Nuclear Operating Company

South Texas Project Electric Generating Station P.O. Box 289 Wadsworth, Texas 77483

January 21, 2002
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File No.: G25
10CFR50.90
STI:31396247

U. S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, DC 20555-0001

South Texas Project
Units 1 & 2
Docket Nos. STN 50-498, STN 50-499
Additional Information to Support the Request for Approval
of Power Uprate and a Revision to the Technical Specifications

References: 1) Letter from J. J. Sheppard to NRC Document Control Desk, "Proposed Amendment to Facility Operating Licenses and Technical Specifications Associated with a 1.4% Core Power Uprate," August 22, 2001 (NOC-AE-01001162)

2) Letter from M. C. Thadani, NRC, to W. T. Cottle, STPNOC, "South Texas Project Units 1 and 2: Request for Approval of Power Uprate and Revision to the Technical Specifications Supporting the Power Uprate," December 20, 2001 (ST-AE-NOC-02000907)

Reference 1 requested approval of increasing the plant operating power level by 1.4 percent and submitted a license amendment supporting associated revisions to Technical Specifications. Reference 2 requested that additional information from South Texas be submitted to the NRC in order for the staff to complete its evaluation. Attachments 1 and 2 provide a partial response to the NRC request. Response to the remainder of the NRC questions is targeted for the first week of February 2002.

There are no licensing commitments in this letter. If you should have any questions concerning this matter, please contact Mr. Ken Taplett at (361) 972-8416 or me at (361) 972-8757.

A001

I declare under penalty of perjury that the foregoing is true and correct.

Executed on: 1/21/02

J. J. Sheppard
Vice President,
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KJT/

Attachments: 1. Additional Information
2. UFSAR Tables

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ATTACHMENT 1

ADDITIONAL INFORMATION

By letter dated August 22, 2001, STP Nuclear Operating Company (STPNOC), the licensee for South Texas Project Units 1 & 2, requested a license amendment to raise the plant operating power level by 1.4 percent. The NRC staff reviewed the application and determined that it requires additional information to complete its review. The following are responses to some of the questions that the NRC transmitted to STPNOC by a letter dated December 20, 2001. The final submittal of the responses to the remaining questions is expected to meet the target date stated in the NRC letter.

Note: The numbers below correspond to the question numbers in the December 20th letter.

<u>Instrumentation & Controls</u>

1. **Section 3.6 in Attachment 6 states that with respect to the CROSSFLOW Ultrasonic flow measurement (UFM) uncertainties, Uncertainty calculations have been performed and determined a mass flow accuracy of better than 0.5 percent of rated flow for STP Units 1 and 2. The Westinghouse calculation in WCAP-15633 used 1 percent instrument uncertainty for the CROSSFLOW. Please explain, why the plant could not attain the calculated accuracy of 0.5 percent.**

   Response:

   As documented in WCAP-15633, the calculations used a value for the CROSSFLOW uncertainty of 1.0%, instead of the calculated value of 0.5%, as a conservative allocation of margin for the power calorimetric uncertainty. Plant data and Westinghouse calculations have demonstrated the CROSSFLOW accuracy to 0.5%, or better, as stated. However, based on engineering judgement for margin allocation, 1.0% was used in the uncertainty calculation.

2. **Section 3.3 in Attachment 6 states that with an inoperable CROSSFLOW UFM, plant operation at a core thermal power level of 3853 MWt may continue for 24 hours. If the CROSSFLOW UFM were not restored in 24 hours, plant procedures would require reactor power to be reduced to a level less than or equal to 3838 MWt. During this 24 hours, the power measurement uncertainty is 1.0 percent, as calculated in WCAP-15697. What is the impact of a sustained overpower event during the 24 hour period on core damage frequency and large early release frequency when secondary**

side power calorimetric measurement uncertainty is 1.0 percent, instead of 0.6 percent?

Response:

During the 24-hour UFM allowed outage time, actual reactor power would remain steady at a nominal 3853 MWt or be allowed to slowly decrease to 3838 MWt. There would be no change to initiating event frequency, or new initiating events caused by the loss of UFM. If an over-power event occurred during the time the UFM was unavailable, there would be no change to plant system or operator response to the transient. Reactor trip instrumentation and alternate reactor power indications remain calibrated and available to ensure power limits are not exceeded. In addition, there would be no change to plant system response, function, or capability during the UFM unavailability. If a daily power range nuclear instrumentation (NI) calibration comes due during the time that the UFM is unavailable, reactor power would be slowly reduced to 3838 MWt and the NIs adjusted to the lower % power level. The margin to the NI high flux trip setpoint would not decrease. Also, the small power decrease does not significantly challenge plant control systems nor require extensive operator action, and therefore does not significantly change the plant transient initiating event frequency. Adjusting the NIs to calorimetric is a periodic surveillance, and the likelihood of a reactor trip due to NI failure during calibration does not change as a result of the UFM outage. In summary, there is no impact to core damage or large early release frequencies due to loss of UFM

4. **Provide description of the programs and procedures that will control calibration of the CROSSFLOW system and the pressure and temperature instrumentation whose measurement uncertainties affect the power calorimetric uncertainties determined in the Westinghouse calculations WCAP-15633, Revision 0 and WCAP-15697, Revision 0. In this description, please include the procedures for:**

   a. **Maintaining calibration,**
   b. **Controlling software and hardware configuration,**
   c. **Performing corrective actions,**
   d. **Reporting deficiencies to the manufacturer, and**
   e. **Receiving and addressing manufacturer deficiency reports.**

Response:

a. The Installed Plant Instrumentation Calibration and Verification Program, Plant Surveillance Program, and Preventive Maintenance Program delineate controls for calibration and verification of permanent plant instrumentation, including the instrumentation whose measurement uncertainties affect the power calorimetric uncertainties determined in

WCAP-15633, Revision 0 and WCAP-15697, Revision 0. Calibrations are performed using Surveillance Calibration and Maintenance Calibration procedures. The Plant Surveillance Test Program controls tests, inspections, and analysis required to satisfy surveillance requirements. The Plant Preventative Maintenance Program schedules calibration frequency and identifies procedures for calibration of instrumentation other than surveillance tests. The Installed Plant Instrumentation Calibration and Verification Program lists and controls those instruments that are not a part of the Plant Surveillance Test Program but are used during the performance of a surveillance to obtain data or are Post Accident Monitoring instruments.

b. The STP Software Quality Assurance Program controls the appropriate level of validation, verification and documentation applied to the software. This program complies with the station's Operations Quality Assurance Program. The STP Configuration Management Program establishes responsibilities and requirements for the process of ensuring that permanent plant systems, structures, and components conform to the approved design and that their physical and functional characteristics are correctly reflected in controlled design, maintenance, and operations technical and procedural documents.

c. STP Condition Reporting Process controls the identification, classification, trending, reporting, and timely correction of situations that require further review, evaluation or resolution.

d. STP Condition Reporting Process addresses equipment deficiencies. The investigation of a deficiency would identify the need to report to manufacturers.

e. The STP Vendor Document Control Program establishes the administrative requirements for receipt, review, approval, and processing of vendor documents including vendor technical bulletins/advisories and 10CFR21 Notices. This program addresses manufacturer deficiency reports.

