Duke Power
A Duke Energy Company

Gary R. Peterson
Vice President

Duke Power
Catawba Nuclear Station
4800 Concord Road
York, SC 29745
(803) 831-4251 OFFICE
(803) 831-3426 FAX

September 19, 2000

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
Revised Proposed Amendment to the Facility Operating Licenses Concerning Steam Generator Tube Rupture Licensing Basis
(TAC Nos. MA8499 and MA8500)

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 certain single failure scenarios 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."

This correspondence also amends Duke Energy Corporation's letter of March 1, 2000, concerning Proposed Amendment to the Facility Operating Licenses and Technical Specification 1.1, Definitions - Dose Equivalent I-131. This package replaces the March 1, 2000, package in its entirety.

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The changes contained in this LAR are being proposed because a small number of single failures have 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 scenarios associated with these newly identified failures is assessed to be inconsequential, and it is being requested that these scenarios 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 Changes 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. 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).

Commitments are made in Section 3.5 of Attachment 1.

NRC approval of this LAR is requested by December 15, 2000, or as soon as practical. This LAR requests approval for the deletion of steam generator tube rupture sequences with certain single failures. These sequences do not in themselves pose a significant risk to the public. Retention of these single failures within the design and license bases will pose an overly restrictive burden on the

plant. Resolution of these scenarios would be very expensive and would not significantly reduce risk to the public. In addition, modifications designed to address these low likelihood events may have an adverse effect on the defense-in-depth and safety margin elsewhere. Implementation of this amendment will be completed within the standard 30 day implementation period.

In accordance with Duke 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 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:

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

C. P. Patel (Addressee Only)
NRC Senior Project Manager (CNS)
U. S. Nuclear Regulatory Commission
Mail Stop O-8 H12
Washington, DC 20555-0001

D. J. Roberts
Senior Resident Inspector (CNS)
U. S. Nuclear Regulatory Commission
Catawba Nuclear Site

V. R. Autry, Director
Division of Radioactive Waste Management
Department of Health and Environmental Control
2600 Bull Street
Columbia, SC 29207

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: 9-19-2000
Date

Notary Public

My Commission Expires: 6-26-2002
Date

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ATTACHMENT 1

DESCRIPTION OF PROPOSED CHANGES

AND

TECHNICAL JUSTIFICATION

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1) Description of Change

Duke Energy Corporation has identified a small number of single failures 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 steam generator tube rupture sequences with these failures is assessed to be low. Therefore, Duke Energy Corporation is requesting that these certain failure sequences be eliminated from the design basis steam generator tube rupture analysis. 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 the Updated Final Safety Analysis Report discussion of the steam generator tube rupture accident analysis.

Duke Energy Corporation is requesting that certain failure sequences be eliminated from the design basis steam generator tube rupture analysis. The specific failures to be excluded are:

1. Failure of 125 V dc Vital I&C Power Distribution Center EDE or EDF
2. Inadvertent swap of Auxiliary Feedwater controls from the Control Room to the Auxiliary Feedwater Pump Turbine Control Panel
3. Inability to Close High Pressure Injection Flow Valves NI9A or NI10B from the Control Room
4. Inability to Reset the Safety Injection Signal for a Train of the Solid State Protection System
5. Inability to Reset a Train's Diesel Generator Load Sequencer
6. Inability to Secure a Safety Injection Pump

These changes 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 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 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 S/G. 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 were listed as follows:

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

2) 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 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 TS 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, TS 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 RCS and prevent the ruptured steam generator from filling. Given the limiting single failure known at the time for overfill margin, compliance with TS 3.7.1.6 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 RCS. 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 increasing 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 Vital I&C 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 very 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.

As described in Licensee Event Report 413/1997-009-02, a plant modification was developed to accommodate some of these single failures. The risk from the remainder of these single failures was analyzed and was determined to be low. As a result of this evaluation, Duke Energy Corporation is requesting that these single failure sequences be removed from the design basis steam generator tube rupture analysis as a risk informed licensing action.

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 small number of additional single failures have been identified in a detailed failure analysis. These failures 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 steam generator tube rupture sequences associated with these failures is assessed to be inconsequential. Therefore, it is proposed that these single failures be eliminated from the steam generator tube rupture overfill design basis for Catawba. The single failures already accommodated by the plant design provide the necessary protection to the health and safety of the public. The general approach used to evaluate the risk significance of recently identified failures is summarized as follows:

- Quantify the single failure probabilities,

- Estimate the frequency of the initiating event,
- Screen out low frequency sequences,
- Identify possible recovery actions for unscreened single failures,
- Establish operator action times and quantify the non-recovery probabilities, and
- Evaluate the Core Damage Frequency (CDF) / Large Early Release Frequency (LERF) significance of remaining sequences relative to the criteria of RG-1.174.

3.1) Single Failures to be Excluded

Certain single failures contribute to the potential for steam generator overfill following a steam generator tube rupture. They do so by inhibiting control over either the flow of auxiliary feedwater into the ruptured steam generator or the flow of fluid from the Emergency Core Cooling System into the RCS. The specific failure modes of interest and the impacts on the operator response to a steam generator tube rupture are summarized below. There are 12 (6 per train) failures to be considered. These failures have been identified by means of a failure analysis.

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

3.1.1) Failure of 125 V dc Vital I&C Power Distribution Center EDE (EDF)

Failure of 125 V dc Vital I&C Power 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 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 Class 1E power unavailable, the 600 volt motor-operated valves in the lines from the Auxiliary Feedwater System turbine driven pump to two steam generators can not be closed from the Control Room. In addition, loss of EDE results in loss of motive power to

the turbine driven auxiliary feedwater pump trip and throttle valve SA145. Thus for a loss of offsite power sequence, a consequence could be unchecked flow from the turbine driven auxiliary feedwater pump to two steam generators. Motor operated valves in the line from a motor-driven auxiliary feedwater pump also are affected. However, they are on the same Class 1E electric power train as the affected motor-driven auxiliary feedwater pump and therefore are not a matter of concern because there would be no auxiliary feedwater flow.

Failure of 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 as described below.

