the Records Management Department at the Innsbrook Technical Center.

Appendix C

UFSAR UPDATES

This section has been incorporated in its entirety in the UFSAR Change Request provided as Appendix 4-2 to DCP 9013.

A complete copy of this document including the proposed UFSAR updates from this section is available from the Records Management Department at the Innsbrook Technical Center.

WESTINGHOUSE LETTER CONCERNING ISI REQUIREMENTS FOR THE SG GIRTH WELD

PER THE WESTINGHOUSE LETTER, "VIRGINIA POWER REPORT, 70% DRAFT CHANGE PACKAGE 90-13-1 FOR THE STEAM GENERATOR REPLACEMENT FOR NORTH ANNA UNIT 1," MR. N. J. LIPARULO TO MR. J. E. RICHARDSON (NRC), DATED SEPTEMBER 2, 1992, THIS ATTACHMENT HAS BEEN WITHHELD SINCE THERE IS INFORMATION IN THIS ATTACHMENT THAT COULD BE PROPRIETARY TO WESTINGHOUSE.

WESTINGHOUSE MODEL 51F STEAM GENERATOR THERMAL AND HYDRAULIC DESIGN DATA REPORT

PER THE WESTINGHOUSE LETTER, "VIRGINIA POWER REPORT, 70% DRAFT CHANGE PACKAGE 90-13-1 FOR THE STEAM GENERATOR REPLACEMENT FOR NORTH ANNA UNIT 1," MR. N. J. LIPARULO TO MR. J. E. RICHARDSON (NRC), DATED SEPTEMBER 2, 1992, THIS ATTACHMENT HAS BEEN WITHHELD SINCE THERE IS INFORMATION IN THIS ATTACHMENT THAT COULD BE PROPRIETARY TO WESTINGHOUSE.

CRITICAL COMMODITIES FOR COLD GAP MEASUREMENT-VERIFICATION LOCATION OF TEMPORARY SUPPORTS AND SUPPORT REMOVALS

The following pages provide tables that list the critical commodities for cold measurement verification and the location of temporary supports and support removals. The list considers all primary reactor coolant piping including primary system branches attached to the reactor coolant loop piping and reactor coolant pumps. It also includes secondary plant system piping attached to the steam generators.

Critical commodities to be verified after achieving plant status Mode 6 - Refueling, include pipe rupture restraint cold gaps, rigid restraint cold gaps or clearances, constant and variable spring hanger cold sets, and hydraulic and mechanical shock suppressor (snubber) cold piston or stroke settings. For hydraulic shock suppressors, the setting is determined either by recording the face plate to pin ("C" dimension) dimension or pin to pin dimension. For mechanical snubbers, the setting is determined by recording the pin to pin dimension.

Temporary supports may be established as indicated in the tables by techniques including installing "pins", "blocks", or travel stops at constant or variable spring hangers; shimming (utilizing steel or wood as appropriate) gapped pipe supports, rupture restraints, or wall penetrations by providing zero clearance; or installing a new temporary support.

Pipe supports to be temporarily removed include supports (spring hangers and snubbers) that are installed on piping that is to be removed to facilitate steam generator repair (SGR) or snubbers that may incur damage as a result of being in an active construction area during SGR.

ITEM	NUMBER	PLANT DRAWING	MEASUREMENT	TEMPORARY SUPPORT
MAIN STEAM "A"				
Rupture Restraint	1-MSR-1	11715-FV-76A,B	check cold gaps	shim 3 points
Rupture Restraint	1-MSR-2	11715-FV-76A,B	check cold gaps	shim 3 points
Rupture Restraint	1-MSR-3	11715-FV-76A,B	check cold gaps	
Rupture Restraint	1-MSR-4	11715-FV-76A,B	check cold gaps	shim 3 points
Constant Spring	1-SHP-SH-42	11715-PSSK-102B.02	check cold setting	
Constant Spring	1-SHP-SH-45	11715-PSSK-102B.07	check cold setting	
Variable Spring	1-SHP-SH-46	11715-PSSK-102B.05	check cold setting	pin variable spring
Snubber	1-SHP-HSS-200	11715-PSSK-101B.01	check cold dimension	
Snubber	1-SHP-HSS-201A	11715-PSSK-101B.04	check cold dimension	
Snubber	1-SHP-HSS-201B	11715-PSSK-101B.04	check cold dimension	
Snubber	1-SHP-HSS-202A	11715-PSSK-101B.04	check cold dimension	
Snubber	1-SHP-HSS-202B	11715-PSSK-101B.04	check cold dimension	
Penetration	Crane wall			shim to bottom of pipe
MAIN STEAM "B"				
Rupture Restraint	1-MSR-6	11715-FV-76A,B	check cold gaps	
Rupture Restraint	1-MSR-7	11715-FV-76A,B	check cold gaps	shim 3 points
Rupture Restraint	1-MSR-8	11715-FV-76A,B	check cold gaps	
Rupture Restraint	1-MSR-9	11715-FV-76A,B	check cold gaps	shim 3 points
Variable Spring	1-SHP-SH-48	11715-PSSK-102C.06	check cold setting	pin variable spring
Constant Spring	1-SHP-SH-49	11715-PSSK-102C.05	check cold setting	
Snubber	1-SHP-HSS-209A	11715-PSSK-101C.04	check cold dimension	
Snubber	1-SHP-HSS-209B	11715-PSSK-101C.04	check cold dimension	
Snubber	1-SHP-HSS-210	11715-PSSK-102C.03	check cold dimension	
Snubber	1-SHP-HSS-223A	11715-PSSK-101C.02	check cold dimension	
Snubber	1-SHP-HSS-223B	11715-PSSK-101C.02	check cold dimension	
Penetration	Crane wall			shim to bottom of pipe
MAIN STEAM "C"				
Rupture Restraint	1-MSR-12	11715-FV-76A,B	check cold gaps	shim 3 points
Variable Spring	1-SHP-SH-47	11715-PSSK-102D.03	check cold setting	pin variable spring
Snubber	1-SHP-HSS-205	11715-PSSK-101D.01	check cold dimension	
Snubber	1-SHP-HSS-206	11715-PSSK-101D.02	check cold dimension	
Penetration	Crane wall			shim to bottom of pipe

ITEM	NUMBER	PLANT DRAWING	MEASUREMENT	TEMPORARY SUPPORT
FEEDWATER "A"				
Rupture Restraint	1-FWR-1	11715-FV-77A,B	check cold gaps	shim 3 points
Rupture Restraint	1-FWR-2	11715-FV-77A,C	check cold gaps	shim 3 points
Rupture Restraint	1-FWR-3	11715-FV-77A,C	check cold gaps	
Variable Spring	1-WFPD-SH-36	11715-PSSK-102A.05	check cold setting	
Constant Spring	1-WFPD-SH-37	11715-PSSK-102A.02	check cold setting	
Snubber	1-WFPD-HSS-200	11715-PSSK-102A.06	check cold dimension	
Snubber	1-WFPD-HSS-201	11715-PSSK-102A.06	check cold dimension	
Snubber	1-WFPD-HSS-231A	11715-PSSK-102A.03	check cold dimension	
Snubber	1-WFPD-HSS-231B	11715-PSSK-102A.03	check cold dimension	
Snubber	1-WFPD-HSS-231C	11715-PSSK-102A.03	check cold dimension	
Snubber	1-WFPD-HSS-231D	11715-PSSK-102A.03	check cold dimension	
Snubber	1-WFPD-HSS-235	11715-PSSK-102A.04	check cold dimension	
Snubber	1-WFPD-HSS-600	11715-PSSK-102A.01	check cold dimension	
Penetration	Crane Wall			shim to bottom of pipe
FEEDWATER "B"				
Rupture Restraint	1-FWR-10	11715-FV-77A,C	check cold gaps	shim 3 points
Rupture Restraint	1-FWR-11	11715-FV-77A,C	check cold gaps	
Rupture Restraint	1-FWR-12	11715-FV-77A,C	check cold gaps	
Rupture Restraint	1-FWR-13	11715-FV-77A,B	check cold gaps	shim 3 points
Variable Spring	1-WFPD-SH-38	11715-PSSK-102B.06	check cold setting	
Variable Spring	1-WFPD-SH-39	11715-PSSK 102B.04	check cold setting	
Constant Spring	1-WFPD-SH-40	11715-PSSK-102B.02	check cold setting	
Snubber	1-WFPD-HSS-210	11715-PSSK-102B.05	check cold dimension	
Snubber	1-WFPD-HSS-211	11715-PSSK-102B.05	check cold dimension	
Snubber	1-WFPD-HSS-212	11715-PSSK-102B.03	check cold dimension	
Snubber	1-WFPD-HSS-238	11715-PSSK-102B.01	check cold dimension	
Penetration	Crane Wall			shim to bottom of pipe

ITEM	NUMBER	PLANT DRAWING	MEASUREMENT	TEMPORARY SUPPORT
FEEDWATER "C"				
Rupture Restraint	1-FWR-4	11715-FV-77A,J	check cold gaps	shim 3 points
Rupture Restraint	1-FWR-5	11715-FV-77A,H	check cold gaps	
Rupture Restraint	1-FWR-7	11715-FV-77A,G	check cold gaps	shim 3 points
Rupture Restraint	1-FWR-9	11715-FV-77A,F	check cold gaps	
Variable Spring	1-WFPD-SH-41	11715-PSSK-102C.05	check cold setting	
Variable Spring	1-WFPD-SH-42	11715-PSSK-102C.04	check cold setting	
Constant Spring	1-WFPD-SH-43	11715-PSSK-102C.02	check cold setting	
Snubber	1-WFPD-HSS-204	11715-PSSK-102C.01	check cold dimension	
Snubber	1-WFPD-HSS-205	11715-PSSK-102C.03	check cold dimension	
Snubber	1-WFPD-HSS-206	11715-PSSK-102C.03	check cold dimension	
Snubber	1-WFPD-HSS-207	11715-PSSK-102C.03	check cold dimension	

CHEMICAL FEED "A"

Snubber	1-FPH-CFPD-1-18	11715-MFSK-5293	check cold setting	remove snubber
Snubber	1-FPH-CFPD-1-19	11715-MFSK-5295	check cold setting	remove snubber

Verify pipe locations and support installation.
Remove piping to coupling inside crane wall.
Rigid pipe support 1-FPH-CFPD-1-17 (dwg. 11715-MFSK-867A) provides temporary dead weight support for cut pipe.

CHEMICAL FEED "B"

Snubber	1-FPH-CFPD-2-21	11715-MFSK-5294	check cold setting	remove snubber
Snubber	1-FPH-CFPD-2-24	11715-MFSK-2757	check cold setting	remove snubber

Verify pipe locations and support installation.
Remove piping to coupling inside crane wall.
Rigid pipe support 1-FPH-CFPD-2-19 (dwg. 11715-MFSK-2756) provides temporary dead weight support for cut pipe.

CHEMICAL FEED "C"

Rigid Pipe Support	1-FPH-CFPD-3-5	11715-MFSK-5292	check cold clearance

Verify pipe locations and support installation.
Remove piping to cut location between supports 1-FPH-CFPD-3-5 and 1-FPH-CFPD-5-5.
Rigid pipe support 1-FPH-CFPD-3-3 (dwg. 11715-MFSK-2190A) provides temporary dead weight support for cut pipe.

ITEM	NUMBER	PLANT DRAWING	MEASUREMENT	TEMPORARY SUPPORT

SG BLOWDOWN "A"
Verify pipe and support locations.
Cut at existing tee.
Install temporary dead weight support at hanger 1-WGCB-R-450.

SG BLOWDOWN "B"
Verify pipe and support locations.
Cut at existing tee.
Install temporary dead weight support at hanger 1-WGCB-R-MFSK-753A.

SG BLOWDOWN "C"
Verify pipe and support locations.
Cut at existing tee.
Rigid pipe support 1-FPH-WGCB-11-3 provides temporary dead weight support for cut pipe.

ITEM	NUMBER	PLANT DRAWING	MEASUREMENT	TEMPORARY SUPPORT
SG WET LAY UP "A"				
Variable Spring	1-FPH-SGD-4-117A	11715-MFSK-5197	check cold setting	
Constant Spring	1-FPH-SGD-4-117D	11715-MFSK-5200	check cold setting	install travel stop at constant spring
Variable Spring	1-FPH-SGD-10-115	11715-MFSK-5195	check cold setting	
Snubber	1-FPH-SGD-4-117C	11715-MFSK-5199	check cold dimension	
Snubber	1-FPH-SGD-4-117E	11715-MFSK-5201	check cold dimension	
Rigid Pipe Support	1-FPH-SGD-4-117B	11715-MFSK-5198	check cold clearance	
Verify pipe and support locations				
Cut piping on 4 foot pipe segment adjacent to 1-FPH-SGD-4-117D				
SG WET LAY UP "B"				
Variable Spring	1-FPH-SGD-5-150A	11715-PSSK-102H.02	check cold setting	pin variable spring
Snubber	1-FPH-SGD-5-150B	11715-PSSK-102H.01	check cold dimension	remove snubber
Snubber	1-FPH-SGD-5-148	11715-PSSK-102H.04	check cold dimension	
Verify pipe and support locations				
Cut piping on 9 inch pipe segment adjacent to 1-FPH-SGD-5-150A				
SG WET LAY UP "C"				
Variable Spring	1-FPH-SGD-6-178	11715-PSSK-102J.07	check cold setting	
Variable Spring	1-FPH-SGD-6-182	11715-PSSK-102J.03	check cold setting	pin variable spring
Variable Spring	1-FPH-SGD-6-184	11715-PSSK-102J.01	check cold setting	remove variable spring
Snubber	1-FPH-SGD-6-180	11715-PSSK-102J.05	check cold dimension	
Snubber	1-FPH-SGD-6-181	11715-PSSK-102J.04	check cold dimension	
Snubber	1-FPH-SGD-6-183	11715-PSSK-102J.02	check cold dimension	remove snubber
Verify pipe and support locations				
Cut piping on pipe segment between 1-FPH-SGD-6-182 and 1-FPH-SGD-6-183				

ITEM	NUMBER	PLANT DRAWING	MEASUREMENT	TEMPORARY SUPPORT
RC LOOP "A"				
Rupture Restraint	1-RC-PRR-1A	11715-FV-82A-F	check cold gap	
Rupture Restraint	1-RC-PRR-4A	11715-FV-86A,B,E	check cold gap	shim to bottom of pipe
--------	-------	----------	-------	support (no welding) hot leg pipe from SG lower support
RC LOOP "B"				
Rupture Restraint	1-RC-PRR-1B	11715-FV-82A-F	check cold gap	
Rupture Restraint	1-RC-PRR-4B	11715-FV-86A,B,E	check cold gap	shim to bottom of pipe
----------	----------	----------	----------	support (no welding) hot leg pipe from SG lower support
RC LOOP "C"				
Rupture Restraint	1-RC-PRR-1C	11715-FV-82A-F	check cold gap	
Rupture Restraint	1-RC-PRR-4C	11715-FV-86A,B,E	check cold gap	shim to bottom of pipe
----------	----------	----------	----------	support (no welding) hot leg pipe from SG lower support
PRESSURIZER SURGE LINE, LOOP "C"				
Rupture Restraint	1-RC-PRR-62	11715-FV-83A,G	check cold gap	
Rupture Restraint	1-RC-PRR-63	11715-FV-83A,G	check cold gap	
Rupture Restraint	1-RC-PRR-68	11715-FV-83A,G	check cold gap	
LETDOWN, LOOP "A"				
Rupture Restraint (H)	1-CH-PRR-15	11715-FV-167F,G,N	check cold gap	
Rupture Restraint (H)	1-CH-PRR-15A	11715-FV-167F,G,N	check cold gap	
CHARGING, LOOP "B"				
Rigid Pipe Support	1-FPH-CH-1-2	11715-PSSK-1116B.02	check cold gaps	

ITEM	NUMBER	PLANT DRAWING	MEASUREMENT	TEMPORARY SUPPORT
RHR/LHSI, LOOP "A"				
Variable Spring	1-SI-SH-13	11715-PSSK-113A.41	check cold setting	
Rupture Restraint (V)	1-RC-PRR-14	11715-FV-167A-C	check cold gaps	
RHR/LHSI, LOOP "B"				
Rupture Restraint (V)	1-SI-PRR-24	11715-FV-167A-C	check cold gaps	
Spring Hanger	1-SI-SH-001	11715-PSSK-113B.01	check cold setting	
Snubber	1-SI-HSS-100A	11715-PSSK-113B.02	check cold dimension	
RHR/LHSI, LOOP "C"				
Rupture Restraint (V)	1-SI-PRR-34	11715-FV-167D,E	check cold gap	
Spring Hanger	1-SI-SH-7	11715-PSSK-113C.13	check cold setting	
RCL ISOLATION VALVE BYPASS, LOOP "A"				
Snubber LSS	1-FPH-RC-45-2	11715-PSSK-103AR.1	check cold dimension	
Rigid Pipe Support	1-FPH-RC-45-3	11715-PSSK-103AR.3	check cold gap	
Variable Spring	1-FPH-RC-47-1	11715-MFSK-1257A-2	check cold setting	
Snubber VSS	1-FPH-RC-47-2	11715-MFSK-1240A-2	check cold dimension	
Snubber LSS	1-FPH-RC-47-3	11715-MFSK-1241A-2	check cold dimension	
Snubber LSS	1-FPH-RC-63-1	11715-MFSK-1346A	check cold dimension	
RCL ISOLATION VALVE BYPASS, LOOP "B"				
Variable Spring	1-FPH-RC-48-1	11715-PSSK-103AQ.1	check cold setting	
Snubber LSS	1-FPH-RC-48-2	11715-PSSK-103AQ.2	check cold dimension	
Snubber VSS	1-FPH-RC-48-3	11715-PSSK-103AQ.3	check cold dimension	
Rigid Pipe Support	1-FPH-RC-46-3	11715-PSSK-103AQ.4	check cold gap	
Snubber LSS	1-FPH-RC-64-1	11715-PSSK-103AQ.5	check cold dimension	
Snubber LSS	1-FPH-RC-46-2	11715-PSSK-103AQ.7	check cold dimension	
RCL ISOLATION VALVE BYPASS, LOOP "C"				
Snubber LSS	1-FPH-RC-44-2	11715-PSSK-103AS.1	check cold dimension	
Snubber LSS	1-FPH-RC-65-1	11715-PSSK-103AS.3	check cold dimension	
Snubber VSS	1-FPH-RC-49-3	11715-PSSK-103AS.5	check cold dimension	
Snubber LSS	1-FPH-RC-49-2	11715-PSSK-103AS.6	check cold dimension	
Variable Spring	1-FPH-RC-49-1	11715-PSSK-103AS.7	check cold setting	

ITEM	NUMBER	PLANT DRAWING	MEASUREMENT	TEMPORARY SUPPORT
COLD LEG LHSI, LOOP "A"				
Rigid Pipe Support	1-FPH-SI-131-2	11715-PSSK-103N.04	check cold gap	
COLD LEG LHSI, LOOP "B"				
Rigid Pipe Support	1-FPH-SI-133-1	11715-PSSK-103Y.07	check cold gap	
COLD LEG LHSI, LOOP "C"				
Rigid Pipe Support	1-FPH-SI-132-1	11715-PSSK-103W.03	check cold gap	
HOT LEG LHSI, LOOP "A"				
Rigid Pipe Support	1-SI-21-1	11715-PSSK-103AJ.01	check cold gap	
Rigid Pipe Support	1-SI-21-2	11715-PSSK-103AJ.02	check cold gap	
HOT LEG LHSI, LOOP "B"				
Rigid Pipe Support	1-FPH-SI-59-19	11715-PSSK-103AF.06	check cold gap	
HOT LEG LHSI, LOOP "C"				
Rupture Restraint	1-RC-PRR-34	11715-FV-167D,E	check cold gap	
Rigid Pipe Support	1-SI-R-213	11715-PSSK-103AD.05	check cold gap	
Rigid Pipe Support	1-FPH-SI-16-2	11715-PSSK-103AD.06	check cold gap	
Variable Spring	1-FPH-SI-16-1	11715-PSSK-103AD.07	check cold setting	
Variable Spring	1-FPH-SI-63-5	11715-PSSK-103AD.08	check cold setting	
PRESSURIZER SPRAY, LOOP "A"				
Snubber LSS	1-FPH-RC-14-101	11715-PSSK-110B.45	check cold dimension	
Snubber LSS	1-FPH-RC-14-100	11715-PSSK-110B.46	check cold dimension	
Variable Spring	1-FPH-RC-14-1	11715-PSSK-110B.47	check cold setting	
PRESSURIZER SPRAY, LOOP "C"				
Variable Spring	1-FPH-RC-15-28	11715-PSSK-110B.04	check cold setting	
Snubber	1-FPH-RC-15-112	11715-PSSK-110B.05	check cold dimension	
Snubber	1-FPH-RC-15-111	11715-PSSK-110B.06	check cold dimension	

ITEM	NUMBER	PLANT DRAWING	MEASUREMENT	TEMPORARY SUPPORT
RHR SUCTION, LOOP "A"				
Variable Spring	1-RH-SH-19	11715-PSSK-113A.13	check cold setting	
Snubber	1-RH-HSS-116A	11715-PSSK-113A.24	check cold dimension	
Snubber	1-RH-HSS-108A	11715-PSSK-113A.25	check cold dimension	
Snubber	1-RH-HSS-108B	11715-PSSK-113A.26	check cold dimension	
Snubber	1-RH-HSS-109C	11715-PSSK-113A.27	check cold dimension	
Snubber	1-RH-HSS-109A	11715-PSSK-113A.28	check cold dimension	
Snubber	1-RH-HSS-109B	11715-PSSK-113A.28	check cold dimension	
Snubber	1-RH-HSS-109D	11715-PSSK-113A.28	check cold dimension	
RC LOOP BYPASS, LOOP "A"				
Variable Spring	1-RC-SH-3	11715-PSSK-109A.01	check cold setting	pin variable spring
Snubber	1-RC-HSS-104	11715-PSSK-109A.02	check cold dimension	
Snubber	1-RC-HSS-105	11715-PSSK-109A.02	check cold dimension	
RC LOOP BYPASS, LOOP "B"				
Variable Spring	1-RC-SH-4	11715-PSSK-109B.01	check cold setting	pin variable spring
Snubber	1-RC-HSS-106	11715-PSSK-109B.02	check cold dimension	
Snubber	1-RC-HSS-107	11715-PSSK-109B.02	check cold dimension	
RC LOOP BYPASS, LOOP "C"				
Variable Spring	1-RC-SH-5	11715-PSSK-109C.01	check cold setting	pin variable spring
Snubber	1-RC-HSS-108	11715-PSSK-109C.02	check cold dimension	
Snubber	1-RC-HSS-109	11715-PSSK-109C.02	check cold dimension	

