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SHEARON HARRIS NUCLEAR POWER PLANT
DOCKET NO. 50-400/LICENSE NO. NPF-63
SUPPLEMENT TO CYCLE 3 RELOAD AMENDMENT REQUEST

Gentlemen:

Carolina Power & Light Company (CP&L) hereby submits a supplement to the April 17, 1989 license amendment request concerning Technical Specification changes in support of the Cycle 3 reload for the Shearon Harris Nuclear Power Plant. This letter provides CP&L's response to NRC staff review questions transmitted by letter dated August 31, 1989. Attachment 1 provides the NRC staff review questions and CP&L's responses. In addition, this letter provides confirmation of a telephone conversation between Mr. M. R. Oates of CP&L, Mr. R. A. Becker of NRC, and other members of the respective staffs on September 7, 1989. The discussion concerned a staff inquiry as to why the Cycle 3 Reload Amendment Request did not include marked-up FSAR pages specifically concerning peak containment pressures. The following is a restatement of CP&L's response.

The reanalysis of the LOCA event presented in Attachment 4 of the April 17, 1989 reload amendment request included a revised minimum containment backpressure calculation. Containment pressure is important because it controls the downcomer pressure and therefore the core water level during reflood. As a result, FSAR Sections 6.2.1.5 and 15.6.5, which describe the minimum containment backpressure and LOCA analyses, will be revised accordingly. The containment design basis analysis presented in FSAR Sections 6.2.1.1.3 and 6.2.1.3, however remain unchanged. This analysis was not revised because the introduction of Vantage 5 fuel into the core has a negligible impact on the peak temperature and pressure, which typically occur prior to the end of blowdown.

In addition, references to the Core Operating Limits Report (COLR) in Specifications 3.1.3.1 and 3.2.1 are being revised and a typographical error is being corrected on page B 2-1. These changes are administrative in nature and as such the 10CFR50.92 Evaluation and the Environmental Evaluation provided in the Company's April 17, 1989 submittal remain valid.

A001 Add: NRR Chatterton Ltr Encl
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411 Fayetteville Street • P. O. Box 1551 • Raleigh, N. C. 27602

Attachment 2 contains revisions of three marked-up Technical Specification pages from the original April 17, 1989 submittal of the Technical Specification change request.

Please refer any questions regarding this submittal to Mr. John Eads at (919) 546-4165.

Yours very truly,

A. B. Cutter

ABC/SDC/rlj (470CRS)

Attachments

cc: Mr. R. A. Becker
Mr. W. H. Bradford
Mr. Dayne H. Brown
Mr. S. D. Ebneter

A. B. Cutter, having been first duly sworn, did depose and say that the information contained herein is true and correct to the best of his information, knowledge and belief; and the sources of his information are officers, employees, contractors, and agents of Carolina Power & Light Company.

Ruby R. Morgan

Notary (Seal)
My commission expires: 11/27/89

RUBY R. MORGAN
NOTARY
PUBLIC
WAKE COUNTY, N.C.

RESPONSES TO NRC QUESTIONS ON CYCLE 3 TECHNICAL SPECIFICATION CHANGE REQUEST

NRC QUESTION 1

The Licensee should address the conformance with restrictions specified in the NRC SER on WCAP-10444; items 1, 2, 4 and 10 in the summary and conclusion.

SER Restriction (1): The statistical convolution method described in WCAP-10125 for evaluation of initial fuel rod to nozzle growth gap has not been approved. This method should not be used in VANTAGE 5.

CP&L Response

The statistical convolution method described in WCAP-10125 was not used for evaluation of initial fuel rod to nozzle growth gap. Worst case fabrication tolerances were used to determine the initial fuel rod to nozzle growth gaps in the evaluation of fuel rod performance.

SER Restriction (2): For each plant application, it must be demonstrated that the LOCA/seismic loads considered in WCAP-9401 bound the plant in question; otherwise additional analysis will be required to demonstrate the fuel assembly structural integrity.

CP&L Response

An evaluation of VANTAGE 5 fuel assembly structural integrity considering the lateral effects of a LOCA and a seismic accident has been performed. The safe shutdown earthquake and LOCA comparative analyses indicated that the flow mixers will share some grid load among the structural grids. The grid load comparison study results show that the VANTAGE 5 fuel assembly has more margin in withstanding the faulted condition transient load than the LOPAR fuel assembly.

