UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 101 TO FACILITY OPERATING LICENSE NPF-35

AND AMENDMENT NO. 95 TO FACILITY OPERATING LICENSE NPF-52

DUKE POWER COMPANY, ET AL.

CATAWBA NUCLEAR STATION, UNITS 1 AND 2

DOCKET NOS. 50-413 AND 50-414

1.0 INTRODUCTION

By letter dated April 13, 1992 (Reference 1), as supplemented July 8 and August 26, 1992, Duke Power Company, et al. (DPC or the licensee) submitted amendments to change the Catawba Nuclear Station, Units 1 and 2 Technical Specifications (TSs). The amendments consist of changes to the TSs for Catawba Unit 1 Cycle 7 reload. The April 13, 1992, submittal also contains changes to the Core Operating Limits Report (COLR), markups of the appropriate Final Safety Analysis Report (FSAR) chapters, and design information relative to Cycle 7 reload. The Catawba Unit 1 plant recently completed operating in Cycle 6 with a complete batch (1/3 core) of B&W Mark-BW 17x17 fuel design. Catawba Unit 1 Cycle 7 will include a second complete batch of B&W Mark-BW 17x17 fuel, resulting in a core that is 2/3 loaded with B&W Mark-BW 17x17 fuel. The use of Mark-BW fuel design in Catawba and the McGuire plants has been previously approved by the NRC via the topical reports BAW-10173-P-A, Revision 2 and BAW-10174-A, Revision 1, which were prepared by the B&W Fuel Company utilizing B&W's methods (References 2 and 3). The August 26, 1992, letter provided clarifying information that did not change the initial no significant hazards consideration determination.

The McGuire Cycle 8 was the first time DPC had performed reload safety analysis for its Westinghouse Units. The reload design and all the analysis for normal and off-normal operations will be carried out inhouse by DPC. The methods and analytical models used by DPC for Catawba Unit 1 Cycle 7 fuel assembly mechanical design, nuclear design, thermal-hydraulic analyses, and non-LOCA safety analyses have been approved by the NRC (References 4 to 7).

2.0 STAFF EVALUATION

2.1 Mechanical Design

The Cycle 7 reload will be the second time that the Mark-BW fuel will be used in Catawba Unit 1. This fuel is similar to the Westinghouse standard assembly design. The core consists of 72 fresh Mark-BW fuel assemblies, 49 Westinghouse optimized fuel assemblies and 72 Mark-BW fuel assemblies, for a total core loading of 193 Fuel Assemblies (FA). Out of 193 FAs, 121 are burned FAs. The unique features of the Mark-BW 17x17 design include the Zircaloy intermediate spacer grids, the spacer grid restraint system, and the use of Zircaloy grids with standard lattice design.

The mechanical analyses and thermal performance for the Mark-BW 17x17 design were performed by DPC with the methodology described in the approved topical report DPC-NE-2001-P-A, Revision 1 (Reference 8); and therefore, are acceptable.

3.0 FUEL SYSTEM DESIGN

3.1 Fuel Management

A general description of the Cycle 7 core is given in section 5.0. The Cycle 7 core uses a low-leakage fuel management scheme where previously burned assemblies are placed on the periphery and most of the fresh assemblies are located throughout the core interior in a pattern which minimizes power peaking. With this loading and a Cycle 6 endpoint of 350 effective full-power days (EFPD), the Cycle 7 reactive lifetime for full power operation is expected to be 350 EFPD. A comparison of Cycle 7 nominal characteristic physics parameters with those used in the safety analyses show that the latter are conservative in all cases.

3.2 Nuclear Design

The core physics parameters for Cycle 7 were generated by DPC with the PDQ07 and EPRI-NODE-P computer codes using the methodology described in the approved topical report DPC-2010-A (Reference 9). The Reactor Protection System setpoint limits and technical specification operating limits for the core were verified through analysis of the Cycle 7 nuclear design using methodology described in the approved topical report DPC-NE-2011-P-A (Reference 10).

3.3 Control Requirement

The value of the required shutdown margin varies throughout core life with the most restrictive value occurring at end of cycle and at hot zero power conditions. Sufficient boration capability and net available control element assembly (CEA) worth, including a minimum worth stuck CEA and appropriate calculation uncertainties, exist to meet all the shutdown margin requirements. These results were derived by approved methods and incorporate appropriate assumptions and are, therefore, acceptable.

