Duke Power
A Duke Energy Company

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May 9, 2002

U. S. Nuclear Regulatory Commission
Washington, DC 20555-0001
ATTENTION: Document Control Desk

SUBJECT: Duke Energy Corporation
Catawba Nuclear Station Unit (s) 1 and 2
Docket Numbers 50-413 and 50-414
Proposed Amendment to the Facility
Operating Licenses Concerning Steam Generator
Tube Rupture Licensing Basis

Pursuant to 10CFR50.90, Duke Energy Corporation is requesting an amendment to the Catawba Nuclear Station Facility Operating Licenses (FOL) to revise the licensing basis Steam Generator Tube Rupture sequences. Specifically, it is requested that a certain single failure scenario potentially leading to steam generator overfill be excluded from the design basis steam generator tube rupture analysis. The justification for this change includes risk-informed evaluations performed using the guidance of Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant Specific Changes to the Licensing Basis."

The change contained in this LAR is being proposed because a single failure has been identified that may be more limiting than the single failure assessed in the original design basis evaluation of the steam generator tube rupture event. The risk from the steam generator tube rupture scenario associated with this failure is assessed to be

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insignificant, and it is being requested that the scenario be deleted from consideration in the design basis steam generator tube rupture.

The contents of this amendment package are as follows:

Attachment 1 provides a Description of the Proposed Change and Technical Justification.

Pursuant to 10CFR50.92, Attachment 2 documents the determination that the amendment contains No Significant Hazards Considerations.

Pursuant to 10CFR51.22(c)(9), Attachment 3 provides the basis for the categorical exclusion from performing an Environmental Assessment/Impact Statement.

Implementation of this amendment to the Catawba Facility Operating License will impact the Catawba UFSAR. At a minimum, Section 15.6.3, Steam Generator Tube Failure, will be revised following approval of the amendment request. Necessary changes will be made in accordance with 10CFR50.71(e).

No commitments are made in this correspondence.

NRC approval of this LAR is requested by November 30, 2002, or as soon as practical. This LAR requests approval for the deletion of a steam generator tube rupture sequence with a certain single failure. This sequence does not in itself pose a significant risk to the public. Retention of this single failure within the licensing basis will pose an overly restrictive burden on the plant. Resolution of the scenario associated with this single failure would be very expensive and would not significantly reduce risk to the public. Implementation of this amendment will be completed following completion of plant modifications to resolve separate issues that could affect steam generator overfill prevention. It is anticipated that this will occur early in 2003.

In accordance with Duke administrative procedures and the Quality Assurance Program Topical Report, this proposed amendment has been reviewed and approved by the Catawba Plant Operations Review Committee and the Duke Corporate Nuclear Safety Review Board.

Pursuant to 10CFR50.91, a copy of this proposed amendment is being sent to the appropriate state officials.

Inquiries on this matter should be directed to M.H. Chernoff at (803) 831-3414.

Very truly yours,

G.R. Peterson

Attachments

xc w/attachments:

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AFFIDAVIT

G. R. Peterson, being duly sworn, states that he is Site Vice President of Duke Energy Corporation; that he is authorized on the part of said corporation to sign and file with the Nuclear Regulatory Commission this amendment to the Catawba Nuclear Station(s) Facility Operating Licenses Numbers NPF-35 and NPF-52 and Technical Specifications; and that all statements and matters set forth herein are true and correct to the best of his knowledge.

G.R. Peterson, Site Vice President

Subscribed and sworn to me: 5-9-2002
Date

Notary Public

My Commission Expires: 6-26-2002
Date

SEAL

# ATTACHMENT 1

# DESCRIPTION OF PROPOSED CHANGE

# AND

# TECHNICAL JUSTIFICATION

## 1) Description of Change

Duke Energy Corporation has identified a single failure that may be limiting with respect to the single failure assessed in the design basis evaluation of the steam generator tube rupture. The risk from a steam generator tube rupture sequence with this failure is assessed to be low. Therefore, Duke Energy Corporation is requesting that this failure sequence be eliminated from the design basis steam generator tube rupture analysis. The specific failure to be excluded is failure of 125 VDC Distribution Center EDE or EDF.

The discussion of this topic provides an evaluation in accordance with Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant Specific Changes to the License Basis." This change will be reflected in site calculation packages and/or the Updated Final Safety Analysis Report (UFSAR) following approval of this amendment. At a minimum, UFSAR Section 15.6.3, Steam Generator Tube Failure, will be revised.

## 2) Background

The steam generator tube rupture analysis was pursued generically by the Westinghouse Owners Group Steam Generator Tube Rupture Subgroup. On March 30, 1987, the NRC Staff issued a Safety Evaluation Report (SER) accepting the Subgroup's analysis methodology documented in WCAP-10698, "SGTR Analysis Methodology to Determine the Margin to Steam Generator Overfill", December 1984. One of the acceptance criteria for the analysis is the existence of margin to overfill of the ruptured steam generator.

The Staff's SER required additional plant specific input for each utility referencing WCAP-10698. One requirement was to list the "systems, components, and instrumentation which are credited for accident mitigation in the plant specific SGTR EOP(s)." For each function required to prevent overfill of the ruptured steam generator (S/G), it was acceptable to identify a non-safety system or piece of equipment if a safety related system train or piece of equipment was identified as a backup. This effectively extended the single failure requirement to allow credit for non-safety equipment as part of the array of the redundant equipment to mitigate the consequences of this accident. The Staff also required that each licensee determine the limiting single failure with respect to margin to overfill of the ruptured steam generator if the limiting failure was not the limiting failure of WCAP-10698 (failure of a power operated relief valve on an intact steam generator to open to establish a subcooled margin for the reactor coolant), then the effect of the limiting failure identified by the licensee on the margin to overfill was to be evaluated.

In a letter dated December 7, 1987, Duke Power Company submitted the plant specific information for Catawba Nuclear Station Units 1 and 2. The limiting single failure with respect to margin to overfill of the ruptured steam generator was a failure of a power operated relief valve on an intact steam generator to open on demand. A "Design Basis Equipment List for Catawba" was provided to list the equipment credited for preventing overfill of the ruptured steam generator. Some non-safety equipment was listed as follows:

1) Auxiliary Feedwater (AFW) flow control valves (AFW isolation valves as backup).

2) Main Feedwater (MFW) flow control valves (MFW isolation valves as backup).

