VIRGINIA ELECTRIC AND POWER COMPANY
RICHMOND, VIRGINIA 23261

March 30, 1999

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

Serial No.: 99-134
NL&OS/GDM: R0'
Docket Nos.: 50-280, 281
50-338, 339
License Nos.: DPR-32, 37
NPF-4, 7

Gentlemen:

VIRGINIA ELECTRIC AND POWER COMPANY
SURRY AND NORTH ANNA POWER STATIONS UNITS 1 AND 2
SUPPLEMENTAL RESPONSE TO GENERIC LETTER (GL) 96-06
STRUCTURAL INTEGRITY EVALUATION OF THERMALLY INDUCED OVER PRESSURIZATION OF CONTAINMENT PENETRATION PIPING DURING DBA

In a letter dated February 25, 1998 (Serial No. 96-516C), Virginia Electric and Power Company (Virginia Power) proposed acceptance criteria to be used for evaluating design adequacy for thermally induced overpressurization of piping systems that penetrate the containments of both North Anna and Surry Power Stations during a postulated design basis accident (DBA). Specifically, we proposed to use the ASME Code Section III, Appendix F, "Rules for Evaluation of Service Loading with Level D Service Limits." The proposed acceptance criteria were provided to address NRC Generic Letter 96-06, "Assurance of Equipment Operability and Containment Integrity During Design Basis Accident Conditions." Detailed criteria for the use of the linear elastic analysis method of the code was proposed and was subsequently discussed in a telephone conversation with members of the NRC staff.

We have completed our structural integrity evaluations for the piping systems that are susceptible to such a postulated loading for both Surry and North Anna Power Stations to address the concerns raised in GL 96-06. Attachment 1 to this letter provides: (1) a summary of the method used for determining thermally induced pressure in isolated piping with confined fluid, (2) identification of the piping with its associated containment penetration numbers, (3) thermally induced pressure in the identified piping, and (4) a summary of stresses in the susceptible piping components and associated valves along with the Code allowable stress. The pressure determination in the pipe considered the differential expansion between the confined fluid and the pipe metal, and also took credit for a limited amount of circumferential strain in the pipe wall due to pressure. Detailed verification of structural integrity was not considered necessary when the faulted pressure was less than 1.2 times the design pressure. Detailed linear elastic

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stress analysis was performed for the components when the faulted pressure exceeded 1.2 times the design pressure.

The results of the analysis and determination of Code compliance were based upon the ASME Boiler and Pressure Vessel Code, Section III, Appendix F. The results demonstrate that the structural integrity of the containment penetration piping and associated valves will be maintained when subjected to thermally induced over-pressurization during postulated DBA conditions (i.e., Loss of Coolant Accident/Main Steam Line Break). Adequate margin exists between the applied stresses and the Appendix F allowable stresses. The deformation of the piping components will be limited to the amount of strain listed in the attachment, and a gross failure during this faulted event is not considered credible. Adequate technical basis exists to conclude that there are no safety significant issues that could affect containment integrity or equipment operability during DBA conditions. Thus, the results obtained provide adequate assurance of continued equipment operability and containment integrity during DBA conditions addressing the concern raised in GL 96-06.

Our review did not result in any physical modifications to our plant facilities. However, Surry and North Anna Power Stations were not originally licensed to use Appendix F of the ASME Section III Code. Consequently, we request your approval to use the 1989 version of the ASME Boiler and Pressure Vessel Code, Section III, Appendix F, as the applicable code for Surry and North Anna Power Stations for this particular faulted loading event as modified and detailed in Attachment 1. Drafts of the proposed changes to the Updated Final Safety Analysis Reports (UFSAR) for Surry and North Anna Power Stations are provided in Attachments 2 and 3, respectively, for your information. A revision to the respective UFSARs for each station will be implemented upon NRC closeout of GL 96-06 in accordance with 10 CFR 50.71(e).

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

Very truly yours,

D. A. Christian

D. A. Christian
Vice President - Nuclear Operations

Attachments

cc: U.S. Nuclear Regulatory Commission
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Mr. M. J. Morgan
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NRC Resident Inspector
Surry Power Station

Commitment Summary

The following represents the specific commitment made by the subject correspondence (Serial No. 99-134):

1. A revision to the respective UFSARs for each station will be implemented upon NRC closeout of GL 96-06 in accordance with 10 CFR 50.71(e).