ITEM	NUMBER	PLANT DRAWING	MEASUREMENT	TEMPORARY SUPPORT
SEAL WATER INJECTION (INLET), LOOP "A"				
Snubber	1-FPH-CH-96-2	11715-PSSK-103AT.1	check cold dimension	
Snubber	1-FPH-CH-96-1	11715-PSSK-103AT.2	check cold dimension	
Snubber	1-FPH-CH-96-3	11715-PSSK-103AT.3	check cold dimension	
Variable Spring	1-FPH-CH-96-4	11715-PSSK-103AT.4	check cold setting	
SEAL WATER INJECTION (INLET), LOOP "B"				
Variable Spring	1-FPH-CH-94-21	11715-PSSK-103AU.21	check cold setting	
Snubber	1-FPH-CH-94-22	11715-PSSK-103AU.22	check cold dimension	
SEAL WATER INJECTION (INLET), LOOP "C"				
Snubber	1-FPH-CH-92-18	11715-PSSK-103AV.16	check cold dimension	
SEAL WATER INJECTION (OUTLET), LOOP "A"				
Rigid Pipe Support	1-FPH-CH-14-1	11715-PSSK-111AK.01	check cold gap	
Snubber	1-FPH-CH-14-2	11715-PSSK-111AK.02	check cold dimension	
Snubber	1-FPH-CH-14-3	11715-PSSK-111AK.03	check cold dimension	
SEAL WATER INJECTION (OUTLET), LOOP "B"				
Rigid Pipe Support	1-FPH-CH-15-15	11715-PSSK-111AL.24	check cold gap	
Rigid Pipe Support	1-FPH-CH-15-16	11715-PSSK-111AL.25	check cold gap	
SEAL WATER INJECTION (OUTLET), LOOP "C"				
Snubber	1-FPH-CH-16-16	11715-PSSK-111AN.23	check cold dimension	
Rigid Pipe Support	1-FPH-CH-16-17	11715-PSSK-111AN.24	check cold gap	

ITEM	NUMBER	PLANT DRAWING	MEASUREMENT	TEMPORARY SUPPORT
RC LOOP DRAIN, LOOP "A"				
Snubber	1-FPH-RC-215-1	11715-PSSK-103BB.01	check cold dimension	
Variable Spring	1-FPH-RC-215-2	11715-PSSK-103BB.02	check cold setting	
Snubber	1-FPH-RC-216-6	11715-PSSK-103BB.03	check cold dimension	
Snubber	1-FPH-RC-53-44	11715-PSSK-103BB.09	check cold dimension	
Snubber	1-FPH-RC-53-45	11715-PSSK-103BB.10	check cold dimension	
Snubber	1-FPH-RC-53-46	11715-PSSK-103BB.11	check cold dimension	
Snubber	1-FPH-RC-53-46A	11715-PSSK-103BB.11	check cold dimension	
Variable Spring	1-FPH-RC-53-42	11715-PSSK-103BB.12	check cold setting	
Variable Spring	1-FPH-RC-53-43	11715-PSSK-103BB.24	check cold setting	
RC LOOP DRAIN, LOOP "B"				
Snubber LSS	1-FPH-RC-217-1	11715-PSSK-103BB.18	check cold dimension	
Snubber VSS	1-FPH-RC-217-4	11715-PSSK-103BB.19	check cold dimension	
Variable Spring	1-FPH-RC-217-3	11715-PSSK-103BB.20	check cold setting	
Variable Spring	1-FPH-RC-57-3	11715-PSSK-103BB.21	check cold setting	
Snubber	1-FPH-RC-57-1	11715-PSSK-103BB.22	check cold dimension	
Snubber	1-FPH-RC-57-2	11715-PSSK-103BB.23	check cold dimension	
Rigid Pipe Support	1-FPH-RC-218-1	11715-PSSK-103BB.35	check cold gap	
Snubber LSS	1-FPH-RC-217-2	11715-PSSK-103BB.17	check cold dimension	
RC LOOP DRAIN, LOOP "C"				
Variable Spring	1-FPH-RC-219-1	11715-PSSK-103BB.25	check cold setting	
Snubber	1-FPH-RC-219-2	11715-PSSK-103BB.26	check cold dimension	
Snubber	1-FPH-RC-220-6	11715-PSSK-103BB.27	check cold dimension	
Variable Spring	1-FPH-RC-220-11	11715-PSSK-103BB.31	check cold setting	
Variable Spring	1-FPH-RC-58-1	11715-PSSK-103BB.32	check cold setting	
Variable Spring	1-FPH-RC-220-5	11715-PSSK-103BB.33	check cold setting	
Snubber	1-FPH-RC-58-2	11715-PSSK-103BB.13	check cold dimension	
Snubber	1-FPH-RC-58-3	11715-PSSK-103BB.14	check cold dimension	

STEAM GENERATOR INSULATION DEBRIS ANALYSES LETTER REPORT

STONE & WEBSTER ENGINEERING CORPORATION

245 SUMMER STREET, BOSTON, MASSACHUSETTS 02210

TELEPHONE 617-589-5111 FAX 617-589-2156 8U71

BOSTON MA
CHATTANOOGA TN
CHERRY HILL NJ
CHICAGO IL
DALLAS TX
DECATUR AL
DENVER CO
FT LAUDERDALE FL
HOUSTON TX

NEW YORK NY
OAK RIDGE TN
PORTLAND ME
PORTLAND OR
RICHLAND WA
RICHMOND VA
PLEASANTON CA
TAMPA FL
WASHINGTON DC

Mr. M. W. Gettler
Manager - Steam Generator Repair Project
Virginia Power
Innsbrook Technical Center
5000 Dominion Boulevard
Glen Allen, VA 23060
ATTN: Mr. R. K. Bayer

July 15, 1992

J.O.No. 02072.1610
NAS-20,897

LETTER REPORT
STEAM GENERATOR INSULATION DEBRIS
NORTH ANNA POWER STATION - UNIT 1

Enclosed as Attachment 1 is a letter report summarizing the results of the analyses and assessments performed in support of the replacement of the existing reflective insulation with fiberglass-blanket type insulation. The analyses assume the RWST LHSI switchover level setpoint change is implemented. Comments received from Virginia Power on Friday July 10, 1992, during a telephone conversation between Mr. L. Viens and Mr. D. Lippard of Virginia Power and R. Bain and M. Collins of S&W have been incorporated.

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

D. E. McLellan
Project Engineer

Enclosure

MKC:CM

LETTER REPORT
TECHNICAL SUMMARY STEAM GENERATOR
INSULATION DEBRIS ANALYSES AND ASSESSMENTS
NORTH ANNA POWER STATION - UNIT 1

1.0 Statement of Problem

The existing reflective insulation on the North Anna Unit 1 steam generators and associated piping will be replaced with fiberglass-blanket type insulation when the steam generators are replaced. The acceptability of this replacement is dependent on the successful results of the effect the new insulation has on insulation debris transport analysis and available net positive suction head (NPSH) for the recirculation spray (RS) and low head safety injection (LHSI) pumps.

2.0 Summary of Analyses and Assessments

2.1 Initial Assessments

2.1.1 NPSH

An initial assessment was performed to compare the impact of the new blanket insulation and the existing metal-encapsulated fiberglass insulation on RS pumps and LHSI pumps available NPSH using the latest NRC requirements. It was concluded that the new blanket insulation would cause a head loss of 0.4 feet while the existing insulation causes a head loss of approximately 0.12 feet.

2.1.2 NRC Requirements

Generic Letter 85-22 (Ref. 2.7.6) recommends that Regulatory Guide 1.82, Rev. 1 (Ref. 2.7.1) be used as guidance for the conduct of 10CFR 50.59 reviews dealing with the changeout and/or modification of thermal insulation installed on primary coolant system piping and components.

Based on this recommendation, it was concluded that RG 1.82, Rev. 1 would be utilized in the insulation debris transport analysis for either the existing or new insulation design. In addition NUREG/CR-2791 and NUREG-0897 were utilized for guidance in determining the type and amount of debris formed due to jet impingement as well as how the sump screens are affected by this debris. It should be noted that this methodology has been previously used on Millstone 3.

2.2 Insulation System

The replacement steam generators employ a removable encapsulated fiberglass insulation system. The insulation system consists of a light density fiberglass insulating material encapsulated in a tough woven fiberglass cloth to form a blanket or pillow. The pillows are attached together with Velcro and are covered with a protective and removable stainless steel sheathing. Encapsulation of the fiberglass results in a significant increase in insulation system strength and resistance to the impinging jet forces which emanate from a postulated pipe rupture.

2.3 Insulation Debris Transport to ESF Sump

In accordance with the requirements of Reference 2.7.1, an insulation debris inventory and transport analysis was performed on Unit 1 containment emergency sump to evaluate the insulation installed on the replacement steam generators. The results of the analysis are shown on Exhibit 1 which gives the transient head loss at the containment emergency sump fine mesh screens due to the accumulation of insulation debris.

In order to analyze the effect of insulation debris buildup on the emergency sump fine screens due to a pipe break in the containment, the worst pipe break and the worst location of the break with respect to the sump fine screens was identified.

The primary system pipe breaks are the worst credible pipe breaks for debris generation since the jet contacts the greatest insulation surface.

Jet impingement break at the steam generator nozzle crossover pipe to the cooling pump was selected for this analysis. The worst location for the above break is the closest proximity to the emergency sump. This was taken to be steam generator cubicle "B" which is located directly above the sump and the t ugh. There is no direct route from the pipe break in steam generator cubicle "B" to the fine screens. A walkdown (Reference C.7.7) was performed to confirm the direct openings in the steam generator cubicle to the containment floor El. 716'-11". The only direct openings in cubicle "B" to the containment floor are:

a. Cubicle "B" floor opening by reactor cavity wall which is directly above the containment trough.

b. Mesh Door in the Cubicle "B" wall which opens outward into the containment.

The steam generator cubicle "B" is very congested area. The pipe break is located right next to door in cubicle "B". During the pipe break the mesh door was assumed to be blown off and most of the transportable debris blown out of the door into the containment and upward inside steam generator cubicle "B" toward the operating floor. Some of the debris may go through the floor opening and out of cubicle "B", but it must go through a very congested path. The ratio of the amount of debris that will go through the door and through the floor to the containment is unknown. Transportable debris may ultimately migrate to the emergency sump fine screens.

A portion of the insulation fragments or small pieces were assumed to be carried immediately to the floor by the pressure difference resulting from the rupture. Additional fragments were directed away from the containment floor and emergency sump by the pressure difference and some were assumed to remain in the subcompartment on gratings and horizontal surfaces. After the blowdown terminates, the spray systems begin to wash down the small fragments of insulation. The wash-down process is inefficient and, therefore, time consuming since most of the fragments, due to their location are not in the direct path of the spray droplet, ; the flow of water For conservatism, however the analysis assumes all transportable debris, including large fragments and "as fabricated" sections reach the containment sump screens in 1800 seconds.

The volume of insulation contributing to the total debris inventory was determined by the intersection of the jet emerging from the rupture with the insulated surfaces of equipment. The insulation was assumed to be severely damaged and removed from the equipment up to a distance of seven pipe diameters from the jet.

Three (3) types debris are formed as a result of jet impingement (Reference 2.7.3):

1. 30% are fine suspended fibers

2. 40% are fluffy fragments which immediately sink.

3. 30% are large floating fragments which will float for days.

Reference 2.7.2 indicates that:

1. 30% of the fine suspended fibrous debris builds up uniformly on the fine screens.

2. 40% of the fluffy fragments which immediately sink will transport and accumulate at the lower portion of the fine screens if the velocity of water is greater than 0.2 fps.

3. 30% of the large floating fragments which will float for days before sinking and therefore have no impact on head loss.

There are three stages of screening prior to the inlet to the pumps. The first stage is at the containment sump with a coarse mesh of 0.588 inch openings. The second stage is at the containment sump with a fine mesh of

0.12 inch openings. The third and final stage of screening utilizes cylindrical screens mounted over the intakes at each LHSI, ORS, and IRS pump. This is a fine mesh screen with 0.12 inch openings.

Exhibit 1 depicts two head loss curves, each employing different assumptions relative to the number of Engineered Safety Features (ESF) trains operating. The maximum ESF case assumes both trains are operating. Since this case exhibits the highest water withdrawal rate from the sump, it also exhibits the greatest head loss across the screens. The minimum case assumes one ESF train is operating. The additional head loss due to insulation debris results in a reduction in the NPSH available to all pumps drawing water from the containment emergency sump. The available NPSH for the RS and LHSI pumps is discussed in Section 2.5.1.

2.4 Containment LOCA Analysis

The results of the containment analysis (Ref.2.7.5) based on the replacement of the Unit 1 steam generators with new Model 51F steam generators concludes that all containment acceptance criteria are satisfied. This is contingent upon the refueling water storage tank (RWST) LHSI switchover (S/O) setpoint being adjusted to -5.5% from the current setpoint. The following is a summary of the Containment Design Bases acceptance criteria results:

Criteria	Design	Worst Case		Margin	
		Old	New	Old	New
Containment Peak Pressure, psig	45	44.1	43.93	0.90	1.07
Containment Pressurization time, sec.	3600	3,310	3370	290	230
Containment Subatmospheric Peak Pressure, psig	<0.00	-0.02	-0.04	.02	.04
Containment Peak Temp, °F	280	270.3	272.3	9.7	7.7

2.5 Assessment of RS and LHSI System Performance

2.5.1 NPSH

The containment analysis (Reference 2.7.5) determined the IRS and ORS pumps available NPSH ($NPSH_A$) and margin were reduced. The old and new $NPSH_A$, required NPSH ($NPSH_R$) and margin are as follows:

	$NPSH_R$, ft	$NPSH_A$, ft		MARGIN, ft	
		Old	New	Old	New
ORS Pump	11.0	16.8	15.3	5.8	4.3
IRS Pump	9.4	11.9	10.4	2.5	1.0

For the LHSI pumps, the containment analysis determined that there was an increase in $NPSH_A$ and margin as follows:

	$NPSH_R$, ft		$NPSH_A$, ft		MARGIN, ft	
	Old	New	Old	New	Old	New
LHSI Pump	13.4	13.1	13.5	14.1	0.1	1.0

2.5.2 Pump Suction Screen Blockage and Pump Clearances

Insulation debris generated by jet impingement from a postulated pipe rupture would exist in a wide spectrum of sizes and shapes. However, most of the debris would be filtered out prior to the intake of the recirculation and low head safety injection pumps by the containment sump fine mesh screens. The fine mesh screens have an open mesh size of 0.12 inches.

The proposed insulation system single glass fiber size is about 2-3 mils. The insulation fibers may or may not be coated with organic binders designed to hold the fibers together. The open mesh would allow passage of single and multiple fibers (bundles) up to 0.12 inches in diameter or width. The individual pump screens also have a mesh size of 0.12 inches. If small insulation fragments pass through the sump screens, they will also pass through the individual pump screens.

Although there are tight clearances found at the wearing ring locations with a pump, the majority of material that might be temporarily suspended in the pumped fluid will actually pass through the much larger passages out the discharge. Therefore, operability and performance of the pumps will not be measurably affected by short term exposure of low levels of debris.

2.5.3 Spray Header Nozzle Blockage

The smallest spray header nozzles are 0.25 inches in diameter which is larger than the smallest sump and pump screen mesh size of 0.12 inches. The small insulation particles will not block the spray nozzles due to the fluid turbulence at the nozzles, the large pressure differential at the nozzles and the opening being larger than the smallest screen mesh size.

2.6 Conclusions

There will be an increase in head loss across the sump screens due to the build-up of insulatio debris during a LOCA. The increase in sump screen head loss was utilized in the containment LOCA analysis performed with the replacement steam generators. The containment LOCA analysis determined there is a reduction in the margin between NPSH available and NPSH required for the inside and outside recirculation spray pumps. In addition, the containment LOCA analysis determined there is an increase in NPSH margin for the low head safety injection pumps, assuming the RWST LHSI switchover level setpoint change is implemented. Adequate NPSH margin is still available for these pumps to perform their safety related functions.

Potential long term blockage of the individual suction screens for the inside and outside recirculation spray and low head safety injection pumps due to insulation fragments that pass through the sump screens will not occur since the screen mesh size of the finest sump screen is the same as the screen mesh size of the individual pump suction scre ns.

If small insulation fragments pass through the sump screens they will also pass through the individual pump screens. These small fragments of insulation will have no impact on the operation of the inside and outside recirculation spray and low head safety injection pumps. The pumps will pump water and small insulation fragments into the spray headers and the primary system. The small particles of insulation will not block the spray header nozzles due to fluid turbulence at the nozzle, the large pressure differential at the nozzle and the nozzle opening being larger than the smallest sump screen mesh size. Some settling of insulation fragments may occur in the reactor vessel.

The replacement of the existing Unit 1 steam generators and associated piping reflective and encapsulated fiberglass insulation with fiberglass blanket type insulation will retain acceptable containment analysis and pump NPSH margins and will meet all present day regulatory requirements. The safety related systems and components impacted by this change will continue to perform their safety related functions.

2.7 References

2.7.1 USNRC, Water Sources For Long-Term Recirculation Cooling Following A Loss-of-Coolant-Accident, USNRC Regulatory Guide 1.82, Rev. 1, November 1985.

2.7.2 A. W. Serkiz, Containment Emergency Sump Performance, USNRC, NUREG-0897, October 1985.

2.7.3 J. Wysocki and R. Kolbe, Methodology for Evaluation of Insulation Debris Effects, Burns and Roe, Inc. and Sandia National laboratories, NUREG/CR-2791 and SAND82-7067, September 1982.

2.7.4 S&W Calculation 02072.1610-US(B)-273 "Head Loss Across Emergency Sump Screens Due to Insulation Debris Caused By a LOCA Event," Rev. 0.

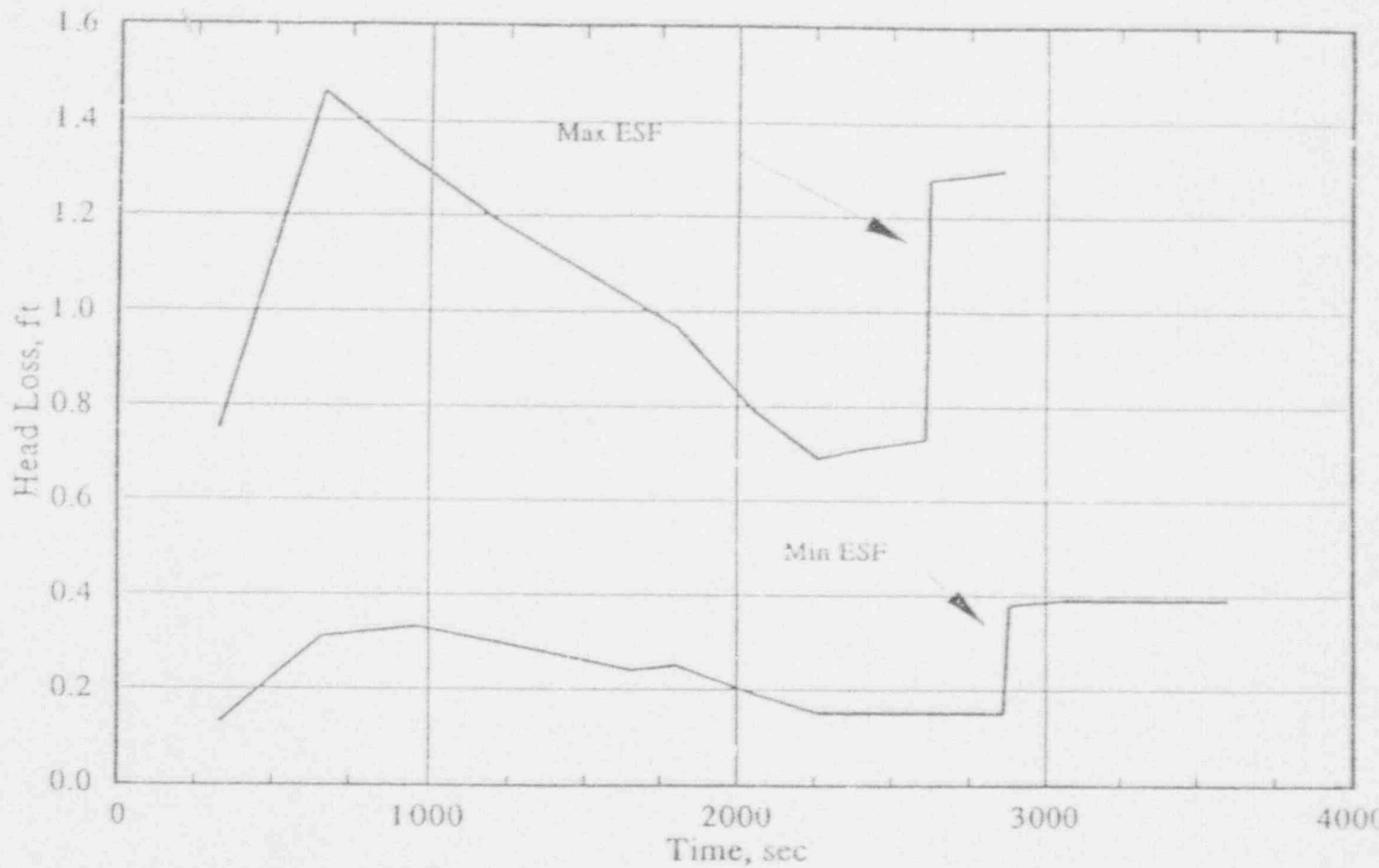
2.7.5 S&W Calculation 02072.201C-US(B)-274, "Containment LOCA Analysis With New Steam Generators", Rev.0.

