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A Duke Energy Company

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March 1, 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
Proposed Amendment to the Facility Operating Licenses and Technical Specification 1.1, Definitions - Dose Equivalent I-131

Pursuant to 10CFR50.90, Duke Energy Corporation is requesting amendments to the Catawba Nuclear Station Facility Operating Licenses (FOL) and Technical Specifications (TS). The changes being proposed by Duke Energy Corporation in this License Amendment Request (LAR) will amend the Facility Operating License to remove the license condition imposing restrictions on dose-equivalent iodine levels. Following approval of this amendment, the limit for dose-equivalent iodine will revert to those values contained in Limiting Condition for Operation (LCO) 3.4.16, Reactor Coolant System Specific Activity. No change is being proposed to this LCO.

This amendment will also revise Technical Specification 1.1, Definitions, by revising the definition of Dose Equivalent I-131. Specifically, this change will revise the source of the thyroid dose conversion factors used in determining Dose Equivalent I-131 from TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites" to Federal Guidance Report No. 11, "Limiting Values

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of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion." Precedent for this change to Federal Guidance Report No. 11 was established in Amendments No. 173 and 177 to Facility Operation License Nos. DPR-24 and DPR-27 for the Point Beach Nuclear Plant Units 1 and 2.

This license amendment request is also requesting that certain single failure scenarios potentially leading to steam generator overfill be excluded from the licensing basis for the 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."

In addition to the change in thyroid dose conversion factors, the analysis used to support this request contains additional features that differ from those in the analysis of record. These features include changes to the Control Room atmospheric dispersion factors, whole body dose conversion factors, and skin dose conversion factors for Control Room operators, and the accident initiated iodine spike factor and include a credit for Reactor Coolant System cleanup prior to isolation of Reactor Coolant System letdown.

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. Based on this assessment, the restrictions on Dose-Equivalent I-131 imposed in the license condition are unnecessarily restrictive, and the license condition may be removed. Also in the re-analysis of the design basis steam generator tube rupture event, the source document for obtaining thyroid dose conversion factors was revised. The definition of Dose-Equivalent I-131 is being revised to reflect this change.

The contents of this amendment package are as follows:

Attachment 1 provides marked up Facility Operating Licenses for Catawba Nuclear Station, Units 1 and 2.

Attachment 2A provides the marked-up Technical Specification page. Attachment 2B provides the reprinted Technical Specification page.

Attachment 3 provides a Description of the Proposed Changes and Technical Justification.

Pursuant to 10CFR50.92, Attachment 4 documents the determination that the amendment contains No Significant Hazards Considerations.

Pursuant to 10CFR51.22(c)(9), Attachment 5 provides the basis for the categorical exclusion from performing an Environmental Assessment/Impact Statement.

Attachment 6 contains the input data used to calculate the Control Room Atmospheric Dispersion Factor with ARCON96. Meteorological data used for the calculation is provided in electronic form.

Attachment 7 contains the detailed dose analysis input used to calculate the doses to the Control Room operators and at the Exclusion Area Boundary and the Low Population Zone.

Implementation of this amendment to the Catawba Facility Operating License and Technical Specifications will impact the Catawba UFSAR. Specifically, Section 2.3, Meteorology, will be revised to reflect revised X/Q values for the control room clean air intakes. Section 15.6.3, Steam Generator Tube Failure, will be revised following approval of the amendment request to document the changes in the analysis approach and assumptions and the derived results of the analysis. Necessary changes will be made in accordance with 10CFR50.71(e).

NRC approval of this LAR is requested by September 1, 2000, or as soon as practical. The analysis of the consequences of the steam generator tube rupture design basis accident has concluded that the limits contained in LCO 3.4.16 are the appropriate limits for operation of Catawba Nuclear Station. Based on the results of the analysis, the limits imposed by the license condition are overly conservative and could result in unnecessary unit shutdowns.

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.

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,

Gary R. Peterson

G.R. Peterson

Attachments

xc w/attachments:

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AFFIDAVIT

G. R. Peterson, being duly sworn, states that he is Site Vice President of Duke Energy Corporation; that he is authorized on the part of said corporation to sign and file with the Nuclear Regulatory Commission this amendment to the Catawba Nuclear Station(s) Facility Operating Licenses Numbers 50-413 and 50-414 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

G.R. Peterson, Site Vice President

Subscribed and sworn to me: 03-01-2000
Date

Micky Love Standrif
Notary Public

My Commission Expires: 06-26-2002
Date

ATTACHMENT 1

MARKED-UP FACILITY OPERATING LICENSE PAGES
FOR CATAWBA NUCLEAR STATION UNITS 1 AND 2

APPENDIX D

ADDITIONAL CONDITIONS

FACILITY OPERATING LICENSE NO. NPF-35

Duke Energy Corporation shall comply with the following conditions on the schedules noted below:

Amendment Number	Additional Condition	Implementation Date
~~159~~	~~This amendment requires the licensee to incorporate in the Updated Final Safety Analysis Report (UFSAR) certain changes to the description of the facility. Implementation of this amendment is the incorporation of these changes as described in the licensee's application dated March 7, 1997, as supplemented by letters dated April 2, 10, 16, 22, and 28, 1997, and evaluated in the staff's Safety Evaluation dated April 29, 1997.~~ (Deleted by Amendment No. 164)	~~Next update of the UFSAR~~
~~159~~	~~This amendment requires the licensee to use administrative controls, as described in the licensee's letter of March 7, 1997, and evaluated in the staff's safety evaluation dated April 29, 1997, to restrict the dose-equivalent iodine levels to 0.46 microCurie per gram (in lieu of the limit in TS Section 3.4.8.a), and to 26 microCurie per gram (in lieu of the limit of TS Figure 3.4-1), until this license condition is removed by a future amendment.~~ (Deleted by Amendment No.).	~~Immediately upon issuance of the amendment~~
173	The licensee is authorized to relocate certain requirements included in Appendix A to licensee-controlled documents. Implementation of this amendment shall include the relocation of these requirements to the appropriate documents, as described in the licensee's letters dated May 27, 1997, as amended by letters dated March 9, March 20, April 20, June 3, June 24, July 7, July 21, August 5, September 8, and September 15, 1998, and evaluated in the NRC staff's Safety Evaluation associated with this amendment.	All relocation to be completed by January 31, 1999.

APPENDIX D

ADDITIONAL CONDITIONS

FACILITY OPERATING LICENSE NO. NPF-52

Duke Energy Corporation shall comply with the following conditions on the schedules noted below:

Amendment Number	Additional Condition	Implementation Date
~~159~~	~~This amendment requires the licensee to incorporate in the Updated Final Safety Analysis Report (UFSAR) certain changes to the description of the facility. Implementation of this amendment is the incorporation of these changes as described in the licensee's application dated March 7, 1997, as supplemented by letters dated April 2, 10, 16, 22, and 28, 1997, and evaluated in the staff's Safety Evaluation dated April 29, 1997.~~ (Deleted by Amendment No. 164)	~~Next update of the UFSAR~~
~~159~~	~~This amendment requires the licensee to use administrative controls, as described in the licensee's letter of March 7, 1997, and evaluated in the staff's safety evaluation dated April 29, 1997, to restrict the dose equivalent iodine levels to 0.46 microCurie per gram (in lieu of the limit in TS Section 3.4.8.a), and to 26 microCurie per gram (in lieu of the limit of TS Figure 3.4-1), until this license condition is removed by a future amendment.~~ (Deleted by Amendment No.)	~~Immediately upon issuance of the amendment.~~
165	The licensee is authorized to relocate certain requirements included in Appendix A to licensee-controlled documents. Implementation of this amendment shall include the relocation of these requirements to the appropriate documents, as described in the licensee's letters dated May 27, 1997, as amended by letters dated March 9, March 20, April 20, June 3, June 24, July 7, July 21, August 5, September 8, and September 15, 1998, and evaluated in the NRC staff's Safety Evaluation associated with this amendment.	All relocation to be completed by January 31, 1999.

ATTACHMENT 2A

MARKED-UP TECHNICAL SPECIFICATION PAGE

1.1 Definitions (continued)

CHANNEL CHECK	A CHANNEL CHECK shall be the qualitative assessment, by observation, of channel behavior during operation. This determination shall include, where possible, comparison of the channel indication and status to other indications or status derived from independent instrument channels measuring the same parameter.
CHANNEL OPERATIONAL TEST (COT)	A COT shall be the injection of a simulated or actual signal into the channel as close to the sensor as practicable to verify the OPERABILITY of required alarm, interlock, and trip functions. The COT shall include adjustments, as necessary, of the required alarm, interlock, and trip setpoints so that the setpoints are within the required range and accuracy.
CORE ALTERATION	CORE ALTERATION shall be the movement of any fuel, sources, or reactivity control components, within the reactor vessel with the vessel head removed and fuel in the vessel. Suspension of CORE ALTERATIONS shall not preclude completion of movement of a component to a safe position.
CORE OPERATING LIMITS REPORT (COLR)	The COLR is the unit specific document that provides cycle specific parameter limits for the current reload cycle. These cycle specific parameter limits shall be determined for each reload cycle in accordance with Specification 5.6.5. Unit operation within these limits is addressed in individual Specifications.
DOSE EQUIVALENT I-131	DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcuries/gram) that alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in ~~Table III of TID-14844, AEC 1962, "Calculation of Distance Factors for Power and Test Reactor Sites."~~ Federal Guidance Report No. 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion".
$\bar{E}$—AVERAGE DISINTEGRATION ENERGY	$\bar{E}$ shall be the average (weighted in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling) of the sum of the average beta and gamma energies per disintegration (in MeV/d) for isotopes, other than iodines, with half lives > 10 minutes, making up at least 95% of the total noniodine activity in the coolant.

