UNITED STATES
NUCLEAR REGULATORY COMMISSION
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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

MASS AND ENERGY RELEASE AND

CONTAINMENT RESPONSE METHODOLOGY

MCGUIRE NUCLEAR STATION, UNITS 1 AND 2

CATAWBA NUCLEAR STATION, UNITS 1 AND 2

DOCKET NOS. 50-369, 50-370, 50-413, AND 50-414

1.0 INTRODUCTION AND BACKGROUND

By letter dated September 9, 1994, the Duke Power Company (DPC) submitted proprietary Topical Report DPC-NE-3004 for staff review and approval. Additional information, in response to staff questions, was provided in a letter dated May 12, 1995. Additional clarifications were provided with respect to proprietary information in a submittal dated July 25, 1995. The submittal followed a meeting between the staff and Duke held on September 1, 1994 (Ref.: Meeting Summary dated October 31, 1994). The report describes Duke's methodology for analyzing; (1) the mass and energy release from high energy line breaks in containment, and (2) the resulting long-term containment response, for the Catawba and McGuire Nuclear Stations. The methodology is applicable to the licensee's two 3411 megawatts thermal (MWt) McGuire units near Charlotte, NC, and two 3411 MWt Catawba units near Rock Hill, SC. These facilities have ice condenser containments. The free-standing, cylindrical steel containments are designed for a peak internal pressure of 15 psi and a negative (external) pressure of 1½ psi. The qualification temperature for electrical equipment within the containments is 340° F.

Containment peak pressure and temperature analyses are performed to determine the pressure and temperature loads on a containment that would result from postulated pipe breaks inside the containment. The results of the containment pressure and temperature analyses establish minimum design criteria, test criteria and environmental qualification criteria for containment systems, structures and components. Containment pressure and temperature analyses are of two main types; short-term and long-term. Short-term analyses encompass the early blowdown phase of a line break during the period when the break results in the sudden release of a large amount of stored energy which must be accommodated primarily by the containment volume and passive heat sinks. Long-term analyses encompass the post-blowdown period when decay and residual heat sources become dominant and the active containment cooling systems become the major method of heat removal. For ice condenser containments, a large quantity of ice is provided as a pressure suppressant to permit use of a smaller containment having a lower design pressure. Ice melt therefore plays a significant role in containment pressure suppression during both the short and long terms.

In addition to containment short-term and long-term pressure and temperature analyses described above, other types of pressure analyses are required for licensing. These include a negative pressure analysis to assure that a containment will not implode due to rapid cooling, and minimum pressure analyses to assure adequate net positive suction head for ECCS pumps and to assure adequate ECCS performance in core reflood. As discussed in Section 1 of the Topical Report, of these additional types of containment pressure analyses, the topical report encompasses only the latter and does not include the short term blowdown peak/subcompartment analysis or the negative pressure analysis.

The methodology described in the report has been used in support of replacement of the original preheater-type Westinghouse steam generators (S/Gs) with new feedring-type Babcock and Wilcox (BWI) S/Gs. These modifications will result in increased mass and energy in the primary and secondary systems. DPC-NE-3004-P is one of several topical reports relating to the S/G replacement program. Others include: (a) DPC-NE-3000, "Thermal-Hydraulic Transient Analysis Methodology;" through Revision 1 as approved November 15, 1991. Revision 3 of this report is currently under staff review. (b) DPC-NE-3001-PA, "Multidimensional Reactor Transients and Safety Analysis Physics Parameters Methodology," dated November 1991, and (c) DP-NE-3002-A ,"FSAR Transient Analysis Methodology", November 1991. Revision 1 of DPC-NE-3002 is currently under staff review. In addition to S/G replacement, the new methodology has potential future use in the evaluation of ice inventory requirements and analyses supporting power uprate.

The current licensing basis analyses described in the FSARs are 1970s-vintage analyses performed by Westinghouse. In addition to using the new methodology for performing new analyses for the replacement S/Gs, the licensee has re-performed the original licensing analyses using the new methodology.

The staff has reviewed the licensee's submittal using the guidance and criteria of the following sections of the Standard Review Plan (SRP): (a) 6.2.1, "Containment Functional Design;" (b) 6.1.1.B, "Ice Condenser Containments," (c) 6.2.1.3, "Mass and Energy Release Analysis for Postulated Loss of Coolant Accidents," and (d) 6.2.1.4, "Mass and Energy Release Analysis of Postulated Secondary System Releases." In addition, ANSI/ANS 56.4-1983, "Pressure and Temperature Transient Analysis for Light Water Reactor Containments" has been used for guidance by both the staff and the licensee as a reference for identification of standard and accepted practices used in performing containment pressure and temperature analyses.

Based on the staff's earlier review of DPC-NE-3003, "Mass and Energy Release and Containment Response Methodology" for Oconee, Letter of L.A. Wiens, NRC, to M.S. Tuckman, DPC, dated March 15, 1995, the licensee's technical support organization has been found to meet the criteria of Generic Letter (GL) 83-11, "Licensee Qualification for Performing Safety Analyses in support of Licensing Actions," and is competent to develop, verify, validate, use and maintain computer codes for the purpose of containment pressure and temperature analysis.

The format and organization of this SE, including the paragraph/section numeration is consistent with that of the topical report.

2.0 COMPUTER CODES

The mass and energy releases associated with various postulated pipe breaks are analyzed using appropriate thermal-hydraulic analysis codes. The resulting mass and energy profiles are then input to containment analysis codes to determine the containment pressure and temperature responses. LOCA mass and energy releases are simulated with the RELAP5/MOD3.1DUKE code. Main Steamline Break (MSLB) mass and energy release analyses are simulated with the RETRAN-02 MOD5.1DKE code. The containment pressure and temperature responses to mass and energy releases in containment are simulated with the GOTHIC4.0/DUKE code. These codes are Duke modifications of well-known generic thermal-hydraulic analysis codes that are widely used for these purposes. Although descriptions are provided below, the staff's review did not focus on the basic governing equations, constitutive models, component models, and numerical methods utilized in the codes, but instead focused on the licensee's development of plant-specific models to be analyzed by the codes.

