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September 2, 2008

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

Subject: Duke Energy Carolinas, LLC (Duke)
Catawba Nuclear Station, Units 1 and 2
Docket Numbers 50-413 and 50-414
Technical Specifications (TS) and/or Bases Sections:
3.3.2, Engineered Safety Feature Actuation System (ESFAS) Instrumentation
3.3.3, Post Accident Monitoring (PAM) Instrumentation
3.5.4, Refueling Water Storage Tank (RWST)
3.6.6, Containment Spray System
License Amendment Request for Emergency Core Cooling System (ECCS) Water Management Initiative

Reference: Letter from Dhiaa M. Jamil to NRC, ECCS Water Management Initiative, dated September 13, 2006

Pursuant to 10 CFR 50.90, Duke hereby requests a license amendment to revise the Unit 1 and Unit 2 TS and associated Bases to allow manual operation of the Containment Spray System and to revise the upper and lower limits on the RWST. Affected sections of the TS and/or Bases are:

1. Table 3.3.2-1 Function 2b and 2c to delete automatic actuation logic for containment spray
2. Table 3.3.2-1 Function 7b to lower the allowable value and the nominal trip setpoint for RWST Level - Low (this change also incorporates, on a limited basis, the footnotes contained in Technical Specification Task Force (TSTF)-493, Rev. 3, "Clarify Application of Setpoint Methodology for LSSS Functions" for this function only)
3. Bases 3.3.3 to revise the role of the containment sump level instrumentation
4. Surveillance Requirement (SR) 3.5.4.2 to raise the RWST volume requirement

A001
NRR

5. SR 3.6.6.4 to delete the automatic containment spray pump start verification

The objectives of this amendment request are to maximize the amount of water available for emergency core cooling from the RWST, to reduce the probability of transfer to containment sump recirculation, to increase operator response time before the transfer to containment sump recirculation conditions are satisfied, and to eliminate a Catawba non-conforming item. Additional benefits are to reduce the debris loading on the containment sump strainers as recommended in NRC Bulletin 2003-01 and to reduce the diesel generator automatic loading conditions.

Following implementation of these changes, significant improvements will be gained in plant safety based on the Catawba probabilistic risk assessment. It is estimated that the implementation of this initiative will result in approximately an 18% reduction in core damage frequency. This amendment request is based on the Nuclear Energy Institute (NEI) and the Pressurized Water Reactor (PWR) Owners Group initiative to extend the post-Loss of Coolant Accident (LOCA) injection phase and to delay the onset of the containment sump recirculation phase. Catawba is serving as the lead ice condenser plant for this initiative.

Duke analyzed the plant response resulting from the changes proposed in this amendment request. The safety and accident analysis concluded that the plant response remained within the current design and licensing limits. Attachment 1 provides the technical and regulatory evaluations of the proposed changes.

As part of this submittal, Duke is also requesting NRC approval of methodology report DPC-NE-3004-PA, Revision 2, "Mass and Energy Release and Containment Response Methodology".

Duke requests NRC approval of these proposed changes by August 31, 2009. Following NRC approval, Catawba will implement the associated modifications on a staggered basis for each unit. The Unit 1 modifications are currently scheduled to be implemented prior to the first entry into Mode 4 following the end-of-cycle refueling outage 18 (scheduled for Fall 2009). The Unit 2 modifications are currently scheduled to be implemented prior to the first entry into Mode 4 following the end-of-cycle refueling outage 17 (scheduled for Fall 2010).

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

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

Attachment 2 contains a marked-up version of the affected TS and Bases pages. Reprinted (clean) TS and Bases pages will be provided to the NRC prior to issuance of the approved amendment.

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

This amendment request contains NRC commitments as discussed in Attachment 3.

Attachment 4 contains the proposed changes associated with Duke methodology report DPC-NE-3004-PA, Revision 2 that will be implemented following NRC approval of this amendment request.

Pursuant to 10 CFR 170.11(a)(1)(iii), since this amendment request is being submitted on behalf of the industry as a lead plant submittal, Duke is requesting an exemption from licensing fees associated with the review and approval of this request. Consistent with the cited regulation, this amendment request represents a means of exchanging information between industry organizations and the NRC for the specific purpose of supporting the NRC's generic regulatory improvements or efforts.

If you have any questions or require additional information, please contact L.J. Rudy at (803) 831-3084.

Very truly yours,

James R. Morris

James R. Morris

LJR/s

Attachments

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

James R. Morris

James R. Morris, Vice President

Subscribed and sworn to me: 9/2/08
Date

Anthony P. Jackson

Notary Public

My commission expires: 7/2/2014
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Attachment 1

Evaluation of the Proposed Changes

Subject: Application for License Amendment for ECCS Water Management Initiative

1. SUMMARY DESCRIPTION

2. DETAILED DESCRIPTION

3. TECHNICAL EVALUATION

4. REGULATORY EVALUATION

5. ENVIRONMENTAL CONSIDERATION

6. REFERENCES

1. SUMMARY DESCRIPTION

This evaluation supports a request to amend Operating Licenses NFP-35 (Catawba Nuclear Station Unit 1) and NFP-52 (Catawba Nuclear Station Unit 2).

Catawba proposes to revise the TS to remove the automatic start signal for the Containment Spray System. The ability to manually operate the system when the pump suction is aligned to the containment sump will be maintained. Currently, TS 3.3.2 and TS 3.6.6 require automatic Containment Spray System operation to reduce containment pressure and temperature following a high energy line break inside containment. Reanalysis of the containment response, crediting the ice condenser safety systems and the Alternate Source Term (AST) methodology, has concluded that automatic containment spray operation is not required during the injection phase of accident mitigation and can be manually initiated later in the event once the ECCS has been realigned to the recirculation mode of operation.