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ATTACHMENT 2B

REPRINTED TECHNICAL SPECIFICATION PAGE

1.1 Definitions (continued)

CHANNEL CHECK	A CHANNEL CHECK shall be the qualitative assessment, by observation, of channel behavior during operation. This determination shall include, where possible, comparison of the channel indication and status to other indications or status derived from independent instrument channels measuring the same parameter.
CHANNEL OPERATIONAL TEST (COT)	A COT shall be the injection of a simulated or actual signal into the channel as close to the sensor as practicable to verify the OPERABILITY of required alarm, interlock, and trip functions. The COT shall include adjustments, as necessary, of the required alarm, interlock, and trip setpoints so that the setpoints are within the required range and accuracy.
CORE ALTERATION	CORE ALTERATION shall be the movement of any fuel, sources, or reactivity control components, within the reactor vessel with the vessel head removed and fuel in the vessel. Suspension of CORE ALTERATIONS shall not preclude completion of movement of a component to a safe position.
CORE OPERATING LIMITS REPORT (COLR)	The COLR is the unit specific document that provides cycle specific parameter limits for the current reload cycle. These cycle specific parameter limits shall be determined for each reload cycle in accordance with Specification 5.6.5. Unit operation within these limits is addressed in individual Specifications.
DOSE EQUIVALENT I-131	DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcuries/gram) that alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in Federal Guidance Report No. 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion."
$\bar{E}$ — AVERAGE DISINTEGRATION ENERGY	$\bar{E}$ shall be the average (weighted in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling) of the sum of the average beta and gamma energies per disintegration (in MeV/d) for isotopes, other than iodines, with half lives > 10 minutes, making up at least 95% of the total noniodine activity in the coolant.

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ATTACHMENT 3

DESCRIPTION OF PROPOSED CHANGE
AND TECHNICAL JUSTIFICATION

Table of Contents

1) Overview

Duke Energy Corporation is requesting removal of the License Condition imposed in License Amendment 159 for Unit 1 and License Amendment 151 for Unit 2. An analysis of the consequences of a design basis steam generator tube rupture has been completed and it has been determined that the limits on Reactor Coolant System specific activity in Technical Specification (TS) 3.4.16 (previously TS 3.4.8) are the appropriate limits. Therefore, no changes to Improved TS Limiting Condition for Operation 3.4.16, Reactor Coolant System (RCS) Specific Activity, are being requested. This License Amendment Request provides the justification used to reach this conclusion.

The analysis described in this License Amendment Request is based on some assumptions that differ from the assumptions made in the current analysis. One of the changes involves a change to the definition of Dose Equivalent Iodine, contained in Technical Specification 1.1, Definitions.

Duke Energy Corporation also 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.

2) Description of Changes

Duke Energy Corporation requests removal of the license conditions concerning dose-equivalent iodine specific activity in the Reactor Coolant System (RCS). The license condition was imposed in Amendment 159 for Catawba Nuclear Station Unit 1, Facility Operating License NPF-35 and

Amendment 151 for Catawba Nuclear Station Unit 2, Facility Operating License NPF-52. The Facility Operating License for Unit 2 contains a typographical error. The License indicates that this license condition is associated with Amendment 159, rather than 151, which is the correct amendment number for Unit 2. The license condition requires Duke Energy Corporation to use administrative controls to restrict dose-equivalent iodine levels to 0.46 microCurie per gram (in lieu of the limit in TS Section 3.4.8a) and to 26 microCurie per gram (in lieu of the limit of TS Figure 3.4-1), until this license condition is removed by a future amendment. It should be noted that the references to specific Technical Specifications are out of date. Subsequent to creation of this license condition, Improved Technical Specifications were implemented at Catawba Nuclear Station, but the text of the license condition was not revised. In the Improved Technical Specifications, the limits for dose-equivalent iodine are contained in Limiting Condition for Operation 3.4.16.

This amendment request presents the technical justification for removal of this license condition for both units. Following approval of this amendment, restrictions on dose-equivalent iodine will revert to the requirements of Technical Specification Limiting Condition for Operation 3.4.16.

Duke Energy Corporation is proposing an amendment to the Definition of Dose Equivalent I-131 contained in Section 1.1 of the plant TS. The definition currently states that the thyroid dose conversion factors used for the calculation of Dose Equivalent I-131 shall be those listed in Table III of TID-14844, AEC, 1962, "Calculation of Distance Factors for Power and Test Reactor Sites." This amendment is requesting that the definition be changed to identify the source of the thyroid dose conversion factors as Federal Guidance Report No. 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion." Precedent for this change to Federal Guidance Report No. 11 was established in Amendments No. 173 and 177 to Facility Operation License Nos. DPR-24 and DPR-27 for the Point Beach Nuclear Plant Units 1 and 2.

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. Spurious 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

Additionally, the analysis contains some new features. The Control Room atmospheric dispersion factors have been revised. Meteorological data used in the calculation of the new atmospheric dispersion factors are attached to this License Amendment Request. The whole body and skin dose conversion factors for the Control Room operators were taken from Federal Guidance Report Nos. 11 and 12 except as delineated in Section 4.2.1.2. The accident initiated iodine spike factor is set to 335 in the analyses of the consequences of the design basis Steam Generator Tube Rupture. This is the value presented in draft Regulatory Guides DG-1074, "Steam Generator Tube Integrity" and DG-1081, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors." A methodology change involving credit for reactor coolant system radioactivity cleanup prior to letdown isolation is also contained in the analysis. These changes will be reflected in site calculation packages and/or the Updated Final Safety Analysis Report (UFSAR) following approval of this amendment. UFSAR Section 2.3, Meteorology, will be revised to reflect revised X/Q values for the Control Room clean air intakes. UFSAR Section 15.6.3, Steam Generator Tube Failure, will also be revised.

3) Background

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.

The Staff's SER required additional plant specific input for each utility referencing WCAP-10698. 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 failures were identified as failure of a power operated relief valve on an intact steam generator to open on demand (limiting for steam generator overfill) and a stuck open power operated relief valve on a ruptured Steam Generator (limiting for radiological consequences).

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 3.7.16 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 result in having perhaps only the Power Operated Relief Valve for the ruptured steam generator available for remote operation thereby extending 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 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 Steam Generator because auxiliary feedwater flow 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. These limits are more restrictive than those contained in the license condition for dose equivalent iodine.

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. Duke Energy Corporation is requesting that these single failure sequences be removed from the Steam Generator Tube Rupture design basis as a risk informed licensing action.

4) Technical Justification

The technical justification consists of two major sections. In the first, Section 4.1, the case is presented for the exclusion of design basis Steam Generator Tube Rupture sequences including certain single failures from the plant licensing basis. The limiting sequences of single failures that are retained in the licensing basis have been developed for the analysis of the radiological consequences of the design basis Steam Generator Tube Rupture Analysis. This analysis is discussed in Section 4.2. Together, these sections provide the information and evaluation from which the determination of no significant hazards is made.

4.1) Risk Based Exclusion of Single Failures

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. 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.

4.1.1) 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.

4.1.1.1) Failure of 125 V dc Vital I&C Power Distribution Center EDE or 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 a spurious swap of the auxiliary feedwater controls from the Control Room to the Auxiliary Feedwater Pump Turbine Control Panel as described below.

4.1.1.2) Spurious 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.

4.1.1.3) 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.

4.1.1.4) 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.

4.1.1.5) 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 trip 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).

4.1.1.6) 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.

4.1.2) 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. 17 - 19). 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. 20). 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 identified.

In the staff's SER for this study, licensees were required to perform analyses to verify that the conclusions of the generic study (margin to steam generator overfill) applied to each plant (Ref. 21). 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. Duke Energy Corporation responded that the results of the Westinghouse generic study, including the single failure analysis, bounded Catawba Nuclear Station (Ref. 18).

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. 15). 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. 15), 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. 16). The effects of some of these failures will be 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. (This modification has been installed on Unit 1 and is scheduled to be installed on Unit 2 during its spring 2000 Refueling Outage.) The effects of some failures have been removed 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 licensing 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 licensing 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.

4.1.3) 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 licensing 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 licensing 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.

4.1.3.1) 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 4.1.1). 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. The effects of any of these single failures in concurrence with a design basis event in UFSAR Chapter 15 are bounded by the results of the safety analyses of these design basis events. 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 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 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" sequence. 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. 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 sequence.

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 somewhat. 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 4.1.4). 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. 11, cf. Ref. 21). 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 seen 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 would 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 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 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 may include activities such as administrative controls associated with limits on initial and boundary conditions assumed in the analysis of design basis events. They also may 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. They included 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 licensing 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. A combination of modifications and administrative controls have been developed and/or implemented to mitigate the consequences of some of these single failures. Air accumulator tanks will have been installed on the lines supplying instrument air to the auxiliary feedwater flow control valves. (This modification has been installed on Unit 1 and is scheduled to be installed on Unit 2 during its spring 2000 Refueling Outage.) 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 23), 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. Also approved was 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 isolation valves for the Steam Generator Power Operated Relief Valves which may be closed has been restricted (to one per Class 1E train). 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. It is concluded that over reliance on programmatic activities to compensate for weaknesses 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 licensing 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 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. It 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 (GDC) (e.g., GDC 20 - GDC 24, GDC 34, GDC 35, GDC 38, GDC 41) 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.

4.1.3.2) 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. 25). 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 noted above affect the ability to throttle or stop some of the engineered safeguards equipment, not to start them. The ability of the Solid State Protection System to perform its safety function is not degraded with any of the failures noted above. 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. 25) 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 bound 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. 3, 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 (cf. Section 4.2.3). The risk evaluation in Section 4.1.4.3 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 (Section 4.1.4.4) 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 4.1.4.4).

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.

4.1.4) Evaluation of Risk Impact

As stated in Section 4.1, 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.

4.1.4.1) 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. The probability that the bus is unavailable following a Steam Generator Tube Rupture is estimated to be 1.6E-06.

Spurious 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. 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 a mission time of 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 a spurious 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**
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

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

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 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 total 7.2E-05.

Inability to Secure a Safety Injection (SI) pump

The following components contribute to an inability to secure the SI-A 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.

4.1.4.2) 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. 8). 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.

4.1.4.3) 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 51.5-1983 (Ref. 9) 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 Spurious 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, a spurious 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. 10) 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.

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 by considering the volume of steam in the generator at the time that termination begins, subtracting the volume transferred following termination and dividing by the break flow during the time period.

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 13. 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. 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 non-response probability is estimated as:

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

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

4.1.4.4) 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 Ref. 11, 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 Ref. 11, 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.

4.1.4.5) 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.

4.1.5) 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 spurious 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. 5), or will be added to the program during implementation of this license amendment.

