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

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U. S. Nuclear Regulatory Commission
Attention: Document Control Desk
One White Flint North
11555 Rockville Pike
Rockville, MD 20852-2738

South Texas Project
Units 3 and 4
Docket No. PROJ0772
Response to Request for Additional Information

Reference: Letter from Tekia Govan to Mark McBurnett, "Request for Additional Information Re: South Texas Project Nuclear Operating Company Topical Report (TR) WCAP-17116-P Revision 0, Supplement 5 – Application to the Advanced Boiling Water Reactor" (TAC No. RG0012), June 7, 2010 (ML101580249)

Attached are the 60-day responses to NRC staff questions included in the reference. The responses to the following RAI questions are provided:

RAI-9	
RAI-23	RAI-29
RAI-27	RAI-33

There are no commitments in this letter.

If you have any questions, please contact Scott Head at (361) 972-7136, or Bill Mookhoek at (361) 972-7274.

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

Executed on 8/4/2010

Mark McBurnett
Vice President, Oversight and Regulatory Affairs
South Texas Project Units 3 & 4

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Attachments:

1. RAI-9
2. RAI-23
3. RAI-27
4. RAI-29
5. RAI-33

cc: w/o attachment except*
(paper copy)

Director, Office of New Reactors
U. S. Nuclear Regulatory Commission
One White Flint North
11555 Rockville Pike
Rockville, MD 20852-2738

Regional Administrator, Region IV
U. S. Nuclear Regulatory Commission
611 Ryan Plaza Drive, Suite 400
Arlington, Texas 76011-8064

Kathy C. Perkins, RN, MBA
Assistant Commissioner
Division for Regulatory Services
Texas Department of State Health Services
P. O. Box 149347
Austin, Texas 78714-9347

Alice Hamilton Rogers, P.E.
Inspection Unit Manager
Texas Department of State Health Services
P. O. Box 149347
Austin, Texas 78714-9347

C. M. Canady
City of Austin
Electric Utility Department
721 Barton Springs Road
Austin, TX 78704

*Steven P. Frantz, Esquire
A. H. Gutterman, Esquire
Morgan, Lewis & Bockius LLP
1111 Pennsylvania Ave. NW
Washington D.C. 20004

*Tekia Govan
Two White Flint North
11545 Rockville Pike
Rockville, MD 20852

(electronic copy)

*George F. Wunder
*Tekia Govan
Loren R. Plisco
U. S. Nuclear Regulatory Commission

Steve Winn
Joseph Kiwak
Eli Smith
Nuclear Innovation North America

Peter G. Nemeth
Crain, Caton & James, P.C.

Richard Peña
Kevin Pollo
L. D. Blaylock
CPS Energy

RAI-9

QUESTION

WCAP-17116-P indicates that the objective of the ECCS analysis for the ABWR is to "simply show that either the core does not uncover or it uncovers minimally so that there is no appreciable cladding heat-up after the initial boiling transition to demonstrate the adequate performance of the ECCS equipment."

The ECCS design objective noted above, however, leaves open a number of questions regarding the definition of the terms for successful completion of the design objectives. *Minimal Core Uncovery* could translate into a two-phase mixture level 6-inches below the top of active fuel, one foot below the top of active fuel, or up to three feet below the top of active fuel. In each of these instances, cladding heat-up may not result if there is sufficient steam cooling and entrained liquid impingent upon spacer grids which would provide rewet cooling. Similarly, No *appreciable cladding heat-up* after the initial boiling transition can have a number of interpretations. It could mean that any secondary cladding heat-up is lower than the initial boiling transitions heat-up. This would still be well below the regulatory limit, but could translate into a two-phase mixture level up to 4 feet below the top of active fuel under steam cooling conditions. It could mean that the cladding temperature only exceeds the normal operating temperature by 50°F. In fact, the two-phase mixture level could be below the top of the core during the transient with the cladding temperature being below the normal operating temperature.

a) Provide quantitative explanations of the design objectives:

Minimal core uncovery, and

ii. No appreciable cladding heat-up after initial boiling transition

RESPONSE

Only one ABWR LOCA scenario shows complete depletion of upper plenum inventory and partial uncovery of fuel assemblies which is the High Pressure Core Flooder (HPCF) line break with failure of the emergency diesel generator (EDG) that powers the unaffected HPCF loop. The equipment available for this case is 1 RCIC + 2 LPFL + 8 Automatic Depressurization System (ADS).