In addition, Catawba proposes to lower the TS 3.3.2 allowable value and nominal trip setpoint for the RWST Level - Low function. This change is based on the reduced tank depletion rate following the removal of the automatic Containment Spray System operation and changes in the vortexing allowance based on testing and analytical refinements. Currently, the TS 3.3.2 setpoint is calculated based on automatic containment spray pump operation with alignment to the RWST.

Lastly, Catawba proposes to raise the TS 3.5.4 RWST minimum volume limit. This change is based on a plant modification to install new narrow range level instruments. Currently, TS 3.5.4 RWST level is based on the existing wide range level instruments.

Benefits following approval of these TS changes include:

- Significant improvement in plant safety through reduced core damage frequency (approximately 18% reduction) (the proposed changes have no meaningful impact on the estimated large early release frequency)
- Maximum available RWST inventory for ECCS coolant injection
- Reduction in the probability of transfer to sump recirculation
- Increase in allowable operator response times

- Resolution of a current Catawba non-conforming item that a subset of LOCA scenarios may utilize procedural guidance other than that described in the UFSAR to transfer to containment sump recirculation
- Consistency with the intent of NRC Bulletin 2003-01
- Reduction in containment sump debris and loading on the containment sump strainers
- Consistency with the NEI and the PWR Owners Group initiative to extend the post-LOCA injection phase and to delay the onset of the containment sump recirculation phase via status as lead ice condenser plant

The following plant hardware modifications are associated with the proposed TS changes:

- Deletion or disabling the containment spray automatic actuation circuitry
- Adjustment of the RWST low and low-low level alarm setpoints
- Installation of a new non-safety related dual channel narrow range RWST level instrument loop
- Installation of a new redundant non-safety related wide range RWST level annunciator alarm
- Changing the containment isolation signal for the non-safety related normal containment cooling units from containment high-high pressure to containment high pressure

In summary, following approval of the proposed TS changes, the plant response to a high energy line break is acceptable for the following considerations:

- Containment pressure and temperature structural limits
- Component environmental qualification for temperature and radiation
- Component submergence
- Containment sump pH
- Offsite and control room dose radiological consequences
- Pipe stress for sump temperature
- LOCA peak clad temperature

Duke requests that the NRC approve the proposed amendment based on the improvement in plant safety.

2. DETAILED DESCRIPTION

TS currently require automatic Containment Spray System actuation following a containment pressure high-high signal. Following the actuation signal, both trains of containment spray pumps start to transfer water from the RWST to the upper containment spray nozzles to reduce the containment pressure, temperature, and radioactive fission product airborne concentration.

The proposed change will remove the automatic start signal for the Containment Spray System. The Containment Spray System will continue to be operated manually after the Residual Heat Removal (RHR) pumps have completed the injection phase of the accident. After RHR pump suction is aligned to the containment sump, one containment spray pump may be manually started after adequate sump level is verified and if containment pressure remains greater than 3 psig. Specifically, the following two requirements are requested to be deleted from the TS. (Since the associated modifications will be implemented on a staggered basis for each unit during refueling outages, the deletion of these two requirements is being accomplished via the use of temporary footnotes. This will allow the requirements to be either applicable or non-applicable, depending upon whether the modifications have not been implemented or implemented, respectively.)

1) Table 3.3.2-1, ESFAS Instrumentation for Function 2, Containment Spray

   b. Automatic Actuation Logic and Actuation Relays

   c. Containment Pressure - High-High

2) SR 3.6.6.4 Verify each containment spray pump starts automatically on an actual or simulated actuation signal.

TS also currently require a minimum volume of RWST inventory to be available for accident mitigation. During an accident, the initial suction source for ECCS pumps and containment spray pumps is the RWST. Following a low level signal in the RWST, the suction of the RHR pumps will automatically transfer from the RWST to the containment sump. When the RWST water level reaches the low-low setpoint, operator action is required to manually realign the containment spray pump suction to the containment sump. The minimum volume in the RWST and the low level setpoint are currently defined in the TS.

The proposed amendment will raise the RWST minimum volume limit and lower the low level setpoint. Specifically, the two requirements requested to be changed are as follows. (The identical methodology of employing temporary footnotes is being utilized in conjunction with these two changes.)

1) Change:

SR 3.5.4.2 Verify RWST borated water volume is ≥ 363,513 gallons.

To state:

SR 3.5.4.2 Verify RWST borated water volume is ≥ 377,537 gallons.

2) Change Table 3.3.2-1, ESFAS Instrumentation for Function 7, Automatic Switchover to Containment Sump, b. Refueling Water Storage Tank (RWST) Level - Low:

Allowable value ≥ 162.4 inches and nominal trip setpoint 177.15 inches

To state:

Allowable value ≥ 91.9 inches and nominal trip setpoint 95 inches

(Note: In conjunction with this proposed change, Duke is also proposing to adopt the SR 3.3.2.7 and SR 3.3.2.9 footnotes contained in Technical Specification Task Force (TSTF)-493, Rev. 3, "Clarify Application of Setpoint Methodology for LSSS Functions" for this function only. Generic adoption of TSTF-493 for all other Functions contained in TS 3.3.1 and TS 3.3.2 is planned for a future license amendment request.)

