March 30, 2005

Mr. James J. Sheppard
President and Chief Executive Officer
STP Nuclear Operating Company
South Texas Project Electric
Generating Station
P. O. Box 289
Wadsworth, TX 77483

SUBJECT: SOUTH TEXAS PROJECT, UNITS 1 AND 2 - ISSUANCE OF NONPROPRIETARY SAFETY EVALUATION RE: REVISION TO RETRAN-2 METHODOLOGY (TAC NOS. MC0759 AND MC0760)

Dear Mr. Sheppard:

On March 7, 2005, the Commission issued Amendment No. 171 to Facility Operating License No. NPF-76 and Amendment No.159 to Facility Operating License No. NPF-80 for the South Texas Project, Units 1 and 2, respectively. The amendments approved revisions to the RETRAN-02 methodology in response to your application dated May 13, 2003, as supplemented by letters dated October 6, 2004, November 30, 2004, and January 20, 2005.

A copy of the related Safety Evaluation (SE) was provided to you at that time for review to assure that any information that Westinghouse claims to be proprietary would not be inadvertently released to the general public. During a telephone conference on March 15, 2005, with Mr. S. Head, STP Nuclear Operating Company, information that Westinghouse claims to be proprietary was identified on page 3 of the SE. This information, which has been redacted from the enclosed nonproprietary SE, is indicated on page 3 by a vertical marginal line. The subject information was previously withheld from public disclosure in accordance with our letter to Mr. J. A. Gresham, Westinghouse Electric Company dated December 13, 2004.

Sincerely,

/RA/

David H. Jaffe, Senior Project Manager, Section 1
Project Directorate IV
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-498 and 50-499

Enclosure: Safety Evaluation

cc w/encl: See next page

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cc w/encl: See next page
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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NOS. 171 AND 159 TO

FACILITY OPERATING LICENSE NOS. NPF-76 AND NPF-80

STP NUCLEAR OPERATING COMPANY, ET AL.

SOUTH TEXAS PROJECT, UNITS 1 AND 2

DOCKET NOS. 50-498 AND 50-499

1.0 INTRODUCTION

In a letter dated May 13, 2003, as supplemented by letters dated October 6, 2004, November 30, 2004, and January 20, 2005 (References 1 through 4, respectively), South Texas Project Nuclear Operating Company (the licensee) requested revisions to the RETRAN-02 methodology that is used to evaluate certain design basis transients and accidents for the South Texas Project (STP), Units 1 and 2. In particular, the licensee believes that the current RETRAN-02 methodology is overly conservative for evaluation of certain design basis events involving loss of normal feedwater (LOFW), loss of offsite power (LOOP), and feedwater line breaks (FWLB). These events all involve reduction in the ability of the steam generators (SGs) to remove reactor heat causing the reactor temperature and pressure to increase.

The supplements dated October 6, 2004, November 30, 2004, and January 20, 2005, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the Nuclear Regulatory Commission (NRC) staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on November 12, 2003 (68 FR 64138).

2.0 REGULATORY EVALUATION

The staff reviewed the licensee's request in accordance with the guidance in the Standard Review Plan (SRP) (Reference 5) Sections 15.2.6 “Loss of Nonemergency AC Power to the Station Auxiliaries,” 15.2.7 “Loss of Normal Feedwater Flow,” and 15.2.8 “Feedwater System Pipe Breaks Inside and Outside Containment.”

The LOFW and the LOOP are classified as incidents of moderate frequency. The key SRP acceptance criteria for events of moderate frequency are summarized as follows:

1. Pressures in the reactor coolant system (RCS) and in the main steam system should be maintained below 110 percent of the design pressure.

2. Fuel cladding integrity shall be maintained by ensuring that the departure from nucleate boiling ratio (DNBR) limit is maintained to ensure a 95 percent

probability that critical heat flux (CHF) will not occur with a confidence of 95 percent for the hottest fuel pins of the reactor core.

The occurrence of FWLB up to the double ended guillotine severance of a main feedwater line are considered by the NRC staff to be design basis accidents. The key SRP acceptance criteria for the analysis of FWLBs are as follows:

1. Pressure in the RCS and main steam system should be maintained below 110 percent of the design pressures for most break sizes and below 120 percent of the design pressures for very low probability events such as the occurrence of a double ended guillotine break.

2. The potential for core damage that may occur during the transient is evaluated on the basis that it is acceptable if the minimum DNBR remains above the 95/95 DNBR limit. If fuel damage is calculated to occur, the damage must be of sufficiently limited extent that the core will remain in place and intact with no loss of core cooling capability.

3. Any activity release must be such that the calculated dose at the site boundary is a small fraction of the guidelines in Part 100 of Title 10 of the Code of Federal Regulations (10 CFR).