3) Turbine stop valves, steam dump valves, reheater steam supply valves, auxiliary steam supply valve, steam line drains, steam traps, condenser air ejector valves (backup to the Main Steam Isolation Valves (MSIV's)).

The Staff found the equipment on this list to be acceptable (Ref. 14).

During a self-initiated review to verify compliance with the UFSAR and accuracy of the UFSAR, it was determined that Technical Specification 3/4.7.1.6 was not restrictive enough to ensure that the consequences of the steam generator tube rupture accident could be mitigated. Additionally, single failures not analyzed for effect on the consequences of the steam generator tube rupture accident were found. At the time, Technical Specification 3/4.7.1.6 required that at least three steam generator power operated relief valves be operable. In the analysis in existence at the time, it was assumed that two steam generator power operated relief valves on two intact steam generators were available for remote operation to establish a subcooled margin in the Reactor Coolant System and prevent the ruptured steam generator from filling. Given the limiting single failure known at the time for overfill margin, compliance with Technical Specification 3/4.7.1.6 would only ensure that the power operated relief valve for at most one intact steam generator would be available for establishing a subcooled margin in the Reactor Coolant System. A single failure consisting of a loss of control power to two steam generator power operated relief valves could have resulted in a sequence in which only the power operated relief valve for the ruptured steam generator is available for remote operation. This could have extended the time needed for plant cooldown and increase the likelihood of steam generator overfill. Prevention of steam generator overfill is one of the acceptance criteria for the steam generator tube rupture analysis.

In order to ensure that a power operated relief valve on at least one intact steam generator is available for remote operation during unit cooldown following a steam generator tube rupture considering the newly identified single failures, all four power operated relief valves were required to be operable. Administrative restrictions were put in place to require all four power operated relief valves be maintained operable and to restrict dose equivalent iodine concentration to a conservatively low value. The restriction for I-131 was intended to ensure that the latest dose analysis of record remained bounding.

In a letter dated March 7, 1997, and as supplemented by letters dated April 2, 10, 16, 22, and 28, 1997, Duke Energy Corporation requested changes to Technical Specification 3/4.7.1.6 to require operability of all four steam generator power operated relief valves and changes to the UFSAR to resolve this issue. The license amendment was issued on April 29, 1997. In addition to the requirement to have all four steam generator power operated relief valves operable, the license amendment also allowed credit for local manual operation of a steam generator power operated relief valve on an intact steam generator to prevent steam generator overfill. Additionally, the Staff imposed a license condition to affirm Duke Energy's self-imposed restriction on dose equivalent iodine in lieu of the Technical Specification limits. It was determined that the adequacy of the Technical Specification limits was an unreviewed issue pending a determination of their validity or revision thereto based on future thermal hydraulic assessment results.

On November 11, 1997, during an additional design review of the auxiliary feedwater system initiated by Duke Energy Corporation, the existence of a more limiting single failure was postulated. A failure of 125 VDC Distribution Center EDE or EDF results in the inability to isolate auxiliary feedwater flow to two steam generators and the inability to control two steam generator power operated relief valves remotely. If a steam generator tube rupture were to occur on one of the affected steam generators, there would be a potential to overfill the ruptured steam generator because auxiliary feedwater flow to it could not be remotely isolated from the Control Room.

Conservative administrative controls on primary and secondary system equilibrium and transient specific activities were established. The administrative controls were calculated using conservative assumptions and limited reactor coolant dose equivalent iodine to ensure the consequences of the steam generator tube rupture would remain within the appropriate guideline values.

This discovery was reported pursuant to 10 CFR 50.72 and 50.73 in Licensee Event Report 413/1997-009-02, "Unanalyzed Postulated Single Failure Affecting Steam Generator Tube Rupture Analysis."

In response to this discovery, a failure analysis on the equipment needed for prevention of steam generator overfill was done to ensure equipment failure effects are clearly identified and properly considered in the analysis. The failure analysis revealed several single failures that had not been evaluated for the steam generator tube rupture accident.

The effects of several of these single failures have since been nullified by a combination of plant modifications, administrative controls, procedure revisions, and training. Two classes of single failures have not yet been dispositioned in their entirety. It is planned to implement a plant modification to disposition one of these failure classes. The remaining failure sequence is the subject of this License Amendment Request.

## 3) Technical Justification

Steam generator overfill can occur following a steam generator tube rupture when there is a failure to control the flow of liquid into the steam generator. Control of both the flow of auxiliary feedwater and the break flow into the steam generator must be effective in order to prevent overfill. Failures that inhibit the control of these functions may lead to steam generator overfill.

A failure of 125 VDC Distribution Center EDE or EDF may be limiting with respect to the single failure assessed in the design basis evaluation of the steam generator tube rupture. However, the risk from a steam generator tube rupture sequence associated with this failure is assessed to be insignificant. The general approach used to evaluate the risk significance of this failure sequence is summarized as follows:

- Quantify the single failure probability,
- Estimate the frequency of the initiating event,
- Estimate the Core Damage Frequency (CDF) and Large Early Release Frequency (LERF), and
- Evaluate the CDF / LERF significance relative to the criteria of RG-1.174.

### 3.1) Detailed Description of Single Failure

The detailed description of this failure and its effects are predicated on certain assumptions. It is assumed that a steam generator tube rupture occurs and there is a loss of offsite power at the time of reactor trip. It is assumed that the 125 VDC Vital I&C Distribution Center EDE or EDF fails within two hours prior to the initiation of the accident or within 12 seconds of the loss of offsite power. The two hour assumption is based on the Technical Specifications allowed outage time of two hours for an inoperable vital DC electrical power distribution subsystem. The 12 second post event assumption is based on the time required for the Diesel Generator Load Sequencer to complete its load sequencing function. Once the Diesel Generator Load Sequencer has sequenced on Load Group 1, a failure of 125 VDC Distribution Center EDE or EDF will not interrupt control or motive power to the valves needed for remote control of auxiliary feedwater flow.

For simplicity in this discussion, nomenclature applicable to Train A is presented first with the corresponding Train B nomenclature provided in parenthesis.

The 125 VDC Distribution Centers EDE and EDF are powered by redundant power supplies via an auctioneering diode assembly. The only potential credible failure mode is failure of the bus itself, such as an internal short.