Attachment 1

Structural Integrity Evaluation of Thermally Induced Over Pressurization of Containment Penetration Piping During a DBA

Surry and North Anna Power Stations Units 1 and 2

Virginia Electric and Power Company

1.0 Introduction

NRC Generic Letter 96-06: "Assurance of Equipment Operability and Containment Integrity During Design-Basis Accident Conditions" was issued on September 30, 1996 to notify all holders of operating licenses about potentially safety significant issues that could affect containment integrity and equipment operability during design basis accident (DBA) conditions.

As a part of the actions associated with the NRC Generic Letter 96-06 (Ref. 6), it was considered necessary to perform analyses to provide assurance of equipment operability and containment integrity during DBA conditions. Specifically, it was considered necessary to analyze and establish that thermally induced overpressurization of isolated water-filled piping sections in the containment boundary could not jeopardize the ability of the accident mitigating systems to perform their safety functions and could not lead to a breach of containment integrity through bypass leakage. The maximum internal pressure developed inside the isolated containment piping penetrations during a design basis accident (i.e., LOCA or MSLB) was calculated taking into account the differences in the expansion of the fluid and the pipe, the temperature increase immediately following the DBA and crediting a limited amount of radial/circumferential strain of the piping material during the pressurization process.

Piping susceptibility to thermal overpressurization following a DBA was not specifically evaluated prior to the issue of NRC GL 96-06. It should be noted that no specific design criteria exist in the North Anna or Surry design basis or original piping codes for evaluating isolated pipe segments under faulted conditions. Consequently, the loading conditions and the criteria for structural integrity evaluation of such a faulted event were not established or required during initial licensing. Also, the particular sections of pipe and the corresponding penetrations that may be susceptible to overpressurization were not specifically identified earlier. As a part of the response to GL 96-06, the susceptible piping sections were evaluated (Ref. 12). Many sections were not susceptible to overpressurization because of their configurations or the operating conditions during the DBA. Detailed thermal analysis of the remaining sections was performed (Ref. 7 & 8) to determine the extent of overpressurization. A detailed evaluation to establish structural integrity was not considered necessary for sections subjected to 1.2 times the design pressure or less during the event. Structural evaluation was performed for remaining sections using linear elastic analysis method stipulated in ASME Boiler and Pressure Vessel Code, Section III, Appendix F.

2.0 Summary of the Method for Determination of Thermally Induced Pressure in Isolation Piping with Confined Fluid

The heat transferred to isolated sections of containment piping during a Design Basis Accident (DBA) was conservatively evaluated using the post-accident containment bulk atmospheric temperature profile for minimum engineered safeguards response. It should be noted that the conservatism in the analysis that produces the bulk atmospheric temperature profile was appropriate for sizing the safeguards systems, the design of the containment structure and pressure boundary components. Lacking more realistic containment response analysis, these highly conservative temperature profiles were used. The location of the containment piping penetrations of concern, relatively low in the structure and outside the crane wall, would also result in their being exposed to lower peak temperatures. However, this analysis assumed no temperature reduction based on piping location.

The maximum internal pressure developed inside the isolated containment piping penetrations during a DBA was calculated as follows:

- Determine piping parameters (length of piping inside and outside, system operating conditions at time of DBA, etc.)
- Determine the time dependent, heat transfer to the piping/fluid inside containment, associated peak temperature, the heat content and mass of the water in the pipe
- Determine the temperature distribution along the piping which passes through the containment wall by numerical relaxation method for two dimensional heat transfer in solids, and determine the heat content and mass of the water in the pipe
- Model the outside containment penetration piping as a fin and calculate the temperature distribution along the piping, determine heat content and mass of the water in the pipe
- Determine the total heat content and total mass of water in the entire length of isolated piping
- Calculate the bulk temperature of the isolated piping
- The expansion of the piping circumference and length resulting from the fluid temperature increase is applied to the piping circumference and length
- A final pressure is assumed and the stress resulting from the pressure is calculated
- The strain resulting from the calculated stress is then calculated and the strain is applied to the piping circumference to determine the increase in volume resulting from the increase in temperature (pressure)
- Calculate the final pressure rise in the isolated piping based on the change in volume of an incompressible fluid
- The final pressure is checked against the assumed pressure
- The iteration is repeated until the assumed final pressure and the calculated final pressure are equal; this results in the final equilibrium state for the piping.

3.0 Identification of Piping with Its Identifying Containment Penetration Numbers

The following penetrations have been evaluated using the methodology in ASME Section III, Appendix F, to determine whether any modifications are required. The thermally induced pressure on this piping was determined to be more than 1.2 times the design pressure.

Surry Power Station, Units 1 & 2 – Penetrations 20, 28 and 46.

