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May 18, 1994
Refer to: RC-94-0141

Document Control Desk
U. S. Nuclear Regulatory Commission
Washington, DC 20555

Attention: Mr. G. F. Wunder

Gentlemen:

Subject: VIRGIL C. SUMMER NUCLEAR STATION
DOCKET NO. 50/395
OPERATING LICENSE NO. NPF-12
RESPONSE TO NRC REQUEST FOR ADDITIONAL INFORMATION
FOR STEAM GENERATOR REPLACEMENT REVIEW (TAC NO. M88172)
DATED APRIL 25, 1994

On April 25, 1994, the NRC issued a request for additional information for the Steam Generator Replacement (SGR) review. Attachment I provides South Carolina Electric & Gas Company's (SCE&G) response to the NRC questions. Also attached is a revision to pages xiv and 3.2-3 of the October 29, 1993, submittal supporting SGR Technical Specification changes. A minor revision to the LOCA hydraulic loads on the core barrel has been made. The revised loads remain well below the current design basis analysis.

I declare that the statements and matters set forth herein are true and correct to the best of my knowledge, information, and belief. If you have any questions, please contact April Rice at (803) 345-4232.

Very truly yours,

John Skolds

John L. Skolds

ARR:lcd
Attachments

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1. **QUESTION - Provide a list of all NSSS and NSSS/BOP piping systems that are affected by the replacement of the steam generators.**

RESPONSE - The following portions of NSSS and BOP piping systems are directly affected by the replacement of the steam generators:

NSSS:

Reactor Coolant System piping encompassing the pressurizer surge line and the hot leg pipe, cold leg pipe, and cross over leg pipe of the three Reactor Coolant Loops (RCL).

BOP:

- Main Steam (MS) System piping from the Steam Generators (SG) to the Reactor Building penetrations (RBP).
- Feedwater (FW) System piping from the SGs to the RBPs.
- Emergency Feedwater (EF) System piping from the SGs to the RBPs.
- Blowdown (BD) System piping (including secondary drain lines) from the SGs to the RBPs.

In addition, the results of the RCL analysis were incorporated into a snubber reduction program for the ASME Class 1 auxiliary piping (including their ASME Class 2 and 3 extensions up to the first anchor) connected to the RCL.

2. **QUESTION - For each system, please provide the following information:**

   a. **The number of mechanical/hydraulic snubbers, rigid supports, and pipe whip restraints before SG replacement.**

   b. **The number of mechanical/hydraulic snubbers, rigid supports, and pipe whip restraints after SG replacement.**

   c. **The number of snubbers replaced by rigid supports, if any.**

RESPONSE - NOTE: The information requested is provided in the table below. Numbers in parentheses refer to NOTES following this table.

Piping System (as described in response to Question 1)	Snubbers Before SG Replcmnt	Rigid (4) Before SG Replcmnt	Whip Res Before SG Replcmnt	Snubbers After SG Replcmnt	Rigid (4) After SG Replcmnt	Whip Res After SG Replcmnt	Snubbers Replaced by Rigid Strut (3)
RCL	15 (1)	Static-0 Dynamic-6 (6) (2)	0	6 (1)	Static-0 Dynamic-0 (2)	0	0
Surge Line	4	2	6	1	2	6	0
MS	12	3	15	9	6	15	3
FW (5)	11	7	22	0	22	11	0
EF (5)	9	30	0	2	38	0	0
BD	32	11	4	32	11	4	0

NOTES:

(1) These are hydraulic snubbers supporting the upper portion of the Steam Generators.

(2) Structural Members (e.g., columns, bumpers, etc.) supporting the SGs, Reactor Coolant Pumps, and the Reactor Vessel are not included.

(3) Snubbers were not directly replaced with a rigid strut without a supporting piping reanalysis. Based on the results of the reanalysis, snubbers were left in place, removed, or replaced by a rigid support.

(4) This represents the number of locations where a rigid support is attached to the pipe. The rigid support at each of these locations could restrain the pipe in any combination of the three orthogonal directions.