<u>**Reactor Systems:**</u>

4. Attachment 6, Section 6.2.2, Steam Dump System - Condenser Steam Dump Valves, states that the condenser steam dump capacity continues to meet the sizing criterion for the uprated plant conditions, however, sufficient bases were not provided to support that conclusion. Please state the total uprated

steam dump capacity, in percentage of the maximum steam flow and in lb/hr, at the uprated conditions.

Response:

The comment requested that steam dump capacity be provided in percentage of maximum steam flow and in lb/hr at the uprated conditions. The criterion requires that the steam dump system be capable of discharging 40 percent of the rated steam flow at full load steam pressure. With respect to criteria compliance, the lowest analyzed full power operating pressure and associated highest steam flow govern. At an uprated power of 1.4%, these conditions correspond to a full power operating steam pressure of 957 psia and a total steam flow of 17.11 x 10^6 lb/hr. Based on these full power operating conditions, the evaluation determined that the steam dump system would discharge steam at a rate of 7.74 x 10^6 lb/hr, or 45.2%, of rated full power steam flow. Since this exceeds the 40% criteria, the steam dump system is adequately sized for the uprating.

7. **Attachment 6, Section 7.2, Vessel Integrity - Neutron Irradiation, states that the 32 effective full power year fluence was reevaluated to account for the power uprate in the revised pressure temperature curves and the calculation of the RT_{PTS}. Did the methodology used in this reevaluation comply with the guidance in Regulatory Guide (RG) 1.190, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I"? Provide a reference to the approved methodology used.**

Response:

It is confirmed that the calculated fluences used in the re-evaluation complied with Draft Reg. Guide DG-1053, which in turn was approved and became RG 1.190. As these calculations are performed on a plant-by-plant basis, there is no generic topical for approved method – the methodology used is that of RG 1.190.

9. **It appears that the South Texas reactor cores will consist of 17x17 XL robust fuel assembly (RFA) and 17x17 XL V5H fuel after the power uprate. Please provide the licensing bases and justifications that the fuel will conform to all the applicable regulations so that the (1) fuel system is not damaged as a result of normal operation and anticipated operational occurrences, (2) fuel system damage is never so severe as to prevent control rod insertion when it is required, (3) number of fuel rod failures is not underestimated for postulated accidents, and (4) core coolability is always maintained. Also, please describe the licensing rationale that was used to justify the transition from V5H to RFA fuel.**

Response:

The following discussion describes the licensing rationale that was used to justify the transition from V5H to RFA fuel.

South Texas loaded RFA fuel beginning with Unit 2 Cycle 7 in October 1998 and with Unit 1 Cycle 9 in April 1999. This fuel design change was the long-term resolution to address the root cause for Incomplete Rod Insertion (IRI) experienced in Unit 1 during an event on December 18, 1995. Following this event, a first set of fuel assembly design changes called "Phase I" changes was implemented for Unit 1 Cycle 8 reload fuel in September 1997 as an interim measure.

By a letter dated August 19, 1997, the NRC issued Amendment Nos. 89 and 76 to the Operating Licenses to allow use of an alternate zirconium-based fuel cladding, ZIRLO™, and limited substitution of fuel rods by ZIRLO™ filler rods. By a letter dated March 11, 1998, the NRC requested information regarding the potential for atypical fuel rod bowing to occur with the fuel approved by Amendment Nos. 89 and 76. By a letter dated February 4, 1999, STP provided a response that concluded fuel rod bow performance was acceptable for the VANTAGE + (V+) fuel with PERFORMANCE + (P+) features (includes the fuel approved by Amendment Nos. 89 and 76) and the Phase 2 ZIRLO™-clad fuel (i.e., RFA design). In addition, South Texas met with the NRC staff on January 26, 1999, to discuss fuel product summary, IRI experience summary, post-irradiation examination results, mechanical analysis of span/assembly bow, fuel analysis of the previous and the improved fuel designs, IRI susceptibility thresholds, an independent contractor's review of Westinghouse's IRI analyses, and STP's rod drop testing plan for the RFA design. The RFA design includes the V+/P+ fuel and the RFA design. By a letter dated February 25, 2000, the NRC staff concluded that fuel rod bowing should not be increasing by the use of the phase 1 or phase 2 (RFA) fuel introduced into the STP units.

By a letter dated November 17, 1998, pursuant to 10CFR50.46, South Texas reported the changes in ECCS performance criteria as a result of consolidating the large break loss of coolant accident (LBLOCA) analyses and to support use of the RFA fuel assemblies. The revised LBLOCA analysis peak clad temperature remained below the regulatory limit.

Westinghouse, STP's fuel vendor, evaluated each fuel change leading up to the current generation RFA fuel design under the NRC-approved Fuel Criteria Evaluation Process (FCEP). Following is a brief synopsis of FCEP notification letters to the NRC that starts with the development of the MV5H/MIFM grid designs and progresses into the RFA design that utilizes the MV5H/MIFM grid designs with the thicker guide thimble tube wall.

- NSD-NRC-96-4694, 4/22/96, "Transmittal of Presentation Material from NRC/Westinghouse Fuel Design Change Meeting on April 15, 1996" (Note 1)

  This notification establishes the applicability of WRB-2 to the newly developed Modified V5H (MV5H) Low Pressure Drop (LPD) Mid-grid with the use of Modified Intermediate Flow Mixers (MIFM). The presentation material also provides details of the newly developed MV5H/MIFM design.

- NSD-NRC-97-5189, 6/24/97, "Transmittal of Response to NRC Request for Information on Wolf Creek Fuel Design Modifications"

  This notification justifies the applicability of WRB-2 to the RFA design for Wolf Creek. The RFA design incorporates the use of MV5H/MIFM grids and a thicker guide thimble tube.

- NSD-NRC-98-5618, 3/25/98, "Notification of FCEP Application for WRB-1 and WRB-2 Applicability to the 17x17 Modified LPD Grid Design for Robust Fuel Assembly Application" (Note 1)

  This notification justifies the applicability of both WRB-1 and WRB-2 to the RFA design that incorporates the MV5H mid-grid. This justification is applicable to both 12 foot and 14 foot core designs.

