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DUKE POWER

May 12, 1995

U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Attention: Document Control Desk

Subject: Duke Power Company
McGuire Nuclear Station
Docket Numbers 50-369 and -370
Catawba Nulear Station
Docket Numbers 50-413 and -414
Topical Report DPC-3004-P, "Mass and Energy Release and Containment Response Methodology"; Reponse to NRC Questions

On September 30, 1994, Duke Power Company submitted the subject topical report for review and approval. By letter dated May 3, 1995, the NRC staff requested additional information about the report. Attachment II provides responses to the Staff's questions.

Please note that the responses to several of the questions contain information that Duke considers proprietary. In accordance with 10CFR 2.790, Duke requests that this information be withheld from public disclosure. An affidavit which attests to the proprietary nature of this information is included as Attachment I. Attachment III contains a non-proprietary version of the responses.

The May 3, 1995 request for additional information refers to McGuire Units 1 and 2, and Catawba Unit 1 only. It should be emphasized that, as indicated in the September 30, 1994 letter which submitted the topical report, the analyses contained therein are also applicable to Catawba Unit 2. While the primary purpose of the topical report is to support the replacement of the steam generators at McGuire Units 1 and 2, and Catawba Unit 1, the topical will also be used to support activities at all four units which are unrelated to steam generator replacement; such as a change in ice condenser loading.

Change: NRC PDR 1 Encl w/out Prop

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If you have any questions, or need more information, please call Scott Gewehr at (704) 382-7581.

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Attachment I

Affidavit to Support Proprietary Designation

AFFIDAVIT OF M. S. TUCKMAN

1. I am Senior Vice President, Nuclear Generation Department, Duke Power Company ("Duke"), and as such have the responsibility of reviewing the proprietary information sought to be withheld from public disclosure in connection with nuclear plant licensing, and am authorized to apply for its witholding on behalf of Duke.

2. I am making this affidavit in conformance with the provisions of 10 CFR 2.790 of the regulations of the Nuclear Regulatory Commission ("NRC") and in conjunction with Duke's application for withholding which accompanies this affidavit.

3. I have knowledge of the criteria used by Duke in designating information as proprietary or confidential.

4. Pursuant to the provisions of paragraph (b)(4) of 10 CFR 2.790, the following is furnished for consideration by the NRC in determining whether the information sought to be withheld from public disclosure should be withheld.

(i) The information sought to be withheld from public disclosure is owned by Duke and and has been held in confidence by Duke and its consultants.

(ii) The information is of a type that would customarily be held in confidence by Duke. The information consists of analysis methodology details, analysis results, supporting data, and aspects of development programs, relative to a method of analysis that provides a competitive advantage to Duke.

(iii) The information was transmitted to the NRC in confidence and under the provisions of 10 CFR 2.790, it is to be received in confidence by the NRC.

(iv) The information sought to be protected is not available in public to the best of our knowledge and belief.

(v) The proprietary information sought to be withheld in this submittal is that which is marked in the proprietary version of the report DPC-NE-3004, "Mass and Energy Release and Containment Response Methodology" and supporting documentation, and omitted from the non-proprietary versions.

M.S. Tuckman

M. S. Tuckman

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This information enables Duke to:

(a) Simulate the mass and energy release rates from loss-of-coolant accidents and steam line break accidents in pressurizer water reactors of the Westinghouse design.

(b) Simulate the response of an ice condenser containment design to a high-energy line break inside containment.

(vi) The proprietary information sought to be withheld from public disclosure has substantial commercial value to Duke.

(a) It allows Duke to reduce vendor and consultant expenses associated with supporting the operation and licensing of nuclear power plants.

(b) Duke intends to sell the information to nuclear utilities, vendors, and consultants for the purpose of supporting the operation and licensing of nuclear power plants.

(c) The subject information could only be duplicated by competitors at similar expense to that incurred by Duke.

5. Public disclosure of this information is likely to cause harm to Duke because it would allow competitors in the nuclear industry to benefit from the results of a significant development program without requiring a commensurate expense or allowing Duke to recoup a portion of its expenditures or benefit from the sale of the information.

M. S. Tuckman

M. S. Tuckman

(continued)

AFFIDAVIT OF M. S. TUCKMAN (Page 3)

M. S. Tuckman, being duly sworn, on his oath deposes and says that he is the person who subscribed his name to the foregoing statement, and that the matters and facts set forth in the statement are true.