3.1.2) Inadvertent Swap of Auxiliary Feedwater Controls from the Control Room to the Auxiliary Feedwater Pump Turbine Control Panel

The auxiliary feedwater flow control valves (pneumatic) and isolation valves (motor-operated) are interfaced with the auxiliary feedwater pump turbine control panel to allow for remote operation to cool and shut down the units should the Control Room become unusable. The panel is grouped into two trains. Each train is equipped with a transfer switch and associated transfer circuit.

The transfer circuits are normally closed, with the associated relays in an energized state for Control Room mode control. Should the transfer switch contacts spuriously open, or a transfer relay spuriously become de-energized, control would be transferred to the auxiliary feedwater pump turbine control panel. Recovery requires local manual operation of the affected equipment at the auxiliary feedwater pump turbine control panel.

3.1.3) Inability to Close High Pressure Injection Flow Valve NI9A or NI10B from the Control Room

Inability to terminate Safety Injection when the termination criteria of the emergency operating procedure are satisfied can result in overfilling the ruptured steam generator. In the steam generator tube rupture emergency procedure, the operators are instructed to isolate high

pressure injection flow from the centrifugal charging pumps to the RCS cold legs by closing valves NI9A and NI10B. One centrifugal charging pump is to remain running in order to provide normal charging and maintain reactor coolant pump seal injection. The second centrifugal charging pump is to be turned off. Failure of either valve to close represents a single failure that prevents terminating centrifugal charging pump injection with the potential of leading to steam generator overfill. Recovery requires successful local manual closure of the affected valve.

3.1.4) Inability to Reset the Safety Injection Signal for a Train of the Solid State Protection System

The inability to reset the Safety Injection Signal for one train of the Solid State Protection System results in the inability to reset the diesel generator load sequencer and affects the ability to turn off the associated safety injection pump and centrifugal charging pump as directed by procedure. In addition, this failure mode results in the inability to maintain NI9A (Solid State Protection System train A) or NI10B (Solid State Protection System train B) closed. Recovery requires local manual operation of the affected valve. Recovery is completed by closing the isolation valves in the line from the affected safety injection pump to the RCS cold leg injection headers and by tripping the redundant centrifugal charging pump.

3.1.5) Inability to Reset a Train's Diesel Generator Load Sequencer

The controls to start the centrifugal charging pumps and safety injection pumps on a safety injection signal are routed through the diesel generator load sequencers. Thus, the operators must reset the safety injection signal and the diesel generator load sequencers before they can secure the centrifugal charging pumps and safety injection pumps to terminate safety injection. The operators may recover from this failure either by opening the feeder breaker to the affected diesel generator load sequencer (a local action), or by closing the safety injection pump discharge valves and tripping the redundant centrifugal charging pump (from within the Control Room).

3.1.6) Inability to Secure a Safety Injection Pump

Once safety injection termination criteria are satisfied, the operators are instructed to secure the safety injection pumps. If the pumps fail to trip and are not isolated, continued break flow to the ruptured steam generator could lead to overfill. The Control Room operators can recover from this failure by closing the isolation valves in the lines from the affected safety injection pump to the RCS cold leg injection headers.

3.2) Affected License Basis

Prevention of overfill of the ruptured steam generator following a design basis steam generator tube rupture is part of the license basis of Catawba Nuclear Station as summarized in the Updated Final Safety Analysis Report (Ref. 12 - 14). 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 list of "Design Basis List for Catawba." In this list a number of non-safety pieces of equipment were identified as cited above.

In January, 1997, a set of single failures limiting with respect to those evaluated in WCAP-10698 were reported in Licensee Event Report 413/1997-002-00 (Ref. 10). The limiting single failure was failure of the power supply to the controls of the power operated relief valves of two intact steam generators. The potential effects of these failures were mitigated by requiring that all four steam generator power operated relief valves be operable and by taking credit for local operation of one of the two failed closed steam generator power operated relief valves. It was shown that following a design basis steam generator tube rupture with the limiting single failure of those identified in Licensee Event Report 413/1997-002-00(Ref. 17), there would be margin to steam generator overfill.

A number of additional single failures have been identified that may degrade the ability of the Control Room operators to prevent steam generator overfill following a design basis steam generator tube rupture as reported in Licensee Event Report 1997-009-02(Ref. 11). In the analyses of margin to overfill of the ruptured steam generator, credit is taken for the equipment associated with the single failure to operate adequately to complete their functions as noted above (consistent with credit taken for some non-safety related pieces of equipment). In addition, failures to terminate safety injection, in particular, the failure to close NI9A and NI10B and the failure to reset the safety injection signal were evaluated (Ref 13, 15). In additional failure analysis, it has been determined that the original evaluations either were less than adequate or not entirely appropriate to Catawba.

The effects of some of these failures were mitigated by a modification consisting of the addition of air accumulator tanks to provide backup to the normal air supply of the auxiliary feedwater flow control valves for a limited

period of time. The effects of some failures have been mitigated by administrative controls on the position of the isolation valves of the steam generator power operated relief valves.

Steam generator tube rupture sequences with the remainder of these additional failures are the subject of this License Amendment Request (LAR). This LAR requests approval for the deletion of the steam generator tube rupture sequences with these single failures from the license bases of Catawba Nuclear Station. From the evaluation below, it will be shown that steam generator tube rupture sequences with these single failures do not in themselves pose a significant risk to the public or the Control Room operator. Retention of these single failures within the license bases will pose an overly restrictive burden on the plant. Resolution of these sequences 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 LAR, that is removal of certain steam generator tube rupture sequences from the license bases of Catawba Nuclear Station. These sequences include the single failures identified above. 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 license bases. No changes to the plant TS are part of this risk-informed resolution. In particular, no changes to TS 3.7.4 (requiring all four steam generator power operated relief valves of each nuclear unit to be operable) are proposed with this LAR.

