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

Subject: Catawba Nuclear Station
Docket Nos. 50-413 and 50-414
Technical Specification Amendment Supplement
Unit 2 Cycle 6 Reload

In a letter to the NRC dated December 15, 1992, Catawba Nuclear Station submitted the Technical Specification changes that would be necessary for the operation of Catawba Unit 2 Cycle 6. Subsequent discussions with the NRC Staff have indicated that the original submittal needed to be supplemented with the following:

1. A revised no significant hazards analysis which replaces the original no significant hazards analysis in its entirety (Attachment I).

2. A mark-up of Technical Specification 2 1.1. (Attachment II).

3. A revised Table 8-1 on page 8-3 of the reload report that changes decreased $F_{\Delta H}$ to increased $F_{\Delta H}$ (Attachment II).

In addition, the original submittal used 3.3.3.12 as the technical specification number for the Boron Dilution Mitigation System. Since that time, Amendments 103/97 changed the technical specification number for the Boron Dilution Mitigation System to 3.3.3.11. Accordingly, any reference to Technical Specification 3.3.3.12 in the original submittal should be changed to 3.3.3.11.

The original submittal also contained an administrative change to Technical Specification 6.9.1.9 "Core Operating Limits Report" which added a recently approved topical (DPC-NE-1004A) to those already referenced. Although none of the current changes are based on this topical, the addition of this topical is necessary to allow future reloads to be carried out under 10 CFR 50.59. If the topical was not added at this time, a future technical specification change would be necessary for the sole purpose of adding this topical.

A001 1/1 Add: NRR Chatterton

If there are any further questions or comments, please contact Chuck Lewis at (803) 831-3076.

Very truly yours,

M. S. Tuckman

M. S. Tuckman

Attachments

CRL/C2C6SUPP.1

M. S. Tuckman, being duly sworn, states that he is Vice President, Catawba Nuclear Station; that he is authorized on the part of said company to sign and file with the Nuclear Regulatory Commission this revision to the Catawba Nuclear Station Technical Specifications, Appendix A to License Nos. NPF-35 and NPF-52; and that all statements and matters set forth therein are true and correct to the best of his knowledge.

M. S. Tuckman

M. S. Tuckman

Subscribed and sworn to before me this 5th day of Feb, 1993.

Yvonne H Jackson

Notary Public

My Commission Expires:

Nov 21, 2000

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SREC
Group File: CN-801.01
Master File (801.01)

ATTACHMENT I

Revised No Significant Hazards Analysis

NO SIGNIFICANT HAZARDS ANALYSIS

The following analysis, required by 10 CFR 50.91, concludes that the proposed amendment will not involve significant hazards consideration as defined by 10 CFR 50.92.

10 CFR 50.92 states that a proposed amendment involves no significant hazards consideration if operation in accordance with the proposed amendment would not:

1) Involve a significant increase in the probability or consequences of an accident previously evaluated; or

2) Create the possibility of a new or different kind of accident from any previously evaluated; or

3) Involve a significant reduction in the margin of safety.

POWER DISTRIBUTION AND SAFETY LIMITS

Catawba Unit 1 Cycle 6 was the first Duke Power Nuclear Station for which B&W Fuel Company (BWFC) supplied the reload fuel. The Catawba Unit 1, Cycle 6 Reload Report presented an evaluation that concluded the core reload using Mark-BW fuel would not adversely impact the safety of the plant. The Catawba Unit 1, Cycle 7 report was similar, but reflected that Duke Power performed the analyses in support of the operation of Cycle 7 rather than BWFC. This reload for Catawba Unit 2, Cycle 6 is a compilation of the changes made for Unit 1 during Cycles 6 and 7 in that it justifies the use of Mark-BW fuel using Duke Power analysis.

The Catawba Unit 2, Cycle 6 Reload Safety Evaluation Report presents an evaluation which demonstrates that the core reload using Mark-BW fuel will not adversely impact the safety of the plant. During Cycle 6, the core will contain 76 fresh fuel assemblies supplied by B&W and 117 Westinghouse supplied Optimized Fuel Assemblies (OFA).

