Duke Power
A Duke Energy Company

M. S. Tuckman
Executive Vice President
Nuclear Generation

Duke Power Company
A Duke Energy Company
EC07H
526 South Church Street
P.O. Box 1006
Charlotte, NC 28201-1006

(704) 382-2200 OFFICE
(704) 382-4360 FAX

April 19, 2000

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

Subject: Duke Energy Corporation
McGuire Nuclear Station, Units 1 and 2
Docket Nos. 50-369 and 370
Catawba Nuclear Station, Units 1 and 2
Docket Nos. 50-413 and 414
Topical Report DPC-NE-3002-A, Revision 4

Reference: Letter, Duke Energy Corporation to U.S. NRC, March 1, 2000, License Amendment Request for FOL and Technical Specification 1.1 Definitions - Dose Equivalent Iodine

Attached for review is Revision 4 to the Duke Energy Corporation Topical Report DPC-NE-3002-A, "UFSAR Chapter 15 Transient Analysis Methodology." This topical report revision consists of two minor changes, which are attached in the form of markups to three affected pages of the previously approved Revision 3 (NRC SER dated February 5, 1999). The approved version of Revision 3 was submitted to the NRC Document Control Desk on May 13, 1999.

The first change (see attached page 5-18) corrects the description of the primary coolant volume that is used in the UFSAR Section 15.4.6 Boron Dilution Accident Analysis for Mode 4 for Catawba Nuclear Station. The current topical report description of the primary coolant volume used in the analysis includes the Reactor Coolant System excluding the pressurizer, the pressurizer surge line, and the reactor vessel upper head. It was later determined that the correct minimum primary coolant volume for the

A001

Mode 4 boron dilution analysis would include only those regions of the Reactor Coolant System which circulate during the residual heat removal mode. The proposed change reflects the correct minimum mixing volume. The need for this topical report change was identified during the Catawba UFSAR verification project, and will update DPC-NE-3002-A to be consistent with the UFSAR that was previously revised by Revision 6. The need for the topical report revision was not identified at the time that the UFSAR revision was implemented. The change in the methodology is a conservative change, in that the mixing volume for the Mode 4 boron dilution accident is being revised to a smaller volume. The results of the analysis continue to meet the acceptance criterion.

The second change (see attached pages 7-8 and 7-9) is required to support the above referenced Catawba FOL and Technical Specifications license amendment request (LAR). The referenced submittal describes reanalysis of the UFSAR Section 15.6.3 Steam Generator Tube Failure Accident for Catawba. The details of the reanalysis are not repeated in this submittal. The revisions in the attachment are necessary to support the reanalyses. The first part of this change involves an increase from two to three in the number of main steam line PORVs credited in the Catawba analysis. This change is consistent with the current Technical Specifications which requires all four main steam line PORVs to be operable, with one PORV being the limiting single failure. The second part of this change specifies three minute operator response times for depressurizing the primary system and for initiating safety injection termination. The topical report revisions are being submitted to maintain consistency with the referenced LAR submittal. These changes are not applicable to McGuire, and the revisions include separating the McGuire and Catawba methodology assumptions as necessary.

Approval of this topical report revision is requested concurrent with, or prior to, the approval of the referenced LAR submittal. This submittal has a requested review/approval date of September 1, 2000.

Please address any questions to J. S. Warren (704) 382-4986 or G. B. Swindlehurst (704) 382-5176.

Very truly yours,

M. S. Tuckman

M. S. Tuckman

Attachments

xc w/Attachments:

L. A. Reyes, Regional Administrator
U. S. Nuclear Regulatory Commission
Region II
Atlanta Federal Center
61 Forsyth St., SW, Suite 23T85
Atlanta, GA 30303

Mr. C. P. Patel, Project Manager (CNS)
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Mail Stop O-8 H12
Washington, D. C. 20555-0001

Mr. F. Rinaldi, Project Manager (MNS)
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Mail Stop O-8 H12
Washington, D. C. 20555-0001

Mr. D. J. Roberts
NRC Senior Resident Inspector
Catawba Nuclear Station

Mr. S. M. Shaeffer
NRC Senior Resident Inspector
McGuire Nuclear Station

bxc:

<u>w/o Attachment</u>
G. B. Swindlehurst
C. J. Thomas
M. T. Cash
K. L. Crane
G. D. Gilbert
K. E. Nicholson
T. K. Pasour (2)
J. S. Warren
ELL

5.6.1 Initial Conditions

Dilution Volume

A postulated dilution event progresses faster for smaller RCS water volumes. Therefore, the analysis considers the smallest RCS water volume in which the unborated water is actively mixed by forced circulation. For Modes 1-3, the Technical Specifications require that at least one reactor coolant pump be operating. This forced circulation will mix the RCS inventory in the reactor vessel and each of the four reactor coolant loops. The pressurizer and the pressurizer surge line are not included in the volume available for dilution in Modes 1-3. For normal operation in Mode 4, forced circulation is typically maintained, although the Technical Specifications do not require it. The volume available for dilution in Mode 4 is therefore conservatively assumed to not include the upper head of the reactor vessel, a region which has reduced flow in the absence of forced circulation, or the pressurizer and the pressurizer surge line. ~~Since the Technical Specifications do require operability of all four steam generators during Mode 4, all four of the reactor coolant loops, in addition to the remainder of the reactor vessel, are included in the RCS volume available for dilution.~~ "Since the Technical Specifications allow for only a single train of the Residual Heat Removal System (RHRS) to be in operation, the Mode 4 dilution volume is assumed to be comprised of the reactor vessel (excluding the upper head), the RHR System, and portions of the hot and cold legs between the RHR inlet and outlet connections." For Modes 5 and 6, the reactor coolant water level may be drained to below the top of the main coolant loop piping, and at least one train of the Residual Heat Removal System (RHRS) is operating. The volume available for dilution in these modes is limited to the smaller volume RHRS train plus the portions of the reactor vessel and reactor coolant loop piping below the minimum water level and between the RHRS inlet and outlet connections. The minimum water level used to calculate this volume is corrected for level instrument uncertainty.