2.7.6 USNRC, Potential For Loss of Post-LOCA Recirculation Capability Due to Insulation Debris Blockage, USNRC Generic Letter 85-22, December 3, 1985.

2.7.7 Walkdown Report transmitted by letter NAS-20,672 dated February 14, 1992.

2.8 Exhibits

1. Insulation Debris Head Loss Following a LOCA.



Insulation Debris Head Loss
Following a LOCA

EXHIBIT 1

NUCLEAR PLANT RELIABILITY DATA SYSTEM GENERAL REPORTS

Nuclear Plant Reliability Data System
General Report

For: Nancy Chapman
Bechtel Corporation

Report-Id: NPRG00AA
Job Number: 9064
Run Date: 08/17/92
Run Time: 09:16

Introduction:

The attached report was generated by your query of the NPRDS data base.
A summary of your query is listed below.

QUERY:

You selected the following search condition(s).

Selected Component is HTEXCH
Selected HTEXCH Subcategory is Steam Generator
Selected Manufacturer is Westinghouse Elec Corp / Hagan
Find Mfr Model Numbers that contain 51F

There were no records meeting the search condition(s).

DISPLAY AND SORT:

You selected to run general report 4:

Component Failure Brief Report

You chose to sort the report by:

Sort Sequence	Field Name

NPRG04AA
Date: 08/17/92

Nuclear Plant Reliability Data System - Failure Brief Report

Run

Job Number: 9064

_Unit__ _Comp_ __Utility Component Id___ __Dates_____
Narratives

Nuclear Plant Reliability Data System
General Report

For: Nancy Chapman
Bechtel Corporation

Report-id: NPRG00AA
Job Number: 8271
Run Date: 08/14/92
Run Time: 07:54

Introduction:

The attached report was generated by your query of the NPRDS data base.
A summary of your query is listed below.

QUERY:

You selected the following search condition(s):

Selected Component is VALVE
Selected Manufacturer is Conval Inc
Find Mfr Model Numbers that are equal to 11G2J-S16-3D

There were no records meeting the search condition(s).

DISPLAY AND SORT:

You selected to run general report 4:

Component Failure Brief Report

You chose to sort the report by:

Sort Sequence	Field Name

NPRG04AA Nuclear Plant Reliability Data System - Failure Brief Report Run Date: 08/14/92
Job Number: 8271

Unit	Comp	Utility Component Id	Dates

Narratives

Nuclear Plant Reliability Data System
General Report

For: Nancy Chapman
Bechtel Corporation

Report-Id: NPRG00AA
Job Number: 8275
Run Date: 08/14/92
Run Time: 07:58

Introduction:

The attached report was generated by your query of the NPRDS data base.
A summary of your query is listed below.

QUERY:

You selected the following search condition(s):

Selected Component is VALVE
Selected Manufacturer is Conval Inc
Find Mfr Model Numbers that are equal to 11G3J-S16-3D

There were no records meeting the search condition(s).

DISPLAY AND SORT:

You selected to run general report 4:

Component Failure Brief Report

You chose to sort the report by:

Sort Sequence	Field Name

NPRG04AA Nuclear Plant Reliability Data System - Failure Brief Report Run Date: 08/14/92
Job Number: 8275

Unit Comp Utility Component Id Dates
Narratives

LINE DESIGNATION TABLE MARK-UPS - FEEDWATER AND BLOWDOWN

SIZE-SYST-UNID-CLASS-LO	FROM	TO	FLUID	OPPRES	OPTEMP	DESPRES	DESTEMP	TSTPRES	VER
14.00-WER-7-151	1-WT-P-2B	16-WER-8-151	WATER	15.00	220.00	50.00	220.00	.	N
16.00-WER-8-151	14-WER-6-151	1-WT-EV-1	WATER	15.00	220.00	50.00	220.00	.	N
8.00-WER-9-151	1-WT-EV-1	1-WT-EV-1	WATER	15.00	220.00	50.00	220.00	.	N
3.00-WFPD-1-301	1-AB-P-14A	4-WFPD-42-301		.	.	.	.	.	N
18.00-WFPD-1-901	26-WFPD-3-901	18-WFPD-1-901	WATER	1170.00	338.00	1800.00	338.00	.	N
6.00-WFPD-10-901	16-WFPD-8-901	1-FW-FCV-1499	WATER	1170.00	440.00	1800.00	440.00	.	N
6.00-WFPD-11-601	1-FW-FCV-1499	16-WFPD-9-601	WATER	850.00	440.00	1100.00	440.00	.	N
16.00-WFPD-12-901	26-WFPD-6-901	1-FW-78	WATER	1170.00	440.00	1800.00	440.00	.	N
16.00-WFPD-13-601	1-FW-78	1-FW-79	WATER	850.00	440.00	1100.00	440.00	.	N
6.00-WFPD-14-901	16-WFPD-12-901	1-FW-FCV-1489	WATER	1170.00	440.00	1800.00	440.00	.	N
6.00-WFPD-15-601	6-WFPD-14-901	16-WFPD-13-601	WATER	850.00	440.00	1100.00	440.00	.	N
6.00-WFPD-15-901	1-FW-FCV-1489	16-WFPD-13-601		.	.	.	.	.	N
18.00-WFPD-16-901	16-WFPD-20-901	1-FW-E-1B	WATER	1170.00	338.00	1800.00	338.00	.	N
16.00-WFPD-17-601	1-FW-46	1-FW-47	WATER	850.00	440.00	1100.00	440.00	.	N
6.00-WFPD-18-901	16-WFPD-7-901	1-FW-FCV-1479	WATER	1170.00	440.00	1800.00	440.00	.	N
6.00-WFPD-19-601	1-FW-FCV-1479	16-WFPD-17-901	WATER	850.00	440.00	1100.00	440.00	.	N
3.00-WFPD-2-301	1-AP-P-14B	4-WFPD-42-301		.	.	.	.	.	N
18.00-WFPD-2-901	26-WFPD-3-901	1-FW-E-1A	WATER	1170.00	338.00	1800.00	338.00	.	N
16.00-WFPD-20-901	18-WFPD-15-901	16-WFPD-4-901	WATER	1170.00	440.00	1800.00	440.00	.	N
16.00-WFPD-21-901	18-WFPD-2-901	18-WFPD-5-901	WATER	1170.00	440.00	1800.00	440.00	.	N
16.00-WFPD-22-601-02	1-FW-111	1-RC-E-1C	WATER	850.00	440.00	1100.00	440.00	.	N
16.00-WFPD-23-601-02	1-FW-79	1-RC-E-1B	WATER	850.00	440.00	1100.00	440.00	.	N
16.00-WFPD-24-601-02	1-FW-47	1-RC-E-1A	WATER	850.00	440.00	1100.00	440.00	.	N
26.00-WFPD-3-901	18-WFPD-1-901	18-WFPD-31-901	WATER	1170.00	338.00	1800.00	338.00	.	N
18.00-WFPD-30-901	1-FW-P-1A	26-WFPD-3-901	WATER	1170.00	338.00	1800.00	338.00	.	N
18.00-WFPD-31-901	1-FW-P-1B	26-WFPD-3-901	WATER	1170.00	338.00	1800.00	338.00	.	N
18.00-WFPD-32-901	1-FW-P-1C	26-WFPD-3-901	WATER	1170.00	338.00	1800.00	338.00	.	N
18.00-WFPD-4-901	1-FW-E-1B	26-WFPD-6-901	WATER	1170.00	440.00	1800.00	440.00	.	N
18.00-WFPD-401-901	26-WFPD-403-901	18-WFPD-414-901		.	.	.	.	.	N
18.00-WFPD-402-901	18-WFPD-430-901	2-FW-E-1A		.	.	.	.	.	N
26.00-WFPD-403-901	18-WFPD-402-901	18-WFPD-401-901		.	.	.	.	.	N
18.00-WFPD-404-901	2-FW-E-1B	26-WFPD-408-901		.	.	.	.	.	N
18.00-WFPD-405-901	2-FW-E-1A	26-WFPD-406-901		.	.	.	.	.	N
26.00-WFPD-406-901	18-WFPD-5-901	LINE CAPPED		.	.	.	.	.	N
16.00-WFPD-407-901	26-WFPD-406-901	2-FW-50		.	.	.	.	.	N
16.00-WFPD-408-901	26-WFPD-406-901	16-WFPD-409-601		.	.	.	.	.	N

601C

NORTH ANNA LINE DESIGNATION TABLE UNITS 1 & 2

10/23/91

SIZE-SYST-UN/O-CLASS-LO	FROM	TO	FLUID	OPPRES	OPTEMP	DESPRES	DESTEMP	TSTPRES	VER
18.00-WFPR-14-901	12-WFPR-18-901	8-WFPR-3-901		.	.	.	.	.	N
12.00-WFPR-16-901	8-WFPR-19-901	18-WFPR-14-901		.	.	.	.	.	N
12.00-WFPR-17-901	8-WFPR-21-901	18-WFPR-14-901		.	.	.	.	.	N
12.00-WFPR-18-901	8-WFPR-23-901	18-WFPR-14-901		.	.	.	.	.	N
8.00-WFPR-19-901	12-WFPR-27-901	12-WFPR-16-901		.	.	.	.	.	N
8.00-WFPR-2-901	18-WFPD-14-901	1-CH-SC-1B	WATER	1170.00	388.00	1800.00	388.00	.	N
8.00-WFPR-20-901	12-WFPR-27-901	12-WFPR-16-901		.	.	.	.	.	N
8.00-WFPR-21-901	12-WFPR-26-901	12-WFPR-17-901		.	.	.	.	.	N
8.00-WFPR-22-901	12-WFPR-26-901	12-WFPR-17-901		.	.	.	.	.	N
8.00-WFPR-23-901	12-WFPR-25-901	12-WFPR-18-901		.	.	.	.	.	N
8.00-WFPR-24-901	12-WFPR-25-901	12-WFPR-18-901		.	.	.	.	.	N
12.00-WFPR-25-901	18-WFPD-30-901	8-WFPR-23-901		.	.	.	.	.	N
12.00-WFPR-26-901	18-WFPD-31-901	8-WFPR-21-901		.	.	.	.	.	N
12.00-WFPR-27-901	18-WFPD-32-901	8-WFPD-19-901		.	.	.	.	.	N
8.00-WFPR-3-901	18-WFPR-14-901	1-CH-SC-1A	WATER	1170.00	388.00	1800.00	388.00	.	N
8.00-WFPR-401-901	18-WFPD-430-901	2-CH-SC-1B	WATER	1170.00	388.00	1800.00	388.00	.	N
8.00-WFPR-402-901	18-WFPD-431-901	2-CH-SC-1B	WATER	1170.00	388.00	1800.00	[illegible]	.	N
8.00-WFPR-403-901	18-WFPD-432-901	2-CH-SC-1A	WATER	1170.00	388.00	1800.00	[illegible]	.	N
18.00-WFPR-404-901	12-WFPR-4[illegible]-901	8-WFPR-403-901	WATER	1170.00	388.00	1800.00	388.00	.	N
12.00-WFPR-405-901	18-WFPD-432-901	18-WFPR-404-901	WATER	1170.00	388.00	1800.00	388.00	.	N
12.00-WFPR-406-901	2-FW-P-1B	18-WFPD-404-901	WATER	1170.00	388.00	1800.00	388.00	.	N
12.00-WFPR-407-901	18-WFPD-430-901	18-WFPR-404-901	WATER	1170.00	388.00	1800.00	388.00	.	N
3.00-WGCB-1-601-Q3			WATER	818.00	521.00	1085.00	560.00	.	N
~~2.00-WGCB-10-601-Q3~~ 2.50-WGCB-10-601C-Q3			WATER	818.00	521.00	1085.00	560.00	.	N
~~2.00-WGCB-11-601-Q3~~ 2.50-WGCB-11-601C-Q3			WATER	818.00	521.00	1085.00	560.00	.	N
1.00-WGCB-12-~~601~~-Q3 601C			WATER	818.00	521.00	1085.00	560.00	.	N
4.00-WGCB-13-902			WATER	9.00	300.00	165.00	365.00	.	N
3.00-WGCB-14-601-Q2			WATER	818.00	521.00	1085.00	560.00	.	N
3.00-WGCB-15-601-Q2			WATER	818.00	521.00	1085.00	560.00	.	N
3.00-WGCB-16-601-Q2			WATER	818.00	521.00	1085.00	560.00	.	N
2.00-WGCB-17-601-Q3			WATER	818.00	521.00	1085.00	560.00	.	N
2.00-WGCB-18-601-Q3			WATER	818.00	521.00	1085.00	560.00	.	N
2.00-WGCB-19-601-Q3			WATER	818.00	521.00	1085.00	560.00	.	N
3.00-WGCB-2-601-Q3			WATER	818.00	521.00	1085.00	560.00	.	N
3.00-WGCB-20-601			WATER	818.00	521.00	1085.00	560.00	.	N
3.00-WGCB-21-601			WATER	818.00	521.00	1085.00	560.00	.	N

SIZE-SYST-UNID-CLASS-LQ	FROM	TO	FLUID	OPPRES	OPTEMP	DESPRES	DESTEMP	TSTPRES	VER
3.00-WGCB-22-601			WATER	818.00	521.00	1085.00	560.00	.	N
3.00-WGCB-23-601			WATER	818.00	521.00	1085.00	560.00	.	N
3.00-WGCB-24-601			WATER	818.00	521.00	1085.00	560.00	.	N
3.00-WGCB-26-601			STEAM	75.00	320.00	150.00	366.00	.	N
3.00-WGCB-27-601			WATER	818.00	521.00	1085.00	560.00	.	N
3.00-WGCB-3-601-03			WATER	818.00	521.00	1085.00	560.00	.	N
3.00-WGCB-30-601			WATER	818.00	521.00	1085.00	560.00	.	N
3.00-WGCB-31-601			WATER	818.00	521.00	1085.00	560.00	.	N
4.00-WGCB-33-601			WATER	818.00	521.00	1085.00	560.00	.	N
6.00-WGCB-34-151			WATER	75.00	320.00	150.00	366.00	.	N
4.00-WGCB-35-151			WATER	75.00	320.00	150.00	366.00	.	N
4.00-WGCB-36-151			WATER	75.00	320.00	150.00	366.00	.	N
4.00-WGCB-37-151			WATER	75.00	320.00	150.00	366.00	.	N
2.00-WGCB-38-151			WATER	75.00	320.00	150.00	366.00	.	N
2.00-WGCB-39-151			WATER	75.00	320.00	150.00	366.00	.	N
2.50-WGCB-4-601C-Q3 ~~2.00-WGCB-4-601-03~~			WATER	818.00	521.00	1085.00	560.00	.	N
8.00-WGCB-40-151			WATER	75.00	320.00	150.00	366.00	.	N
3.00-WGCB-401-601-03	2-WGCB-405-601-03	2-BD-TV-200B	WATER	818.00	521.00	1085.00	560.00	.	N
3.00-WGCB-402-601-03	2-WGCB-408-601-03	2-BD-TV-200D	WATER	818.00	521.00	1085.00	560.00	.	N
3.00-WGCB-403-601-03	2-WGCB-411-601-03	2-BD-TV-200F	WATER	818.00	521.00	1085.00	560.00	.	N
2.00-WGCB-404-601-03	2-RC-E-1A	3-WGCB-401-601-03	WATER	818.00	521.00	1085.00	560.00	.	N
2.00-WGCB-405-601-03	2-RC-E-1A	3-WGCB-401-601-03	WATER	818.00	521.00	1085.00	560.00	.	N
1.00-WGCB-406-601-03	2-RC-E-1A	2-WGCB-404-601-03	WATER	818.00	521.00	1085.00	560.00	.	N
2.00-WGCB-407-601-03	2-RC-E-1B	3-WGCB-402-601-03	WATER	818.00	521.00	1085.00	560.00	.	N
2.00-WGCB-408-601-03	2-RC-E-1B	3-WGCB-402-601-03	WATER	818.00	521.00	1085.00	560.00	.	N
1.00-WGCB-409-601-03	2-RC-E-1B	2-WGCB-407-601-03	WATER	818.00	521.00	1085.00	560.00	.	N
8.00-WGCB-41-151			WATER	75.00	320.00	150.00	366.00	.	N
2.00-WGCB-410-601-03	2-RC-E-1C	3-WGCB-403-601-03	WATER	818.00	521.00	1085.00	560.00	.	N
2.00-WGCB-411-601-03	2-RC-E-1C	3-WGCB-403-601-03	WATER	818.00	521.00	1085.00	560.00	.	N
1.00-WGCB-412-601-03	2-RC-E-1C	2-WGCB-411-603-03	WATER	818.00	521.00	1085.00	560.00	.	N
4.00-WGCB-413-902	12050-FM-098A, SH. 4	2-BD-TK-1	WATER	9.00	300.00	165.00	365.00	.	N
3.00-WGCB-414-601-02	2-BD-TV-200B	2-BD-TV-200A	WATER	818.00	521.00	1085.00	560.00	.	N
3.00-WGCB-415-601-02	2-BD-TV-200D	2-BD-TV-200C	WATER	818.00	521.00	1085.00	560.00	.	N
3.00-WGCB-416-601-02	2-BD-TV-200F	2-BD-TV-200E	WATER	818.00	521.00	1085.00	560.00	.	N
2.00-WGCB-417-601-03	3-WGCB-401-601-03	2-BD-3	WATER	818.00	521.00	1085.00	560.00	.	N
2.00-WGCB-418-601-03	3-WGCB-402-601-03	2-BD-12	WATER	818.00	521.00	1085.00	560.00	.	N

SIZE-SYST-UNID-CLASS-LQ	FROM	TO	FLUID	OPPRES	OPTEMP	DESPRES	DESTEMP	TSTPRES	VER
2.00-WGCB-454-601	3-WGCB-420-601	11715-FM-102B, SH. 1	WATER	50.00	150.00	60.00	150.00	.	N
2.00-WGCB-455-601	3-WGCB-421-601	11715-FM-102B, SH. 1	WATER	50.00	150.00	60.00	150.00	.	N
2.00-WGCB-456-601	3-WGCB-422-601	11715-FM-102B, SH. 1	WATER	50.00	150.00	60.00	150.00	.	N
1.00-WGCB-457-601	3-WGCB-423-601	2-WGCB-448-601	WATER	818.00	521.00	1085.00	560.00	.	N
1.00-WGCB-458-601	3-WGCB-427-601	2-WGCB-449-601	WATER	818.00	521.00	1085.00	560.00	.	N
1.00-WGCB-459-601	3-WGCB-430-601	2-WGCB-450-601	WATER	818.00	521.00	1085.00	560.00	.	N
4.00-WGCB-46-151			WATER	75.00	320.00	150.00	366.00	.	N
6.00-WGCB-462-602	3-WGCB-424-601	10-WGCB-426-602	WATER	75.00	320.00	150.00	366.00	.	N
6.00-WGCB-463-602	3-WGCB-428-601	10-WGCB-426-602	WATER	75.00	320.00	150.00	366.00	.	N
4.00-WGCB-47-151			WATER	75.00	320.00	150.00	366.00	.	N
2.00-WGCB-48-601			WATER	818.00	521.00	1085.00	560.00	.	N
2.00-WGCB-49-601			WATER	818.00	521.00	1085.00	560.00	.	N
2.50-WGCB-5-601C-Q3 ~~2.00-WGCB-5-601-Q3~~			WATER	818.00	521.00	1085.00	560.00	.	N
2.00-WGCB-50-601			WATER	818.00	521.00	1085.00	560.00	.	N
2.00-WGCB-51-601			WATER	818.00	521.00	1085.00	560.00	.	N
2.00-WGCB-52-601			WATER	818.00	521.00	1085.00	560.00	.	N
2.00-WGCB-53-601			WATER	818.00	521.00	1085.00	560.00	.	N
2.00-WGCB-54-601			STEAM	75.00	320.00	150.00	366.00	.	N
6.00-WGCB-54-602				.	.	.	.	.	N
2.00-WGCB-55-601			STEAM	75.00	320.00	150.00	366.00	.	N
6.00-WGCB-55-602				.	.	.	.	.	N
2.00-WGCB-56-601				.	.	.	.	.	N
601C 1.00-WGCB-6-~~601~~-Q3			WATER	818.00	521.00	1085.00	560.00	.	N
~~2.00-WGCB-7-601-Q3~~ 2.50-WGCB-7-601C-Q3			WATER	818.00	521.00	1085.00	560.00	.	N
~~2.00-WGCB-8-601-Q3~~ 2.50-WGCB-8-601C-Q3			WATER	818.00	521.00	1085.00	560.00	.	N
1.00-WGCB-9-~~601~~-Q3 601C			WATER	818.00	521.00	1085.00	560.00	.	N
2.00-WHDW-1-301	2-WHDW-1-301		WATER	600.00	393.00	650.00	410.00	.	N
1.00-WHDW-2-301	1-WHDW-3-301	1-SD-TE-111A		.	.	.	.	.	N
1.00-WHDW-3-301	10-WHPD-2-301	1-WHDW-3-301		.	.	.	.	.	N
0.75-WHDW-4-301	1-WHDW-3-301			.	.	.	.	.	N
2.00-WHDW-401-301	FM-75A	12-SLPD-401-151	WATER	.	.	.	.	.	N
1.00-WHDW-402-301	0.75-WHDW-409-301	2-WHDW-401-301	WATER	.	.	.	.	.	N
1.00-WHDW-403-301	0.75-WHDW-406-301	0.75-WHDW-404-301	WATER	.	.	.	.	.	N
0.75-WHDW-404-301	1-WHDW-403-301	2-SD-792	WATER	.	.	.	.	.	N
0.75-WHDW-405-301	2-SD-790	1-WHDW-403-301	WATER	.	.	.	.	.	N
0.75-WHDW-406-301	2-SD-791	1-WHDW-403-301	WATER	.	.	.	.	.	N

DC 90-13-1, Appendix 4-30, Page 4

SPECIAL OVERRATED-LOAD LIFT QUALIFICATION, REACTOR CONTAINMENT POLAR CRANE BRIDGE STRUCTURE

J.O. No. 01040.0810

SPECIAL OVER RATED-LOAD LIFT QUALIFICATION
REACTOR CONTAINMENT POLAR CRANE
BRIDGE STRUCTURE
NORTH ANNA POWER STATION

PREPARED BY
STONE & WEBSTER ENGINEERING CORPORATION
JANUARY 1991

TABLE OF CONTENTS

1.0 STATEMENT OF PROBLEM

In order to accomplish the steam generator repair effort, the use of the existing Reactor Containment Polar Crane is required. Although the two main hoisting units provide sufficient rated capacity, the bridge structure does not, and therefore requires evaluation, in order to be qualified in accordance with ASME B30.2 (Reference 5.1) to make the necessary special over rated-load lifts.