2.1 RELAP5/MOD3.1DUKE

2.1.1 CODE DESCRIPTION

The Westinghouse LOCA mass and energy release methodology used for the original licensing analyses is described in WCAP-8264-P-A, "Westinghouse Mass and Energy Release Data for Containment Design," Revision 1, August 1975). The SATAN-V computer code was used to determine the blowdown phase mass and energy release. The WREFLOOD code was used to calculate the reflood phase mass and energy release, and the FROTH code was used for the post-reflood phase. That methodology was approved in a staff SE dated March 12, 1975. For purposes of future calculations of LOCA mass and energy release data for McGuire and Catawba containment long-term pressure and temperature analysis it is to be replaced by methodology based on RELAP5/MOD3.1DUKE. RELAP5/MOD3.1DUKE is the licensee's variant of the RELAP5/MOD3 code described in NUREG/CR-5535.

The RELAP5/MOD3.1DUKE code is derived from RELAP5/MOD3.1 which was developed by EG&G Idaho under NRC sponsorship. The code models the steady-state and transient behavior of a hydraulic system that may contain a mixture of steam, water, non-condensible gas or nonvolatile solute. The fluid system is modeled by discretizing the system into control volumes (nodes) joined by flow junctions. The hydraulic flow field treats the liquid and steam phases as separate fluids in a nonhomogeneous, non-equilibrium manner, solving the mass, energy and momentum equations for each phase. Constitutive relationships are used to define flow regimes and to model interphase drag, vapor generation and interphase heat and mass transfer, and horizontal and vertical stratification. Empirical relationships are used to model convective heat transfer, energy partitioning between phases, choked flow and wall friction. The code supports simulation of the primary system, secondary system, feedwater train, automatic control systems and core neutronics. Available component models include reactor point kinetics, pumps, valves, heat structures, heat exchangers, turbines, separators and accumulators.

2.1.2 RELAP5/MOD3.1DUKE ("RELAP") LOCA SIMULATION MODELS

Nodalization: The licensee has developed a two-loop base model for SBLOCA and LBLOCA mass and energy analyses. The base model uses [ ] downcomer nodes and [ ] core nodes, with [ ] nodalization for the core region and [ ] for the downcomer. A third loop is used for cases where asymmetric boundary conditions may occur among the intact loops. The nodalization is depicted in Figure 2.1.2-1 of the topical report. Appropriate S/G nodalization is used to reflect BWI feedring or Westinghouse preheater S/Gs as desired for a specific analysis. The vessel, piping and S/G nodalization are of sufficient detail to assure that the requirements of ANS-56-4-1983 are met (i.e., break flow quality is not overpredicted, and core-to-coolant, metal-to-coolant, and S/G-to-coolant heat transfer will conservatively predict high containment peak pressures).

Validation: The CNS/MNS (Catawba Nuclear Station/McGuire Nuclear Station) RELAP models have been benchmarked against available plant transient data (RCP coastdown and loss of offsite power natural circulation events) and against the original Westinghouse analyses described in the FSAR. This information is discussed in the topical report. A graphical comparison was provided of the FSAR analysis and the new analysis, for the pump suction break event. (The pump suction break produced the highest peak containment pressure in the FSAR analyses.) Based on relatively good agreement between the FSAR analyses and the new Duke analyses for the suction break, and the licensee's discussion (Ref. May 12, 1995 letter) of reasons for the differences, the RELAP model is considered suitable for use in the calculation of LOCA mass and energy releases for use in long-term containment analyses.

2.2 RETRAN-02 MOD5.1DKE

Secondary system mass and energy release analyses are performed using the RETRAN-02 MOD 5.1DKE (RETRAN) code, a modified version of the widely used RETRAN-02 MOD 5.1 code. RETRAN was developed by Energy Incorporated for the Electric Power Research Institute to provide utilities with a code capable of simulating thermal-hydraulic transients of interest for both PWRs and BWRs. It can be used to model a general fluid system by partitioning the system into one-dimensional volumes and connecting flowpaths or junctions. The code solves the mass, energy and momentum equations using numerical methods. Although the RETRAN-02 equations describe homogenous equilibrium fluid volumes, phase separation can be modeled by separated bubble-rise volumes and by a dynamic slip model. Heat transfer across steam generators and to or from structures can be modeled. Component models for heat exchangers, pumps, and valves, are available in RETRAN. Control system and trip logic capability are also provided. A general transport model capable of modeling the distribution of boron is included. RETRAN-02 MOD5.0 has been reviewed and approved by the staff for use in the analysis of non-LOCA transients (Ref.: Letter w/SE from A. Thadani to W.J Boatright dated November 1, 1991). The DKE version of the code incorporates corrections which are to be incorporated in future versions of the generic EPRI version of RETRAN.

The licensee's MSLB mass and energy release analysis RETRAN models for Catawba and McGuire were developed from earlier models developed for transient analysis purposes. These models were described in DPC-NE-3000, "Thermal-

Hydraulic Transient Analysis Methodology." That topical report, through Revision 1, was approved by the staff for use in non-LOCA transient analysis of the McGuire and Catawba facilities on November 15, 1991 (Letter T. Reed, NRC, to H. Tucker, DPC). The models used for containment mass and energy release analyses have been modified to: (a) model breaks, (b) model the asymmetry, (c) model thermal mixing in the vessel, and (d) add additional detail to the S/G heat conductors. They are otherwise the same as the DPC-NE-3000 transient analysis models.

Validation: There are no suitable MSLB data available for validation of the Catawba and McGuire MSLB models for the replacement S/Gs. The licensee has attempted to ensure conservatism through selection of initial conditions and boundary conditions.

2.3 GOTHIC

The containment response calculations for the original Catawba nd McGuire licensing analyses were performed using the Westinghouse TMD and LOTIC codes. These codes were used for the blowdown and post-blowdown phases respectively. A non-proprietary description of the Westinghouse models is provided in "The Ice-Condenser System for Containment Pressure Suppression," *Nuclear Safety*, Vol. 17, No. 6 December 1976.

The new containment response methodology utilizes GOTHIC (Generation of Thermal Hydraulic Information for Containments) to calculate the long-term containment pressure and temperature responses to the mass and energy inputs from high energy primary and secondary reactor coolant system breaks. GOTHIC is a derivative of FATHOMS, which in turn was a derivative of the NRC's COBRA-NC thermal-hydraulic code. GOTHIC is capable of modeling all containment types (i.e., ice condenser, large dry, subatmospheric and suppression pool).