4.2) Analysis of Radiological Consequences of the design basis Steam Generator Tube Rupture

As noted above, Duke Energy Corporation requests approval for the removal of certain design basis Steam Generator Tube Rupture sequences from the license basis of Catawba Nuclear Station. The sequences to be removed are associated with certain single failures. With this amendment, Duke Energy Corporation will retain a number of design basis Steam Generator Tube Rupture / single failure sequences within the license basis of Catawba Nuclear Station.

The analyses of radiological consequences of the design basis Steam Generator Tube Rupture were performed with the use of the LOCADOSE computer code (References 32-34). LOCADOSE has been purchased from the Bechtel Corporation by Duke Energy Corporation under a software license agreement (Reference 35). This computer code was developed to be used for the analyses of radiological consequences of any design basis event as it allows the user to specify a network of volumes and flow paths through which activity may be transported. LOCADOSE calculates the activity transport through these user-defined volumes and from there to the environment and to the control room. In particular, radioactivity in the Control Room is computed using the time-dependent Murphy-Campe Equation (Radioactive decay of the specific radionuclide of interest is not accounted for in the Murphy-Campe formulation found in Reference 27. LOCADOSE not only solves for Control Room specific activity in a time dependent fashion, but it also includes the effects of radioactive decay).

Failure analyses performed at Catawba determined that the radiological consequences of design basis Steam Generator Tube Rupture scenarios with the following single failures bounded the radiological consequences of all other design basis Steam Generator Tube Rupture scenarios:

1) False high indication of chlorine from a Class 1E chlorine detector of the Control Room Area Ventilation System (CRAVS). This causes the inadvertent closure of the associated outside air valve of the CRAVS, degrading Catawba from a dual intake plant to a single intake plant.

2) Stuck open Power Operated Relief Valve on the ruptured Steam Generator (Ref. 14).

3) Failure of Class 1E Power to the controls of the Power Operated Relief Valves of two intact Steam Generators (Ref. 15).

Radiological consequences of design basis Steam Generator Tube Rupture scenarios including these single failures have been analyzed. These analyses are reported below.

While not specifically called out in the following section as a new feature of the analysis, the dose calculation results submitted in this license amendment include necessary features to bring the Steam Generator Tube Rupture dose calculation into conformance to EPRI TR-107621, "Steam Generator Tube Integrity Assessment Guidelines" (Ref. 6). Specifically, Reactor Coolant System leakage as allowed by plant Technical Specifications has been heretofore inadvertently omitted from the equilibrium iodine production rate term, and letdown density being different than Reactor Coolant System density has also been unaccounted for in previous calculations of equilibrium iodine production rate. The analysis described in this license amendment request include these two effects.

4.2.1) New Features of the Analysis

The analysis of radiological consequences of the design basis Steam Generator Tube Rupture incorporates a number of new features. First, new values of atmospheric dispersion factors (*X/Q*'s) have been calculated. These include a new *X/Q* for release of radioactivity from the Steam Generator Doghouse. A new set of dose conversion factors has been developed and employed in this analysis. A new value of the multiplier for the production rate for the accident-initiated iodine spike has been used. Finally, credit is taken for letdown cleanup for the time span between the initiating Steam Generator Tube Rupture and letdown isolation. These new features are discussed below.

4.2.1.1) Control Room Atmospheric Dispersion Factors

Values of the atmospheric dispersion factors (*X/Q*'s) at the location of the Control Room air intakes are obtained for releases of radioisotopes from the unit vent stacks, Steam Generator Power Operated Relief Valves, Main Steam Code Safety Valves, and Auxiliary Feedwater Pump turbine exhausts. The *X/Q* values are calculated with the ARCON96 Computer Code.

ARCON96 has three release types as options in the computer code. These options are: 1)ground, 2) vent, and 3)elevated releases. No radionuclide release points at Catawba Nuclear Station can be considered elevated. The ground release option is more conservative in calculated *X/Q* results when compared to identical releases using the vent release option and as a result only the ground release option is used.

The Control Room air intakes at Catawba constitute a dual intake arrangement per Standard Review Plan 6.4.

The *X/Q* at the Control Room intakes for releases from the Unit Vents is determined from the highest of the following source-receptor pairs:

1) Unit 1 Vent to Control Room Unit 1 Intake
2) Unit 2 Vent to Control Room Unit 2 Intake

The *X/Q* from the Unit 1 Vent to Control Room Unit 2 Intake and from the Unit 2 Vent to Control Room Unit 1 Intake were not calculated. Since in these cases the source-receptor distance is much greater, the *X/Q* values for these source-receptor pairs are bounded by those listed above, even when considering potential wind directional frequency differences.

The *X/Q* values for the Control Room Intakes for releases from the Doghouses were determined from the highest of the following source-receptor pairs:

1) Unit 1 Inboard Doghouse Auxiliary Feedwater Turbine exhaust to the Control Room Unit 1 Intake
2) Unit 2 Inboard Doghouse Auxiliary Feedwater Turbine exhaust to the Control Room Unit 2 Intake

The X/Q's for the Doghouse source-receptor pairs listed below are considered bounded by the above *X/Q*'s. This is explained in the following paragraphs:

1) Unit 1 Outboard Doghouse to Control Room Unit 1 Intake
2) Unit 1 Outboard Doghouse to Control Room Unit 2 Intake
3) Unit 1 Inboard Doghouse to Control Room Unit 2 Intake
4) Unit 2 Outboard Doghouse to Control Room Unit 2 Intake
5) Unit 2 Outboard Doghouse to Control Room Unit 1 Intake
6) Unit 2 Inboard Doghouse to Control Room Unit 1 Intake

There are three potential release points from the Doghouse roofs. These are

1) Steam Generator Power Operated Relief Valve
2) Main Steam Safety Valve
3) Auxiliary Feedwater Turbine exhaust

The Outboard Doghouses have both Main Steam Safety Valves and Power Operated Relief Valves but do not have auxiliary feedwater turbine exhausts. Steam releases from both the Power Operated Relief Valves and Main Steam Safety Valves are ejected upwards with high velocity. The ARCON96 computer code allows releases of this type to be modeled as either a ground release or a vent release. In the ground release option ARCON96 assumes that the receptor is on the axis of the effluent plume. This assumption ignores the effect of vertical separation. In the case of an Outboard Doghouse release (Main Steam Safety Valve or Power Operated Relief Valve) to the nearest Control Room Intake this is overly conservative. The over conservatism arises because the closest Control Room Intake will not be on the effluent plume axis due to the combination of 1) a 50′-6″ vertical

Doghouse releases. An additional physical consideration when examining possible contamination of a Control Room Intake from the closest Outboard Doghouse is that the effluent plume will have significant plume rise due to the large vertical velocity and the high temperature of the steam effluent. The large vertical velocity will eject the steam high above the initial release point at which point the steam's high temperature will cause the plume to rise even more. The plume rise of the effluent steam precludes contamination of a Control Room intake from main steam safety valve or power operated relief valve releases from the closest Outboard Doghouse.

Both of the Inboard Doghouses have Auxiliary Feedwater Turbine exhausts in addition to Main Steam Safety Valves and Power Operated Relief Valves as potential release points. The volumetric flow and hence the vertical velocity of the Auxiliary Feedwater Turbine exhaust releases are much less than the Main Steam Safety Valves or the Power Operated Relief Valves. In addition, the Auxiliary Feedwater Turbine exhaust release points are closer to the Control Room Intakes than the Main Steam Safety Valves and the Power Operated Relief Valves. For these reasons, the Auxiliary Feedwater Turbine exhaust releases result in the highest Control Room X/Q's of all release points on the Doghouse.

In addition, the distance from the Unit 1 Auxiliary Feedwater Turbine exhaust to the Control Room Unit 1 Intake is substantially closer and in the same general direction as both the Unit 2 Inboard and Outboard Doghouses to Control Room Unit 1 Intake. Similarly, the Unit 2 Auxiliary Feedwater Turbine exhaust to the Control Room Unit 2 Intake is substantially closer and in the same general direction as both the Unit 1 Inboard and Outboard Doghouses to Control Room Unit 2 Intake. Hence, the Unit 1 Inboard Doghouse to Control Room Unit 1 Intake and Unit 2 Inboard Doghouse to Control Room Unit 2 Intake become the limiting source-receptor pairs for all the Doghouses.

Due to a single failure potential, the dose analysis for Control Room habitability at Catawba assumes that the clean Control Room air intake is isolated for the first ten hours of a radiological release event and only the contaminated Control Room Intake is available. After ten hours both intakes are assumed to be open (Reference 2).

As a result the averaging periods for Catawba are:

0-8 hours
8-10 hours
10-24 hours
1-4 days
4-30 days.

The meteorological data used for determining the *X/Q* values at the Control Room intakes is in compliance with Regulatory Guide 1.23.

The following table contains the ARCON96 input used in the evaluation of the Control Room (CR) intake *X/Q* values.

ARCON96 INPUT

Input Parameter	Unit 1 Vent to Unit 1 CR Intake	Unit 2 Vent to Unit 2 CR Intake	Unit 1 AFW Turbine Exhaust to Unit 1 CR Intake	Unit 2 AFW Turbine Exhaust to Unit 2 CR Intake
No. of Meteorological Data Files:	3	3	3	3
Height of upper wind instrument (meter):	10	10	10	10
Height of lower wind instrument (meter):	40	40	40	40
Units of wind speed data:	Miles per hour (mph)	Mph	mph	Mph
Type of Release:	Ground	Ground	Ground	Ground
Release height (meters):	38.2	38.2	16.8	16.8
Building cross sectional area (sq. meter):	1571	1571	1571	1571
Effluent vertical velocity (meter/sec):	0	0	0	0
Vent flow rate (cubic meter/sec)	2.8	2.8	11.0	11.0
Vent radius (meters):	0	0	0	0
Direction from intake to source (degrees):	45.2	134.8	10.8	169.2
Distance to intake (meter):	45.2	45.2	38.2	38.2
Intake height (meter):	1.4	1.4	1.4	1.4
Terrain level difference (meter):	0.0	0.0	0.0	0.0
Minimum wind speed (meters/second):	0.5	0.5	0.5	0.5
Surface roughness length (meter):	0.1	0.1	0.1	0.1
Sector Averaging constant:	4.0	4.0	4.0	4.0
Wind direction sector width (degrees):	90	90	90	90

As noted above, the Control Room *X/Q* values for the first 10 hours after an initiating event represent a single intake configuration. The *X/Q* values for the remaining time of interest (i.e., 10 hr - 720 hr) represent the dual intake configuration. In this manner, the calculation conforms to a commitment made to the NRC in Ref. 2. Specifically, Duke Energy Corporation committed to consider the effects of a failure causing an outside air intake valve of the CRAVS to close and remain closed for the first 10 hours following a postulated accident. In the past, Duke Energy Corporation has kept this commitment by doubling the values of the Control Room *X/Q*'s (calculated for a two intake plant) for the first 10 hours after the initiating event. The values of *X/Q*'s had been doubled regardless of the single failure taken in the analyses of radiological consequences of accidents. Recently, Duke Energy Corporation performed a failure analysis in which the following was determined:

1) Only one failure within the plant design basis causes the spurious closure of a CRAVS outside air intake valve; specifically, the failure of a Class 1E CRAVS chlorine detector.