3.3.1) Defense-in-depth

A number of single failures have been identified which may degrade the ability of the operators to prevent the ruptured steam generator from overfilling following a design basis steam generator tube rupture. These single failures have been identified and presented above (Section 3.1). Of all the failures listed above, only the failure of distribution center EDE (EDF) may be a "transient initiator." The limiting consequences of failure of EDE / EDF during normal unit operations are similar to those of a unit trip. None of the other single failures would precipitate either an accident or an event. Indeed, some of the failures listed above (e.g., failures of safety injection termination) would not be manifested during normal plant operations. Failure of 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 the so-called "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 quite adequate. The frequencies of steam generator tube ruptures or other initiating events are not increased as a result of this license amendment. The remainder of the evaluation of defense-in-depth is focused on the occurrence of these single failures with the design basis steam generator tube rupture.

The design basis steam generator tube rupture sequences with the more limiting failures to be excluded from the plant license basis have been evaluated. (The more limiting of these single failures include failure of EDE or EDF, inadvertent transfer of one train of auxiliary feedwater controls to local mode, failure to close NI9A and NI10B, and failure to reset the safety injection signal for both trains of the Solid State Protection System.) Should one of these single failures follow a design basis steam generator tube rupture, the operators might not recover in time to prevent the ruptured steam generator from filling with water. Thus, the more limiting of these sequences may lead to overfill of the ruptured steam generator. As noted

below, 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.

Also considered in the evaluation were design basis steam generator tube rupture sequences with any one of the failures to be retained in the plant license basis. For any one of these design basis steam generator tube rupture sequences, the operators will be able to prevent the ruptured steam generator from filling with water. To ensure this for some of these single failures, additional equipment was installed by plant modifications. To address some of these single failures, administrative controls were placed into effect. The nature of the additional equipment and administrative controls are discussed below. To ensure margin to overfill for some single failures affecting the ability to open power operated relief valves for two of the intact steam generators, credit has been taken for local operation of one of these valves. The Staff already has approved this credit (Ref. 18).

The single failures presented above affect the Auxiliary Feedwater System and the Emergency Core Cooling System. However, the failures that affect the Auxiliary Feedwater System affect only the ability of the operators to control or stop the flow of auxiliary feedwater to a steam generator. The primary purpose of the Auxiliary Feedwater System is to deliver feedwater to the steam generators to remove residual heat from the RCS should normal feedwater not be available. With respect to this function, the only one of the failures listed above that has an adverse effect on the ability of the Auxiliary Feedwater System to perform this function is the EDE / EDF failure. Its effect on the Auxiliary Feedwater System is the loss of a motor-driven auxiliary feedwater pump. The Auxiliary Feedwater System is capable of providing feedwater to the steam generators to adequately remove decay heat from the RCS 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 RCS 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. The only failure of those listed above with an adverse effect on the ability of the Emergency Core Cooling System to perform this function is the failure of EDE / EDF. Its effect on the Emergency Core Cooling System is the loss of one of the redundant Class 1E trains of Emergency Core Cooling System equipment, precipitating the so-called "minimum safeguards" scenario. One Class 1E train of the Emergency Core Cooling System is sufficient to provide water to the RCS 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.

None of the single failures have any adverse effect on the primary containment shell. Engineered safeguards provided for the protection of the containment include the Containment Spray System and Containment Air Return Fans. These systems will not actuate following a design basis steam generator tube rupture. Therefore, for this design basis event, they are not important to safety. Of the failures listed above, only the EDE / EDF failure has an adverse effect on these containment safeguards. This failure precipitates the loss of one Class 1E train of each these systems - part of 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 RCS 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 one of the above single

failures 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). It is concluded that there is no significant increase in the risk of containment bypass associated with steam generator tube rupture sequences with the single failures described above.

As noted above, the changes proposed in the license amendment do not degrade the ability to maintain a secondary heat sink and provide water to the RCS 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 any of the failures listed above. The limiting level of radioactivity in the RCS available for release in these steam generator tube rupture sequences is the activity allowed by the Technical Specifications (Ref. 1, TS 3.4.16) and augmented by either the pre-accident iodine spike or the accident-initiated iodine spike. As noted above, 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 RCS, etc.), radiological consequences of a steam generator tube rupture with one of the failures above 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. Operator action was credited in only two of the steam generator tube rupture sequences in the analysis of plant risk associated with the failures listed above. These sequences were steam generator tube rupture with (1) inability to secure a safety injection pump and (2) failure of valve NI9A or valve NI10B to close on command from the Control Room.

With approval of this LAR, a number of design basis steam generator tube rupture single failure sequences will be retained in the plant license bases. They include some of the single failures reported in Licensee Event Report 413/1997-009-02 and identified in the failure analysis reported therein. Modifications and administrative controls have been implemented to mitigate the consequences of some of these single failures. Air accumulator tanks have been installed on the lines supplying instrument air to the auxiliary feedwater flow control valves. These tanks will ensure that the operators can maintain the auxiliary feedwater flow control valves to the ruptured steam generator closed for a minimum of 60 minutes, allowing time for an operator to close the downstream motor-operated valve manually following a design basis steam generator tube rupture. Per Facility Operating License Amendment 159/151 (Ref. 18), TS 3.7.4 (Then TS 3.7.1.6) was amended to require that all four steam generator power operated relief valves for each unit be operable. The license amendment included local operation of one of the steam generator power operated relief valves with its handwheel following a failure of a power supply to the controls of the power operated relief valves of two intact steam generators. Finally, the number of steam generator power operated relief valve isolation valves which may be closed has been restricted to one per Class 1E train during normal unit operations. This nullifies one of the consequences of a failure of a common power supply

(e.g., a diesel generator) to the isolation valves of two steam generator power operated relief valves following a design basis steam generator tube rupture. (This was one of the single failures identified in the failure analysis as described in Licensee Event Report 413/1997-009-02.) The operability of the steam generator power operated relief valves is not affected with these administrative controls in place. On the other hand, these administrative controls ensure that the flow path for the power operated relief valves for at least two intact steam generators will be available following a design basis steam generator tube rupture with the failure of any Class 1E power supply at either 600 volts or 4160 volts. It is concluded that over reliance on programmatic activities to compensate for deficiencies in plant design is avoided.