The Auxiliary Feedwater System includes two motor-driven pumps each aligned to separate and independent Class 1E power trains (A and B), and the Turbine-driven Auxiliary Feedwater Pump. The Auxiliary Feedwater Motor-driven Pump A is aligned to Steam Generators A and B while Auxiliary Feedwater Motor-driven Pump B is aligned to Steam Generators C and D. The Turbine-driven Auxiliary Feedwater Pump is aligned to all four steam generators. This constitutes a total of eight auxiliary feedwater lines to the steam generators.

Each line is equipped with an air operated flow control valve and a motor-operated isolation valve in series. Motive power for the control valves is supplied by the Instrument Air System, which is non-safety related and powered by non-Class 1E power supplies. Each motor-operated isolation valve is aligned to a Class 1E power train. Motor-driven Auxiliary Feedwater Pump A and the isolation valves from Motor-driven Auxiliary Feedwater Pump A to Steam Generators A and B are powered from the A train Class 1E power supply system. Motor-driven Auxiliary Feedwater Pump B and the isolation valves from Motor-driven Auxiliary Feedwater Pump B to Steam Generators C and D are powered from the B train Class 1E power supply system. The isolation valves for the Turbine-driven Auxiliary Feedwater Pump to Steam Generators A and B are powered from the B train Class 1E power supply system, and the isolation valves for the Turbine-driven Auxiliary Feedwater Pump to Steam Generators C and D are powered from the A train Class 1E power supply system.

Failure of 125 VDC Distribution Center EDE (EDF) causes a loss of power to A (B) train Diesel Generator Load Sequencer, failing both 4160 and 600 volt power to one train of Emergency Core Cooling System equipment. Power would be lost to the A (B) train Motor-driven Auxiliary Feedwater Pump and its isolation valves to two steam generators. Power to the motor-operated isolation valves in the discharge line of the Turbine-driven Auxiliary Feedwater Pump to the opposite two steam generators would be lost.

If there is a loss of offsite power, instrument air is lost and the non-safety auxiliary feedwater flow control valves fail open. Air accumulators have been installed on the valves to allow a limited period of operation following a Loss of Instrument Air. Failure of 125 VDC Distribution Center EDE or EDF causes the solenoids of the train related auxiliary feedwater flow control valves to de-energize. With their solenoids de-energizing, air in the instrument air lines to the affected auxiliary feedwater flow control valves vent to atmosphere and the valves fail open.

If there is also a failure of 125 VDC Distribution Center EDE (EDF) prior to the Diesel Generator Load Sequencer completing its load sequencing function, one train of Emergency Core Cooling System would be lost. One Motor-driven Auxiliary Feedwater Pump would be supplying auxiliary feedwater to two steam generators, and the Turbine-driven Auxiliary Feedwater Pump would be supplying feedwater to all four generators, but remote operation of two of the isolation valves would be lost due to the loss of 125 VDC Distribution Center EDE (EDF). The ability of the Control Room operators to remotely throttle or isolate auxiliary feedwater flow to two steam generators would be compromised. If the ability to remotely throttle or isolate auxiliary feedwater flow was compromised for a steam generator with a steam generator tube rupture, steam generator overfill could result.

Failure of 125 VDC Distribution Center EDE or EDF also causes an inadvertent swap of the auxiliary feedwater controls from the Control Room to the auxiliary feedwater pump turbine control panel. It is planned to modify the transfer scheme of the auxiliary feedwater pump turbine control panel to an "energize to transfer" scheme, which will nullify these failure effects.

Loss of EDE would additionally result in loss of power to the Trip and Throttle Valve SA145 for the Turbine-driven Auxiliary Feedwater Pump.

The effects of a failure of 125 VDC Distribution Center EDE or EDF have been evaluated for impact on other design basis accident scenarios. For these accident scenarios, the operators have sufficient time to take recovery actions prior to steam generator overfill. In the evaluation, it was determined that procedural guidance exists to direct the appropriate recovery action.

3.2) Affected Licensing Basis

Prevention of overfill of the ruptured steam generator following a design basis steam generator tube rupture is part of the licensing basis of Catawba Nuclear Station as summarized in the Updated Final Safety Analysis Report (Ref. 12). The assumptions concerning loss of offsite power, initial conditions, protection systems and engineered safeguards activation, and operator action are the same as those established by the Steam Generator Tube Rupture Subgroup of the Westinghouse Owner's Group and reported in WCAP-10698 (Ref. 15). In that effort, the design basis steam generator tube rupture was defined, with limiting initial and boundary conditions identified. For the Westinghouse reference plant, the limiting single failure was defined as the failure of a power operated relief valve on one of the intact steam generators to open on demand. Finally, for the occurrence of the design basis steam generator tube rupture and limiting single failure at the reference plant, margin to steam generator overfill was demonstrated.

In the SER for this study, the staff required licensees to perform analyses to verify that the conclusions of the generic study (margin to steam generator overfill) applied to each plant (Ref. 16). One of the requirements was that each licensee referencing WCAP-10698 identify the limiting single failure for its plant(s). If it was different from the limiting single failure of WCAP-10698, then the effect of the limiting single failure on margin to overfill of the ruptured steam generator was to be determined. An additional requirement was to list the systems, components, and instrumentation which are credited for accident mitigation in the plant specific steam generator tube rupture Emergency Operating Procedures. For each function required to prevent overfill of the ruptured steam generator, it was acceptable to identify a non-safety system or piece of equipment if a safety-related system train or piece of equipment was identified as a backup. Duke Energy Corporation responded that the results of the Westinghouse generic study, including the single failure analysis, bounded Catawba Nuclear Station (Ref. 13). Duke Energy Corporation also provided a Design Basis Equipment List for Catawba. In this list a number of non-safety pieces of equipment were identified. In additional failure analysis, it has been determined that the original evaluations were not entirely appropriate to Catawba.

This License Amendment Request requests approval for the deletion of the steam generator tube rupture sequences with this single failure from the licensing basis of Catawba Nuclear Station. In the evaluation below, it will be shown that a steam generator tube rupture sequence with this single failure does not in itself pose a significant risk to the public or the Control Room operator. Retention of this single failure within the licensing basis will pose an overly restrictive burden on the plant. Resolution of this sequence will be very expensive and also may have an adverse effect on the defense-in-depth and safety margin elsewhere without significantly reducing the risk of the plant to the public.