2) Failure of a CRAVS chlorine detector shares no mode of failure with any other failure of a Class 1E component at Catawba Nuclear Station.

As noted in Ref. 2, the non safety related controls to the CRAVS outside air intake valves have been removed by plant modification. Only the associated Class 1E chlorine detector will cause a CRAVS outside air intake valve to close. Thus, the full values of the *X/Q*'s are taken in calculation of post accident radiation doses to the Control Room operators only in the case of failure of a Class 1E CRAVS chlorine detector. In the absence of a failure of a chlorine detector, Catawba Nuclear Station may be seen as a plant with two CRAVS outside air intakes with no capacity for automatic isolation. Therefore, the full *X/Q* values calculated as discussed above will be used in the calculation of radiation doses to Control Room operators following a postulated accident only with failure of a Class 1E CRAVS chlorine detector. When a single failure other than a failure of a chlorine detector of the CRAVS is assumed, the following adjustment will be made in keeping with the corresponding section of Ref. 3, Section 6.4. The values of Control Room *X/Q*'s of the first 10 hours following the initiating event, representative of a single intake plant for that time span, will be reduced by a factor of 2. The Control Room *X/Q* values for the remaining time span (10 hours - 720 hours) representative of a dual intake configuration will be used without adjustment.

4.2.1.2) Dose Conversion Factors

The dose conversion factors used for all radiological analyses were taken from Federal Guidance Report Nos. 11 and 12 (except as delineated below). Dose conversion factors are also used in the relevant Catawba Nuclear Station Chemistry Procedures to collapse the iodine isotope spectrum to a single isotope, where the isotopic spectrum becomes I-131 dose equivalent iodine specific activity for purposes of comparison with the Technical Specifications. The design basis iodine spectrum from the UFSAR was assumed to exist in all radiological analyses, and hence the proposed Technical Specification limit was converted from I-131 dose equivalent specific activity to specific activity of each iodine isotope by use of these dose conversion factors. To ensure consistency between the dose calculation methodology and the plant chemistry program, the definition of "Dose Equivalent I-131" (Definitions, Section 1.1) is being revised to show that the dose conversion factors are as listed in Federal Guidance Report No. 11.

The dose conversion factors from Federal Guidance Report No. 11 are used for all thyroid dose calculations. The "effective" dose coefficients from Federal Guidance Report No. 12 are used for all whole body dose calculations, for both iodine and noble gas radionuclides. The dose coefficients used for calculations of skin radiation doses to the Control Room operators are not taken from Federal Guidance Report No. 12. Rather, an adjusted set of skin dose coefficients has been used, with the adjustments being performed in internal Duke Energy calculations. Federal Guidance Report No. 12 does not specifically list the skin dose coefficients by beta and gamma contribution, or the fractions of the skin dose coefficient contributed by each particle. Duke Energy worked with Dr. Keith Eckerman of Oak Ridge National Laboratory to obtain a complete listing of skin dose coefficients for all significant radionuclides categorized by particle type. The dose coefficients were computed with the same methodology as the original values in Federal Guidance Report No. 12, and the contributions by particle sum to the values printed in this report.

The adjustment made by Duke Energy involves decreasing the skin dose coefficient for photons where the skin dose coefficient is divided by the control room geometry factor as described by Murphy and Campe (Ref. 27). Duke Energy also performed independent shielding and dose calculations to verify the accuracy and validity of the Control Room geometry factor calculation as summarized in Ref. 27.

The reason for this adjustment pertains to the nature of the derivation of the dose coefficients listed in Federal Guidance Report No. 12. All dose coefficients listed in this document are for exposure in a semi-infinite (or infinite hemisphere) cloud uniformly contaminated with a radionuclide. Exposure in the Control Room is not semi-infinite, and most dose calculation computer codes account for this effect by dividing the effective doses by the Murphy and Campe geometry factor, which is computed from the volume of the licensee's control room. However, the LOCADOSE computer code, used for all dose calculations, does not automatically correct skin doses by dividing the computed doses by the geometry factor. The treatment for skin doses must be examined separately. Duke Energy Corporation has concluded that it is appropriate to apply this dose correction factor to the beta particle contribution to skin dose given the range of beta particles in air. The Murphy and Campe geometry factor was computed for photon contributions to whole body dose, rather than for skin doses.

The range of beta particles in air is approximately

$$Range_{AIR} \approx \frac{12 ft.}{MeV}.$$

The maximum range in air is about 20 feet. Hence, consideration of a finite volume such as the Control Room versus an infinite volume yields no discernible decrease in dose to skin from beta particles. So the geometry factor correction is not applied to the skin dose coefficient from beta particles.

However, the application of the geometry factor correction is appropriate for the dose to skin from photons. Therefore the total skin dose coefficient in use in all radiological calculations performed herein is computed as follows:

$$DCF_{SKIN} = DCF_{\beta} + \frac{DCF_{\gamma}}{GF_{MURPHY-CAMPE}}.$$

The geometry factor used was specifically computed for the Catawba Nuclear Station control room, and so the skin dose coefficients in use are specific to Catawba Nuclear Station. This methodology has been discussed with Dr. Keith Eckerman of Oak Ridge National Laboratory, and the computations have been reviewed in Duke Energy calculations.

4.2.1.3) Accident-initiated Iodine Spike Factor

The accident-initiated iodine spike factor is set to 335 in the analyses of radiological consequences of the design basis Steam Generator Tube Rupture. This value is cited by the Nuclear Regulatory Commission Staff in Draft Regulatory Guides DG-1074 and DG-1081. The spike factor will also be set to 335 for the Break of a Small Line Carrying Reactor Coolant Outside Containment (UFSAR Section 15.6.2). The response of the affected nuclear unit to this accident would be similar to the Steam Generator Tube Rupture. In particular, the Small Line Break accident is marked by the absence of any actuation of the Reactor Protection System or any Engineered Safety Features. There would be very little change in the Reactor Coolant System pressure following a Small Line Break accident, since this accident is analyzed with the assumption that the charging is able to replenish the lost inventory, and safety injection is not automatically started. For these reasons, setting the spike factor at 335 for the analysis of radiological consequences of this accident is justified.

In order to verify the validity and accuracy of the spike factor of 335, Duke Energy worked with Staff personnel at the NRC and PNL (Pacific Northwest Laboratory) to obtain the raw data used to develop the recommendation for the new spike factor. Some of the raw information is cited in Ref. 28, while the data reduction is discussed but not demonstrated or performed in this reference. Duke Energy performed the data reduction and computations in order to substantiate the use of this spike factor. Duke Energy agrees that the recommended spike factor represents the spike rate associated with the data cited in Ref. 28, with a statistical adjustment (i.e., the spike factor of 335 is a 95^{th} percentile value).

4.2.1.4) Credit for Letdown Cleanup Prior to Letdown Isolation

Duke Energy is submitting a methodology change, wherein credit is assumed in the dose calculations for cleanup of the Reactor Coolant System radioactivity in the iodine spike prior to letdown isolation. No credit is assumed after letdown isolation, since a safety injection signal is initiated, isolating normal charging and letdown. Duke Energy has performed the necessary assessments to ensure that it is appropriate to assume this credit. These assessments are summarized below.

Duke Energy has a rigorous specification for procurement of demineralizer resins, including, but not limited to, testing and quality control outlined in standards such as ASTM D2687-84, ASTM D5627-94 and ASTM D2187-94. Tested resin decontamination factors (equal to the inlet concentration divided by the outlet concentration) range from 20 to 70, and are as high as 10,000 for fresh ion exchange resin. Based on studies of Reactor Coolant System chemistry, the cited removal rate is very unlikely to degrade due to an increase in specific activity due to defective fuel, pH change within the range of 3 - 9, or aeration of the reactor coolant.

Iodine fission products form a mixture of ions (I^- and IO_3^-) in reactor coolant water. These ions are in chemical equilibrium with dissolved molecular iodine (I_2). Passing the coolant through nuclear grade strong base anion exchange resin results in high removal rates due to the ability of the resin to remove and split the iodine salts. As the resin removes the ions, molecular iodine is forced to form more ions to maintain chemical equilibrium. These ions, in turn, are ion-exchanged. This process continues until a much lower iodine concentration is present. Additionally, iodine forms simple ionic salts, such as with cesium, lithium and sodium in the reactor coolant (sodium is present due to trace impurities in the boric acid, makeup water, and nuclear grade ion exchange resin). The iodide ions in these salts are rapidly and effectively removed by the ion exchange resin. Should the reactor coolant become aerated, iodine removal is generally unaffected.

In EPRI sponsored studies, it has been demonstrated that even under the worst conditions (i.e., aerated radwaste iodine-to-resin interactions) no other radionuclide (transition metal or cesium fission product) was exchanged as rapidly or completely as iodine. Moreover, it would be inconsistent to include iodine removal by the demineralizers as a contribution to the iodine spiking term prior to the accident, yet not credit removal by the same demineralizers after the accident. For purposes of computation of the pre-accident equilibrium release rate of iodine from defective fuel (which is an input into the spike rate), the demineralizer efficiency is conservatively set to 100%. For letdown cleanup of Reactor Coolant System radioactivity after the accident, but prior to reactor trip, the demineralizer efficiency is set to 95%.