North Anna Power Station, Unit 1 – Penetrations 5, 12, 13, 14, 20, 25 and 46.

North Anna Power Station, Unit 2 – Penetrations 5, 12, 13, 14, 20, 25, 26, 27, 46 and 106.

4.0 Thermally Induced Pressure in the Identified Piping

Table 4.1- Pressure Loading Due to Temperature Increase in Isolated Pipe

Penetration No. (1)	Pressure Psi	Predicted Circumferential Percent Strain (2)
NAPS #5 U1	1,247	0.57
NAPS #5 U2	1,242	0.68
NAPS #12 U1	2,176	0.09
NAPS #12 U2	2,555	0.10
NAPS #13 U1	2,765	0.11
NAPS #13 U2	2,499	0.10
NAPS #14 U1	2,574	0.10
NAPS #14 U2	2,499	0.10
NAPS #20 U1	5,503	1.20
NAPS #20 U2	5,337	1.20
NAPS #25 U1	2,069	0.33
NAPS #25 U2	2,347	0.57
NAPS #26 U2	2,230	0.22
NAPS #27 U2	1,664	0.08
NAPS #46 U1	7,582	0.99
NAPS #46 U2	7,936	0.90
NAPS #106 U2	3,171	0.44
SPS #20 U1	5,022	1.67
SPS #20 U2	5,022	1.67
SPS #28 U1	2,960	0.65
SPS #28 U2	2,960	0.65
SPS #46 U1	7,125	1.46
SPS #46 U2	7,125	1.46

Notes:

(1) NAPS - North Anna Power Station
SPS - Surry Power Station
U1 - Unit 1
U2 - Unit 2

(2) Pressure determination utilizes credit of a limited amount of strain in the pipe

5.0 Summary of Stresses in the Susceptible Piping Components and Associated Valves Along with the Code Allowable Stress

Table 5.1 - Pipe Stress Summary

Plant-Unit	Penetration No.	Faulted Pressure (psi)	Applied Membrane Stress (psi)	Allowable Membrane Stress (psi)	Applied Membrane + Bending Stress (psi)	Allowable Membrane + Bending Stress (psi)
SPS-1	20	5022	26360	46200	21620	45000
	28	2960	24900	46200	27170	45000
	46	7125	25340	46200	21520	45000
SPS-2	20	5022	26360	46200	21620	45000
	28	2960	24900	46200	27170	45000
	46	7125	25340	46200	21670	45000
NAPS-1	5	1247	33700	42000	32870	60000
	12, 13, 14	2765	36280	42000	24630	60000
	20	5503	28890	46200	23400	45000
	25	2069	30840	42000	28110	60000
	46	7582	26970	48000	23460	46600
NAPS-2	5	1242	33570	42000	17430	60000
	12, 13, 14	2555	33520	42000	18980	60000
	20	5337	28020	46200	22790	45000
	25, 26, 27	2347	34980	42000	26160	60000
	46	7936	28230	48000	24640	46600
	106	3171	26140	46200	20990	45000

Notes:

(1) Linear Elastic Analysis Method of analysis is used.

(2) Pressure and dead weight loadings are used. Seismic loading is not considered concurrent with the event.

(3) Allowable membrane stress = $2.4S_m$ or $0.7S_u$ whichever is lower.

(4) Allowable membrane plus bending stress = $3.0S_m$ or $2S_y$ whichever is lower.

(5) Only the piping with pressure greater than 1.2 times the design pressure is listed.

(6) Pressure increase is due to temperature effect on confined fluid inside piping on both sides of the containment penetration.

(7) Allowable stresses are taken from ASME B & PV Code Section III, 1989 (Ref. 4).

(8) Adequate margins exist between applied stress and allowable stress.

Table 5.2 – Valve Stress Summary (applicable to weld end valves)

