June 16, 1986

Docket Nos. 50-280
and 50-281

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Mr. W. L. Stewart
Vice President - Nuclear Operations
Virginia Electric and Power Company
Post Office Box 26666
Richmond, Virginia 23261

Dear Mr. Stewart:

The Commission has issued the enclosed Amendment No. 108 to Facility Operating License No. DPR-32 and Amendment No. 108 to Facility Operating License No. DPR-37 for the Surry Power Station, Unit Nos. 1 and 2, respectively. The amendments consist of changes to the license in response to your application transmitted by letter dated April 30, 1986.

These amendments permit plant operation with the reactor coolant pump and steam generator supports redesigned in accordance with the recently noticed amendment to General Design Criterion 4 (GDC-4), 10 CFR Part 50, Appendix A (51 FR 12502), which was effective May 12, 1986.

A copy of the related Safety Evaluation is enclosed. A Notice of Issuance will be included in the Commission's next regular bi-weekly Federal Register notice.

Sincerely,

/s/

Chandu P. Patel, Project Manager
PWR Project Directorate #2
Division of PWR Licensing-A
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 108 to DPR-32
2. Amendment No. 108 to DPR-37
3. Safety Evaluation

cc: w/enclosures
See next page

LA:PAD#2	PM:PAD#2	C:EB:DPL-A	D:PAD#2	OELD
DMiller	CPatel:hc	RBallard	LRubenstein	J. GRAY
6/5/86	6/5/86	6/11/86	6/16/86	6/13/86

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

VIRGINIA ELECTRIC AND POWER COMPANY

DOCKET NO. 50-280

SURRY POWER STATION, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 108
License No. DPR-32

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Virginia Electric and Power Company (the licensee) dated April 30, 1986, 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 the addition of the following license condition:

3M. The design of the reactor coolant pump and steam generator supports may be revised in accordance with the licensee's submittals dated November 5, 1985 (Serial No. 85-136), December 3, 1985 (Serial No. 85-136A), and January 14, 1986 (Serial No. 85-136C).

3. This license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Daniel G. McDonald
Acting for
Lester S. Rubenstein, Director
PWR Project Directorate #2
Division of PWR Licensing-A

Date of Issuance: June 16, 1986

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

VIRGINIA ELECTRIC AND POWER COMPANY

DOCKET NO. 50-281

SURRY POWER STATION, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 108
License No. DPR-37

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Virginia Electric and Power Company (the licensee) dated April 30, 1986, 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 the addition of the following license condition:

3M. The design of the reactor coolant pump and steam generator supports may be revised in accordance with the licensee's submittals dated November 5, 1985 (Serial No. 85-136), December 3, 1985 (Serial No. 85-136A), and January 14, 1986 (Serial No. 85-136C).

3. This license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Daniel G. McDonald

Acting Lester S. Rubenstein, Director
for PWR Project Directorate #2
Division of PWR Licensing-A

Date of Issuance: June 16, 1986

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 108 TO FACILITY OPERATING LICENSE NO. DPR-32

AND AMENDMENT NO. 108 TO FACILITY OPERATING LICENSE NO. DPR-37

VIRGINIA ELECTRIC AND POWER COMPANY

SURRY POWER STATION, UNIT NOS. 1 AND 2

DOCKET NOS. 50-280 AND 50-281

INTRODUCTION

On April 11, 1986, a notice of modification of General Design Criterion 4 (GDC-4) of Appendix A, 10 CFR Part 50, was published in the Federal Register. The modified rule, effective May 12, 1986, allows exclusion from the design basis of the dynamic effects of postulated ruptures in primary coolant loop piping when analyses demonstrate the probability of rupturing such piping is extremely low under design basis conditions. Specifically, leak-before-break (LBB) technology could be employed to demonstrate such low probability. This consists of the use of advanced fracture mechanics analysis techniques to demonstrate the capability to detect leakage well before any cracks in the pipe wall can become unstable and grow to failure. Containment design, emergency core cooling, and environmental qualification requirements are not influenced by this rule modification. The Supplementary Information accompanying the rule indicates that modifications, similar to those covered by this request, may involve an unreviewed safety question, which in turn requires NRC approval.