(5) The FW and EF piping has been significantly rerouted to accommodate the new nozzle locations of the replacement SG. Therefore, a direct before and after comparison of numbers of supports is not meaningful.

(6) Prior to the approval by Reference 1 of leak-before-break technology for the elimination of postulated RCL pipe breaks, these dynamic supports performed a dual role of whip restraint and seismic restraint.

3. **QUESTION - What is the Code of Record for the reevaluated NSSS/BOP piping systems? Is this the same Code to which the systems were originally designed?**

RESPONSE -

NSSS:

The RCL piping, including the surge line, was reevaluated (due to the SG replacement) using the ASME Code, Section III, 1971 Edition, through the Winter of 1971 Addenda, except for fatigue qualification which used the 1977 Edition through the Summer of 1979 Addenda. The original design for RCL piping and the surge line used the same ASME Code Edition (the 1971 Edition through the Winter of 1971 Addenda) for the entire evaluation including fatigue qualification. RCL supports were qualified to ASME Section III, 1974 Edition, in both the original evaluation and the reevaluation of SG replacement.

BOP:

All the BOP Piping was reevaluated to the ASME Code, Section III, 1971 Edition, through the Summer of 1973 Addenda. The BOP pipe supports used the 1971 Edition through the Winter of 1973 Addenda. These were the same ASME Code editions and addenda used for the original design.

4. **QUESTION - Provide justification for basing the design of the SGs on the 1986 Edition of the Code, Section III.**

RESPONSE - The 1986 Edition of ASME Section III was the latest NRC approved edition of the Code (reference 10CFR50.55a) at the time the bid specification for purchasing the replacement SGs was developed. A reconciliation to the original SG Code of Record (1971 Edition through Summer 1971 Addenda) was performed.

5. **QUESTION - Provide the basis for the postulation of pipe breaks at VCSNS.**

RESPONSE -

NSSS:

Reference 1 approved the use of leak-before-break technology for the elimination of the dynamic effects of postulated RCL pipe breaks from the design basis of VCSNS. Therefore, RCL pipe breaks are not considered in the analysis of RCL piping or the qualification of RCL supports. However, the effects of pipe breaks at RCL branch piping nozzles are still considered.

BOP:

Pipe break postulation for BOP piping is based upon VCSNS FSAR Sections 3.6.2.1.1 and 3.6.2.1.2. A copy of these FSAR sections is attached for your convenience.

Reference 1 - NRC SER from G. Wunder to J. Skolds dated 1/11/93.

NOTE: Instrumentation and sample lines are not included in this response. These lines consist of short lengths of pipe (3/4" for instrumentation and 2" for sampling) from the SG to an isolation valve. Downstream of the isolation valve is 3/8" tubing.

3.6.2.1.1 High Energy System Piping Outside Containment

Breaks are postulated to occur in ASME Code, Section III, Class 2 and 3 piping and branch runs at the following locations:

1. At terminal ends. The terminal end for piping which penetrates containment is the pipe to penetration weld (see Figure 3.8-15) outside the reactor building.

2. At intermediate locations selected by either one of the following criteria:

   a. At each pipe fitting.

   b. At each location where the stresses exceed 0.8 ($1.2S_h + S_a$). Stresses are determined under the combination of loadings associated with the OBE and the nominal and upset plant condition loadings.

Breaks in non-nuclear safety class piping are postulated to occur at the following locations in each piping or branch run:

1. At terminal ends.

2. At each intermediate pipe fitting, welded attachment and valve.

Circumferential breaks are postulated to occur in fluid system piping and branch runs with nominal pipe size in excess of 1 inch.

Where a pipe elbow break location is selected without benefit of stress calculations, the pipe-to-elbow weld that joins the elbow to the shorter straight piping run is considered as the location of the break. Where break locations are selected in full size branch connection tees without benefit of stress calculations, the two pipe-to-tee welds that join the tee to the shorter straight piping runs are considered to be break locations. Where break locations are selected in reduced size branch connection tees without benefit of stress calculations, the pipe to tee weld that joins the tee to the shorter, straight, main piping run and the pipe to tee weld that joins the tee to the branch piping run are considered to be break locations.