With GOTHIC, a containment is modeled as a network of computational volumes connected by fluid flow path junctions. GOTHIC solves the mass, energy and momentum equations for multi-component, two-phase flow. The control cells may be lumped parameter, one, two or three dimensional, or any combination. Velocity fields are provided for: (1) vapor/non-condensible gases, (2) continuous liquid, and (3) liquid droplet. Up to eight non-condensible gases may be modeled. Temperature fields are provided for: (1) vapor/non-condensible gas mixture, (2) continuous liquid, and (3) liquid droplets. The temperature fields may be in thermal non-equilibrium within the same calculational volume. A mass balance is solved for solid ice. The simplified ice model does not provide for transport. If ice exists, its temperature is set to a constant value by code input and it remains at that temperature until it changes phase to liquid. Passive thermal conductors, flat plate, cylindrical tube or solid rod models, are simulated with finite-difference conduction models. Active heat sources and sinks may also be included in the volumes. Valves, heat exchangers, pumps, spray nozzles and fans may be included in the flow paths.

2.3.1 GOTHIC ICE CONDENSER MODEL

GOTHIC has the capability to model ice condensers, including such phenomana as steam condensation, ice melt, the spring-loaded doors at the inlet to the ice condenser, and the drainage spray into the lower containment. The ice is not modeled as a component but as a feature of the associated volume. For each computational volume modeled to contain ice, the user inputs initial ice conditions and control parameters, including the amount and location of ice, and its temperature and density, in a manner similar to fluids. Ice condenser heat transfer is calculated explicitly in GOTHIC using the equation:

$$Q_i = \lambda_i(t) A_i H_i (T_v - T_i)$$

where

Q_i is the heat from vapor to ice
$\lambda_i(t)$ is a user-specified time-dependent multiplier
A_i is the ice surface area
H_i is the vapor/ice heat transfer coefficient
T_v is the vapor temperature
T_i is the specified ice temperature

The ice is assumed to be in solid, smooth-surfaced cylinders. The time-dependent multiplier is used to account for the difference in actual and effective ice surface area due to the form of the ice in the baskets (i.e., the ice is in the form of flakes which increases the effective area). The heat transfer coefficient is described in Duke's letter of May 15, 1995. An ice melt rate is calculated for each time step. The ice melt rate, in turn is used to calculate a rate of change of ice volume fraction. That, in turn, is used to calculate the change in ice volume fraction for each time step. From this, the change in ice surface area is calculated for each time step.

2.3.2 MCGUIRE AND CATAWBA GOTHIC MODELS

NODALIZATION

The MNS and CNS containments consist of four different regions; the lower compartment containing the NSSS (Nuclear Steam Supply System), the dead-ended regions, the ice condenser regions, the upper compartment. The lower compartment volume is modeled [ ] being subdivided into [ ] regions. The upper compartment is modeled as a volume with [ ] nodes representing the refueling canal, the operating floor space and the upper dome space. The ice condenser is modeled [ ] nodes. The seventeen dead-ended compartments, which are separate rooms or compartments within the lower compartment, are modeled as [ ] volumes. This is a high degree of nodalization for long-term analyses.

INITIAL CONDITIONS

The GOTHIC input files are written to specify an initial lower containment temperature of 100° F, an upper containment temperature of 75° F, dead-ended compartment temperature of 100° F, and ice condenser compartment temperature of 30° F. An initial containment pressure of 0.3 psig and humidity of 100% are assumed. Since it is conservative to assume initial conditions that produce a high mass of non-condensible gases (Ref.: ANS-56.4 para. 4.3.2), and minimize the warming of ice prior to melting, these initial conditions are conservative. (Information regarding the sensitivity of peak pressure to changes in initial conditions is provided in the licensee's May 15, 1995 response to staff questions.)

HEAT STRUCTURES

For simplification, the MNS and CNS GOTHIC models assume that the outer surfaces of the containment walls and dome are insulated. This assumption is conservative since heat is retained in the containment. The other structures (inside the containment) are represented as a combined one-sided slab. The Direct Uchida heat transfer correlation is used for heat transfer from the containment atmosphere to condensing surfaces. Use of the Uchida correlation is consistent with accepted practice. The standard GOTHIC interfacial heat transfer models, as described in the GOTHIC Technical Manual (EPRI), are used for heat transfer between droplets, liquid and vapor. These models are considered valid on the basis of the results of tests and benchmarking as described in the EPRI Qualification Manual and on the basis of comparison of results with original licensing analyses.

BOUNDARY CONDITIONS

Boundary conditions are: (1) the energy sources that transfer mass and energy to the modeled system during the event, and (2) assumptions that specify how the mass and energy are distributed and/or how the components will respond during the event. Boundary conditions are specified by the code user for parameters governed by conditions outside of the problem boundaries. The code imposes the boundary conditions on the system model at the beginning of the transient after the initial conditions have been established. Boundary conditions for the GOTHIC analyses include such items as: (1) break mass flow rate and energy input data, (2) building spray mass flow rate and energy input, (3) containment RHR system sump drainage pumped through the RHR heat exchanger and branching to [a] the primary system, [b] RHR spray nozzles or [c] the containment floor, and (4) nitrogen addition to containment from accumulators via the RCS break. All containment spray is assumed to enter the containment at the dome with an average droplet size of 700 μm. (Justification for this assumption is provided in the licensee's May 15, 1995 letter responding to staff questions.) Nitrogen flowrate and timing is based on the LOCA analysis. Liquid in the break flow is assumed to be in the form of 20 μm diameter droplets during blowdown and continuous liquid thereafter.

2.3.3 VALIDATION OF CODE AND MODEL

CODE VALIDATION

The GOTHIC code has been subjected to sample problem testing and benchmarking as described in the EPRI GOTHIC Qualification Manual. However, validation of the ice condenser heat transfer model is not encompassed by those actions. Accordingly, the licensee instituted an effort to obtain data that would demonstrate the suitability of GOTHIC for its ice condenser facilities. This effort is described in 2.3.4 below.

