Duke Power®

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July 25, 2005

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

Subject: Duke Energy Corporation
Catawba Nuclear Station, Units 1 and 2
Docket Numbers 50-413 and 50-414
Proposed Change to Technical Specification (TS) 3.7.9, Standby Nuclear Service Water Pond (SNSWP)

Pursuant to 10 CFR 50.4, 10 CFR 50.90, and 10 CFR 50.91(a)(5), Duke Energy Corporation is submitting the attached proposed revision to the temperature limit for the SNSWP. This amendment is being submitted on an emergency basis.

As a result of extended hot and humid weather conditions experienced at the Catawba site, the SNSWP is approaching the 91.5°F TS temperature limit. On July 23, 2005, the peak SNSWP temperature was 87.7°F. Based on predictions of continued high ambient air temperatures, high humidity, and no appreciable precipitation, the limit could be exceeded by early August 2005. Additionally, the SNSWP has historically experienced temperature excursions of up to 3°F following late afternoon rain showers as a result of rain runoff absorbing heat from the SNSWP surroundings. The details of this amendment request, including mitigating actions taken to date to decrease the SNSWP temperature trend, are explained fully in the attachments.

This emergency situation is being caused by environmental factors beyond the control of Catawba Nuclear Station. Therefore, Duke Energy Corporation requests approval of this license amendment application on an emergency basis by July 29, 2005 in order to avoid an unnecessary shutdown of Units 1 and 2.

Attachment 1 provides a marked copy of the affected TS and Bases pages for Catawba, showing the proposed changes. Attachment 2 contains reprinted pages of the affected TS and Bases pages for Catawba. Attachment 3 provides the technical justification, No

A001

Significant Hazards Consideration Determination, and Environmental Analysis that revising the SNSWP temperature limit as indicated does not create any safety concerns. In accordance with Duke Energy Corporation administrative procedures and the Quality Assurance Program Topical Report, this proposed amendment has been previously reviewed and approved by the Catawba Plant Operations Review Committee and the Corporate Nuclear Safety Review Board.

Implementation of this amendment request will require changes to the Catawba Updated Final Safety Analysis Report (UFSAR). Necessary UFSAR changes will be submitted to the NRC in accordance with 10 CFR 50.71(e).

Pursuant to 10 CFR 50.91, a copy of this proposed amendment is being sent to the appropriate State of South Carolina official.

Should you have any questions concerning this information, please call L.J. Rudy at (803) 831-3084.

Very truly yours,

D.M. Jamil

Attachments

LJR/s

D.M. Jamil affirms that he is the person who subscribed his name to the foregoing statement, and that all the matters and facts set forth herein are true and correct to the best of his knowledge.

D.M. Jamil, Site Vice President

Subscribed and sworn to me: 7-25-2005
Date

Notary Public

My commission expires: 7-10-2012
Date

SEAL

xc (with attachments):

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ATTACHMENT 1

MARKED-UP TS AND BASES PAGES FOR CATAWBA

3.7 PLANT SYSTEMS

3.7.9 Standby Nuclear Service Water Pond (SNSWP)

LCO 3.7.9 The SNSWP shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. SNSWP inoperable.	A.1 Be in MODE 3. AND A.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.9.1	Verify water level of SNSWP is ≥ 571 ft mean sea level.	24 hours
SR 3.7.9.2	------------------------------------NOTE------------------------------------ Only required to be performed during the months of July, August, and September. --- Verify average water temperature of SNSWP is ≤ 91.5°F at an elevation of 568 ft. in SNSWP. 95	24 hours
SR 3.7.9.3	Verify, by visual inspection, no abnormal degradation, erosion, or excessive seepage of the SNSWP dam.	12 months

B 3.7 PLANT SYSTEMS

B 3.7.9 Standby Nuclear Service Water Pond (SNSWP)

BASES

BACKGROUND

The SNSWP provides a heat sink for processing and operating heat from safety related components during a transient or accident, as well as during normal operation. This is done by utilizing the Nuclear Service Water System (NSWS) and the Component Cooling Water (CCW) System.

The SNSWP has been defined as the water source, including necessary retaining structure, but not including the cooling water system intake structures as discussed in the UFSAR, Section 9.2 (Ref. 1). The principal functions of the SNSWP are the dissipation of sensible heat during normal operation, and dissipation of residual and sensible heat after an accident or normal operation.

The basic performance requirements are that a 30 day supply of water be available, and that the design basis temperatures of safety related equipment not be exceeded.

Additional information on the design and operation of the SNSWP can be found in Reference 1.

APPLICABLE SAFETY ANALYSES

The SNSWP is the sink for heat removed from the reactor core following all accidents and anticipated operational occurrences in which the unit is cooled down and placed on residual heat removal (RHR) operation.

INSERT 1

The peak containment pressure analysis assumes the NSWS flow to the Containment Spray and Component Cooling Water heat exchangers has a temperature of 92°F. (100) To ensure that this condition is not exceeded, and to ensure that long term NSWS temperature does not exceed the 100°F design basis of NSWS components a limit of 91.5°F (95) is conservatively observed for the SNSWP. This temperature is important in that it, in part, determines the capacity for energy removal from containment. The peak containment pressure occurs when energy addition to containment (core decay heat) is balanced by energy removal from these heat exchangers. This balance is reached far out in time, INSERT 1 for APPLICABLE SAFETY ANALYSES:

NSWS temperature influences containment pressure following a Loss of Coolant Accident and offsite dose following a Main Steam Line Break. The containment peak pressure analysis can accommodate NSWS temperatures up to 100°F. However, offsite dose requirements limit NSWS temperature to 95.5°F to prevent extending Reactor Coolant System cooldown time to a value greater than that currently assumed. Therefore, offsite dose following a Main Steam Line Break is more limiting with respect to NSWS temperature.

