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October 12, 1985
ST-HL-AE-1394
File No.: G9.17

Mr. George W. Knighton, Chief
Licensing Branch No. 3
Division of Licensing
U. S. Nuclear Regulatory Commission
Washington, DC 20555

South Texas Project
Units 1 and 2
Docket Nos. STN 50-498, STN 50-499
Responses to DSER/FSAR Item; Fuel Design

Dear Mr. Knighton:

The attachments enclosed provide STP's response to Draft Safety Evaluation Report (DSER) or Final Safety Analysis Report (FSAR) items.

The item numbers listed below correspond to those assigned on STP's internal list of items for completion which includes open and confirmatory DSER items, STP FSAR open items and open NRC questions. This list was given to your Mr. N. Prasad Kadambi on October 8, 1985 by our Mr. M. E. Powell.

The attachments include mark-ups of FSAR pages which will be incorporated in a future FSAR amendment unless otherwise noted below.

The items which are attached to this letter are:

Attachment	Item No.*	Subject
1	D 4.2-3	Fuel Rod conformance of the revised internal rod pressure design basis described in WCAP 8963 and 8964.
	D 4.2-2	Provide enthalpy limit for irradiated fuel
	C 4.2-5	Confirm that load analysis for guide thimble tube degradation was performed using 6 g.
	C 4.2-6	Provide reference to WCAP-9220-P-A (Rev. 1) and WCAP-9221-A in the FSAR

* Legend
D - DSER Open Item　　C - DSER Confirmatory Item
F - FSAR Open Item　　Q - FSAR Question Response Item

L1/DSER/m

B001
1/1

If you should have any questions concerning this matter, please contact Mr. Powell at (713) 993-1328.

Very truly yours,

M. R. Wisenburg
Manager, Nuclear Licensing

JSP/bl

Attachments: See above

L1/DSER/m

cc:

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Revised 9/25/85

Attachment 1

Reference 4.2-2 provides the basis for justifying that no adverse chemical interactions occur between the fuel and its adjacent material.

4.2.1.3 Fuel Rod Performance.

a. Fuel Rod Models

The basic fuel rod models and the ability to predict operating characteristics are given in Reference 4.2-5 and the Design Evaluation Section 4.2.3.

b. Mechanical Design Limits

Cladding collapse shall be precluded during the fuel rod design lifetime. The models described in Reference 4.2-6 shall be used for this evaluation.

The rod internal gas pressure shall remain below the value which causes the fuel-clad diametral gap to increase due to outward cladding creep during steady-state operation (See Reference 4.2-16). Rod pressure is also limited such that extensive DNB propagation shall not occur during normal operation and accident events. 1

4.2.1.4 Spacer Grids.

a. Mechanical Limits and Materials Properties

Lateral loads resulting from seismic/LOCA events will not cause unacceptably high plastic grid deformation. Each fuel assembly's geometry will be maintained such that the full rods remain in an array amenable to cooling. The behavior of the grids under loading has been studied experimentally. (See Reference 4.2-12) 1

The grid material and chemical properties are given in Reference 4.2-2.

b. Vibration and Fatigue

The grids shall provide sufficient fuel rod support to limit fuel rod vibration and maintain clad fretting wear to within acceptable limits (defined in Subsection 4.2.1.1).

4.2.1.5 Fuel Assembly.

1. Structural Design

As previously discussed in Section 4.2.1, the structural integrity of the fuel assemblies is assured by setting design limits on stresses and deformations due to various non-operational, operational and accident loads.

REFERENCES (Continued)

SECTION 4.2:

4.2-15 Anderson, T. M., Westinghouse, April 21, 1981, letter to J. R. Miller of the NRC. 45

4.2-16 Risher, D. H. (Ed) "Safety Analysis for the Revised Fuel Rod Internal Pressure Design Basis," WCAP-8963-P-A (Proprietary), August, 1978, and WCAP-8964, August, 1978.

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shield. The top end plates of the position indicator coil assemblies would prevent the broken piece from directly hitting the rod travel housing of a second drive mechanism. Even if a direct hit by the rebounding piece were to occur, the low kinetic energy of the rebounding projectile would not be expected to cause significant damage.

Possible Consequences

From the above discussion, the probability of damage to an adjacent housing must be considered remote. However, even if damage is postulated, it would not be expected to lead to a more severe transient since RCCAs are inserted in the core in symmetric patterns, and control rods immediately adjacent to worst ejected rods are not in the core when the reactor is critical. Damage to an adjacent housing could, at worst, cause that RCCA not to fall on receiving a trip signal, however this is already taken into account in the analysis by assuming a stuck rod adjacent to the ejected rod.

