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Robert A. Fenech
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July 28, 1993

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

Gentlemen:

In the Matter of	)	Docket Nos. 50-327
Tennessee Valley Authority	)	50-328

SEQUOYAH NUCLEAR PLANT (SQN) - 10 CFR 50.46 ANNUAL REPORT

References:
1. TVA letter to NRC dated July 21, 1992, "Sequoyah Nuclear Plant (SQN) - 10 CFR 50.46 Annual Report"
2. TVA letter to NRC dated November 16, 1992, "Sequoyah Nuclear Plant (SQN) - Technical Specification (TS) Change 92-14, 'Departure From Nucleate Boiling Ratio (DNBR) Limits Bases Change'"
3. NRC letter to TVA dated December 8, 1992, "Issuance of Bases Change (TAC NOS. M84955 and M84956) (TS 92-14)"

10 CFR 50.46 requires reporting, on at least an annual basis, each change to or error discovered in an acceptable loss-of-coolant accident evaluation model or in the application of such a model that affects the peak clad temperature (PCT) calculation. The purpose of this letter is to provide that report. It should be noted that a significant change in the PCT for the large break loss-of-coolant accident (LOCA) has occurred as a result of the reanalysis performed during this reporting period. The analysis was performed to support increases to the core peaking factor and the hot channel enthalpy factor that will be used for core designs beginning with Cycle 7. A detailed discussion of the reanalysis was included in the correspondence related to Technical Specification Change 92-14 (references 2 and 3).

The enclosed documentation contains the recent changes to SQN's emergency core cooling system evaluation model and the effect on the peak fuel cladding temperature during the reporting period from July 29, 1992, to July 29, 1993. Reference 1 supplied SQN's last 10 CFR 50.46 update.

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Additionally, several potential issues are under investigation by Westinghouse Electric Corporation that may impact the PCT for both large and small break LOCA. The potential issues have had PCT margin temporarily allocated to ensure that the cumulative effects are tracked such that the 10 CFR 50.46 PCT limit of 2200 degrees Fahrenheit is not exceeded. Upon their resolution, these issues will continue to be reported as appropriate.

Please direct questions concerning this issue to W. C. Ludwig at (615) 843-7460.

Sincerely,

Robert A. Fenech

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Enclosure
cc (Enclosure):

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ENCLOSURE

10 CFR 50.46 ANNUAL REPORT DOCUMENTATION

Large Break Loss of Coolant Accident (LOCA)

	PCT	Attachment
Previous Licensing Basis Peak Fuel Cladding (PCT) (July 1, 1992)	2043°F	
1. Analysis of Record Revision - Peaking Factor/Hot Channel Factor Increase - Mini-Revised Thermal Design Procedure	+126°F	1
New Design Basis PCT	2169°F	
2. Structural Metal Heat Modeling	-25°F	2
3. Steam Generator Seismic LOCA Assumptions	+20°F	3
Updated Licensing Basis PCT	2164°F	
Net Change	+121°F	

Small Break LOCA

	PCT	Attachment
Previous Licensing Basis PCT (July 1, 1992)	2142°F	
1. Bessel Function Error	+11°F	4
2. Fuel Rod Initial Condition Inconsistency	-37°F	5
Updated Licensing Basis PCT	2116°F	
Net Change	-26°F	

A detailed discussion of each of the changes outlined above is included in the indicated attachment.

Large Break LOCA Analysis - FSAR CHAPTER 15.4.1

A large rupture of the reactor coolant system (RCS) piping is a hypothetical event postulated to demonstrate that the calculated performance of the emergency core cooling system is adequate to mitigate the consequences of such a scenario. The effect of increased core peaking factors during a hypothetical large rupture of the RCS piping is examined to ensure that the bases and assumptions of the calculation remain valid, and that a conservative approach will yield values that conform to 10CFR 50.46 standards. Following a large rupture of the cold leg RCS piping, the RCS depressurizes in approximately 30 seconds to a pressure nearly equal to the containment pressure. During this time the core flow reverses and the core is cooled by a two-phase mixture flowing down through the core, up the downcomer and out the break. When the reverse core flow period ends, end of bypass occurs, and the lower plenum can begin filling with cold safety injection water. After the lower plenum fills, and the bottom of the core is reached, the process of reflooding the core begins. The peak cladding temperature (PCT) for large break LOCAs occurs during the reflooding portion of the transient at elevations near or above the mid-plane of the core (6 feet) for the Westinghouse ECCS Evaluation Models.