The changes to the Safety Limit and Power Distribution Technical Specifications presented in Section 8 of the Reload Report represent the application of previously approved methodology to Catawba Unit 2. The changes to remove the power range neutron flux negative rate reactor trip, increase the low steam line pressure setpoint, increase feedwater isolation response time, increase steam line isolation response time, increase pressurizer safety valve lift setpoint tolerance, remove steam line pressure dynamic compensation, increase pressurizer safety valve lift setpoint tolerance, and increase main steam line isolation valve stroke time reflect the use of Duke analysis, and have already been approved for Catawba Unit 1. The changes described above include the deletion of references to specific units on individual Technical Specification pages, and delete pages which were

previously for Unit 2 only. The implementation of unit specific references became necessary due to the transition from Westinghouse to B&W supplied fuel during Unit 1 Cycle 6 and for the Unit 1 Cycle 7 Reload due to the transition to Duke analysis methodology. The analysis which made the changes necessary in the Unit 1 reload submittal is generic, and as described in the technical justification, is equally applicable to both McGuire and Catawba units.

A LOCA evaluation for operation of Catawba Nuclear Station with Mark-BW fuel has been completed (BAW 10174, Mark-BW Reload LOCA Analysis for the Catawba and McGuire Units). Operation of the station while in transition from Westinghouse supplied OFA fuel to B&W supplied Mark-BW fuel is also justified in this topical.

BAW-10174 demonstrates that Catawba Nuclear Station continues to meet the criteria of 10 CFR 50.46 when operated with Mark-BW fuel. Large Break LOCA calculations completed consistent with an approved evaluation model (BAW-10168P and revisions) demonstrate compliance with 10 CFR 50.46 for breaks up to and including the double ended severance of the largest primary coolant pipe. The small break LOCA calculations used to license the plant during previous fuel cycles are shown to be bounding with respect to the new fuel design. This demonstrates that the plant meets 10 CFR 50.46 criteria when the core is loaded with Mark-BW fuel.

During the transition from Westinghouse OFA fuel to Mark-BW fuel, both types of fuel assemblies will reside in the core for several fuel cycles. Appendix A to BAW-10174 demonstrates that results presented above apply to the Mark-BW fuel in the transition core, and that insertion of the Mark-BW fuel will not have an adverse impact on the cooling of the Westinghouse fuel assemblies.

Duke Power Company's Topical Reports DPC-NE-3000, DPC-NE-3001-PA, and DPC-NE-2004-PA provide evaluations and analyses for non-LOCA transients which are applicable to Catawba. The scope of these analyses includes all events specified by sections 15.1-15.6 of Regulatory Guide 1.70 (Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants) and presented in the Final Safety Analysis Report for Catawba. The analysis and evaluations performed for these topicals confirm that operation of Catawba Nuclear Station for reload cycles with Mark-BW fuel will continue to be within the previously reviewed and licensed safety limits.

One of the primary objectives of the Mark-BW replacement fuel is compatibility with the resident Westinghouse fuel assemblies. The description of the Mark-BW fuel design and the thermal-hydraulics and the core physics performance evaluation demonstrate the similarity between the reload fuel and the resident fuel. The extensive testing and analysis summarized in BAW-10173P shows that the Mark-BW fuel design performs, from the standpoint of neutronics and thermal-hydraulics, within the bounds and limiting design criteria applied to the resident Westinghouse fuel for the Catawba plant safety analysis.

Each FSAR accident has been reviewed to determine the effects of Cycle 6 operation and to ensure that the radiological consequences of postulated accidents are within applicable regulatory guidelines, and do not adversely affect the health and safety of the public. The design basis LOCA evaluations assessed the radiological impact of differences between the Mark-BW fuel and Westinghouse OFA fuel fission product core inventories. Also, the dose calculation effects from non-LOCA transients reanalyzed by Duke Power were evaluated using Cycle 6 characteristics. The calculated radiological consequences are all within specified regulatory guidelines and contain significant levels of margin.

The analyses contained in the referenced Topical Reports indicate that the existing design criteria will continue to be met. Therefore, the enclosed TS changes will not increase the probability or consequences of an accident previously evaluated.

As stated in the above discussion, normal operational conditions and all fuel-related transients have been evaluated for the use of Mark-BW fuel at Catawba Nuclear Station. Testing and analysis was also completed to ensure that, from the standpoint of neutronics and thermal-hydraulics, the Mark-BW fuel would perform within the limiting design criteria. Because the Mark-BW fuel performs within the previously licensed safety limits, the possibility of a new or different accident from any previously evaluated is not created.