The two proposed TS changes will increase the amount of water available for ECCS injection into the Reactor Coolant System (RCS). The minimum RWST volume increase is based on a plant modification to install two redundant narrow range level instruments. The new level instruments will reduce the amount of instrument uncertainty currently applied to the level measurement. The lower RWST level setpoint is based on removing the containment spray pump suction from the RWST, thereby reducing the tank drawdown rate by between approximately 3000 gpm (one train minimum spray) to approximately 8000 gpm (two trains maximum spray). The reduction in the RWST drawdown rate permits a longer response time to transfer ECCS pump suction to the containment sump. In addition, the combination of reduced flow and utilization of a plant specific RWST vortex formation correlation reduces the penalty applied in the determination of the low level setpoint. Containment spray operation from the RWST will be precluded by a combination of system alignment and procedural guidance. Procedural guidance will be provided to ensure that operator action is taken to manually start one containment spray pump only when aligned to the containment sump. In addition, the normal Containment Spray System alignment is such that no single failure will result in the depletion of RWST inventory by containment spray pump operation. Therefore, the RWST low level setpoint may be reduced accordingly.

The future TS 3.3.2 setpoint is calculated assuming the containment spray pumps do not deplete the available RWST volume. Containment spray operation from the RWST will be precluded by a combination of system alignment and procedural guidance. Procedural guidance will be provided to ensure that operator action is taken to manually start one containment spray pump only when aligned to the containment sump. In addition, the normal Containment Spray System alignment is such that no single failure will result in the depletion of RWST inventory by containment spray pump operation.

These changes are submitted to improve plant safety. One of the larger contributors to the overall plant risk is the sequence of plant operations to transfer ECCS suction from the RWST to the containment sump and to ensure the containment sump inventory is sufficient to provide long term core cooling and containment cooling. Following implementation of this amendment, the plant vulnerability to this evolution is reduced. If the conditions for transferring suction to the containment sump are met, the vulnerability associated with operating in the ECCS recirculation mode will be minimized due to (1) increased RWST inventory transferred into containment; (2) possible decreased flow through the sump strainers with less than two containment spray pumps running; (3) elimination of the upper containment holdup volume prior to initiation of sump recirculation; and (4) possible reduced debris loading on the containment sump strainers from containment washdown. Additionally, the operator response time is enhanced by (1) providing additional time before reaching swapover conditions; and (2) providing additional time to complete the manual swapover actions. The improvements in operator response times enhance the likelihood of successful control board switch manipulations.

The amendment request will also resolve a current operable but non-conforming condition concerning a narrowly defined postulated LOCA scenario. In 2004, Catawba identified a condition whereby a postulated break in either a bottom mounted instrumentation nozzle, a reactor vessel nozzle, or certain reactor vessel head locations may lead to the accumulation of water in the incore instrumentation room. The inventory of water lost from the containment sump could potentially result in containment sump level indicating less than required at the low level RWST setpoint. This could result in a condition where inadequate sump inventory exists to support the operation of the RHR pumps. As a result of this issue, Duke evaluated a spectrum of small break LOCA mass and energy releases. The evaluation indicated that there are certain small break LOCA scenarios for which alternate Emergency Procedure (EP) coping strategies had to be developed that are different than what is described in the UFSAR. This amendment request will resolve this operable but non-conforming condition.

The proposal to remove the automatic containment spray operation was discussed during a meeting with NEI and the NRC on May 11, 2006. Following the meeting, the NRC indicated an interest in receiving pilot plant applications to eliminate automatic containment spray initiation and to implement this ECCS Water Management Initiative. On September 13, 2006, Duke requested to be considered as the pilot plant for PWRs with ice condenser containments. On February 26, 2007, Duke presented the NRC with the scope of the license amendment request, preliminary plant analysis results, and requested NRC feedback on the proposed submittal. This amendment request is in response to these meetings and communications with the NRC.

Also included in this submittal are proposed changes to the TS Bases that reflect the proposed TS changes. The proposed TS Bases changes are shown in the form of marked-up versions of the affected TS Bases pages. This aids in identifying the areas of proposed change. Following NRC approval of this amendment request, for ease of operator use, two complete versions of each corresponding TS Bases section will be physically utilized. One version will be applicable to the existing plant configuration and one version will be applicable to the proposed plant configuration. This will reflect the fact that the proposed modifications will be implemented on a staggered basis for each unit during refueling outages.

3. TECHNICAL EVALUATION

Following a high energy line break, the containment pressure and temperature conditions are currently maintained within design limits by the Containment Spray System, Ice Condenser System, air return fans, and, if necessary, RHR auxiliary spray. Offsite dose is currently controlled by the containment design, containment isolation, containment spray, ice condenser, containment annulus ventilation system, and, if necessary, RHR auxiliary spray.

Methodology to evaluate containment response following a high energy line break, described in Duke Topical Report DPC-NE-3004-PA, has been approved by the NRC, with Safety Evaluations dated September 6, 1995 for Revision 0 and February 29, 2000 for Revision 1. Calculations of post-accident radiation dose using AST methodology was approved for Catawba on September 30, 2005 (License Amendments 227 and 222). Calculation of the RWST low level setpoint is based on current setpoint methodology.

The design basis functions for containment pressure control systems are discussed in UFSAR Section 6.2. An overview of the applicable plant systems is discussed below.

3.1 Systems

3.1.1 Containment Structure

The containment is used to limit the release of radioactivity to the environment.

The primary containment vessel is a free standing steel structure that encloses the RCS. Primary containment is further divided into upper and lower compartments such that any high energy line break flow within lower containment is routed through the ice condenser before entering upper containment. The free volume within the upper containment is approximately 670,000 cubic feet (nominal) and the free volume within the lower containment is approximately 200,000 cubic feet (nominal).