Penetration No.	Valve No.	Crotch Stress Intensity, psi	Allowable Crotch Stress Intensity, psi
NAPS #5 U1	1-CC-764	18,275	73,500
NAPS #12 U1	1-CC-568	20,262	73,500
NAPS #12 U1	1-CC-715	20,262	73,500
NAPS #12 U1	1-CC-718	20,262	73,500
NAPS #13 U1	1-CC-581	21,522	73,500
NAPS #13 U1	1-CC-721	21,522	73,500
NAPS #14 U1	1-CC-555	21,114	73,500
NAPS #14 U1	1-CC-710	21,114	73,500
NAPS #20 U1	1-SI-58	16,726	72,000
NAPS #20 U1	1-SI-59	19,875	72,000
NAPS #20 U1	1-SI-111	12,942	72,000
NAPS #20 U1	1-SI-110	16,725	72,000
NAPS #20 U1	1-SI-245	14,451	72,000
NAPS #20 U1	1-SI-HCV-1851A	33,335	72,000
NAPS #20 U1	1-SI-HCV-1851B	33,335	72,000
NAPS #20 U1	1-SI-HCV-1851C	33,335	72,000
NAPS #25 U1	1-CC-108	20,033	73,500
NAPS #25 U1	1-CC-754	20,033	73,500
NAPS #46 U1	1-CH-FCV-1160	38,454	72,000
NAPS #46 U1	1-CH-330	30,606	72,000
NAPS #46 U1	1-CH-331	17,958	72,000
NAPS #46 U1	1-CH-332	17,958	72,000
NAPS #46 U1	1-CH-476	22,195	72,000
NAPS #46 U1	1-CH-488	22,195	72,000
NAPS #46 U1	1-CH-489	22,195	72,000
NAPS #46 U1	1-RC-HCV-1556A	38,454	72,000
NAPS #46 U1	1-RC-HCV-1556B	38,454	72,000
NAPS #46 U1	1-RC-HCV-1556C	38,454	72,000
NAPS #5 U2	2-CC-329	18,265	73,500
NAPS #5 U2	2-CC-705	18,265	73,500
NAPS #12 U2	2-CC-298	21,073	73,500
NAPS #12 U2	2-CC-750	21,073	73,500
NAPS #12 U2	2-CC-751	21,073	73,500
NAPS #13 U2	2-CC-311	20,953	73,500
NAPS #13 U2	2-CC-756	20,953	73,500
NAPS #14 U2	2-CC-284	20,953	73,500
NAPS #14 U2	2-CC-285	20,953	73,500
NAPS #20 U2	2-SI-47	16,427	72,000
NAPS #20 U2	2-SI-48	12,674	72,000
NAPS #20 U2	2-SI-136	20,704	72,000

NAPS #20 U2	2-SI-137	14,284	72,000
NAPS #20 U2	2-SI-243	12,674	72,000
NAPS #20 U2	2-SI-HCV-2851A	32,883	72,000
NAPS #20 U2	2-SI-HCV-2851B	32,883	72,000
NAPS #20 U2	2-SI-HCV-2851C	32,883	72,000
NAPS #25U2	2-CC-103	20,628	73,500
NAPS #25 U2	2-CC-712	20,628	73,500
NAPS #26 U2	2-CC-177	20,378	73,500
NAPS #26 U2	2-CC-349	20,378	73,500
NAPS #26 U2	2-CC-722	20,378	73,500
NAPS #27 U2	2-CC-140	19,167	73,500
NAPS #27 U2	2-CC-717	19,167	73,500
NAPS #27 U2	2-CC-718	19,167	73,500
NAPS #46 U2	2-CH-FCV-2160	39,578	72,000
NAPS #46 U2	2-CH-259	17,012	72,000
NAPS #46 U2	2-CH-332	31,522	72,000
NAPS #46 U2	2-CH-333	18,507	72,000
NAPS #46 U2	2-CH-334	18,507	72,000
NAPS #46 U2	2-CH-380	18,507	72,000
NAPS #46 U2	2-CH-406	19,078	72,000
NAPS #46 U2	2-CH-385	18,507	72,000
NAPS #46 U2	2-RC-HCV-2556A	39,576	72,000
NAPS #46 U2	2-RC-HCV-2556B	39,576	72,000
NAPS #46 U2	2-RC-HCV-2556C	39,576	72,000
NAPS #106 U2	2-SI-TV-2842	19,022	72,000
NAPS #106 U2	2-SI-248	6,138	72,000
NAPS #106 U2	2-SI-231	10,753	72,000
NAPS #106 U2	2-SI-TV-2859	19,023	72,000
NAPS #106 U2	2-SI-49	24,600	72,000
SPS #20 U1	1-SI-HCV-1851A	29,556	72,000
SPS #20 U1	1-SI-HCV-1851B	29,556	72,000
SPS #20 U1	1-SI-HCV-1851C	29,556	72,000
SPS #20 U1	1-SI-32	15,860	72,000
SPS #20 U1	1-SI-181	13,796	72,000
SPS #28 U1	1-CH-TV-1204A	17,099	72,000
SPS #28 U1	1-CH-TV-1204B	32,628	72,000
SPS #28 U1	1-CH-419	10,598	72,000
SPS #28 U1	1-CH-440	10,598	72,000
SPS #28 U1	1-CH-457	31,704	72,000
SPS #46 U1	1-CH - FCV-1160	30,260	72,000
SPS #46 U1	1-RC-HCV-1556A	30,260	72,000
SPS #46 U1	1-RC-HCV-1556B	30,260	72,000
SPS #46 U1	1-RC-HCV-1556C	30,260	72,000
SPS #46 U1	1-CH-444	18,486	72,000