Summary

The considerations given above lead to the conclusion that failure of a control rod housing, due either to longitudinal or circumferential cracking, would not cause damage to adjacent housings that would increase severity of the initial accident.

15.4.8.1.2 Limiting Criteria: This event is classified as an ANS Condition IV incident. See Section 15.0.1. Due to the extremely low probability of a RCCA ejection accident, some fuel damage could be considered an acceptable consequence.

Comprehensive studies of the threshold of fuel failure and of the threshold of significant conversion of fuel thermal energy to mechanical energy have been carried out as part of the SPERT project by the Idaho Nuclear Corporation (Ref. 15.4-8). Extensive tests of UO_2 zirconium clad fuel rods representative of those in pressurized water reactor type cores have demonstrated failure thresholds in the range of 240 to 257 cal/gm. However, other rods of a slightly different design have exhibited failures as low as 225 cal/gm. These results differ significantly from the TREAT (Ref. 15.4-9) results, which indicated a failure threshold to 280 cal/gm. Limited results have indicated that this threshold decreases by about 10 percent with fuel burnup. The clad failure mechanism appears to be melting for zero burnup rods and brittle fracture for irradiated rods. Also important is the conversion ratio of thermal to mechanical energy. This ratio becomes marginally detectable above 300 cal/gm for unirradiated rods and 200 cal/gm for irradiated rods; catastrophic failure (large fuel dispersal, large pressure rise) for irradiated rods did not occur below 300 cal/gm.

In view of the above experimental results, criteria are applied to ensure that there is little or no possibility of fuel dispersal in the coolant, gross lattice distortion, or severe shock waves. These criteria are:

1. Average fuel pellet enthalpy at the hot spot below 225 cal/gm for unirradiated fuel and 200 cal/gm for irradiated fuel.

As discussed in Section 4.2.1 and Section 4.2.1.5, the fuel assembly design loads for shipping and handling have been established at ~~6 g's~~ 6 g lateral and 4 g axial while maintaining dimensional stability. Accelerometers are permanently placed into the shipping cask to monitor and detect fuel assembly accelerations that would exceed the criteria. Past history and experience has indicated that loads which exceed the allowable limits rarely occur. Exceeding the limits requires reinspection of the fuel assembly for damage. Tests on various fuel assembly components such as the grid assembly, sleeves, inserts and structure joints have been performed to assure that the shipping design limits do not result in impairment of fuel assembly function. As discussed in Section 9.1.4, the Fuel Handling System is designed such that the inertial loads imparted to the fuel assemblies during handling operations are less than the loads which could cause damage. Seismic analysis of the fuel assembly is presented in Reference 4.2-12.

4.2.3.5.2 Dimensional Stability: A prototype fuel assembly has been subjected to column loads in excess of those expected in normal service and faulted conditions (Ref. 4.2-12).

No interference with control rod insertion into thimble tubes will occur during a postulated LOCA transient due to fuel rod swelling, thermal expansion, or bowing. In the early phase of the transient following the coolant pipe break, the high axial loads, which could be generated by the difference in thermal expansion between fuel clad and thimbles, are relieved by slippage of the fuel rods through the grids. The relatively low drag force restraint on the fuel rods will induce only minor thermal bowing, which is insufficient to close the fuel rod-to-thimble tube gap.

Reference 4.2-12 shows that the fuel assemblies will maintain a geometry amenable to cooling during a combined seismic and double-ended LOCA.

4.2.3.6 Incore Control Components. The components are analyzed for loads corresponding to normal, upset, emergency and faulted conditions. The analysis performed depends on the mode of operation under consideration.

The scope of the analysis requires many different techniques and methods, both static and dynamic.

Some of the loads that are considered on each component where applicable are as follows:

1. Control rod trip (equivalent static load)
2. Differential pressure
3. Spring preloads
4. Coolant flow forces (static)
5. Temperature gradients
6. Differences in thermal expansion
   a. Due to temperature differences
   b. Due to expansion of different materials
7. Interference between components

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The containment airborne iodine inventory available for release is assumed to be the flashed portion of the total primary coolant iodine inventory based on a pre-existing iodine spike level of 60 μCi/g dose equivalent I-131. For noble gases, 100 percent of the primary coolant inventory based on 1 percent failed fuel is assumed to be available for release. No failed fuel is assumed since isolation occurs prior to the core reaching a temperature which could cause a fuel failure. 45

15.6.5.3.3.1 Containment Purge Doses - The offsite doses calculated due to Containment purging are presented in Table 15.6-11 for the exclusion zone boundary (EZB) of 1,430 meters and low population zone (LPZ) outer boundary distance of 4,800 meters. 45

15.6.5.4 Core and System Performance.

15.6.5.4.1 Mathematical Model: The requirements of an acceptable ECCS Evaluation Model are presented in Appendix K of 10CFR50 (Ref. 15.6-2).