### 3.3) Traditional Engineering Evaluation

An evaluation has been performed to show that sufficient defense-in-depth and safety margins are retained with the change proposed in this License Amendment Request. Effectively, it is requested that the design, configuration, and operation of the plant be left unchanged with respect to these steam generator tube rupture sequences. No changes to the plant systems, structures, and components are associated with the removal of these steam generator tube rupture sequences from the licensing basis. No changes to the plant Technical Specifications are part of this risk-informed resolution. In particular, no changes to Technical Specification 3.7.4 (requiring all four steam generator power operated relief valves of each nuclear unit to be operable) are proposed with this License Amendment Request.

#### 3.3.1) Defense-in-depth

A failure of 125 VDC Distribution Center EDE/EDF may degrade the ability of the operators to prevent the ruptured steam generator from overfilling following a design basis steam generator tube rupture. The limiting consequences of failure of 125 VDC Distribution Center

EDE/EDF during normal unit operations are similar to those of a unit trip. Failure of 125 VDC Distribution Center EDE or EDF, in addition to degrading the ability of the operators to stop the flow of auxiliary feedwater to the ruptured steam generator, may result in loss of electric power to one Class 1E train of safety-related systems and equipment. This is termed the "minimum safeguards" scenario. The minimum safeguards configuration does not have a significant effect on the ability to prevent the ruptured steam generator from filling. With respect to other design basis events, the plant is designed for the minimum safeguards scenario. The response of the plant to any of these design basis events with the minimum safeguards failure has been shown to be adequate. The frequencies of steam generator tube ruptures or other initiating events are not increased as a result of this license amendment.

The consequences of steam generator overfill following a steam generator tube rupture will almost invariably be limited to consequential failure of a main steam safety valve or a steam generator power operated relief valve for the ruptured steam generator. The consequences of this in terms of the fission product barriers are discussed below. Also, the radiological consequences of such an event under both design basis conditions and nominal conditions are discussed below.

Failure of 125 VDC Distribution Center EDE/EDF affects the ability of the operators to control or stop the flow of auxiliary feedwater to a steam generator. It also causes the loss of a Motor-driven Auxiliary Feedwater Pump. The primary purpose of the Auxiliary Feedwater System is to deliver feedwater to the steam generators to remove residual heat from the Reactor Coolant System should normal feedwater not be available. The Auxiliary Feedwater System is capable of providing feedwater to the steam generators to adequately remove decay heat from the Reactor Coolant System with the loss of any one pump. Therefore, the ability of the Auxiliary Feedwater System to maintain a secondary heat sink is not degraded by the proposed license amendment.

The Emergency Core Cooling System is designed to provide water to the Reactor Coolant System following a design basis event for the purpose of makeup, cooling of the reactor core, and preservation of shutdown margin. The design basis steam generator tube rupture is one of the design basis events for the Emergency Core Cooling System. Failure of 125 VDC Distribution Center EDE/EDF causes loss of one of the redundant Class 1E trains of Emergency Core Cooling System equipment, precipitating the minimum safeguards scenario. One Class 1E train of the Emergency Core Cooling System is sufficient to provide water to the Reactor Coolant System for makeup, cooling of the core, and shutdown margin following any design basis event, including the design basis steam generator tube rupture. From the above evaluation, it is concluded that the proposed license amendment does not degrade the ability of the Emergency Core Cooling System and Auxiliary Feedwater System to maintain core integrity and prevent fuel damage following the design basis steam generator tube rupture.

Engineered safeguards provided for the protection of the containment include the Containment Spray System and Containment Air Return Fans. These systems are not required to actuate following a design basis steam generator tube rupture. Therefore, for this design basis event, they are not important to safety. Failure of 125 VDC Distribution Center EDE/EDF also precipitates the loss of one Class 1E train of each these systems. This is the minimum safeguards scenario.

The design basis steam generator tube rupture includes a pathway for bypass of the reactor containment. This pathway includes flow of reactor coolant from the Reactor Coolant System through the break to the secondary side of the ruptured steam generator, where it is available for release to the environment through the relief valves of the ruptured steam generator (e.g., the steam generator power operated relief valve). Should the operators be unable to prevent the ruptured steam generator from filling following this event, the potential for containment bypass may be increased. However, the frequency of overfill events due to a steam generator tube rupture with failure of 125 VDC Distribution Center EDE/EDF has been found to be low, as shown below (Section 3.4). In addition, the most likely consequence of a steam generator tube rupture with steam generator overfill is the consequential failure of a steam generator relief valve (steam generator power operated relief valve or main steam safety valve). As noted below, another potential failure mode, steam line failure, is significantly less likely (Ref. 7, cf. Ref. 16). Additional evaluations have indicated that in the unlikely event of its occurrence, waterhammer would be small so as to not degrade the steam lines. It is concluded that there is no significant increase in the risk of containment bypass.

As noted above, the change proposed in this license amendment does not degrade the ability to maintain a secondary heat sink and provide water to the Reactor Coolant System for makeup, cooling of the core, and shutdown margin following a design basis steam generator tube rupture. Neither fuel damage nor clad damage is expected to occur for steam generator tube rupture sequences as a result of a failure of 125 VDC Distribution Center EDE/EDF. The limiting level of radioactivity in the Reactor Coolant System available for release in the steam generator tube rupture sequence is the activity allowed by the Technical Specifications (Ref. 1, Technical Specification 3.4.16) and augmented by either the pre-accident iodine spike or the accident-initiated iodine spike. The most likely consequence of a design basis steam generator tube rupture with overfill of the ruptured steam generator is a consequential failure of a main steam safety valve or steam generator power operated relief valve.

Should the ruptured steam generator overfill following a design basis steam generator tube rupture with one of the failures listed above, radioactivity could be released to the environment in increased amounts and over a longer time span than predicted in the safety analysis. Again, the frequency of occurrence of these steam generator tube rupture sequences is low, as shown below. In addition, should such an event occur, the radiological consequences are projected to be below the guidelines of 10 CFR 100 and General Design Criteria 19. Under nominal conditions, (e.g., nominal atmospheric dispersion factors, nominal levels of radioactivity in the Reactor Coolant System, etc.), radiological consequences of a steam generator tube rupture with failure of 125 VDC Distribution Center EDE/EDF are expected to be small compared to even the guideline values of the Standard Review Plan, Section 15.6.3. There is no significant adverse effect on the mitigation of consequences following a steam generator tube rupture by the proposed license amendment.