- NSD-NRC-98-5722, 6/23/98, "Fuel Criteria Evaluation Process Notification for the Revised Guide Thimble Dashpot Design for the 17x17 XL Robust Fuel Assembly Design" (Note 1)

  This notification discusses the change in the guide thimble dashpot design from a double dashpot to a single dashpot. This is one of the changes associated with the 14-foot (XL design) RFA design to reduce the potential for IRI.

- NSD-NRC-98-5796, 10/13/98, "Fuel Criteria Evaluation Process Notification for the 17x17 Robust Fuel Assembly with IFM Grid Design"

  This notification justifies the applicability of WRB-2 to the RFA design generically for all MV5H plants. The Robust design incorporates the use of MV5H/MIFM grids and a thicker guide thimble tube.

Note (1) - denotes those FCEP notifications that were directly applicable to the South Texas 17x17XL RFA design.

The first set of RFA design changes, called "Phase I" changes, were implemented for the Unit 1 Cycle 8 reload fuel in September 1997 as an interim measure to mitigate IRI until a long-term fix could be implemented. These changes were implemented by the STP pursuant to 10CFR50.59.

The Phase I change consisted of:

- A Protective Bottom Grid (P- Grid) was added to form a double grid with the Standard Bottom Grid
- The Standard Bottom Grid was moved upward (inserts were lengthened) to accommodate the P- Grid
- The fuel rods were re-positioned to be in contact with the bottom nozzle within a few thousand MWD/MTU cycle exposure (e.g., a longer bottom end-plug was used – fuel stack elevation remained unchanged).
- Long fuel rod bottom end plugs were adopted in place of the standard short end plugs.
- The internal fuel rod plenum spring was changed to a non-linear, shorter spring (e.g., a variable-pitch plenum spring).
- The thimble screws were changed to high-strength steel
- The Mid-Grids, Guide Tubes and the Instrument Tubes were changed to the ZIRLO™ alloy
- The Fuel Rod Cladding was changed to ZIRLO™ alloy (subject of approved Amendment Nos. 89 & 76)

The 10CFR50.59 evaluation concluded that the individual mechanical and material changes for the Phase I fuel assembly will have no impact on the outcome of the UFSAR Chapter 15 analysis of record. The Phase I fuel assembly thermal-hydraulic performance will not be significantly different from previous fuel designs. Therefore, the structures, systems and components upstream and downstream of the core will not be adversely impacted. The structural integrity of the Phase I fuel assemblies and their fuel rods will not be reduced by the individual mechanical and material changes. Therefore, there will be no reduction in the assemblies' ability to perform during normal operating conditions or during a UFSAR Chapter 15 event. The Phase I fuel assembly mechanical and material changes were expected to have the beneficial effects of reducing dimensional distortion to re-mediate IRI and of reducing assembly corrosion effects as well as avoiding flow-induced vibration and fretting.

As part of each cycle's reload safety evaluation process, the overall core design and fuel configuration is verified to meet all applicable design criteria and ensure that all pertinent licensing basis acceptance criteria are met. The Phase I fuel assembly design did not impose any new performance requirements. No new modes or new limiting single failures were created by the mechanical changes noted above. Adherence to the licensing basis design criteria ensures that the fuel assembly fission product barrier capability is maintained, and that a coolable core geometry and sub-criticality will be maintained during the postulated accidents of Chapter 15 of the UFSAR. It was determined that the STP V5H reload design and safety analysis limits remain applicable for the mixed core of XLR, V5H, and Phase I fuel assemblies. In summary, the 10CFR50.59 evaluation concluded that the fuel design change did not meet any of the rule's criteria for requesting NRC approval of the change.

STP loaded RFA fuel beginning with Unit 2 Cycle 7 in October 1998 and with Unit 1 Cycle 9 in April 1999. This fuel design change was the long-term resolution to address the root cause for IRI experienced in Unit 1 during an event on December 18, 1995. These changes were also implemented by the STP pursuant to 10CFR50.59.

The RFA fuel loaded beginning with the Unit 2 Cycle 7 in October 1998 was the Phase II fuel. The most significant changes introduced by this RFA design were:

- The guide thimble and instrumentation tube outside diameter was increased
- The dashpot section of the guide tube was redesigned to provide additional assembly strength
- The mid-grids were modified to improve DNB performance

The fuel vendor performed several tests to demonstrate that the RFA design met established criteria, and to ensure safe and reliable fuel performance. The test results validated that design criteria were met and that the design change would not result in a failure of a different kind than previously analyzed or an increase in the probability of a failure. No abnormal or detrimental fuel rod vibration behavior was observed. The test results concluded that the analytical models used with the existing fuel to predict scram time were applicable to the RFA design. The design changes associated with the larger thimble and instrument tubes were found to have a negligible effect on the hydraulic characteristics of the 17XL RFA as compared to similar tests on existing fuel. Therefore, the RFA fuel was tested for thermal-hydraulic compatability with previous fuel designs and determined that no appreciable mixed core effects would be introduced.

The RFA design only changes certain structural elements of the fuel assemblies. It does not impact the nuclear characteristics of the reactor core or transient response of the reactor coolant system to accidents described in UFSAR Sections 6.2 and 15. With respect to post-LOCA long-term core cooling analysis, the only parameters of interest that changed were the thickness of the thimble and instrument tubes and the resulting changes to reactor coolant system volume. Conservatively applying the diameter increase over the length between the core plates results in a very small reduction in volume, less than 0.5%. The reduction in volume is marginally conservative for this calculation, but the magnitude of the change results in a negligible effect. Therefore, the existing analysis remained bounding.

With respect to the non-LOCA safety analysis,

1. The slight change to the pressure differential across the upper head spray nozzles had no impact on the safety analysis, since the upper head fluid conditions are explicitly defined to yield the most conservative results for those events that are sensitive to the upper head conditions.
2. The results of a DNB evaluation showed that the RFA fuel was less limiting than the V5H fuel.
3. Data demonstrated that the rod control cluster assembly (RCCA) insertion time for the RFA fuel was more rapid than that used in the existing licensing basis.

In conclusion, adherence to the design criteria of the RFA fuel in combination with testing demonstrated that the RFA fuel is not more prone to fuel failure and does not introduce any new failure modes. No new modes or new limiting single failures were created with the mechanical changes made. The adherence to existing standards and design criteria precludes new challenges to components or systems that could introduce a new type of accident. The RFA fuel conforms to the requirements of Section 5.3.1 of the Technical Specifications. Cycle-specific parameters that are a function of the fuel enrichment and location in the reactor core are evaluated as part of the reload safety analysis process. In summary, the 10CFR50.59 evaluation concluded that the fuel design change did not meet any of the rule's criteria for requesting NRC approval of the change.