The secondary containment is a reinforced concrete structure that surrounds the primary containment vessel. The secondary containment creates an approximate six foot annulus region around the primary containment such that any primary containment leakage is filtered by the Annulus Ventilation System prior to release to the environment.

Containment leakage limits are specified within the TS.

During normal plant operations, TS define the upper and lower containment pressure and temperature limits such that containment conditions will remain below the design limits following a high energy line break. The containment pressure design limit is 15.0 psig and the containment design temperature is 120°F for normal conditions and 328°F for design basis accident conditions. The environmental qualification temperature for components within containment is 340°F.

3.1.2 Ice Condenser

The ice condenser is used to limit containment pressure, temperature, and radioactivity following a LOCA.

The ice condenser encompasses 300 degrees of the containment circumference. Within the ice condenser are 1944 ice baskets with each basket 48 feet long and 12 inches in diameter. There are 48 lower inlet doors separating lower containment from the ice condenser. Inlet doors are a passive design feature that will open when lower containment pressure increases following a steam leak. As steam is directed through the ice condenser, it is condensed to limit containment pressure. Condensed steam and melted ice are routed into lower containment by the ice condenser floor drains. Melted ice provides an inventory of borated water into the containment sump. Melted ice is also used to control the containment sump pH levels. Non-condensed steam will relieve into upper containment.

3.1.3 Air Return Fans

The air return fans route the air and steam within upper containment into lower containment for recirculation through the ice condenser for further reduction in containment pressure and temperature.

Two safety related fans will automatically start and circulate the air from upper containment into lower containment following a high-high containment pressure signal (nominally 3 psig) and an approximate 10 minute time delay. The control room operator may manually start one air return fan at 1 psig containment pressure as approved in the NRC Safety Evaluation Report for Catawba License Amendments 231 and 227, dated September 25, 2006.

3.1.4 Containment Pressure Control System (CPCS)

The CPCS functions to prevent a vacuum condition inside containment which would cause containment design negative pressure to be exceeded. The CPCS provides permissive, inhibit, and termination signals for the containment spray and air return fans based on containment pressure. The CPCS inhibits containment spray and air return operation until containment pressure is greater than a 0.9 psid nominal trip setpoint. When containment pressure is reduced below 0.35 psid nominal trip setpoint, the CPCS terminates containment spray and air return operation.

3.1.5 Containment Spray System

The Containment Spray System is currently credited with reducing containment pressure, temperature, and radioactivity following a LOCA and after containment pressure reaches the high-high containment pressure setpoint.

Following approval of this amendment request, the containment spray automatic start signal will be removed and spray operation will be manually controlled when pump suction is aligned to the containment sump. Credit will be taken for reducing containment pressure, temperature, and radioactivity by containment spray during the cold leg recirculation and hot leg recirculation phases of the accident.

The current Containment Spray System has two trains of safety related pumps, heat exchangers, upper containment spray header nozzles, and associated valves and piping. The system will automatically start following a high-high containment pressure of 3.0 psig nominal trip setpoint. Pump suction is initially aligned to the RWST and will transfer water into upper containment. Following a low-low RWST level, pump suction is manually transferred from the RWST to the containment sump.

3.1.6 RHR Auxiliary Spray

In the current plant EPs, a portion of the RHR system flow may be aligned to another, separate upper containment spray header as an independent method of providing additional spray flow. Auxiliary spray may be manually placed in service after RHR is aligned to the sump and a minimum of 50 minutes have elapsed since the plant shutdown.

The proposed change to the containment spray actuation logic will impact the use of RHR auxiliary spray. The plant EPs will be revised such that RHR auxiliary spray is manually aligned based upon reaching a containment pressure setpoint. For the containment response analyses provided in Section 3.2.1 of this amendment request, RHR auxiliary spray is not aligned because the containment pressure remains below the setpoint selected. The nominal setpoint selected in the analysis for aligning RHR auxiliary spray is equal to the containment design pressure. The proposed plant EP setpoint for manually aligning RHR auxiliary spray may be decreased in the future to accommodate plant changes or to provide additional peak containment pressure margin.

The containment response analyses presented in Section 3.2.1 of this amendment request demonstrate that RHR auxiliary spray is not required to obtain acceptable peak containment pressure results for the current plant configuration with the proposed changes. When instrument uncertainty is considered, operator action could be taken to align RHR auxiliary spray for design basis events. The consequences of taking action to align RHR auxiliary spray would not adversely impact either containment pressure or core cooling. While aligning RHR auxiliary spray would reduce the flow injected to the cold legs and the associated condensation rate in the RCS, the increase in the spray condensation rate in containment will more than compensate, resulting in a slightly reduced containment pressure. Secondly, should the alignment of RHR auxiliary spray occur prior to the intact loop seal refilling, this will reduce the chance of this phenomenon. Avoiding this phenomenon will reduce the peak containment pressure. Core cooling is assured for the current plant configuration by the analysis that assumed RHR auxiliary spray is manually aligned at 50 minutes. Since any potential alignment of RHR auxiliary spray with the proposed changes described in this amendment request will occur at a later point in time, this confirms that core cooling would be assured.

The proposed setpoint for aligning RHR auxiliary spray may be decreased in the future to accommodate plant changes or to provide additional peak containment pressure margin.

The ability to align auxiliary spray will also be retained for use with the functional restoration guidelines. This will maintain the RHR auxiliary spray system capability as a contingency for beyond design basis events, such as loss of both trains of normal containment spray.

3.1.7 Annulus Ventilation System

During a LOCA, the Annulus Ventilation System maintains a negative pressure within the annulus such that primary containment leakage will be into the annulus volume. The system reduces the concentration of airborne activity within the annulus and filters any air discharged from the annulus to the environment.