From this evaluation, it is concluded that a reasonable balance is preserved among prevention of core damage, prevention of containment failure, and consequence mitigation.

Programmatic activities include activities such as administrative controls associated with limits on initial and boundary conditions assumed in the analysis of design basis events. They also include operator actions taken pursuant to abnormal or emergency procedures following a design basis event. Local remote operator action was not credited in the steam generator tube rupture sequence associated with a failure of 125 VDC Distribution Center EDE/EDF.

It is concluded that over reliance on programmatic activities to compensate for deficiencies in plant design is avoided.

The failure of 125 VDC Distribution Center EDE or EDF does not degrade the ability to prevent core damage consistent with the single failure criterion. This failure may degrade the ability of the Control Room operators to prevent the ruptured steam generator from overfilling following a design basis steam generator tube rupture. The result may be consequential failure of the steam generator power operated relief valve or main steam safety valve for the ruptured steam generator - a degradation in the containment boundary for the design basis steam generator tube rupture. However, the frequencies of a steam generator tube rupture with these failures have been shown to be low, as reported below. It follows that no "risk outliers" are associated with this License Amendment Request. System redundancy, independence, and diversity are preserved. The failure of 125 VDC Distribution Center EDE or EDF does not include any common cause failures of equipment in independent and redundant Class 1E trains.

As noted above, no changes to any structure, system or component are associated with the design basis steam generator tube rupture sequences proposed for exclusion from the licensing basis. No changes in the operation of any structure, system or component are associated with the changes proposed in this License Amendment Request. Defenses against human error are preserved.

The equipment associated with the single failure listed above is evaluated for conformance to the General Design Criteria (GDC) of Appendix A to 10 CFR 50. From the evaluation above, it follows that the ability of the Emergency Core Cooling System to "provide abundant emergency core cooling … to transfer any heat from the reactor core following any loss of coolant…" is not degraded. The Emergency Core Cooling System also remains capable of "poison addition." Compliance with GDC 27 and GDC 35 is not degraded with the proposed change in this License Amendment Request. The ability of the Auxiliary Feedwater System "to transfer fission product heat and other residual heat from the reactor core at a rate such that specified acceptable fuel design limits … are not exceeded" is not degraded.

Failure of 125 VDC Distribution Center EDE or EDF does not degrade the ability of the Auxiliary Feedwater System to "transfer heat from systems, components, and structures important to safety to an ultimate heat sink." Therefore, conformance of the Auxiliary Feedwater System to General Design Criteria 34 and General Design Criteria 44 is not degraded. The ability of the Auxiliary Feedwater System to be controlled outside the Control Room as described in the UFSAR is not degraded with any of the above single failures. Therefore, the system remains in conformance with the germane requirements of General Design Criteria 19.

The ability of the Solid State Protection System (includes the Engineered Safety Features Actuation System - ESFAS) to activate the Emergency Core Cooling System, Auxiliary Feedwater System, and other engineered safeguards on the appropriate signals given a single failure is not degraded. Therefore, compliance of the ESFAS with its applicable General Design Criteria (e.g., GDC 20 - GDC 24, GDC 34, GDC 35, GDC 38, GDC 44) is not degraded.

Failure of 125 VDC Distribution Center EDE or EDF in concurrence with a design basis steam generator tube rupture or other design basis event may result in the "minimum safeguards" scenario. The ability of the remaining Class 1E train of equipment to function to protect the reactor has been demonstrated. For this reason, the Electric Power System at Catawba remains in conformance with GDC 17 given the failure of 125 VDC Distribution Center EDE/EDF.

Again, no hardware change is associated with this LAR. Therefore, conformance to applicable General Design

Criteria including those concerning inspections, testability, and separation of control systems from protection systems, is not degraded. For the reasons given above, no deviation from the General Design Criteria of Appendix A to 10 CFR Part 50 is associated with the change proposed in this LAR.

It is concluded that sufficient defense-in-depth is retained with the exclusion of the steam generator tube rupture sequences associated with the failure of 125 VDC Distribution Center EDE/EDF from the licensing basis of Catawba Nuclear Station.

3.3.2) Safety Margin

As noted above, no change to any structure, system or component is associated with the proposed removal of the steam generator tube rupture single failure sequence associated with the failure of 125 VDC Distribution Center EDE/EDF from the licensing basis.

The Solid State Protection System and the control interfaces with the Emergency Core Cooling System and Auxiliary Feedwater System (including the diesel generator load sequencers) have been designed in conformance with IEEE Std 279-1971 (Ref. 9). The Solid State Protection System activates the Class 1E components of the Emergency Core Cooling System on a safety injection signal and the auxiliary feedwater pumps on any of the appropriate automatic start signals. The Solid State Protection System has been designed to activate at least one Class 1E train of equipment even if it is affected by a random single failure. The failure of 125 VDC Distribution Center EDE/EDF affects the ability to throttle or stop some of the engineered safeguards equipment, not to start them. The failure would not prevent the Solid State Protection System from fulfilling its intended safety function. Conformance of the Solid State Protection System and other ESFAS equipment to IEEE Std 279-1971 is not degraded. The Class 1E electric power systems would not be degraded so as to cause "loss of power to … devices sufficient to jeopardize the safety of the station." Failure of 125 VDC Distribution Center EDE/EDF would not "prevent satisfactory performance of the minimum Class 1E loads required for safe shutdown and maintenance of post shutdown or post-accident station security." With any of these failures following a design basis steam generator tube rupture (or any other design basis event), conformance to IEEE Std 308-1971 (Ref. 5) is not degraded. The affected mechanical equipment (i.e., auxiliary feedwater and emergency core cooling system pumps, valves, etc.) remains in conformance with the applicable clauses of ASME Section III, Class 2 and Class 3. It is concluded that Codes and Standards approved by the NRC are met.

The standards by which the consequences of the design basis steam generator tube rupture at Catawba Nuclear Station are evaluated are as follows:

1) Departure from Nucleate Boiling Ratio (DNBR) is greater than the limit value.

2) There is margin to steam generator tube rupture overfill.

3) Radiological consequences are within the appropriate guideline values (Ref. 2, Sections 6.4 and 15.6.3).