SPS #46 U1	1-CH-316	17,250	72,000
SPS #20 U2	2-SI-HCV-2851A	29,566	72,000
SPS #20 U2	2-SI-HCV-2851B	29,566	72,000
SPS #20 U2	2-SI-HCV-2851C	29,566	72,000
SPS #20 U2	2-SI-32	15,860	72,000
SPS #20 U2	2-SI-181	13,796	72,000
SPS #28 U2	2-CH-TV-2204A	17,099	72,000
SPS #28 U2	2-CH-TV-2204B	35,529	72,000
SPS #28 U2	2-CH-414	10,599	72,000
SPS #28 U2	2-CH-441	31,551	72,000
SPS #28 U2	2-CH-453	31,704	72,000
SPS #46 U2	2-CH-FCV-2160	30,260	72,000
SPS #46 U2	2-RC-HCV-2556A	30,260	72,000
SPS #46 U2	2-RC-HCV-2556B	30,260	72,000
SPS #46 U2	2-RC-HCV-2556C	30,260	72,000
SPS #46 U2	2-CH-444	40,386	72,000
SPS #46 U2	2-CH-316	17,250	72,000

Note:

(1) The valve body structural integrity is verified by comparison of valve section and material properties with that of the connected pipe.

(2) Allowable stresses are from Reference 4.

(3) Adequate margin exists between applied stress and allowable stress.

Table 5.3 – Valve Stress Summary (applicable to flange end valves)

Penetration No.	Valve No.	Stress on Flange Bolts (psi)	Allowable (1) (psi)
NAPS #5 U1	1-CC-MOV-100A	25,535	87,500
NAPS #5 U1	1-CC-TV-103A	25,535	87,500
NAPS #12 U1	1-CC-TV-100B	26,000	87,500
NAPS #12 U1	1-CC-TV-105B	26,000	87,500
NAPS #13 U1	1-CC-TV-100C	32,070	87,500
NAPS #13 U1	1-CC-TV-105C	32,070	87,500
NAPS #14 U1	1-CC-TV-100A	30,102	87,500
NAPS #14 U1	1-CC-TV-105A	30,102	87,500
NAPS #25 U1	1-CC-TV-102E	44,646	87,500
NAPS #25 U1	1-CC-TV-102F	44,646	87,500
NAPS #5 U2	2-CC-MOV-200A	21,758	87,500
NAPS #5 U2	2-CC-TV-203A	21,758	87,500
NAPS #12 U2	2-CC-TV-200B	28,001	87,500
NAPS #12 U2	2-CC-TV-205B	28,001	87,500
NAPS #13 U2	2-CC-TV-200C	27,423	87,500
NAPS #13 U2	2-CC-TV-205C	27,423	87,500
NAPS #14 U2	2-CC-TV-200A	27,423	87,500
NAPS #14 U2	2-CC-TV-205A	27,423	87,500
NAPS #25 U2	2-CC-TV-202E	47,142	87,500
NAPS #25 U2	2-CC-TV-202F	47,142	87,500
NAPS #26 U2	2-CC-TV-202A	45,110	87,500
NAPS #26 U2	2-CC-TV-202B	45,110	87,500
NAPS #27 U2	2-CC-TV-202C	35,281	87,500
NAPS #27 U2	2-CC-TV-202D	35,281	87,500

Note:

(1) $0.7S_u$ or S_y (A193 Gr. B7 Bolts) whichever is lower.

6.0 CONCLUSIONS:

[1]. The pressure determination of the pipe takes credit for a limited amount of strain in the pipe at equilibrium state.

[2] A detailed evaluation of piping for verification of structural integrity was not considered necessary when the faulted pressure was less than 1.2 times the design pressure.

[3] The results of the analysis and determination of Code compliance were based upon the ASME Boiler and Pressure Vessel Code, Section III, Appendix F. These results demonstrate that the structural integrity of the containment penetration piping and associated valves will be maintained when subjected to thermally induced overpressurization during postulated DBA conditions (i.e., LOCA/MSLB).

[4] An adequate technical basis exists to conclude that there are no safety significant issues that could affect containment integrity and equipment operability during DBA conditions. Thus, the evaluation results provide assurance of continued equipment operability and containment integrity during DBA conditions and addresses the concern raised in NRC Generic Letter 96-06.