BASES

APPLICABLE SAFETY ANALYSES (continued)

after the transition from injection to cold leg recirculation and after ice melt. Because of the effectiveness of the ice bed in condensing the steam which passes through it, containment pressure is insensitive to small variations in containment spray temperature prior to ice meltout.

Long term equipment qualification of safety related components required to mitigate the accident is based on a continuous, maximum NSWS supply temperature of 100°F.

To ensure that the NSWS initial temperature assumptions in the peak containment pressure analysis are met, Lake Wylie temperature is also monitored. During periods of time while Lake Wylie temperature is greater than 92°F, the emergency procedure for transfer of Emergency Core Cooling System (ECCS) flow paths to cold leg recirculation directs the operator to align at least one train of containment spray to be cooled by a loop of NSWS which is aligned to the SNSWP. Swapover to the SNSWP is required at 92°F rather than 91.5°F because Lake Wylie is not subject to subsequent heatup due to recirculation, as is the SNSWP. Therefore, the 100°F design basis maximum temperature is not approached.

limiting

95.5

95

The operating limits are based on conservative heat transfer analyses for the worst case LOCA. Reference 1 provides the details of the assumptions used in the analysis. The SNSWP is designed in accordance with Regulatory Guide 1.27 (Ref. 2), which requires a 30 day supply of cooling water in the SNSWP.

accident

The SNSWP satisfies Criterion 3 of 10 CFR 50.36 (Ref. 3).

LCO

The SNSWP is required to be OPERABLE and is considered OPERABLE if it contains a sufficient volume of water at or below the maximum temperature that would allow the NSWS to operate for at least 30 days following the design basis LOCA without the loss of net positive suction head (NPSH), and without exceeding the maximum design temperature of the equipment served by the NSWS. To meet this condition, the SNSWP temperature should not exceed 91.5°F at 568 ft mean sea level and the level should not fall below 571 ft mean sea level during normal unit operation.

accident

95

BASES

APPLICABILITY

In MODES 1, 2, 3, and 4, the SNSWP is required to support the OPERABILITY of the equipment serviced by the SNSWP and required to be OPERABLE in these MODES.

In MODE 5 or 6, the requirements of the SNSWP are determined by the systems it supports.

ACTIONS

A.1

If the SNSWP is inoperable the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours and in MODE 5 within 36 hours.

The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.9.1

This SR verifies that adequate long term (30 day) cooling can be maintained. The specified level also ensures that sufficient NPSH is available to operate the NSWS pumps. The 24 hour Frequency is based on operating experience related to trending of the parameter variations during the applicable MODES. This SR verifies that the SNSWP water level is ≥ 571 ft mean sea level.

SR 3.7.9.2

This SR verifies that the NSWS is available to cool the CCW System to at least its maximum design temperature with the maximum accident or normal design heat loads for 30 days following a Design Basis Accident. The 24 hour Frequency is based on operating experience related to trending of the parameter variations during the applicable MODES. This SR verifies that the average water temperature of the SNSWP is ≤ 91.5°F (95). The SR is modified by a note that states the Surveillance is only required to be performed during the months of July, August, and September. During other months, the ambient temperature is below the surveillance limit.

NO CHANGES THIS PAGE.
FOR INFORMATION ONLY

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.9.3

This SR verifies dam integrity by inspection to detect degradation, erosion, or excessive seepage. Operating experience has shown that these components usually pass the Surveillance when performed at the 12 month Frequency. Therefore, the Frequency is acceptable from a reliability standpoint.

REFERENCES

1. UFSAR, Section 9.2.
2. Regulatory Guide 1.27.
3. 10 CFR 50.36, Technical Specifications, (c)(2)(ii).

ATTACHMENT 2

REPRINTED TS AND BASES PAGES FOR CATAWBA

3.7 PLANT SYSTEMS

3.7.9 Standby Nuclear Service Water Pond (SNSWP)

LCO 3.7.9 The SNSWP shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. SNSWP inoperable.	A.1 Be in MODE 3. AND A.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.9.1	Verify water level of SNSWP is ≥ 571 ft mean sea level.	24 hours
SR 3.7.9.2	------------------------------NOTE------------------------------ Only required to be performed during the months of July, August, and September. -- Verify average water temperature of SNSWP is ≤ 95°F at an elevation of 568 ft. in SNSWP.	24 hours
SR 3.7.9.3	Verify, by visual inspection, no abnormal degradation, erosion, or excessive seepage of the SNSWP dam.	12 months

B 3.7 PLANT SYSTEMS

B 3.7.9 Standby Nuclear Service Water Pond (SNSWP)

BASES

BACKGROUND

The SNSWP provides a heat sink for processing and operating heat from safety related components during a transient or accident, as well as during normal operation. This is done by utilizing the Nuclear Service Water System (NSWS) and the Component Cooling Water (CCW) System.