The results of the two 10CFR50.59 evaluations discussed above were reported to the NRC in the South Texas' 10CFR50.59 Summary Reports of November 19, 1998 (USQE # 97-0029) and January 15, 2001 (USQE # 98-12879-2).

The summary above provides the bases for concluding that the RFA fuel will meet performance requirements during normal operation and anticipated operational occurrences. The RFA fuel design was specifically designed to ensure control rod insertion when it is required. The UFSAR Chapter 6 and 15 accident analyses remain applicable to the RFA fuel design and core coolability is maintained with this type of fuel loading.

Section 7.10 of Attachment 6 to the August 22, 2001 application summarizes the 1.4-percent uprate fuel evaluation that is applicable to the standard, V5H and RFA fuel types. Future reload designs will be evaluated to confirm that the loading patterns and associated fuel types meet all design and performance criteria.

16. **Attachment 6, Section 7.3.1 states that the revised design conditions (primarily T_{cold}) will have a negligible effect on the rod control cluster assembly (RCCA) drop time, and the time will still be less than the current value of 2.8 seconds required by the Technical Specifications. However, sufficient technical bases were not provided to support this conclusion. Provide the technical bases that support these conclusions.**

Response:

The revised design conditions for the RCCA drop time consist of the core power and the core inlet temperature (T_{cold}). The core power increased by 1.4% from 3800 MWt to 3853 MWt. The lowest core inlet temperature remained unchanged at 549.8°F for the uprate conditions. The percent change in RCCA drop time for the 1.4-percent uprating at normal operating conditions was calculated to be 0.1-percent. This change is considered to have a negligible effect.

17. **Attachment 6, Section 7.3.1 states that the resulting bypass flow velocities exiting or entering the pressure relief holes on the baffle plates did not significantly change and still meet the fuel interface requirements for the 1.4-percent uprate conditions. Provide the technical bases that support these conclusions.**

Response:

The percent change in the bypass flow velocities exiting or entering the pressure relief holes on the baffle plates due to the 1.4-percent power uprating was insignificant (calculated to be -0.6%). Based on this insignificant change, the bypass flow velocities exiting of entering the pressure relief holes on the baffle plant will still meet the fuel interface requirements for the 1.4-percent power uprate conditions.

19. **The NRC concluded that you have not provided enough information for us to agree with your conclusion that South Texas continues to comply with the ATWS Rule. In sensitivity studies performed by Westinghouse as the basis for the ATWS rule, an increase in rated power for units similar to South Texas can result in significantly higher peak RCS pressures. This condition, coupled with a positive moderator temperature coefficient operation may result in the South Texas units exceeding the American Society of Mechanical Engineers (ASME) Stress Level C Limit of 3200 psig during an ATWS event. This limit is the basis for the ATWS rule. Provide a plant-specific analysis demonstrating the ability of the South Texas units to meet the basis for the ATWS Rule, 10 CFR 50.62. In the analysis, state the assumed initial conditions for power level, moderator temperature coefficient**

and plant operating conditions for each of the units. Show that the units will not exceed the ASME Stress Level C Limit of 3200 psig. In addition, include a discussion and applicable values of the unfavorable exposure time, if any, and ATWS Core Damage Frequency for the South Texas units as a result of the proposed power uprate.

Response:

The current AMSAC design for South Texas with the Model Δ94 steam generators installed is based on the Logic 1 generic AMSAC design for Westinghouse pressurized water reactors (PWRs) as described in WCAP-10858P-A, Revision 1. The NRC concluded in their Safety Evaluation Report, NUREG-0781, Supplement 6, that the South Texas AMSAC design is acceptable and in compliance with 10CFR50.62. The AMSAC design is not effected by the 1.4-percent power uprate. South Texas Units 1 and 2 will maintain and operate AMSAC consistent with the AMSAC design as specified in WCAP-10858-A, Revision 1 for the 1.4-percent power uprate.

The generic Westinghouse analysis which is considered the basis for the ATWS Mitigation System is provided in Westinghouse Letter NS-TMA-2182 "ATWS Submittal", December 1979. At South Texas, operator response to an ATWS event can be enhanced by the capability to trip the control rod drive motor-generator sets from the control room. This improves response time and maximizes the probability of success of this step, if required, as opposed to the need for an operator to transit to the station to locally trip these power supplies.

The Core Damage Frequency (CDF) for an ATWS-type event at the South Texas units is 8.3E-07 per year. This CDF value is not impacted by the power uprate. This low risk value reflects the high availability and reliability of the reactor protection system to perform its design function when required.

The 1.4-percent power uprate is being requested because improved instrumentation accuracy reduced the uncertainty in determining rated thermal power. If uncertainties are taken into account for the current licensed power level versus the requested revised licensed power level, the core thermal output remains unchanged in accident analyses if the worst-case condition is assumed (i.e., 3878 MWt). Regardless, the revised uprated power is small and is not considered a significant change that impacts peak pressure in the extremely low likelihood that an ATWS event would occur. Based on this small increase in power level and the low risk of an ATWS event occurring at South Texas, a plant-specific analysis that is costly and could impact the planned implementation of this change is not considered justified.

The Westinghouse Owner's Group has an industry initiative to address concerns regarding challenges to the ASME Stress Level C Limit for NSSS components and potentially increased unfavorable exposure times during an ATWS event. STP plans to follow this initiative to ascertain if future plant-specific actions are prudent.

In summary, South Texas is not planning to perform a plant-specific analysis because:

(1) South Texas will remain in compliance with the ATWS Rule, 10CFR50.62, following implementation of the 1.4-percent power uprate,
(2) The requested power increase is small and is the result of improved instrumentation with less uncertainty , and
(3) There is low risk for occurrence of an ATWS event at South Texas.

If South Texas decides to modify the plant in the future to achieve a higher rated power level, then a plant-specific analysis for an ATWS event will be considered.

23. You reference CENPD-397-P-A Revision 01 in your amendment request. This topical report, including the NRC staff's safety evaluation, contains criteria that shall be addressed by licensees referencing the topical report. Please list each criterion and state how each is satisfied. (Where appropriate, you may reference specific sections of your application. Also, provide the technical bases that support the use of a Combustion Engineering methodology at a Westinghouse plant.)

Response:

Criterion 1: The licensee should discuss the development of the maintenance and calibration procedures that will be implemented with the Crossflow UFM installation. These procedures should include process and contingencies for an inoperable Crossflow UFM and the effect on thermal power measurement and plant operation.