M. S. Tuckman

M. S. Tuckman

Sworn to and subscribed before me this 15th day of May, 1995. Witness my hand and official seal.

Mary P. Helms

Notary Public

My commission expires JAN 22, 1996

Attachment III
Responses to Questions
(Non-Proprietary Version)

Response to NRC Questions on DPC-NE-3004

Question 1:

Figure 2.1.3-12 of the topical report illustrates a difference between the FSAR analysis and RELAP analysis for the mass and energy release rate during the period from 300 to 1500 seconds. In Section 2.1.3 of the topical report, this difference is attributed to differences in intact S/G heat transfer due to RELAP using mechanistic modeling in which heat transfer is dependent on mass flow rate. Please provide a more detailed discussion of these differences.

A description of the FSAR methodology "Westinghouse Mass and Energy Release Data for Containment Design", WCAP-8264-P-A (Proprietary) and WCAP-8312 (Non-Proprietary) is provided in the NRC letter of March 12, 1975 (reference 1). Also the NRC staff letter of February 17, 1987 (reference 2) provides an evaluation of WCAP-10325, "Westinghouse LOCA Mass and Energy Release Model for Containment Design" relating to the vendor methodology. To what extent are the differences to which you refer related to issues discussed in these two references.

Response:

The March 12, 1975 safety evaluation provides details regarding the treatment of heat transfer from the steam generators during reflood. The W REFLOOD code is used to calculate the mass and energy release during this portion of the analysis. The W REFLOOD code utilizes a loop hydraulic resistance model and an energy balance model. The steam generators are assumed to be cooled to the temperature of saturated steam at the containment pressure during the reflood phase of the transient. This process is determined to have been completed after 960 seconds, as stated on page 6. This assumption is based upon "the maximum steam flow based upon the hydraulic resistance and steam generator heat transfer." During the post-reflood phase of the transient, heat transfer from the steam generators is determined by the containment depressurization rate. The February 17, 1987 staff evaluation describes changes to the original methodology. The change made with respect to steam generator heat transfer is to assume that only saturated steam exits the steam generators. Eliminating the release of superheated steam from the steam generator side of the break increases the overall steam mass flow released to containment.

In the McGuire/Catawba RELAP5 model, heat transfer from the steam generators is determined, in part, by the mass flow through each steam generator. The flow split between the broken and intact loops is determined by the hydraulic resistances in the model, and is not modified in the analysis. The broken steam generator rapidly cools down to the temperature of saturated steam in the RCS as expected, due to high flow rates in the affected loop. Heat transfer from the intact steam generators is limited by the mass flow through the primary side of the tube bundle and the degree of superheat that can be achieved. The steam generator secondary side initial conditions are specified to maximize the available stored energy, thereby maximizing the heat transfer to the primary and the mass and energy release out the break.

Question 2:

Referring to the discussion of the ice condenser heat transfer correlations available in GOTHIC-4.0/DUKE (pg. 2-16 of DPC-NE-3004), describe which correlations and/or heat transfer coefficient values are actually selected. Indicate if an upper limit value for heat transfer coefficient is input to the program. Refer to the staff's SER "Staff Evaluation of the Tests conducted to Demonstrate the Functional Adequacy of the Ice Condenser Design," of April 25, 1974, (PDR #8603040079) in which an upper limit coefficient of 10,000 BTU/hr-sq.ft was established.

Response:

The [ ] is used to determine a heat transfer coefficient for the ice condenser heat transfer equations in all GOTHIC-4.0/DUKE containment analyses discussed in DPC-NE-3004. This correlation [ ] provided the best matches with the data generated in the Ice Condenser Test Facility heat transfer tests. Heat transfer from the steam to the ice is calculated in GOTHIC for each node containing ice. Separate heat transfer coefficients are calculated for each node. This is different then the LOTIC-1 ice condenser heat transfer logic, where the ice melt rate (and hence, the heat transfer coefficient to the ice) is dependent on the steam flow rate into the condenser, assuming constant meltwater temperatures and complete condensation of steam prior to ice meltout. Even if one or more nodes exceeded the 10,000 BTU/hr-sq.ft-F heat transfer coefficient value (the limit imposed on ice condenser heat transfer in LOTIC-1, per PDR #8603040079) in a GOTHIC analysis, the heat transfer coefficient for the entire ice condenser would still be well below the value of 10,000.