MODEL VALIDATION

The licensee benchmarked the MNS GOTHIC model against the Westinghouse LOTIC-1 results for the RCP suction break. (The RCP suction line break is the FSAR limiting break location.) It was found that GOTHIC and LOTIC-1 results exhibited reasonably similar pressure trends. The LOTIC-1 peak pressure (14.0 psig) was 1.1 psi greater than the GOTHIC peak pressure (12.9 psig). Significant is the relative agreement on a steady pressure of 6-9 psig during ice melt over the first hour of the event. For the upper containment temperature response, there were discrepancies between the GOTHIC results and LOTIC-1 results for the ice melt phase. These discrepancies are attributed to LOTIC-1 simplifications. GOTHIC more conservatively models the escape of steam from the ice condenser area into the upper containment. The lower compartment temperature stabilized at 200-210° F, during the period of peak pressure, for both GOTHIC and LOTIC-1. There was good agreement with the ice-melt curve for GOTHIC and LOTIC-1. Ice melt completion occurred at about 3500 seconds for both. The sump temperature curves also compared reasonably well. Both models indicated a temperature of approximately 190° F at 1000 seconds. Following completion of ice melt at 3500 seconds, the GOTHIC results showed a stable sump temperature about 10° F higher than LOTIC-1.

2.3.4 ICE CONDENSER HEAT TRANSFER VALIDATION

2.3.4.1 ICTF PROGRAM

The ice condenser heat transfer equation in GOTHIC was described in 2.3.1. It was noted that the GOTHIC code qualification program did not encompass the ice condenser models. The licensee therefore undertook such a program using an Ice Condenser Test Facility (ICTF) that had been developed and previously used to obtain data on aerosol particle transport and retention in an ice condenser (Ref: NUREG/CR-5768, "Ice Condenser Aerosol Tests," September 1991). The purpose of the ICTF program was to demonstrate that GOTHIC has the capability to accurately simulate the flow and heat transfer processes that occur in an ice condenser, particularly during the post-blowdown/post-reflood long-term phase of a large break DBA-LOCA. It is during this phase, subsequent to ice bed meltout, that the peak accident pressure occurs.

The ICTF is a full-height, four equivalent ice basket, scaled representation of a Westinghouse 1944-basket ice condenser system. For a description of the ICTF, the reader is referred to Section 2.3.4 of the topical report.

2.3.4.2 GOTHIC ICTF MODEL

For the GOTHIC model of the ICTF the test assembly was divided into 13 axial nodes, 10 of which represented the 48-foot high portion contain ice. Horizontally (north-south), the test assembly was divided into six equal-width channels representing the flow from the diffuser outlet box which contains five turning vanes. The ice was modeled as being symmetrically and uniformly distributed among the six channels.

2.3.4.3 COMPARISONS OF GOTHIC PREDICTIONS TO ICTF TEST RESULTS

The tests conducted at the ICTF and the results of the calculations using the GOTHIC model of the ICTF tests indicate that GOTHIC has the capability to accurately simulate the mass and heat transfer processes that occur in an ice condenser during the long-term phase of a LBLOCA. The ICTF data were used to determine an appropriate ice area multiplier, which was input to GOTHIC. This multiplier was [ ]. Using this multiplier GOTHIC produced analytical results which compared well with the ICTF data. It is significant that the [ ] multiplier appears valid throughout the duration of the ice meltout period. The ICTF test did not encompass steam flow conditions at a scale equivalent to blowdown rates. Accordingly, the resultant conclusions regarding GOTHIC's capability to model ice condenser heat transfer are limited in applicability to analysis of the long-term phase. (The FSAR short-term analyses remain applicable for blowdown effects.)

2.3.4.4 ACCEPTABILITY OF GOTHIC FOR USE IN CATAWBA/MCGUIRE LONG-TERM ANALYSES

With the [ ] multiplier in the heat transfer model as discussed in 2.3.1, the GOTHIC model of the ICTF was found to produce ice melt predictions acceptably consistent with the ICTF test results. Also, using the same multiplier, the GOTHIC model produced timing/trend results consistent with that of the LOTIC licensing analysis. Based on these findings, the GOTHIC ice model, with the [ ] multiplier, is considered validated for use in long-term pressure and temperature analyses.

3.0 <u>LARGE BREAK LOCA MASS AND ENERGY RELEASE ANALYSES</u>

3.1 OVERVIEW

For an ice condenser containment, the limiting LOCA containment peak pressure will occur during the long-term period following melt of most or all of the ice. Due to the sensitivity of the peak pressure value to the time of ice meltout, small breaks need not be examined since it is recognized that a large break produces the limiting peak containment pressure.

Using the RELAP code discussed in 2.1 above, the licensee calculated mass and energy release rates for double-ended guillotine breaks of the RCS in three locations: (1) the hot leg, (2) RCP suction leg, and (3) the RCP discharge leg. For (2) and (3), the break flow quality effects of ECCS recirculation phase injection to the hot leg was analyzed in addition to normal cold leg injection. The result of these analyses were input to the GOTHIC code as data for the four boundary conditions representing the liquid and vapor flow from each side of the break into the containment. The Ransom and Trapp critical flow model option was selected as the break flow model. Flow discharge coefficients were applied so as to provide break flow results equivalent to that of the Moody/Henry-Fauske critical flow models. Flow coefficients are chosen according to the type of break. In RELAP, the Ransom and Trapp choked-flow model is in equation form, whereas other models such as Henry-Fauske, Moody, Homogenous-Equilibrium, and Murdock-Bauman are in tabular form. The license's methodology enables the Ransom and Trapp option to be used, but provides results equivalent to that of the Standard (ANS 56.4-1984) with a smooth transition between phases.

3.1 INITIAL CONDITIONS

The initial conditions for the RELAP LBLOCA mass and energy release analyses were chosen with consideration of the guidance presented in ANS-56.4-1983, paragraph 3.2.2. The intent is to select initial conditions that will maximize the stored energy in the reactor primary and secondary coolant systems and thus contribute to a conservatively high peak pressure in the subsequent containment analysis.