The SNSWP has been defined as the water source, including necessary retaining structure, but not including the cooling water system intake structures as discussed in the UFSAR, Section 9.2 (Ref. 1). The principal functions of the SNSWP are the dissipation of sensible heat during normal operation, and dissipation of residual and sensible heat after an accident or normal operation.

The basic performance requirements are that a 30 day supply of water be available, and that the design basis temperatures of safety related equipment not be exceeded.

Additional information on the design and operation of the SNSWP can be found in Reference 1.

APPLICABLE SAFETY ANALYSES

The SNSWP is the sink for heat removed from the reactor core following all accidents and anticipated operational occurrences in which the unit is cooled down and placed on residual heat removal (RHR) operation.

NSWS temperature influences containment pressure following a Loss of Coolant Accident and offsite dose following a Main Steam Line Break. The containment peak pressure analysis can accommodate NSWS temperatures up to 100°F. However, offsite dose requirements limit NSWS temperature to 95.5°F to prevent extending Reactor Coolant System cooldown time to a value greater than that currently assumed. Therefore, offsite dose following a Main Steam Line Break is more limiting with respect to NSWS temperature.

The peak containment pressure analysis assumes the NSWS flow to the Containment Spray and Component Cooling Water heat exchangers has a temperature of 100°F. To ensure that this condition is not exceeded, and to ensure that long term NSWS temperature does not exceed the

BASES

APPLICABLE SAFETY ANALYSES (continued)

100°F design basis of NSWS components a limit of 95°F is observed for the SNSWP. This temperature is important in that it, in part, determines the capacity for energy removal from containment. The peak containment pressure occurs when energy addition to containment (core decay heat) is balanced by energy removal from these heat exchangers. This balance is reached after the transition from injection to cold leg recirculation and after ice melt. Because of the effectiveness of the ice bed in condensing the steam which passes through it, containment pressure is insensitive to small variations in containment spray temperature prior to ice meltout.

Long term equipment qualification of safety related components required to mitigate the accident is based on a continuous, maximum NSWS supply temperature of 100°F.

To ensure that the NSWS initial temperature assumptions in the limiting analysis are met, Lake Wylie temperature is also monitored. During periods of time while Lake Wylie temperature is greater than 95.5°F, the emergency procedure for transfer of Emergency Core Cooling System (ECCS) flow paths to cold leg recirculation directs the operator to align at least one train of containment spray to be cooled by a loop of NSWS which is aligned to the SNSWP. Swapover to the SNSWP is required at 95.5°F rather than 95°F because Lake Wylie is not subject to subsequent heatup due to recirculation, as is the SNSWP. Therefore, the 100°F design basis maximum temperature is not approached.

The operating limits are based on conservative heat transfer analyses for the worst case accident. Reference 1 provides the details of the assumptions used in the analysis. The SNSWP is designed in accordance with Regulatory Guide 1.27 (Ref. 2), which requires a 30 day supply of cooling water in the SNSWP.

The SNSWP satisfies Criterion 3 of 10 CFR 50.36 (Ref. 3).

LCO

The SNSWP is required to be OPERABLE and is considered OPERABLE if it contains a sufficient volume of water at or below the maximum temperature that would allow the NSWS to operate for at least 30 days following the design basis accident without the loss of net positive suction head (NPSH), and without exceeding the maximum design temperature of the equipment served by the NSWS. To meet this condition, the SNSWP temperature should not exceed 95°F at 568 ft mean sea level and the level should not fall below 571 ft mean sea level during normal unit operation.

BASES

APPLICABILITY

In MODES 1, 2, 3, and 4, the SNSWP is required to support the OPERABILITY of the equipment serviced by the SNSWP and required to be OPERABLE in these MODES.

In MODE 5 or 6, the requirements of the SNSWP are determined by the systems it supports.

ACTIONS

A.1

If the SNSWP is inoperable the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours and in MODE 5 within 36 hours.

The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.9.1

This SR verifies that adequate long term (30 day) cooling can be maintained. The specified level also ensures that sufficient NPSH is available to operate the NSWS pumps. The 24 hour Frequency is based on operating experience related to trending of the parameter variations during the applicable MODES. This SR verifies that the SNSWP water level is ≥ 571 ft mean sea level.

SR 3.7.9.2

This SR verifies that the NSWS is available to cool the CCW System to at least its maximum design temperature with the maximum accident or normal design heat loads for 30 days following a Design Basis Accident. The 24 hour Frequency is based on operating experience related to trending of the parameter variations during the applicable MODES. This SR verifies that the average water temperature of the SNSWP is ≤ 95°F. The SR is modified by a note that states the Surveillance is only required to be performed during the months of July, August, and September. During other months, the ambient temperature is below the surveillance limit.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.9.3

This SR verifies dam integrity by inspection to detect degradation, erosion, or excessive seepage. Operating experience has shown that these components usually pass the Surveillance when performed at the 12 month Frequency. Therefore, the Frequency is acceptable from a reliability standpoint.

REFERENCES

1. UFSAR, Section 9.2.
2. Regulatory Guide 1.27.
3. 10 CFR 50.36, Technical Specifications, (c)(2)(ii).

ATTACHMENT 3

TECHNICAL, REGULATORY, AND ENVIRONMENTAL ANALYSIS

1.0 Description:

This submittal is a request to amend Operating Licenses NPF-35 and NPF-52 for Units 1 and 2, respectively. The purpose of this change is to revise the temperature limit for the SNSWP from the existing value of 91.5°F to the proposed value of 95°F.