In this case, the break is located in the upper plenum region, and, with both Reactor Core Isolation Cooling (RCIC) and Low Pressure Flooder (LPFL) delivering water to the Reactor Pressure Vessel (RPV) annular downcomer region, core inventory is replenished solely by reflooding from the bottom. Due to the small break size and the break elevation, core inventory is maintained until ADS is actuated. After the ADS is actuated, the loss of system inventory accelerates, and the core region becomes partially uncovered until LPFL refills the core from the bottom. During this period of partial core uncovery, the average core is filled with a two-phase

mixture and the severity of individual fuel channel uncovery depends on the local bundle power level. For the very low-power fuel channels, the lower rate of steam generation due to the removal of decay heat may not sustain a two- phase mixture level to the top of the fuel channel. For example, the top 20% of a low power channel would be cooled by steam generated in the lower 80% of the channel.

As documented in WCAP-17116-P, sensitivity studies of the impact of fuel bundle power on the severity of fuel channel uncovery and subsequent fuel rod heat up were performed by varying the single GOBLIN channel power peaking factor from 0.3 to 1.7. The results show that, while the two-phase mixture level decreases below the top of the low power fuel assemblies, the fuel rod temperature increase in those assemblies remains well below the peak clad temperature (PCT) that occurs during the initial dryout, which is well below the 10CFR50.46 acceptance criterion.

There are no design criteria for minimal core uncovery, since the real design criteria remain the acceptance criteria of 10CFR50.46.

RAI-23

QUESTION

In WCAP-17116-P Section 6.1.6, Westinghouse indicates that heat transfer from piping, vessel walls, and non-fuel internal hardware is accounted for according to the method described in Sections 3.5 and 3.6 of WCAP-11284-P-A. This is in compliance with the requirement of Section I.A.6 of Appendix K, "Heat transfer from piping, vessel walls, and non-fuel internal hardware shall be taken into account."

Section II.1.a of Appendix K also requires sufficiently complete description of the evaluation model to permit a technical review of the values of all parameters or the procedure for their selection. Appendix B of WCAP-17116-P only provides information regarding the surface areas and thicknesses of the metal heat structures. However, it is not completely clear how the metal heat structures are modeled; single and doubly exposed slabs, lumped parameter models, or more correct fin models for some of the internals structures. Furthermore, detailed information on heat structure nodalization is not available. Therefore, describe the heat structure model input in detail.

RESPONSE

GOBLIN can model any number of heat-transferring plates simulating different parts of the vessel or its internals that are in contact with the coolant. The plates may be insulated on either side or in contact with two different volume cells. The one-dimensional heat conduction equation is solved using a finite difference technique and a user-specified nodal subdivision of each plate. Table B-1 and Figures B-1 through B-4 of WCAP-17116-P document the dimensions and geometry of the core structures. These core structures are then broken down into many individual heat slabs. There are over 160 such heat slabs in the ABWR model that are used to represent the reactor vessel and its internals. The volume nodes that border the left and right side of each slab are defined, as well as the heat transfer area of each side, based upon the dimensions and geometry of the structure modeled by each heat slab. This modeling could include the same volume on both sides of the heat slab as is the case of the steam dryer, which as Figure B-1 illustrates, borders volume WV-P on both sides. A heat slab can also be defined as insulated on one side such as is the case for the reactor pressure vessel (RPV) wall, which as Figure B-1 illustrates, borders volume WV-Q on one side without bordering another volume node on the other. No heat slab is modeled in such a way that multiple volume nodes border a single side of the heat slab. For example the RPV wall is broken down into 5 heat slabs for the region where it borders volume nodes WV-S1 through WV-S5 as illustrated by Figure B-1, with one heat slab corresponding to each individual volume node. The top and bottom elevations of each individual heat slab are then input such that the heat slabs stack on top of one another to provide the overall model of each core structure. The material type(s) and number of mesh points for each heat slab are also defined.

RAI-27

QUESTION

How has the CCFL modeling been adjusted to account for the present ABWR fuel design?