Addressed in Sections 3.3 and 3.4 of Attachment 6 to Reference 1.

Criterion 2: For plants that currently have the Crossflow UFM installed, the licensee should provide an evaluation of the operational and maintenance history of the installed UFM and confirm that the instrumentation is representative of the Crossflow UFM and is bounded by the requirements set forth in Topical Report CENPD-397-P.

Addressed in Section 3.5 of Attachment 6 to Reference 1.

Criterion 3: The licensee should confirm that the methodology used to calculate the uncertainty of the Crossflow UFM in comparison to the current feedwater flow instrumentation is based on accepted plant setpoint methodology (with regard to the development of instrument uncertainty). If an alternative methodology is used, the application should be justified and applied to both the venturi and the Crossflow UFM for comparison.

Addressed in Section 3.6 of Attachment 6 to Reference 1.

Criterion 4: The licensee of a plant at which the installed Crossflow UFM was not calibrated to a site-specific piping configuration (flow profile and meter factors not representative of the plant-specific installation), should submit additional justification. This justification should show that the meter installation is either independent of the plant-specific flow profile for the stated accuracy, or that the installation can be shown to be equivalent to known calibration and plant configurations for the specific installation, including the propagation of flow profile effects at higher Reynolds numbers. Additionally, for previously installed and calibrated Crossflow UFM, the licensee should confirm that the plant-specific installation follows the guidelines in the Crossflow UFM topical report.

Addressed in Section 3.7 of Attachment 6 to Reference 1.

The generic CROSSFLOW topical, CENPD-397-P-A, was submitted to, and approved by, the NRC prior to the nuclear division of Combustion Engineering becoming part of Westinghouse Electric, LLC – thus explaining the document numbering convention. There is nothing in this topical that is plant- or vendor-specific; the generic topical is strictly based on flow phenomena. The principal NRC reviewer for this Topical Report was Iqbal Ahmed.

24. Attachment 6, Section 3.3 states that when the CROSSFLOW equipment is out-of-service "... power level is based upon the methodology and instrumentation configuration assumed in WCAP-15697...." WCAP-15697 also uses "assume" and provides values, including a power calorimetric uncertainty. Please:

A. Confirm that your methodology is actually what is provided in WCAP-15697 or identify any differences.

Response:

The word "assumed" as used in the context of this sentence is the same as "defined". This should be considered an editorial preference by the authors. It is confirmed that the methodology is as provided in WCAP-15697.

B. Where "assume" is used, please justify this usage or provide plant-specific information.

Response:

Upon review of the WCAP, the word "assume" or variations of assume should be considered an editorial preference of the authors of the WCAP. The phrase "is based on" could have been substituted for "assume". It is common phraseology to use "assume" as defining the basis for how the calculations were performed. It was not intended for the statements where these words are used to be interpreted as unverified assumptions or open items. For example the first sentence under the Reactor Power Measurement title on page 6 uses the word "assumes", but the calculation as shown on pages 10, 11, and 12 actually do use the feedwater flow ΔP transmitter errors. In part, the WCAP is written in a manner so the licensee understands the methods and values behind the calculation, and is satisfied that there are no misunderstandings relative to what is "assumed" or "the basis" for the calculations.

C. For each item contributing to the power calorimetric uncertainty determination, provide a comparison of the uncertainty-of-record used in your licensing basis and the values provided in WCAP-15697 and justify any differences.

Response:

A comparison of differences is not practical in this case, since WCAP – 15697 is essentially a complete re-work of the power calorimetric uncertainty based on the currently installed instrumentation. New feedwater temperature and feedwater flow instruments have been installed since WCAP-13441 Rev. 0 was prepared. New steam pressure instruments will be installed prior to implementation of the 1.4-percent power uprate. Also, the previous WCAP-13441 Rev. 0 did not include the affects of steam generator blowdown flow because the contribution of these uncertainties to the total power calorimetric was so small that it did not affect the final power calorimetric uncertainties relative to the 2.0% used in the safety analysis limit. However, with the reduction in uncertainties to increase plant power, it is appropriate to include the affects of steam generator blowdown flow. WCAP-15697 also reflects an update to the net pump heat addition calculations, and improved moisture carryover due to new steam generators.

The only portion of the calculation that remains the same as previous calculations is the uncertainties associated with the venturis. These inputs are the same as previous calculations, because this part of the plant configuration remains unchanged.

The uncertainties as documented in WCAP-15697 are based on a review of plant procedures and practices and manufacturer's specifications.

South Texas will be submitting WCAP 13441 to the NRC as requested by Reference 2. Revision 0 to WCAP 13441 pertains to the power calorimetric uncertainty at South Texas. Revision 1 to WCAP 13441 has other sections that have been applied at STP for instrumentation that does not pertain to the power calorimetric. Revision 1 is not used for the power calorimetric since not all of the instrumentation assumed in this revision for the power calorimetric has been upgraded yet.

D. **WCAP-15697 states "... no allowance is made for venturi fouling." We found no other reference to venturi fouling in your request. Please describe how you address this topic.**

Response:

Venturi fouling causes calculated calorimetric power to indicate higher than actual. It is always conservative when higher than actual calorimetric power is used to monitor licensed power level or adjusts excore power indications. Therefore, making no allowance for venturi fouling is conservative. Using no allowance for venturi fouling is consistent with the currently approved Revised Thermal Design Procedure (RTDP) setpoint methodology (WCAP-13441). WCAP-13441 will be submitted by separate correspondence. The methodology in WCAP-15697 is consistent with that of WCAP-13441.

E. **The WCAP-15697 power calorimetric uncertainty appears to be inconsistent with the 3838MWt you state will be the maximum allowable power if the CROSSFLOW system is unavailable and is not restored within 24 hours. Please explain.**

Response:

The power uncertainty determined by WCAP-15697 is ±1% of rated thermal power. The ECCS evaluation assumes that the reactor operates at a margin of 2% above rated thermal power. WCAP-15697 uncertainty does not take credit for reduced uncertainties associated with the CROSSFLOW system. Based on a core power level power measurement uncertainty of 1%, a power uprate of 1% is achievable using current NRC-approved methodologies. A power uprate of 1% over the current core power limit of 3800 MWt is 3838 MWt. Therefore, the maximum allowable core power of 3838 MWt with the CROSSFLOW system unavailable is consistent with WCAP-15697.