As a result of hot weather conditions at the Catawba site, the temperature of the SNSWP is approaching the existing 91.5°F limit. Should the SNSWP temperature reach this limit, both Catawba units would be forced to shut down. Duke Energy Corporation requests approval of this emergency TS change by July 29, 2005 in order to avoid an unnecessary shutdown of Units 1 and 2.

2.0 Proposed Change:

Duke Energy Corporation proposes to revise the temperature limit for the SNSWP as delineated in Surveillance Requirement (SR) 3.7.9.2 from 91.5°F to 95°F. Corresponding changes are also made to the TS Bases to reflect this proposed change.

3.0 Background:

System description

The SNSWP serves as the safety related ultimate heat sink for processing and operating heat from safety related components during a transient or accident, as well as during normal operation. This is accomplished by utilizing the Nuclear Service Water System (NSWS) and the Component Cooling Water (CCW) System.

Lake Wylie serves as the normal (i.e., non-safety related) source of water for the NSWS. Should Lake Wylie be lost due to a seismic event in excess of the design of Wylie Dam, the SNSWP, formed by the Class 1 seismically designed SNSWP Dam, contains sufficient water to bring the station safely to a cold shutdown condition under all normal, transient, and accident conditions. The transfer from Lake Wylie to the SNSWP occurs automatically upon loss of Lake Wylie.

The basic performance requirement of the SNSWP is that a 30-day supply of water will be available and that the design basis temperatures of safety related equipment will not be exceeded. The SNSWP has been sized and analyzed to ensure that the 30-day cooling supply is available at a temperature compatible with design basis temperatures for NSWS supplied equipment. A point near the surface of the SNSWP, at elevation 568 feet, is monitored in order to ensure that the UFSAR assumption for the containment peak pressure analysis will be preserved. By selection of the elevation, it is assured by water density differences that the intake to the NSWS, located at the bottom, will be bounded by the SNSWP equilibrium temperature limit. To ensure that this initial condition is not exceeded, and to ensure that long term NSWS temperature does not exceed the 100°F design basis of NSWS components, the average water temperature must be less than or equal to 91.5°F at elevation 568 feet in the SNSWP. Long term equipment qualification of safety related components required to mitigate the accident is based on the continuous, maximum NSWS supply temperature of 100°F. Swapover from Lake Wylie to the SNSWP is required at a temperature of 92°F in the NSWS supply header rather than 91.5°F because Lake Wylie is not subject to subsequent heatup due to recirculation, as is the SNSWP. Hence, the 100°F design basis maximum temperature is not approached while operating from Lake Wylie.

TS 3.7.9 delineates the requirements for the SNSWP. The SNSWP is required operable in Modes 1, 2, 3, and 4. With the SNSWP inoperable, the affected unit(s) is(are) required to be in Mode 3 within 6 hours and in Mode 5 within 36 hours. SR 3.7.9.2 requires that during the months of July, August, and September, verification that average water temperature of the SNSWP is less than or equal to 91.5°F at an elevation of 568 feet in the SNSWP be performed. The SR frequency is 24 hours.

Reason for requesting change on an emergency basis

This change is being requested on an emergency basis to prevent the unnecessary shutdown of both units. The SNSWP conditions currently being experienced at Catawba could not have been reasonably anticipated based on plant history. Over the 20-year history of Catawba, there has never been a forced shutdown due to exceeding the SNSWP temperature limit. Catawba has experienced extended hot, humid weather with no appreciable precipitation. Approximately one more week of similar conditions where daytime ambient air temperature is in excess of the SNSWP limit is predicted. Hot weather conditions have resulted in a heatup rate of the SNSWP of approximately 0.4°F per day for the past two weeks. Given the anticipated weather conditions at Catawba over the next week, the temperature of the SNSWP could reach the existing SR 3.7.9.2 limit of 91.5°F by early August 2005. Additionally, the SNSWP has historically experienced temperature excursions of up to 3°F following late afternoon rain showers as a result of rain runoff absorbing heat from the SNSWP surroundings. Catawba has not made any changes to the topography surrounding the SNSWP which are believed to have contributed to this situation.

Actions taken in response to this situation

Catawba is performing periodic makeup to the SNSWP in an effort to replace the warm surface water of the SNSWP with cooler water from Lake Wylie. This action has slowed the rate of temperature increase of the SNSWP, although it will not likely preclude a temperature spike following any rain shower runoff. Duke Energy Corporation is also attempting to raise and maintain the level of Lake Wylie in order to preserve the cooler water near the bottom of the lake, which serves as makeup to the SNSWP. However, Lake Wylie water temperature is also increasing and therefore this mitigation strategy will become less effective at higher lake temperatures.

Condition that the proposed amendment is intended to resolve

The purpose of this proposed amendment is to revise the SR 3.7.9.2 temperature limit from 91.5°F to 95°F so as to avoid an unnecessary shutdown of Catawba Units 1 and 2.