It is not until the Control Room operators attempt to stop the flow of auxiliary feedwater to the ruptured steam generator that the effects of a failure of 125 VDC Distribution Center EDE or EDF relative to control of auxiliary feedwater flow would be manifested. Minimum DNBR would occur within seconds after reactor trip. Therefore, the criterion concerning DNBR is met. For all cases to be retained within the licensing basis with approval of this LAR, there is margin to steam generator overfill. In addition, radiological consequences of the design basis steam generator tube rupture retained in the licensing basis are within the appropriate guideline values. The risk evaluation in this License Amendment Request demonstrates that the frequency of steam generator overfill associated with the steam generator tube rupture sequences to be excluded is low (approximately 3.7 E-11 per reactor year). Additionally, the frequency of a large early release is shown to be very low (approximately 3.7 E-15 per reactor year). It is concluded that sufficient margin exists to account for analytical and data uncertainty for these steam generator tube rupture sequences.

It is concluded that sufficient safety margin with respect to the consequences of the design basis steam generator tube rupture is retained with the removal of the selected steam generator tube rupture sequences from the licensing basis as proposed in this LAR.

### 3.4) Evaluation of Risk Impact

The process of evaluating the risk significance of these failures includes the following steps:

- Quantify the single failure probability,
- Estimate the frequency of the initiating event,
- Estimate the Core Damage Frequency (CDF) and Large Early Release Frequency (LERF), and
- Evaluate the CDF / LERF significance relative to the criteria of RG-1.174.

The estimates for the relevant parameters are developed as follows.

#### 3.4.1) Failure of 125 VDC Vital I&C Power Distribution Center EDE or EDF

The hardware failure that is the subject of this License Amendment Request is failure of a dc distribution center that may result in the potential for leading to steam generator overfill. The bus failure rate has been estimated by performing a Bayesian update of a generic value from industry data with plant specific experience collected as part of the maintenance rule periodic assessments. The generic value has been taken from a database developed by an independent contractor; this same database formed the basis for Revision 2 of the Catawba Probabilistic Risk Assessment. The plant specific experience used in the update is from the time period December 1995 through March 1999. A log-normal distribution is assumed.

Failure of 125 VDC Distribution Center EDE (EDF) results in a loss of control power to 4160 volt Switchgear ETA (ETB) breakers, such as the feeder breaker from diesel generator A (B). If a loss of offsite power (LOOP) occurs, then the normal power path to ETA (ETB) is also unavailable. Consequently, one train of Class 1E equipment is unavailable for performing its intended safety functions.

With one train of safety power unavailable, the 600 volt motor-operated isolation valves (MOVs) in the lines for the Turbine-driven Auxiliary Feedwater Pump to two S/Gs could not be closed from the control room. In addition, loss of EDE results in loss of motive power to Turbine-driven Auxiliary Feedwater Pump Trip and Throttle (T&T) Valve SA145. Thus for a LOOP scenario, a consequence could be unchecked flow from the Turbine-driven Auxiliary Feedwater Pump to two S/Gs.

The timing of this failure is critical if it is to have the impact described above. Motor control centers (via the essential load centers) are sequenced onto the essential switchgear in the first load group at 1 second. Therefore, for the Auxiliary Feedwater isolation valves to be left unavailable the bus failure must occur within the first few seconds of the LOOP or the failure must be pre-existing.

A generic bus failure rate of 6.1E-07/hr with an error factor 5.2 is assumed as a prior distribution. The plant specific information is 0 failures in approximately 8.07E+05 bus-hours of operation. The resulting failure rate for "DC Bus Fails" is 3.9E-07/hr. Failure of 125 VDC Distribution Center EDE (EDF) during normal operation would be readily apparent through the undervoltage alarms that would be actuated. The failure probability is estimated assuming a pre-mission exposure time of two hours, based on the Technical Specification Allowable Outage Time. The post initiating event exposure time is only a few seconds and is assumed to make a negligible contribution. The probability that the bus is unavailable following a steam generator tube rupture is estimated to be 7.8E-07.

### 3.4.2) Initiating Event Frequency

The frequency of the steam generator tube rupture initiating event is estimated by updating a generic steam generator tube rupture frequency with Catawba specific experience. Both the generic frequency parameters and the Catawba critical hours have been taken from NUREG/CR-5750 (Ref. 4). The frequency estimate for this analysis is derived from a prior distribution based on the generic parameters, mean and 95$^{th}$ percentile values of 7.0E-03/RY and 1.4E-02/RY respectively, with a Bayesian update using the Catawba experience of 0 steam generator tube rupture events in 14.4 reactor-years (RYs) of operation. It is recognized that the Catawba experience is also included in the generic data calculation. Because the Catawba experience represents only a small fraction of the industry experience, this double counting of the Catawba experience is assumed to represent a negligible change from the condition where the Catawba experience is removed from the generic estimate. The estimated steam generator tube rupture initiating event frequency for this analysis is 6.8E-03/RY.

### 3.4.3) Overfill Frequency Sequence Analysis

Coincident with an SGTR and a LOOP, failure of 125 VDC distribution center EDE leads to an inability to control flow from the Turbine-driven Auxiliary Feedwater Pump to the C and D steam generators. If the rupture is on one of these generators then overfill is assumed to result. No credit for operator recovery is assumed.

The coincidence of a LOOP with the SGTR is required in order for the 125 VDC Distribution Center EDE/EDF failure to result in a failure of the isolation valves. NUREG/CR-6538 (Ref. 6) provides an analysis of the probability of a LOOP conditional on a reactor trip and ECCS actuation. The resulting probability of 0.014 has been adopted here for estimating the frequency of sequences consisting of an SGTR with LOOP and a single failure.

The overfill sequence is quantified as follows for 125 VDC Distribution Center EDE. The 125 VDC Distribution Center EDF sequence quantification would proceed similarly.

| Single Failure | SGTR Frequency (per RY) | Conditional Probability of a LOOP | 125 VDC Bus Failure Probability | Probability that SGTR is on SGs C or D | Frequency of Overfill (per RY) |
|---|---|---|---|---|---|
| Failure of EDE | 6.80E-3 | 1.4E-2 | 7.8E-07 | 0.5 | 3.7E-11 |

3.4.4) Significance of Steam Generator Overfill

Steam generator overfill can lead to higher than expected offsite consequences if the release of reactor coolant activity is greater than assumed in the design basis analysis. Steam generator overfill could contribute to an increased release by creating a condition, water in the steam lines, that would increase the probability of a loss of the secondary system integrity. The most likely cause is expected to be a stuck open relief valve.