F. **WCAP-15697 states "Reactor power is monitored by the performance of a secondary side heat balance (power calorimetric) at least once every 24 hours." In Section 3.2, you state a comparison is made periodically. If the NRC staff assumes periodically means every 24 hours, and considers the Section 3.3 statement referenced in Item (e), immediately above, approximately 48 hours may pass between a calibration of thermal power and a power reduction. Please address this observation and justify the actual times.**

Response:

South Texas Project Technical Specification surveillances require power range nuclear instrumentation to be compared to calorimetric power at least once per 24 hours (plus 6 hours of grace). Therefore, the maximum time between required calorimetric power determinations is 30 hours. If the CROSSFLOW system becomes unavailable, then plant procedures would require that reactor power be reduced to less than 3838 MWt before the next required Technical Specification calorimetric. If the CROSSFLOW system becomes unavailable and the next Technical Specification required calorimetric is due in 2 hours, then reactor power would be reduced to less than 3838 MWt within 2 hours. If the CROSSFLOW system becomes unavailable and the next Technical Specification required calorimetric is due in 30 hours (24 hour surveillance requirement plus 6 hours grace), then reactor power would be reduced to less than 3838 MWt within 30 hours.

25. **WCAP-15697 references Prairie Island data and other plant reactor coolant pump data regarding pump thermal energy generation. Please provide references for the plant data other than Prairie Island. The data we have seen appear to indicate that the reactor coolant pump thermal energy changes early in plant life due to impeller smoothing. Explain how this effect is taken into account in determining the pump thermal energy contribution for the power uprate?**

Response:

The statement referred to in WCAP-15697, "Reactor coolant pump hydraulics are known to a relatively high confidence level, supported by system hydraulics tests performed at Prairie Island Unit 2 and by input power measurements from several other plants." refers to reactor coolant pump input power as well as hydraulics measurements. It means that the measured pump input power was found to be relatively close to the predicted curve generated by the pump vendor. With respect to impeller smoothing, Westinghouse has considered the measured decrease in pump input power at Prairie Island and concluded that the magnitude of the decrease is well within the uncertainties allowed for the pump input power (5%) and total net heat input power (20%). Since impeller smoothing has a negligible impact on plant thermal output, Westinghouse has not adjusted or corrected the thermal energy contribution of the reactor coolant pumps to account for impeller smoothing.

Structural Questions:

1. **The licensee evaluated the impact of the power uprate on the minimum wall thickness of the steam generator tubes. However, the licensee did not provide the value of the minimum wall thickness. In addition, the licensee did not relate the minimum wall thickness to the tube repair limits in the South Texas Units 1 and 2 technical specifications. The licensee needs to provide the value of the minimum wall thickness and confirm that the minimum wall thickness is within the tube repair limits in the South Texas technical specifications.**

   Response:

   A Regulatory Guide 1.121 analysis applicable to the South Texas Units 1 and 2 Delta 94 steam generators is documented in WCAP-15095, Revision 1. The analysis calculates the minimum acceptable wall thickness for several different tube locations and two different tube plugging levels, 0% and 10%. Of the several sets of conditions and locations considered, the bounding requirement for tube minimum wall thickness is 0.015 inch, which translates to a tube structural limit of 62.5%. The minimum tube wall thickness is a conservatively calculated (uniform thinning model assumed) number that represents an acceptable wall thickness that meets all specified Code criteria using minimum Code material properties. As such, this limit represents a true, conservative structural limit.

   The tube repair limit is derived from the structural limit by subtracting allowances for measurement uncertainty. The Technical Specification requires that indications equal to or greater than 40% through-wall depth must be plugged or repaired. Thus, there is a margin of 22.5% wall loss, assumed to be a uniform wall loss over an infinite length, between the conservative structural limit and the specified repair limit.

   The Degradation Assessment provided prior to each inspection, and the Condition Monitoring and Operational Assessment required during and after each inspection, provide degradation-specific measurement uncertainties and repair limits. Typically only volumetric tube degradation can be adequately sized for volumetric types of degradation, the measurement uncertainties are less than about 10% at a 95% confidence limit (that is, a measured depth of 40% may be 50% at the 95% confidence limit). Therefore, there is significant margin between the specified repair limit and the conservative structural limit.

For degradation mechanisms that cannot be adequately sized, either a "plug-on detection" approach is used or alternate repair criteria may be licensed for plant-specific use that justifies degradation specific repair limits.

2. **NRC has issued the following generic communications regarding SG tube plugs: NRC Information Notice 89-65, "Potential for Stress Corrosion Cracking in Steam Generator Tube Plugs Supplied by Babcock and Wilcox;" NRC Information Notice 89-33, "Potential Failure of Westinghouse Steam Generator Tube Mechanical Plugs;" NRC Bulletin No. 89-01, "Failure of Westinghouse Steam Generator Tube Mechanical Plugs," and Supplements 1 and 2; NRC Information Notice 94-87, "Unanticipated Crack in A Particular Heat of Alloy 600 Used for Westinghouse Mechanical Plugs for Steam generator Tubes".**

   The licensee discussed evaluation of mechanical tube plugs under the power uprated conditions.

   (a) **Clarify if tube plugs have been used in the South Texas Unit 1 replacement SGs,**

   Response:

   Westinghouse Alloy 690 thermally treated tapered welded plugs were installed during fabrication of the STP Unit 1 Delta 94 replacement steam generators.

   (b) **Discuss if any of the above NRC generic communications are applicable to the tube plugs used in the South Texas replacement SGs and the steps that have been taken to meet the NRC staff's recommendations in the above generic communications,**

   Response:

   The above generic communications are not applicable to the tube plugs used in the South Texas replacement steam generators as discussed below.

NRC Information Notice 89-65: This information notice applies to steam generator tube plugs of the "rolled," "ribbed," and "taper welded" designs fabricated from alloy 600 mill annealed material and supplied by Babcock & Wilcox. This Notice addressed the occurrence of PWSCC cracking which was the result of the plug alloy 600 mill annealed material and it's low mill annealing temperature. This Notice does not apply to the STP Unit 1 alloy 690 thermally treated welded plugs. This plug material has higher chromium content and higher annealing temperature that results in improved carbide precipitation on the grain boundaries that preclude the occurrence of PWSCC.

NRC Information Notice 89-33: The Information Notice addressed alloy 600 mill annealed mechanical plugs with PWSCC cracks associated with minimal intergranular carbide precipitation, which was a result of the material selection and a low mill annealing temperature. This Notice does not apply to the STP Unit 1 installed alloy 690 thermally treated welded plugs, which have a higher chromium content and higher annealing temperature that results in improved carbide precipitation on the grain boundaries to preclude the occurrence of PWSCC.