This letter report addresses the over rated-load lift qualification of the Polar Crane's bridge structure, as authorized by the Virginia Power July 30, 1990 letter NP-1486-S07-004 (Reference 5.2).

2.0 DISCUSSION

2.1 Equipment Operation

Lower sections of the existing steam generators will be removed and replaced by utilizing the Reactor Containment Polar Crane. The crane's bridge structure supports two main hoist & trolley units which connect to a lift beam, which in turn connects to the lifting trunnions on the lower steam generator sections.

Upper sections of the existing steam generators could be handled using the two main hoists as described above, or they may be handled using a single main hoist provided the total lifted weight does not exceed the single hoist's capacity.

2.2 Equipment in Question

The Reactor Containment Polar Crane, built by the Harnischfeger Corporation, is located within the containment on a separate crane wall at elevation 343'-0", as is depicted in Attachment 1. The span of the crane bridge, and centerline diameter of the crane wall, is 104'-0". The circular runway rail upon which the bridge operates is a 175 lb./yd. Bethlehem crane rail. The configuration of the crane with its two hoist and trolley units mounted on a bridge structure is shown on the crane clearance diagram, Attachment 2.

2.2 Equipment in Question (Continued)

The original (1970) design capacities of the Polar Crane are as follows:

Permanent Nameplate Ratings

Main Hoists (two) .... 125 tons each
Aux Hoist .... 15 tons
Bridge .... 250 tons

Temporary Construction Ratings

Main Hoists (two) .... 195 to 197 1/2 tons each
Aux Hoist .... 30 tons
Bridge .... 395 tons

Note, to obtain the hoist construction ratings, the wire ropes were rereeved, increasing the mechanical advantage. However, no modifications were performed on the bridge structure to obtain it's construction rating.

The upgraded (1977) design capacities of the Polar Crane are as follows:

Permanent Nameplate Ratings

Main Hoists (two) .... 140 tons each
Aux Hoist .... 15 tons (Unchanged)
Bridge .... 250 tons (Unchanged)

Note, that although the bridge rated capacity was not increased, it is evaluated for a seismic event occurring with a 140 ton load as opposed to the originally specified 125 ton load.

3.0 SUMMARY OF STUDY

3.1 Introduction

In accordance with Section 2-3.2.1.1 of ASME B30.2, (Attachment 3) infrequent over rated-load lifts may be performed provided specific requirements are met. These ASME (a) through (h) requirements are summarized as follows:

(a) A written review of the crane service history shall be made.

(b) Crane structural, mechanical and electrical components are to be evaluated for the higher loads according to accepted standards.

(c) The crane supporting structure shall be checked for the higher loads according to accepted standards.

(d) The crane shall be inspected prior to making the over rated-load lifts.

(e) The over rated-load lifts shall be made under the direction of a designated person in accordance with a prepared lift plan (procedure).

(f) The operator shall test the crane at the special over rated-load lift by lifting the actual load a short distance and setting the brakes.

(g) The crane shall be inspected after making the over rated-load lifts.

(h) A record of the over rated-load lifts, including distances moved, shall be maintained on file.

Concerning the steam generator replacement effort and use of the existing Polar Crane, the maximum load to be lifted will be 280 tons or less (See Assumption 3.6.1). Based upon this load, the existing main hoisting units have sufficient capacity, and therefore, it is only the bridge which requires qualification for the special over rated-load lifts.

3.1 Introduction (Continued)

This letter report addresses the ASME criteria for over rated-load lifts as follows:

Section 3.2 Review of Crane Service History (ASME item a)
Section 3.3 Evaluation of Crane Bridge (ASME item b)
Section 3.4 Evaluation of Crane Supporting Structure (ASME item c)
Section 3.5 Crane Load Testing (ASME item f)

ASME items d, e, g and h are appropriately discussed under Section 3.6 Assumptions, Section 4.1 Conclusions, and Section 4.2 Recommendations.

3.2 Review of Crane Service History

In accordance with the criteria of ASME B30.2, the review of the crane service history considered the following:

1. Previous Over Rated-Load Lifts;
2. Structural Repairs;
3. Modifications of Original Design.

The following is a list of all known over rated load lifts including load weight, number of hoists used to make the lifts, and appropriate references describing the lift.

- Steam Generators, three items lifted during initial plant construction, load weight 390-395 tons design, (Actual weight 348.5 tons per Field Installation Procedure No. 58, Reference 5.3) utilizing two main hoists at 11'-3" centers, in accordance with Specification NAS-142, Reference 5.4.

- Reactor Vessel, one item lifted during initial plant construction, load weight 370-371 tons design, utilizing two main hoists at 11'-3" centers, in accordance with Specification NAS-142, Reference 5.4.

3.2 Review of Crane Service History (Continued)

- Neutro.. Shield Tank, one item lifted during initial plant construction, load weight 181 tons, utilizing one main hoist to fully support the load, in accordance with Specification NAS-142, Reference 5.4.

- Reactor Vessel Head, one item raised and lowered during initial plant construction, load weight 135 tons, utilizing one hook, in accordance with Specification NAS-142, Reference 5.4. (Note, subsequent Reactor Vessel Head lifts made after commercial operation, June 1978, are not considered over rated load lifts, since the main hoist units were upgraded to 140 tons in 1977.)

- Reactor Internals, one item raised and .owered during initial plant construction, load weight 135 tons, utilizing one hook, in accordance with Specification NAS-142, Reference 5.4. (Note, subsequent lifts of the Reactor Internals made after commercial operation, June 1978, are not considered over rated load lifts, since the main hoist units were upgraded to 140 tons in 1977.)

- Initial Load Test, one set of lifts, load weight 418 tons maximum, utilizing two main hoists, reference January 1973 Stone & Webster Quality Control Inspection Report on testing of Unit 1 Polar Crane Main Hoists, Reference 5.5, and Procedure for Testing Unit 1 & 2 Polar Crane dated December 6, 1972, Reference 5.6.

- Load Test for Upgraded Hoist Capacity, one lift on each hoist, load weight 171 tons, reference Marc 1977 Test Certification Report and Stone & Webster Quality Control Inspection Report transmitted by Stone & Webster March 11, 1977 letter to M.J.V. Harrison, Reference 5.7.

3.2 Review of Crane Service History (Continued)

With regard to structural repairs, the maintenance history of the Polar Crane was reviewed by investigating all work orders relating to the bridge from 08/22/78 to 12/18/89 (Reference 5.8). Except for the removal and subsequent reinstallation of the girder's box beam vent covers, maintenance performed upon the bridge was either electrically or mechanically oriented, and basically related to normal crane maintenance. It is therefore concluded that there have been no major structural repairs to the bridge structure.

With regard to modifications of the original crane design, the only significant change that was made was the 1977 upgrading of each main hoist's capacity from 125 tons to 140 tons. This modification, covered by Harnischfeger's Uprate Calculation (Ref. 5.9), had no impact on the bridge structure, and therefore does not factor into the qualification of the bridge for making a special over rated-load lift.

3.3 Evaluation of Crane Bridge

In accordance with the criteria of ASME B30.2, the evaluation of the crane bridge considered the following:

1. Structural components
2. Mechanical components
3. Electrical components

With regard to the structural components, the original manufacturer's calculations (Reference 5.10) were reviewed. The construction lift of 370 tons (185 tons on each main hoist spaced at 11'-3") resulted in the most severe non seismic loading condition. As per Assumption 3.6.2, seismic loads shall not be considered during steam generator replacement. For the 370 ton construction lift, designated as Case Vb in the manufacturer's calculation, the maximum bending stresses at mid span are 17.9 ksi in compression and 18.1 ksi in tension. Although the original allowable stress established for a construction lift was 80% of yield, the construction lift stresses are still less than the present CMAA Spec # 70 (Reference 5.11) minimum allowable stress which is 60% of yield, or 21.6 ksi. Intuitively, reducing the 370 ton load to 280 tons will

3.3 Evaluation of Crane Bridge (Continued)

significantly reduce the bending stresses below the present minimum CMAA allowable. Based upon the present condition of the bridge (see Assumption 3.6.3), it is concluded that the bridge is structurally adequate for the 280 ton special over-rated load lifts.

With regard to the mechanical and electrical components, it can be concluded that since the original steam generator and reactor vessel construction lifts were performed without any mechanical or electrical modifications of the bridge drives, that they are also sufficiently designed to make lower capacity special over rated-load lifts of 280 tons.

It is therefore concluded that the crane bridge is structurally, mechanically and electrically qualified for making the indicated special over rated-load lifts.

3.4 Evaluation of Crane Supporting Structure

In accordance with the criteria of ASME B30.2, the evaluation of the crane supporting structure considered the following:

1. Conformance to accepted standards
2. Consideration for the effects of deterioration

Concerning the conformance to accepted standards, an evaluation of the Polar Crane supporting structure was performed to verify that the loads and stresses imposed by the crane during replacement of the steam generators are within the original design limits. Based upon the present condition of the supporting structure (See Assumption 3.6.4), it is concluded that the supporting structure is structurally adequate for the 280 ton special over rated load lifts.

The maximum initial design construction load of 390 tons and its associated wheel load of 130.25 kips (Reference 5.10), was evaluated for the bearing stress at the top of the wall under the crane rail anchor plate, and for the critical crane wall column load.

3.4 Evaluation of Crane Supporting Structure (Continued)

Drawing 11715-FC-16Z (Reference 5.12) provides the detail used to anchor the rail to the top of the crane wall. By evaluation, the maximum bearing stress on the 12" X 12" anchor plates, spaced 2'-0" on center, is approximately 1400 psi. (The 12" X 12" plate which is subjected to a vertical construction wheel load of 130.25 kips is assumed to be a strip with an effective width of 8 inches and length of 12 inches.) This is well below the maximum bearing stress of 1785 psi permitted by Paragraph 10.14, "Bearing" of the ACI 318-71 Building Code (Reference 5.13). These results are consistent with Stone & Webster calculation 11715-BK-5B, Section 4B, "Anchors for Polar Crane Rail" (Reference 5.14).

The crane wall and crane wall columns were originally designed for the maximum construction lift of 390 tons in Stone & Webster calculation 11715-BK-5B, Section 8A, "Crane Wall - Investigation of Stability for Construction Condition" (Reference 5.15). In this calculation, the axial loads in the critical crane wall columns were determined and found to be less than one third of the allowable capacity.

Utilizing the Harnischfeger crane calculations (References 5.9 and 5.10), and a load of 140 tons on each main hoist, the maximum "special over rated-load lift" wheel load is 106.3 kips/wheel, which is substantially less than the maximum construction lift wheel load. It is therefore concluded that the crane supporting structure can sustain all loads and stresses associated with the maximum lift of 280 tons occurring during the steam generator repair effort.

3.5 Crane Load Testing

In accordance with the criteria of ASME B30.2, Section 2-3.2.1.1 item f, load testing of the bridge structure for the special over rated-load lifts shall be accomplished by raising the over rated-load lift (the steam generator section itself) a short distance and then setting the brakes.

Note, that as specifically stated under item i of this section of the ASME standard, a 125% rated load test is not required for a "Special Over Rated-Load Lift".

3.6 Assumptions

As addressed in this report, the qualification of the Polar Crane bridge structure for the indicated special over rated-load lifts is based upon the following assumptions:

3.6.1 The maximum load to be lifted for steam generator replacement, including all rigging, is assumed to be 280 tons or less, with the load divided evenly between the two main hoists spaced no closer than 11'-3".

3.6.2 During the Steam Generator repair effort, a seismic occurrence is not required to be considered since all fuel shall have been removed from containment.

3.6.3 The bridge evaluation assumes that the bridge structure is in good condition, and that there is no deterioration which could reduce the structure's strength; or that if deterioration is found, it is either corrected, or the effects of the deterioration are evaluated and found to be acceptable.

3.6.4 The runway supporting structure evaluation assumes that the structure is in good condition, and that there is no deterioration which could reduce it's load carrying capability; or that if deterioration is found, it is either corrected, or the effects of the deterioration are evaluated and found to be acceptable.

3.6.5 It is assumed that the criteria of ASME B30.2, Section 2-3.2.1.1, items d and g, on crane inspections (but including only those applicable to the bridge), and items e and h, on lift procedures and lift records, are to be addressed elsewhere by Virginia Power.

4.0 CONCLUSIONS AND RECOMMENDATIONS

4.1 Conclusions

Based upon the criteria of ASME B30.2, and in accordance with this letter report, which includes a review of the crane service history, an evaluation of the crane bridge, an evaluation of the crane supporting structure, and consideration for crane load testing; it is concluded that the existing crane bridge is qualified for making the required 280 ton special over rated-load lifts for steam generator replacement without any modifications to the bridge, and without the need for a crane load test, other than that specified by Paragraph (f) of Section 2-3.2.1.1 of ASME B30.2.

This conclusion is based upon the assumption that the criteria of ASME B30.2, Section 2-3.2.1.1 items d and g, on crane inspections (but including only those applicable to the bridge), and items e and h, on lift procedures and lift records, are to be addressed elsewhere. (See Assumption 3.6.5).

4.2 Recommendations

4.2.1 It is recommended that the bridge inspections, that are required to be made before and after the special over rated-load lifts (see Conclusion Section 4.1 and Assumption 3.6.5), be in accordance with the Stone & Webster Recommended Crane Inspections transmitted to Virginia Power by letter NAS-19, 731 (Ref. 5.16).

4.2.2 With regard to testing by means of lifting the actual load, Item (f) of Section 2-3.2.1.1 of ASME B30.2, it is recommended that the specified short distance be approximately 6 inches, and that the load be suspended for approximately 5 minutes.

5.0 REFERENCES

5.1 ASME B30.2-1990, Overhead and Gantry Cranes (Top Running Bridge, Single or Multiple Girder, Top Running Trolley Hoist)

5.2 Virginia Power letter NP-1486-S07-004, dated July 30,1990, from R.C. Carroll Jr. to D.E. McLellan

5.3 Procedure for Setting the Unit No. 2 Steam Generators, Field Procedure No. 58, North Anna Power Station, May 8, 1973

5.4 Specification for Reactor Containment Crane for North Anna Power Station, NAS-142, through Revision 3 dated June 10, 1977.

5.5 Stone & Webster Quality Control Inspection Report on Testing the Unit 1 Polar Crane Main Hoists, January 1973

5.6 Procedure for Testing the Unit 1 and 2 Reactor Containment Polar Cranes, 12/6/72

5.7 Stone & Webster letter, March 11, 1977, To J.V. Harrison from W.M. Sweeter, Test Certificate Report for Unit 1 Reactor Containment Polar Crane

5.8 North Anna Unit 1 Work Orders on the Reactor Containment Polar Crane, Mark # 01-MH-CR-1, from 08/22/78 to 12/18/89

5.9 Harnischfeger Corporation, Uprate Calculation for Crane No. C-23860 and 61, dated December 16, 1976, Stone & Webster approved April 6, 1977.

5.10 Harnischfeger Corporation, Calculation for Crane No. C-23860 and 61, dated April 1971, Stone & Webster approved November 7, 1971

5.11 CMAA Specification # 70, Revised 1988, Specifications for Electric Overhead Traveling Cranes

5.0 REFERENCES (Continued)

5.12 Stone & Webster Dwg. No. 11715-FC-16Z, Crane Wall - Elev. & Details, sheet 5, Interior Concrete Reactor Containment, Rev. 6

5.13 ACI Standard 318-71, Building Code Requirements for Reinforced Concrete

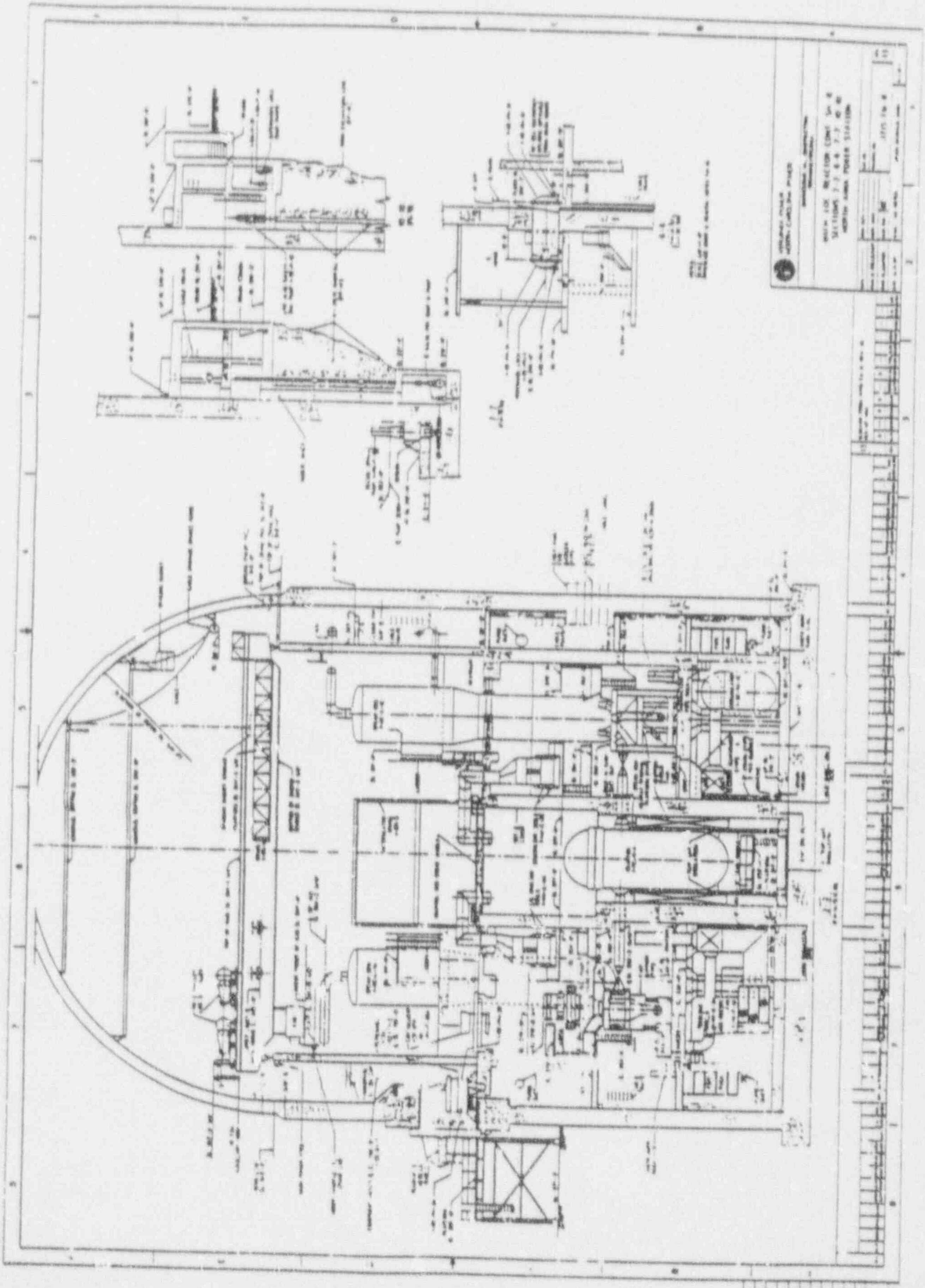
5.14 Stone & Webster Concrete Calculation 11715-BK-5B, Section 4B, Anchors for Polar Crane Rail, 2/22/72

5.15 Stone & Webster Concrete Calculation 11715-BK-5B, Section 8A, Crane Wall Investigation of Stability for Construction Condition, 7/12/72

5.16 Stone & Webster letter, NAS-19,731, August 10, 1990, to R.C. Carroll, Jr. from D.E. McLellan, including Transmittal of Stone & Webster Recommended Crane Inspections (Periodic), Reactor Containment Crane, Prior to Steam Generator Replacement, North Anna Power Station, Revision 1 DRAFT