7.0 REFERENCES

[1]. ANSI B31.7, 1969 w/addenda through 1970 "Nuclear Power Piping Code".

[2] ANSI B31.1, 1967, "Standard Code for Pressure Piping."

[3] ANSI B31.1, 1955, "Code for Pressure piping up to and including Code Case N7."

[4] ASME B & PV Code, Section III, Appendix I, 1989.

[5] ASME B & PV Code, Section III, Appendix F, 1989.

[6] The NRC Generic Letter 96-06, "Assurance of Equipment Operability and Containment Integrity During Design Basis Accident Conditions, September 30, 1996."

[7] Virginia Power Calculation ME-0526, "Containment Piping Penetration Overpressurization During a DBA for Containment Penetrations 5, 12, 13, 14, 20, 25, 26, 27, 28 and 46 for North Anna Power Station, Units 1 & 2 and Penetration 106 for North Power Station, Unit 2."

[8] Virginia Power Calculation ME-0527, "Containment Piping Penetration Overpressurization During a DBA for Containment Penetration 20, 28, 46 and 110 for Surry Power Station Units 1 & 2."

[9] Letter from Virginia Power to NRC, Serial No. 96-516C, dated February 25, 1998, "Supplemental Response to Generic Letter (GL) 96-06 Acceptance Criteria for Design Adequacy Evaluation."

[10] Virginia Power Calculation ME-0569, "ASME B & PV Code, Section III, Appendix F, Valve Analysis for NRC Generic Letter 96-06."

[11] Letter from Virginia Power to NRC, Serial No. 96-516A, dated January 28, 1997, "NRC Generic Letter 96-06, Assurance of Equipment Operability and Containment Integrity During Design-Basis Accident."

[12] Virginia Power Technical Report No. CE-0102, Rev. 0, "Evaluation of Structural Integrity of Containment Penetration Piping and Isolation Valves During Postulated LOCA/MSLB Scenarios Using ASME Code Section III, Appendix F Criteria."

Attachment 2

Draft of Proposed UFSAR Revision to Reflect the Resolution of GL 96-06

Surry Power Station Units 1 and 2

Virginia Electric and Power Company

Attachment 2

Draft of Proposed UFSAR Revision to Reflect the Resolution of GL 96-06 Surry Power Station Units 1 and 2

- Insert 1

Insert in SPS UFSAR Revision 30-09/1/98, Section 15.5, after the 4th paragraph in page 15.5-18:

As a part of the issues identified in NRC GL 96-06, isolated containment penetration piping with confined fluid was reviewed for susceptibility to thermal overpressurization following a DBA. The linear elastic analysis criteria stipulated in the 1989 version of the ASME Boiler and Pressure Vessel Code Section III, Appendix F, were used for structural integrity evaluation. The internal pressure in piping penetrations during a design basis accident (LOCA or MSLB) was calculated by taking into account the differences in the expansion of the fluid and the pipe, the temperature increase immediately following the DBA and credit for a limited amount of circumferential strain in the pipe. The analysis established that thermally induced overpressurization of isolated water-filled piping sections in the containment boundary could not jeopardize the ability of the accident mitigating systems to perform their safety functions and could not lead to a breach of containment integrity (Reference XX).

- Insert 2

Add in SPS UFSAR Revision 30-09/1/98, Section 15.5 References, the following new reference:

XX. Letter Dated March XX, 1999, Serial No. 99-134, from Virginia Power to the NRC, Supplemental Response to Generic Letter 96-06.

The circumferential groove in the attachment plate, between the sleeve and penetration with its outside threaded connection, serves as a test chamber for the testing of the welds joining the attachment plate and penetration.

All penetrations are anchored in the reinforced concrete containment wall. The anchor strength is equal to the full yield strength of the pipe with regard to torsion, bending, and shear, and to the maximum possible pipe jet reaction. All stresses induced in the liner by these combinations of loadings are only those reflected by the resulting distortions in the reinforced concrete containment wall, and are minor in intensity. So, loads will not be imposed on the liner, thereby preserving its integrity.

All highly stressed insert plates at penetrations and equipment supports that are welded into the liner to transfer loads into the concrete have been ultrasonically tested to check for possible laminations. Tests were conducted on all plates where analysis showed a higher than average stress field, although all such plates are stressed well below the allowable limits for the materials. These tests show that no faults exist in the insert plates.