Computer code methodology used for analyses of design basis transients and accidents including the computer code input and calculational assumptions are to be reviewed and assured to be conservative for showing compliance with the acceptance criteria. The NRC staff concludes that, when the licensee utilizes the revised methodology, the licensee will meet the acceptance criteria.

3.0 TECHNICAL EVALUATION

The licensee believes that the current analyses for LOFW, LOOP, FWLB events are too conservative in the following two respects: (1) in the heat absorption from the reactor coolant to the thick structural metal of the reactor system pressure boundary and (2) in the determination of initial SG water mass.

The current methodology, which the licensee uses to evaluate LOFW, LOOP, and FWLB is described in WCAP-14882-P-A "RETRAN-02, Modeling and Qualification for Westinghouse Pressurized Water Reactor Non-LOCA Safety Analysis" (Reference 6) which has been approved by the NRC staff. RETRAN-02 is a flexible, general purpose, thermal/hydraulic computer code that is used to evaluate the effect of various upset reactor conditions on the RCS. WCAP-14882-P-A describes the input assumptions and code options that are used to simulate non-LOCA transients and accidents for 2-, 3-, and 4-loop reactors designed by Westinghouse including STP, Units 1 and 2.

The methodology for the revised RETRAN-02 input to take credit for the more realistic initial SG water mass as well as for thick metal heat absorption is described in WCAP-14882-S1-P and WCAP-15234-S1-NP (Reference 7) which are referenced by the licensee.

The RETRAN-02 modeling described in WCAP-14882-P-A currently includes modeling of the thick metal structures for events for which it is conservative to release heat from the metal structures into the reactor coolant. This is accomplished by a lumped node technique which maximizes heat transfer into the coolant from the metal structures. For events for which it is conservative to limit the amount of heat transfer from the coolant to the thick metal structures, the current model assumes there is no heat transfer. To take credit for a portion of the heat that is transferred to the metal surfaces, the licensee will use fine mesh detail which will more accurately calculate the amount of heat flow. The NRC staff questioned the adequacy of the noding detail used by the licensee and the heat transfer correlations that will be used. The licensee responded by comparing the results from the RETRAN-02 thick metal model with those calculated by the LOFTRAN code. The thick metal model had previously been approved for use with the LOFTRAN code for Byron Station, Units 1 and 2 and Braidwood Station, Units 1 and 2 (Reference 8). The RETRAN-02 results were found to be in good agreement with those calculated by LOFTRAN. The thick metal heat transfer model in LOFTRAN has been benchmarked against text book data (Reference 9) and found to be conservative. In applying the thick metal model to analyses for STP, Units 1 and 2, the licensee will only utilize a portion of the actual thick metal of the reactor system. In addition, a thermal conductivity of steel will be used which is much less than that of the STP reactor system piping. The NRC staff concludes that the RETRAN-02 thick metal model as described by the licensee is conservative for analysis of LOFW, LOOP, and FWLB events at STP, Units 1 and 2.

The licensee seeks to provide input to the RETRAN-02 code which better represents the initial water mass on the secondary side of the SGs. This is because the homogeneous flow of steam and water assumed in the RETRAN–02 input under-predicts the initial water mass, and is thus conservative. The water in the SGs acts as a heat sink to mitigate the predicted consequences of the LOFW, LOOP, and FWLB events for which the licensee proposes to utilize the revised model. The licensee proposes to utilize a better prediction of SG water mass from NOTRUMP. NOTRUMP SG modeling has previously been used in conjunction with LOFTRAN for FWLB analysis as described in WCAP-9230 (Reference 10). The results were accepted by the NRC staff in the safety evaluations for several operating plants. Instead of assuming homogeneous flow, NOTRUMP utilizes a drift flux model which calculates the individual velocities of steam and water. Since steam generally has a higher velocity than the water within a SG, the resulting water fraction is larger for the same amount of steam flow. Thereby, a greater amount of water is predicted to be in the SG nodes. The drift flux model was derived from data taken at the Westinghouse MB-2 scale model SG test facility. The test facility was designed to model a SG of the feedring type which is the design of the SGs at STP, Units 1 and 2. The total SG water mass predicted by NOTRUMP was compared with that predicted by the Westinghouse SG design codes and found to be acceptable. To provide conservatism in the calculation, the licensee will reduce the SG water masses calculated by NOTRUMP by [ ] before inputting the nodal masses into RETRAN-02. The NRC staff concludes that this approach is acceptable.

To provide a more accurate determination of reactor trip on low SG level, the licensee will determine the SG water mass at the time of reactor trip from the NOTRUMP analysis with allowance for instrument uncertainty and an additional reduction to provide conservative margin. This mass will then be used as the trip parameter in the RETRAN-02 model. The less accurate determination of SG level by RETRAN-02 will therefore not be utilized to determine the time of reactor trip. The NRC staff concludes that this approach is acceptable.