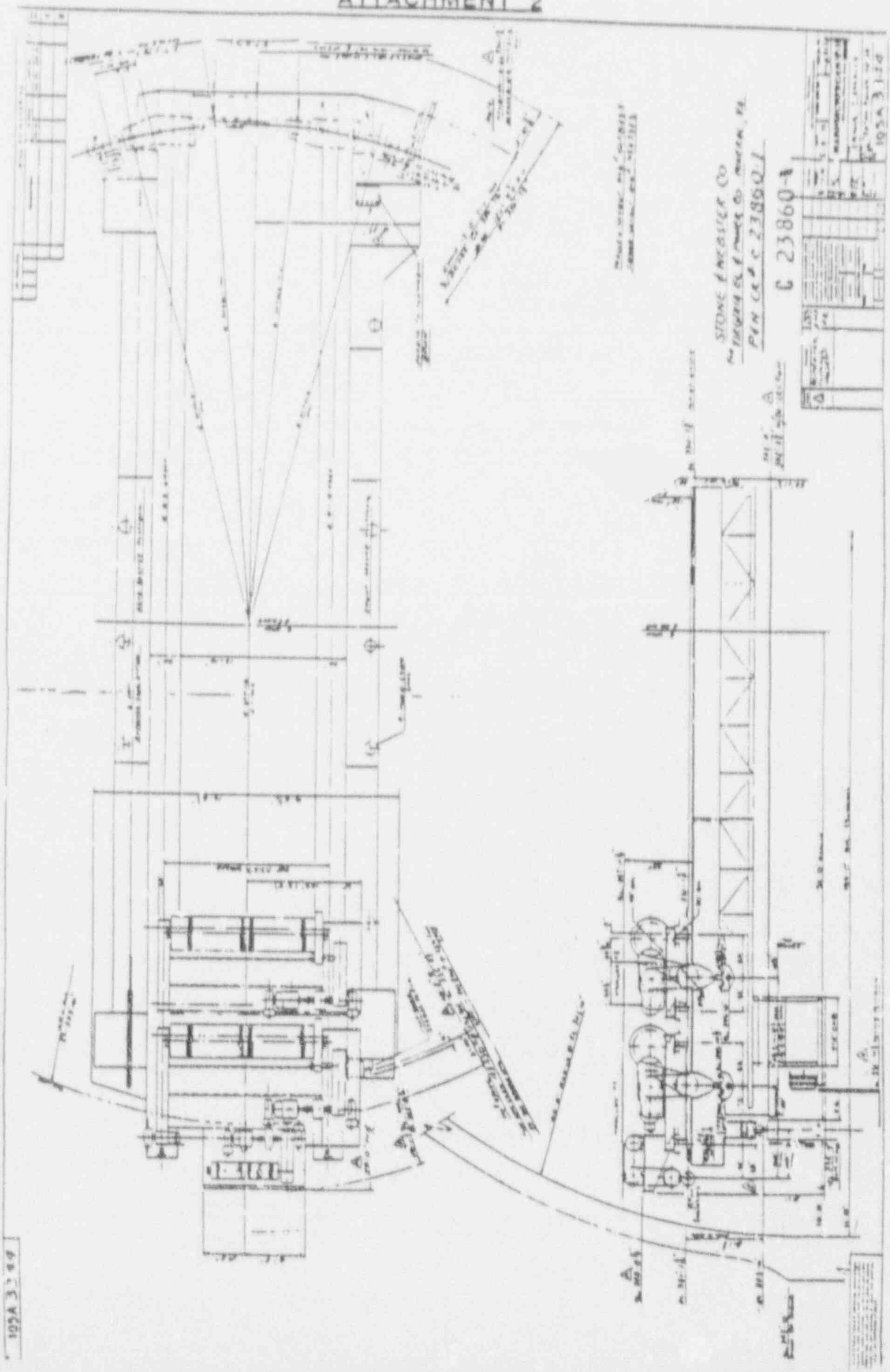
J.O. No. 01040.0810

ATTACHMENT 1



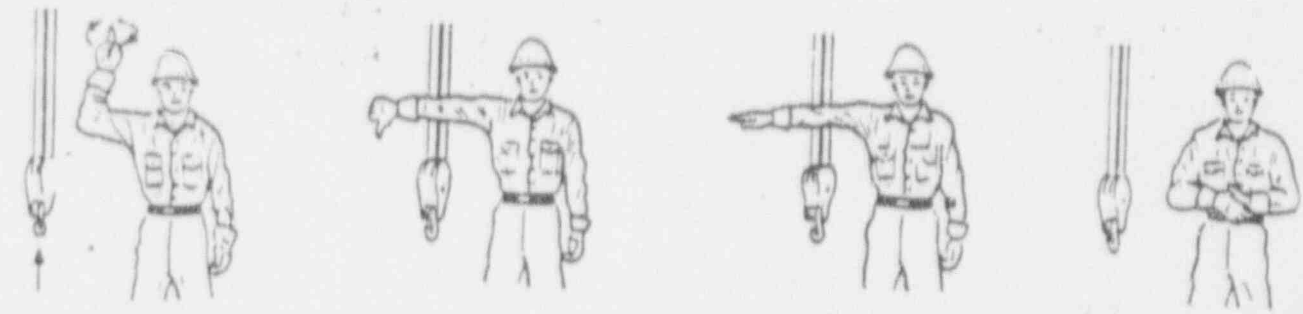
J.O. No. 01040.0 0

ATTACHMENT 2



J.O. No. 01040.0810

ATTACHMENT 3

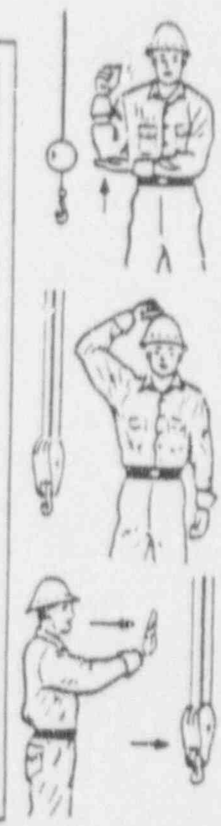


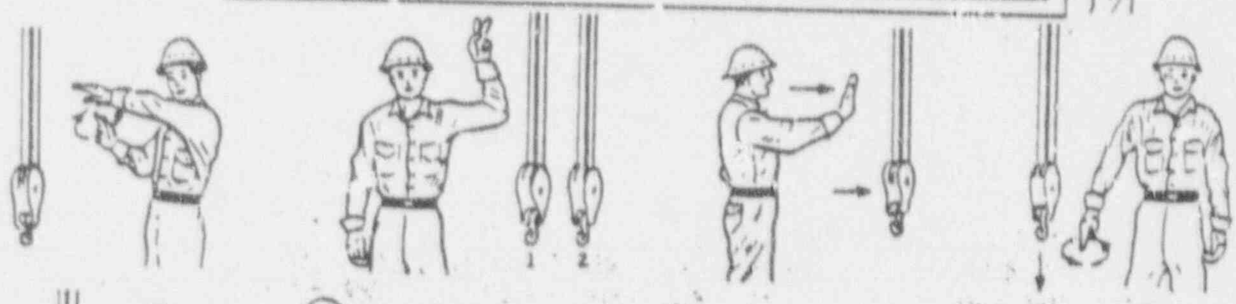
ASME B30.2-1990

(REVISION OF ANSI/ASME B30.2-1983)

OVERHEAD AND GANTRY CRANES (Top Running Bridge, Single or Multiple Girder, Top Running Trolley Hoist)

AN AMERICAN NATIONAL STANDARD





The American Society of Mechanical Engineers

ATTACHMENT 3 (cont.)

2-3.1.7 Conduct of Operators

(a) The operator shall not engage in any practice that will divert attention while actually engaged in operating the crane.

(b) When physically or otherwise unfit, an operator shall not engage in the operation of the equipment.

(c) The operator shall respond to signals from the person who is directing the lift, or an appointed signalperson. When a signalperson or a crane follower is not required as part of the crane operation, the operator is then responsible for the lifts. However, the operator shall obey a stop signal at all times, no matter who gives it.

(d) Each operator shall be responsible for those operations under the operator's direct control. Whenever there is doubt as to safety, the operator shall consult with the supervisor before handling the loads.

(e) If a warning device is furnished, it shall be activated each time before traveling, and intermittently when approaching workpersons.

(f) Before leaving a cab-operated crane unattended, the operator shall land any attached load, place controllers in the off position, and open the main line disconnect device of the specific crane.

(90) *(g)* The operator shall not close the main switch (crane disconnect) until certain that no worker is on or adjacent to the crane. If there is a warning sign or lock on the device, it shall not be energized until the sign or lock is removed by the person who placed it there, or by an authorized person.

(90) *(h)* Before closing the main switch (crane disconnect), the operator shall be sure that all controllers are in the off position.

(i) If power goes off during operation, the operator shall immediately place all controllers in the off position. Prior to re-use of the crane, operating motions shall be checked for proper direction.

(j) The operator shall be familiar with the equipment and its proper care. If adjustments or repairs are necessary, or any defects are known, the operator shall report the same promptly to the appointed person who shall be responsible for the operation and maintenance repairs of the crane. The operator shall also notify the next operator of any remaining uncorrected defects upon changing shifts.

(k) Contacts with runway stops or other cranes shall be made with extreme caution. The operator shall do so with particular care for the safety of persons on or below the crane, and only after making certain that any persons on the other cranes are aware of what is being done.

(l) Operators of outdoor cranes shall secure them when leaving.

(m) When the wind-indicating device of a cab-operated outdoor crane gives the alarm, crane operation shall be discontinued and the crane shall be prepared and stored for excessive wind conditions. (90)

(n) Before the operator performs any maintenance work on the crane, the operator shall lock, tag, or flag the main switch (see para. 2-2.3.2) in the de-energized position.

(o) All controls shall be tested by the operator before beginning a new shift. If any controls do not operate properly, they should be adjusted or repaired before operations are begun.

(p) Persons boarding or leaving overhead cranes should do so only at authorized locations and designated boarding entrances.

Section 2-3.2 — Handling the Load

2-3.2.1 Load Weight. The crane shall not be loaded beyond its rated load except for test purposes as provided in para. 2-2.2.2, or for special over rated-load lifts as provided in para. 2-3.2.1.1.

2-3.2.1.1 Special Over Rated-Load Lifts. Lifts in excess of the rated load may be required from time to time for specific purposes such as new construction or major repairs. Each instance of exceeding the rated load shall be treated as a specific problem and the following requirements shall be met for each instance. If special lifts are to be made frequently, the crane should be rerated for the load being handled (see para. 2-1.4.3 for rerating).

(a) A written review of the crane service history shall be prepared, including reference to previous over rated-load lifts, structural repairs, and modifications of original design.

(b) Structural, mechanical, and electrical components of the crane design shall be checked for the load to be lifted by a crane manufacturer or other qualified person, according to accepted crane design standards, such as CMAA #70 or AISE #6.

(c) The crane supporting structure design shall be checked for conformance to AISC or other applicable design criteria. The crane support shall be inspected and any deterioration or damage shall be taken into consideration in design calculations for the load to be lifted.

(d) A complete inspection of the crane as described in para. 2-1.1.3 shall be made just prior to making the lift. 2

(e) The lift shall be made under controlled con-

ATTACHMENT 3 (cont.)

ditions under the direction of a designated person in accordance with a previously prepared lift plan. All persons in the area of the crane runway shall be alerted.

(f) The operator shall test the crane at the special over rated-load by lifting the load a short distance and setting the brakes. Any failure to hold the load shall be corrected before proceeding with the lift.

(g) After the special over rated-load lift is concluded, an inspection shall be made in accordance with para. 2-2.1.3.

(h) A record of the over rated-load lift, including all distances moved, shall be placed on file where available to appointed personnel.

(i) The rated load test that is specified in para. 2-2.2.2 is not applicable to special over rated-load lifts.

2-3.2.2 Attaching the Load

(a) The hoist rope shall be free from kinks or twists and shall not be wrapped around the load.

(b) The load shall be attached to the load block hook by means of slings or other devices.

(c) Care shall be taken to make certain that the sling clears all obstacles.

2-3.2.3 Moving the Load

(a) The appointed person directing the lift shall ascertain that:

(1) the load, sling, or lifting device is seated in the bowl of the hook;

(2) the load is secured, balanced, and positioned in the hook, sling, or lifting device before the load is lifted more than a few inches (millimeters);

(3) the hoist rope is not kinked;

(4) multiple part lines are not twisted around each other;

(5) the hook is brought over the load in such a manner as to minimize swinging;

(6) the rope is seated in the drum grooves and in the sheaves, if there is or has been a slack rope condition.

(b) During lifting, care shall be taken that:

(1) there is no sudden acceleration or deceleration of the moving load;

(2) load does not contact any obstructions.

(c) Cranes shall not be used for side pulls except when specifically authorized by a qualified person who has determined that:

(1) the various parts of the crane will not be overstressed;

(2) the hoist rope will not bear or rub against other members of the crane, such as girders or trolley frame, except members specifically designed for such contact;

(3) such side pulls will not cause the hoist rope to be pulled out of the sheaves or across drum grooves;

(4) such side pulls will not result in excessive swinging of the load block or load.

(d) The operator shall not cause the crane to lift, lower, or travel while anyone is on the load or hook.

(e) The operator should avoid carrying loads over people.

(f) The operator of a floor-operated crane having a lifting magnet should exercise caution due to the hazard of possible falling metal.

(g) The operator shall check the hoist brake(s) at least once each shift if a load approaching the rated load is to be handled. This shall be done by lifting the load a short distance and applying the brake(s).

(h) The load shall not be lowered below the point where two wraps of rope remain on each anchorage of the hoisting drum unless a lower limit device is provided, in which case, no less than one wrap shall remain.

(i) When two or more cranes are used to lift a load, one qualified person shall be in charge of the operation. This person shall analyze the operation and instruct other personnel involved in the proper positioning, rigging of the load, and the movements to be made.

(j) The operator shall not leave the position at the controls while the load is suspended over an area accessible to people.

2-3.2.4 Hoist Limit Devices (Switches)

(a) Prior to the initial use of any hoist during each shift, the operator shall verify operation of the primary upper limit device under no-load conditions. Care shall be exercised; the block shall be inched into the limit or run in at slow speed. If the device does not operate properly, the operator shall immediately notify the appointed person.

(b) The hoist limit device that controls the upper limit of travel of the load block shall not be used as an operating control in normal operation unless additional means are provided to prevent damage from overtravel.

Section 2-3.3 — Signals

2-3.3.1 Standard Signals (90)

(a) Signals to the operator shall be in accordance with this volume, unless voice communication (telephone, radio, or equivalent) is utilized.

(b) Signals should be discernible or audible to the operator.

MATERIALS SUBSTITUTION RECONCILIATION

Material Substitution Reconciliation Summary

Steam Generator Lower Assembly

The replacement SG lower assemblies have been fabricated in compliance with the 1986 Edition of ASME Section III and have been "NPT" stamped. The Stress Report is based on the 1968 Edition of the ASME Section III, including all addenda through the Winter of 1968. For materials whose strength properties do not exist in the earlier edition, the material strength properties used in the Stress Report were determined from Section III, 1986 Edition. Additionally, in cases where the later edition specifies more conservative material strength properties than the earlier edition, the more conservative values were used in the Stress Report. Hence, the replacement SGs are fabricated and analyzed to standards which are, at a minimum, equivalent to the existing units. The design changes made to the new steam generator lower assemblies and their reconciliations are evaluated in the Westinghouse Safety Evaluation (Appendix 4-19).

Steam Generator Upper Restraint

The materials purchased and used in the replacement design of the steam generator upper restraint have been reconciled to the design code for North Anna Unit 1. The fabrication of the upper restraint was performed in accordance with Specification No. NAP-0C33. Later versions of codes and standards included as references in the specification were reconciled to the design codes and standards. Inspection and non-destructive examination techniques specified in the specification meet the requirements of the design code. Materials and their appropriate stress allowable limits have been utilized in the analysis that demonstrates the acceptability of the upper restraint. This qualification is documented in Calculation No. 02072.13-NP(B)-003-BA.

Steam Generator Lower Supports

A design code and nondestructive examination reconciliation was performed for the spare parts for the steam generator lower supports. A complete evaluation of spare parts material requirements and design code and examination reconciliation is contained in an attachment to SWEC letter no. 02072.2210 and is included in this appendix.

Main Steam Piping

Existing main steam piping material is SA155, Grade CMS75. Replacement piping that may be utilized if necessary is SA691, Grade CMS75 material. Allowable stress values for these two materials are the same and have no impact on the results of the stress analysis. This is documented in Calculation Nos. 13075.62-NP(B)-4-X2, 11715-X1-3, and 11715-X1-4. Additionally, the design conditions of the main steam system are unchanged.

Feedwater Loop Seal Elbows and Drain Valves

Replacement chrome-moly material A335 P22 will be used for the feedwater loop seal elbows. The replacement of carbon steel material with chrome-moly material was determined to be acceptable by the Materials Engineering Group in Innsbrook (see attached memorandum). This replacement material has an allowable stress value of 14500 psi which is less than the allowable limit of 15000 psi for the existing A106 Grade B elbows. This piping system has been evaluated considering the lower allowable value and has been demonstrated to be acceptable. Other mechanical properties including thermal coefficient of expansion and modulus of elasticity also vary, but have been determined to be have no impact on system acceptability. The qualification of the replacement feedwater loop seal elbows is documented in Calculation Nos. 13075.62-NP(B)-5-X2, 13075.62-NP(B)-38-X2, and 11715-X2-16.

The reconciliation of replacement feedwater drain valves (Conval Figure No. 3/4" 11G2J-105) determined that the weight of these valves is 11 pounds as compared to the weight of 5 pounds for the existing valves. This additional weight for the replacement valves was evaluated and demonstrated to be acceptable. the qualification is documented in Calculation Nos. 13075.62-NP(B)-5-X2, 13075.62-NP(B)-38-X2, and 11715 X2-16. The design conditions of the feedwater system remain unchanged and the ANSI B16.34 rating of the valves exceed the system design conditions.

Steam Generator Blowdown Piping, Valves, Reducers and Tees

The SG blowdown system modification was designed to the original code. The replacement of carbon steel material with chrome-moly material was determined to be acceptable by the Materials Engineering Group in Innsbrook (see attached memorandum).

Replacement chrome-moly material A335 P22 will be used for the steam generator (SG) blowdown piping, material A234 WP22 will be used for the SG blowdown elbows, reducers and tees, and material A182 Grade F22 will be used for the valves. This replacement material has an allowable stress value of 14500 psi which is less than the allowable limit of 15000 psi for the existing A106 Grade

B piping, elbows, etc. This piping system has been evaluated considering the lower allowable value and has been demonstrated to be acceptable. Other mechanical properties including thermal coefficient of expansion and modulus of elasticity also vary, but have been determined to be have no impact on system acceptability. The qualification of the steam generator blowdown piping is documented in Calculation Nos. 02072.07-NP(B)-001-XE, 02072.07-NP(B)-002-XE, and 02072.07-NP(B)-003-XE.

Wet Layup and Sample System Piping

The design conditions of the wet layup and sample systems remain unchanged. The code reconciliation of SA106 Grade B and ASTM A106 Grade B has indicated that the stress allowables are unchanged and that the weight variance permitted has increased from not more than 6.5 percent over that specified to not more than 10 percent over that specified. Piping analyses have been performed in accordance with the design code that specifies nominal dimensions and weights to be utilized. Nominal dimensions and weights have not changed for this material. In addition, the amount and lengths of replacement material is not significant. Therefore, as documented in Calculation No. 02072.13-NP(B)-004-X, this code change for SA106/A106 material is acceptable and does not impact the qualification of wet layup or sample system piping.

SG Level Instrumentation Piping, Root Valves and Condensate Pots

The design conditions of the SG level instrumentation system remain unchanged. The existing carbon steel level instrumentation piping, root valves, and condensate pots are being replaced with stainless steel material manufactured to later editions of ASTM/ASME material specifications and ANSI standards. A reconciliation of the replacement material stress allowables and design ratings demonstrates that the replacement materials meet or exceed the requirements of the original design codes and the original design requirements. This evaluation is documented in Calculation 21809-M-03.

ATTACHMENT 1

EVALUATION OF SPARE PARTS MATERIAL REQUIREMENTS
AND DESIGN CODE AND EXAMINATION RECONCILIATION
FOR THE STEAM GENERATOR LOWER SUPPORT
STEAM GENERATOR REPAIR PROJECT - NORTH ANNA UNIT 1

The following pages contain a design code and nondestructive examination reconciliation for spare parts for the steam generator lower supports for North Anna Unit 1. Reference to item numbers in Attachment 1 utilizes the item numbering sequence from Drawing No. 11715-FV-17_.

This is an attachment from Stone and Webster Engineering Corporation letter no. 02072.2210 from D. E. McLellan to M. W. Gettler, dated August 20, 1992.

EVALUATION OF SPARE PARTS MATERIAL REQUIREMENTS
AND DESIGN CODE AND EXAMINATION RECONCILIATION
FOR THE STEAM GENERATOR LOWER SUPPORT
STEAM GENERATOR REPAIR PROJECT - NORTH ANNA UNIT 1

AISI 4340	This material is used for the vertical support plate (Item 7). Heat treatment has been indicated as required to obtain 140 ksi to 160 ksi yield strength mechanical properties. This provides for the replacement material to be of equivalent strength to the original material. This piece is to be examined by ultrasonic test in accordance with ASME Section III, Subsection NF, Paragraph NF-5111 and Subsubarticle NF-5330. The material will be purchased in accordance with the latest material specification requirements. Any variance between examination practices or acceptance standards provided in the original and current material specifications and codes will not impact the design function or capability of the steam generator (SG) lower support. This material should be marked by the electrochemical etch method only.
ASTM A36	This material is used for the shims (Items 23, 24, 25, 26, 27, 29 and 30) and the mating piece for the Lubrite plates (Items 9 and 10). The original material requirements were for AISI C1018 steel or equivalent for shims and ASTM A36 or AISI 4340 for the mating piece. ASTM A36-91 is considered equivalent to AISI C1018 for the intended function of shims. It is also considered equivalent to AISI 4340 for the function of the Lubrite mating piece. Since these items are loaded only in compression, any variance in chemical or mechanical requirements, including minimum tensile strength, does not impact the design function of the support. No nondestructive examination is required for the plate material to be used as shims. The material used for the Lubrite mating pieces should have magnetic particle or liquid penetrant examination performed.