There is [ ] as there is in the Uchida correlation. Therefore, it is possible for the heat transfer coefficient [ ] to exceed the 10,000 BTU/hr-sq.ft-F value in a single node. In determining what the maximum heat transfer coefficient is in the DPC-NE-3004 GOTHIC analyses, it is observed that the [ ] The highest velocities through the ice condenser occur during the very early stages of blowdown. At this time (0-2 seconds) the highest ice melt rates are also observed. The highest heat transfer coefficient for any ice condenser node was on the order of [ ] BTU/hr-sq.ft-F, directly above the break, at the lowest elevation in the ice condenser. The flow velocity through this node was the highest of any ice condenser location at any time during the transient. The 10,000 value is not approached in any transient.

Question 3:

Section 2.3.2 discusses the spray droplet size used, which is significantly less than the size used for Oconee. Was this decrease intentional? If possible relate your selection to the data in WCAP-8258, "SPRAYCO Model 17143A Nozzle Spray Drop-Size Distribution".

Response:

The difference in the assumed droplet size for the spray droplets between the McGuire/Catawba and Oconee containment models was intentional. The spray headers are required at McGuire and Catawba to produce a droplet size spectrum with a mean diameter of less than 700 μm. Figure 6-195 of the McGuire FSAR shows the distribution produced by the McGuire spray headers. This figure shows that the assumption of 700 μm for an average droplet size is therefore conservative.

At Oconee, there is no corresponding requirement for a maximum spray droplet size. An average droplet size of 700 μm is assumed in the Oconee FATHOMS base model. This is increased to 7000 μm in the long-term containment response analyses, which introduces additional conservatism. The FATHOMS containment response is very insensitive to changes above the 700 μm size.

Question 4:

A low initial containment temperature is generally conservative for the blowdown peak pressure determination. However, if the limiting peak is relatively late in the event, might a high initial containment be conservative due to reduced heat structure heat absorption capacity. Provide assurance that the initial conditions of 2.3.2 are indeed conservative for the LBLOCA analyses.

Response:

A sensitivity run was conducted for the CNS-1 cold leg pump discharge break with the initial lower and upper containment temperatures increased to the upper Tech Spec limits of 120 and 100 °F, respectively. (The ice condenser initial temperature remained unchanged at 30 °F.) The peak pressure decreased in the sensitivity case from 11.76 psig to 10.79 psig. The effect of an increased air mass (low initial temperature) on the eventual peak pressure following a LOCA is greater than that of reduced heat structure heat absorption capacity (high initial temperature), as stated in Section 5.2. It is expected that this trend would hold for all MNS/CNS cases.

Question 5:

It would be useful to include, for each LBLOCA analysis, a table identifying the peak pressure and time of peak pressure for each identifiable peak (similar to those included in DPC-NE-3003P).

Response:

See Table A.

Question 6:

Section 3.3.2.12 states that the long-term analyses are insensitive to a 20 second refill assumption. Since we have no other information readily available to confirm this, and since ANS-56.4-1983 states that justification should be provided for use of a non-zero refill time in long-term analyses, please explain the reason for the non-zero refill assumption.

Response:

The 20 second refill time mentioned in Section 3.3.2.12 is obtained from page 3-3 of NSAC/86 (Reference 3-10). The refill phase of a large break LOCA transient is defined as the period of time between the end of blowdown and the beginning of reflood. The reflood phase of the transient begins when the mixture level in the reactor vessel lower plenum reaches the core inlet. The timing of refill and reflood can be significant in a dry containment design because the peak containment pressure occurs during the early reflood period. For the ice condenser containment design the refill phase and associated phenomena are short-term concerns and the peak pressure occurs much later. Therefore the timing of refill has no effect on the peak pressure response.

Question 7:

Section 3.4 postulates that the LBLOCA case having the greatest integrated steam release will produce the limiting peak pressure (i.e., insensitivity to timing effects). Has this phenomena been shown by analysis?

Response:

Analyses have been performed for each of the three break locations for the durations necessary to establish the long-term mass and energy release trends, and to identify the limiting location. The

key considerations are the time of ice meltout, and the steaming rate out the break after ice meltout. The steaming rate out the break is maximized for the pump discharge break, where ECCS is lost due to spilling out the break and less steam condensation results.(Refer to the response to Question 7 for details) The containment pressure is higher for the pump discharge break at the time that the long-term steaming trends are established. Therefore, it has been shown by analysis that the pump discharge break is the limiting location.