The single failures listed in this LAR do not degrade the ability to prevent core damage consistent with the single failure criterion, as discussed above. As noted above, some of the single failures listed above 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 LAR. System redundancy, independence, and diversity are preserved. The single failures listed above do 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 license bases. No fission product barrier is directly affected. It follows that the independence of the fission product barriers is not affected. The design basis steam generator tube rupture sequences to be excluded do not in themselves lead to any degradation of independence of the fission product barriers. As noted above, the dependence on recovery for the sequences to be removed from the plant license basis is very limited. No changes in the operation of any structure, system or component are

associated with the changes proposed in this LAR. 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 by any of the single failures listed above. The Emergency Core Cooling System also remains capable of "poison addition." Compliance with GDC 27 and GDC 35 is not degraded with the changes in this proposed license amendment. 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 by any of the single failures in this proposed license amendment.

None of these single failures 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 with any of the failures listed above. 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 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 EDE / EDF. The failure of Diesel Generator Load Sequencer reset does not affect the ability of the Diesel Generator Load Sequencers to load engineered safeguards onto the 4160 volt switchgear on the Safety Injection or "blackout" signal. Conformance of the Diesel Generator Load Sequencers to GDC 17 is not degraded with the reset failure.

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 changes proposed within this LAR.

It is concluded that sufficient defense-in-depth is retained with the exclusion of the steam generator tube rupture sequences with the single failures listed above from the license 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 sequences from the license 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. 19). 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 failures described above affect the ability to throttle or stop some of the engineered safeguards equipment, not to start them. None of the failures identified above would 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. None of the failures listed above will degrade the Class 1E electric power systems so as to cause "loss of power to … devices sufficient to jeopardize the safety of the station." None of the single failures noted above will "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. 20) 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. With respect to DNBR, the design basis steam generator tube rupture is determined to be bounded by the Complete Loss of Forced Reactor Coolant Flow (UFSAR Section 15.3.2).

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 any of the single failures listed above would be manifested. Minimum DNBR would occur within seconds after reactor trip. Therefore, for all cases, the criterion concerning DNBR is met. For all cases to be retained within the license 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 license basis are within the appropriate guideline values. The risk evaluation in Section 3.4 demonstrates that the frequency

of steam generator overfill associated with the steam generator tube rupture sequences to be excluded is low (approximately 3.0 E-06 per reactor year). Additionally, the frequency of a large early release is shown to be very low (approximately 3.0 E-10 per reactor year). It is concluded that sufficient margin exists to account for analytical and data uncertainty for these steam generator tube rupture sequences (cf. Section 3.4).

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 license 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 probabilities,
- Estimate the frequency of the initiating event,
- Screen out low frequency sequences,
- Identify possible recovery actions for unscreened single failures,
- Establish operator action times and quantify the non-recovery probabilities, and
- Evaluate the Core Damage Frequency (CDF) / Large Early Release Frequency (LERF) significance of remaining sequences relative to the criteria of RG-1.174.

The estimates for the relevant parameters are developed as follows.

3.4.1) Hardware Failure Probabilities

The hardware failures that are the subject of this license amendment request are failures that may result in the potential for leading to steam generator overfill. The six failure modes of interest have been identified previously. The hardware failure rates have 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 values have 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. The generic prior information and the plant specific failure information are provided for each component considered. Log-normal distributions are assumed unless otherwise noted.

Failure of 125 V dc Vital I&C Power Distribution Center EDE or EDF

Failure of EDE (EDF) during normal operation would be readily apparent through the undervoltage alarms that would be actuated. Since the accident analysis indicates that the leakage through the ruptured tube is terminated within 2 hours, a mission time of 2 hours is assumed for the bus failure. The failure probability estimated below is applicable to each distribution center.

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. The failure probability is estimated assuming a pre-mission exposure time of two hours, based on the Technical Specification Allowable Outage Time, and a mission time of two hours. A failure that occurs beyond two hours has no impact on risk from steam generator overfill following the steam generator tube rupture. The flows into the ruptured steam generator have been stabilized and a satisfactory steady state has been achieved. Failure of the bus beyond this time does not cause any transient to occur to destabilize the condition and result in steam generator overfill. The probability that the bus is unavailable following a steam generator tube rupture is estimated to be 1.6E-06.

Inadvertent swap of auxiliary feedwater controls from the Control Room to the Auxiliary Feedwater Pump Turbine Control Panel

This failure mode may be caused by various transfer circuit failures: loss of power from EDE or EDF, blown control

circuit fuses, or spurious operation of the transfer switch. A spurious swap is the consequence of any of these failure modes. The probability of each of the individual component failures that have this consequence are evaluated and summed in order to estimate the probability of a spurious swap. Failure of EDE or EDF would result in the transfer, but this failure is considered separately because it has consequences beyond the swap of the controls. This failure mode is immediately recognizable in the control room from the alarms that are received after a transfer occurs. The failure probability is estimated assuming a pre-mission exposure time of six hours, based on the mean time to repair experienced in two events, and based on the Technical Specification Allowed Outage Time. For this failure, the motor-driven and turbine-driven auxiliary feedwater pump would be declared inoperable and the associated Action would require the unit be placed in Mode 3, Hot Standby, within six hours. The assumed mission time is two hours. The single failure to be eliminated is the spurious swap of a train of controls from the Control Room to the panel, not the failures of the individual components. Therefore, the sum of the individual contributors is adopted as the probability of the failure mode of interest. The following data has been used in estimating the failure rates for those components whose failure may result in an inadvertent transfer to the auxiliary feedwater pump turbine control panel.

Component	**Prior Mean**	**EF**	**Plant Specific Failures**	**Exposure**	**Posterior Mean**
DC breaker	6.7E-07/hr	32	0	2.42E06 hours	8.5E-08/hr
Fuse	6.3E-07/hr	9.4	N/A[1]		
Switch	8.0E-08/hr	13	0	1.61E06 hours	5.1E-08/hr

Component Failure	**Failure rate (/hr)**	**Exposure time (hr)**	**Failure Probability**
Train A			
Breaker EDE-F01G transfers open	8.5E-08	8	6.8E-07
Transfer switch 1TH contact 4-4C spuriously opens	5.1E-08	8	4.1E-07
Fuse F-39 fails	6.3E-07	8	5.0E-06
Fuse F-40 fails	6.3E-07	8	5.0E-06
Total			**1.1E-05**

[1] The generic value is used. No plant specific data collected.