This is an attachment from Stone and Webster Engineering Corporation letter no. 02072.2210 from D. E. McLellan to M. W. Gettler, dated August 20, 1992.

ATTACHMENT 1

EVALUATION OF SPARE PARTS MATERIAL REQUIREMENTS
AND DESIGN CODE AND EXAMINATION RECONCILIATION
FOR THE STEAM GENERATOR LOWER SUPPORT
STEAM GENERATOR REPAIR PROJECT - NORTH ANNA UNIT 1

ASTM A574	This material is used for the socket head cap screws (Item 11). The current ASTM A574-90 requirements for mechanical properties, chemical requirements, workmanship and finish are identical to or exceed the ASTM A574-67 requirements. The decarburization limit has changed slightly (3/4 of basic thread height versus 2/3 of basic thread height) and additional limits have been placed on the discontinuities. Both of these changes are acceptable. The required length of these cap screws and the required engagement length of them into the screw thread inserts (HeliCoils) and steam generator lower support feet is unchanged for service following the completion of the steam generator repair project.
Thread Inserts	Screw thread inserts (Item 12) utilized in conjunction with the Item 11 socket head cap screws were originally specified as HeliCoil Type 3585 (Screw-Lock type, Unified coarse threads). Type 3591 (Screw-Lock type, Unified fine threads) have been recommended as replacement parts. These Black & Decker stainless steel screw thread inserts were recommended to be thread compatible with the Item 11 cap screws. Westinghouse Drawing No. 6142E90 also identifies HeliCoil Types 3591 and 1191 (Standard type, Unified fine threads) as suitable parts for use. The required length of these screw thread inserts and the required engagement length of them into the socket head cap screws and steam generator lower support feet is unchanged for service following the completion of the steam generator repair project.

This is an attachment from Stone and Webster Engineering Corporation letter no. 02072.2210 from D. E. McLellan to M. W. Gettler, dated August 20, 1992.

ATTACHMENT 1

EVALUATION OF SPARE PARTS MATERIAL REQUIREMENTS
AND DESIGN CODE AND EXAMINATION RECONCILIATION
FOR THE STEAM GENERATOR LOWER SUPPORT
STEAM GENERATOR REPAIR PROJECT - NORTH ANNA UNIT 1

ASTM B152	This material is used for the washers (Item 28) utilized in conjunction with the Item 11 socket headed cap screws and Item 12 screw thread inserts. The current ASTM B152-88 requirements for annealed material has insignificant changes from ASTM B152-66 requirements for annealed material. The chemical composition, mechanical strength, and hardness remain identical. The material designation now corresponds to the UNS system (C10200). Resistivity tests and embrittlement tests are not required.
ASTM A354 Gr BD	The original standard specification for the 2 inch hex head bolts (Items 14 and 15) was ASTM A490 (1967). However, the current (1991) edition of the ASTM standard only covers bolts from 1/2 inch to 1 1/2 inches, inclusive, in diameter. Quenched an' tempered 2 inch alloy steel bolts are now covered in Specification A354 Gr. BD. The chemical and mecnanical requirements of ASTM A354 Gr. BD (1991) are no less stringent than those of ASTM A490 (1967). Torquing requirements for these replacement bolts are the same as for the original A490 bolts (snug tight plus 1/2 turn (30 degrees)). Retorquing of these A354 Gr. BD bolts or the original A490 bolts after they have been fully torqued - not recommended.
ASTM A563 Gr DH	This material is used for the heavy hex nuts (Item 18). The current ASTM A563 91c specification has a lower proof load stress than the earlier edition ASTM A563-66 (175 ksi versus 200 ksi). The reason for this relaxation is the nut requirements were made to agree with the bolt requirements. Since the bolt requirements for this design have not changed and the strength of the nut material is not the critical design parameter, the relaxation is acceptable for this application.

This is an attachment from Stone and Webster Engineering Corporation letter no. 02072.2210 from D. E. McLellan to M. W. Gettler, dated August 20, 1992.

VIRGINIA POWER
NORTH CAROLINA POWER

Memorandum

To: R. K. Bayer, NAPS-SGRP

From: R. W. Calder, IN/3NW

Innsbrook Technical Center

July 21, 1992

SGRP MATERIALS SUBSTITUTION
Steam Generator Replacement Project
(North Anna Power Station, Unit 1)

As part of the Steam Generator Replacement Design Change Package, the main feedwater loop seal and the blowdown piping from the steam generator to the 3 inch header will be replaced. The existent corroded carbon steel piping (pipe class 601) is scheduled to be replaced with FAC[1] resistant chrome-molybdenum alloy steel (class 601C). Materials Engineering was asked if the proposed replacement alloy 601C was acceptable from a materials standpoint.

There is no evidence that significant galvanic corrosion occurs at 601/ 601C interfaces at other NAPS locations. Welding procedures are already in place for 601C. 601C will not undergo appreciable radiation embrittlement in this application. If the elevated temperature mechanical properties of 601C are adequate for your purpose (per Stone and Webster pipe stress analysis)[2], Materials Engineering has no objection substituting 601C for 601.

R. W. Calder

COPY:
J. I. Bennetch, JIB IN/3NW
I. L. Breedlove, IN/1NW
A. T. Vig, A IN/3NW
S. L. Wilkie, NAPS-SGRP
Project File:
Records Management: (NP-1486), IN/GW

[1] Flow accelerated corrosion

[2] Virginia Power memo to R. W. Calder from R. K. Bayer, July 15, 1992

FOR INFORMATION

VIRGINIA POWER COMPANY
ENGINEERING TECHNICAL BULLETIN

Bulletin No: MA - 1

___ Action ___ X Information

Manager ISI/NDE & Engineering Programs Date: April 10, 1990

Subject: Chrome-moly pipe welding

Station: Surry and North Anna Unit(s): 1 & 2

REFERENCES:

None

DISCUSSION:

(See Attached)

Action Parties: Station Managers, Discipline Managers, Supervisors and A/Es

Date Response Required: N/A

By: L N Hartz Date: 4/12/90

L. N. Hartz
Manager - ISI/NDE & Engineering Programs

DISCUSSION SECTION FOR TECHNICAL BULLETIN MA-1

The leak before break fracture mechanics analysis and materials testing for the chrome-moly feedwater pipe replacement for Surry and North Anna (UFSAR update) has revealed lower than expected toughness values at room temperature. The low toughness values cause a concern during hydrostatic testing of the lines welded with the shielded metal arc process and E9018-B3 electrodes. With a large flaw size in the weld metal and without preheating the hydrostatic test medium, a condition exists that may cause a failure of a pipe weld during the hydro test. There is no evidence at this time to indicate that a problem exist with these welds during any other condition of plant operation.

To eliminate any possibility of a failure resulting from this concern, a volumetric and a surface examination should be performed on all future chrome-moly groove welds over 1/2" thick, made with the above process and filler metal. This is in addition to any other ANSI B31.1 requirements. Since hydrogen cracking is not a concern the volumetric examination may be performed on welds immediately following completion of the welding while the surface examination should be performed after any stress relieving process.

The volumetric examination need not be concerned with root geometry since the defect must be greater than four inches long and one inch in depth to create a potential failure. The following should be cause for rejection of chrome-moly welds not requiring volumetric examinations:

1) Any cracking.
2) Elongated indications greater than one inch in length.
3) Any group of indications in a line that have an aggregate length greater than three inches except when the distance between the successive indications exceeds four inches.

Based on the information to date there is no need to perform the above inspections on lines already in service unless the line is to be hydrostatically tested. This information is based on preliminary results of the fracture mechanics analysis. All future DCP'S and EWR'S dealing with the installation of chrome-moly piping should address this concern.

JUSTIFICATION FOR WELDED PIPE SADDLES

Innsbrook Technical Center
Nuclear Electrical Engineering
5000 Dominion Boulevard
Glen Allen, Virginia 23060

BECHTEL
RECEIVED
MAY 13 19__

VIRGINIA POWER

May 12, 1992
PR-BV2001-VP010

North Anna

Mr. R. L. Miller
Bechtel Power Corporation
9801 Washingtonian Blvd.
Gaithersburg, MD 20878-5356

Attention: Mr. C. D. Brown
Project Engineer

S/G BLOWDOWN DRAWINGS AND TEXT COMMENTS
STEAM GENERATOR REPLACEMENT PROJECT
NORTH ANNA POWER STATION - UNIT 1

Gentlemen:

Enclosed please find the Rev. 0, Issued for Construction, drawings for the blowdown portion of DC-90-13-1 as listed on the attached sheets. Also attached are comments received from Stone & Webster on the text portion of DC-90-13-1 and Virginia Power approval letter for use of saddles on the blowdown piping. The exception to our specification NAS 1009 should be noted in the text and this letter attached and referenced. Please incorporate the drawings and text comments into the Design Change Package prior to issuing the 70% draft.

If you have any question, please contact Sheri Wilkie at (703) 894-8010.

Sincerely,

R. K. Bayer
Project Engineer Supv.

cc: B. P. Reilly
E. L. Geiger

File

NOTED APR 27 1992 L.E.V.

VIRGINIA POWER
NORTH CAROLINA POWER

Memorandum

To: Ms. Sheri Wilkie NAPS

Innsbrook Technical Center

From: A. T. Vig - IN/3NW

April 15, 1992

STONE & WEBSTER WELDED SADDLE PROCESS
STEAM GENERATOR REPLACEMENT PROJECT
NORTH ANNA POWER STATION UNIT 1

Materials Engineering has reviewed the S&W proposal for carbon steel saddles welded to 2 1/4 Cr - 1 Mo pipe. This will be acceptable for the Steam Generator Blow Down pipe replacement portion of this project provided the pipe diameter is limited to a maximum 4" and the pipe wall thickness is limited to a maximum 1/2". This deviation to NAS 1009 should be allowed for this project. NAS 1009 should not be permanently changed to allow uncontrolled use of different alloys without engineering approval. E8018, ER70S, or ER80S filler material would also be acceptable for this application. Welding procedures will have to be properly qualified and approved prior to fabrication. If you have any questions contact me at extension 2658 in Innsbrook.

Allen T. Vig

A. T. Vig

pc: Mr. R. W. Calder - IN/3NW

SECONDARY SIDE PRESSURE TEST TEMPERATURE (WCAP-13453)

THIS ATTACHMENT HAS BEEN WITHHELD SINCE THERE IS INFORMATION IN THIS ATTACHMENT THAT IS PROPRIETARY TO WESTINGHOUSE. THE DOCUMENT IS MARKED AS WESTINGHOUSE PROPRIETARY CLASS II.

UNIT 1 - 1993 OUTAGE BASELINE INSPECTION LIST, SECONDARY PIPING INSPECTION PROGRAM

Memorandum

VIRGINIA POWER
NORTH CAROLINA POWER

To Mr. T. Travis September 2, 1992

From R. Lee Design Engineering

UNIT 1 - 1993 OUTAGE BASELINE INSPECTION LIST
SECONDARY PIPING INSPECTION PROGRAM
NORTH ANNA POWER STATION

Attached for your use is a copy of the list of components to be baseline inspected during the 1993 Unit 1 outage as part of the Secondary Piping Inspection Program.

The attached list includes feedwater, main steam, and steam generator blowdown components that will be replaced or removed and then reinstalled under Design Change 90-13-1 "Steam Generator Repair." Copies of the isometric drawing showing the location of the baseline components are also attached.

If you have any questions, please call me at extension 2276.

R. Lee

R. Lee

cc: Mr. I. L. Breedlove
Mr. J. C. Temple
Mr. C. M. Hooper
Mr. W. W. Chaisson
Mr. R. K. Bayer
Ms. S. L. Wilke

Form No. 720003 (Aug 88)
(Formerly 870240201)

SECONDARY PIPING INSPECTION PR[illegible]

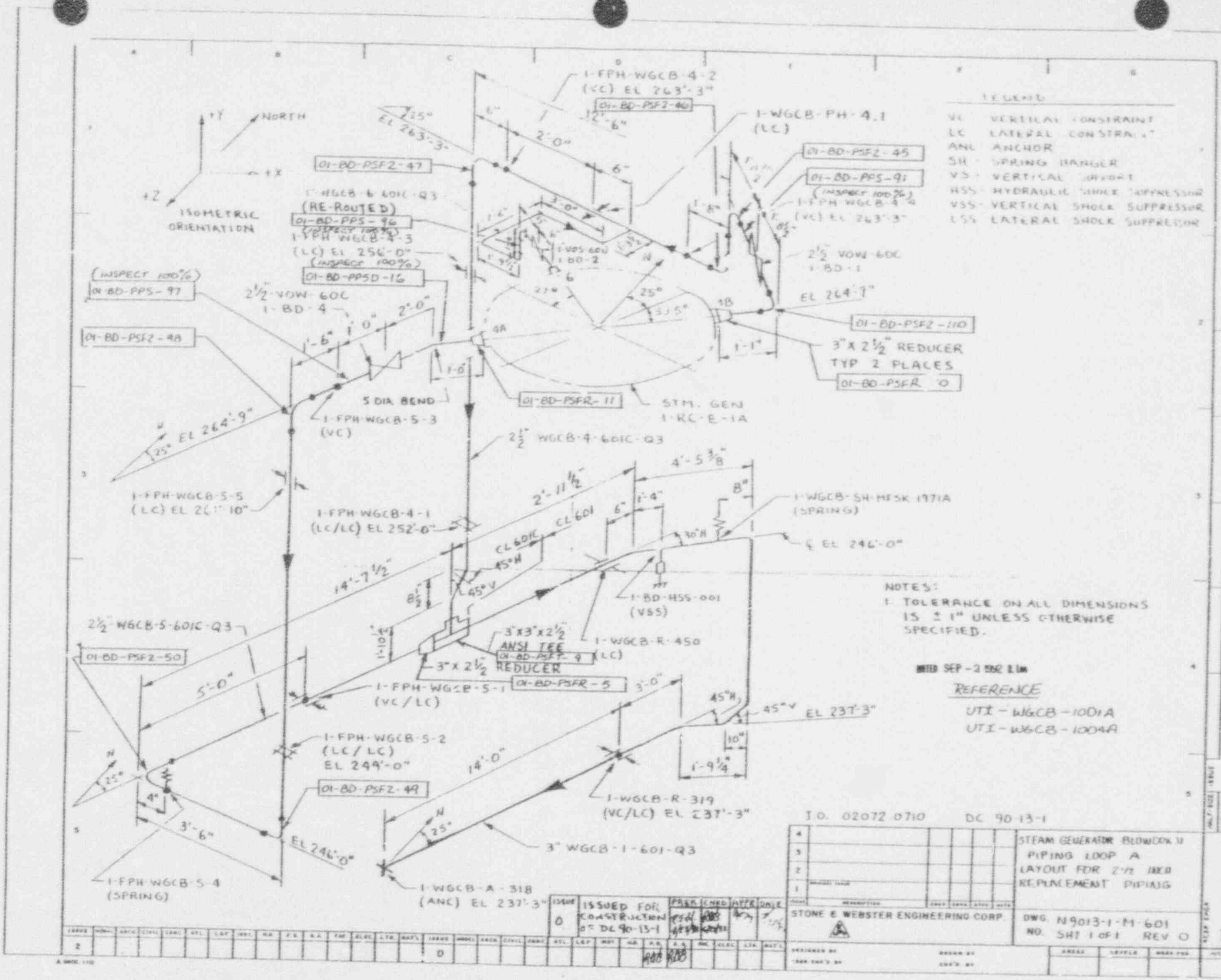
NORTH ANNA - UNIT 1

1993 OUTAGE BASELINE INSPECTION LIST

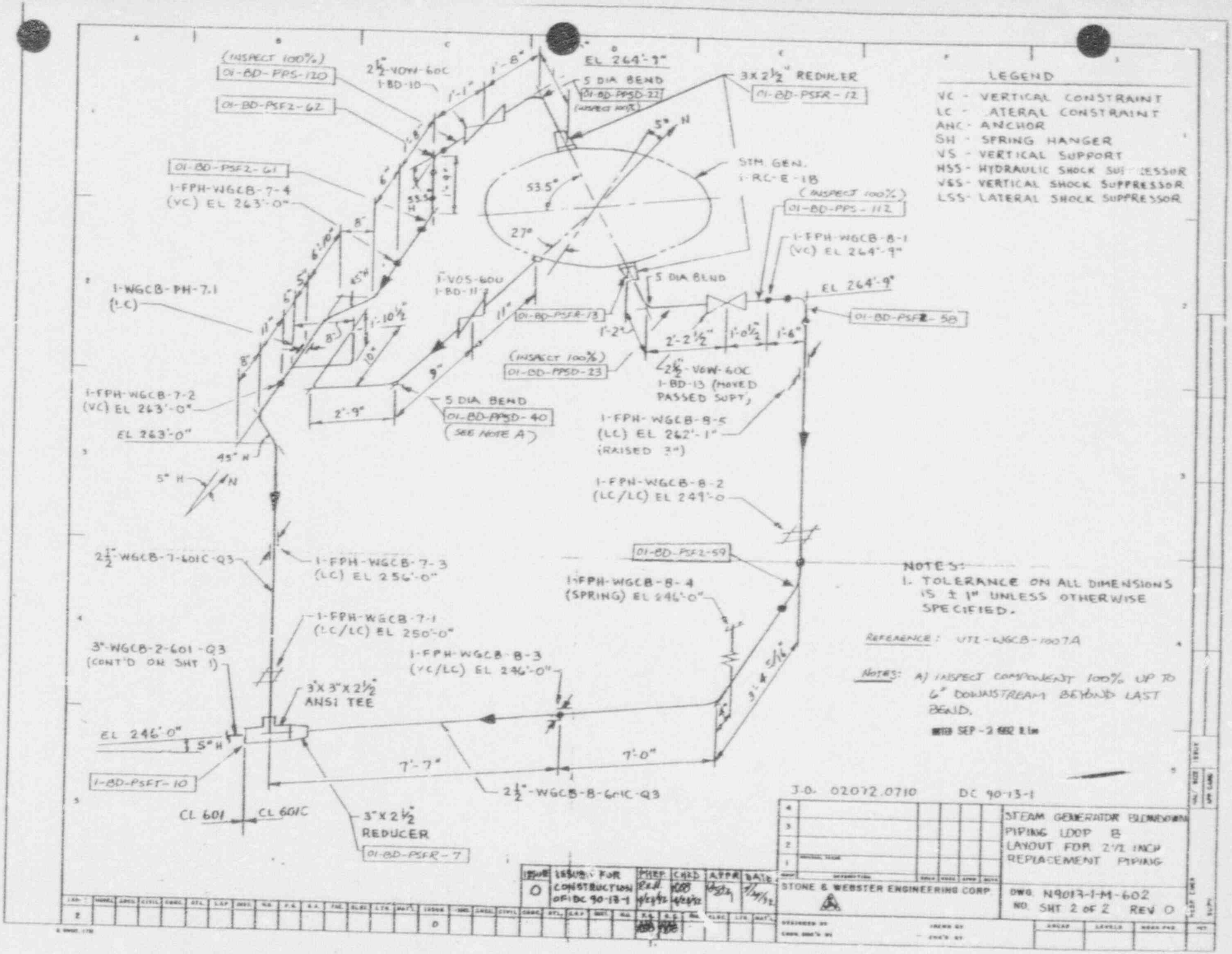
Component	Line	Sketch	Scope	Phase	System	File Name
1-BD-PSFR-11	2.5-WGCB-5-601C-Q3	N9013-1-M-601	O	S	BD	BDPSFR11
1-BD-PP5D-16	2.5-WGCB-5-601C-Q3	N9013-1-M-601	O	S	BD	BDPP5D16
1-BD-PSF2-48	2.5-WGCB-5-601C-Q3	N9013-1-M-601	O	S	BD	BDPSF248
1-BD-PSF2-49	2.5-WGCB-5-601C-Q3	N9013-1-M-601	O	S	BD	BDPSF249
1-BD-PSF2-50	2.5-WGCB-5-601C-Q3	N9013-1-M-601	O	S	BD	BDPSF250
1-BD-PSFR-5	2.5-WGCB-5-601C-Q3	N9013-1-M-601	O	S	BD	BDPSFR5
1-BD-PSFT-9	3-WGCB-1-601-Q3	N9013-1-M-601	O	S	BD	BDPSFT9
1-BD-PSFR-10	2.5-WGCB-4-601C-Q3	N9013-1-M-601	O	S	BD	BDPSFR10
1-BD-PSF2-110	2.5-WGCB-4-601C-Q3	N9013-1-M-601	O	S	BD	BDSF2110
1-BD-PSF2-45	2.5-WCGB-4-601C-Q3	N9013-1-M-601	O	S	BD	BDPSF245
1-BD-PSF2-46	2.5-WGCB-4-601C-Q3	N9013-1-M-601	O	S	BD	BDPSF246
1-BD-PSF2-47	2.5-WGCB-4-601C-Q3	N9013-1-M-601	O	S	BD	BDPSF247
1-BD-PPS-96	1-WGCB-6-601C-Q3	N9013-1-M-601	O	S	BD	BDPPS96
1-BD-PSFR-12	2.5-WGCB-7-601C-Q3	N9013-1-M-602	O	S	BD	BDPSFR12
1-BD-PP5D-22	2.5-WGCB-7-601C-Q3	N9013-1-M-602	O	S	BD	BDPP5D22
1-BD-PPS-120	2.5-WGCB-7-601C-Q3	N9013-1-M-602	O	S	BD	BDPPS120
1-BD-PSF2-62	2.5-WGCB-7-601C-Q3	N9013-1-M-602	O	S	BD	BDPSF262
1-BD-PSF2-61	2.5-WGCB-7-601C-Q3	N9013-1-M-602	O	S	BD	BDPSF261
1-BD-PSFR-13	2.5-WGCB-8-601C-Q3	N9013-1-M-602	O	S	BD	BDPSFR13
1-BD-PP5D-23	2.5-WGCB-8-601C-Q3	N9013-1-M-602	O	S	BD	BDPP5D23
1-BD-PPS-112	2.5-WGCB-8-601C-Q3	N9013-1-M-602	O	S	BD	BDPPS112
1-BD-PSF2-58	2.5-WGCB-8-601C-Q3	N9013-1-M-602	O	S	BD	BDPSF258
1-BD-PSF2-59	2.5-WGCB-8-601C-Q3	N9013-1-M-602	O	S	BD	BDPSF259
1-BD-PSFR-7	2.5-WGCB-8-601C-Q3	N9013-1-M-602	O	S	BD	BDPSFR7
1-BD-PSFT-10	3-WGCB-2-601-Q3	N9013-1-M-602	O	S	BD	BDPSFT10
1-BD-PP5D-40	1-WGCB-9-601C-Q3	N9013-1-M-602	O	S	BD	BDPP5D40
1-BD-PSFR-14	2.5-WGCB-10-601C-Q3	N9013-1-M-603	O	S	BD	BDPSFR14
1-BD-PP5D-1	2.5-WGCB-10-601C-Q3	N9013-1-M-603	O	S	BD	BDPP5D1
1-BD-PSF2-2	2.5-WGCB-10-601C-Q3	N9013-1-M-603	O	S	BD	BDPSF22
1-BD-PSF2-11	2.5-WGCB-[illegible]-601C-Q3	N9013-1-M-603	O	S	BD	BDPSF211
1-BD-PSFR-8	2.5-WGCB-10-601C-Q3	N9013-1-M-603	O	S	BD	BDPSFR8
1-BD-PSFT-11	3-WGCB-3-601-Q3	N9013-1-M-603	O	S	BD	BDPSFT11
1-BD-PSFR-15	2.5-WGCB-11-601C-Q3	N9013-1-M-603	O	S	BD	BDPSFR15
1-BD-PP5D-5	2.5-WGCB-11-601C-Q3	N9013-1-M-603	O	S	BD	BDPP5D5
1-BD-PSF2-4	2.5-WGCB-11-601C-Q3	N9013-1-M-603	O	S	BD	BDPSF24
1-BD-PP5D-41	1-WGCB-12-601C-Q3	N9013-1-M-603	O	S	BD	BDPP5D41
1-BD-PPS-198	1-WGCB-12-601C-Q3	N9013-1-M-603	O	S	BD	BDPPS198
1-BD-PPS-26	2.5-WGCB-11-601C-Q3	N9013-1-M-603	O	S	BD	BDPPS26
1-BD-PPS-19	2.5-WGCB-10-601C-Q3	N9013-1-M-603	O	S	BD	BDPPS19