4.0 THERMAL-HYDRAULIC DESIGN

The thermal performance of Cycle 7 fuel was analyzed using the NRC-approved methodology (Reference 8). The analyses included the power and burnup levels representative of the peak pin at each burnup interval, from the beginning of the cycle to the end of the cycle burnups. Based on this analysis, the internal pressure in the most limiting fuel rod will stay below the nominal reactor coolant system (RCS) pressure throughout the cycle. Because this satisfies Standard Review Plan (SRP) Section 4.2 criteria, the thermal design of the Cycle 7 core is acceptable.

The thermal-hydraulic analysis supporting Cycle 7 operation was performed by DPC with VIPRE-01 computer code and approved statistical core design (SCD) methodology (Reference 4). The SCD methodology is a technique that statistically combines uncertainties associated with the core statepoint parameters, code/model, and Critical Heat Flux (CHF) correlation to determine a statistical departure from nucleate boiling ratio (DNBR) limit. The uncertainties used in Reference 4 bound the uncertainties specifically calculated for Catawba Unit 1. The statistical DNBR limit for use with the BWCMV CHF correlation (Reference 11) in VIPRE-01 is determined to be 1.40. To provide design flexibility, a 10.7% margin is added to the statistical DNBR limit to yield a design DNBR limit of 1.55 for the generic Mark-BW and the Catawba Unit 1 Cycle 7 analyses. The reactor core safety limits for Catawba Unit 1 Cycle 7 were generated utilizing the BWCMV CHF correlation and the SCD methodology for a full core of Mark-BW assemblies and a radial enthalpy rise hot channel factor of 1.50.

The hydraulic compatibility of the Mark-BW fuel and the Westinghouse Optimized Fuel Assemblies (OFA) had been addressed in the approved topical report BAW-10173-P-A Revision 2 (Reference 2). The results of the hydraulic compatibility test indicated that the total pressure drop across the Mark-BW Fuel is 2.4% lower than the total pressure drop across the OFA fuel. The licensee approach to addressing the transition core penalty is presented in detail in Reference 3. The licensee determined a generic transition core penalty by modeling a conservative core configuration with one OFA assembly as the hot assembly located in a Mark-BW core. Bounding power shapes during normal and accident conditions were analyzed yielding a maximum DNBR penalty of 3.8% for OFA fuel. The licensee addressed the transition core penalty for OFA fuel by applying the 3.8% DNBR penalty against the 10.7% generic margin included in the design DNBR limit.

5.0 ACCIDENT ANALYSES

5.1 Non-LOCA Safety Analysis

The design basis events considered in the safety analyses are categorized into two groups: anticipated operational occurrences and postulated accidents (limiting faults). All events were reviewed by the licensee to account for the differences in the core physics parameters of the Mark-BW fuel and the changes to the Technical Specifications. The scope of the events considered is consistent with that addressed in the FSAR for Catawba. The evaluations considered the effects of mixed (transition) cores using Westinghouse and Mark-BW fuel.

The methods and results for the analyses of the Steam System Piping Failure, Rod Ejection and Dropped RCCA/RCCA bank transients, are documented in DPC topical report DPC-NE-3001 and follow-up correspondence from DPC which have been reviewed and found acceptable by the staff, (References 5 and 6). Analytical models and methodology for the statistically misaligned rod accident are provided in approved topical reports, (References 4, 7 and 9). For the generic and single events, a system thermal-hydraulic analysis was performed which bound both Catawba Units 1 and 2, and McGuire Units 1 and 2. Since a single set of generic analyses has been performed for these events, the results for Catawba are identical to those submitted in the approved

McGuire 2 Cycle 8 reload report, (Reference 12). The Catawba 1 Cycle 7 reload core physics parameters were reviewed with respect to the assumptions used in these analyses. The analysis and methodology for these events have been reviewed and found acceptable by the staff (References 5 and 9).

For the remaining FSAR thermal-hydraulic accident analyses sensitive to reload core physics parameters (e.g., LOCA), the current approved licensing bases are being retained. These bases have been reviewed and found acceptable by the staff (Reference 7). In addition, the post-LOCA subcriticality evaluation and the boron precipitation evaluation have been performed by DPC in the FSAR for Catawba (Reference 12). The Catawba Unit 1 Cycle 7 parameters have been reviewed with respect to the assumptions used in the subcriticality analysis.