The system consists of two redundant trains with each train consisting of a fan, filter train, and associated dampers and duct work. The system automatically starts on a safety injection signal.

3.1.8 RWST

During accident conditions, the RWST currently provides a source of borated water to the ECCS and containment spray pumps. The RWST provides water for containment cooling and depressurization, core cooling, and is a source of negative reactivity for reactor shutdown. The TS minimum volume of the RWST is 363,513 gallons. By addition of redundant narrow range level instrumentation, the minimum RWST volume can be increased to 377,537 gallons.

The current low level setpoint actuates at a level corresponding to a remaining water volume of approximately 153,625 gallons. The proposed change to the low level setpoint represents a remaining water volume of approximately 79,600 gallons.

Currently, the RWST inventory between low and low-low level is used to swap containment spray and high and intermediate head injection pumps. Given the high containment spray flow rate, this swap must currently be performed expeditiously by the operators. After the proposed changes are implemented, only the high and intermediate head injection pumps will be operating from the RWST below the RWST low level setpoint, thus allowing for a slower, more controlled depletion. Given the lower flow rates, the low-low level can be reduced from its current value by 6%, or 22,625 gallons, thereby maximizing the available RWST inventory.

In summary, the current RWST inventory available for ECCS operation is 307,951 gallons (TS minimum volume of 363,513 gallons with a volume associated with low-low level of 55,562 gallons). Upon approval of this amendment request and implementation of the associated modifications, the RWST inventory available for ECCS operation will be 354,912 gallons.

3.1.9 Auxiliary Building Filtered Ventilation Exhaust System

The Auxiliary Building Filtered Ventilation Exhaust System filters air exhausted from all potentially contaminated areas of the auxiliary building, which includes the ECCS pump rooms and non-safety related portions of the auxiliary building. The system, in conjunction with other normally operating systems, also provides ventilation for these areas of the auxiliary building.

The safety related portion of the system consists of two redundant trains with each train principally consisting of a fan, filter train, and associated dampers and duct work. Following receipt of a safety injection signal, the system isolates non-safety related portions and exhausts air only from the ECCS pump rooms.

3.1.10 Control Room Area Ventilation System

The Control Room Area Ventilation System ensures that the control room will remain habitable for personnel during and following all credible accident conditions. This function is accomplished by pressurizing the control room to $\geq$ 1/8 inch water gauge with respect to all surrounding areas, filtering the outside air used for pressurization, and filtering a portion of the return air from the control room to clean up the control room environment.

The system consists of two independent, redundant trains of equipment with each train consisting of a pressurizing filter train fan, filter unit, and associated dampers and duct work. One train of the system is in continuous operation (either train can be selected). Upon receipt of an engineered safety features signal, the selected train continues to operate and the pressurizing filter train fan of the non-selected train is started. This assures control room pressurization, assuming an active failure of one of the pressurizing filter train fans.

3.2 Calculations and Analysis Results

3.2.1 Containment Analysis

There are three separate parameters evaluated in the containment analyses: 1) peak containment pressure, 2) maximum sump liquid temperature, and 3) maximum containment vapor temperature. The peak containment pressure and maximum sump liquid temperature result from large break LOCA analyses. The maximum containment vapor temperature results from a large steam line break. The evaluation of each parameter is discussed below.

Large Break LOCA Containment Response

A reanalysis of the containment response to a large break LOCA has been performed. The containment response is determined using the Duke ice condenser containment response methodology described in Topical Report DPC-NE-3004-PA. Catawba Unit 1 is equipped with feedring steam generators (FSGs). Catawba Unit 2 is equipped with Model D5 steam generators. Since the FSGs are limiting for containment response due to higher initial primary and secondary fluid mass, Catawba Unit 1 is selected for the reanalysis. This reanalysis includes the following changes:

- Increased initial RWST TS liquid inventory
- Decreased RWST low level alarm setpoint
- Decreased RWST low-low level alarm setpoint
- Elimination of the automatic containment spray actuation

The reanalysis also incorporates several revisions to the EPs. These revisions are summarized below:

- Operator action to transfer high head and intermediate head safety injection pumps to take suction from RHR pump discharge is taken upon receipt of the RWST low-low level alarm
- Following transfer of the RHR pump suction to the containment sump at the RWST low level alarm, one containment spray pump may be manually started after adequate sump level is verified and if containment pressure remains greater than 3 psig
- The operator does not align one RHR pump for auxiliary containment spray

The timing of when the operator will manually start one containment spray pump is dependent upon the single failure assumed. For most cases, the containment spray pump would be aligned prior to reaching RWST low-low level. When the single failure affects the valves that automatically swap during the RHR transfer to the containment sump, the containment spray pump would be aligned after reaching RWST low-low level.

Two criteria were examined in this reanalysis of the LOCA containment response, peak containment pressure and maximum sump liquid temperature. The results are compared against a containment design pressure of 15.0 psig and a UFSAR sump temperature limit of 190°F. The sump temperature limit is considered applicable when ECCS is aligned to take suction from the containment sump. Each of these criteria is discussed below.

Peak Containment Pressure

The peak containment pressure is obtained as a result of a large break LOCA located on a cold leg pump discharge pipe. This break location evolves to a boiling pot mode of core cooling, where the amount of liquid entering the core is equal to the steaming rate. The remainder of the injected ECCS spills from the broken cold leg. This provides a long term steam source that is condensed by the ice mass in the ice condenser and containment spray. The limiting single failure is the loss of one train of engineered safeguards. This minimizes the available containment spray flow and limits the number of heat exchangers available to remove heat from containment.