INITIAL CONDITIONS		
PARAMETER	**ANS GUIDANCE**	**LICENSEE SELECTION**
Core power level	≥ licensed power level plus an uncertainty allowance (e.g., 102%)	Nominal + 2% (i.e., 3479.22 Mwt)
Core inlet temperature	≥ normal operating temperature for the selected power level plus upward adjustment for uncertainties.	Nominal + 4°
RCS pressure	≥ normal operating pressure for the selected power level plus allowance for uncertainties.	Nominal + 60 psi uncertainty allowance.
RCS flow	No guidance	High design flow rate plus 2.2% uncertainty.
S/G pressure	≥ normal operating pressure plus uncertainty allowance.	S/G pressure will be determined by RELAP initialization control for power level and Tavg.
Pressurizer level	≥ maximum normal operating level plus uncertainty allowance.	Nominal + 9%.
S/G water level	≥ normal level associated with selected power level plus uncertainty allowance.	8% and 10% uncertainty allowances will be applied to the nominal Westinghouse and BWI S/Gs levels respectively.
Safety injection tank pressure and water level	Normal operating values with allowances for uncertainties biased to produce maximum containment pressure.	Based on the results of sensitivity studies, bounding low initial pressure and low liquid volume will be used in the cold leg accumulators for all three break locations.
Safety injection tank temperature	Normal operating value with allowance for uncertainties biased to produce maximum containment pressure.	A bounding high temperature is selected to maximize break flow temperature and energy.
Refueling water storage tank (RWST) liquid volume	Choose ECCS flows and delay times in accordance with single-failure criteria to produce highest peak containment pressure.	A bounding low RWST inventory is selected to minimize the heat sink effect of the a large volume of cold water and to delay recirculation switchover.
Main feedwater temperature	No guidance.	A high MFW temperature is assumed in order to maximize the heat source effect of the S/G.

The licensee's criteria for selection of initial conditions are conservatively selected and thus consistent with the acceptance criteria of Standard Review Plan Section 6.2.1.3.

3.3 BOUNDARY CONDITIONS - ENERGY SOURCES AND ASSOCIATED ASSUMPTIONS

The energy released into a containment by a pipe break is that energy that is (a) initially contained in the primary and secondary coolant systems fluids and the metal components of the system boundaries and the sensible heat stored in the core, plus (b) that additional energy that is produced and released subsequent to the break as a result of continued fission, fission product and actinide decay and metal-water reaction. This section describes how the energy sources are accounted-for in the analysis.

3.3.1 ENERGY SOURCES

The initial conditions described above serve to maximize the stored energy initially present at the time of the break. In addition, the following assumptions are used to conservatively maximize the results of the analysis.

3.3.1.1 RCS AND S/G INVENTORY

The volume of the RCS piping system is increased 1% to account for the increased inventory due to thermal expansion. Also, zero S/G tube plugging is assumed.

3.3.1.2 RCS AND S/G METAL

Heat structures are assumed to be in thermal equilibrium with the coolant in which they are in contact.

3.3.1.3 CORE STORED ENERGY

A core time-of-life is selected such that the combined effects of the core stored energy and decay heat release will provide a core stored energy release that bounds all future core loadings for any point in the fuel cycle.

3.3.1.4 FISSION ENERGY

The RELAP5 code includes a point kinetics reactor model that computes the immediate fission power and the power from decay or fission fragments taking into account moderator density, Doppler and initial boron. Either of two reactivity feedback methods can be implemented. One method involves a calculation in which the effects of rapidly changing boron are not directly computed but which enables boron feedback to be simulated using a control system. (A RELAP *control system* provides the capability to evaluate simultaneous algebraic and ordinary differential equations to simulate control systems and other phenomena). The other method is to provide interpolable data in the form of a lookup table. The licensee's methodology utilizes the latter method with the data being calculated by generating a bounding 2nd order polynomial curve between the reactivity curve's known endpoints.

Because a point kinetics model is not capable of calculating spatial power distributions, nodal reactivities are flux-weighted to obtain a single reactivity value for use in the point kinetics model. A bounding beginning-of-cycle (BOC) β_{eff} is used for conservative moderator density feedback since end-of-cycle β_{eff} would provide a non-conservatively high Doppler effect greater than the increased density feedback effect of a BOC β_{eff}.

3.1.5 FISSION PRODUCT AND ACTINIDES DECAY

Radioactive decay of fission products and actinides is based on the ANSI/ANS-5.1-1979 standard with 2σ uncertainty added to the mean values. This is an approved standard practice.

3.3.1.6 METAL-WATER REACTION RATE

Heat resulting from exothermic metal-water reaction is considered. Although RELAP includes a zirconium-water reaction modeling capability, its use requires detailed thermal-hydraulic modeling of the core. The licensee has elected to using a simplified model that conservatively bounds the expected reaction. The total amount of clad reaction is assumed to be 1% of the amount that would be generated due to reaction of all of the cladding in the active region of all of the fuel rods. The metal-water reaction is assumed to begin when the PCT exceeds 1800° F (~65 seconds based on FSAR information) and follows a parabolic rate for approximately 200 seconds. The hydrogen generated by the reaction is added to the containment atmosphere as a non-condensible gas.

The above methodology provides a conservative consideration of metal-water reaction rate.

3.3.2 ASSUMPTIONS

3.3.2.1 LIMITING SINGLE FAILURE

It is assumed that loss of one train of ESF systems at the beginning of the accident is the limiting single-failure for LBLOCAs and that the resultant effect on maximum peak containment pressure is more severe than for the case of no equipment failures. This assumption is based on the knowledge that peak containment pressure for an LBLOCA occurs relatively late when decay heat is the primary heat source and ESF core cooling and containment spray are in use as heat removal systems. The staff acknowledges that the licensee's assumption is appropriate and that other potential single-failure scenarios need not be individually analyzed.

3.3.2.2 SELECTION OF BROKEN LOOP

The licensee models the break to occur in the loop containing the pressurizer. However, since peak pressure occurs in the long-term, post-reflood period, peak pressure results are not sensitive to the broken loop selection.

3.3.2.3 EMERGENCY CORE COOLING SYSTEM INJECTION FLOW

Initiation time for ECCS is assumed to be consistent with the Technical Specifications delay time. No spilling of injection flow from the break is assumed for the hot leg break and RCP suction break cases. ECCS injection flow spillage is accounted for in the discharge line break flow analyses. To maximize break flow energy, a high value is selected for ECCS suction temperature during injection.

3.3.2.4 RWST DEPLETION & ECCS SWITCHOVER

The available RWST inventory is minimized. Net flow from the RWST is tracked and switchover (manual action) to recirculation initiated when the RWST level has decreased to a minimum volume corresponding to the low level alarm setpoint. RHR suction is switched first, then the intermediate and high head safety injection pumps. At 50 minutes, RHR flow will be diverted to the auxiliary spray header. Operator actions and delay times for pump realignments are accounted for with conservative allowances.

3.3.2.5 RCP TRIP

RCPs are assumed to trip simultaneously with the main turbine and loss of offsite power.