Secondary Integrity

When relief valves designed for steam pass a large quantity of liquid, the failure to close probability has typically been assumed to increase above the normally low random failure rate. A value of 0.1 is assumed in this analysis. This same value is used in NUREG 0844.

Reactor Coolant Activity

Reactor coolant activity during normal operation is restricted by the Technical Specification limits. These limits are set to assure that offsite doses are acceptably small in the case of the design basis accident. The quantity of radioactive material available in the Reactor Coolant System during normal operation is very small compared to the available material that results from a core damage accident. The offsite consequences for a steam generator overfill accident releasing only the normal reactor coolant activity would be much less severe than if core damage is involved.

Offsite Consequences and LERF

With the Reactor Coolant System dose equivalent iodine at historical levels and best estimate meteorology, exposure to the Control Room operator and offsite population as a result of steam generator overfill should be insignificant. With the Reactor Coolant System dose equivalent iodine at the Technical Specification limit, offsite exposures would increase but remain small compared to severe accident consequences. In order to generate a release of fission products comparable to a large early release, core damage must occur as a result of the overfill.

Because steam generator tube rupture results in a loss of reactor coolant outside of the containment, long term cooling via recirculation from the containment sump is not available. Instead, long term cooling is established by cooling down and depressurizing to residual heat removal conditions. Core damage can result if break flow can not be terminated and the refueling water storage tank, the injection source, is depleted. The principal concern with overfill is the loss of secondary integrity. Loss of secondary integrity impacts the ability to mitigate a steam generator tube rupture event by requiring a depressurization to atmospheric pressure to terminate break flow.

Using information contained in NUREG 0844 (Reference 7), a conditional probability of core damage can be estimated. Core damage occurs due to failure to depressurize the Reactor Coolant System to atmospheric conditions prior to refueling water storage tank depletion. The estimate adopted for the conditional probability of core damage for a steam generator tube rupture and a stuck open secondary relief valve is 1E-03. It is assumed for the purpose of this analysis that core damage as a consequence of a steam generator tube rupture and stuck open steam line relief valve constitutes a large early release. This assumption may be conservative.

| Single Failure | Frequency of Overfill (per RY) | Probability of Relief Valve Failure to Reseat | Conditional Probability of Core Damage | Frequency of Uncontrolled Release as a Result of Overfill (per RY) |
|---|---|---|---|---|
| Failure of EDE | 3.7E-11 | 1.0E-1 | 1.0E-3 | 3.7E-15 |

The frequency of a sequence in which a steam generator tube rupture results in steam generator overfill which then proceeds to core damage and containment bypass is calculated to be 3.7E-15. This frequency is a very small fraction of the ΔLERF criterion of 1.0E-07 stated in Regulatory Guide 1.174; and is also a very small fraction of the estimated base case LERF for Catawba Nuclear Station which is 4.8E-07/RY.

Main steam line failure is also a possible (though much less likely) consequence of steam generator overfill. Using the estimates from NUREG 0844 (Reference 7), the LERFs due to steam line failure are a factor of 100 less likely than those presented for the stuck open relief valve.

Discussion of Uncertainty

The sources of uncertainty in the probabilistic analysis include uncertainties that result from modeling assumptions as well as the inherent uncertainties in the data applied to the analysis. No formal uncertainty analysis is included here; rather it is observed that an increase in the sequence frequencies of many orders of magnitude is needed to bring the estimated frequencies into the range of the acceptance criterion for ΔLERF. Such a large uncertainty in the result is very unlikely.

Scope, Level of Detail, and Quality of the PRA

The Catawba PRA model has not been applied to this analysis. The data and sequence analyses included in support of this LAR have adopted a number of PRA techniques in support of this evaluation. The scope of the evaluation is consistent with the objective of addressing the frequency and consequences of steam generator overfill scenarios for the single failures of interest. The level of detail in the analysis is sufficient to support the risk-informed conclusions. Quality of the inputs to the evaluation is maintained by adopting values that are reported in reputable sources that are in most cases publicly available.

3.4.5) Summary of Risk Impact

The total contribution, from both buses, that these sequences are estimated to make to the LERF for Catawba is less than 1E-14 / RY. While uncertainty exists in this estimate, as there is in any probabilistic estimate, there is considerable margin to the criteria set forth in RG 1.174. The probability of failure of a 125 VDC distribution Center coincident with an SGTR/LOOP is very low. These sequences are not expected to contribute meaningfully to the risk estimates for Catawba, and their exclusion from the licensing basis is considered appropriate.

The guidance contained in Regulatory Guide 1.174 calls for the estimation of ΔLERF for comparison to the acceptance criterion. The proposed license amendment does not request any change to the plant. This request asks that the plant be left "as is" with respect to the capability to prevent steam generator overfill following a steam generator tube rupture. In this context, the LERF estimate is best considered as the ΔLERF (reduction) that might be achieved if the plant was modified in order to essentially eliminate these sequences. The actual reduction is expected to be less than the calculated amount since no modification can be perfectly reliable.

## 3.5) Monitoring Program

A risk based evaluation has been performed of a failure of 125 VDC Distribution Center EDE (EDF), a single failure which may degrade the ability of the Control Room operators to prevent the ruptured steam generator from filling following a steam generator tube rupture. System and component functions germane to prevention of steam generator overfill associated with this single failure is to provide uninterruptible power at 125 VDC to controls required to prevent the ruptured steam generator from filling following a design basis steam generator tube rupture. This function is monitored as part of the program put into place at Catawba Nuclear Station for compliance with the Maintenance Rule, 10 CFR 50.63 (Ref. 3).

4) **REFERENCES**


1) Catawba Nuclear Station Technical Specifications, Through Amendments 183/175.

2) USNRC, Standard Review Plan for the Review of Safety Analysis Reports of Nuclear Power Plants, NUREG-0800 (Rev 2).

3) Volume 10 of the Code of Federal Regulations (CFR), Part 50, Section 50.63.

4) NUREG/CR-5750, Rates of Initiating Events at U.S. Nuclear Power Plants: 1987-1995.

5) IEEE Standard Criteria for Class 1E Power Systems for Nuclear Power Generating Stations, IEEE Std 308-1971.

6) NUREG/CR-6538, Evaluation of LOCA With Delayed Loop and Loop With Delayed LOCA Accident Scenarios.

7) NUREG-0844, NRC Integrated Program for the Resolution of Unresolved Safety Issues A-3, A-4, and A-5 Regarding Steam Generator Tube Integrity.