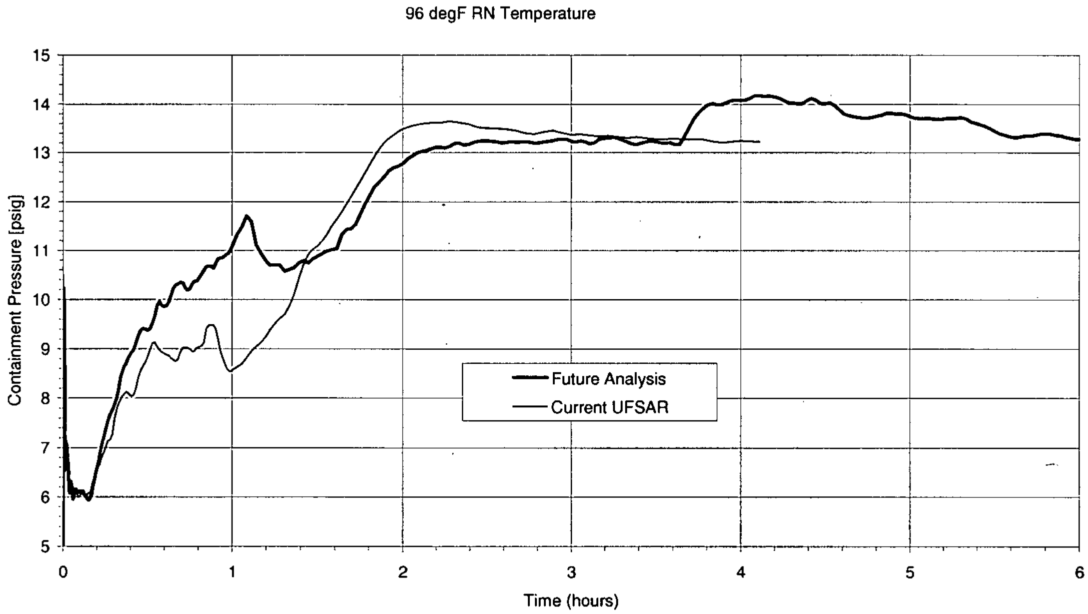
The peak containment pressure obtained for a cold leg pump discharge break with the changes described above is 14.18 psig, which is below the containment design pressure of 15.0 psig. This is an increase from the current peak pressure of 13.65 psig. This pressure increase is expected due to the decrease in total containment spray flow.

Two phenomena previously not included in Catawba peak containment pressure response analyses were observed. The first phenomenon, intact loop seal refilling, directly affects the peak containment pressure. The current containment analysis diverts flow from the RHR pump from the cold legs and core cooling, to be used as auxiliary containment spray. The reanalysis keeps this flow aligned to the cold legs. For a cold leg break, the core steaming rate decreases with decay heat. The majority of the steam generated in the core passes through the broken loop and is released to containment. A fraction of this steam is drawn through the intact loops by the condensation of steam by cooler ECCS injection. Eventually the steam velocity in the intact legs decreases to the point where liquid spills over the reactor coolant pump weirs and refills the intact loop seals. At this point in time, all of the steam generated in the core exits the broken loop into containment to be condensed. In Figure 1 shown below, this occurs just after 3.5 hours.

The second phenomenon observed is a significant increase in the amount of reverse break flow. This is steam predicted to be drawn from containment into the RCS by steam condensed on subcooled ECCS injection. The Duke containment response methodology is an iterative method that uses GOTHIC for the containment response calculation and RELAP5 for the mass and energy release calculation. Steam predicted to be drawn from containment into the RCS in the RELAP5 calculation is not removed from the GOTHIC calculation. This steam flow is non-conserved mass and energy. In the present containment response calculation, the predicted reverse break flow is a penalty incorporated in the results. For the reanalysis, the magnitude of this penalty is such that it was desirable to mitigate this flow. The ECCS steam condensation rate in the cold legs is mitigated by the inclusion of nitrogen in the RELAP5 containment boundary condition. This change reduced the magnitude of the reverse break flow penalty.

Figure 1

CNS LOCA M&E Release
RSG Cold Leg Pump Discharge - Min ECCS Case



Maximum Sump Liquid Temperature

The maximum sump liquid temperature is obtained as a result of a large break LOCA located on a hot leg pipe. This break location allows all of the injected ECCS to flow through the core. Relative to a cold leg break, a greater fraction of the break energy is deposited in the liquid phase for a hot leg break, resulting in a higher sump liquid temperature. For this evaluation, both minimum and maximum ECCS flow is considered.

The maximum sump liquid temperature results from a hot leg break with maximum ECCS flow. A maximum temperature of 198.7°F occurs at the time of transfer to the sump. The temperature remains above the current UFSAR described limit of 190°F for approximately 18.8 minutes. This result is due in part to the absence of containment spray flow mixing with the sump fluid. The impact of the increased maximum sump liquid temperature on NPSH is described in Section 3.2.2, and the impact on the piping analysis is described in Section 3.2.3. Acceptable results are obtained for the increased maximum sump liquid temperature.