Question 8:

Please explain the phenomena of the 3000 sec crossover (3.4.1.4) where spilled ECCS fluid causes the RCP discharge case M&E to be greater than the RCP suction case.

Response:

Both the pump discharge break case and pump suction case are initiated from the same initial conditions. The initial differences in the integrated break vapor mass and energy release observed in Figure 3.4.1.4-2 and Figure 3.4.1.4-4 are a result of break location specific blowdown phenomena. These differences are consistent with those presented in the current FSAR analyses. The integrated steam release for the pump suction case following blowdown initially exceeds that from the pump discharge case. The primary cause for this difference is implicit to the break location assumptions. For the pump discharge break location, all ECCS to the broken loop is assumed to be spilled directly to containment and is therefore not available for condensing steam before it reaches the break. ECCS flow is not spilled directly to containment for the pump suction case, allowing steam to be condensed in the broken cold leg prior to reaching the break. The impact of this difference is that the steam release for the pump discharge case eventually exceeds that of the pump suction case. The timing of the crossover is approximately 3000 seconds for the case illustrated in Figures 3.4.1.4-2 and 3.4.1.4-4. The same phenomena is demonstrated in Figure 3.4.2.4-2 and Figure 3.4.2.4-4, although the timing of the crossover differs.

Question 9:

Paragraph 4.4 states that primary and secondary metal structures are initially in equilibrium with the surrounding coolant, with a constant temperature distribution. Explain how this assumption is applied to structures in contact with both primary and secondary coolant.

Response:

This assumption applies to structures that are only in contact with either the primary or secondary coolant, but not both. The only structures that are physically in contact with both the primary and secondary coolant are the steam generator tubes and the tubesheet. The tubesheet metal is at the temperature of the primary coolant flowing through it. The small amount of primary-to-secondary heat transfer that occurs at the top of the tubesheet is neglected. Therefore, the tubesheet is modeled by a one-sided conductor connected to the primary side. The steam generator tubes are modeled with a linear temperature gradient across the heat conductors, which is determined by the adjacent water temperatures.

Question 10:

Referring to Fig 4-24, explain why such a large percentage of the total AFW flow is delivered to the intact S/Gs (instead of the faulted S/G which is at a much lower pressure). At what time is AFW flow to the faulted S/G terminated? Explain the differences between the AFW flow rate curves for W and BWI S/Gs (Figs 4.5.3 and 4.5.17).

Response:

The AFW flow rates plotted in Figures 4.5.3 and 4.5.17 represent the total AFW flow to the three intact steam generators and the AFW flow to the single faulted steam generator. Thus, AFW flow to the faulted steam generator (~ 1200 gpm) is nearly double that seen by each intact steam generator (~ 2000 gpm total or 670 gpm each).

AFW flow to the faulted steam generator is not terminated prior to the end of the mass and energy release analysis. Isolating AFW flow to the faulted steam generator would non-conservatively terminate the flow of steam through the break. Thus, the mass and energy release is analyzed with continued AFW flow past the time of peak containment temperature.

AFW flow is specified in the RETRAN input deck as a function of steam generator pressure with an enthalpy corresponding to a given temperature. For the Westinghouse preheater steam generators, a purge volume of hot water is assumed to be delivered before the cold AFW reaches the steam generators. For break sizes above 0.6 ft^2, pressure in the faulted steam generator drops below the saturation pressure of the hot AFW before the purge of the hot AFW is completed. When pressure in the faulted generator falls below the saturation pressure of the hot AFW, it is assumed that the water in the AFW piping flashes and is added to the faulted generator as steam. Thus, a sharp increase in the volumetric AFW flow rate is seen between 65 and 85 seconds in Figure 4.5.17 due to the much lower density of the steam. Figure 1 shows the AFW mass flow rate to the faulted steam generator that corresponds to the volumetric flow rate shown in Figure 4.5.17.

Question 11:

For MSLB temperature analysis, it is the reviewer's understanding that the limiting break size is normally that at which entrainment starts to occur. Provide a discussion of how entrainment is accounted for in the RETRAN model for large MSLBs. Also, can you provide the staff with information regarding the sensitivity of MSLB temperature to $\lambda(T)$[ref page 2-16 of DPC-3004].