The radiological consequences for the locked rotor, single rod withdrawal and rod ejection events, were reanalyzed due to differences between the Mark-BW fuel and the OFA fuel. The results, presented in section 8 of Reference 1, were reviewed and found acceptable by the staff as discussed below. Review by the staff of Catawba Unit 1 Cycle 7 reload parameters were found to be bounded by the accident analysis assumptions for all accidents which are sensitive to core physics parameters, and are therefore, acceptable.

5.2 <u>LOCA Analyses</u>

The LOCA analyses for Catawba Unit 1 transition cores with mixed Mark-BW and Westinghouse OFA assemblies and future cores with all Mark-BW fuel have been reviewed previously by the NRC (BAW-10174-A) and found acceptable.

5.3 <u>Radiological Evaluation</u>

In its analysis, the licensee determined that the radiological impact of two accidents, the rod ejection and the locked rotor events, could be impacted as a result of changes related to Cycle 7 operation. The licensee reanalyzed the potential radiological consequences arising from these events.

5.3.1 <u>Locked Rotor</u>

The locked rotor accident is analyzed by postulating an instantaneous seizure of a reactor coolant pump rotor, resulting in a rapid decrease in reactor coolant flow and a reactor trip. In the licensee's analysis, ten percent of the fuel was assumed to fail as a result of the accident. This is to be compared with the previously assumed FSAR value of a failed fuel percentage of 1% arising from a locked rotor.

In the staff's "Safety Evaluation Report related to the operation of Catawba Nuclear Station, Units 1 and 2" (NUREG-0954) and its Supplements, it was concluded that there is no fuel failure calculated to occur for this accident at Catawba.

In the FSAR for Catawba, the licensee indicated previously that 1% fuel failure could be expected for this event and that exclusion area boundary whole body and thyroid doses of 4.4E-01 and 3.6 rem, respectively, were calculated to occur. The licensee's April 13, 1992, submittal, reported whole body and thyroid doses at the exclusion area boundary of 0.253 and 25.5 rem, respectively. However, these doses were calculated assuming a reduced primary to secondary leak rate of 0.5 gpm compared to the 1.0 gpm leak rate previously assumed. The reduced leak rage of 0.5 gpm was adopted so that the licensee's calculated potential offsite doses from a locked rotor event do not exceed the staff acceptance criterion that exclusion area boundary doses should be less than a small fraction (<10%) of Part 100 guideline values. Since the reduced TS allowable primary to secondary leakage value of 0.5 gpm reduces the calculated offsite radiological consequence from a locked rotor event to values which meet NRC regulatory criteria, the staff finds the changes proposed by the licensee related to the locked rotor event acceptable.

5.3.2 Rod Ejection

In analyzing the rod ejection accident, it is assumed that a mechanical failure of the control rod drive mechanism has occurred. Consequently, reactor coolant leaks to the containment and the control rod and control rod drive shaft would be moved to the fully withdrawn positions. As a result of this mechanical failure, a rapid positive reactivity insertion occurs as well as a primary system depressurization. An adverse core power distribution results with localized fuel damage.

In the staff's Safety Evaluation Report for Catawba (NUREG-0954), the staff used a value of 10% fuel failure as a result of the rod ejection accident. In performing its analysis contained in NUREG-0954, the staff calculated offsite doses via two pathways: 1) containment leakage of primary coolant from the ruptured drive mechanism, and 2) secondary system contaminated steam releases.

In the staff's SER analysis of this accident, a partition factor of 100 between the water and vapor phases in the steam generator was assumed. In addition, the staff's SER analysis of the radiological consequences of this event utilized the guidance of SRP 15.4.8 Appendix A and the recommendations of Regulatory Guide 1.77; this analysis indicated that thyroid doses from a postulated rod ejection accident from both secondary side leakage and containment leakage totalled about 7.9 rem. This assumes a failed fuel percentage of 10%. For purposes of this evaluation, the staff considered that a ratio of the assumed fuel failure rates and resultant thyroid doses would provide a suitable indication of calculated thyroid doses at the exclusion area boundary. Based on this technique, calculated exclusion area boundary thyroid doses are 39.5 rem. The licensee's projection of whole body dose is 0.28 rem. These values are well within the guideline values of 10 CFR 100. Consequently, the licensee has demonstrated that the radiological consequences of a rod ejection accident are within the acceptance criteria of Regulatory Guide 1.77 and SRP 15.4.8, and are acceptable.