The reload-related changes to the TSs do not involve a significant reduction in the margin of safety. The calculations and evaluations documented in BAW-10174 show that Catawba will continue to meet the criteria of 10 CFR 50.46 when operated with Mark-BW fuel. The evaluation of non-LOCA transients documented in DPC-NE-3001 also confirms that Catawba will continue to operate within previously reviewed and licensed safety limits. Because of this, the TS changes to support the use of Mark-BW fuel will not involve a significant reduction in the margin of safety.

The technical changes made to Table 2.2-1 reflect the use of the BWCMV CHF correlation and Duke Power's Statistical Core Design methodology with a 1.55 thermal design limit. These changes to Table 2.2-1 will not significantly increase the probability or consequences of an accident previously evaluated. the changes to the K values conservatively bound the allowable operating region, as defined by the new DNBR methodology. It can be concluded that these changes will not create the possibility of any new accident from those previously evaluated. It can also be concluded that since all new TS values are bounded by safety analysis assumptions that this change will not significantly decrease the margin of safety.

DELETION OF NEUTRON FLUX HIGH NEGATIVE RATE TRIP

The removal of the Power Range Neutron Flux High Negative Rate trip will not result in any previously-reviewed accident becoming more probable or more severe. The trip is a response to a pre-existing transient condition and would not initiate any accident. The trip

is designed to provide protection from a dropped control rod. However, in the event of a dropped rod, the reactor is assumed to trip on low pressurizer pressure. Therefore, the protection function is retained. The consequences of a dropped rod have been analyzed and found to be within acceptable limits.

Likewise, the removal of this trip will not create a new accident not previously reviewed. The removal of a response to a transient will not initiate a new transient. There are no credible unanalyzed transients which will occur as a result of a dropped rod. The removal of this trip will reduce the potential for spurious or unnecessary trips which may occur as a result of maintenance or the drop of a low-worth rod. There are no other hardware modifications or procedure changes that will be made as a result of this deletion which could create the possibility of a new accident.

No margin of safety will be reduced by this change. As noted above, if a dropped rod necessitates a trip, the trip function will be accomplished as a result of low pressurizer pressure. For those dropped rods for which no trip is necessary, the removal of this trip will provide protection against an unnecessary transient.

LOW STEAM LINE SETPOINT PRESSURE CHANGE

Changing the Low Steam Line Pressure setpoint and removal of dynamic compensation will not increase the probability or consequences of any previously-reviewed accident. The higher steam line pressure setpoint is consistent with all licensing basis safety analyses. This change, in conjunction with the removal of the dynamic compensation of the steam pressure signal, is intended to reduce or eliminate spurious Engineered Safeguards Features (ESF) actuations which are caused by minor (but rapid) pressure decreases in the secondary system.

The proposed amendment will not result in a new accident not previously reviewed. A change in steam line pressure is a response to an existing transient condition, rather than a precursor or initiating event. A change in the steam line pressure setpoint is also not a precursor or initiating event.

The proposed amendment will not result in a significant decrease in a margin of safety. The reanalysis of the steam line break accident which was performed shows that all imposed Condition II acceptance criteria are met.

Based on the above, it is concluded that no significant hazards exist.

FEEDWATER AND MAIN STEAM LINE ISOLATION VALVE STROKE TIME

The proposed changes to the valve stroke times in Tables 3.3-5 and Table 3.6-2a will not significantly increase the probability or consequences of any previously evaluated accident. The effects of the delays in isolation times on the various transients affected have been analyzed and found to be acceptable. Since these valves do not receive a containment isolation signal, and no credit is taken for operation of these valves in the dose analysis for a containment isolation function, a maximum stroke time does not apply for containment isolation.

The proposed changes will not significantly increase the possibility of a new accident not previously evaluated. Feedwater and main steam isolation are responses to ongoing transients, rather than initiators or precursors of transients. No equipment or component reconfiguration will occur as a result of this change.

The proposed changes will not significantly decrease any margin of safety. As noted above, the effects of the longer isolation times have been evaluated and found to be acceptable.

Based on the above, it is concluded that no significant hazards exist.