Additional analyses have been performed to demonstrate fuel assembly structural integrity. Since the VANTAGE 5 fuel has been shown to have more margin than the LOPAR fuel used in previous cycles, the evaluation of the VANTAGE 5 fuel assembly in accordance with NRC requirements as given in SRP 4.2, Appendix A, shows that the VANTAGE 5 fuel is structurally acceptable for an all VANTAGE 5 core, i.e., the grids will not buckle due to combined impact forces of a seismic/LOCA event. The same conclusion is true for a transition core composed of both VANTAGE 5 and LOPAR assemblies. Thus, the core coolable geometry is maintained. The stresses in the fuel assembly components resulting from seismic and LOCA induced deflections are well within acceptable limits. The reactor can be safely shutdown under the combined faulted condition loads.

SER Restriction (4): For those plants using the ITDP, the restrictions enumerated in Section 4.1 of this report must be addressed and information regarding measurement uncertainties must be provided.

CP&L Response

The ITDP method described in WCAP-8567 has been approved for use in licensing applications subject to the certain restrictions. One of the restrictions requires that if the sensitivity factors are changed as a result of a DNB correlation change, then the use of an uncertainty allowance for application of Equation 3-2 (WCAP-8567) must be reevaluated and the linearity assumption of WCAP-8567 must be validated.

In response to an NRC staff question on WCAP-10444[1], Westinghouse performed the required reevaluation and validation using the same methods described in the staff safety evaluation report for WCAP-8567. This was found acceptable as documented in Section 4.1 of the NRC SER for VANTAGE 5.

Another restriction states that those plants using ITDP provide plant specific design DNBR limits and provide measurement uncertainties for pressurizer pressure, power, coolant flow rate and temperature. This is provided in WCAP-12340 "Westinghouse Improved Thermal Design Procedure Instrumentation Uncertainty Methodology for Carolina Power & Light Company Shearon Harris Nuclear Power Station."

In addition, licensees referencing WCAP-10444 should incorporate in the bases of their plant Technical Specifications the plant-specific safety analysis DNBR limit, the DNBR allowance and the amount of allowance that has been used.

The information was not included in the Technical Specfication bases to preclude making Technical Specification changes on a reload by reload basis if margin was allocated differently. This information was provided in Attachment 1, Section 5.0 of the licensing submittal and identifies margin being allocated to transition core penalties, rod bow penalty, and additional margin reserved for design flexibility.

Recent NRC and utility initiatives, such as the Core Operating Limit Report (COLR) and the MERITS program have strived to simplify the Technical Specifications and reduce unnecessary burdens on the NRC and utilities. Providing the specific information described for the safety analysis DNBR limit in the bases would be contrary to these trends in light of the changing nature of this value during the transition to a full core of VANTAGE 5 fuel. Note that the bases pages for MERITS do not provide this level of detail.

1 [Question 3 in Westinghouse Letter, E. P. Rahe, Jr., to C. O. Thomas (NRC), "Response to Request Number 1 for Additional Information on WCAP-10444 entitled, VANTAGE-5 Fuel Assembly" (Proprietary), NS-NRC-85-3014, dated March 1, 1985].

The information requested is currently supplied by Westinghouse to CP&L on a cycle-by-cycle basis and is maintained in Chapter 4 of the Shearon Harris FSAR. Changes to these values are controlled by performing a 10CFR50.59 safety evaluation.

SER Restriction (10): If a positive MTC is intended for VANTAGE 5, the same positive MTC consistent with the plant Technical Specifications should be used in the plant safety analysis.

CP&L Response

The same positive MTC from the Cycle 1 Technical Specifications was used in the plant specific safety analysis for the VANTAGE 5 fuel.

NRC QUESTION 2

The Licensee should modify the Technical Specification regarding COLR as noted.

CP&L Response

Revised Technical Specification pages reflecting administrative changes are included in Attachment 2.

NRC QUESTION 3

Why use 2785 MWT for power level instead of 2775 MWT used in other analyses (see p. 25, Attachment 1 SGTR)?

CP&L Response

As shown in Attachment 3 (see Footnote 2), Table 15.1-1, 2785 MW represents the NSSS thermal power, and 2775 MW is the power generated in the core. The difference between these values is the thermal power transferred to the primary fluid by the reactor coolant pumps. LOFTRAN requires an input for both NSSS power and pump power. For DNB analyses, the pump power is subtracted from the NSSS power such that the calculated heat flux on the rods reflects core power only (2775 MW). Similarly, FSAR Chapter 15 events which progress very rapidly (Rod Ejection), or that develop after power to the pumps is lost (SBLOCA), ignore pump heat and use 2775 MW for the NSSS power.