8) Regulatory Guide 1.174, An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant Specific Changes to the License basis.

9) IEEE Standard: Criteria for Protection Systems for Nuclear Power Generating Stations, IEEE Std 279-1971.

10) LER 413/97-002-00, "Unanalyzed Postulated Single Failure Affecting the Steam Generator Tube Rupture Analysis".

11) LER 413/97-009-02, "Unanalyzed Postulated Single Failure Affecting the Steam Generator Tube Rupture Analysis".

12) Catawba Nuclear Station Final Safety Analysis Report, 1998 Update.

13) H.B. Tucker, Duke Energy Corporation, to U.S. Nuclear Regulatory Commission, "Catawba Nuclear Station Docket Nos. 50-413 and 50-414 Steam Generator Tube Rupture Analysis," December 7, 1987.

14) R.E. Martin (USNRC) to M.S. Tuckman, Duke Energy Corporation, "Safety Evaluation Report for the Catawba Nuclear Station Units 1 and 2, Steam Generator Tube Rupture Analysis (TAC Nos. 66753 and 66754." May 14, 1991.

15) R.N. Lewis, "SGTR Analysis Methodology to Determine the Margin to Steam Generator Overfill," WCAP-10698, December, 1984.

16) C.E. Rossi (USNRC) to A.E. Ladieu (SGTR Subgroup, Westinghouse Owners Group), "Acceptance of Methodology for Referencing Licensing Topical Report WCAP-10698 'SGTR Analysis Methodology to Determine the Margin to Steam Generator Overfill,' December 1984."

17) W.R. McCollum, Duke Energy Corporation, to U.S. Nuclear Regulatory Commission, "Catawba Nuclear Station Units 1 and 2 Docket Nos. 50-413 and 50-414 Request for Facility Operating License Amendment Steam Generator Tube Rupture Evaluation," March 7, 1997.

18) P.S. Tam, USNRC, to W.R. McCollum, Duke Energy Corporation, "Issuance of Amendments - Catawba Nuclear Station, Units 1 and 2 (TAC Nos. M98107 and M98108)," April 29, 1997.

ATTACHMENT 2

DETERMINATION OF NO SIGNIFICANT HAZARDS

DETERMINATION OF NO SIGNIFICANT HAZARDS

*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 proposed amendment requests that steam generator tube rupture sequences involving a failure of 125 VDC Distribution Center EDE (EDF) be excluded from consideration in the analysis of the design basis steam generator tube rupture event. These sequences involve a single failure that potentially degrades the ability to terminate auxiliary feedwater flow into a ruptured steam generator following a steam generator tube rupture. The inability to terminate auxiliary feedwater flow in a timely manner following a steam generator tube rupture could result in steam generator overfill.

The sequences to be excluded do not involve equipment that can be considered an accident initiator. Implementation of this amendment does not involve any physical changes to the facility. It does not affect basic operation of the facility. The probability of occurrence of a steam generator tube rupture or any other accident previously evaluated will not change following implementation of this amendment.

Elimination of certain sequences from the design basis steam generator tube rupture analysis does not adversely affect the ability to cool the reactor core and prevent core damage following a steam generator tube rupture. The Departure from Nucleate Boiling ratio is not adversely impacted.

The ability to maintain a secondary heat sink and provide water to the Reactor Coolant System for makeup, cooling of the core, and shutdown margin following a design basis steam generator tube rupture is not affected by the changes proposed in this license amendment. Neither fuel damage nor clad damage is expected to occur for the steam generator tube rupture sequences to be eliminated.

Should the ruptured steam generator overfill following a design basis steam generator tube rupture in one of the sequences to be excluded, radioactivity could be released to the environment in increased amounts and over a longer time span than predicted in the safety analysis. The frequency of occurrence of these steam generator tube rupture sequences is low. Should such an event occur, the radiological consequences are expected to be below the guidelines of 10 CFR 100 and General Design Criteria 19. Under nominal conditions, (e.g., nominal atmospheric dispersion factors, nominal levels of radioactivity in the Reactor Coolant System, etc.), radiological consequences of a steam generator tube rupture would be small compared to even the guideline values of the Standard Review Plan, Section 15.6.3. There is no significant adverse effect on the mitigation of consequences following a steam generator tube rupture.

In summary, operation of the facility in accordance with the proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

*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. The proposed amendment involves elimination of certain sequences from the design basis steam generator tube rupture analysis. No physical changes to the facility are associated with the proposed amendment.

The sequences to be eliminated involve single failures that could adversely affect the ability to terminate auxiliary feedwater flow to a ruptured steam generator. The failures associated with these sequences are not accident sequence precursors and do not have an adverse impact on any accident initiator.

No new failure modes are created due to implementation of the change proposed in this License Amendment Request. Therefore, operation of the facility in accordance with the changes proposed in this License Amendment Request does not create the possibility of a new or different kind of accident from any accident previously evaluated.

*Does operation of the facility in accordance with the proposed amendment involve a significant reduction in a margin of safety?* No. One of the standards by which the consequences of the design basis steam generator tube rupture are evaluated is that the Departure from Nucleate Boiling Ratio (DNBR) is greater than the limit value. Should one of the steam generator tube rupture sequences to be excluded occur, the effects relative to steam generator overfill would not be manifested until the Control Room operators attempt to stop the flow of auxiliary feedwater to the ruptured steam generator which is well into the event. The minimum DNBR would occur within seconds after reactor trip. Therefore, the criterion concerning DNBR is met.

The risk evaluation demonstrates that the frequency of steam generator overfill associated with the steam generator tube rupture sequences to be excluded is low (approximately 3.7 E-11 per reactor year per Class 1E Train). Additionally, the frequency of a large early release is shown to be very low (approximately 3.7 E-15 per reactor year per Class 1E Train).

It is concluded that removal of certain steam generator tube rupture sequences from the plant licensing basis as proposed does not constitute a significant reduction in a margin of safety.

Based on this evaluation, it is concluded that operation of the facility in accordance with the proposed amendment constitutes no significant hazard to the public.

ATTACHMENT 3

ENVIRONMENTAL ASSESSMENT/IMPACT STATEMENT

## ENVIRONMENTAL ASSESSMENT/IMPACT STATEMENT

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 significant 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 FOL meets the criteria of 10 CFR 51.22(c)(9) for categorical exclusion from an environmental impact statement.