4.0 Technical Evaluation

In support of this proposed amendment, the following considerations were made and appropriate analysis was performed:

1. Acceptable flow and fouling in the containment spray heat exchangers for one NSWS pump operation (calculation CNC-1223.13-00-0002)
2. Acceptable NSWS flow and fouling in the CCW System heat exchangers (calculation CNC-1223.24-00-0018)
3. Determination of Reactor Coolant System cooldown time following a Main Steam Line Break (calculation CNC-1201.30-00-0027)
4. SNSWP thermal analysis during one unit Loss of Coolant Accident and one unit shutdown (calculation CNC-1150.01-00-0001)
5. Containment response reanalysis (calculation CNC-1552.08-00-0278) and Large Break Loss of Coolant Accident containment response with asymmetric initial ice distribution (calculation CNC-1552.08-00-0314)

Each of these considerations is discussed in detail in the following paragraphs:

1. CNC-1223.13-00-0002

This calculation determines containment spray heat exchanger fouling factors, tube plugging limit, and minimum NSWS flow to satisfy the heat transfer assumptions for the containment spray heat exchangers in the containment pressure analysis.

Calculation CNC-1552.08-00-0314 analyzes containment response during a Loss of Coolant Accident. As part of this analysis, values of the product of the overall heat transfer coefficient (U) and surface area (A) for the containment spray heat exchangers are assumed. Calculation CNC-1223.13-00-0002 analyzes the performance of the containment spray heat exchangers at the flow and temperature conditions of calculation CNC-1552.08-00-0314. The amount of fouling and the number of plugged tubes that can be tolerated is determined to ensure that the actual value of the product of U times A does not decrease below that assumed in calculation CNC-1552.08-00-0314.

The containment spray heat exchanger was analyzed at a NSWS inlet temperature of 100°F, and it was verified that the current acceptance criteria and tube plugging limits maintain the UA assumptions for the containment spray heat exchangers in the containment pressure analysis. In fact, the UA values actually increased slightly (< 1%). This is due to the lower viscosity of the higher temperature water, which results in higher calculated film coefficients. Thus, the proposed 95°F SNSWP temperature limit will be bounded.

2. <u>CNC-1223.24-00-0018</u>

This calculation determines CCW System heat exchanger fouling factors, tube plugging limit, and minimum NSWS flow to satisfy the heat transfer assumptions for the CCW and residual heat removal heat exchangers in the containment pressure analysis.

Calculation CNC-1552.08-00-0314 analyzes containment response during a Loss of Coolant Accident. As part of this analysis, values of the product of the overall heat transfer coefficient (U) and surface area (A) for the CCW System heat exchangers are assumed. Calculation CNC-1223.24-00-0018 analyzes the performance of the CCW and residual heat removal heat exchangers together at the flow and temperature conditions of calculation CNC-1552.08-00-0314. The amount of fouling and the number of plugged tubes that can be tolerated is determined to ensure that the actual value of the product of U times A does not decrease below that assumed in calculation CNC-1552.08-00-0314.

The CCW and residual heat removal heat exchangers were analyzed at a NSWS inlet temperature of 100°F, and it was verified that the current acceptance criteria and tube plugging limits maintain the UA assumptions for the CCW and residual heat removal heat exchangers in the containment pressure analysis. In fact, the UA values actually increased slightly (< 1%). This is due to the lower viscosity of the higher temperature water, which results in higher calculated film coefficients. Thus, the proposed 95°F SNSWP temperature limit will be bounded.

3. CNC-1201.30-00-0027

This calculation determines the time required for steaming through the steam generator power operated relief valves after a Main Steam Line Break Accident. This calculation is an input to dose calculations.

Calculation CNC-1201.30-00-0027 models the cooldown of the Reactor Coolant System following a Main Steam Line Break Accident. Heat inputs to the Reactor Coolant System are decay heat, reactor coolant pump heat, and heat stored in the metallic components of the system. Heat removal is via the steam generator power operated relief valves. Below 350°F, residual heat removal is also included in the heat removal term. The amount of heat that can be removed via residual heat removal is a strong function of NSWS temperature, so any increase in NSWS temperature will increase the amount of time to cool down the unit.

NSWS temperature to the CCW System heat exchangers was raised to 95.5°F. The amount of heat removed via residual heat removal as a function of residual heat removal temperature was input to the cooldown model. Revision 0 of calculation CNC-1201.30-00-0027 determined that the cooldown time to 210°F was 31.2 hours. Revision 1 of this calculation (which supports this amendment request) shows that the revised cooldown time to 210°F is 18.8 hours. By comparison, the dose calculations assume a cooldown time of 37.5 hours. The increase in margin from Revision 0 to Revision 1 results from the fact that Revision 0 assumed 865 plugged tubes in the CCW System heat exchangers, while Revision 1 assumes 400 plugged tubes. Thus, the increase in surface area was more than enough to offset the increase in NSWS temperature.

This calculation assumes at least 18.8 hours of constant NSWS temperature. Per calculation CNC-1150.01-00-0001, the high flow model of the SNSWP assumes 46,000 gpm for the first 4 hours of an accident and 23,000 gpm for the remainder of the 30 days. Thus, in 18.8 hours, 96.6 acre-feet of water would be pumped from the SNSWP, through the plant, and back to the SNSWP. However, the volume versus elevation data contained in calculation CNC-1150.01-00-0001 states that there is 279.45 acre-feet of water contained in the SNSWP up to elevation 566 feet (temperature is measured at elevation 568 feet). Therefore, the CCW System heat exchangers would be supplied constant temperature water for the duration of the cooldown. This maintains the validity of the constant temperature assumption in calculation CNC-1201.30-00-0027.

4. <u>CNC-1150.01-00-0001</u>

This calculation determines the plant intake temperature from the SNSWP during a one unit Loss of Coolant Accident and a one unit shutdown. It verifies the UFSAR requirement that the intake temperature remains below 100°F during the 30-day one unit cooldown period following a one unit Loss of Coolant Accident.