Longitudinal breaks are postulated to occur in high energy system piping at the location of each postulated circumferential break except at the terminal ends under conditions discussed below:

1. Longitudinal breaks are not postulated to occur in high energy system piping and branch runs of nominal 3 inch pipe size and smaller.

2. Longitudinal breaks are postulated to occur in addition to, but not concurrently with, circumferential breaks.

3. Longitudinal breaks are not postulated to occur at terminal ends if the system piping at the terminal ends contains no longitudinal pipe welds.

4. Longitudinal breaks are assumed to result in an axial pipe split without pipe severence. Splits are oriented at two diametrically opposed points on the circumference of the pipe or fitting such that a jet reaction results that is normal to the plane formed by two of the applicable orthogonal axes, x, y, and z, of the piping configuration.

3.6.2.1.1.1 Conformance to Branch Technical Positions APCSB 3-1 [13] and MEB 3-1 [14]

An analysis has been performed which demonstrates that acceptable protection against the effects of piping failures outside containment has been provided. This analysis satisfies the intent of the guidelines of Branch Technical Positions (BTP) APCSB 3-1 and MEB 3-1. Since these positions were published a considerable period of time after the Virgil C. Summer Nuclear Station pipe rupture analysis had commenced, certain requirements could not be followed.

However, in certain respects, the design and analyses to cope with postulated pipe rupture for the Virgil C. Summer Nuclear Station are more stringent than required by the previously referenced BTPs. Specific differences are as follows:

1. APCSB 3-1:

    Paragraph B.3.b(1): Offsite power was assumed to be unavailable for all postulated piping failures.

2. MEB 3-1:

   a. Paragraph B.2.c(2): Cracks are postulated in most moderate energy fluid system piping, even where the maximum stress range is less than 0.4 ($1.2\ S_h + S_A$).

   b. Paragraph B.3.a(1): Circumferential breaks are postulated at locations where the circumferential stress range is at least 1.5 times the axial stress range.

   c. Paragraph B.3.b(1): Longitudinal breaks are postulated at locations where the axial stress range is at least 1.5 times the circumferential stress range.

   d. Paragraph B.3.b(2)(b): Longitudinal breaks are postulated where the criterion for a minimum number of break locations must be satisfied.

Criteria stated in the previously referenced BTPs with which the analysis does not fully comply and the alternative approaches are as follows:

1. APCSB 3-1:

   a. Paragraph B.2.c(1): The fluid system piping between containment isolation valves is not designed to the stress limits specified in Paragraphs B.1.b or B.2.b of BTP MEB 3-1. Breaks or cracks, as appropriate, are postulated in these portions of the fluid system piping in accordance with the criteria stated in Section 3.6.2.1.1.

   b. Paragraph B.2.d(2): For these portions of fluid system piping identified in Paragraph B.2.c, the inservice examination will be that required by the ASME Code, Section XI.

   c. Paragraphs B.3.b and B.3.d: The effects of an environmentally-induced failure caused by a leak or rupture which would not of itself result in protective action may include a loss of redundancy in the protective function, but not a loss of the protective function, as permitted by BTP-APCSB 3-1, Appendix B, paragraph 11.b.[15]. In these cases, plant shutdown is required. The use of Appendix B in lieu of BTP-APCSB 3.1 is permitted by the implementation schedule of paragraph B.4-c, since the V. C. Summer construction permit is dated March, 1973. Other criteria of BTP-APCSB 3.1 for single failure analyses are met, including the paragraph B.3 criteria for evaluating effects of cracks in moderate energy lines.

2. MEB 3-1:
Paragraph B.3.a(2): Break locations are selected in accordance with the criteria stated in Section 3.6.2.1.1.

3.6.2.1.2 High Energy System Piping Inside Containment

Breaks are postulated to occur in reactor coolant piping systems as discussed by WCAP-8172-A[5]. Additional details are presented in Section 3.6.2.1.3.

Breaks are postulated to occur in ASME Code, Section III, Class 1, piping, other than piping discussed in Reference [5], at the following locations in each piping or branch run:

1. At the terminal ends.

2. At any intermediate location between terminal ends where the primary plus secondary stress intensities (circumferential or longitudinal) derived on an elastically calculated basis under loadings associated with specific seismic events and normal and upset operational plant conditions exceed $2.4S_m$.