All of the relevant Chemical and Volume Control System piping is class B or C (ASME III, Classes 2 and 3 respectively), such that it can be relied upon in a seismic event. The demineralizers are a completely passive means of removing iodine from the Reactor Coolant System, and involve no active components. No part of the demineralizers is subject to single failure criteria.

The assumed continued operation of the letdown system is considered reasonable since this would not fall in the category of a demand failure. That is, nothing is required to actively start or change state to perform the function credited in the dose calculation. The letdown mixed bed demineralizers are assumed to operate until letdown isolation. During this period, the simultaneous failure of the components in the letdown system and a Steam Generator Tube Rupture is highly improbable. There are several valves in the letdown flow stream that could sustain a spurious failure. The combined failure frequency of any of these components along with a Steam Generator Tube Rupture during this period of time is 4.0 E-08. The failure frequency discussed here is for non-safety grade components, since failure of safety-grade components in the letdown line would be considered the single active failure for the accident, and would differ from the other failures assumed in the analysis to maximize doses.

There is a possibility for a Steam Generator Tube Rupture following a loss of letdown purification. For those low probability conditions in which this occurs along with a Reactor Coolant System specific activity that is very close to the limit of 1 uCi/gm, and in which the equilibrium concentration would otherwise drift upwards above 1 uCi/gm, the dose results are considered to be enveloped by the pre-existent iodine spike case.

Expected early actions by the operators to trip the reactor and initiate safety injection are not credited in this dose calculation. The time to reactor trip is protracted well beyond what would be expected by operator actions. Even with the absence of letdown flow purification, early reactor trip, safety injection and letdown isolation are considered to be bounded by the assumptions in this dose analysis. Early reactor trip mitigates the releases accounted for in this dose calculation with the assumption of continued operation, and this conservatism obviates the absence of letdown purification in such an event.

4.2.2) Survey of the design basis Steam Generator Tube Rupture Scenarios Analyzed

A number of scenarios were selected for the analysis of radiological consequences of the design basis Steam Generator Tube Rupture at Catawba Nuclear Station. The radiation consequences calculated included the following:

1) Radiation doses to the whole body at the Exclusion Area Boundary (EAB), Low Population Zone boundary (LPZ), and Control Room operators.

2) Radiation doses to the skin and eyes of the Control Room operators.

3) Radiation doses to the thyroid gland at the EAB and LPZ, and in the Control Room.

Radiological consequences of design basis Steam Generator Tube Rupture scenarios were calculated for each nuclear unit at Catawba Nuclear Station.

Two different iodine spikes were postulated to occur in conformance to the Standard Review Plan (Ref. 3). First, a pre-accident iodine spike was assumed to occur before the design basis Steam Generator Tube Rupture. Under this condition, the DEI specific activity in the RCS was assumed to be at the transient limit associated with 100% power (60 uCi/gm, Ref. 1, TS 3.4.16, Action A.1). The same set of design basis Steam Generator Tube Rupture scenarios were analyzed with the assumption that a transient had occurred with the accident, producing the accident-initiated iodine spike. For these design basis Steam Generator Tube Rupture's, the initial dose equivalent Iodine 131 (DEI) specific activity in the RCS was set to the equilibrium limit (1 uCi/gm, cf. Ref. 1, LCO 3.4.16). The accident-initiated iodine spike is marked by an increase in the "production rates" of iodine radioisotopes in the RCS from their equilibrium values. A value of 335 was taken for the multiplier of the equilibrium production rates, as discussed above. The equilibrium production rate for each iodine radioisotope was calculated in conformance to the guidelines of Ref. 6. In particular, a bounding value (125 GPM) was taken to account for the possibility of two letdown flow paths placed in service during unit power operations. In addition, the letdown flow rate was referenced at standard conditions (given that reactor coolant has passed through the letdown heat exchanger and is at 110 ^{o}F as it passed through the flow element). Finally, limiting values of Identified and Unidentified Reactor Coolant System Leak Rates (TS 3.4.13) are taken into account in the calculating the production rate of iodine radioisotopes for the accident-initiated iodine spike.

In all cases, a value of 0.1 uCi/gm was taken for the initial DEI equilibrium specific activity in each Steam Generator (Ref. 1, TS 3.7.17). All four Steam Generator Power Operated Relief Valves were assumed to be initially operable (Ref. 1, TS 3.7.4).

The contributions to radiation doses from noble gases were calculated with the initial gross gamma activity in the RCS set to the limit of LCO 3.4.16 (100/E-bar). Activity of iodine radioisotopes as well as noble gas isotopes was included in the calculation of these external radiation doses. The pre-accident iodine spike was taken in the calculation of external radiation doses from iodine radioisotopes.

The following three single failures were included in the calculation of radiological consequences of the design basis Steam Generator Tube Rupture.

1) A Class 1E CRAVS chlorine detector was assumed to fail high. This causes the inadvertent closure of the associated outside air valve of the CRAVS, degrading Catawba from a two intake plant to a single intake plant (cf. Ref. 3, Section 6.4). Given the relatively short duration of releases of radioactivity from the plant following a design basis Steam Generator Tube Rupture, no recovery from this failure is postulated for the entire time of these releases (Ref. 4).

2) The Power Operated Relief Valve on the ruptured steam generator was assumed to fail to close when the pressure in the ruptured steam generator falls below the close setpoint of the steam generator Power Operated Relief Valve. This results in uncontrolled releases of steam from the ruptured steam generator until the operators close the associated Class 1E isolation valve. For the design basis Steam Generator Tube Rupture with stuck open Power Operated Relief Valve on the ruptured steam generator, the operators close the isolation valve for the stuck open Power Operated Relief Valve 10 minutes after it would have closed on pressure below its CLOSE setpoint and decreasing.

3) Power from 120 VAC Class 1E Panelboard ERPA (ERPD) is assumed to fail. The worst effect of this failure is loss of power to the controls for the Power Operated Relief Valves of two intact Steam Generators. The operators are assumed to recover from this failure by going to a Steam Generator doghouse and operating one of the affected Power Operated Relief Valves with its handwheel. This operator action can be completed within 18 minutes of initiating RCS cooldown.

4.2.3) Radiological Consequences

The regulatory limits and guideline values are as follows:

For the Steam Generator Tube Rupture with a pre-accident iodine spike,

1) Whole body radiation doses at the exclusion area boundary (EAB) should not exceed 25 Rem and the thyroid radiation doses at the EAB should not exceed 300 Rem for an individual located at any point on the EAB or two hours immediately following onset of the postulated fission product release.
2) Whole body radiation doses at the boundary of the low population zone (LPZ) should not exceed 25 Rem and thyroid radiation doses at the LPZ should not exceed 300 Rem for an individual located at any point on the LPZ outer boundary who is exposed to the radioactive cloud resulting from the postulated fission product release during the entire period of its passing.

For the Steam Generator Tube Rupture with accident-initiated iodine spike, the calculated doses should not exceed a small fraction of the above guideline values, i.e.,10 percent or 2.5 Rem and 30 Rem, respectively, for the whole body and thyroid doses.

For the Steam Generator Tube Rupture,

1) Whole body radiation doses in the Control Room should not exceed 5.0 Rem.
2) Skin radiation doses in the Control Room should not exceed 30 Rem.
3) Thyroid Radiation doses in the Control Room should not exceed 30 Rem.

The results of the analysis of the radiological consequences of the limiting of the design basis Steam Generator Tube Rupture scenarios to be retained in the plant license basis are listed below. Radiation doses to the thyroid gland are reported for the cases of design basis Steam Generator Tube Rupture with both the pre-accident iodine spike and accident-initiated iodine spike. However, radiation doses to the whole body and skin are reported for the cases of the design basis Steam Generator Tube Rupture with only the pre-accident iodine spike. For each design basis Steam Generator Tube Rupture sequence analyzed, the radiation doses to the whole body and skin are higher in the cases of the pre-accident iodine spike than in the cases of the accident-initiated iodine spike.

Radiological Consequences of Design Basis Steam Generator Tube Rupture (DB SGTR) Scenarios

Radiation Dose (Rem)

Unit 1 DB SGTR with CRAVS Chlorine Detector Failure	
EAB Whole Body Dose	0.18
EAB Thyroid Dose (Accident-Initiated Iodine Spike)	13.0
EAB Thyroid Dose (Pre-accident Iodine Spike)	31.5
LPZ Whole Body Dose	0.03
LPZ Thyroid Dose (Accident-Initiated Iodine Spike)	2.78
LPZ Thyroid Dose (Pre-accident Iodine Spike)	5.92
Control Room Whole Body Dose	0.03
Control Room Skin Dose	1.06
Control Room Thyroid Dose (Accident-Initiated Spike)	3.49
Control Room Thyroid Dose (Pre-accident Iodine Spike)	15.5
Unit 1 DB SGTR with Stuck Open PORV on the Ruptured S/G	
EAB Whole Body Dose	0.19
EAB Thyroid Dose (Accident-Initiated Iodine Spike)	15.2
EAB Thyroid Dose (Pre-accident Iodine Spike)	36.7
LPZ Whole Body Dose	0.03
LPZ Thyroid Dose (Accident-Initiated Iodine Spike)	3.28
LPZ Thyroid Dose (Pre-accident Iodine Spike)	6.78
Control Room Whole Body Dose	0.02
Control Room Skin Dose	0.55
Control Room Thyroid Dose (Accident-Initiated Spike)	1.81
Control Room Thyroid Dose (Pre-accident Iodine Spike)	6.48
Unit 1 DB SGTR with ERPA / ERPD Failure	
EAB Whole Body Dose	0.19
EAB Thyroid Dose (Accident-Initiated Iodine Spike)	15.0
EAB Thyroid Dose (Pre-accident Iodine Spike)	35.3
LPZ Whole Body Dose	0.03
LPZ Thyroid Dose (Accident-Initiated Iodine Spike)	3.27
LPZ Thyroid Dose (Pre-accident Iodine Spike)	6.58
Control Room Whole Body Dose	0.02
Control Room Skin Dose	0.57
Control Room Thyroid Dose (Accident-Initiated Spike)	1.90
Control Room Thyroid Dose (Pre-accident Iodine Spike)	7.96
Unit 2 DB SGTR with CRAVS Chlorine Detector Failure	
EAB Whole Body Dose	0.19
EAB Thyroid Dose (Accident-Initiated Iodine Spike)	11.1
EAB Thyroid Dose (Pre-accident Iodine Spike)	30.9
LPZ Whole Body Does	0.03
LPZ Thyroid Dose (Accident-Initiated Iodine Spike)	3.10
LPZ Thyroid Dose (Pre-accident Iodine Spike)	6.52
Control Room Whole Body Dose	0.03
Control Room Skin Dose	1.10
Control Room Thyroid Dose (Accident-Initiated Spike)	3.63