NRC Generic Letter 84-04, dated February 1, 1984, informed all operating pressurized water reactor (PWR) licensees that NRC had completed its review of Westinghouse reports WCAP-9958, Revision 2, and WCAP-9787, Revision 0, dealing with elimination of postulated pipe breaks in PWR primary main loops. The Westinghouse reports had been submitted to address asymmetric blowdown loads on PWR primary systems that resulted from a limited number of discrete break locations as stipulated in NUREG-0609, the NRC staff's resolution of Unresolved Safety Issue A-2. The generic letter stated that an acceptable technical basis has been provided so that the asymmetric blowdown loads resulting from double-ended pipe breaks in main coolant loop piping need not be considered as a design basis for certain plants (including Surry Units 1 and 2) provided certain conditions were met.

By letters dated November 5, and December 3, 1985, and January 14, 1986, Virginia Electric and Power Company (the licensee) requested an exemption from certain technical requirements of GDC-4. Specifically, the licensee proposed that protection against the dynamic effects of pipe rupture on primary system components and piping be eliminated, which would permit the removal and elimination of 18 large bore snubbers on each unit's reactor coolant system originally installed solely to mitigate the effects of a pipe rupture event.

By letter dated April 30, 1986, the licensee requested an amendment, in the form of a license condition, to Facility Operating License Nos. DPR-32 and DPR-37 for Surry Units 1 and 2 based on the documentation previously submitted in support of its exemption request. The proposed amendments would add a license condition stating that the design of the reactor coolant pump and steam generator supports may be revised in accordance with the submittals dated November 5, 1985, December 3, 1985, and January 14, 1986.

EVALUATION

The Supplementary Information in the April 11, 1986, notice of modification of GDC-4 provides guidance that the following should be performed in applying the modified rule:

- Plant unique analyses to demonstrate adequate margins for all remaining loads.
- Confirmation that Surry Units 1 and 2 fall within the vendor-calculated envelope.
- Demonstration of improved overall system performance and reliability compared with the previous support system.
- Consideration of independent design and fabrication verification.
- Demonstration that leak detection capability is adequate.

We have reviewed the licensee's submittals with regard to the above guidance. Our evaluation is as follows:

A. Analysis

The licensee has proposed to eliminate snubbers which are required solely to mitigate a pipe rupture event. Specifically, the licensee has requested to:

- eliminate two snubbers per loop which are parallel to the reactor coolant cold leg at the reactor coolant pump support.
- eliminate four snubbers per loop which are parallel to the reactor coolant hot leg at the steam generator lower support ring.
- eliminate the LOCA pipe rupture loads postulated for the four snubbers per loop which are at the steam generator upper support rings.

In addition, the licensee proposes to eliminate four small bore snubbers that act as upper diagonal braces for each of the three reactor coolant pump supports.

The licensee has submitted the results of a structural analysis of the RCS with proposed modifications in the reactor coolant pump and steam generator supports.

With respect to pipe stress and loading, two independent analyses of a representative single primary reactor coolant loop were performed for the licensee. Westinghouse Electric Corporation performed an analysis to obtain piping stresses, and Stone & Webster Engineering Corporation (SWEC) performed an analysis to obtain component support loads. The SWEC analytical model incorporated a detailed representation of the support members. This division of analytical responsibility between the two organizations is similar to the original division of design responsibility. Both analytical models were revisions to existing models and incorporated changes due to earlier steam generator replacement efforts and the proposed snubber removal.