3. At any intermediate location between terminal ends where the cumulative usage factor, U, derived from the piping fatigue analysis under the loadings associated with specified seismic events and normal and upset plant operational conditions exceeds 0.1.

Breaks are postulated to occur in ASME Code, Section III, Class 2 and 3 piping at the following locations in each piping or branch run:

1. At the terminal ends. The terminal end for piping which penetrates containment is the pipe to penetration weld (see Figure 3.8-15) inside the reactor building.

2. At intermediate locations selected by either one of the following criteria:

   a. Each pipe fitting.

   b. Any location where either the circumferential or longitudinal stresses, derived on an elastically calculated basis under loadings associated with specified seismic events and normal and upset operational plant conditions exceeds 0.8 (1.2 S_h + S_a).

The following types of breaks are postulated to occur at locations previously identified for ASME Code, Section III, Class 1, 2, and 3 piping:

1. Circumferential breaks in piping runs and branch runs exceeding 1 inch nominal pipe size.

2. Longitudinal breaks in piping runs and branch runs of 4 inch nominal pipe size and larger except as discussed in item 3, below.

3. Longitudinal breaks are not postulated to occur at terminal ends if system piping at the terminal ends does not contain longitudinal pipe welds.

Where break locations are selected without benefit of stress calculations, breaks are postulated to occur at the piping welds to each fitting or valve.

Longitudinal breaks are assumed to result in an axial split without pipe severence. Splits are oriented at two diametrically opposed points on the circumference of the pipe or fitting such that a jet reaction results that is normal to the plane formed by two of the applicable orthogonal axes, x, y, and z, of the piping configuration.

Revision to 10/29/93 SGR Submittal

Pages xiv and 3.2-3

A brief summary of the results of each analysis, evaluation, and supporting documentation contained in this submittal is as follows:

Basis of Evaluations/Analyses:

The analyses and evaluations performed to support the RSGs bound a range of operating conditions for VCSNS. Four cases are presented which define a range of primary operating temperatures from 572°F to 587.4°F and a range of steam generator tube plugging levels from 0% to 10%. This will provide SCE&G with the flexibility to select the appropriate primary temperatures on a cycle-by-cycle basis necessary to achieve full megawatt electric output and to adjust the temperature as necessary to compensate for steam generator tube plugging or to perform end-of-cycle T_{avg} coastdown.

Large Break LOCA:

The Large Break (LB) LOCA analysis was performed at the current power level of 2775 MWt (core power) with the NRC approved ECCS Evaluation Model using the BASH code. In addition to the 2775 MWt power level assumed, the analysis assumed a total peaking factor (F_Q) of 2.45 and a hot channel enthalpy rise factor ($F_{\Delta H}$) of 1.62. An initial RCS pressure of 2300 psia and a TDF of 277,800 gpm was also assumed. A complete spectrum of breaks was analyzed for the Large Break LOCA along with a maximum safety injection case for a vessel average temperature range of 572°F to 587.4°F. Additional input assumptions for the Large Break LOCA analysis are listed in Table 3.1-1. All LOCA acceptance criteria as described in 10CFR50.46 were met and it was concluded that operation of the VCSNS with the Δ75 steam generators is acceptable with respect to LB LOCA. The detail and results of the Large Break LOCA analysis can be found in Section 3.1 and Appendix 6 (Chapter 15 FSAR writeups).

Small Break LOCA:

The Small Break LOCA analysis was performed using the NOTRUMP code and the small break version of the LOCTA code. Analyses were performed for the 1.5, 2, and 3 inch break sizes. The Small Break LOCA analysis was performed for a core power level of 2900 MWt with a total core peaking factor (F_Q) of 2.45, a hot channel enthalpy rise factor ($F_{\Delta H}$) of 1.62, and a hot assembly average power (P_{HA}) of 1.443. The Δ75 replacement steam generators were modeled in the analysis, assuming a 10% steam generator tube plugging level with a thermal design flow of 277,800 gpm. Analyses were performed for the range of reactor coolant average temperatures from 572.0 to 587.4°F. The major input assumptions for the analysis are summarized in Table 3.1-2. The results of the Small Break LOCA analysis indicate that the LOCA acceptance criteria in 10CFR50.46 will continue to be met. The details and results of the Small Break LOCA analysis can be found in Section 3.1 and Appendix 6 (Chapter 15 FSAR writeups).