3.3.2.6 RCP TWO-PHASE MULTIPLIERS

The RELAP pump component model adds a pump head rise to the mixture momentum equation. To account for degradation due to two-phase flow when void fractions are such that little head is developed, homologous difference curves are provided. Appropriate curves for the Catawba and McGuire RCPs, based on experimental data, are included in RELAP5 and used in the analyses. For two-phase flow with void fractions where the pumps are able to develop head, multipliers based on void fraction are applied.

3.3.2.7 S/G POST-TRIP LEVEL CONTROL

Main feedwater flow is assumed to continue until the feedwater isolation valves close. Post-trip S/G level is then controlled by auxiliary feedwater (AFW) flow under manual control simulated by a RELAP5 control system.

3.3.2.8 AUXILIARY FEEDWATER FLOWRATES AND TEMPERATURE

AFW temperature is conservatively maximized for increased primary to secondary heat transfer. One of the motor-driven AFW pumps is unavailable due to loss of one ESF train. The Technical Specifications (TS) value for startup and loading of the available diesel generator is assumed for the delay in availability of the available diesel generator.

3.3.2.9 POST-TRIP S/G PRESSURE CONTROL

MSIVs and main steam PORVs are assumed to close on the containment high-high pressure signal resulting from the LBLOCAs. Operator action is assumed for any subsequent main steam PORV operation to reduce S/G pressure.

3.3.2.10 COLD LEG ACCUMULATOR NITROGEN

Nitrogen used to pressurize the safety injection accumulators is assumed to be discharged into the RCS cold legs and subsequently into the containment.

3.3.2.11 CONTAINMENT BACKPRESSURE

Containment backpressure affects mass and energy release rates during the reflood and post-reflood phases (Ref.: ANS-56.4-1983, paragraph 3.2.4.6.1). However, the RELAP code used for the mass and energy release analysis is not coupled to the GOTHIC code used for the containment pressure analysis. Accordingly, the analysts must provide GOTHIC output to RELAP in an iterative manner. Conservatively high back pressures are used.

3.3.2.12 REFILL ASSUMPTION

A realistic refill time (i.e., 20 seconds) will be assumed in lieu of the standard zero refill time assumption. As discussed in the licensee's May 15, 1995 reply to the staff's Request for Additional Information - Question #6, this assumption has a minor impact on long-term containment response.

3.3.3 COLD LEG RECIRCULATION BOUNDARY CONDITIONS

See 3.3.2.4 for assumptions regarding pump realignments for ECCS suction switchover.

The temperature of the ECCS injection fluid following switchover is dependent on several factors, one of which is the sump temperature. Since sump temperature is computed by GOTHIC and not by RELAP, it must be obtained through a GOTHIC/RELAP iterative process.

3.4 RESULTS OF MASS AND ENERGY RELEASE ANALYSES

Various break locations were analyzed for Catawba Unit 2 (Westinghouse preheater S/G) and for Catawba Unit 1 (BWI feedring S/G). The results show that for each unit, the cold leg RCP discharge leg break case generates the highest total integrated break vapor mass and energy release. The BWI feedring S/G is more limiting than the Westinghouse preheater S/G apparently due to the larger secondary-to-primary heat transfer surface area of the BWI S/G.

Of the postulated RCS break locations, the hot leg break has the least vent path resistance. As a result, that break path results in the highest blowdown mass and energy release rates. However, following blowdown, the pump suction line break has a greater energy release rate due to the fact that the coolant picks up additional heat from an S/G. Much later (at 3000 seconds for Catawba Unit 2, 1800 seconds for Catawba Unit 1), during recirculation, the mass and energy release rates of the cold leg RCP discharge break exceed those of the RCP suction and the hot leg breaks due to ECCS spillage. As a result of these characteristics and of the heat removal capability of the containment cooling systems, the RCP discharge leg break produces the limiting LBLOCA peak pressure.

3.5 HOT LEG RECIRCULATION ALIGNMENT

In a typical LBLOCA scenario ECCS flow is realigned for hot leg injection after 4 to 24 hours of cold leg recirculation. This is done in order to minimize boron concentration in the core region. Using the RELAP5 model, the licensee has analyzed the potential effects on containment mass and energy input of "early" or "accelerated" recirculation realignment for hot leg injection. It was found that this could result in a significant reduction of vapor mass and energy release with resultant lower peak containment pressures. Further modifications to plant systems and procedures would be required to implement this feature.

4.0 STEAM LINE BREAK MASS AND ENERGY RELEASE ANALYSES

4.1 OVERVIEW

With feedwater flow and unaffected S/Gs promptly isolated, steam line break mass and energy releases are considerably less than that of the limiting LOCA. However, since the blowdown fluid is superheated, the lower containment temperature response must be examined to ensure that environmental qualification limits for safety-related equipment are not exceeded.

The steamline break mass and energy releases are analyzed for both the BWI feedring S/Gs and Westinghouse preheater S/G. For the Westinghouse preheater S/Gs, the analyses encompass both the Catawba Unit 2 S/G design which differs from the replacement S/Gs and the replacement S/G design. The Catawba Unit 2 preheater S/G has taller U-tubes and thus uncovers earlier and thus provides a more limiting temperature response.

4.2 THERMAL-HYDRAULIC ANALYSIS

A discussion of the licensee's RETRAN base model is provided in 2.2 above. Modifications to the base model are described below.

4.2.1 MODIFICATIONS TO THE BASE MODEL - BWI FEEDRING S/G

The base model is modified for the BWI feedring S/G analyses as follows:

> The [ ] to minimize heat transfer oscillations seen when a small feedwater flow is added to an essentially dry S/G, and
>
> The riser walls and primary deck, which are [ ] to more accurately model the heat transfer to this metal following tube bundle uncovery, and
>
> To reduce code errors, the elevation of [ ] subvolume.

4.2.2 MODIFICATIONS TO THE BASE MODEL - WESTINGHOUSE S/G

The [ ] control volumes to minimize heat transfer oscillations seen when a small AFW flow is added to an essentially dry S/G.

The [ ] volumes to more accurately model tube bundle uncovery.

The [ ] to enhance code predictions of the two phase flow that occurs during the transient. (The RETRAN-02 equations describe homogenous equilibrium flow. The dynamic slip model provides a means to model phase separation.)

A conductor is added to model the [ ].

The elevation of the [ ]

4.2.3 BREAK MODELING

The main steam line is modeled as [ ] for the turbine. The break model uses the Moody critical flow model which is appropriate for saturated and two-phase upstream flow conditions.