A large break LOCA reanalysis has been performed for Sequoyah Nuclear Plant to support the Cycle 7 core design. The reanalysis used the 1981 Evaluation Model with BASH and assumed a reactor power level of 102% of 3411 Mwt, F_q=2.40, $F_{\Delta h}$=1.62, and uniform 10% steam generator tube plugging. The previous analysis had assumed a reactor power level of 102% of 3411 Mwt, F_q = 2.32, $F_{\Delta h}$ = 1.55, and uniform 10% steam generator tube plugging. The PCT calculated in this analysis was 2069°F for a V-5H fuel core (that is, no standard fuel is present). With a mixed core (including standard fuel), the PCT was determined to be 2169°F for the same parameters. This reflects a 100°F analysis penalty due to the inconel grids associated with the standard fuel assemblies. Until such time as the standard fuel is completely removed from the core, the new design basis PCT will remain 2169°F. The previous analysis (performed for UHI removal in 1988) calculated a PCT of 2043°F. As such, the revised analysis represents a 126°F increase in calculated PCT for large break LOCA.

PL025N04--3981

Structural Metal Heat Modeling

Background

A discrepancy was discovered during review of the finite element heat conduction model used in the WREFLOOD-INTERIM code to calculate heat transfer from structural metal in the vessel during the reflood phase. It was noted that the material properties available in the code corresponded to those of stainless steel. While this is correct for the internal structures, it is inappropriate for the vessel wall which consists of carbon steel with a thin stainless internal clad. This was defined as a non-discretionary change per Section 4.1.2 of WCAP-13451, since there was thought to be potential for increased PCT with a more sophisticated composite model. The model was revised by replacing it with a more flexible one that allows detailed specification of structures.

Affected Evaluation Models

1981 ECCS Evaluation Model with BART
1981 ECCS Evaluation Model with BASH

Affected Codes

WREFLOOD-INTERIM

Estimated Effects

The estimated effect of this correction is a 25°F PCT benefit.

STEAM GENERATOR FLOW AREA

Background:

Licensees are normally required to provide assurance that there exists only an extremely low probability of abnormal leakage or gross rupture of any part of the reactor coolant pressure boundary (General design criteria 14 and 31). The NRC issued a regulatory guide (RG 1.121) which addressed this requirement specifically for steam generator tubes in pressurized water reactors. In that guide, the staff required analytical and experimental evidence that steam generator tube integrity will be maintained for the combinations of the loads resulting from a LOCA with the loads from a safe shutdown earthquake (SSE). These loads are combined for added conservatism in the calculation of structural integrity. This analysis provides the basis for establishing criteria for removing from service tubes which had experienced significant degradation.

Analyses performed by Westinghouse in support of the above requirement for various utilities, combined the most severe LOCA loads with the plant specific SSE, as delineated in the design criteria and the Regulatory Guide. Generally, these analyses showed that while tube integrity was maintained, the combined loads led to some tube deformation. This deformation reduces the flow area through the steam generator. The reduced flow area increases the resistance through the steam generator to the flow of steam from the core during a LOCA, which potentially could increase the calculated PCT.

The effect of tube deformation and flow area reduction in the steam generator was analyzed and evaluated for some plants by Westinghouse in the late 1970's and early 1980's. The combination of LOCA and SSE loads led to the following calculated phenomena:

1. LOCA and SSE loads cause the steam generator tube bundle to vibrate.

2. The tube support plates may be deformed as a result of lateral loads at the wedge supports at the periphery of the plate. The tube support plate deformation may cause tube deformation.

3. During a postulated large LOCA, the primary side depressurizes to containment pressure. Applying the resulting pressure differential to the deformed tubes causes some of these tubes to collapse, and reduces the effective flow area through the steam generator.

4. The reduced flow area increases the resistance to venting of steam generated in the core during the reflood phase of the LOCA, increasing the calculated peak cladding temperature (PCT).

The ability of the steam generator to continue to perform its safety function was established by evaluating the effect of the resulting flow area reduction on the LOCA PCT. The postulated break examined was the steam generator outlet break, because this break was judged to result in the greatest loads on the steam generator, and thus the greatest flow area reduction. It was concluded that the steam generator would continue to meet its safety function because the degree of flow area reduction was small, and the postulated break at the steam generator outlet resulted in a low PCT.

In April of 1990, in considering the effect of the combination of LOCA + SSE loadings on the steam generator component, it was determined that the potential for flow area reduction due to the contribution of SSE loadings should be included in other LOCA analyses. With SSE loadings, flow area reduction may occur in all steam generators (not just the faulted loop). Therefore, it was concluded that the effects of flow area reduction during the most limiting primary pipe break affecting LOCA PCT, i.e., the reactor vessel inlet break (cold leg break LOCA), had to be evaluated to confirm that 10CFR 50.46 limits continue to be met and that the affected steam generators will continue to perform their intended safety function.