Response:

From a peak containment temperature perspective, it is conservative to assume that all of the break flow is released in the form of steam. Therefore, the steam dome volume is modeled as a bubble rise volume with a very large separation velocity. This provides nearly instantaneous and complete separation between the liquid and vapor phases in that volume and precludes any liquid from being entrained in the flow leaving through the steam generator outlet nozzle.

The [ ] assumed in the GOTHIC containment model has a negligible impact on the lower containment temperature response following a MSLB. The peak temperature in lower containment is reached in the first few minutes following a steam line break. This is well before the time of ice meltout. [ ] It is also well before the time when ice above any one sector of lower containment would be melted. [ ]

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Question 12:

Section 5.2 indicates that the initial ice mass is the Technical Specification (TS) value (which includes a sublimation allowance). Is the sublimation ice mass allowance assumed to be available for containment heat removal? Is the ice mass assumption for McGuire greater than FSAR/LOTIC-1 values?

Response:

The allowance for ice sublimation is not assumed to be available in the GOTHIC containment model. Although this sublimed ice may still be present in the condenser, in the form of water vapor/frost, no credit is taken for it in the GOTHIC model. The ice mass in the GOTHIC analyses is 13.9% below the Tech Spec value of 2.475 E6 lbm. The GOTHIC ice mass assumptions in both McGuire and Catawba analyses are the same as those in the most recent respective FSAR analyses, performed with LOTIC-1.

Question 13:

The RETRAN-02 code has been approved by the staff for generic use in 1984 and 1991 (references 3 and 4). Section II.C of the Safety Evaluation (SE) identifies "General Limitations" regarding use of the code. Provide a general discussion of the extent, if any, to which your use of RETRAN-02 deviates from the limitations.

Response:

The conclusions regarding the limitations specific to the steam line break mass and energy release analysis are given below. The numbers and letters correspond to those listed in the SERs.

<u>Limitations from the MOD5.0 SER:</u>

2.1 The general transport model is used to simulate the injection of boron into the primary system. The conservative application of this model with respect to purge volumes and assumed boron concentrations is discussed in Section 5.3.2.5 of DPC-NE-3001.

2.2 The 1979 ANS Standard decay heat model is used in the analysis with an added two-sigma uncertainty to bound all uncertainties associated with the input parameters to the decay heat model.

<u>Limitations from the MOD2.0 SER:</u>

i, n3. [

] The mass and energy release results are relatively unaffected by the use of this model since liquid is not allowed to exit the steam generator as discussed in the response to Question 10 above. The AFW flow rate and primary-to-secondary heat transfer are of primary importance in this analysis, and their conservative application produces a limiting mass and energy release.

Question 14:

Provide a general discussion of the extent to which DPC's use of the RELAP-5 code deviates, if any, from the recommended uses and practices described in the Code Manual (NUREG/CR-5535, Vol. 5 User's Guidelines).

Response:

In general, the practices presented in the User's Guidelines (Vol. 5 of the RELAP5/MOD3 Code Manual) represent the basic information required to perform transient analyses. These practices are consistent with those used at Duke Power. A review of Vol. 5 identified three instances where our use of RELAP5 differed from the author's recommendations. These instances are discussed below.

In Section 4.6.8.1, it is recommended that the frictional torque specified for a pump component be divided equally between TF0 and TF2. The frictional torque input to the McGuire/Catawba RELAP5 model is specified entirely as TF2. The pump model used in the RELAP5 model is the same as that described in the Duke Power RETRAN model for McGuire/Catawba, which is documented in the NRC-approved topical report DPC-NE-3000-PA.

In Section 4.6.9, the author of Vol.5 recommends against using the multiple junction component, based upon the potential for confusion in identifying the location of a specific junction in the output. The multiple junction component is used in the secondary side of the tube bundle in the BWI FSG model.

In Section 5.1.8, the final recommendations made in this section are to use a two-component representation for the core if simulation of the high-powered fuel rod behavior is important in meeting the analysis objectives, to not model crossflow in the downcomer, and use a simplified system nodalization if possible. The first recommendation does not apply for a mass and energy release analysis. In the McGuire/Catawba RELAP5 large break LOCA model, the reactor vessel

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were both necessary to accurately model the LOCA mass and energy release.

Question 15:

Provide a discussion of the significant reasons for the change in the break location of the limiting LBLOCA from the pump suction to pump discharge.