Train B			
Breaker EDF-F01G transfers open	8.5E-08	8	6.8E-07

It is seen that train A is the limiting case, and that train will be addressed in the sequence frequency analysis.

Inability to close High Pressure Injection Flow Valve NI9A or NI10B from the Control Room

The probability of this failure mode is dominated by failure of the valves themselves to close. The failure to close is assumed to be entirely random and independent of the fact that the valve has recently opened successfully. That is, no reduction in the failure to close probability is assumed because of the success to open on demand (i.e., this valve is closed prior to the accident, and is automatically opened upon a safety injection signal). The only individual control component considered in the failure probability is the Solid State Protection System relay in the valve close circuit. Failure of this relay can prevent the valve from closing. Other control and operator (e.g., motor, torque switch) component failure rates are assumed to be included in the motor operated valve (MOV) failure rate. This is consistent with the plant specific data collection process and no detailed modeling of components in the MOV controls is explicitly included in the MOV failure rate. The failure probability estimated below is applicable to each valve.

A generic valve failure rate of 3.5E-03/demand, mean value, with an error factor of 2.2 is assumed as a prior distribution. Based on Maintenance Rule failure data, the plant specific information is 0 failures in approximately 454 demands. The resulting failure rate for "MOV Fails to Close" is 2.6E-03/demand. The relay failure makes an insignificant contribution to the overall failure probability. With a single demand on the valve, the failure probability becomes 2.6E-03.

Inability to reset a Safety Injection Signal

Reset of the safety injection signal is accomplished by picking-up the unlatch coil of the actuation relay. A generic relay failure rate of 1.9E-04/demand, mean value, with an error factor of 9.0 is assumed as a prior

distribution. Based on Maintenance Rule failure data, the plant specific information is 0 failures in approximately 18,683 demands. The resulting failure rate for "Relay Fails on Demand" is 3.6E-05/demand. Demand failure rates for pushbuttons are typically much smaller (estimated to be 1E-05/D) than for relays and this failure is neglected. With a single demand on the reset function for the Safety Injection Signal, the failure probability is estimated to be 3.6E-05.

Inability to reset the Diesel Generator Load Sequencer

Reset of the diesel generator load sequencer is accomplished by picking-up the reset relay (RRA2 for train A). The same data used to quantify the Solid State Protection System reset failure is used here. Single failure of the diesel generator load sequencer reset is dominated by a failure of the diesel generator load sequencer reset relay RRA2 or fuse HK. The relay demand failure rate has been estimated to be 3.6E-05. The fuse is not normally carrying current; however, after the reset pushbutton is pressed, it must carry current to energize the reset relay coil. For this condition a demand failure rate is appropriate. The databases reviewed do not include data of this type for fuses. A fuse is a simple passive device with no moving parts and would be expected to have a low failure rate compared to more complicated active components. As a screening value, the fuse is assigned the same demand failure rate as the reset relay, resulting in an estimated reset failure probability of 7.2E-05/demand. With a single demand on the reset function for the diesel generator load sequencer (DGLS), the failure probability is estimated to be 7.2E-05.

Inability to Secure a Safety Injection (SI) pump

The following components contribute to an inability to secure the A train safety injection pump. The B train pump failure is estimated in the same manner. The trip switch failure is assumed to make a negligible contribution.

Component	**Failure Rate**
Pump Breaker (4 kV) fails to trip	9.2E-04/demand
DGLS A relay ESGAX1 fails to de-energize	3.6E-05/demand

DGLS A relay SAA2 fails to de-energize	3.6E-05/demand
DGLS A relay SAA5 fails to de-energize	3.6E-05/demand
Relay K608 fails to de-energize	3.6E-05/demand
Total	**1.1E-03/demand**

The relay failure rate estimation has been previously described.

Failure of the 4 kV breaker to trip is quantified from the following information. A generic breaker failure rate of 1.2E-03/demand, mean value, with an error factor of 4.0 is assumed as a prior distribution. Based on Maintenance Rule failure data, the plant specific information is 0 failures in approximately 316 demands. The resulting failure rate for "Breaker (4 kV) Fails to Trip" is 9.2E-04/demand.

With a single demand, the failure probability for failure to stop the safety injection pump is estimated to be 1.1E-03.

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 95th percentile values of 7.0E-03 and 1.4E-02 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) Sequence Analysis

The sequence analysis is performed, to the extent practical, using best estimate parameters. Some sequences may be of such low frequency that detailed evaluation is not warranted. Sequence frequencies are evaluated relative to an appropriate screening criterion to identify sequences that warrant detailed evaluation.

Sequence Screening

American National Standard ANSI/ANS 51.1-1983 (Ref. 5) provides a framework for determining which accidents are of sufficiently high frequency to warrant consideration in the design. From Section 3.2.3 Optional Approach,

"… a probabilistic assessment may be performed to determine the likelihood of the combination of the initiating occurrence plus a single failure or the coincident occurrences, or both. … If the frequency of occurrence of an event is shown to be $<10^{-6}$/reactor year on a best estimate basis, this event shall not be considered for the design. …"

This screening criterion is applied to the sequences of interest in this analysis. Sequences which are identified to have frequencies greater than 1E-06/RY are evaluated in greater detail. Those falling below the criterion are assumed to contribute negligibly to risk and require no further analysis. The following table presents the results of considering the steam generator tube rupture frequency when combined with the single failure probabilities estimated previously. The results given are for one train.