5.3.3 Conclusions

Based on the foregoing, the staff concludes that the FSAR changes proposed by the licensee with respect to the radiological consequence analysis of the locked rotor and rod ejection accidents meet NRC regulatory criteria and are, therefore, acceptable.

6.0 TECHNICAL SPECIFICATION CHANGES

(1) Core Safety Limit (Figure 2.1-1a)

Figure 2.1-1a of the TS is revised to reflect the use of the BWCMV CHF correlation and the licensee's statistical core design (SCD) methodology with a 1.55 thermal design DNBR limit. The revised core safety limits are based on a full core of Mark-BW assemblies. The licensee addresses the transition core penalty for OFA fuel by applying a 3.8% DNBR penalty against the 10.7% generic margin included in the design DNBR limit. Since the BWCMV CHF correlation and the SCD methodology are based on the approved Topical Report DPC-NE-2004, we conclude that the revised core safety limits are acceptable.

(2) Revision to TS 2.2.1

The change to TS 2.2.1 will delete ACTION 2.2.1.b.1 and equation 2.2-1. ACTION 2.2.1.B.1 provides the option of declaring an instrumentation channel operable by the use of equation 2.2-1 when the Reactor Trip System Instrumentation (RTSI) or Interlock Setpoint (IS) is less conservative than the allowable value. This option requires the tabulation of the Total Allowance, the value Z, and the Sensor error terms in table 2.2-1. However, the deletion of this ACTION and equation 2.2-1 makes TS 2.2.1 more restrictive, in that the channel must be declared inoperable with the RTSI or the IS less conservative than the allowable value. This revision to TS 2.2.1 is acceptable.

(3) Changes to TS Table 2.2-1

The overtemperature and over power delta T trip function K values in TS Table 2.2-1 are revised to reflect the use of BWCMV CHF correlation and the statistical core design methodology with a 1.55 thermal design DNBR limit. In addition, an axial imbalance penalty, f2(delta I), is applied to (OP delta T) reactor trip.

The changes to delete the Power Range Neutron Flux Negative Rate reactor trip function from TS Tables 2.2-1, 3.3-1, 3.3-2, and 4.3-1 are acceptable, since it is no longer needed for the control rod drop event, when analyzed with the approved DPC methodology.

Also, the table is revised to delete the Total allowance (TA) column, the Z column, and the Sensor Error column from Table 2.2-1. This change is consistent with the proposed change to TS 2.2.1 and is acceptable.

(4) Changes to TS Table 3.1-1

TS Table 3.1-1 is revised to include all accident analyses that would require reevaluation in the event that one full-length Rod Cluster Control Assembly is inoperable. Deletion of large break LOCA (LBLOCA) analysis from TS Table 3.1-1 is acceptable since LBLOCA analysis does not take credit for any control rod insertion. This change is the same as that approved for McGuire Units 1 and 2. This change is acceptable.

(5) Changes to TS 3/4.2.2

The changes to TS 3/4.2.2 were made to provide surveillance requirements consistent with DPC methodology for core power distribution control and surveillance of power peaking as described in DPC-NE-2011PA. This change is acceptable.

(6) Changes to TS 3/4.2.2 and 3/4.2.3

TS 3/4.2.2 was revised to provide required actions and surveillance requirements consistent with DPC methodology for core power distribution control and surveillance of the nuclear enthalpy rise hot channel factor, as discussed in DPC-NE-2011PA. Also, the nomenclature is changed to reflect differences between DPC and the fuel vendor (B&W), and to clarify the surveillance requirements making them consistent with DPC's methods as described in DPC-NE-2011PA. Specification 3/4.2.3 was also revised to reflect the power peaking surveillance methods described in DPC-NE-2011PA. These changes are the same as those approved for McGuire 1. They have been appropriately described and justified in accordance with the approved DPC methodology and are, therefore, acceptable.

(7) Changes to TS 3/4.2.5

The change of the letter "a" to "b" in TS 3.2.5 ACTION item c.1.b connects the requirement of ACTION c.1.b to ACTION "b" instead of ACTION "a", which was incorrect. Therefore, it corrects a typographical error and does not represent an actual change to the requirements of TS 3/4.2.5. This change is acceptable.