Break sizes of 0.4 ft.^{2}, 0.6 ft.^{2}, 0.86 ft.^{2}, 1.1 ft.^{2}, and 2.4 ft.^{2} were analyzed.

4.3 INITIAL CONDITIONS

Initial conditions specified in the RETRAN models for MSLB mass and energy analyses are selected to maximize the energy/superheat of the break fluid. A discussion of the initial conditions is provided below.

4.3.1 CORE POWER

Initial core power is selected to maximize the S/G primary fluid inlet temperature. Because the plants operate with a constant cold leg temperature program, the inlet/hot-leg temperature rises with power. With a 2% measurement uncertainty the initial power level is 3411 x 102% = 3479.2 Mwt. A hot zero power break is not analyzed as the higher S/G inventory and lower hot leg temperature would result in reduced break flow enthalpy.

4.3.2 RCS TEMPERATURE

An initial RCS temperature corresponding to 102% power plus 4° F for measurement uncertainty is selected.

4.3.3 RCS PRESSURE

The analyses are initialized with a pressurizer pressure at the nominal 102% power value plus an allowance of +60 psi. These are conservative analytical assumptions since higher pressure reduces safety injection flow and thus minimizes S/G inlet temperature resulting in an increased break flow.

4.3.4 PRESSURIZER LEVEL

The pressurizer is assumed to have an initial level corresponding to 100% power plus 9%. The additional inventory mixes with the hot leg inventory and increases S/G inlet temperature.

4.3.5 RCS FLOW RATE

A high primary loop flow rate, 420,000 gpm, is assumed to maximize primary to secondary heat transfer in the S/G.

4.3.6 MIXING IN THE RPV

The return flow from the four cold legs is assumed to [

]

4.3.7 S/G INVENTORY

Initial S/G inventory is minimized at nominal level minus 8%. The reduced inventory assumption provides earlier tube uncovery which in turn increases break flow enthalpy.

4.3.8 S/G TUBE PLUGGING

No tube plugging is assumed. This provides increased heat transfer surface area which, in turn, provides increased break flow enthalpy.

4.4 BOUNDARY CONDITIONS AND RELATED ASSUMPTIONS

This section discusses the boundary conditions and related assumptions for the MSLB mass and energy release analyses. Each item below discusses an individual mass and/or energy source that contributes to or subtracts from the net break flow mass and energy release seen by the containment atmosphere.

4.4.1 RCS PRIMARY SYSTEM WATER AND METAL

The volume of the RCS is assumed to be the calculated cold volume plus a 1% allowance for thermal expansion due to heatup from the cold condition. No tube plugging is assumed.

The metal components in contact with the primary coolant are modeled as heat conductors initially in equilibrium with the primary coolant and with a constant temperature distribution across each conductor to maximize the stored energy.

4.4.2 SECONDARY SYSTEM WATER AND METAL

The volume of the secondary system has not been increased to account for thermal expansion, but the main feedwater flow and initial S/G inventory are selected to ensure a conservative calculation.

The metal components in contact with the secondary coolant are modeled as heat conductors initially in equilibrium with the coolant and with a constant temperature distribution across each conductor to maximize the stored energy. Zero tube plugging is assumed.

4.4.3 CORE STORED ENERGY

Core stored energy is based on end-of-cycle fuel parameters. This is non-conservative with respect to the sensible heat, however, EOC conditions provide increased decay heat and a greater return to power. The overall effect of the EOC assumption is thus conservative.

4.4.4 FISSION HEAT INPUT

Fission heat input is calculated using the RETRAN built-in point kinetics model. The option employed uses one prompt neutron group, six delayed neutron groups, eleven gamma emitters, plus U-239 and Np-239. A low effective delayed neutron fraction and prompt neutron lifetime are selected to maximize the reactivity addition rate. [

] This provides conservatively high feedback effects.

The control rods, with the exception of the most negative rod, are assumed to insert at the time of reactor trip and make the core subcritical by the shutdown margin specified in the Technical Specifications (TS).

Initial boron concentration is assumed to be zero, consistent with the assumed EOC condition. The negative reactivity insertion due to boron injection is modeled by computing the average boron concentration in the [ ] and multiplying the concentration by boron worth to give the reactivity.

4.4.5 DECAY HEAT INPUT

The ANSI/ANS-5.1-1979 decay heat values for EOC conditions are used with a $+2\sigma$ uncertainty.

4.4.6 ROD CONTROL

No rod motion is assumed to occur prior to reactor trip.

4.4.7 SAFETY INJECTION

Safety injection initiates on a containment high pressure trip. The delay associated with emergency bus load sequencing is accounted for. Flow is computed to reflect the changing RCS pressure. Pump characteristics account for degradation in the head-flow performance with time. A single train is assumed to operate. This conservatively minimizes the injection flow. No credit for boron is taken in the initial injection flow until the piping is purged of unborated water.

The RETRAN boron transport model is used in the analysis (See 4.4.4). RWST boron concentration is assumed to be at the TS lower limit minus 1% measurement allowance.

4.4.8 REACTOR TRIP AND MSL ISOLATION

The safety injection signal resulting from containment high pressure provides a reactor trip signal with a 2-second response time. MSL isolation occurs as a result of the containment high-high pressure signal. The modeling is based on use of setpoint limits specified in the TS. A range of reactor trip delays is analyzed for sensitivity effects to investigate the competing effects of late reactor trip (which provides additional fission heat) and early MSIV isolation (which provides earlier S/G tube uncovery).

4.4.9 MAIN FEEDWATER

Main feedwater flow is assumed to remain at its initial flow rate until reactor trip at which time the flow is quickly isolated. These assumptions minimize the time to tube bundle uncovery.

4.4.10 AUXILIARY FEEDWATER

AFW is initiated by the safety injection signal from the reactor trip on high containment pressure. A relatively long (i.e., 60-sec.) delay time is assumed in order to speed tube bundle uncovery of the faulted S/G. All three AFW pumps are assumed to deliver flow to all four S/Gs in order to maximize the available mass. Sensitivity studies were performed to investigate the effects of AFW flow. A conservatively high AFW temperature is assumed.

4.4.11 OFFSITE POWER ASSUMPTION

Offsite power is assumed to remain available. This enables the RCPs to continue operating and thereby maximize primary to secondary heat transfer.

4.5 RESULTS

Mass and energy release data were computed for the various break sizes. The sensitivity studies confirmed that an assumption of maximum AFW flow is conservative, and that the benefits of early trip-isolation outweigh the penalty of the associated delay in tube bundle uncovery.