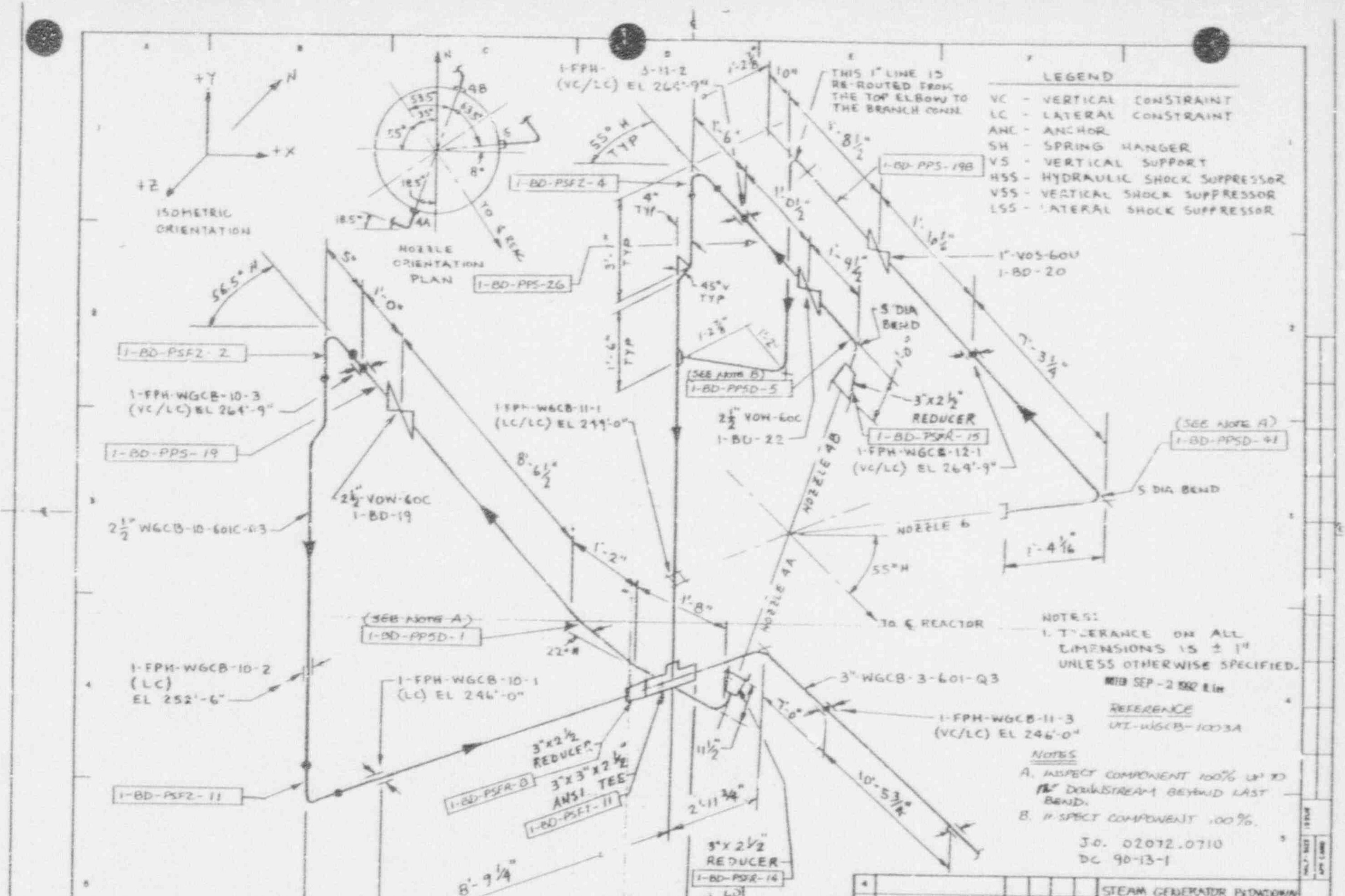
SECONDARY PIPING INSPECTION PROGRAM
NORTH ANNA - UNIT 1
1993 OUTAGE BASELINE INSPECTION LIST

Component	Line	Sketch	Scope	Phase	System	File Name
1-BD-PPS-97	2.5-WGCB-5-601C-Q3	N9013-1-M-603	O	S	BD	BDPPS97
1-BD-PPS-91	2.5-WGCB-4-601C-Q3	N9013-1-M-603	O	S	BD	BDPPS91
1-MS-PSF2-19	32-SHP-1-601-Q2	UTI-SHP-1001A	O	S	MS	MSPSF219
1-MS-PSF2-22	32-SHP-2-601-Q2	UTI-SHP-1002A	O	S	MS	MSPSF222
1-MS-PSF2-25	32-SHP-3-601-Q2	UTI-SHP-1003A	O	S	MS	MSPSF225
1-FW-PSF2-60	16-WFPD-22-601-Q2	UTI-WFPD-1022A	O	S	FW	FWPSF260
1-FW-PSF2-61	16-WFPD-22-601-Q2	UTI-WFPD-1022A	O	S	FW	FWPSF261
1-FW-PSF2-62	16-WFPD-22-601-Q2	UTI-WFPD-1022A	O	S	FW	FWPSF262
1-FW-PSF2-63	16-WFPD-22-601-Q2	UTI-WFPD-1022A	O	S	FW	FWPSF263
1-FW-PPS-31	16-WFPD-22-601-Q2	UTI-WFPD-1022A	O	S	FW	FWPSF231
1-FW-PPS-32	16-WFPD-22-601-Q2	UTI-WFPD-1022A	O	S	FW	FWPPS32
1-FW-PSF2-66	16-WFPD-23-601-Q2	UTI-WFPD-1023A	O	S	FW	FWPSF266
1-FW-PSF2-67	16-WFPD-23-601-Q2	UTI-WFPD-1023A	O	S	FW	FWPSF267
1-FW-PSF2-68	16-WFPD-23-601-Q2	UTI-WFPD-1023A	O	S	FW	FWPSF268
1-FW-PSF2-69	16-WFPD-23-601-Q2	UTI-WFPD-1023A	O	S	FW	FWPSF269
1-FW-PPS-107	16-WFPD-23-601-Q2	UTI-WFPD-1023A	O	S	FW	FWPPS107
1-FW-PPS-33	16-WFPD-23-601-Q2	UTI-WFPD-1023A	O	S	FW	FWPPS33
1-FW-PSF2-72	16-WFPD-24-601-Q2	UTI-WFPD-1024A	O	S	FW	FWPSF272
1-FW-PSF2-73	16-WFPD-24-601-Q2	UTI-WFPD-1024A	O	S	FW	FWPSF273
1-FW-PSF2-74	16-WFPD-24-601-Q2	UTI-WFPD-1024A	O	S	FW	FWPSF274
1-FW-PSF2-75	16-WFPD-24-601-Q2	UTI-WFPD-1024A	O	S	FW	FWPSF275
1-FW-PPS-37	16-WFPD-24-601-Q2	UTI-WFPD-1024A	O	S	FW	FWPPS37
1-FW-PPS-175	16-WFPD-24-601-Q2	UTI-WFPD-1024A	O	S	FW	FWPPS175
1-FW-PPS-176	16-WFPD-24-601-Q2	UTI-WFPD-1024A	O	S	FW	FWPPS176

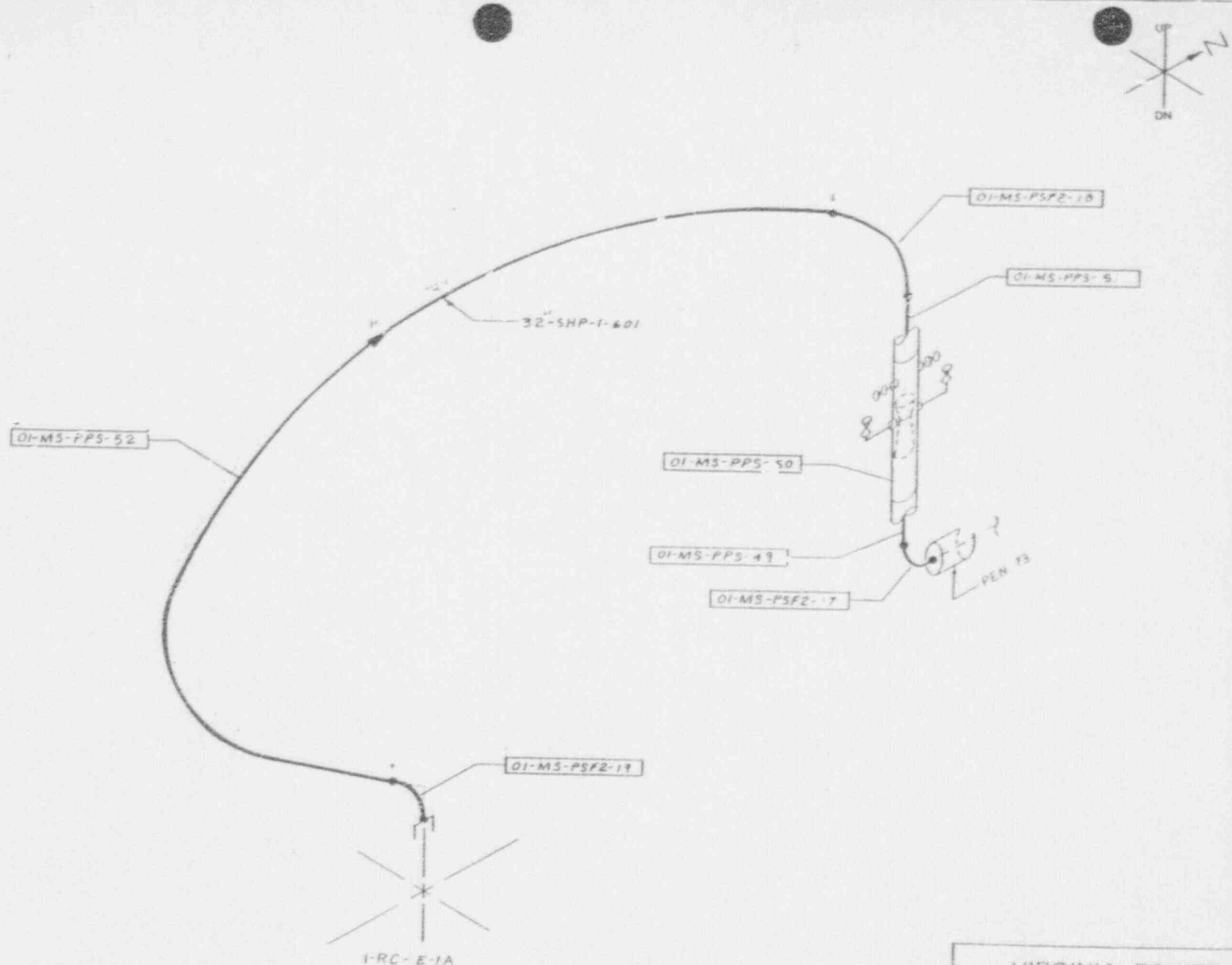


LEGEND
VC VERTICAL CONSTRAINT
LC LATERAL CONSTRAINT
ANC ANCHOR
SH SPRING HANGER
VS VERTICAL SUPPORT
HSS HYDRAULIC SHOCK SUPPRESSOR
VSS VERTICAL SHOCK SUPPRESSOR
LSS LATERAL SHOCK SUPPRESSOR
+Y
NORTH
+X
+Z
ISOMETRIC ORIENTATION
1-FPH-WGCB-4-2 (VC) EL 263'-3"
01-BD-PSF2-46
1-WGCB-PH-4.1 (LC)
01-BD-PSF2-45
01-BD-PPS-91 (INSPECT 100%)
1-FPH-WGCB-4-4 (VC) EL 263'-3"
01-BD-PSF2-47
1-WGCB-6-601C-Q3 (RE-ROUTED)
01-BD-PPS-96 (INSPECT 100%)
1-FPH-WGCB-4-3 (LC) EL 256'-0" (INSPECT 100%)
01-BD-PPSD-16
(INSPECT 100%)
01-BD-PPS-97
2½" VOW-60C 1-BD-4
2½" VOW-60C 1-BD-1
EL 264'-7"
01-BD-PSF2-110
01-BD-PSF2-48
3" X 2½" REDUCER TYP 2 PLACES
01-BD-PSFR-10
5 DIA BEND
01-BD-PSFR-11
STM. GEN 1-RC-E-1A
1-FPH-WGCB-5-3 (VC)
EL 264'-9"
2½" WGCB-4-601C-Q3
1-FPH-WGCB-5-5 (LC) EL 261'-10"
1-FPH-WGCB-4-1 (LC/LC) EL 252'-0"
1-WGCB-SH-MFSK-1971A (SPRING)
℄ EL 246'-0"
CL 601C
CL 601
1-BD-HSS-001 (VSS)
NOTES:
1. TOLERANCE ON ALL DIMENSIONS IS ± 1" UNLESS OTHERWISE SPECIFIED.
2½" WGCB-5-601C-Q3
01-BD-PSF2-50
3"X3"X2½" ANSI TEE
01-BD-PSFT-9
1-WGCB-R-450 (LC)
3" X 2½ REDUCER
01-BD-PSFR-5
1-FPH-WGCB-5-1 (VC/LC)
REFERENCE
UTI-WGCB-1001A
UTI-WGCB-1004A
EL 237'-3"
1-FPH-WGCB-5-2 (LC/LC) EL 249'-0"
01-BD-PSF2-49
1-WGCB-R-319 (VC/LC) EL 237'-3"
EL 246'-0"
3" WGCB-1-601-Q3
1-FPH-WGCB-5-4 (SPRING)
1-WGCB-A-318 (ANC) EL 237'-3"
I.O. 02072 0710 DC 90-13-1
STEAM GENERATOR BLOWDOWN PIPING LOOP A LAYOUT FOR 2½ INCH REPLACEMENT PIPING
ISSUED FOR CONSTRUCTION OF DC 90-13-1
STONE & WEBSTER ENGINEERING CORP.
DWG. NO. N9013-1-M-601 SHT 1 OF 1 REV 0


LEGEND
VC - VERTICAL CONSTRAINT
LC - LATERAL CONSTRAINT
ANC - ANCHOR
SH - SPRING HANGER
VS - VERTICAL SUPPORT
VSS - VERTICAL SHOCK SUPPRESSOR
LSS - LATERAL SHOCK SUPPRESSOR
(INSPECT 100%)
01-BD-PPS-120
2½"-VOW-60C
1-BD-10
01-BD-PSF2-62
EL 264'-9"
5 DIA BEND
01-BD-PPSD-22
3 X 2½" REDUCER
01-BD-PSFR-12
01-BD-PSF2-61
1-FPH-WGCB-7-4
(VC) EL 263'-0"
STM. GEN.
1-RC-E-1B
53.5°
27°
(INSPECT 100%)
01-BD-PPS-112
1-FPH-WGCB-8-1
(VC) EL 264'-9"
5 DIA BEND
EL 264'-9"
01-BD-PSF2-58
1-WGCB-PH-7.1
(LC)
1-VOS-60U
1-BD-11
01-BD-PSFR-13
(INSPECT 100%)
01-BD-PPSD-23
2½"-VOW-60C
1-BD-13 (MOVED PASSED SUPT)
1-FPH-WGCB-7-2
(VC) EL 263'-0"
5 DIA BEND
01-BD-PPSD-40
(SEE NOTE A)
EL 263'-0"
1-FPH-WGCB-8-5
(LC) EL 262'-1"
(RAISED 3")
1-FPH-WGCB-8-2
(LC/LC) EL 249'-0
01-BD-PSF2-59
2½"-WGCB-7-601C-Q3
1-FPH-WGCB-7-3
(LC) EL 256'-0"
1-FPH-WGCB-8-4
(SPRING) EL 246'-0"
NOTES:
1. TOLERANCE ON ALL DIMENSIONS IS ± 1" UNLESS OTHERWISE SPECIFIED.
REFERENCE: UTI-WGCB-1007A
NOTES: A) INSPECT COMPONENT 100% UP TO 6" DOWNSTREAM BEYOND LAST BEND.
1-FPH-WGCB-7-1
(LC/LC) EL 250'-0"
3"-WGCB-2-601-Q3
(CONT'D ON SHT 1)
1-FPH-WGCB-8-3
(VC/LC) EL 246'-0"
3"X 3"X 2½" ANSI TEE
EL 246'-0"
1-BD-PSFT-10
7'-7"
7'-0"
2½"-WGCB-8-601C-Q3
CL 601
CL 601C
3" X 2½" REDUCER
01-BD-PSFR-7
J.O. 02072.0710
DC 90-13-1
STEAM GENERATOR BLOWDOWN PIPING LOOP B LAYOUT FOR 2½ INCH REPLACEMENT PIPING
ISSUE 0 ISSUED FOR CONSTRUCTION OF IDC 90-13-1
STONE & WEBSTER ENGINEERING CORP.
DWG. NO. N9013-1-M-602 SHT 2 OF 2 REV 0