NRC Bulletin No. 89-01: This Bulletin discussed the susceptibility to PWSCC of Westinghouse mechanical plugs made from various specific heats of alloy 600 mill annealed material. It also discusses the algorithm for determining when a plug of a specific heat should be preventively removed and replaced with a new plug. This Bulletin does not apply to the STP Unit 1 installed alloy 690 thermally treated welded plugs which have a higher chromium content and higher annealing temperature which results in improved carbide precipitation on the grain boundaries that preclude the occurrence of PWSCC.

NRC Information Notice 94-87: This Notice addresses operating experience affecting the predicted service life of mechanical tube plugs fabricated from alloy 600 mill annealed material supplied by Westinghouse. This Notice does not apply to the STP Unit 1 installed alloy 690 thermally treated welded plugs which have a higher chromium content and higher annealing temperature that results in improved carbide precipitation on the grain boundaries to preclude the occurrence of PWSCC.

Unit 2 steam generator applicability**:** As per definition 1.27 of STP's Technical Specifications in Reference 1, the uprate condition only applies to Delta 94 replacement steam generators. The STP Unit 2 Delta 94 replacement steam generators are under fabrication and to date no tube plugs have been installed. Should the up coming preservice inspection identify the need to plug tubes, only alloy 690 thermally treated material will be used and thus the above Notices and Bulletin do not apply.

(c) **Discuss any degradation detected in tube plugs and the associated repair method other than those discussed in Item (b).**

Response:

The welded solid tapered alloy 690 thermally treated plugs in service in the Unit 1 replacement Delta 94 steam generators were visually inspected during the last refueling outage in accordance with the guidelines of EPRI TR-107569" PWR Steam Generator Examination Guidelines" section 3.2.3, " Examination of Plugs" This 100% visual examination of all hot and cold leg plugs in all four steam generators detected no degradation in tube plugs. The STP Unit 2 Delta 94 replacement steam generators are under fabrication and are planned for installation in late 2002.

3. **Clarify whether the SG tubes under the power uprated conditions satisfy the structural integrity discussed in RG 1.121, "Bases for Plugging Degraded PWR Steam Generator Tubes".**

Response:

The RG 1.121 analysis applicable to the South Texas Delta 94 steam generators is documented in WCAP-15095, Revision 1. The analysis calculates the minimum acceptable wall thickness for several different tube locations and two different tube plugging levels, 0% and 10%. The transient loading conditions analyzed as part of the RG 1.121 analysis bound the conditions applicable to the uprating, and the analysis is therefore applicable to the uprated conditions.

4. **The licensee evaluated structural integrity of the steam generator tubes under the power uprated conditions; however, the NRC staff is not clear if the structural integrity evaluation included evaluating leakage integrity of SG tubes. Confirm the acceptability of the leakage integrity of the SG tubes under the power uprated conditions.**

Response:

The steam line break (SLB) differential pressure is controlled by the pressurizer power-operated relief valve (PORV) setpoint which does not change with the power uprate. Leakage integrity of SG tubing is provided through application of the EPRI PWR Primary to Secondary Leak Guidelines - Revision 2. Table 2-3 of this document indicates that application of the guideline implies that there is a

90.4% probability that the burst pressure of a single indication leaking at 75 gpd (recommended administratively controlled leak rate per the EPRI guideline) will be greater than the SLB pressure differential. At the Technical Specification normal operating condition LCO leak rate of 150 gpd, there is a 79.4% probability that the burst pressure of a single leaking indication will be greater than the SLB pressure differential.

Additionally, the 100% through-wall (TW) outside diameter stress corrosion cracking (ODSCC) flaw length that would provide primary-to-secondary leakage of 150 gpd at normal operating conditions was estimated using a nominal leakage prediction for 11/16" OD x 0.040" tubing, using the lower tolerance limit flow stress, and tortuosity assumptions consistent with recent pulled tube destructive examination results. In this (recent) pulled tube examination, the estimated 100%TW flaw length based on laboratory leak test results for the pulled tube provided an excellent match with the 100% TW flaw length measured by destructive examination. Using the calculated 100% TW ODSCC flaw length that provides a normal operating condition leak rate of 150 gpd, the nominal predicted burst pressure is well above the SLB pressure differential. If partial TW flaw depths are included such that the overall flaw length is twice the 100% TW flaw length, predicted burst pressure using lower tolerance limit flow stress is still greater than the SLB pressure differential. Thus, leakage integrity is expected to be provided for the STP tubing. An ODSCC leakage model was used since expected leak rates for equal 100%TW lengths will be less for ODSCC than for PWSCC. Thus the ODSCC model will predict a greater 100% TW flaw length for leakage at 150 gpd.

Thus, the leakage integrity of the steam generator tubes has been confirmed under the power uprated conditions.

5. **Discuss the impact of the power uprated conditions on (a) the degradation in the steam generator tubes, (b) the intervals of SG tube inspections, and (c) the condition monitoring and operational assessments of the SG tubes.**

Response:

The operating parameters that would affect the corrosion degradation potential of the steam generator tubing are temperature and pressure differential (ΔPp-s) across the tubes. Both the temperature and ΔPp-s at uprated operating conditions are essentially the same as the reference operating conditions. For example, for the zero-plugging condition, the maximum steam temperature at the uprated condition is 1°F higher than the pre-uprate (RSG) design condition. This slightly increased temperature has negligible effect on the corrosion potential of the Alloy 690TT tubing. Similarly, the uprate analyses have shown that there is essentially no change in the ΔPp-s for the uprated condition compared to the reference conditions. Therefore, the uprate operating conditions represent no significant increase in the corrosion degradation potential of the tubes.

The inspection interval for the SG tubing is defined in the Technical Specifications. The operating experience for Alloy 690TT tubing has been flawless, with no corrosion degradation reported. Both the operating experience and laboratory data comparing corrosion potential of Alloy 690 TT to Alloy 600 MA and Alloy 600TT tubing indicates that inspection intervals much longer than specified in the Technical Specifications are technically justified. Since the uprated operating conditions are insignificant relative to the corrosion potential of the Alloy 690TT tubing, and the inspection intervals required by the Technical Specifications are very conservative, the uprated conditions have no negative effect on the currently required inspection intervals.

The South Texas Project steam generator program complies with the requirements of NEI 97-06, which requires Condition Monitoring and Operational Assessments (CMOA) be performed at each inspection of the SGs. The condition monitoring assessment considers the actual condition of the SG tubing at the current inspection and compares it to the structural criteria on a degradation specific basis. The operational assessment considers the current condition of the SG tubes, and projects forward to the next planned inspection to conservatively show that degradation specific structural criteria will be met at the next inspection for the planned operating conditions. Since no significant change in corrosion potential has been identified (noting that the corrosion potential of Alloy 690TT is very low), and since the inspection intervals identified in the technical specification is very conservative, the uprate conditions will have no impact on the CMOA.