Evaluation of Containment Spray Pump EP Change

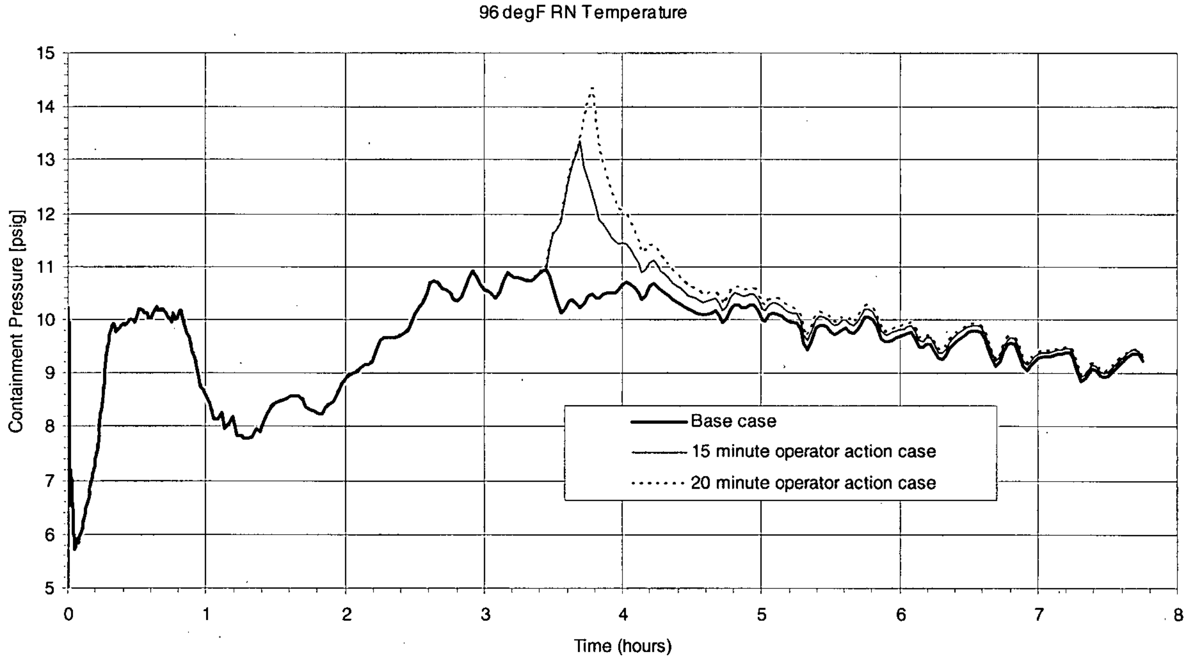
The proposed changes to the EPs include instructing the operator to start only one containment spray pump from the containment sump. This is a change from the current EP strategy that starts both containment spray pumps if they are available. The change creates a new potential single failure scenario, the failure of the operating containment spray pump.

If the operating containment spray pump were to fail upon demand, while the pump is being initially started by procedure, the operator would proceed to start the second containment spray pump. This action will occur within the expected operator action time to start spray flow. Thus, this case would not be limiting from a containment pressure response perspective, as two trains of ECCS flow would be available, and more importantly, three heat exchangers would be available for rejecting heat from containment.

A sensitivity study has been performed to evaluate the containment pressure response to a containment spray pump run failure. The pump is assumed to fail at the time of peak pressure for the event. The results of this sensitivity study, shown on Figure 2, indicate that 20 minutes are available for operator action to initiate containment spray flow from the idle spray pump. If containment spray flow is initiated within 20 minutes, the peak containment pressure will remain below the containment design pressure. Operator action is reasonably achievable within the 20 minute timeframe.

Figure 2

CNS LOCA M&E Release
RSG Cold Leg Pump Discharge - 2 ECCS trains



<u>Large Steam Line Break Containment Response</u>

The current steam line break analysis temperature results are well below the 340°F equipment qualification (EQ) limit in lower containment. This analysis demonstrates that the average lower containment vapor temperatures peak within the first 30 seconds and return to between 250°F and 240°F by 60 seconds. The current analysis does not allow containment spray flow to enter lower containment. The duration of the analysis is not sufficient to include the actuation of the containment air return fans. Thus, it can be concluded that the current steam line break analysis bounds the plant response with the proposed modification to remove the automatic containment spray actuation logic.

The proposed modifications to the RWST will not have an impact on the peak containment vapor temperature result, as the impact of these changes is to prolong the cold leg injection phase of an event. Therefore, the proposed modifications will not impact the current peak containment vapor temperature results.

DPC-NE-3004-PA Methodology Revision 2

The Duke ice condenser containment response methodology described in Topical Report DPC-NE-3004-PA Revision 1 is used to perform the analyses described above. This version of the report does not describe the modifications to the ECCS alignments incorporated in these analyses. Following approval of this amendment request, Topical Report DPC-NE-3004-PA will be revised to include the following information. The majority of this information is already described in this amendment request.

Revision 2 of the containment response methodology will describe the modeling changes required to perform mass and energy release calculations with the containment spray automatic actuation logic removed. These changes include:

- Removal of the containment spray automatic actuation logic precludes containment spray flow until operator action is taken
- Operator action to align safety injection and centrifugal charging pump suction to RHR pump discharge is delayed from RWST low level until RWST low-low level
- Operator action to align RHR pump to auxiliary containment spray header is changed from 50 minutes to be based upon a containment pressure setpoint plus a delay for operator action and the RHR pumps taking suction from the sump
- Revised modeling approach that includes nitrogen to mitigate the effects of reverse break steam flow
- Description of new phenomena

The information to be included in DPC-NE-3004-PA Revision 2 is provided in Attachment 4.

Duke is requesting review of DPC-NE-3004-PA Revision 2 to extend the ice condenser containment response methodology to include the changes described above.

3.2.2 NPSH Analysis

In current ECCS pump NPSH analyses, no credit has been taken for either containment pressure being above atmospheric conditions or for the static head of the sump inventory above the elevation of the containment sump screen structure. Therefore, there is not an impact on NPSH as a result of the proposed changes due to either increased sump inventory or to higher containment elevations.