(8) Changes to TS Table 3.3-2

The reactor trip on power range neutron flux negative rate is deleted. This is acceptable as explained in the changes to TS Table 2.2-1. The response times associated with the Resistance Temperature Detection (RTD) bypass system for the OT delta T and OP delta T are deleted, as is the footnote regarding the RTD bypass system. This is acceptable as the RTD bypass has been removed. Also, the neutron detector response time deletion which is applicable to the OT delta T trip is also deleted for the OP delta P trip due to the addition of the f2(delta I) function to the OP delta T trip. This is acceptable as explained in the changes to TS Table 2.2-1.

(9) Changes to TS 3/4.3.3.2

The changes to TS 3/4.3.3.2 include deletion of ACTION 3.3.2.b.1 and equation 2.2-1. These changes are consistent with the changes to TS 2.2.1, and the removal of the TA, Z, and S columns from tables 2.2-1 and 3.3-4. Deletion of ACTION 3.3.2.b.1 and equation 2.2-1 makes TS 3.3.2 more restrictive, in that the channel must be declared inoperable with the Engineered Safety Feature Actuation System (ESFAS) instrumentation or interlock setpoint less conservative than the allowable value. The staff finds the deletion of ACTION 3.3.2.b.1 and equation 2.2-1 to be acceptable as it improves the safe operation of the plant by reducing the complexity of the technical specifications.

(10) Changes to Table 3.3-4

Several categories of changes are made to table 3.3-4: (1) based on a reanalysis, the low steam line pressure setpoint for safety injection and main steam line isolation is increased from 725 psig to 775 psig and the allowable value of this trip function is changed from 694 psig to 744 psig, maintaining the same 31 psig allowance for rack uncertainties. The lead-lag controller for steam line pressure-low is deleted which eliminates spurious Engineered Safety Feature (ESF) actuation on minor pressure increases in the secondary system; (2) the TA, Z, and S columns are deleted from table 3.3-4 which is consistent with the removal of these items from table 2.2-1; and (3) the allowable values associated with the RTD bypass system for the feedwater isolation on Tavg-Low (item 5.c) and ESFAS P-12 interlock on Low-Low Tave (item 18.c) are deleted as a result of the RTD bypass system being removed. We have found the above changes to be acceptable.

(11) Change to TS 3.4.1.2

This specification is changed to require that three of the four loops be in operation for Mode 3. This restriction is imposed in order to make the specification consistent with the reanalysis of the uncontrolled bank withdrawal from subcritical or low power startup condition. The staff finds this change to be acceptable.

(12) Changes to TS 3/4.4.2.1 and TS 3/4.4.2.2

This modification changes the tolerance on the pressurizer safety valve lift setpoint from plus or minus 1% to +3% , -2% in all modes of operation. After verifying that the valves remain within tolerance over several cycles, this larger tolerance will enable reduction of work in a dangerous work environment by requiring only one valve to be tested per outage instead of three. The larger allowable deviation from the nominal lift setting is consistent with the licensing basis analyses. Three accident categories involving heat transfer mismatches, decrease in secondary heat removal, decrease in reactor coolant system flow rate and reactivity, and power distribution anomaly transients were analyzed by the licensee assuming a lift setpoint of 3 percent above the normal value. The analyses, which are the same as previously performed for McGuire and which used the approved methodology of topical reports DPC-NE-3001P (Reference 6) and DPC-NE-3002 (Reference 7), showed that the following peak RCS pressure acceptance criteria were met: 110% of design pressure for the feedline break and locked rotor transients and 120% of design pressure for the rod ejection transient. The amount by which the safety valve lift setpoint is allowed to drift downward is restricted to 2 percent of nominal in order to ensure that safety valve lift cannot preclude reactor trip on high pressurizer pressure. The licensee stated that reanalysis of departure from nucleate boiling (DNB) transients and the uncontrolled bank withdrawal at power and single rod withdrawal events showed that all acceptance criteria are met with the 2% downward setting. Based on meeting the acceptance criteria in the above analyses, the staff finds these changes to be acceptable.