Control Room Thyroid Dose (Pre-accident Iodine Spike)	16.3

Unit 2 DB SGTR with Stuck Open PORV on the Ruptured S/G

EAB Whole Body Dose	0.22
EAB Thyroid Dose (Accident-Initiated Iodine Spike)	13.7
EAB Thyroid Dose (Pre-accident Iodine Spike)	38.3
LPZ Whole Body Dose	0.03
LPZ Thyroid Dose (Accident-Initiated Iodine Spike)	3.86
LPZ Thyroid Dose (Pre-accident Iodine Spike)	7.81
Control Room Whole Body Dose	0.02
Control Room Skin Dose	0.63
Control Room Thyroid Dose (Accident-Initiated Spike)	2.06
Control Room Thyroid Dose (Pre-accident Iodine Spike)	8.55

Unit 2 DB SGTR with ERPA / ERPD Failure

EAB Whole Body Dose	0.21
EAB Thyroid Dose (Accident-Initiated Iodine Spike)	12.1
EAB Thyroid Dose (Pre-accident Iodine Spike)	37.1
LPZ Whole Body Dose	0.03
LPZ Thyroid Dose (Accident-Initiated Iodine Spike)	3.55
LPZ Thyroid Dose (Pre-accident Iodine Spike)	7.58
Control Room Whole Body Dose	0.02
Control Room Skin Dose	0.61
Control Room Thyroid Dose (Accident-Initiated Spike)	1.96
Control Room Thyroid Dose (Pre-accident Iodine Spike)	8.48

For each design basis Steam Generator Tube Rupture sequence analyzed, the radiological consequences are within the appropriate guideline values.

7) REFERENCES

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24) Volume 10 of the Code of Federal Regulations, Part 50.

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

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

27) K. G. Murphy and K. M. Campe, "Nuclear Power Plant Control Room Ventilation System Design for Meeting General Design Criterion 19," 13th AEC Air Cleaning Conference.

28) James P. Adams and Corwin L. Atwood, "The Iodine Spike Release Rate During a Steam Generator Tube Rupture," Nuclear Technology, Vol. 94, June 1991.

29) Draft Regulatory Guide DG-1081, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors".

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conversion Factors for Inhalation Submersion and Ingestion".

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33) Bechtel Corporation, LOCADOSE, NE-319 Theoretical Manual, Rev. 6, October, 1998. (Proprietary to Bechtel Corporation.)

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ATTACHMENT 4

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 is affirmed in the evaluation of the Steam Generator Tube Rupture sequences to be removed from the license basis and the Steam Generator Tube Rupture sequences to be retained in the license basis.

No changes to facility structures, systems, or components or in the operation of the plant are directly associated with the changes requested in this license amendment request. The amendment proposes only the removal from the license basis of the plant a number of single failures following a Steam Generator Tube Rupture. Of the failures in the Steam Generator Tube Rupture sequences proposed for removal from the license basis, none are "accident initiators" although failure of Distribution Center EDE (EDF) could be considered a "transient initiator". The consequences of this failure are similar to the consequences of a unit trip. The anticipated frequency of failure of EDE / EDF is insignificant compared to the anticipated frequency of a unit trip. Therefore, operation of the plant in accordance with this amendment does not involve a significant increase in the probability of an accident previously evaluated.

Balance among prevention of core damage, prevention of containment failure, and mitigation of consequences is retained. In particular, a design basis evaluation considering the single failure criterion has shown that core damage can be prevented. Reliance on programmatic activities has been kept to a minimum and defense against human errors has been retained, as discussed above. System redundancy, independence, and diversity have been preserved. Conformance with applicable general design criteria has not been degraded. In summary, it has been shown above that defense in depth is preserved with the proposed amendment. Furthermore, it has been shown that safety margin has been retained. In particular, the structure, system or components associated with the single failures in the Steam Generator Tube Rupture sequences proposed for removal remain in conformance to the germane codes and standards.

The probability of overfill of the ruptured steam generator following a Steam Generator Tube Rupture has been shown to be extremely low. The risk of consequential failure of a Power Operated Relief Valve or Main Steam Safety Valve of the ruptured steam generator, with its ancillary effects on containment bypass and consequence mitigation is acceptably low. The risk-informed analysis has shown that probability of core damage or large early releases is acceptably low. Therefore, removal of certain Steam Generator Tube Rupture sequences from the plant license basis proposed above does not constitute a significant increase in risk of the occurrence of a Steam Generator Tube Rupture at the plant.

Radiological consequences have been analyzed for design basis Steam Generator Tube Rupture scenarios with the limiting of the single failures to be retained in the license basis. All radiation doses are calculated to be less than the appropriate guideline values, as listed above. Radiation doses to the Exclusion Area Boundary and Low Population Zone are less than the values reported in earlier updates of the Updated Final Safety Analysis Report and other licensing documents. For this reason and those given above, the changes proposed in this License Amendment Request constitute no significant increase in consequences of an accident previously evaluated. 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. No changes to either any structure, system or component or to any procedure are associated with the changes proposed in this License Amendment Request. Therefore, no new failure modes are created with these changes. Therefore, neither any new accident nor an accident different from any previously evaluated is associated with the changes proposed in this License Amendment Request.

Does operation of the facility in accordance with the proposed amendment involve a significant reduction in a margin of safety? No. This conclusion is reached in consideration of the separate Steam Generator Tube Rupture sequences to be removed from the license basis and the Steam Generator Tube Rupture scenarios to be retained in the license basis.

Sufficient level of defense in depth has been retained with the changes proposed in this License Amendment Request. Balance among prevention of core damage, prevention of containment failure, and mitigation of consequences is retained. Again, it has been shown in a "traditional" (design basis) evaluation that core damage does not occur following any of the Steam Generator Tube Rupture sequences proposed from removal from the license basis. An evaluation from a risk-informed perspective has shown that neither core damage nor large early release of radioactivity following any of the Steam Generator Tube Rupture sequences proposed for removal from the license basis is credible. Removal of these sequences from consideration does not degrade confidence in the containment as a fission product barrier, as shown in the risk-informed analysis. The safeguards provided for post accident protection of the core and containment remain capable of their protective functions given any of the single failures listed above. In particular, conformance of these systems with applicable general design criteria and codes and standards is not degraded. 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.

As noted above, radiological consequences of the design basis Steam Generator Tube Rupture sequences with the limiting of the single failures retained in the license basis are less than both values previously reported in licensing documents and the germane guideline values. The analyses of radiological consequences of design basis Steam Generator Tube Rupture incorporated a number of new features.

Control room atmospheric dispersion factors for releases from the unit vent stack and steam generator Doghouses have been derived with appropriately conservative input and assumptions. The use of these control room factors reflects the effects of single failures following design basis events. Conversion factors for thyroid radiation doses are based on data from Federal Guidance Report No. 11. The basis for revision to the definition of the definition of "Dose Equivalent Iodine" is seen to be adequate. Dose conversion factors for whole body and skin doses, derived from the information of Federal Guidance Report No. 12 incorporate an acceptable level of conservatism. The new factor for the accident-initiated iodine spike has been cited by the staff and validated within Duke Energy Corporation. Its use for the design basis Steam Generator Tube Rupture and "Break of a Small Line Carrying Reactor Coolant Outside Containment" has been justified. Credit taken for letdown cleanup prior to letdown isolation is justified, as explained above. Therefore, use of these new features is appropriate for the analyses of radiological consequences of the design basis Steam Generator Tube Rupture and other design basis events.

Based on the above summary, operation of the facility in accordance with the proposed amendment does not involve 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 5

ENVIRONMENTAL ASSESSMENT/IMPACT STATEMENT

ENVIRONMENTAL ASSESSMENT/IMPACT STATEMENT

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

Implementation of this amendment will have no adverse impact upon the Catawba units; neither will it contribute to any 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.

ATTACHMENT 6

Input Data For Calculation of Control Room Atmospheric Dispersion Factor with ARCON96

Note: Meteorological data used in ARCON96 is being provided in electronic form.

APPPENDIX B ARCON96 OUTPUT FILES

B.1 CASE 49U1VNT1 OUTPUT FILES

```
Program Title:  ARCON96.

Developed For:  U.S. Nuclear Regulatory Commission
                Office of Nuclear Reactor Regulation
                Division of Reactor Program Management

Date:           June 25, 1997    11:00 a.m.

NRC Contacts:   J. Y. Lee       Phone: (301) 415 1080
                                e-mail: jyl1@nrc.gov
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                                e-mail: lab2@nrc.gov

Code Developer: J. V. Ramsdell  Phone: (509) 372 6316
                                e-mail: j_ramsdell@pnl.gov

Code Documentation:  NUREG/CR-6331 Rev. 1

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liability or responsibilities for any third party's use, or the results of such
use, of any portion of this program or represents that its use by such third
party would not infringe privately owned rights.

Program Run   9/23/1999  at  15:33:38

 ******* ARCON INPUT **********

    Number of Meteorological Data Files  =  3
    Meteorological Data File Names
      C:\ARCON96\CNS94MET.MET
      C:\ARCON96\CNS95MET.MET
      C:\ARCON96\CNS40MET.MET

    Height of lower wind instrument (m)  =     10.0
    Height of upper wind instrument (m)  =     40.0
    Wind speeds entered as miles per hour

    Ground-level release
```

```
Release height (m)                      =     38.2
Building Area (m^2)                     =   1571.0
Effluent vertical velocity (m/s)        =      .00
Vent or stack flow  (m^3/s)             =     2.80
Vent or stack radius (m)                =      .00

Direction .. intake to source (deg)     =      045
Wind direction sector width (deg)       =       90
Wind direction window (deg)             =   000 - 090
Distance to intake (m)                  =     45.2
Intake height  (m)                      =      1.4
Terrain elevation difference (m)        =       .0

Output file names
  49u1vnt1.log
  49u1vnt1.cfd

Minimum Wind Speed (m/s)                =       .5
Surface roughness length (m)            =      .10
Sector averaging constant               =      4.0

Initial value of sigma y                =      .00
Initial value of sigma z                =      .00

Expanded output for code testing not selected

Total number of hours of data processed =  21426
Hours of missing data                   =    356
Hours direction in window               =   5681
Hours elevated plume w/ dir. in window  =      0
Hours of calm winds                     =    108
Hours direction not in window or calm   =  15281
```