The pipes anchored to the containment penetrations between containment isolation valves constitute an extension of the containment, and are designed in accordance with the *USA Standard Code for Pressure Piping - Power Piping,* USAS B31.1.0-1967, with respect to materials and allowable stress. Analyses of stresses due to thermal expansion and shock loadings from earthquake, pipe jet reaction, and other causes were made using established digital computer calculation techniques.

In order to determine the loading combinations that act on a penetration, the pipe line passing through the penetration sleeve was assumed to have failed transversely at several locations along its run. The location at which the reaction of the ensuing jet of fluid flowing from the broken end first causes the pipe to completely yield, in either bending or torsion, was taken as the design case from which all resultant combinations of penetration loading were determined for that particular pipe line. The maximum stress allowed on any individual element of the penetration is 90% of the minimum yield point.

The intent of this criterion is to keep the material assembly components within the elastic range of the material. Under operating conditions of pressure, temperature, and external loads, the stresses in the assembly will be within the limits established in Section III of the ASME Pressure Vessel Code.

Insert 1

All liner seams were strength-welded. Small steel channels welded continuously along the edges of their flanges to the liner plate cover the plate weld seams, in a manner similar to those installed at the Connecticut Yankee Station. These channels are zoned into test areas by dams welded to the ends of the sections of the channels. Fittings are provided in the channels for periodic testing of the weld seams for leaktightness under pressure. Typical liner details are shown in Figure 15.5-12. Testing of the liner is described in Section 5.5.

periodic testing for structural purposes could be duplicated if at any time further tests were required. The minimum test level required to verify continued structural integrity would be no less than the 115%, or 52-psig initial test pressure.

Periodic inspection of the steel liner is accomplished by a type A leak rate test in accordance with 10 CFR 50 Appendix J. All welded joints and all penetrations of the liner are designed for periodic halogen gas testing.

In summary, no basis exists for attempting to develop structural performance information from leak rate tests conducted at moderate pressures.

15.5 REFERENCES

1. G. N. Bycroft, *Forced Vibrations of a Rigid Circular Plate on a Semi-Infinite Elastic Space and on an Elastic Stratum, Philosophical Transactions*, Royal Society, London, Series A, Vol. 248, pp. 327-368.
2. Karl Terzaghi, *Evaluation of Coefficients of Subgrade Reaction, Geotechnique,* Vol. 5, pp. 297-326, 1955.
3. B. O. Hardin and W. L. Black, *Vibration Modulus of Normally Consolidated Clay, Symposium on Wave Propagation and Dynamic Properties of Soils*, University of New Mexico, 1967.
4. Stone & Webster Engineering Corporation, *Nuclear Containment Structure Access Opening*.
5. B. Budiansky and P. Radkowski, *Numerical Analysis of Unsymmetrical Bending of Shells of Revolution, AIAA Journal*, August 1963.
6. S. Gere and S. Timoshenko, *Theory of Elastic Stability*, second edition.
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8. Stone & Webster Engineering Corporation, *Report on Pressure Testing of Reactor Containment for Connecticut Yankee Atomic Power Plant, Connecticut Yankee Atomic Power Company, Haddam, Connecticut*, 1967.
9. D. A. Davenport, *Penetration of Reactor Containment Shells, Nuclear Safety*, Vol. 2, No. 2, December 1960.

Insert 2

Attachment 3

Draft of Proposed UFSAR Revision to Reflect the Resolution of GL 96-06

North Anna Power Station Units 1 and 2

Virginia Electric and Power Company

Attachment 3

Draft of Proposed UFSAR Revision to Reflect the Resolution of GL 96-06 North Anna Power Station Units 1 and 2

- Insert 1

Insert in NAPS UFSAR Revision 34-09/1/98, Section 6.2.4.1, before the last paragraph in page 6.2-92:

As a part of the issues identified in NRC GL 96-06, isolated containment penetration piping with confined fluid was reviewed for susceptibility to thermal overpressurization following a DBA. The linear elastic analysis criteria stipulated in the 1989 version of the ASME Boiler and Pressure Vessel Code Section III, Appendix F, were used for structural integrity evaluation. The internal pressure in piping penetrations during a design basis accident (LOCA or MSLB) was calculated by taking into account the differences in the expansion of the fluid and the pipe, the temperature increase immediately following the DBA and credit for a limited amount of circumferential strain in the pipe. The analysis established that thermally induced overpressurization of isolated water-filled piping sections in the containment boundary could not jeopardize the ability of the accident mitigating systems to perform their safety functions and could not lead to a breach of containment integrity (Reference XX).