(13) Changes to TS 3.5.1

This change raises the required average cold leg accumulator boron concentration in ACTIONS c.2 and c.3 from 1500 to 1800 ppm, and bases this average on all four accumulators instead of just the limiting three. Regardless of the break location, the contents of each accumulator will be emptied, either directly or indirectly into the containment sump. Therefore, calculating the volumetric average boron concentration based on all four cold leg accumulators is valid. The licensee has increased the volumetric average boron concentration from 1500 to 1800 ppm to ensure long-term subcriticality following a LOCA. Based on the licensee's analysis, the staff finds these changes to be acceptable.

(14) Changes to TS 4.5.2

The ECCS pump performance requirements in TS 4.5.2 (f) were modified by decreasing the centrifugal charging pump required head from 2380 to 2223 psid and the safety injection pump required head from 1420 to 1341 psid. Also, the ECCS delivered flow requirements in TS 4.5.2 (h) were modified by decreasing the centrifugal charging pump total flow rate from 565 to 560 gpm and increasing the safety injection total flow rate from 660 to 675 gpm. These revisions apply to Units 1 and 2. The licensee stated that these changes are consistent with revised pump vendor information and that margin is available to enable sufficient allowances for instrument errors and to permit reasonable test acceptance criteria. The pump performance at the new specification values is said to be sufficient to meet all acceptance criteria in both the current FSAR analyses, and in the Catawba 1 Cycle 7 reload. The staff, therefore, finds these changes to be acceptable.

(15) Change to TS 3/4.6.3, Table 3.6-2a

The proposed changes in Table 3.6-2a were related to isolation valve operation maximum response time. The licensee proposed that the numerical value of the stroke time of these valves be changed to NA partly on the basis that these valves do not receive a containment isolation signal. This information is inconsistent with the FSAR. Accordingly, the licensee withdrew this proposed change in its letter of August 26, 1992.

(16) Change to TS 3/4.7.1.4

As stated in the technical justification for the proposed revision to TS Table 3.3-4, the valve stroke time, when added to the applicable instrumentation delays, yields the overall ESF response time. This response time is input to the steam line break transient analysis. Analysis using the DPC approved SCD methodology shows this Condition IV transient does not violate the imposed Condition II acceptance criteria of no DNB. This change is acceptable.

(17) Changes to TS 6.9.1.9

The changes to TS 6.9.1.9 simply reflected the changes to the COLR due to the implementation of DPC core operating limit methodology.

The safety and control bank insertion limits were revised to reflect a minimum rod withdrawal limit of 222 steps and a maximum rod withdrawal limit of 230 steps. Analysis by Westinghouse showed that the various core physics parameters remained bounded by previous analysis. These changes are acceptable.

7.0 STATE CONSULTATION

In accordance with the Commission's regulations, the South Carolina State official was notified of the proposed issuance of the amendments. The State official had no comments.

8.0 ENVIRONMENTAL CONSIDERATION

The amendments change requirements with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and change surveillance requirements. The NRC staff has determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration, and there has been no public comment on such finding (57 FR 32240 dated July 21, 1992). Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

9.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

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Date: September 14, 1992

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3. BAW-10174P-A, MARK-BW Reload LOCA Analysis for Catawba and McGuire, Babcock & Wilcox, Revision 1, November 1990.

4. DPC-NE-2004P-A, McGuire and Catawba Nuclear Stations Core Thermal-Hydraulic Methodology using VIPRER-01, Duke Power Company, December 1991.

5. DPC-NE-3000P, Duke Power Company, Thermal-Hydraulic Transient Analysis Methodology, Revision 2, February 20, 1991.

6. DPC-NE-3001P, Duke Power Company, Multidimensional Reactor Transient Safety Analysis Parameters Methodology, Revision 2, November 1991.

7. DPC-NE-3002, McGuire Nuclear Station/Catawba Nuclear Station FSAR Chapter 15 System Transient Analysis Methodology, Revision 1, November 1991.

8. DPC-NE-2001P-A, Revision 1, Fuel Mechanical Reload Analysis Methodology, for Mark-BW Fuel, Duke Power Company, October 1991.

9. DPC-NF-2010A, McGuire Nuclear Station/Catawba Nuclear Station Nuclear Physics Methodology for reload Design, Duke Power Company, June 1985.

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11. BAW-10159P-A, BWCMV Correlation of Critical Heat Flux in Mixing Vane Grid Fuel Assemblies, Babcock & Wilcox, July 1990.