Single Failure	SGTR Frequency	Estimated Single Failure Probability per Train	Frequency of Plant Condition (per RY)
Isolation Valve Fails to Close	6.8E-03	2.6E-03	1.8E-05
Safety Injection Pump Fails to Trip	6.8E-03	1.1E-03	7.5E-06
Diesel Generator Load Sequencer Fails to Reset	6.8E-03	7.2E-05	**4.9E-07**
Safety Injection Signal Train Fails to Reset	6.8E-03	3.6E-05	**2.4E-07**
Train Inadvertent Swap to Auxiliary Feedwater Pump Turbine Control Panel	6.8E-03	1.1E-05	**7.5E-08**
Failure of Distribution Center	6.8E-03	1.6E-06	**1.1E-08**

It is seen that the frequency of a steam generator tube rupture combined with either a failure of the diesel generator load sequencer to reset, a failure of the safety injection signal to reset, an inadvertent swap to the auxiliary feedwater pump turbine control panel, or a failure of the distribution center falls below the screening criterion of 1E-06/RY. On the basis of satisfying this screening criterion, it is judged that these failure modes are not risk significant and no further analysis of these 4 failure modes is presented.

The design basis analysis also assumes the occurrence of a loss of offsite power (LOOP) coincident with the steam generator tube rupture. The occurrence of a loss of offsite power impacts the analysis by making the reactor coolant pumps and instrument air unavailable. This extends the time required to cool down the RCS. While a loss of offsite power is the conservative assumption for the design basis analysis, the likelihood of occurrence should be considered in a probabilistic analysis that is intended to be a best estimate evaluation. Therefore, the coincidence of a loss of offsite power with a steam generator tube rupture is investigated for significance against the adopted screening criterion for the remaining failures. NUREG/CR-6538 (Ref. 6) provided an analysis of the probability of a loss of offsite power conditional on a Reactor trip and Emergency Core Cooling System actuation. The resulting probability of 0.014 has been adopted here for estimating the frequency of sequences consisting of a

steam generator tube rupture with loss of offsite power and a single failure.

Single Failure	**SGTR Frequency**	**Conditional Probability of a LOOP**	**Estimated Single Failure Probability per Train**	**Frequency of Plant Condition (per RY)**
Isolation Valve Fails to Close	6.8E-03	1.4E-02	2.6E-03	**2.5E-07**
Safety Injection Pump Fails to Trip	6.8E-03	1.4E-02	1.1E-03	**1.0E-07**

The frequency of sequences involving a loss of offsite power fall below the screening criterion of 1E-06/RY. Therefore, no loss of offsite power is assumed to occur when considering the impact of the single failures on the response to a steam generator tube rupture. This is important in the thermal hydraulic analysis for estimating the time available to the operators to take compensatory action. A best estimate time can be arrived at by assuming the availability of the reactor coolant pumps and instrument air.

In conclusion, of the 12 (6 per train) single failures originally identified, only 4 (2 per train) represent failure probabilities that result in transients of meaningful frequency. When evaluating the significance of the remaining failures, no loss of offsite power needs to be assumed as these sequences would be probabilistically insignificant.

The failures to be considered further affect the steam generator tube rupture response by inhibiting the rapid termination of safety injection when the necessary conditions in the RCS have been established. Remedial action, sometimes outside the Control Room, is required by the operating crew in order to terminate safety injection. The time available for recovery from the failure is estimated from a thermal hydraulic analysis of the steam generator tube rupture event.

Steam Generator Tube Rupture Thermal Hydraulic Analysis

The thermal hydraulic analyses discussed below are based on the Catawba Unit 1 steam generators. These generators have

been identified to be limiting with respect to Unit 2 with respect to steam generator overfill.

Typically, the design basis steam generator tube rupture overfill analysis adopts a number of conservative assumptions. These include:

- Steam generator level instrument errors that maximize the initial liquid inventory
- Loss of offsite power
- Double-ended guillotine break of a tube

For the purposes of this analysis, these assumptions are modified to represent a best estimate set of criteria. This allows the estimation of the important human error probabilities to be conducted on a realistic basis. While conditions other than those assumed are certainly possible, the likelihood of having one or more of these parameters significantly deviating from the best estimate value reduces the overall frequency of occurrence for such sequences. For this analysis the following conditions are assumed:

- Steam generator initial liquid level is nominal
- Reactor coolant pumps and instrument air are available
- An average value for steam generator tube rupture flow rate occurs

Not all of the tube rupture events that have occurred in the industry have exhibited flow rates representative of a double ended guillotine break of a tube, the usual design basis assumption. For a risk informed analysis, a best estimate flow is desirable as a modeling approach. There is inadequate plant specific experience on which to develop a best estimate break flow. As an alternative, the historical evidence from actual steam generator tube ruptures is used as a basis for developing a best estimate flow. The steam generators in the population that have experienced steam generator tube ruptures are a mix of designs with a variety of tube diameters, materials, water chemistry, and age. It is likely that few, if any, actually represent a set of conditions that accurately represent the Catawba generators (which are themselves different on the two units). By limiting the size of the sample in an attempt to find generators most like one of Catawba units, the uncertainty in the average obtained

increases due to the limited data included in the estimate. The most unbiased process is to assume the average flow from the actual events. This flow data has been obtained from NUREG/CR-6365 and is presented in the table below. The average flow rate is found to be 388 gallons per minute (gpm). The break flows calculated in the design basis analyses for the Catawba units are 440 gpm and 560 gpm for Units 1 and 2 respectively. The estimate used in this analysis represents 88% and 69% of the design basis flow rates and this is judged to be a reasonable range for this parameter.

Plant	**Flow Rate (gpm)**
Fort Calhoun	112
Point Beach Unit 1	125
Doel Unit 2	135
Palo Verde Unit 2	240
Surry Unit 2	330
Prairie Island Unit 1	336
McGuire Unit 1	500
North Anna Unit 1	637
Mihama Unit 2	700
Ginna Unit 1	760

The prevention of steam generator overfill following a steam generator tube rupture involves three important operator evolutions:

- Identification and isolation of the ruptured generator
- Cooldown and depressurization of the RCS
- Termination of safety injection

The failures being evaluated affect only the termination of safety injection. However, the time frames for accomplishing these actions are not completely independent of each other. Significant delays in the isolation of the generator or depressurization of the RCS clearly impact the time available to terminate safety injection. In establishing the time available to the operators for remedial action, it has been assumed that the first 2 actions have occurred in a manner consistent with the design basis steam generator tube rupture overfill analysis. This does not mean that the actions occur at the same time as in the design basis analysis. It means that the actions are taken when the same conditions and indications, used to determine the operator action time in the design basis calculation, are satisfied. It is assumed

that this process provides a time available that is appropriate for the calculation of the failure to recover probabilities.