The 100°F limit is imposed primarily to assure long term equipment qualification criteria are met. In addition, the diesel generator jacket water cooling heat exchanger and control room chiller analyses currently assume a 100°F NSWS inlet temperature. Recent revisions to calculations CNC-1223.13-00-0002 and CNC-1223.24-00-0018 have raised the assumed NSWS inlet temperature for the containment spray and CCW heat exchangers, respectively, from 92°F to 100°F.

In NUREG-0954, "Safety Evaluation Report related to the operation of Catawba Nuclear Station, Units 1 and 2", Supplements 1 and 2, Section 2.4.4.2, the NRC imposed a 2.4°F margin (penalty) on the maximum SNSWP return temperature during the 30-day one unit cooldown period following a one unit Loss of Coolant Accident. With this penalty, the maximum SNSWP temperature was limited to 97.6°F. Given this limit, the maximum initial SNSWP temperature that could be assumed and still remain below the 30-day limit was 91.5°F. Thus, this became the TS limit. By letter dated February 5, 1997 (Peter S. Tam to William R. McCollum, "Catawba Nuclear Station - Standby Nuclear Service Water Pond Analysis Model"), the NRC withdrew the 2.4°F penalty. While this was reflected in Revision 10 of calculation CNC-1150.01-00-0001, the SNSWP analysis was not rerun at that time to determine if the initial SNSWP temperature could be raised.

Due to the recent increase in SNSWP temperature in July 2005, rerunning the SNSWP analysis at a higher initial temperature was revisited to determine how much additional margin could be gained. Runs were made at 93°F through 100°F, and the maximum initial temperature which would maintain the 30-day pond temperature below 100°F was determined to be 95°F.

Therefore, in this calculation, the initial SNSWP temperature was raised from 91.5°F to 95°F. It should be noted that this reanalysis was run using the same initial pond elevation of 571 feet as was used in the previous analysis. However, since then, three additional feet of water have been added to the SNSWP, such that the normal SNSWP elevation is now 574 feet. No credit was taken for this additional level in this reanalysis, thus providing additional margin. The reanalysis was performed in accordance with the NRC Safety Evaluation for the Catawba SNSWP analysis model (letter from Peter S. Tam to William R. McCollum dated November 19, 1996, "Catawba Nuclear Station - Standby Nuclear Service Water Pond Analysis Model").

The impact of increasing the initial SNSWP temperature from 91.5°F to 95°F was to increase the maximum SNSWP intake temperature during the first 30 days following a one unit cooldown concurrent with a one unit Loss of Coolant Accident from 97.0°F to 98.2°F, thus remaining below the limit of 100°F.

The change in SNSWP temperature will provide additional margin for short term SNSWP temperature spikes and thus reduce the probability of plant shutdown transients.

5. CNC-1552.08-00-0278 and CNC-1552.08-00-0314

The current UFSAR peak containment pressure calculation is performed utilizing methodology developed by Duke Energy Corporation for analyzing the mass and energy release and containment response for the McGuire and Catawba Nuclear Stations. This methodology is described in topical report DPC-NE-3004-P. This methodology utilizes the RELAP5/MOD3.1DUKE and GOTHIC4.0/DUKE computer codes, and is available for use in reanalyzing the long term containment pressure response. NRC approval for this topical report was received by letter, R.E. Martin to M.S. Tuckman, dated September 6, 1995.

Topical report DPC-NE-3004-P describes the methodology developed by Duke Energy Corporation for simulating the mass and energy release from high energy line breaks and the resulting containment response for the McGuire and Catawba Nuclear Stations. The mass and energy release resulting from Loss of Coolant Accidents is simulated with the RELAP5/MOD3.1DUKE computer code for a spectrum of break locations. The mass and energy release resulting from Steam Line Breaks is simulated with the RETRAN-02 MOD5.1DUKE computer code for a spectrum of break sizes. The ice condenser containment response is simulated with the GOTHIC4.0/DUKE computer code. The methodology includes models for both the Unit 1 Babcock & Wilcox International feedring steam generators and the Unit 2 Westinghouse Model D5 steam generators. These methods are used to demonstrate that the containment peak pressure and temperature limits are not exceeded. This methodology is approved for use in predicting the containment pressure and temperature responses to design basis accidents for the McGuire and Catawba Nuclear Stations.

Mass and Energy Release Methodology:

The methodology described in topical report DPC-NE-3004-P for simulating the mass and energy release resulting from a design basis Loss of Coolant Accident utilizes the RELAP5/MOD3.1DUKE computer code for a spectrum of break locations. This code is derived from RELAP5/MOD3.1, which is an advanced thermal-hydraulic computer code developed by EG&G Idaho for the NRC. Duke Energy Corporation has modified the RELAP5/MOD3.1 code by including error corrections provided by EG&G Idaho to obtain RELAP5/MOD3.1DUKE.

The energy released into containment by a Large Break Loss of Coolant Accident is that energy that is initially contained in the primary and secondary coolant systems fluid, associated metal components of the system boundaries, and sensible heat stored in the core, plus the additional energy that is produced and released subsequent to the break as a result of continued fission, fission product decay, and metal-water reaction. The initial conditions for Large Break Loss of Coolant Accident analyses are chosen to maximize the stored energy in both the primary and secondary systems. Maximizing the stored energy will ensure that conservative mass and energy boundary conditions are provided to the containment response analyses. Guidance and criteria for selecting the initial values for the principal system parameters are provided in ANSI/ANS-56.4-1983.