DISTRIBUTION SUMMARY DATA BY AVERAGING INTERVAL

AVER. PER.	1	2	4	8	10	24	96	168	360	720
UPPER LIM.	1.00E-02	1.00E-02	1.00E-02	1.00E-02	1.00E-02	1.00E-02	1.00E-02	1.00E-02	1.00E-02	1.00E-02
LOW LIM.	1.00E-06	1.00E-06	1.00E-06	1.00E-06	1.00E-06	1.00E-06	1.00E-06	1.00E-06	1.00E-06	1.00E-06
ABOVE RANGE	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
IN RANGE	5789.	6744.	8001.	9742.	10712.	14235.	19719.	20264.	20708.	20414.
BELOW RANGE	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
ZERO	15281.	14247.	12832.	10779.	10247.	6598.	974.	205.	0.	0.
TOTAL X/Qs	21070.	20991.	20833.	20521.	20959.	20833.	20693.	20469.	20708.	20414.
% NON ZERO	27.48	32.13	38.41	47.47	51.11	68.33	95.29	99.00	100.00	100.00

95th PERCENTILE X/Q VALUES

1.77E-03	1.71E-03	1.62E-03	1.51E-03	1.37E-03	9.67E-04	6.05E-04	5.12E-04	4.51E-04	4.23E-04

B.3 49U2VNT2 OUTPUT FILES

```
Program Title:  ARCON96.

Developed For:  U.S. Nuclear Regulatory Commission
                Office of Nuclear Reactor Regulation
                Division of Reactor Program Management

Date:           June 25, 1997    11:00 a.m.

NRC Contacts:   J. Y. Lee       Phone: (301) 415 1080
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                                e-mail: lab2@nrc.gov

Code Developer: J. V. Ramsdell  Phone: (509) 372 6316
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Code Documentation:  NUREG/CR-6331 Rev. 1

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use, of any portion of this program or represents that its use by such third
party would not infringe privately owned rights.

Program Run   9/23/1999  at  15:36:43

 ******* ARCON INPUT **********

    Number of Meteorological Data Files  =  3
    Meteorological Data File Names
      C:\ARCON96\CNS94MET.MET
      C:\ARCON96\CNS95MET.MET
      C:\ARCON96\CNS40MET.MET

    Height of lower wind instrument (m)  =     10.0
    Height of upper wind instrument (m)  =     40.0
    Wind speeds entered as miles per hour

    Ground-level release
    Release height (m)                   =     38.2
```

```
Building Area (m^2)                     =   1571.0
Effluent vertical velocity (m/s)        =       .00
Vent or stack flow  (m^3/s)             =      2.80
Vent or stack radius (m)                =       .00

Direction .. intake to source (deg)     =      135
Wind direction sector width (deg)       =       90
Wind direction window (deg)             =   090 - 180
Distance to intake (m)                  =     45.2
Intake height  (m)                      =      1.4
Terrain elevation difference (m)        =       .0

Output file names
  49u2vnt2.log
  49u2vnt2.cfd

Minimum Wind Speed (m/s)                =       .5
Surface roughness length (m)            =       .10
Sector averaging constant               =      4.0

Initial value of sigma y                =       .00
Initial value of sigma z                =       .00

Expanded output for code testing not selected

Total number of hours of data processed =  21426
Hours of missing data                   =    356
Hours direction in window               =   3403
Hours elevated plume w/ dir. in window  =      0
Hours of calm winds                     =    108
Hours direction not in window or calm   =  17559
```

DISTRIBUTION SUMMARY DATA BY AVERAGING INTERVAL

AVER. PER.	1	2	4	8	10	24	96	168	360	720
UPPER LIM.	1.00E-02	1.00E-02	1.00E-02	1.00E-02	1.00E-02	1.00E-02	1.00E-02	1.00E-02	1.00E-02	1.00E-02
LOW LIM.	1.00E-06	1.00E-06	1.00E-06	1.00E-06	1.00E-06	1.00E-06	1.00E-06	1.00E-06	1.00E-06	1.00E-06
ABOVE RANGE	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
IN RANGE	3511.	4415.	5744.	7631.	8581.	12562.	19283.	20309.	20708.	20414.
BELOW RANGE	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
ZERO	17559.	16576.	15089.	12890.	12378.	8271.	1410.	160.	0.	0.
TOTAL X/Qs	21070.	20991.	20833.	20521.	20959.	20833.	20693.	20469.	20708.	20414.
% NON ZERO	16.66	21.03	27.57	37.19	40.94	60.30	93.19	99.22	100.00	100.00

95th PERCENTILE X/Q VALUES

1.70E-03	1.59E-03	1.40E-03	1.22E-03	1.08E-03	6.79E-04	3.75E-04	3.11E-04	2.71E-04	2.26E-04

95% X/Q for standard averaging intervals

C.3 49U2AFW2 OUTPUT FILES

```
Program Title:  ARCON96.

Developed For:  U.S. Nuclear Regulatory Commission
                Office of Nuclear Reactor Regulation
                Division of Reactor Program Management

Date:           June 25, 1997    11:00 a.m.

NRC Contacts:   J. Y. Lee       Phone: (301) 415 1080
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Code Developer: J. V. Ramsdell  Phone: (509) 372 6316
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Code Documentation:  NUREG/CR-6331 Rev. 1

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party would not infringe privately owned rights.

Program Run   9/23/1999  at  15:31:40

 ******* ARCON INPUT **********

    Number of Meteorological Data Files  =  3
    Meteorological Data File Names
      C:\ARCON96\CNS94MET.MET
      C:\ARCON96\CNS95MET.MET
      C:\ARCON96\CNS40MET.MET

    Height of lower wind instrument (m)  =     10.0
    Height of upper wind instrument (m)  =     40.0
    Wind speeds entered as miles per hour

    Ground-level release
    Release height (m)                   =     16.8
```

```
Building Area (m^2)                   =   1571.0
Effluent vertical velocity (m/s)      =      .00
Vent or stack flow  (m^3/s)           =    11.00
Vent or stack radius (m)              =      .00

Direction .. intake to source (deg)   =     169
Wind direction sector width (deg)     =      90
Wind direction window (deg)           =   124 - 214
Distance to intake (m)                =     38.2
Intake height  (m)                    =      1.4
Terrain elevation difference (m)      =       .0

Output file names
  49u2afw2.log
  49u2afw2.cfd

Minimum Wind Speed (m/s)              =       .5
Surface roughness length (m)          =      .10
Sector averaging constant             =      4.0

Initial value of sigma y              =      .00
Initial value of sigma z              =      .00

Expanded output for code testing not selected

Total number of hours of data processed =  21426
Hours of missing data                   =    356
Hours direction in window               =   7152
Hours elevated plume w/ dir. in window  =      0
Hours of calm winds                     =     87
Hours direction not in window or calm   =  13831
```

DISTRIBUTION SUMMARY DATA BY AVERAGING INTERVAL

AVER. PER.	1	2	4	8	10	24	96	168	360	720
UPPER LIM.	1.00E-02	1.00E-02	1.00E-02	1.00E-02	1.00E-02	1.00E-02	1.00E-02	1.00E-02	1.00E-02	1.00E-02
LOW LIM.	1.00E-06	1.00E-06	1.00E-06	1.00E-06	1.00E-06	1.00E-06	1.00E-06	1.00E-06	1.00E-06	1.00E-06
ABOVE RANGE	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
IN RANGE	7239.	8584.	10182.	11945.	12885.	16053.	20010.	20392.	20708.	20414.
BELOW RANGE	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
ZERO	13831.	12407.	10651.	8576.	8074.	4780.	683.	77.	0.	0.
TOTAL X/Qs	21070.	20991.	20833.	20521.	20959.	20833.	20693.	20469.	20708.	20414.
% NON ZERO	34.36	40.89	48.87	58.21	61.48	77.06	96.70	99.62	100.00	100.00

95th PERCENTILE X/Q VALUES

3.37E-03	3.27E-03	3.16E-03	2.92E-03	2.62E-03	1.76E-03	1.16E-03	1.06E-03	9.03E-04	8.52E-04

95% X/Q for standard averaging intervals

APPENDIX C AFW TURBINE EXHAUST OUTPUT FILES

C.1 49U1AFW1 Output Files

```
Program Title:  ARCON96.

Developed For:  U.S. Nuclear Regulatory Commission
                Office of Nuclear Reactor Regulation
                Division of Reactor Program Management

Date:           June 25, 1997    11:00 a.m.

NRC Contacts:   J. Y. Lee       Phone: (301) 415 1080
                                e-mail: jyl1@nrc.gov
                J. J. Hayes     Phone: (301) 415 3167
                                e-mail: jjh@nrc.gov
                L. A  Brown     Phone: (301) 415 1232
                                e-mail: lab2@nrc.gov

Code Developer: J. V. Ramsdell  Phone: (509) 372 6316
                                e-mail: j_ramsdell@pnl.gov

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party would not infringe privately owned rights.

Program Run   9/23/1999  at  15:28:58

 ******* ARCON INPUT **********

    Number of Meteorological Data Files  =  3
    Meteorological Data File Names
      C:\ARCON96\CNS94MET.MET
      C:\ARCON96\CNS95MET.MET
      C:\ARCON96\CNS40MET.MET

    Height of lower wind instrument (m)  =     10.0
    Height of upper wind instrument (m)  =     40.0
```

```
Wind speeds entered as miles per hour

Ground-level release
Release height (m)                  =     16.8
Building Area (m^2)                 =   1571.0
Effluent vertical velocity (m/s)    =       .00
Vent or stack flow  (m^3/s)         =     11.00
Vent or stack radius (m)            =       .00

Direction .. intake to source (deg) =     011
Wind direction sector width (deg)   =      90
Wind direction window (deg)         =   326 - 056
Distance to intake (m)              =     38.2
Intake height  (m)                  =      1.4
Terrain elevation difference (m)    =       .0

Output file names
  49u1afw1.log
  49u1afw1.cfd

Minimum Wind Speed (m/s)            =       .5
Surface roughness length (m)        =       .10
Sector averaging constant           =      4.0

Initial value of sigma y            =       .00
Initial value of sigma z            =       .00

Expanded output for code testing not selected

Total number of hours of data processed =  21426
Hours of missing data                   =    356
Hours direction in window               =   7053
Hours elevated plume w/ dir. in window  =      0
Hours of calm winds                     =     87
Hours direction not in window or calm   =  13930
```