Both of these analyses incorporated the loads from: dead weight, thermal expansion, internal pressure, seismic events (OBE and DBE), and dynamic effects of pipe ruptures of other systems (controlling breaks, for example, in the main steam line, pressurizer surge line, etc). For additional conservatism, these analyses used low equipment damping values, i.e., 0.5 percent for OBE and 1 percent for DBE. These values are lower than those in Regulatory Guide 1.61, "Damping Values for Seismic Design of Nuclear Power Plants" and ASME Code Case N-411, "Alternative Damping Values for Seismic Analysis of Classes 1, 2 and 3 Piping."

The results of these analyses were compared with the allowable stresses and loads which are currently documented in the Updated FSAR for Surry Power Station, Units 1 and 2. All of the calculated stresses and loads from both of the analyses were below these allowable values.

In addition, the maximum resultant bending moment in the primary coolant loop piping was calculated to be 28,860 in-kips in the steam generator outlet nozzle/piping junction. This value is less than the 42,000 in-kips moment which was identified in NRC Generic Letter 84-04 as the maximum allowable moment for the Westinghouse Owner's Group plants for justifying that pipe rupture need not be postulated in the main reactor coolant loop piping.

In a letter dated January 14, 1986, the licensee provided additional information in support of the request for approval to remove snubbers. This additional information pertained to a summary of peak pipe stresses resulting from controlling pipe breaks in the main steam and the pressurizer surge lines. The stress summary indicates all stress values to be within the allowable limits permitted by the conservative licensing criteria applicable to Surry Units 1 and 2.

The licensee also made a comparison of the maximum pipe stress locations before and after the elimination of selected snubbers. For seismic loads, the location of the maximum stress is the steam generator outlet elbow of the reactor coolant system (RCS) piping; for the main steam line rupture loading, the location of maximum stress is in the steam generator inlet elbow of the RCS; for pressurizer surge line break, the maximum stress occurs in the hot leg where the surge line intersects. These locations for maximum stresses did not change as a result of the proposed elimination of the selected snubbers.

The licensee provided a comparison of the factors of safety for the component supports with and without snubber configurations for the load cases involving required pipe break loads. This comparison shows that acceptable factors of safety exist under the original licensing criteria, which is more conservative than currently used criteria.

The licensee also provided additional information regarding the elimination of small bore diagonal snubbers from the reactor coolant pump (RCP) support and possible pump displacements under the rocking mode of vibration under earthquake loading, which show that the natural frequencies for significant motion of the RCP did not change substantially (17% drop in the frequency of rocking mode of vibration). The dynamic behavior of the RCP for the applicable loads remains virtually unaffected by the proposed removal of the snubbers.

Based on the review of the above information, the staff concludes that the licensee has provided acceptable plant specific analysis.

B. Vendor-Calculated Envelope

As discussed previously, the maximum resultant bending moment in the primary coolant loop piping was demonstrated to be within the envelope of the maximum allowable moment for the Westinghouse Owner's Group plants for justifying that pipe rupture need not be postulated in the main reactor coolant loop piping. The staff finds this to be acceptable.

C. Overall System Performance and Reliability

The licensee indicated that the large bore snubbers, being active components, require periodic removal for functional testing and implementation of a seal service life program. Also, removal/inspection activities of large bore snubbers have exposed maintenance personnel to high radiation because the snubbers are located in the reactor containment cubicles. The deletion of these snubbers will eliminate this source of occupational exposure and facilitate maintenance and in-service inspections of piping and components by reducing plant congestion.

The licensee also indicated that support system reliability is also increased with the removal of these active elements. Inadvertent lockup, bleed rate variance and hydraulic fluid leakage are possible problems related to larger bore snubbers that are eliminated.

The staff agrees with the above assessment and concludes that removal of large-bore snubbers will result in net safety benefit.

D. Independent Review

The snubber elimination program involves relatively simple construction activity and does not involve any modification of existing supports. The geometry of the RCS piping and components was thoroughly checked during the steam generator replacement effort. Also, the previous RCS piping analyses activities and the current analytical effort for the snubber elimination program was performed by almost totally new personnel with years intervening between the two events. Therefore, a substantial amount of scrutiny and checking of the old as well as the new analyses took place.