6. In your submittal you have indicated that the models in the CHECWORKS predictive code were revised to incorporate flow and process system conditions that are determined for 1.4% power uprate conditions. What was the predicted change of the wear rates calculated by the revised code for the components most susceptible to flow-accelerated corrosion?

Response:

The table below consists of heat balance system/lines in descending order of Flow Accelerated Corrosion (FAC) system susceptibility.

SYSTEM	DESCRIPTION	% CHANGE IN PREDICTED WEAR RATE	CHANGE IN PREDICTED WEAR RATE mils/yr
ES	High Pressure Extraction Steam to Deaerator	+ 0.015%	+ 0.68
ES	High Pressure Extraction Steam to High Pressure Feedwater Heater 11	- 0.004%	- 0.13
ES	Extraction Steam to Feedwater Heater 14	+ 0.102%	+ 0.29
FW	Feedwater from FW Pump to High Pressure Feedwater Heater 11	+ 0.003%	+ 0.06
FW	Feedwater from Deaerator to FW Pump	+ 0.002%	+ 0.05
FW	High Pressure Feedwater Heater 11 to SG	+ 0.027%	+ 0.45
CD	Condensate from Feedwater Heater 14 to Feedwater Heater 13	+ 0.011%	+ 0.32
CD	Condensate from Feedwater Heater 15 to Feedwater Heater 14	+ 0.013%	+ 0.36
CD	Condensate from Feedwater Heater 13 to Deaerator	+ 0.003%	+ 0.06
HD	Heaters Drips from Moisture Separator Drip Tank to Condensate System	+ 0.005%	+ 0.03
HD	Heater Drips from Reheater to Reheater Drip Tank	- 0.005%	- 0.08
HD	Heater Drips from Reheater Drip Tank to High Pressure Feedwater Heater 11	- 0.005%	- 0.08
HD	Heater Drips from Moisture Separator to Moisture Separator Drip Tank	+ 0.005%	+ 0.05
HD	Heater Drip from Feedwater Heater 14 to Feedwater Heater 15	+ 0.016%	+ 0.13
HD	Heater Drip from Feedwater Heater 13 to Feedwater Heater 14	+ 0.004%	+ 0.03
HD	Heater Drip from Feedwater Heater 15 to Flash Tanks	+ 0.019%	+ 0.12
HD	Heater Drip from Feedwater Heater 11 to Deaerator	+ 0.042%	+ 0.15
SB	Steam Generator Blowdown Tank Vent	+ 0.001%	+ 0.002
SB	Steam Generator Blowdown Tank Drain	- 0.006%	- 0.01

<u>Accident Analyses</u>

During a conference call on December 19, 2001, the NRC staff pointed out discrepancies in the licensee's statement in Attachment 6, Section 11.2, "Accident Analyses," and the information contained in the updated safety analysis report (SAR). The licensee has stated in Section 11.2 that radiological source terms for all the analyses (except small line failure outside the containment) were determined at power level of 4100 MWt. Contrary to the above, the NRC staff noted, and informed the licensee during the December 19, 2001 phone call, that some accidents reported in the updated SAR were not labeled as evaluated at power level of 4100 MWt. The licensee indicated that the radiological source terms used in the analyses are based on 4100 MWt, and the SAR page(s) have either been updated or will be updated accordingly. Please provide the updated SAR page(s) reflecting the correct power level used to evaluate the radiological consequences resulting from these design-basis accidents and any other additional information concerning use of updated power level of 4100 MWt.

Response:

Updated UFSAR Table 15.6-2, "Parameters Used in Sample Line Failure Radiological Analysis", and Table 15.6-3, "Parameters Used in Steam Generator Tube Rupture Analyses" are included as Attachment 2 to this letter. These tables have been updated to reflect that the radiological analyses were performed at a power level of 4100 MWt. These tables will be included in the next UFSAR revision submitted to the NRC pursuant to 10CFR50.71(e).

Section 11.2 of Attachment 6 of Reference 1 indicated that the small line failure analysis had been performed at 3800 MWt. Further review determined that this analysis was performed at 4100 MWt. The dose results previously reported in Table 15.6-14 of the UFSAR by an update submittal to the NRC were found to be a small fraction of the guideline values of 10CFR100. These previously reported dose results were based on a power level of 4100 MWt.

ATTACHMENT 2

UFSAR TABLES

Table 15.6-2

PARAMETERS USED IN SAMPLE LINE FAILURE RADIOLOGICAL ANALYSIS

CN-2582

Core thermal power for radiological source term, MWt	4,100
Fuel defects prior to accident	1.0%
GWPS operating prior to accident	No
Time assumed for operator to close isolation valves, min	30
Mass of primary coolant release, lbm*	1.6×10^4
Primary coolant concentrations	
pre-existing iodine spike	Table 15.A-4
current iodine spike	Table 15.A-6
Meteorology	5 percentile Table 15.B-1
Dose model	Appendix 15.B
Flashing fraction	0.57

* Evaluated for line of maximum release, the pressurizer sample line.

CN-2582

Table 15.6-3

PARAMETERS USED IN STEAM GENERATOR TUBE RUPTURE RADIOLOGICAL ANALYSES

Core thermal power for radiological source term, MWt	4,100
SG tube leak prior to and during accident	1.0 gal/min*
Offsite power	Lost
Primary coolant concentrations	
preexisting iodine spike spike caused by accident	Table 15.A-4 Appendix 15.A.3
Secondary coolant concentrations	Table 15.A-5
Iodine partition factor in SGs during accident	0.01
Duration of plant cooldown by secondary system after accident, hrs	8
Steam release from defective SG, lb	194,000 (Model E steam generators) 248,200 (Delta 94 steam generators)
Steam release from 3 unaffected SGs, lb	
(0-2 hr)	640,400** (Model E steam generators) 852,600 (Delta 94 steam generators)
(2-8 hrs)	1,051,000 (Model E steam generators) 1,103,300 (Delta 94 steam generators)
Steam release from the four orifices in the above MSIV seat drain lines (0-36 hrs)	1.93 lb/sec/orifice
Reactor coolant released to the defective SG, lb	186,000 (Model E steam generators) 136,100 (Delta 94 steam generators)
Meteorology	5 percentile Table 15.B-1
Dose models	Appendix 15.B

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* For Model E steam generators 1.0 gal/min leakage is assumed to be 0.30 gal/min in defective SG and 0.233 gal/min per intact SG. For Delta 94 steam generators, 1.0 gal/min is split equally between the intact SGs.

** The condenser is assumed to be unavailable for steam dump.