DISTRIBUTION SUMMARY DATA BY AVERAGING INTERVAL

AVER. PER.	1	2	4	8	10	24	96	168	360	720
UPPER LIM.	1.00E-02	1.00E-02	1.00E-02	1.00E-02	1.00E-02	1.00E-02	1.00E-02	1.00E-02	1.00E-02	1.00E-02
LOW LIM.	1.00E-06	1.00E-06	1.00E-06	1.00E-06	1.00E-06	1.00E-06	1.00E-06	1.00E-06	1.00E-06	1.00E-06
ABOVE RANGE	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
IN RANGE	7140.	8205.	9630.	11513.	12561.	16067.	20295.	20407.	20708.	20414.
BELOW RANGE	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
ZERO	13930.	12786.	11203.	9008.	8398.	4766.	398.	62.	0.	0.
TOTAL X/Qs	21070.	20991.	20833.	20521.	20959.	20833.	20693.	20469.	20708.	20414.
% NON ZERO	33.89	39.09	46.22	56.10	59.93	77.12	98.08	99.70	100.00	100.00

ATTACHMENT 7

List of Inputs for the Analysis of Radiological Consequences of A Design Basis Steam Generator Tube Rupture At Catawba Nuclear Station

Table 1) Data Pertaining to Source Radioactivity

Radioisotope half lives and decay constants	Table of the Isotopes, 8th Edition.
Whole body radiation dose conversion factors	Federal Guidance Report No. 12. "Effective" dose coefficients were used.
Thyroid radiation dose conversion factors	Federal Guidance Report No. 11.
Skin radiation dose conversion factors	Federal Guidance Report No. 12. The separate constituents of the dose conversion factors were obtained from Oak Ridge National Laboratory. The constituents associated with gamma radiation were adjusted for Control Room geometry effects.
Gross gamma specific activity in the Reactor Coolant System (RCS)	100/E-bar uCi/gm (TS 3.4.16, SR 3.4.16.1).
Equilibrium Dose Equivalent I-131 (DEI) specific activity in the RCS	1 uCi/gm (TS 3.4.16, Condition A).
Transient DEI specific activity in the RCS	60 uCi/gm (TS 3.4.16, Action A.1, Figure 3.4.16-1).
Equilibrium DEI specific activity in the S/G secondary coolant	0.1 uCi/gm TS 3.7.17, LCO 3.7.17).
Equilibrium DEI specific activity in the condenser hotwell	0.085 uCi/gm. It is assumed that 85% of the iodine activity in the steam flow is scrubbed within the condenser. The remainder is assumed to be released through the Condensate Steam Air Ejectors.
Iodine spectrum	UFSAR Table 15-13.
Mass of water in the RCS	539,793 lbm (Unit 1), 481,637 lbm (Unit 2).
Mass of water in each steam generator (S/G)	100,000 lbm (Unit 1), 60,000 lbm (Unit 2).
Mass of water in the condenser hotwell	1,190,400 lbm (each unit).
The following information is used to compute the production rate for the accident initiated iodine spike.	
Rate of leakage from the RCS	11 GPM (TS 3.4.13, LCO's 3.4.13.b and 3.4.13.c
Letdown flow rate	125 GPM.
Reference density	62.4 lbm/ft^3.
Efficiency of the letdown demineralizers	100%.
Multiplier for the accident initiated iodine spike	335.

Table 1, (continued)

The following information is used in association with credit for letdown cleanup prior to letdown isolation.	
Letdown flow rate	75 GPM.
Reference density	62.4 lbm/ft^3.
Efficiency of the letdown demineralizers	95%.

Table 2) Data Pertaining to Unit 1 DB SGTR Break Flow Before Reactor Trip.

Time span after DB SGTR (min)	Break flow rate (lbm/min)	flash fraction
0 - 5	2667.7	0.17616
5 - 10	2598.4	0.17685
10 - 15	2530.5	0.17862
15 - 20	2468.0	0.18049

Data concerning Unit 1 DB SGTR break flow and steam releases from the ruptured S/G and the intact S/G's after unit trip are presented on Tables 8 - 11.

Table 3) Data Pertaining to Unit 2 DB SGTR Break Flow Before Reactor Trip.

Time span after DB SGTR (min)	Break flow rate (lbm/min)	Flash fraction
0 - 4.16667	3421.3	0.15789
4.16667 - 8.33333	3269.5	0.15906
8.33333 - 12.5	3141.7	0.16029
12.5 - 16.76	3016.4	0.16146

Data concerning Unit 2 DB SGTR break flow and steam releases from the ruptured S/G and the intact S/G's after unit trip are presented on Tables 8 and 12 - 14.

Table 4) Additional Data Pertaining to Transport Radioactivity and Release of Radioactivity to the Environment

Noble gas "flash fraction"	1. All noble gases entrained in the DB SGTR break flow are assumed to be released to the environment.
Fraction of iodine partitioning with steam flow from the S/G's	0.01
Steam flow rate prior to reactor trip	66179 lbm/min per S/G.
Fraction of iodine scrubbed from steam flow in the condenser	0.85
Main feedwater flow rate prior to reactor trip	66179 lbm/min per S/G.
Time after the initiation of the DB SGTR to reactor trip	20 min (Unit 1), 16.76 min (Unit 2). Trip of Unit 2 on low pressurizer pressure was predicted in the RETRAN02 calculations.
Primary to secondary leak rate (PSLR) for each intact S/G	150 GPD.
Reference conditions for PSLR	The average temperature of the reactor coolant is 350 °F. At Catawba, the measured value of PSLR is adjusted for conditions associated with normal full power operation.
PSLR for all three S/G's	2.322 lbm/min.
Time assumed for actuation of the Auxiliary Feedwater System (AFWS)	Unit trip. Offsite power is assumed to be lost at this time.
Volume of condensate grade sources	225,000 gallons.
Maximum AFWS flow rate before DB SGTR break flow is terminated	1896 GPM (Unit 1), 1894 GPM (Unit 2).

Table 5) Offsite Atmospheric Dispersion Factors

Dispersion Factor at the Exclusion Area Boundary	
Time span (hr)	X/Q (sec/m^3)
0 - 2	4.78×10^{-4}
Dispersion Factor at the Boundary of the Low Population Zone	
Time span (hr)	X/Q (sec/m^3)
0 - 8	6.85×10^{-5}
8 - 24	4.00×10^{-5}
24 - 96	2.00×10^{-5}
96 - 720	7.35×10^{-6}

Table 6
Control Room Dispersion Factors

	X/Q (sec/m^3)	
Time Span (hr)	DB SGTR with CRAVS chlorine detector failure (Note 1)	DB SGTR with stuck open S/G PORV or ERPA / ERPD failure (Note 1)
0 - 0.3333333	1.51×10^{-3} (Note 2)	7.55×10^{-4} (Note 2)
0.3333333 - 8	3.12×10^{-3}	1.56×10^{-3}
8 - 10	1.70×10^{-3} (Note 3)	8.50×10^{-4}
10 - 24	7.27×10^{-4}	7.27×10^{-4}
24 - 96	5.95×10^{-4}	5.95×10^{-4}
96 - 720	4.78×10^{-4}	4.78×10^{-4}

Notes on Table 6

1) Catawba Nuclear Station has two control room outside air intakes. A false high signal from a Class 1E CRAVS chlorine detector may cause its associated outside air intake valve to close, isolating one of the outside air intakes. The operators are assumed to recover within 10 hours after the accident by opening the affected intake. The values for X/Q's for DB SGTR with CRAVS chlorine detector failure are calculated as follows: The assumption is made that the releases are directed to one intake while the air in the vicinity of the second intake is uncontaminated. The uncontaminated intake is assumed to close following the failure of one of its chlorine detectors. Only the contaminated intake is assumed to be open. After 10 hours, it is assumed that the operators open the uncontaminated outside air intake. The failure of a Class 1E CRAVS chlorine detector is the only valid failure mode causing the inadvertent closure of an outside air intake valve. Specifically, the other failures analyzed for effect on radiation doses following a DB SGTR do not affect the control room outside air intake valves. The values of control room X/Q's for these failures reflect this.

2) Between the initiation of the DB SGTR and unit trip, it is assumed that radioactivity is released to the environment from the Condensate Steam Air Ejectors to the Auxiliary Building and the unit vent stack. Offsite power is assumed to be lost at unit trip. Therefore, all radioactivity from the DB SGTR after unit trip is assumed to be released from the S/G doghouses. It is assumed that the operators trip the reactor at 20 minutes (0.333333 hours) after the initial DB SGTR at Unit 1. RETRAN-02 calculations project reactor trip on low pressurizer pressure at 16.76 minutes (0.279333 hours) after the initial DB SGTR at Unit 2.

3) As noted above, if a control room outside air intake valve is assumed to close (due to a false high signal from its Class 1E chlorine detector), it is assumed that the operators open it 10 hours after the initiating event. Cf. LER 413/91-08.

Table 7
Control Room Data

Rate of unfiltered inleakage to the control room (CFM)	30
Rate of CRAVS pressurized air flow (CFM)	4,000
Rate of CRAVS recirculation air flow (CFM)	1,400
Control room volume (ft^3)	117,920
Filter efficiency for removal of elemental iodine, %	99
Filter efficiency for removal of organic iodine, %	95
Filter efficiency for removal of particulate iodine, %	99