Consequently, the action was taken to address the safety significance of steam generator tube collapse during a cold leg break LOCA. The effect of flow area reduction from combined LOCA and SSE loads was estimated. The magnitude of the flow area reduction was considered equivalent to an increased level of steam generator tube plugging. Typically, the area reduction was estimated to range from 0 to 7.5%, depending on the magnitude of the seismic loads. Since detailed non-linear seismic analyses are not available for Series 51 and earlier design steam generators, some area reductions had to be estimated based on available information. For most of these plants, a 5 percent flow area reduction was assumed to occur in each steam generator as a result of the SSE. For these evaluations, the contribution of loadings at the tube support plates from the LOCA cold leg break was assumed negligible, since the additional area reduction, if it occurred, would occur only in the broken loop steam generator.

Westinghouse recognizes that, for most plants, as required by GDC 2, "Design Basis for Protection against Natural Phenomena", that steam generators must be able to withstand the effects of combined LOCA + SSE loadings and continue to perform their intended safety function. It is judged that this requirement applies to undegraded as well as locally degraded steam generator tubes. Compliance with GDC 2 is addressed below for both conditions.

For tubes which have not experienced cracking at the tube support plate elevations, it is Westinghouse's engineering judgment that the calculation of steam generator tube deformation or collapse as a result of the combination of LOCA loads with SSE loads does not conflict with the requirements of GDC 2. During a large break LOCA, the intended safety functions of the steam generator tubes are to provide a flow path for the venting of steam generated in the core through the RCS pipe break and to provide a flow path such that the other plant systems can perform their intended safety functions in mitigating the LOCA event.

Tube deformation has the same effect on the LOCA event as the plugging of steam generator tubes. The effect of tube deformation and/or collapse can be taken into account by assigning an appropriate PCT penalty, or accounting for the area reduction directly in the analysis. Evaluations completed to date show that tube deformation results in acceptable LOCA PCT. From a steam generator structural integrity perspective, Section III of the ASME Code recognizes that inelastic deformation can occur for faulted condition loadings. There are no requirements that equate steam generator tube deformation, per se, with loss of safety function. Cross-sectional bending stresses in the tubes at the tube support plate elevations are considered secondary stresses within the definitions of the ASME Code and need not be

considered in establishing the limits for allowable steam generator tube wall degradation. Therefore, for undegraded tubes, for the expected degree of flow area reduction, and despite the calculation showing potential tube collapse for a limited number of tubes, the steam generators continue to perform their required safety functions after the combination of LOCA + SSE loads, meeting the requirements of GDC 2.

During a November 7, 1990 meeting with a utility and the NRC staff on this subject, a concern was raised that tubes with partial wall cracks at the tube support plate elevations could progress to through-wall cracks during tube deformation. This may result in the potential for significant secondary to primary inleakage during a LOCA event; it was noted that inleakage is not addressed in the existing ECCS analysis. Westinghouse did not consider the potential for secondary to primary inleakage during resolution of the steam generator tube collapse item. This is a relatively new item, not previously addressed, since cracking at the tube support plate elevations had been insignificant in the early 1980's when the tube collapse item was evaluated in depth. There is ample data available which demonstrates that undegraded tubes maintain their integrity under collapse loads. There is also some data which shows that cracked tubes do not behave significantly differently from uncracked tubes when collapse loads are applied. However, cracked tube data is available only for round or slightly ovalized tubes.

It is important to recognize that the core melt frequency resulting from a combined LOCA + SSE event, subsequent tube collapse, and significant steam generator tube inleakage is very low, on the order of 10^{-8} /RY or less. This estimate takes into account such factors as the possibility of a seismically induced LOCA, the expected occurrence of cracking in a tube as a function of height in the steam generator tube bundle, the localized effect of the tube support plate deformation, and the possibility that a tube which is identified to deform during LOCA + SSE loadings would also contain a partial through-wall crack which would result in significant inleakage. To further reduce the likelihood that cracked tubes would be subjected to collapse loads, eddy current inspection requirements can be established. The inspection plan would reduce the potential for the presence of cracking in the regions of the tube support plate elevations near wedges that are most susceptible to collapse which may then lead to penetration of the primary pressure boundary and significant inleakage during a LOCA + SSE event.

Change Description

As noted above, detailed analyses which provide an estimate of the degree of flow area reduction due to both seismic and LOCA forces are not available for all steam generators. The information that does exist indicates that the flow area reduction may range from 0 to 7.5 percent, depending on the magnitude of the postulated forces, and accounting for uncertainties. It is difficult to estimate the flow area reduction for a particular steam generator design, based on the results of a different design, due to the differences in the design and materials used for the tube support plates.

While a specific flow area reduction has not been determined for some earlier design steam generators, the risk associated with flow area reduction and tube leakage from a combined seismic and LOCA event has been shown to be exceedingly low. Based on this low risk, it is considered adequate to assume, for those plants which do not have a detailed analysis, that 5 percent of the tubes are susceptible to deformation.