Response:

The mass and energy release analyses in the current FSAR encompass the blowdown phase for the hot leg break, the blowdown and reflood phase for the pump discharge break and long-term for the pump suction break case only. If the limiting transient were to be selected based solely on the current FSAR analyses, the same conclusion would be reached. The RELAP5 analyses presented for each break location are analyzed further out in time, thereby gaining additional insights into the long-term response for different break locations. A crossover in the integrated mass and energy release occurs for the two cold leg break locations during the cold leg recirculation phase of the transient. The phenomena behind the crossover in the integrated mass and energy release are discussed in the response to Question #7 above.

Question 16:

The new LBLOCA analyses produce significantly lower peak pressure. This can be attributed to both the mass and energy release modeling and the containment modeling. Has the GOTHIC model been run with the FSAR mass and energy data to determine the relative contribution of each?

Response:

The GOTHIC runs conducted to benchmark the containment simulation model, described in Section 2.3.3, use the FSAR mass and energy release data. As mentioned at the bottom of p. 2-22, there are some uncertainties as to the exact data used in some phases of the LOTIC-1 FSAR analyses. The results of this run indicate approximately how much the change in containment analysis methods alone improved the peak pressure result. In this GOTHIC run, a peak pressure of 12.90 psig was calculated. Compared with the FSAR result of 14.07 psig, a decrease of 1.17 psig is achieved in the peak containment pressure when utilizing GOTHIC alone, without new mass and energy release data.

The CNS-2 cold leg pump discharge run documented in Section 5.4.2 utilizes RELAP5 mass and energy release data and the GOTHIC containment analysis model. In addition, the CNS-2 analysis is for the Westinghouse preheater steam generator type; the impact of the BWI feedring S/G is not present in this analysis. The results of this analysis shows the significance of the RELAP5/GOTHIC analysis package, without the differing S/G geometry effect. As documented in Section 5.4.2, the CNS-2 cold leg pump discharge analysis resulted in a peak containment pressure of 10.29 psig. When subtracted from the Catawba FSAR peak pressure of 14.05 psig, a decrease of 3.76 psig is achieved when utilizing the RELAP5/GOTHIC analysis package. This is roughly 3 times the decrease when using GOTHIC alone, as mentioned above. Therefore, the relative contribution of each code may be generalized as 2 parts RELAP5, 1 part GOTHIC.

Question 17:

Describe DPC's intentions with respect to Appendix K minimum pressure analyses. Does DPC seek approval for use of GOTHIC to establish a higher Appendix K minimum pressure? Would structural heat transfer coefficients be consistent with the guidance of ANS-56.4-1983? Is the 70^{o}F spray temperature Key Assumption in Section 5.6 conservative for winter conditions?

Response:

The minimum containment pressure analysis methodology described in Section 5.6 will be used to calculate the minimum containment backpressure boundary condition for future LOCA PCT analyses. The GOTHIC model and inputs used for the peak pressure analyses will be significantly modified as described in order to conservatively predict a minimum pressure response. The predicted pressure response may be higher or lower than the current FSAR analysis, depending on the initial and boundary conditions assumed.

Due to the large heat sink effect of the ice condenser, the modeling of structural heat sinks plays a much smaller role than in conventional dry containment design. A +5% allowance for structural heat sinks is included. The ANS-56.4-1983 heat transfer coefficient modeling guidance is unnecessary, and is not used.

The 70^{o}F spray temperature assumption is consistent with the minimum temperature specified in Technical Specification 3.5.4 for the Catawba Nuclear Station and Technical Specification 3.5.5 for the McGuire Nuclear Station.

Question 18:

Section 6.1 states that "once the affected steam generator is isolated, the release of steam to the containment is essentially finished." Is this statement intended to refer to the feedwater flow?

Response:

Yes. Main feedwater flow is automatically isolated, and once the auxiliary feedwater flow is manually isolated from the faulted generator, the steam release to containment is finished.

Question 19:

From Section 6.5 it is not clear as to what was the highest peak temperature among all the cells of the lower compartment. The fact that the break volume was cooled by jetting raises the question as to whether other areas were subjected to increased local temperatures. What is the peak temperature experienced in the lower compartment?