Attachment 3 (see Footnote 2), Table 15.1-3 summarizes the NSSS power assumed for each event. For the non-ITDP analyses, the NSSS powers shown in this table are increased by 2% to account for calorimetric measurement uncertainty. This additional 2% power is assumed to account for uncertainty only, and does not represent available margin for power uprating.

2 See CP&L Cycle 3 Reload submittal dated April 17, 1989, NLS-89-087.

NRC QUESTION 4

The Licensee should address the conformance with restrictions specified in the NRC's SER on LOFTR2 (WCAP-11704). (See Section 15.6.3.3 of Attachment 3)

CP&L Response

Conformance with the restrictions specified in the NRC safety evaluation report on LOFTTR2 are provided below. Conformance with SER Restrictions 1, 3, 4, and 5 for the Cycle 3 SGTR analysis remained unchanged from the Cycle 2 conformance which was documented in a CP&L letter dated February 1, 1988.

It should be noted that CP&L is currently performing additional SGTR analyses based on revised operator action times as documented in a CP&L letter dated May 19, 1989. This effort is ongoing and independent of the Cycle 3 SGTR analysis.

SER Restriction (1): Each utility in the SGTR subgroup must confirm that they have in place simulators and training programs which provide the required assurance that the necessary actions and times can be taken consistent with those assumed for the WCAP-10698 design basis analysis. Demonstration runs should be performed to show that the accident can be mitigated within a period of time compatible with overfill prevention using design basis assumptions regarding available equipment and to demonstrate that the operator action times assumed in the analysis are realistic.

CP&L Response

SHNPP has in place a plant-specific simulator and operator training program. Both the classroom and simulator training include an SGTR as an event for which the operators are trained to respond. This training includes emphasis on the necessary operator actions and the time constraints. Simulator and classroom training materials will be reviewed for changes that may be required by WCAP-11703.

To demonstrate that the operator action times assumed in the analysis are realistic, several SGTR simulator runs were conducted during annual operator requalification training in the fall of 1987. The crews being trained on the simulator were not aware of the type of event to expect nor that their actions were being timed to validate an analysis. The simulator was programmed using the conservative assumptions of the SGTR analysis: loss of off-site power and a stuck control rod both occurring at the time of reactor trip, along with lack of pressurizer control and failure of an intact steam generator PORV to open. A total of three separate crews were timed under these conditions with the results demonstrating that the times assumed in the analysis are realistic.

SER Restriction (2): A site-specific SGTR radiation off-site consequence analysis which assumes the most severe failure identified in WCAP-10698, Supplement 1. The analysis should be performed using the methodology in SRP, Section 15.6.3, as supplemented by the guidance previously provided by the NRC in their Safety Evaluation Report on WCAP-10698, Supplement 1.

CP&L Response

- Section 15.6.3.4 of Attachment 4 (see Footnote 2) is a site-specific SGTR off-site radiation dose analysis which assumes the most severe failure identified in WCAP-10698, Supplement 1. The analysis was performed consistent with the methodology in SRP, Section 15.6.3 as supplemented by the guidance previously provided by the NRC in their Safety Evaluation Report on WCAP-10698, Supplement 1.

SER Restriction (3): An evaluation of the structural adequacy of the main steam lines and associated supports under water-filled conditions as a result of SGTR overfill.

CP&L Response

The SHNPP specific SGTR analysis demonstrates that the steam generators do not overfill such that no water will accumulate in the main steam lines; however, as required by the NRC, stress analysis has been performed on the main steam lines to confirm their structural adequacy under water-filled conditions. This analysis was performed using an approved Ebasco stress analysis program. The steam lines were assumed to be full of water from the steam generator nozzles to the MSIVs under otherwise normal operating conditions. The results of the analysis show that the pipe stress, the steam generator nozzle loads, the containment penetration loads, and hanger loads remain within code allowables with the piping full of water; therefore, structural adequacy of the main steam lines and associated supports is assured.

SER Restriction (4): A list of systems, components, and instrumentation which are credited for accident mitigation in the plant-specific SGTR EOPs. Specify whether each system and component specified is safety grade. For primary and secondary PORVs and control valves, specify the valve motive power and state whether the motive power and valve controls are safety grade. For nonsafety grade systems and components, state whether safety grade backups are available which can be expected to function or provide the desired information within a time period compatible with prevention of SGTR overfill or justify that nonsafety grade components can be utilized for the design basis event. Provide a list of all radiation monitors that could be utilized for identification of the accident and the ruptured steam generator and specify the quality and reliability of this instrumentation if possible. If the EOPs specify steam generator sampling as a means of ruptured SG identification, provide the expected time period for obtaining the sample results and discuss the effect on the duration of the accident.