RESPONSE

The ABWR fuel design to be used is the SVEA-96 Optima2 fuel, and the Countercurrent Flow Limitation (CCFL) modeling for this fuel was reviewed and accepted by NRC in Supplement 3 to WCAP-16078-P-A titled "Westinghouse BWR Emergency Core Cooling System (ECCS) Evaluation Model: Supplement 3 to Code Description, Qualification and Application to SVEA-96 Optima2 Fuel". Since the same fuel is being utilized in the ABWR as was previously approved by the NRC, no change to the CCFL modeling is necessary.

RAI 29

QUESTION

The acceptance criteria specified in 10CFR50.46 for the ECCS performance evaluation includes maximum cladding oxidation and maximum hydrogen generation. Provide maximum cladding oxidation and maximum hydrogen generation calculated for each LOCA scenario analyzed in WCAP-17116-P.

RESPONSE

WCAP-17116-P is intended to demonstrate the LOCA methodology that will be applied to the ABWR rather than being an analysis for a specific plant. Since some of the inputs to the calculations are typical, the results presented in the LTR should be interpreted as being representative. Plant specific analyses will be performed after the actual inputs are available.

As documented in WCAP-17116-P, two cases were shown to have the highest Peak Clad Temperature (PCT) as calculated by the GOBLIN hot assembly analysis: a small feedwater line break and a small RHR suction line break initiated from 90% core flow rate. In both cases, the cladding heatup is short-lived. As a result, the transient oxidation is small. The small feedwater line break was selected for the heatup analysis using CHACHA.

The local maximum oxidation for this case was shown to be < 1% which is well below the 10CFR50.46 acceptance criteria of 17%.

The core-wide oxidation (CWO) of <0.2% is calculated very conservatively by assuming that all bundles at all axial locations behave like the most limiting lattice. A more detailed core-wide oxidation calculation, as described in Section 8.2.3.4 of ABB Combustion Engineering Topical Report No. CENPD-300-P-A, "Reference Safety Report for Boiling Water Reactor Fuel," July 1996, would show that CWO is negligible and far less than the acceptance criterion of 1%.

RAI-33

QUESTION

Appendix B of WCAP-17116-P provides input parameters that were used for the LOCA analysis. The values of some parameters specified in Appendix B are different from those of the ABWR Design Certification Document (DCD). Please explain the apparent discrepancy between the ABWR DCD and WCAP-17116-P values for the following input parameters:

In addition, please verify if the GOBLIN inputs (both the design specific and model parameters) have been subjected to an internal Westinghouse Quality Assurance (QA) review and verification.

Input Parameter	Appendix B of WCAP-17116	ABWR DCD
Initial water level m	12.85	13.42 DCD Table 12.2-1a
Rm3~h RIP flow rate	6750	6912 (DCD Table 5.4-1)
Rated RIP head m	34.0	32.6 DCD Table 5.4-1
Scram rod worth table	See Table B-1 of WCAP-17116-P	See Table 15.0-5 of ABWR DCD

RESPONSE

The LOCA analysis performed in WCAP-17116-P demonstrates Westinghouse LOCA methodology and does not represent the actual plant or mimic or defend the information and analysis documented in the ABWR DCD. The LOCA inputs for WCAP-17116-P were based on the most current Toshiba design reports at the time of the analysis. All GOBLIN inputs and analyses have undergone internal Westinghouse Quality Assurance (QA) review and verification.

Furthermore, the apparent discrepancies between the WCAP-17116-P inputs and the ABWR DCD inputs are explained as follows:

Initial Water Level:

WCAP-17116-P LOCA analysis utilizes an initial water level conservatively set to LWL-3 (scram water level), whereas ABWR DCD Table 12.2-1a provides a normal water level.

Rated RIP Flow Rate / Head:

The rated Reactor Internal Pump (RIP) flow rate and head used in the LOCA analysis are typical safety analysis values for the ABWR. The ABWR DCD Table 5.4-1 values appear to only be representative of the 100% reactor power / 100% core flow case.

Scram Rod Worth Table:

The scram rod worth table is dependent upon the fuel and core design. As such the WCAP-17116-P and ABWR DCD inputs are expected to differ as the WCAP-17116-P LOCA analysis assumes SVEA-96 Optima2 fuel, whereas the ABWR DCD assumes 8x8 fuel.