The most limiting single failure assumed for minimum safeguards situations is the loss of one emergency diesel generator in conjunction with a loss of offsite power. This failure minimizes the capability to mitigate the Loss of Coolant Accident mass and energy release and the resulting containment response. Other conservative assumptions include Emergency Core Cooling System (ECCS) injected flowrates, available refueling water storage tank volume, steam generator pressure and level control, main and auxiliary feedwater flowrates and temperatures, and containment backpressure. The ECCS injection temperature during the containment sump recirculation phase is obtained through an iterative process using RELAP5/GOTHIC results.

Containment Response Methodology:

The methodology described in topical report DPC-NE-3004-P for simulating the containment response to high energy line breaks utilizes the GOTHIC4.0/DUKE computer code. The GOTHIC code, derived from the COBRA-NC thermal-hydraulic code, was developed by Numerical Applications, Inc. (NAI), under contract from EPRI, for performing thermal-hydraulic analysis of nuclear power plant containment and auxiliary buildings. Duke Energy Corporation has modified the GOTHIC Version 4.0 code by including minor code changes provided by NAI to obtain GOTHIC4.0/DUKE.

The Catawba GOTHIC model simulates the four different regions in an ice condenser containment building. These are lower containment, upper containment, ice condenser, and dead-ended compartments. The ice condenser and passive heat structures are modeled in detail. The initial conditions that result in a conservative peak containment pressure analysis produce a high mass of non-condensible gases, and minimize the warming of ice prior to melting. The boundary conditions for the GOTHIC analyses include break mass flow rate and energy input data, containment spray mass flow rate and energy input, containment sump cooling, and nitrogen addition to containment from the cold leg accumulators. The most limiting single failure results in one train of containment spray available to mitigate the pressurization transient.

Containment Response Analysis:

The mass and energy release and containment response methodology described above is utilized to reanalyze the long term peak containment pressure response for both the Babcock & Wilcox International feedring steam generators installed in Unit 1 and the Westinghouse Model D5 steam generators installed in Unit 2. The limiting peak containment pressure case, a Unit 1 double-ended guillotine cold leg reactor coolant pump discharge break, is analyzed using the current TS minimum ice bed ice weight of 2,132,000 lbm (SR 3.6.12.4), and an increased SNSWP temperature of 100°F.

The impact of the increased SNSWP temperature occurs in the GOTHIC analysis during the containment sump recirculation phase of the event. The containment spray heat exchangers are cooled directly by the NSWS. The residual heat removal heat exchangers are cooled by the CCW System (a closed cooling system), which in turn is cooled by the NSWS. Thus, the cold leg injection portion of the analysis is unaffected by this change.

The peak containment pressure obtained during the reanalysis increases from the current conservative UFSAR value of 13.16 psig to 14.04 psig (which is within the maximum allowable value of 14.68 psig specified by TS 5.5.2, "Containment Leakage Rate Testing Program"). Therefore, this reanalysis utilizing the NRC-approved methodology justifies an increase in the NSWS temperature to 100°F.

The maximum containment sump temperature is obtained from a large double-ended hot leg break. The sump temperature must remain low enough to ensure stable residual heat removal pump operation during the recirculation phase. The limiting point for this transient is reached immediately after the residual heat removal pumps swap to sump recirculation. The sump temperature is decreasing at this point in the transient, as ice melt and containment spray flow accumulates in the sump. The proposed change to the NSWS temperature will not impact the maximum calculated containment sump temperature for the hot leg break, since this is reached before the increased NSWS temperature can have any impact on the containment sump temperature response.

A review of the revised cold leg reactor coolant pump discharge break with the increased NSWS temperature verified that the cold leg reactor coolant pump discharge break case remains bounded by the hot leg break case, with regard to maximum sump temperature. The maximum sump temperature is therefore unaffected by an increase in the NSWS temperature. The UFSAR maximum credible sump temperature of 190°F is not exceeded.

The Catawba UFSAR will be revised to reflect the containment response reanalysis in the applicable annual UFSAR update following NRC approval of these requested amendments.

Short Term Blowdown Peak Pressure:

The blowdown peak containment pressure analysis is unaffected by an increase in the allowable SNSWP temperature. The current method used to calculate the blowdown peak pressure consists of the calculation of the air mass compression ratio, using the polytropic exponent for this compression process taken from the Waltz Mill results and compartment volumes taken from TMD input data. This method is described in Section 6.2.1.1.3 of the UFSAR. The time frame for this analysis is extremely brief, and does not extend to sump recirculation.

Therefore, it is concluded that the short term containment pressure response is not affected by an increase in the assumed SNSWP temperature and the analysis currently presented in the UFSAR remains valid.

Loss of Coolant Accident Peak Cladding Temperature:

The minimum containment pressure used in the Loss of Coolant Accident peak cladding temperature analysis is unaffected by an increase in the SNSWP temperature. This is primarily because the time frame of interest in this analysis is prior to transfer to containment sump recirculation. Therefore, it is concluded that the minimum containment pressure analysis and the Loss of Coolant Accident peak cladding temperature analysis currently presented in the UFSAR remain valid.