The effect of potential steam generator area reduction on the cold leg break LOCA peak cladding temperature has been either analyzed or estimated for each Westinghouse plant. A value of 5 percent area reduction has been applied, unless a detailed non-linear analysis is available. The effect of tube deformation and/or collapse will be taken into account by allocating the appropriate PCT margin, or by representing the area reduction by assuming additional tube plugging in the analysis.

Affected Evaluation Models:

1978 Large Break ECCS Evaluation Model
1981 Large Break ECCS Evaluation Model
1981 Large Break ECCS Evaluation Model with BART
1981 Large Break ECCS Evaluation Model with BASH

Status:

Complete.

SG TUBE SEISMIC/LOCA ASSUMPTION ΔPCT= + 20 °F

NOTRUMP BESSEL FUNCTION ERROR

Resolution

INTRODUCTION

Westinghouse has completed its evaluation of an issue affecting the NOTRUMP small break LOCA Evaluation Model. This information is being provided to allow affected utilities to assess individual reporting requirements which may exist due to changes in Peak Cladding Temperature (PCT) in their small break LOCA analysis results.

BACKGROUND

During a recently completed effort, anomalous behavior was noted in the NOTRUMP runs. This behavior was eventually traced to an error in SUBROUTINE BESSJ0 which calculates Bessel Function values used during the transient solution. During the time before this error was corrected, convergence anomalies were observed in NOTRUMP. It has been determined that this error was introduced in Cycle 23 of the NOTRUMP code and that only analyses performed with this version of the code are affected. Subsequent reruns with a corrected version of NOTRUMP (cycle 24) showed that the convergence abnormalities were indeed the result of the Bessel Function error, and that the standard convergence criteria used for Evaluation Model calculations continue to be valid when the corrected code is used.

EFFECTS OF ISSUE

The effect of this issue on Sequoyah Unit 1 (TVA) has been determined by a plant specific calculation to be a change of +11°F. This result should be evaluated to determine reporting requirements under 10 CFR 50.46.

FUEL ROD MODEL REVISIONS

During the review of the original Westinghouse ECCS Evaluation Model following the promulgation of 10CFR50.46 in 1974, Westinghouse committed to maintain consistency between future loss-of-coolant accident (LOCA) fuel rod computer models and the fuel rod design computer models used to predict fuel rod normal operation performance. These fuel rod design codes are also used to establish initial conditions for the LOCA analysis.

Change Description:

It was found that the large break and small break LOCA code versions were not consistent with fuel design codes in the following areas:

1. The LOCA codes were not consistent with the fuel rod design code relative to the flux depression factors at higher fuel enrichment.

2. The LOCA codes were not consistent with the fuel rod design code relative to the fuel rod gap gas conductivities and pellet surface roughness models.

3. The coding of the pellet/clad contact resistance model required revision.

Modifications were made to the fuel rod models used in the LOCA Evaluation Models to maintain consistency with the latest approved version of the fuel rod design code.

In addition, it was determined that integration of the cladding strain rate equation used in the large break LOCA Evaluation Model, as described in Reference 3, was being calculated twice each time step instead of once. The coding was corrected to properly integrate the strain rate equation.

Affected Evaluation Models:

1981 Large Break LOCA Evaluation Model
1981 Large Break LOCA Evaluation Model, With BART
1981 Large Break LOCA Evaluation Model, With BASH
1975 Small Break LOCA Evaluation Model
1985 Small Break LOCA Evaluation Model

Effect of Changes:

The changes made to make the LOCA fuel rod models consistent with the fuel design codes were judged to be insignificant, as defined by 10CFR50.46(a)(i). To quantify the effect on the calculated peak cladding temperature (PCT), calculations were performed which incorporated the changes, including the cladding strain model correction for the large break LOCA. For the large break LOCA Evaluation Model, additional calculations, incorporating only the cladding strain corrections were performed and the results supported the conclusion that compensating effects were not present. The PCT effects reported below will bound the effects taken separately for the large break LOCA.

Small Break LOCA

The effect of the changes on the small break LOCA analysis peak cladding temperature calculations was determined using the 1985 small break LOCA Evaluation Model by performing a computer analysis calculations for a typical three-loop plant and a typical four-loop plant. The analysis calculations confirmed that the effect of the changes on the small break LOCA ECCS Evaluation Model were insignificant as defined by 10CFR50.46(a)(3)(i). The calcul tions showed that 37°F would bound the effect on the calculated peak cladding temperatures for the four-loop plants and the three-loop plants. It was judged that an increase of 37°F would bound the effect of the changes for the 2-loop plants.

Status:

Changes completed and implemented.

TVA Note - Upon completion of the Sequoyah plant specific calculations necessary to address the NOTRUMP bessel function error, it was determined that the fuel rod modeling revisions had no impact on calculated PCT. As such, the 37 F PCT permanent assesment discussed above was deleted from the Sequoyah SB LOCA analysis .