- Insert 2

Add in NAPS UFSAR Revision 34-09/1/98, Section 6.2 References, the following new reference:

XX. Letter Dated March XX, 1999, Serial No. 99-134, from Virginia Power to the NRC, Supplemental Response to Generic Letter 96-06.

6.2.4 Containment Isolation System

6.2.4.1 Design Bases

The containment isolation system has the following design bases:

1. For pipe penetrations through the containment, it provides, during accident conditions, at least two barriers between the atmosphere outside the containment structure and
   a. The atmosphere inside the containment structure, or
   b. The fluid inside the reactor coolant pressure boundary.
2. The design pressure of all piping and connecting component forming the isolation boundary is greater than the 45-psig design pressure of the containment. Piping forming the isolation boundary is designed to Class I or II of the American Standard Code for Pressure Piping - ANSI B31.7-1969 Nuclear Power Piping.
3. Failure of a single valve or barrier does not prevent isolation.
4. Operation of the containment isolation system is automatic.
5. All isolation valves and equipment are protected from missiles and water jets originating from the reactor coolant system (RCS).
6. All remotely actuated valves and automatically operated isolation valves have their positions indicated in, and can be operated from, the main control room.

All isolation valves located outside the containment in accordance with General Design Criteria 55, 56, and 57 are located as close to the penetration as possible without limiting the service accessibility of the valves or interfering with other valves, piping, or structural members. Approximately 70% of all outside isolation valves are located within 10 feet of the penetration. The six valves not within about 20 feet of their penetration are on 3/8-inch lines and are located 50 and 60 feet from their penetration. These six valves are located at this distance to maintain separation of components, as in the leakage monitoring system, or due to the physical size of the isolation valve, such as in the sampling system.

The pressure retaining integrity of the containment pipe penetrations will be maintained under an applicable pressure, temperature, and mechanical load combination, including SSE effects. The intent of Regulatory Guide 1.29 for these penetrations is met by the load combinations and elastic stress limits specified in Table 3.8-8. The plastic pipe loads M_P and T_P, which are far greater than the actual calculated pipe seismic loads, plus pipe design pressure and temperature effects, are each sufficient to fully yield the loaded pipe across its entire cross section at the penetration. The resulting penetration assembly stresses for these loads are limited to elastic ~~stress limits~~ such as 3S.

Insert 1

~~All~~ containment pipe penetrations are designed, built, inspected, and tested to the requirements of B31.7-1969, Class I or II. In 1971, these requirements were incorporated into

48. USNRC, *Water Sources for Long-Term Recirculation Cooling Following a Loss-of-Coolant-Accident,* USNRC *Regulatory Guideline 1.82, November 1985.*

49. A.W. Serkiz, *Containment Emergency Sump Performance,* USNRC, NUREG-0897, October 1985.

50. *Westinghouse LOCA Mass and Energy Release Model for Containment Design - March 1979 Version*, WCAP-10325-A, April 1979. (Proprietary)

51. *Westinghouse ECCS Evaluation Model - 1981 Version*, WCAP-9220-P-A, Rev. 1 (Proprietary), WCAP-9211-A, Rev. 1, February 1982.

52. *Mixing of Emergency Core Cooling Water with Steam: 1/3 Scale Test and Summary,* (WCAP-8423), EPRI 294-2, Final Report, June 1975.

53. *Topical Report Westinghouse Mass and Energy Release Data for Containment Design*, WCAP-8264-P-A, Rev. 1, August 1975. (Proprietary)

54. *American National Standard for Decay Heat Power in Light Water Reactors*, ANSI/ANS-5.1-1979, August 1979.

55. Amendment No. 126 to Facility Operating Licence No. DPR-58 (TAC No. 7106), for D. C. Cook Nuclear Plant Unit 1, Docket No. 50-315, June 9, 1989.

56. 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.

57. Letter Dated January 28, 1997, Serial No. 96-516A, From Virginia Power to the NRC, Generic Letter 96-06.

58. T.G. Carson, *Critical Calculation Review, Systems Design Basis Documents, North Anna Power Station,* Stone & Webster Engineering Corporation, to D.L. Benson, Virginia Power, November 20, 1989.

59. J. G. Knudsen and R. K. Hilliard, 1969, *Fission Product Transport by Natural Processes in Containment Vessels,* Battle-Northwest, Richland, Washington, BNWL-943.

60. *CORHYD - A Computer Program to Calculate Hydrogen Concentrations After a Design Basis Accident,* NU-111, User's Manual Reissued 1983, Stone & Webster Engineering Corporation. (Proprietary)

Insert 2