Peak Containment Temperature:

The peak containment temperature determined by the Main Steam Line Break analysis is unaffected by an increase in the SNSWP temperature. This is primarily because the time frame of interest in this analysis is prior to transfer to containment sump recirculation. Therefore, it is concluded that the maximum containment temperature analysis currently presented in the UFSAR remains valid.

Peak Reverse Differential Pressure:

The peak reverse differential pressure analysis currently presented in the UFSAR is unaffected by an increase in the allowable SNSWP temperature. The time frame for this analysis is extremely brief, and does not involve a transfer to containment sump recirculation. Therefore, it is concluded that the short term containment pressure response is not affected by an increase in the assumed SNSWP temperature and the analysis currently presented in the UFSAR remains valid.

Summary:

The NRC-approved mass and energy release and containment response methodology was utilized to reanalyze the long term Loss of Coolant Accident peak containment pressure response. The results of this reanalysis demonstrate that the applicable acceptance criteria are satisfied while maintaining the operational and safety margins.

5.0 Regulatory Evaluation:

No Significant Hazards Consideration Determination

The following discussion is a summary of the evaluation of the changes contained in this proposed amendment against the 10 CFR 50.92(c) requirements to demonstrate that all three standards are satisfied. A no significant hazards consideration is indicated if operation of the facility in accordance with the proposed amendment would not:

1. Involve a significant increase in the probability or consequences of an accident previously evaluated, or

2. Create the possibility of a new or different kind of accident from any accident previously evaluated, or

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

First Standard

Does operation of the facility in accordance with the proposed amendment involve a significant increase in the probability or consequences of an accident previously evaluated? No.

This license amendment request proposes a change to the SNSWP TS requirement for maximum temperature. The SNSWP is the safety related ultimate heat sink utilized by the NSWS. Neither the NSWS nor the SNSWP is capable of initiating an accident. Therefore, the probability of initiation of any accident cannot be affected. The technical evaluation provided in support of this amendment request demonstrated that with a maximum allowable SNSWP temperature of 95°F as specified in SR 3.7.9.2, the environmental qualification limit for applicable safety related equipment is not reached and the peak containment pressure remains below the TS limit. This amendment request does not involve any change to previously analyzed dose analysis results. The accident of interest from a dose perspective is the Main Steam Line Break Accident. The dose release path during this accident is via steaming of the Reactor Coolant System through the steam generator power operated relief valves. The results of this accident have been reviewed with the revised SNSWP temperature limit and it has been determined that the Reactor Coolant System cooldown is terminated early enough such that the dose analysis results are not adversely impacted. Therefore, there is no increase in any accident consequences.

Second Standard

Does operation of the facility in accordance with the proposed amendment create the possibility of a new or different kind of accident from any accident previously evaluated? No.

This proposed amendment does not involve addition, removal, or modification of any plant system, structure, or component. This change will not affect the operation of any plant system, structure, or component as directed in plant procedures. Operation of the facility in accordance with this amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.

Third Standard

Does operation of the facility in accordance with the proposed amendment involve a significant reduction in the margin of safety? No.

Margin of safety is related to confidence in the ability of the fission product barriers to perform their design functions following any design basis accident. These barriers include the fuel cladding, the Reactor Coolant System, and the containment. The proposed changes have no impact on fuel cladding performance. In addition, Reactor Coolant System performance (as determined by its impact on dose analysis results) continues to be acceptable as indicated above. Finally, containment performance (as determined by calculated containment peak pressure) remains acceptable. Therefore, the performance of these fission product barriers either during normal plant operations or following an accident will not be affected by the changes associated with this license amendment request. In addition, the operation of the NSWS and the SNSWP either during normal plant operations or following an accident will not be adversely impacted by implementation of the proposed amendment.

Based upon the preceding discussion, Duke Energy Corporation has concluded that the proposed amendment does not involve a significant hazards consideration.

6.0 Environmental Evaluation:

Pursuant to 10 CFR 51.22(b), an evaluation of this license amendment request has been performed to determine whether or not it meets the criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9) of the regulations.

Implementation of this amendment will have no adverse impact upon the Catawba units; neither will it contribute to any additional quantity or type of effluent being available for adverse environmental impact or personnel exposure.

It has been determined there is:

1. No significant hazards consideration,

2. No significant change in the types, or significant increase in the amounts, of any effluents that may be released offsite, and

3. No significant increase in individual or cumulative occupational radiation exposures involved.

Therefore, this amendment to the Catawba TS meets the criteria of 10 CFR 51.22(c)(9) for categorical exclusion from an environmental impact statement.

7.0 References:

1) Catawba Nuclear Station Technical Specifications, with Amendments through 225/220.

2) Catawba Nuclear Station UFSAR, Section 9.2.

8.0 Precedents:

A number of licensees have requested and received amendment of their operating licenses authorizing increases in the temperature limit of their ultimate heat sink. As a result of the different formats of the TS that are part of the operating licenses for these various licensees, there have been wide variations in the approaches proposed by the licensees and in the format of the increased ultimate heat sink temperature.

However, the following precedents involve increases in the ultimate heat sink temperature limit:

1. H.B. Robinson, Amendment 191, issued August 9, 2001
2. Hope Creek, Amendment 120, issued April 19, 1999
3. Davis-Besse, Amendment 242, issued September 12, 2000
4. Palisades, Amendment 202, issued June 4, 2001
5. Indian Point Unit 2, Amendment 149, issued March 27, 1990
6. Braidwood Station, Amendment 107, issued June 13, 2000