Response:

The steam line break is assumed to occur in the node closest to the crane wall, pointed towards the crane wall. As discussed in Section 6.2, there are [ ] the GOTHIC steam line break containment model. Due to the high pressure at which the steam is injected through the break in the early stages of the transient, all [ ] are at essentially the same temperature. The cooler air coming in behind the break due to the jetting effect is forced along with the steam to the next node downstream of the break until hitting the crane wall. All [ ] in this sector of lower containment, are at the same pressure, temperature, and steam/air concentration. Upon hitting the crane wall, the steam/air jet is re-directed in all directions and mixes around containment. The break compartment peak temperature is referred to as the peak temperature in lower containment, although the nodes adjacent to the break compartment in the direction of the break flow are at the same temperature.

This does not hold true throughout the transient, however. As the steam line pressure decreases, the jetting force decreases, and the jetting flows decrease relative to the break flow. The node adjacent to the crane wall is slightly hotter than the node at the break location later in the transient. However, the peak temperature for the entire transient has already been reached by the time this jetting decrease occurs.

Question 20:

The Safety Evaluation Report, NUREG-0847 Supplement 7, for Watts Bar describes COBRA-NC analyses performed by Westinghouse for Catawba and subsequently applied also to Watts Bar. TVA found that the MSLB hot spot locations are not locations containing environmentally-qualified equipment. To what degree can DPC state that the Catawba COBRA-NC analyses are consistent with DPC GOTHIC analyses? To what extent can DPC state that the COBRA-NC methods described in WCAP-10988-P (proprietary version) and WCAP-10989 (non-proprietary version) are consistent with DPC's GOTHIC code and models?

Response:

The location of environmentally-qualified equipment within the Catawba/McGuire containment buildings would not be significant since the maximum temperatures calculated by GOTHIC following a MSLB do not exceed the EQ limit of 340°F.

The GOTHIC analyses are consistent with the COBRA-NC results in the Westinghouse WCAP-10988-P insofar as the COBRA-NC code is a precursor of the GOTHIC code. The prediction of the jetting effect and its subsequent cooling of lower containment are consistent between the analyses, although the magnitude of the cooling may be different between analyses. Other consistencies include the closing of the ice condenser doors above the break due to the jet-induced pressure decrease. The maximum bulk temperature of 291°F in the COBRA-NC analysis (Model 2) is very close to the 297°F peak in the GOTHIC analysis. The peak temperature is about 15°F warmer in the COBRA-NC results, which can be attributed to the smaller node sizes used.

Many other modeling factors are involved which could impact the differences in flow patterns and longer-term temperature increase in the COBRA-NC analysis which is not present in the GOTHIC analysis. Among these factors is the[

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The GOTHIC code itself is technically more advanced than COBRA-NC. The separate set of energy equations present in GOTHIC for the droplet phase could have a major impact on the analysis results. The modeling of flow paths (junctions) in GOTHIC is completely different than COBRA-NC, in which these flow paths were simply left as gaps in the calculational mesh. The interfacial heat transfer routines have been fine-tuned with an additional ten years of comparisons with test data during the development from COBRA-NC to FATHOMS to GOTHIC 4.0. All differences between the COBRA-NC and GOTHIC codes would provide a higher degree of accuracy and certainty with the GOTHIC code. The conservatism required to ensure a conservatively high peak building temperature is applied through the selection of conservative initial and boundary conditions throughout all GOTHIC analyses, as well as in the mass and energy release calculations.

Question 21:

McGuire Unit 2 LER 85-29 of October 31, 1985 describes how spray (NS) pump switchover initiation time and delay interval could cause LOCA peak pressure to exceed the containment design pressure depending on initial FWST level. Does the DPC-NE-3004-P methodology account for the concerns identified in this LER?

Response:

The primary concern identified is the potential for ice meltout prior to completing the transfer of the containment spray pump suction from the FWST to the containment sump. This would result in a time frame during which no ice remained without containment spray. The time at which auxiliary containment spray is initiated was changed from 60 minutes to 50 minutes to address this concern. The same timing assumptions regarding auxiliary containment spray initiation are made in the DPC-NE-3004 methodology.