As specified in plant emergency procedures, the actions to terminate safety injection begin following the RCS depressurization.

Following safety injection termination, break flow continues until primary and secondary pressures equalize. Therefore, termination must occur while there is sufficient steam space available in the generator to absorb this flow. This volume is determined from the thermal-hydraulic analysis. The time available is estimated from the thermal-hydraulic simulations of the scenario. The steam volume at the end of the Reactor Coolant System depressurization (safety injection termination criteria satisfied) is taken from the thermal-hydraulic calculation results. The amount of water that flows to the steam generator following safety injection termination is obtained from the results and this volume is subtracted. The remaining volume is available in the generator to accommodate the break flow while safety injection termination is attempted. The average break flow for the 15 minutes prior to reaching the safety injection termination criteria is used as the estimated break flow rate. The time available to terminate safety injection is then estimated as the volume of steam available divided by the estimated break flow and is found to be approximately 14 minutes.

Using the assumptions described above, the margin to steam generator overfill available when the safety injection termination criteria are met leaves approximately 14 minutes for the operators to take the appropriate action. This is an increase of approximately 7 minutes over the time estimated using the conservative boundary conditions. The time available is considered along with the expected time to complete the recovery to estimate the non-recovery probabilities.

There is some small conservatism introduced by the manner in which the thermal hydraulic analysis is conducted. The analysis assumes that all of the Emergency Core Cooling System flow continues regardless of which single failure is being evaluated. For example, when considering the failure of valve NI9A to close, it should be assumed that the

safety injection pumps have been stopped. This would reduce the safety injection flow and increase the time available to close the valve. However, the thermal hydraulic analysis assumes that all pumps are providing flow. The simplified approach taken here reduces the required number of thermal hydraulic analyses.

The method selected for determining a best estimate flow has limitations regarding its applicability, as do any of the alternative methods. The degradation mechanisms that are dominant in a specific generator may likely have the most influence on what kind of break a particular generator may experience. The break flow used in the evaluation, while judged to be a reasonable estimate, may or may not be a good estimate for the specific conditions of the Catawba steam generators. As such, the break flow rate, through its impact on the human reliability analysis, represents an important source of uncertainty in the estimated frequencies. Some perspective on the significance of this uncertainty is included in the "Discussion of Uncertainty" section of the LAR.

Human Reliability Analysis - Recovery From Inability to close Valves NI9A or NI10B

Should NI9A or NI10B fail to close, the response not obtained (RNO) instruction calls for operators to be dispatched to the valves to close the affected valve(s). There are two components to the failure to recover probability. First, there is the human response to identify the need and then to correctly take action to close the valve. The human response is conveniently broken down into a cognitive phase and an action phase. The second component is the hardware failure probability.

The human response is analyzed using the human cognitive reliability (HCR) methodology, Reference 9. It is estimated that 14 minutes are available after the Safety Injection termination criteria are satisfied before the steam generator would overfill. Once the operator determines that the criteria have been met, it is estimated that 2 minutes will be required to get to the step to close the subject valves. The execution time is estimated to be 9 minutes. Several walkdowns of these valves have been performed. The radiation levels in the rooms containing these valves are invariably low. That is, an operator

performing the action within the time required would receive a radiation dose far below the limit of NUREG-0737 (5 Rem). Valves 1NI10B, 2NI9A, and 2NI10B are easily accessible with the use of ladders. Valve 1NI9A may be somewhat more difficult to access as an equipment operator will have to wear fall protection to reach it. Ladder storage racks exist in the general area near both valve rooms, increasing the likelihood of ladder availability. A test has been performed to show that an operator can reach the handwheel of one of these valves in 4.5 minutes. A separate test was conducted to show that an operator could close one of these valves with its handwheel against flow from one high pressure injection pump in 45 seconds. This justifies the assumption concerning the time an operator would take to execute this step in this risk informed analysis. The execution time is not the same for all of the valves (1NI9A, 1NI10B, 2NI9A, and 2NI10B) due to variations in location. The assumed time is in the middle of the expected range of execution times. The resulting error probability for this phase is estimated to be 6.0E-02. The response has been evaluated as rule-based. An additional error contribution during the action phase of 3.0E-03 has been included. The action is assessed to be a simple action that occurs outside the Control Room. The total human error probability is estimated to be approximately 6.3E-02 for this recovery.

The hardware failure is assessed as the conditional probability that the valve fails to close locally given that it has failed to close remotely. For the sequences under consideration, the valve opened successfully less than 2 hours prior to the attempt to close. This suggests that the failure to close might be reduced from the nominal value for failure on demand since those components required for travel in both directions are demonstrated to be functioning during the opening of the valve. However, since no data is available to provide a basis for the magnitude of such a reduction, none is assumed here.

The probability of failure of the valve to close locally given that it failed to close remotely is assumed to be related to the failure rate of a manual valve. Failure rates for MOV's are substantially higher than those for manual valves. The difference is judged to reflect the influence of the valve operator failures on the failure rate. The failure of a manual valve to close is estimated to be 2.9E-04/demand. For the sequence under

consideration, failure of the operator or the controls can be recovered by local manual operation of the valve. The demand failure rate for a manual valve is approximately 11% of the rate of a motor operated valve. Therefore, NI9A or NI10B is assumed to be non-recoverable through local manual operation 11% of the time.

The total non-recovery probability is the sum of the human and hardware failure probabilities. For this analysis, the failure to recover from failure of NI9A or NI10B is estimated as:

$$P_{non\text{-}rec} = 0.063 + 0.11 = 0.17$$

Recovery From Failure of Safety Injection Pump to Trip

Should the safety injection pumps fail to trip, no procedural guidance is provided; there is no RNO for this step. Flow from the safety injection pumps to the RCS is easily terminated by closing the valves downstream of the pumps. Multiple valves operable from the Control Room are available for accomplishing this action. Closing these valves does not threaten the pumps since the minimum flow path is still available.