INCREASE IN PRESSURIZER CODE SAFETY VALVE SETPOINT TOLERANCES

The proposed amendment will not result in a significant increase in the probability or consequences of any previously analyzed accident. The valve lift setting is challenged only after a transient has been initiated and is not a contributor to the probability of any transient or accident. The transients which involve pressure increases which would potentially challenge the safety valves have been analyzed to determine the consequences of delayed or premature valve actuation at the extremes of the new setpoint tolerances. These analyses show that all applicable acceptance criteria are met using the wider tolerances.

The proposed amendment will not result in the creation of any new accident not previously evaluated. As noted above, the setpoint tolerance only affects the time at which the safety valve opens following or during a transient, and is not a contributor to the probability of an accident.

The proposed amendment will not result in a significant decrease in a margin of safety. The limiting transient in each accident category has been analyzed to determine the effect

of the change in lift setpoint tolerance on the transient. In each case, the results of the analyses met all acceptance criteria.

Based on the above, it is concluded that no significant hazards exist.

CONTAINMENT ISOLATION VALVES

The proposed changes to the valve stroke times in Table 3.6-2a and 3.6.2b will not significantly increase the probability or consequences of any previously evaluated accident. The effects of the delays in isolation times on the various transients affected have been analyzed and found to be acceptable. Since these valves do not receive a containment isolation signal, and no credit is taken for operation of these valves in the dose analysis for a containment isolation function, a maximum stroke time does not apply for containment isolation.

The proposed changes will not significantly increase the possibility of a new accident not previously evaluated. Feedwater and main steam isolation are responses to ongoing transients, rather than initiators or precursors of transients. No equipment or component reconfiguration will occur as a result of this change.

The proposed changes will not significantly decrease any margin of safety. The isolation times which are applicable to these valves are specified in Table 3.3-5, Engineered Safety Features Response Times. The effects of the isolation of these valves was evaluated based on their ESF function, not a containment isolation function, and determined to be acceptable, therefore there is no significant decrease in the margin of safety.

BORON DILUTION MITIGATION SYSTEM

TS 3.3.3.11.a.2 is changed to reduce the allowable Reactor Makeup Water Pump flow in Mode 5 from 75 gpm to 70 gpm. In the event that the Boron Dilution Mitigation System (BDMS) is inoperable the Reactor Makeup Water Pump flowrates are limited to ensure that operator action times required to terminate a dilution event can be met. The limits on reactor makeup water pump flowrates when the BDMS is inoperable are verified each cycle to ensure that the safety analysis assumptions for these parameters remain valid. When the calculated Reactor Makeup Water Pump flowrate is found to be less than the existing flowrate limits, the flowrate limit must be reduced so that the operator action time acceptance criteria of Standard Review Plan 15.4.6 can be met.

Reducing the allowable Reactor Makeup Water Pump flow in Mode 5 does not involve a significant increase in the probability or consequences of an accident previously evaluated. The current TS flowrate does not allow enough time for the operator to terminate an

uncontrolled dilution event when required operator response times are assumed. The lower flowrate allows needed operator response times and is therefore more conservative.

Reducing the allowable Reactor Makeup Water Pump flow in Mode 5 does not change the way that any plant equipment is operated or maintained, therefore it does not create the possibility of a new or different accident.

Reducing the Allowable Reactor Makeup Water Pump Flow in Mode 5 will not involve a significant reduction in the margin of safety. This flowrate is more conservative, and ensures that safety analysis assumptions regarding operator actions times in response to the termination of an uncontrolled dilution event can be met.

CORE OPERATING LIMITS REPORT

The proposed change to TS 6.9.1.9 adds approved topical DPC-NE-1004A to the list of analytical methods used to determine core operating limits. This change is administrative, adding a topical report which has been approved for use on Catawba to the list of analytical methods used to determine core operating limits. Since this change is administrative it has been determined that no significant hazards are involved.

ENVIRONMENTAL IMPACT STATEMENT

The proposed Technical Specification change has been reviewed against the criteria of 10 CFR 51.22 for environmental considerations. As shown above, the proposed change does not involve any significant hazards consideration, nor increase the types and amounts of effluents that may be released offsite, nor increase the individual or cumulative occupational radiation exposures. Based on this, the proposed Technical Specification change meets the criteria given in 10 CFR 51.22 (c) (9) for categorical exclusion from the requirement for an Environmental Impact Statement.