The data were input to GOTHIC in tabular form as boundary conditions for containment analyses. The GOTHIC results are described in Section 6.0 of this report.

5.0 LARGE-BREAK LOCA CONTAINMENT ANALYSES

5.1 OVERVIEW

The results of the RELAP mass and energy release analyses discussed in Section 3 were input to GOTHIC to determine the long-term containment response. The break flow consists of a steam portion which passes through the ice condenser and melts ice, and a liquid portion which enters the containment sump. An LBLOCA typically has two associated peak pressures, a short-term peak occurring during blowdown, the magnitude of which is primarily affected by blowdown rate, and a later and greater peak, following ice meltout, which is more sensitive to the timing of ice meltout. It is the latter peak that is the primary subject of a longterm analysis. The highest peak containment pressure associated with any LBLOCA establishes "P_a" a parameter defined in 10 CFR 50, Appendix J, for containment leakage testing. During the initial design phase of a facility an estimated P_a is used for purposes of containment design and a 20% margin is applied (Ref.: SRP 6.2.1.1). During the operational phase of the plant P_a is recalculated as necessary to reflect modifications to the facility. P_a must be less than the containment design pressure.

Section 2.3 provided a description of the GOTHIC code and the simulation model including initial and boundary conditions.

5.2/5.3 INITIAL AND BOUNDARY CONDITIONS

See Section 2.3.2

5.4 RESULTS OF GOTHIC LBLOCA ANALYSES

In the FSAR licensing analyses, the limiting peak containment pressures were the results of double-ended pump suction breaks.

	PEAK PRESSURE
CATAWBA	14.05 psig @ 7308 sec
McGUIRE	14.07 psig @ 6454 sec

For the new analyses the pump discharge break is the limiting case and has the following results:

	PEAK PRESSURE
CATAWBA-1 (BWI S/G)	11.77 psig @ 5600 sec
CATAWBA-2 (W S/G)	10.29 psig @ 6800 sec

5.6 MINIMUM CONTAINMENT PRESSURE ANALYSIS

Containment pressure has a positive effect on ECCS reflood performance. For this reason, a conservative minimum containment pressure is calculated and used in the Appendix K analysis. The licensee indicates, in the May 15, 1995 letter, that GOTHIC will be used to revise the minimum containment pressure assumption in Appendix K analyses. To ensure a conservative minimum pressure calculation key analytical assumptions for containment volume, initial pressure and temperature, ice inventory, spray flowrate, service water temperature, spray initiation, bypass leakage, heat sink data, and ice condenser drain droplet size would be changed from those used in the current FSAR analysis and the peak pressure analyses. The changes are consistent with the guidance of ANS 56.4-1983, Para. 4.3.3 for secondary containment analyses and minimum pressure analyses.

6.0 STEAM LINE BREAK CONTAINMENT ANALYSIS

6.1 OVERVIEW

For some facilities, a steam line break may produce the limiting peak accident pressure. However, for ice condenser facilities a LBLOCA is limiting for pressure and steam line breaks are examined for the purpose of defining a bounding temperature profile for use in the design and qualification of systems, structures and components in the lower containment. Steam line breaks produce higher temperatures in the lower containment than LOCAs due to superheating of the break flow. Whereas, the LOCA peak pressure is sensitive to total ice mass, the MSLB temperature is sensitive to ice condenser heat transfer rate. The latter effect is due to the cooling effect of ice condenser drain flow to the lower compartment.

As noted in 2.3, GOTHIC is derived of the COBRA-NC code. COBRA-NC has previously been used for a three-dimensional analysis of a MSLB at Catawba. This analysis is described in WCAP-10988P. That analysis was in turn applied to Watts Bar (Ref.: Watts Bar SER Supplement 7). The licensee compared the GOTHIC analyses with the COBRA-NC analyses and found the temperature predictions to be in good agreement (Ref.: Discussion provided in May 15, 1995 letter responding to staff questions).

6.2 CONTAINMENT MODEL MODIFICATIONS

Section 2.3.2 described the [ ] GOTHIC nodalization for the lower containment for LOCA analyses. However, for the MSLB analyses nodalization is enhanced to model jetting effects.

6.3 INITIAL CONDITIONS

6.3.1 INITIAL BUILDING PRESSURE

A conservatively high initial building pressure of 0.3 psig is assumed.

6.3.2 INITIAL BUILDING TEMPERATURE

Initial temperatures of 120° F in the lower compartment and 100° F in the upper compartment are assumed. These values are consistent with TSs limitations.

6.3.3 ICE CONDENSER DRAIN FLOW

Ice condenser drain droplet discharge into the lower compartment is assumed to be [ ] The GOTHIC model accounts for the fact that the ice drains in the ice compartment must build up a certain head to initiate flow into the lower compartment.

6.4 BOUNDARY CONDITIONS

Because there is no liquid in the break flow, the break flow mass and energy is input to GOTHIC as a single boundary condition. Pressure and flow from the RETRAN analyses are input in tabular form.

6.5 RESULTS

The peak temperature reached in the break compartment is 313° F for a 2.4 $ft.^2$ break. The peak pressure is only about 7 psig. The analyses demonstrate that the EQ temperature limit of 340° F is not exceeded.

7.0 SUMMARY AND CONCLUSIONS

DPC-NE-3004-P describes the Duke Power Company's methodology for simulating the mass and energy releases and containment response to LOCAs and MSLBs for the McGuire and Catawba facilities. The analyses encompass both the current and future S/G installations. The staff found that the licensee's proposed analytical methodology (a) utilizes conservative initial plant conditions, (b) accounts for all significant heat sources and heat sinks, (c) utilizes conservative heat transfer coefficients and correlations, (d) encompasses a complete spectrum of break sizes and locations, (e) encompasses postulated single failures of mitigating equipment, (f) utilizes well-known, well-maintained thermal-hydraulic computer codes having a strong user-base, and (g) utilizes conservative modelling techniques (e.g., nodalization, break flow modelling, ESF pump performance characteristics, operator action credits). The experimental data from the ICTF provided input information needed for GOTHIC to properly calculate long-term ice condenser performance.

The staff has therefore concluded that the DPC-NE-3004-P analytical methodology is acceptable for use in predicting Catawba and McGuire containment pressure and temperature responses to design basis accidents.

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