CP&L Response

1. A listing of the systems, components and instrumentation which are credited for mitigating an SGTR event utilizing Harris Nuclear Plant's Emergency Operating Procedures are provided below. Motive power for PORVs, control valves, or other valves that may need to be operated during the event are provided in parenthesis. The systems/equipment listed below are required to function for an SGTR and are safety related.

A. Systems

Charging/Safety Injection System
AFW
Main Steam up to and including the MSIVs
Reactor Protection System
RVLIS

B. Components

AFW Flow Control Valves (Electro-Hydraulic Operator)
Motor-Driven AFW Pumps
Turbine-Driven AFW Pump
Main Steam PORVs (Electro-Hydraulic Operator)
Main Steam Isolation Valves
MSIV Bypass Valves
MS Isolation Valves to AFW Turbine-Driven Pump (Motor Operator)
SG Blowdown Valves
Turbine-Driven AFW Pump Isolation Valves to SG
SI Reset Control Switch
Emergency Diesel Generators
CSIP Isolation Valves (Motor Operator)
BIT Isolation Valves (Motor Operator)
Emergency Diesel Generator Control Switches
Emergency Bus Voltmeter Phase Selector Control Switches
Service Transformer Breaker Control Switches

C. Instrumentation

Steam Generator Level Instrumentation
Steam Generator Pressure Indication
Motor-Driven AFW Pump Status Indication
Turbine-Driven AFW Pump Status Indication
MS PORV Position Indication
MSIV Position Indication
MSIV Bypass Valve Position Indication
SG Blowdown Isolation Valve Position Indication
Position Indication for Steam Supply Valves to Turbine-Driven AFW Pump
Position Indication for Turbine-Driven AFW Pump Isolation Valves to SG
AFW Flow Indication
AFW Flow Control Valve Indication
RCS Temperature Indication
Emergency Bus Voltage Indication
Emergency Diesel Generator Indication
Neutron Flux Monitoring System Indication

2. The following systems/equipment may also be utilized or monitored during the SGTR event, but they would not need to function to mitigate the event. Those items which are safety related are designated below in parenthesis.

A. Systems

Instrument Air System

B. Equipment and/or Associated Instrumentation

Containment Phase A and B Reset Switch (Safety Related)
P11 (Low Steamline Pressure - SI Control Block) (Safety Related)
RHR Pumps (Safety Related)
Steam Dump Valves
Normal Pressurizer Spray Valves
Pressurizer PORVs
Pressurizer Auxiliary Spray Valves
Charging Flow Control Valve
Pressurizer Pressure Indication
Pressurizer Level Indication

3. The following radiation monitors may be used to assist in identifying an SGTR event:

A. Main Steam Line Radiation Monitors

Safety Related, Seismic Category I
Electrical Class IE

(The detectors are specified as being able to detect the state of isotopic concentration within one hour with an error no greater than ±5 percent of the net count rate with a confidence level equal to or greater than 95.5 percent. Nominal performance at setpoints shall have smaller than an error of ±2.5 percent at one standard deviation in one hour.)

B. SG Blowdown Monitor

Not Nuclear Safety Related

(Determination of counting time length shall be done through the use of a programmed algorithm which shall ensure statistical significance at a 95 percent confidence level with a maximum error of ±2.5 percent for count rates between 10^2 and 10^5 cpm.)

C. Condenser Vacuum Pump Effluent Monitor

Not Nuclear Safety Related

(The confidence level shall be 95 percent for the minimum detectable concentration with a maximum error of ±2.5 percent of net count for counts ranging between 10^2 and 10^5 cpm.)

4. The Harris Nuclear Plant SGTR EOP contains a step that directs sampling of the steam generators to check activity. The plant Chemistry Department estimates a two-hour time period from being requested to take a sample until the results could be reported. The estimate included travel to and from the sampling panel, sample line

flushing, and count time. As sampling of the steam generator fluid is not the primary method for determining an SGTR event, it is expected that the sampling time span would not impact accident duration.

<u>SER Restriction (5)</u>: A survey of plant primary and "balance of plant" systems design to determine the compatibility with the bounding plant analysis in WCAP-10698. Major design differences should be noted. The worst single failure should be identified if different from the WCAP-10698 analysis and the effect of the differences on the margin to overfill should be provided.

<u>CP&L Response</u>

A comparison of the SHNPP to the bounding plant analysis including the worst single failure in WCAP-10698 is provided in Sections II A and B of WCAP-11703, "LOFFTTR2 Analysis for a Steam Generator Tube Rupture: Shearon Harris Nuclear Power Plant".