Large Break LOCA Evaluation Model

The analysis of a large break LOCA transient is divided into three phases: 1) blowdown, 2) refill, and 3) reflood. There are three distinct transients analyzed in each phase, including the thermal hydraulic transient in the RCS, the pressure and temperature transient within the Containment, and the fuel and clad temperature transient of the hottest fuel rod in the core. Based on these considerations, a system of inter-related computer codes has been developed for the analysis of the LOCA.

The description of the various aspects of the LOCA analysis methodology is given in WCAP-8339 (Ref. 15.6-4). This document describes the major phenomena modeled, the interfaces among the computer codes, and the features of the codes which ensure compliance with the Acceptance Criteria. The SATAN-VI, WREFLOOD, COCO, and LOCTA-IV codes, which are used in the LOCA analysis, are described in detail in References 15.6-5 through 15.6-8. Modifications to these codes are specified in References 15.6-9, 15.6-10, and 15.6-11. The BART code is described in References 15.6-11e and 15.6-11f. These codes are 51|18 used to assess the core heat transfer geometry and to determine if the core remains amenable to cooling throughout and subsequent to the blowdown, refill, and reflood phases of the LOCA. The SATAN-VI computer code analyzes the thermal-hydraulic transient in the RCS during blowdown and the WREFLOOD computer code is used to calculate this transient during the refill and reflood phases of the accident. The COCO computer code is used to calculate the Containment pressure transient during all three phases of the LOCA analysis. Similarly, the LOCTA-IV computer code is used to compute the thermal transient of the hottest fuel rod during the three phases.

The large-break analysis was performed with the approved December, 1981 version of the Evaluation Model (Reference 15.6-10), with BART (Reference 15.6-11g) which includes modifications delineated in Reference 15.6-11h. 51

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REFERENCES

SECTION 15.6:

15.6-1 Burnett, T. W. T., et al., "LOFTRAN Code Description," WCAP-7907-P-A (Proprietary Class 2), WCAP-7907-A (Proprietary Class 3), April 1984 45

15.6-2 "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Cooled Nuclear Power Reactors," 10 CFR 50.46 and Appendix K of 10 CFR 50. 45

15.6-3 "Reactor Safety Study - An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants," WASH-1400, NUREG-75/014, October, 1975.

15.6-4 Bordelon, F. M., Massie, H. W., and Borden, T. A. "Westinghouse ECCS Evaluation Model-Summary," WCAP-8339 (Non-proprietary), July, 1974.

15.6-5 Bordelon, F. M., et al., "SATAN-VI Program: Comprehensive Space Time Dependent Analysis of Loss of Coolant," WCAP-8302, June, 1974 (Proprietary) and WCAP-8306, June, 1974 (Non-Proprietary).

15.6-6 Kelly, R. D., et al., "Calculated Model for Core Reflooding After a Loss of Coolant Accident (WREFLOOD Code)," WCAP-8170, June, 1974 (Proprietary) and WCAP-8171, June, 1974 (Non-Proprietary).

15.6-7 Bordelon, F. M. and Murphy, E. T., "Containment Pressure Analysis Code (COCO)," WCAP-8327, June, 1974 (Proprietary) and WCAP-8326, June, 1974 (Non-Proprietary).

15.6-8 Bordelon, F. M., et al., "LOCTA-IV Program: Loss of Coolant Transient Analysis," WCAP-8301, June, 1974 (Proprietary) and WCAP-8305, June, 1974 (Non-Proprietary).

15.6-9 Bordelon, F. M., et al., "Westinghouse ECCS Evaluation Model - Supplementary Information," WCAP-8471-P-A, April, 1975 (Proprietary) and WCAP-8472-A, April, 1975 (Non-Proprietary). 18

15.6-10 ~~"Westinghouse ECCS Evaluation Model October 1975 Version," WCAP-8622, November 1975 (Proprietary), and WCAP-8623, November 1975 (Non-Proprietary).~~

"Westinghouse ECCS Evaluation Model, 1981 Version", WCAP-9220-P-A, Rev. 1 (Proprietary), WCAP-9221-A, Rev. 1 (nonproprietary), February 1982.

15.6-11 Letter from C. Eicheldinger of Westinghouse Electric Corporation to D. B. Vassallo of the Nuclear Regulatory Commission, Letter Number NS-CE-924 dated January 23, 1976.