Westinghouse Preheater SG 1.4 Ft2 Steam Line Break Mass and Energy Release

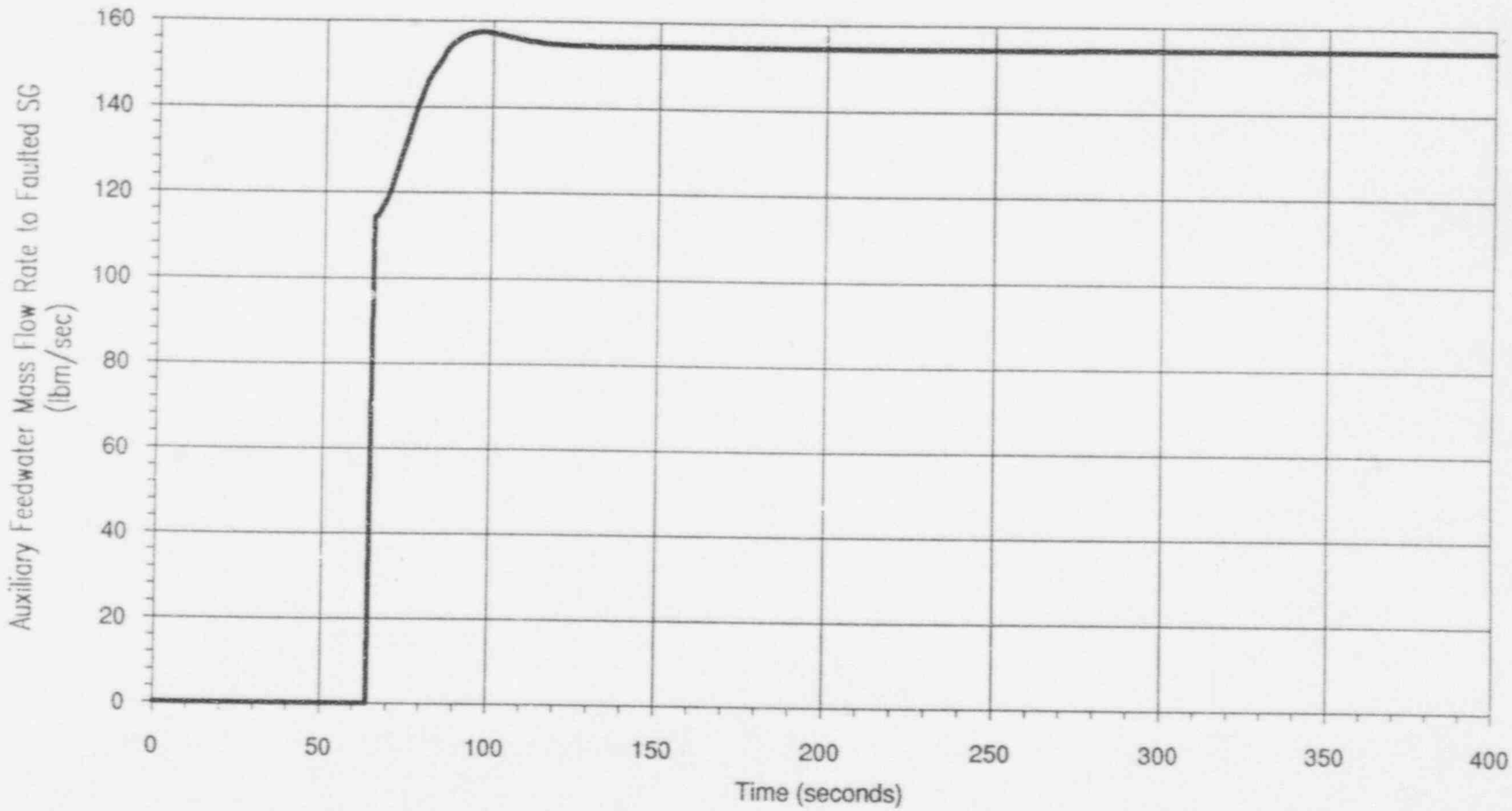

Figure 1

TABLE A
DPC-NE-3004 GOTHIC Analyses - Peak Pressures

Title	Figure No.	Peak pressure (psig)	Time of peak pressure (sec)
MNS FSAR CLPS Break	2.3.3-1	12.90	4900
CNS-1 Pump Discharge Break	5.4.4.1-1	11.77	5600
CNS-2 Pump Discharge Break	5.4.2.1-1	10.29	6850

Note: Only cases run past the time of ice meltout have peak pressures reported. Cases which were not the limiting cases were not run to this point. Initial peaks (within a few seconds of the end of initial NC System blowdown) are not reported, as the maximum pressure in all cases is expected to be reached after ice meltout.