E. Leak Detection Capability

The leakage detection capability at Surry Power Station, Units 1 and 2, for unidentified leakage, includes the following:

a) containment air particulate monitor,
b) containment radioactive gas monitor,
c) monitoring for abnormal addition of make-up water,
d) containment sump level monitor, and
e) containment pressure, temperature and humidity monitors.

The Surry Technical Specifications (TS) require two leakage detection systems to be operable, with one based on the detection of radionuclides in the containment. Both the containment air particulate monitoring system and the containment radioactive gas monitoring system have the capability to detect leakage of 1 gpm in 4 hours; indication, recording and alarm are provided in the control room. The containment sump level monitoring system can also detect leakage of 1 gpm in 4 hours; the unidentified leakage rate is calculated based on the sump level change and the time interval between sump pump actuations. Sump level indication and a high level alarm are provided in the control room. A leak rate of 1 gpm in 4 hours can also be detected by the abnormal make-up water monitoring system. The primary grade water and concentrated boric acid make-up flow rates, to maintain pressurizer level, are both recorded and alarmed in the control room. Parameters are monitored every 4 hours to calculate the leakage rate. Plant operation is governed by the Technical Specifications through a prescribed leakage limit such that if the leakage rate exceeds 1 gpm and the source of leakage can not be identified in 4 hours, the reactor must be brought to hot shutdown.

Based on the above discussion, the staff concludes that leakage detection capability at Surry Power Station, Units 1 and 2 meets the staff guidelines of 1 gpm in 4 hours, as stated in Generic Letter 84-04.

Based on our evaluation as summarized above, we conclude that the licensee has satisfied the guidance accompanying the modified GDC-4 and therefore may exclude from the design basis for the reactor coolant pump and steam generator supports the dynamic effects associated with postulated ruptures of primary coolant loop piping and may modify the reactor coolant pump and steam generator supports as described in the references herein.

Environmental Consideration

These amendments involve a change in the installation or use of the facilities components located within the restricted areas as defined in 10 CFR 20. The staff has determined that these amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that these amendments involve no significant hazards consideration and there has been no public comment on such finding. Accordingly, these amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR §51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of these amendments.

Conclusion

We have concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (2) such activities will be conducted in compliance with the Commission's regulations and the issuance of these amendments will not be inimical to the common defense and security or to the health and safety of the public.

Dated: June 16, 1986

Principal Contributors:

E. Sullivan
G. Bagchi
C. Patel

References

1. Generic Letter 84-04, February 1, 1984, "Safety Evaluation of Westinghouse Topical Reports Dealing with Elimination of Postulated Pipe Breaks in PWR Primary Main Loops."

2. WCAP-9558, Revision 2 (May 1981) "Mechanistic Fracture Evaluation of Reactor Coolant Pipe Containing a Postulated Circumferential Through-wall Crack."

3. WCAP-9787 (May 1981) "Tensile and Toughness Properties of Primary Piping Weld Metal for Use in Mechanistic Fracture Evaluation."

4. Letter Report NS-EPR-2519, E.P. Rahe to D.G. Eisenhut (November 10, 1981) Westinghouse Response to Questions and Comments Raised by Members of ACRS Subcommittee on Metal Components During the Westinghouse Presentation on September 25, 1981.

5. Letter Serial No. 85-136, W.L. Stewart to H.R. Denton, "Request for Partial Exemption from General Design Criterion 4," dated November 5, 1985.

6. Letter Serial No. 85-136A, W.L. Stewart to H.R. Denton, "Request for Partial Exemption from General Design Criterion 4 - Supplement," dated December 3, 1985.

7. Letter Serial No. 85-136C, W.L. Stewart to H.R. Denton, "Partial Exemption from General Design Criterion 4 - Request for Additional Information," dated January 14, 1986.