For analysis of events of moderate frequency such as LOOP and LOFW, the licensee will apply the acceptance criteria from the Standard Review Plan and will, in addition, apply a Westinghouse acceptance criterion which requires that complete filling of the pressurizer will not be predicted. This restriction prevents water discharge from the pressurizer safety or relief valves which might cause damage to the valve seats.

Following a FWLB, the rapid reduction of water inventory from the affected SG causes a reduction of heat removal capability, thereby causing reactor system heatup. The RETRAN-02 code generally predicts a more rapid discharge of water from the affected SG than does the NOTRUMP code because in the RETRAN-02 model, the water exiting the SG is assumed to have the same velocity as the steam. The licensee will continue to use the more conservative RETRAN-02 model to predict water loss from the affected SG with the initial water mass determined using NOTRUMP. For analysis of postulated main feedwater line breaks, the licensee will apply the acceptance criteria from the SRP and will, in addition, apply a Westinghouse acceptance criterion which requires that the temperature of the water in the hot legs remains less than the boiling temperature. Meeting this criterion is one way of ensuring that any damage to the core following a FWLB will be minimal.

Based on the supporting information provided by the licensee which demonstrates the conservatism in the models, the NRC staff accepts use of the methodology in WCAP-14882-S1-P and WCAP-15234-S1-NP for analysis of LOOP, LOFW, and FWLB for the STP, Units 1 and 2. The NRC staff review utilized analyses and supporting experimental data supplied by the licensee that are specific to reactor system designs similar to STP, Units 1 and 2. The NRC staff will therefore, require that licensees seeking to apply this methodology for analyses of other nuclear power plants provide supporting justification that use of this methodology is appropriate and conservative for their designs.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Texas State official was notified of the proposed issuance of the amendment. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendments change a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the 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 the amendments involve no significant hazards consideration, and there has been no public comment on such finding published on November 12, 2003 (68 FR 64138). Accordingly, the 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 the amendments.

6.0 CONCLUSION

The Commission has 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, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

7.0 REFERENCES

1. Letter from T. J. Jordan, South Texas Project Nuclear Operating Company, to U. S. Nuclear Regulatory Commission, “License Amendment Request for Approval of a Change in Analytical Methodology,” May 13, 2003, Accession No. ML031400401.

2. Letter from T. J. Jordan, South Texas Project Nuclear Operating Company, to U. S. Nuclear Regulatory Commission, “Request for Additional Information Regarding a License Amendment Request for Approval of a Change in Analytical Methodology,” October 6, 2004, Accession No. ML042860042.

3. Letter from T. J. Jordan, South Texas Project Nuclear Operating Company, to U. S. Nuclear Regulatory Commission, “Complete Response to Request for Additional Information Regarding a License Amendment Request for Approval of a Change in Analytical Methodology”, November 30, 2004, Accession No. ML043410306.

4. Letter from T. J. Jordan, South Texas Project Nuclear Operating Company, to U. S. Nuclear Regulatory Commission, “Revised Response to Request for Additional Information Question 30 regarding Correlations Used in Computer Codes,” January 20, 2005, Accession No. ML050250195.

5. NUREG-0800, Revision 2, “U.S. Nuclear Regulatory Commission Standard Review Plan,” July 1981.

6. D. S. Huegel, et. al., “RETRAN-02 Modeling and Qualification for Westinghouse Pressurized Water Reactor Non-LOCA Safety Analysis,” WCAP-14882-P-A, Westinghouse Electric Corporation, April 1999.

7. WCAP-14882-S1-P and WCAP-15234-S1-NP, “RETRAN-02 Modeling and Qualification for Westinghouse Pressurized Water Reactors Non-LOCA Safety Analysis Supplement 1-Thick Metal Mass Heat Transfer Model and NOTRUMP-Based Steam Generator Mass Calculation Method,” December 2002.

8. Safety Evaluation by the Office of Nuclear Reactor Regulation Related to Amendment No. 119 to Facility Operating License No. NPF-37, Amendment No. 119 to Facility Operating Licensee No. NPF-66, Amendment No. 113 to Facility Operating License NO. NPF-72, and Amendment No. 113 to Facility Operating License No. NPF-77 - EXELON Generation Company, LLC - Byron Station Unit Nos. 1 and 2, Braidwood

Station Units Nos. 1 and 2, Docket Nos. STN 50-454, STN 50-455, STN 50-456 and STN 50-457, May 2001, Accession No. ML011420274.

9. P. J. Schneider, “Temperature Response Charts,” John Wiley and Sons, Inc., 1963.

10. G. E. Lang, et. al., “Report on the Consequences of a Postulated Main Feedline Rupture,” WCAP-9230, Westinghouse Electric Corporation, January 1978.

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South Texas Project, Units 1 & 2

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