Boron Concentrations

The Technical Specifications require that the shutdown margin in the various modes be above a certain minimum value. The difference in boron concentration, between the value at which the relevant alarm function is actuated and the value at which the reactor is just critical, determines the time available to mitigate a dilution event. Mathematically, this time is a function of the ratio of these two concentrations, where a large ratio corresponds to a longer time. During the reload safety analysis for each new core, the above concentrations are checked to ensure that the value of this ratio for each mode is larger than the corresponding ratio assumed in the accident analysis. Each mode of operation covers a range of temperatures. Therefore, within that mode, the temperature which minimizes this ratio is used for comparison with the accident analysis ratio. For accident initial conditions in which the control rods are withdrawn, it is conservatively assumed, in calculating the critical boron concentration, that the most reactive rod does not fall into the core at reactor trip. This assumption is also

For McGuire,

For Catawba, three steam line PORVs on the intact steam generators are assumed to be operable

Steam Line PORVs
~~Only~~ two of the three steam line PORVs on the intact steam generators are assumed to be operable. ~~This lengthens the cooldown time, thereby maximizing the atmospheric steam releases.~~ A negative bias is applied to the ruptured steam generator PORV control signal. This results in an earlier opening time which maximizes atmospheric releases and delays operator identification of the failed open steam line PORV. A positive bias is applied to the intact SG PORV control signals to maximize secondary side post-trip pressurization. This delays operator identification of the failed open steam line PORV.

Decay Heat
End-of-cycle decay heat, based upon the ANSI/ANS-5.1-1979 standard plus a two-sigma uncertainty, is employed.

Offsite Power
Offsite power is assumed to be lost coincident with turbine trip. This isolates steam flow to the condenser, thereby maximizing the atmospheric steam releases.

Break Model
The break is assumed to be a double-ended guillotine break of a single steam generator tube at the tubesheet surface on the steam generator outlet plenum. This location maximizes the mass flow through the break.

RCP Operation
The reactor coolant pumps are assumed to operate normally until offsite power is lost coincident with turbine trip.

ECCS Injection
SI actuation is assumed to occur on low pressurizer pressure at a setpoint with an applied positive uncertainty or on manual operator action. Maximum ECCS injection flow is assumed to maximize the primary-to-secondary leakage.

Main Feedwater
Main feedwater flow is assumed to terminate coincident with the loss of offsite power to minimize the secondary inventory available to mix with and dilute primary-to-secondary leakage.

Charging Flow
A conservatively high charging flow capacity is modeled to delay reactor trip and maximize total primary-to-secondary leakage.

Manual Actions
- Immediate action to maximize charging flow (penalty).

- Immediate action to energize pressurizer heater banks (penalty).

- Operators identify the abnormal condition of the RCS at 20 minutes and manually trip the reactor if not already tripped.

- Identify and isolate ruptured steam generator consistent with assumptions in WCAP-10698 (Reference 5), 15 minute minimum delay (credit).

- Isolate failed open steam line drains upstream of the main steam isolation valves. This action occurs 10 minutes after the ruptured steam generator is identified.

- Isolate the steam supply to the turbine-driven auxiliary feedwater pump from the ruptured steam generator after identification of the ruptured steam generator. An operator action delay time of 30 minutes is assumed (credit).

- Isolate failed open steam line PORV on the ruptured steam generator with an operator action delay time from when it should have closed normally. The delay times assumed are 10 minutes for control room and 30 minutes for local operation (credit).

- Manually control auxiliary feedwater to maintain zero power steam generator levels (nominal).

- Using the steam line PORVs, initiate natural circulation cooldown of the primary system after identification of the ruptured steam generator. Operator action delay times of 15 minutes for control room action and 45 minutes for local action are assumed (credit).

- ~~Initiate~~ For McGuire, initiate depressurization of the primary system using the pressurizer PORVs to terminate break flow 10 minutes after the primary system is 20°F subcooled at the ruptured steam generator pressure (credit). For Catawba, this action is initiated 3 minutes after the primary system is 20°F subcooled (credit).

- Initiate SI termination 3 minutes after completing the depressurization of the primary system (credit).

7.2.2.4 Control, Protection, and Safeguards System Modeling

Reactor Trip
A reactor trip occurs on either low pressurizer pressure or manual operator action at 20 minutes. A negative uncertainty is applied to the low pressurizer pressure trip setpoint to delay reactor trip. The overtemperature ΔT trip function is not credited.

Pressurizer Pressure Control
This control system is assumed to be in manual and therefore is not modeled. Operator action is assumed to energize the pressurizer heaters and control the PORVs. Pressurizer spray is not available for the duration of this transient.

Pressurizer Level Control
This control system is assumed to be in manual and therefore is not modeled. Operator action is assumed to maximize charging flow.