Other LOCA Analyses:

Post-LOCA Long Term Core Cooling Subcriticality (Section 3.1.3) and Hot Leg Switchover to Prevent Boron Precipitation (Section 3.1.4) analyses were performed, incorporating revised operating conditions and the Δ75 steam generators. The analysis for the post-LOCA long term core cooling showed that the reactor core remains subcritical assuming all control rods out. The analysis for the hot leg switchover to prevent potential boron precipitation showed that switchover to hot leg recirculation within 8 hours of a LOCA will prevent boron precipitation in the reactor vessel. Both analyses concluded that operation of the VCSNS with the RSGs and a power level up to 2912 MWt is acceptable.

The LOCA hydraulic forces analysis performed for the RSG conditions (Table 2.1-1) postulated auxiliary line breaks and used the NRC approved Leak-Before-Break (LBB) methodology. The LOCA hydraulic forces were generated for the vessel, loop, and the RSG (Δ75). The peak lateral LOCA hydraulic load
5/94 on the core barrel was determined to be 4.99×10^6 lbf, which is approximately 19.4% lower in magnitude than the previous analysis, which considered a 150 in^2 rupture of the reactor vessel inlet nozzle. The analysis, thus, yielded considerable margin to the analyses contained in Sections 3.6.2.2.1 and 3.9.3.5

that breaks in the cold leg produce greater peak loads than those postulated elsewhere in the RCS. The most limiting break location for the hot leg is the pressurizer surge line. These two breaks form the break spectrum analyzed for the VCSNS RSG conditions.

In addition to the postulated break location and area, the severity of the postulated pipe rupture is a function of the decompression path through the system, the break opening time, and the operating conditions of the plant at the time of the postulated rupture. The break opening time used in the analysis was a conservative one millisecond as required by the NRC in their topical evaluation report for Reference 3. The thermal parameters of case 4, Table 2.1-1 were incorporated in the analysis. This case corresponded to Vantage+ fuel with IFM grids, thimble plugs removed, 10% steam generator tube plugging and a low vessel/core inlet temperature of 536.6°F. Additionally, the analysis incorporated RCS temperature and pressure uncertainties of ±5.5°F and ±50.0 psi. These parameters form a conservative lower bound RCS temperature, upper bound RCS pressure case for the revised plant conditions.

The LOCA hydraulic forces were generated for the vessel, loop and the replacement (Δ75) steam generator. The peak lateral LOCA hydraulic load on the core barrel was determined to be
5/94 4.99×10^6 lbf, which is approximately 19.4% lower in magnitude than the previous analysis, which considered a 150 in^2 rupture of the reactor vessel inlet nozzle. The analysis, thus, yielded considerable margin to the FSAR analyses contained in Sections 3.6.2.2.1 and 3.9.3.5. The Model D3 steam generator structural analysis was performed with double-ended guillotine inlet and outlet nozzle breaks, which yield far larger loads than auxiliary line breaks analyzed. The new LOCA hydraulic forcing functions were used for qualification of components in combination with other applicable design basis loads.

3.2.4 References

1. WCAP-13206, "Technical Justification for Eliminating Large Primary Loop Rupture as the Structural Design Basis for the Virgil C. Summer Nuclear Power Plant," April 21, 1992.

2. NRC Docket No. 50-395, "Safety Evaluation of Request to Use Leak-Before-Break for Reactor Coolant System Piping - Virgil C. Summer Nuclear Station, Unit No. 1 (TAC No. M83971)," G. F. Wunder, 1/11/93.

3. WCAP-8708, "MULTIFLEX, a FORTRAN-IV Computer Program for Analyzing Thermal-Hydraulic-Structure System Dynamics," 1977.