The human response is also analyzed with the HCR methodology, in this case using the knowledge-based curve. Once the safety injection termination criteria are satisfied, approximately 14 minutes are available for re-aligning the necessary valves. The estimated median time to accomplish this is two minutes. The operators are specifically directed to trip the safety injection pumps in the emergency procedures for the steam generator tube rupture. Failure of one of these pumps to trip in response to an operator pressing the trip pushbutton would be clearly indicated in the control room. The operators are aware of the option to close the isolation valve from the pump discharge to the Reactor Coolant System. This justifies the time span of two minutes assumed for the operators to recover from failure of a safety injection pump to trip when an operator presses its trip pushbutton. The non-response probability is estimated as:

$$P_{non\text{-}rec} = 0.01$$

Overfill Frequency Analysis

The frequency of steam generator overfill sequences as a result of a steam generator tube rupture are quantified as the product of the steam generator tube rupture frequency and the probabilities of subsequent failures, hardware and human, that result in overfill. The relevant sequences for this analysis are presented in the following table.

Single Failure	**SGTR Frequency**	**Estimated Single Failure Probability per Train**	**Failure to Recover From ECCS Termination Failure**	**Frequency of Overfill (per RY)**
Isolation Valve Fails to Close	6.8E-03	2.6E-03	1.7E-01	3.0E-06
Safety Injection Pump Fails to Trip	6.8E-03	1.1E-03	1.0E-02	7.5E-08

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 RCS 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 RCS 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 inconsequential. With the RCS dose equivalent iodine at the Technical Specification limit, offsite exposures would increase but remain quite 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 Reference 7, a conditional probability of core damage can be estimated. Core damage occurs due to failure to depressurize the RCS 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	Probability of Relief Valve Failure to Reseat	Conditional Probability of Core Damage	Frequency of Uncontrolled Release as a Result of Overfill
Isolation Valve Fails to Close	3.0E-06	1.0E-01	1E-03	3.0E-10
Safety Injection Pump Fails to Trip	7.5E-08	1.0E-01	1E-03	7.5E-12

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 very small. Furthermore, this frequency is a very small fraction of the ΔLERF criterion of 1.0E-07 stated in Regulatory Guide 1.174. The estimated base case LERF for Catawba Nuclear Station is 4.3E-07/year.

Main steam line failure is also a possible (though much less likely) consequence of steam generator overfill. Using the estimates from Reference 7, the LERF's 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.

The best estimate tube rupture flow rate, and ultimately the non-recovery probabilities, are judged to be one of the more significant sources of uncertainty in the analysis. This is especially true in the analysis for the recovery from failure of the safety injection isolation valves (NI9A and NI10B). The execution time for this action is long

relative to the estimated time available. Rupture flow rates that are significantly smaller would add considerable time to the estimate; the assumption of a guillotine break and the maximum possible flow through the rupture would result in a non-recovery probability of 1 for this sequence. This worst case assumption would result in an increase in the frequency for this sequence by a factor of approximately 6. The LERF contribution from the "Isolation Valve Fails to Close" sequence would increase to 1.8E-09/RY with this assumption. However, even in this case the sequence frequency is low and remains an insignificant contributor to LERF.

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 recently identified single failures have been reviewed for risk significance. A few were screened out due to low frequency of the initiating sequence. Those for which a more detailed evaluation has been developed were found to contribute very little to CDF and LERF. The total contribution that these sequences is estimated to make to the LERF for Catawba is approximately 6.2E-10 / 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. These sequences are not expected to contribute meaningfully to the risk estimates for Catawba, and their exclusion from the license 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. Furthermore, the addition of additional components or controls needed to make termination of the safety injection and auxiliary feedwater functions more reliable, may actually reduce their reliability for the more risk significant sequences.

3.5) Monitoring Program

A risk based evaluation has been performed of a number of single failures 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 may be associated with these single failures as follows:

1) 125 VDC Vital I&C Distribution Centers EDE and EDF: 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.

2) Auxiliary Feedwater Pump Turbine Control Panel Transfer Circuits: Preclude inadvertent transfer of control of auxiliary feedwater control and isolation valves from the Control Room to the auxiliary feedwater pump turbine control panel.

3) Solid State Protection System Trains A and B: Manual reset of the safety injection signal.

4) Diesel Generator Load Centers A and B: Manual reset of the diesel generator load sequencers following reset of the safety injection signal.

5) Safety Injection Pumps: Manual trip.

6) Motor Operated Isolation Valves NI9A and NI10B: remote manual closure (i.e., from the Control Room).

These functions either are 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), or will be added to the program during implementation of this license amendment.

1) **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) American National Standard ANSI/ANS-51.1-1983, Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor.

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) Hannaman, G. W., et al., Human Cognitive Reliability Model for PRA Analysis, EPRI Report RP-2847-1 (Interim Report), December 1984.

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.

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

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

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 certain sequences be excluded from consideration in the analysis of the design basis steam generator tube rupture event. These sequences involve single failures that potentially degrade the ability to terminate auxiliary feedwater flow into a ruptured steam generator following a steam generator tube rupture or that potentially degrade the ability to terminate Emergency Core Cooling System injection flow into the Reactor Coolant System, thereby prolonging primary system flow into the ruptured steam generator. The inability to terminate auxiliary feedwater flow or emergency core cooling system injection 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 accident initiators. 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 or to terminate Emergency Core Cooling System flow into the Reactor Coolant System. 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. With respect to DNBR, the design basis steam generator tube rupture is bound by the Complete Loss of Forced Reactor Coolant Flow. Should one of the steam generator tube rupture sequences to be excluded occur, the effects 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, for all cases, 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 3.0 E-06 per reactor year per Class 1E Train). Additionally, the frequency of a large early release (Section 3.4.4) is shown to be very low (approximately 3.0 E-10 per reactor year per Class 1E Train).

It is concluded that removal of certain steam generator tube rupture sequences from the plant license 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

EVNIRONMENTAL 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 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.