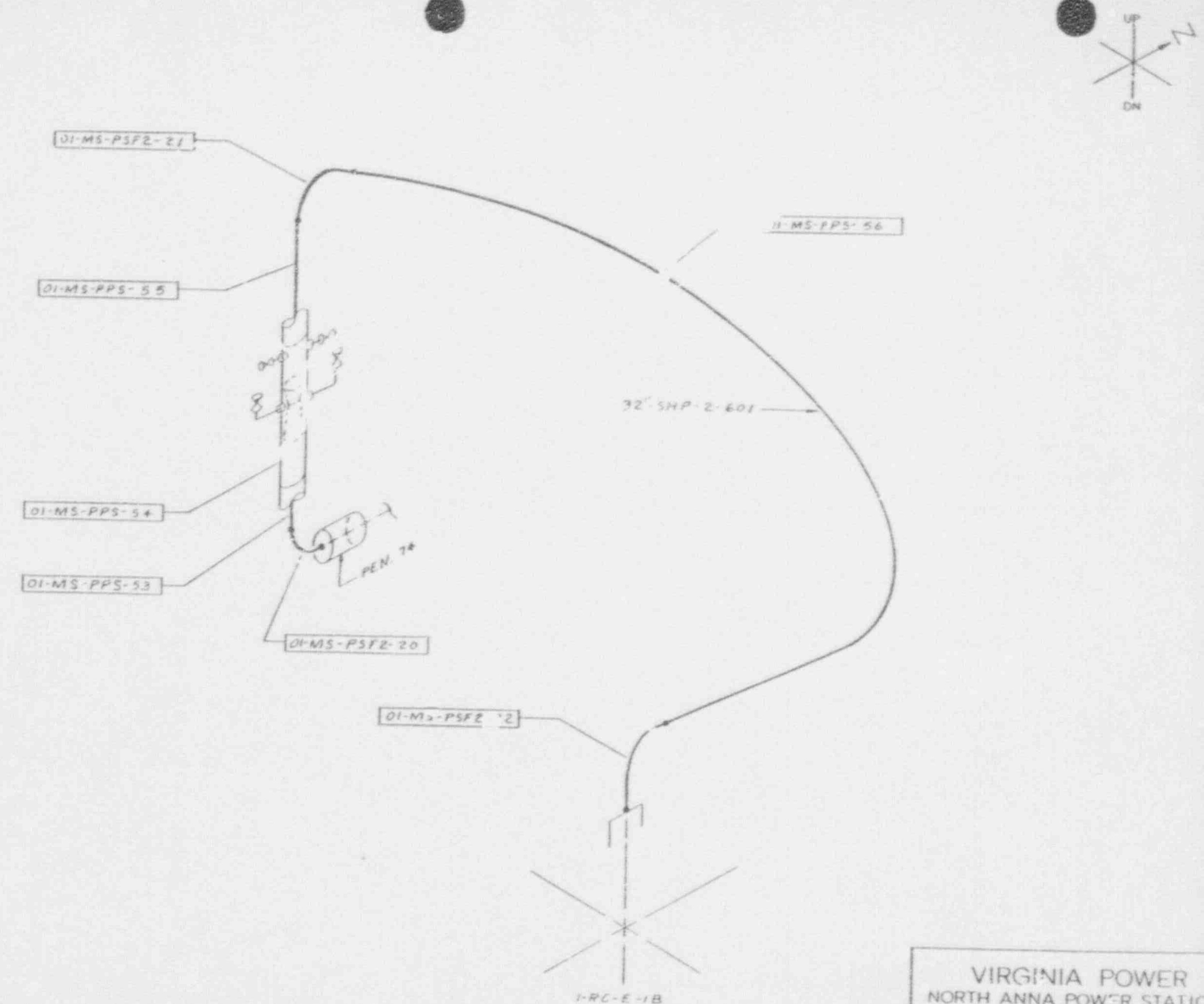


LEGEND
VC - VERTICAL CONSTRAINT
LC - LATERAL CONSTRAINT
ANC - ANCHOR
SH - SPRING HANGER
VS - VERTICAL SUPPORT
HSS - HYDRAULIC SHOCK SUPPRESSOR
VSS - VERTICAL SHOCK SUPPRESSOR
LSS - LATERAL SHOCK SUPPRESSOR
ISOMETRIC ORIENTATION
NOZZLE ORIENTATION PLAN
THIS 1" LINE IS RE-ROUTED FROM THE TOP ELBOW TO THE BRANCH CONN.
1-BD-PSFZ-4
1-BD-PPS-19B
1-BD-PPS-26
1-BD-PSFZ-2
1-FPH-WGCB-10-3 (VC/LC) EL 264'-9"
1-BD-PPS-19
2½" VOW-60C 1-BD-19
2½" WGCB-10-601C-A3
1-FPH-WGCB-11-1 (LC/LC) EL 249'-0"
(SEE NOTE B) 1-BD-PPSD-5
2½" VOW-60C 1-BD-22
3"x2½" REDUCER
1-BD-PSFR-15
1-FPH-WGCB-12-1 (VC/LC) EL 264'-9"
1" VOS-60U 1-BD-20
5 DIA BEND
(SEE NOTE A) 1-BD-PPSD-41
NOZZLE 4B
NOZZLE 6
NOZZLE 4A
TO ℄ REACTOR
(SEE NOTE A) 1-BD-PPSD-1
1-FPH-WGCB-10-2 (LC) EL 252'-6"
1-FPH-WGCB-10-1 (LC) EL 246'-0"
3" WGCB-3-601-Q3
1-FPH-WGCB-11-3 (VC/LC) EL 246'-0"
1-BD-PSFZ-11
3"x2½ REDUCER
1-BD-PSFR-D
3"x3"x2½ ANSI TEE
1-BD-PSFI-11
3"x2½ REDUCER
1-BD-PSFR-14
CL 601C
CL 601
NOTES:
1. TOLERANCE ON ALL DIMENSIONS IS ± 1" UNLESS OTHERWISE SPECIFIED.
REFERENCE
NOTES
A. INSPECT COMPONENT 100% UP TO 12" DOWNSTREAM BEYOND LAST BEND.
B. INSPECT COMPONENT 100%.
J.O. 02072.0710
DC 90-13-1
STEAM GENERATOR BLOWDOWN PIPING LOOP C LAYOUT FOR 2-1/2 INCH REPLACEMENT PIPING
STONE & WEBSTER ENGINEERING CORP.
DWG. N9013-1-M-603
NO. SHT 2 of 2 REV 0
ISSUED FOR CONSTRUCTION OF DC 90-13-1



VIRGINIA POWER
NORTH ANNA POWER STATION
SECONDARY SIDE PIPING INSPECTION

LINE DESIGNATION	Tnom	Tacc(1)	
32" SHP-1-601			UNIT No.: 1 LINE Nos: SHP-1
			REF DWG: 11715-FP-1A
			DRAWN: REM APPR:
			CHKD: DATE:
			DRAWING NUMBER: UTI-SHP-1001A REV: 0



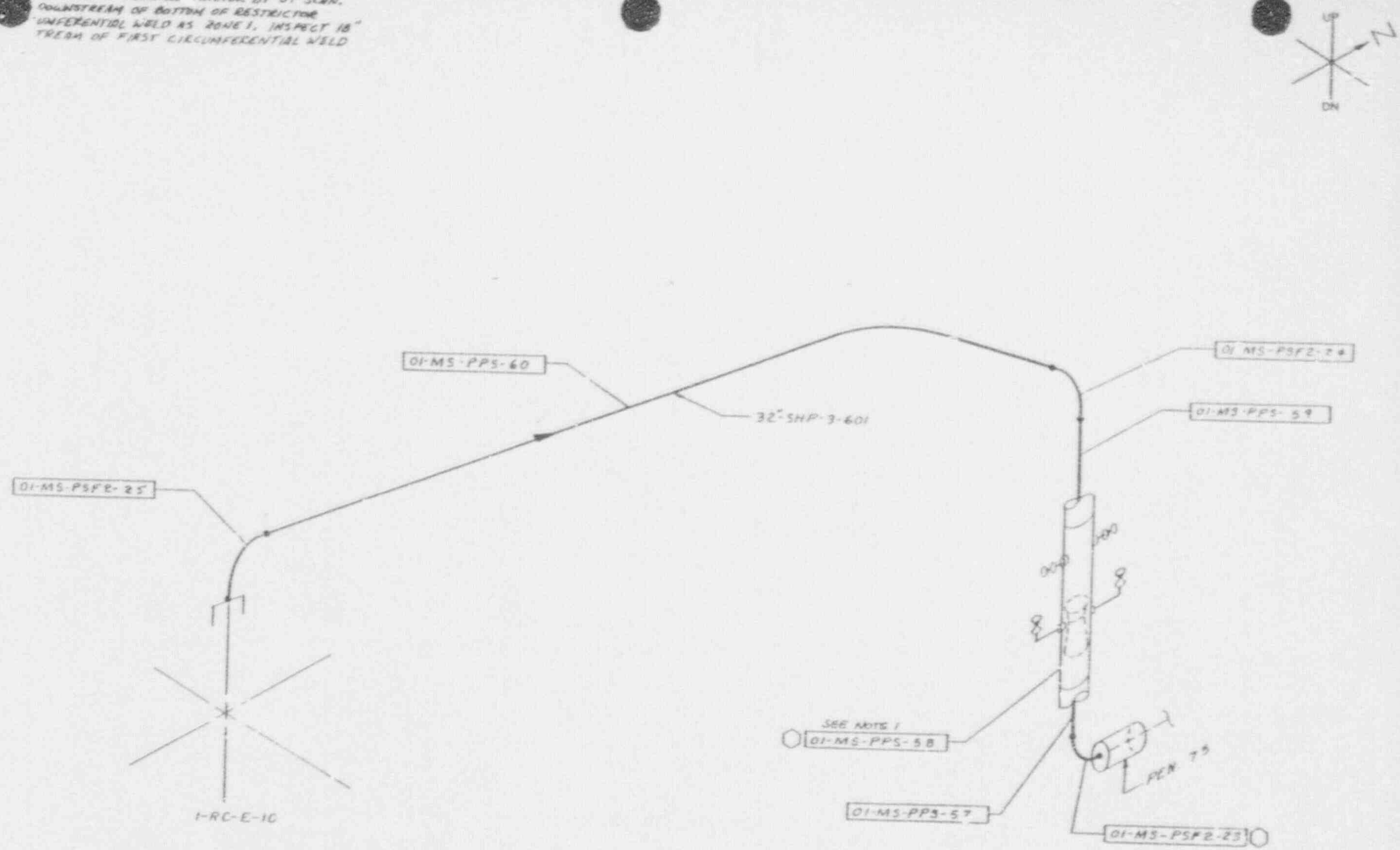
VIRGINIA POWER
NORTH ANNA POWER STATION
SECONDARY SIDE PIPING INSPECTION

LINE DESIGNATION	Tnom	Tacc(1)
32" SHP-2-601		

UNIT No.: 1	LINE Nos.: SHP-2
REF DWG.: 11715-FP-1A	
DRAWN: R E M	APPR:
CHKD:	DATE:
DRAWING NUMBER: UTI-SHP-1002A	REV: 0

NOTE:

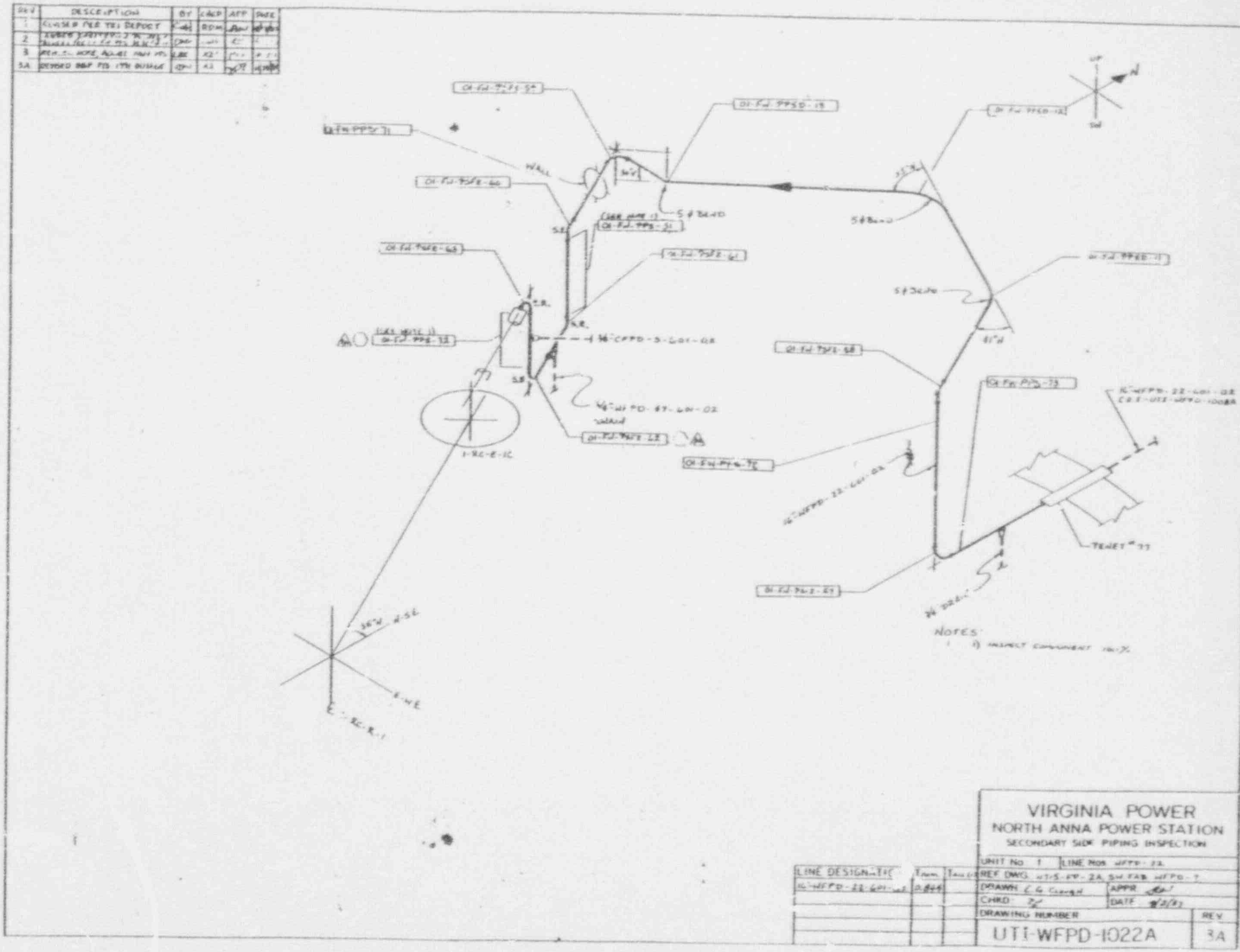
1. ADE TO L[illegible]E BOTTOM OF FLOW RE TRICTOR BY UT SCAN. INSPECT [illegible] DOWNSTREAM OF BOTTOM OF RESTRICTOR TO FIRST [illegible] UMFERENTIAL WELD AS ZONE 1, INSPECT 18" BALIN D[illegible] TREAM OF FIRST CIRCUMFERENTIAL WELD

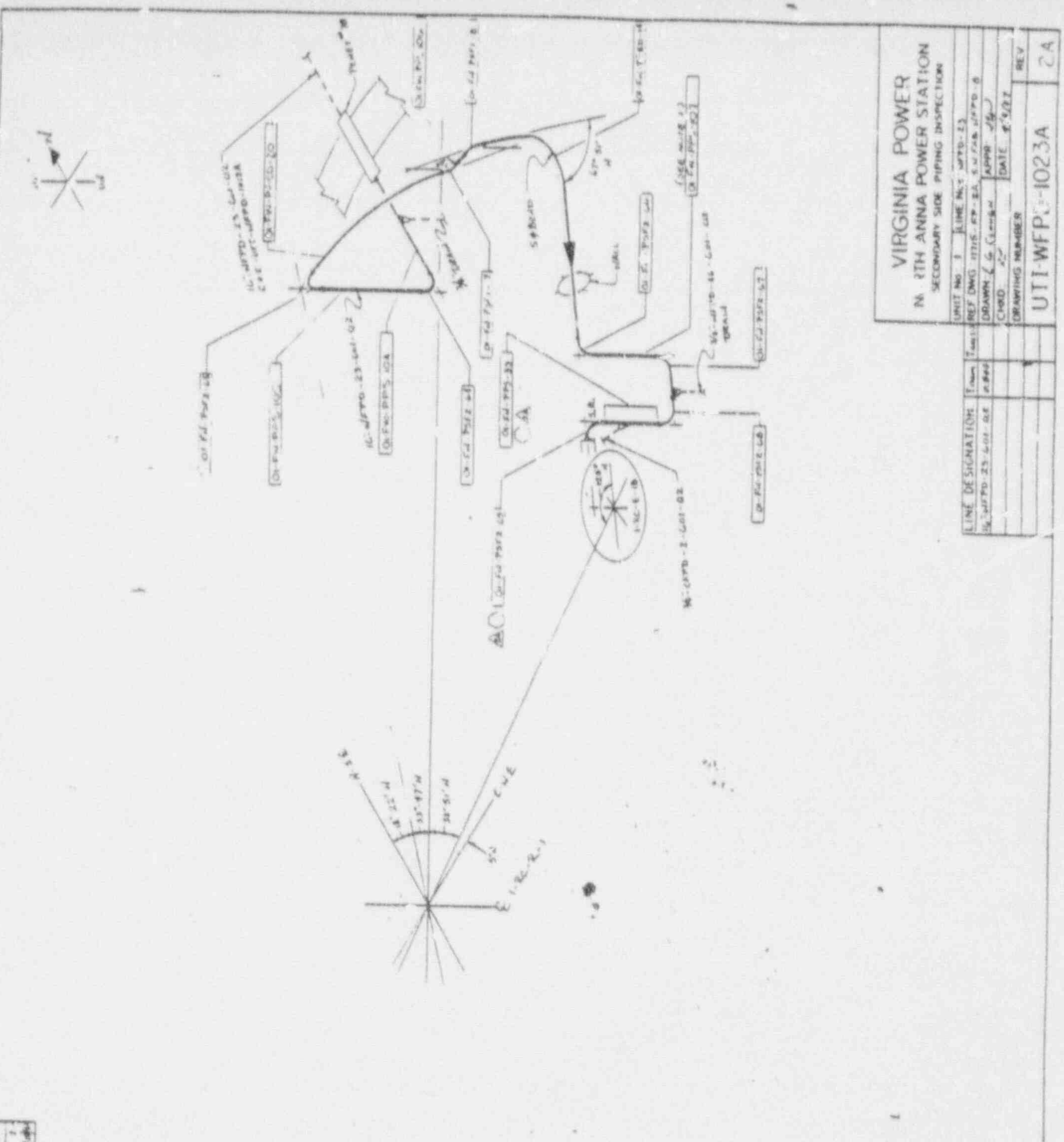


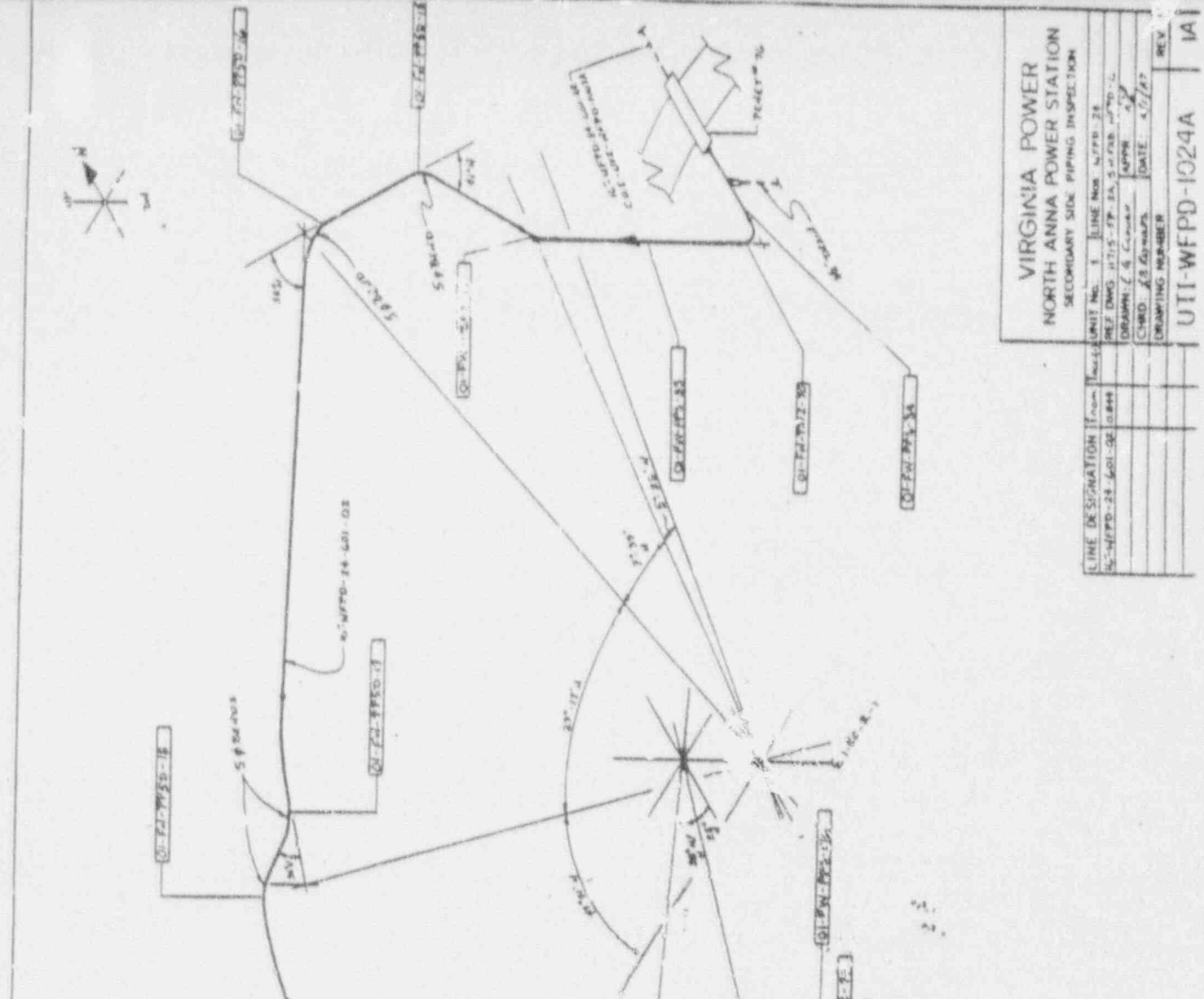
LINE DESIGNATION	Tnom	Tacc(1)
32"-SHP-3-601	1.000	

VIRGINIA POWER
NORTH ANNA POWER STATION
SECONDARY SIDE PIPING INSPECTION

UNIT No.: 1	LINE Nos.: SHP-3
REF. DWG.: 11715-FP-1A	
DRAWN: REM	APPR.: [illegible]
CHKD: RV	DATE: 4/22/87
DRAWING NUMBER: UTI-SHP-1003A	REV: 0


VIRGINIA POWER
NORTH ANNA POWER STATION
SECONDARY SIDE PIPING INSPECTION
UNIT No 1
DRAWING NUMBER
UTI-WFPD-1022A
REV
3A
NOTES:
WALL
LINE DESIGN-TIC
DESCRIPTION


VIRGINIA POWER
N. RTH ANNA POWER STATION
SECONDARY SIDE PIPING INSPECTION
NOTES:
1 INSPECT COMPONENT 100%


VIRGINIA POWER
NORTH ANNA POWER STATION
SECONDARY SIDE PIPING INSPECTION
UTI-WFPD-1024A
REV
1A

TESTING PROCEDURE

The Test Procedure has been prepared under ENAP-0025 and will be approved and maintained separate from the DCP.

OVERSIZE DOCUMENT PAGE PULLED

SEE APERTURE CARDS

NUMBER OF OVERSIZE PAGES FILMED ON APERTURE CARDS 53

APERTURE CARD/HARD COPY AVAILABLE FROM

RECORDS AND REPORTS MANAGEMENT BRANCH