The elevated sump temperatures discussed above in Section 3.2.1 reduces the available NPSH margin; however, sufficient NPSH remains available. Once containment sump temperatures are below previously analyzed temperatures, the existing NPSH analysis remains unchanged as a result of this amendment request.

3.2.3 Piping Analysis

Delaying containment spray actuation until after swapping to the containment sump can cause sump temperature to exceed 190°F (the maximum evaluated sump temperature is 198.7°F for approximately 18.8 minutes). The resultant increase in sump temperature has been evaluated for its effect on piping and other components, including the replacement ECCS sump strainers.

The affected piping is shown on flow diagrams for the Catawba RHR and safety injection systems. The current design pressure and temperature of the identified piping is 65 psia and 190°F.

The increase in design temperature affects only a small segment of piping from the ECCS sump inside containment to valves 1/2NI184B and 1/2NI185A in the auxiliary building. Downstream of the ECCS sump isolation valves, the design temperature of the piping is 400°F and this value has been used in the analysis. The piping configuration associated with the Train A ECCS sump includes approximately 20 feet from penetration M303 to valve 1/2NI185A. The piping configuration for the Train B ECCS sump includes approximately 20 feet from penetration M210 to valve 1/2NI184B. The small amount of piping inside containment upstream of each penetration was included in the analysis. The resultant temperature increase has no effect, since any increase in thermal expansion is unrestrained and thus produces no loads or stresses within the piping or the penetrations.

Since the increase in temperature is only 10°F (i.e., from 190°F to 200°F) and is limited to only approximately 20 feet of piping for each ECCS train, the impact on the existing qualification is determined to be negligible. Furthermore, this condition is associated with an ASME Service Level D (faulted) event. As such, it is not necessary (although it is typically Duke's practice) to include this increase in the qualification of the piping or the penetrations. It is necessary to consider the effects on support loads. Again, since the resultant temperature increase is minor and since the adjacent downstream piping has been analyzed for a design temperature of 400°F, it is determined that the increase in temperature for this limited scope of piping has no impact on the qualification of the piping, penetrations, supports, or any other components.

3.2.4 Equipment Qualification

As a result of the proposed modifications, the environmental accident profiles based on a LOCA scenario for areas located inside containment and the annulus were revised. The steam line break based environmental accident profiles remain bounding and were not revised. The environmental qualification program related electrical equipment located inside containment and the annulus was evaluated against the revised environmental profiles to ensure qualification is maintained under the proposed revised conditions.

The affected equipment located within the specified areas remains qualified for its respective applications, and there is no adverse impact on the existing qualifications with the proposed revised environmental conditions associated with the proposed modifications.

For determining the effect of the proposed modifications on post-accident air temperatures in the annulus, the computer code CANVENT was used. Only post-LOCA temperatures in the annulus were calculated. It was determined that the current steam line break containment analysis bounds the plant response with the proposed modifications (refer to the discussion in Section 3.2.1 under the heading Large Steam Line Break Containment Response). For this reason, post-steam line break temperatures in the annulus were not calculated with the proposed modifications assumed to be in place.

The method of the analysis was the same as that reported in Section 3.2.8 to determine Annulus Ventilation System operation for limiting post-LOCA radiation doses with two exceptions. First, the initial conditions in the annulus were calculated based on setting the outside air temperature to a high value instead of a low value. Specifically, the outside air temperature was set to a value bounding the 99th percentile outside air temperature in the Charlotte, North Carolina area. This is conservative compared to the NRC regulatory positions concerning the assumptions taken for dual containments in the analysis of radiological consequences of the LOCA (Ref. 11 Section 4.3). Second, the conservative assumption was made that the reactor building does not leak. The maximum post-LOCA annulus air temperature was found to be 171°F.

3.2.5 RWST Minimum Level Calculation

By installation of the proposed narrow range RWST level instrumentation, the RWST volume SRs can be greatly enhanced. Currently, the full span level instrumentation is used to satisfy SR 3.5.4.2 for minimum RWST volume. The proposed narrow range level instrument loop has an accuracy of 3.5 inches, compared to the existing wide range accuracy of approximately 14.5 inches. The volume difference associated with the improved accuracy, as well as revising the margin between RWST makeup and TS minimum level, will allow the RWST minimum level to be increased from 363,513 gallons to 377,537 gallons.

3.2.6 RWST Low Level Allowable Value and Nominal Trip Setpoint Analysis

Upon reaching the RWST low level setpoint, the suction source for the RHR pumps will automatically transfer to the containment sump. The high head and intermediate head safety injection pumps are transferred to the containment sump by manual operator action once the RWST low-low level setpoint is reached. The RWST low and low-low level setpoints are calculated such that the RHR pumps and the high head and intermediate head safety injection pumps will be aligned to the containment sump prior to reaching the RWST level at which air entrainment due to vortexing is predicted to occur.

The RWST low level setpoint provides a volume above the no air entrainment level to account for the maximum RWST outflow during switchover with the most limiting single failure, plus an allowance for instrument error. The most restrictive single failure is the failure of one of the RWST-to-RHR suction isolation valves to close, thus maximizing RWST outflow during switchover. To mitigate this failure, manual operator action is required to close the corresponding sump valve.

Removal of the automatic start of the containment spray pumps reduces the outflow from the RWST during switchover. This reduction in the RWST draindown rate permits a longer response time to transfer the ECCS pump suction to the containment sump, and it also decreases the RWST level needed to preclude air entrainment due to vortexing. Therefore, the RWST low level setpoint may be reduced to 95 inches with an allowable value of 91.9 inches.

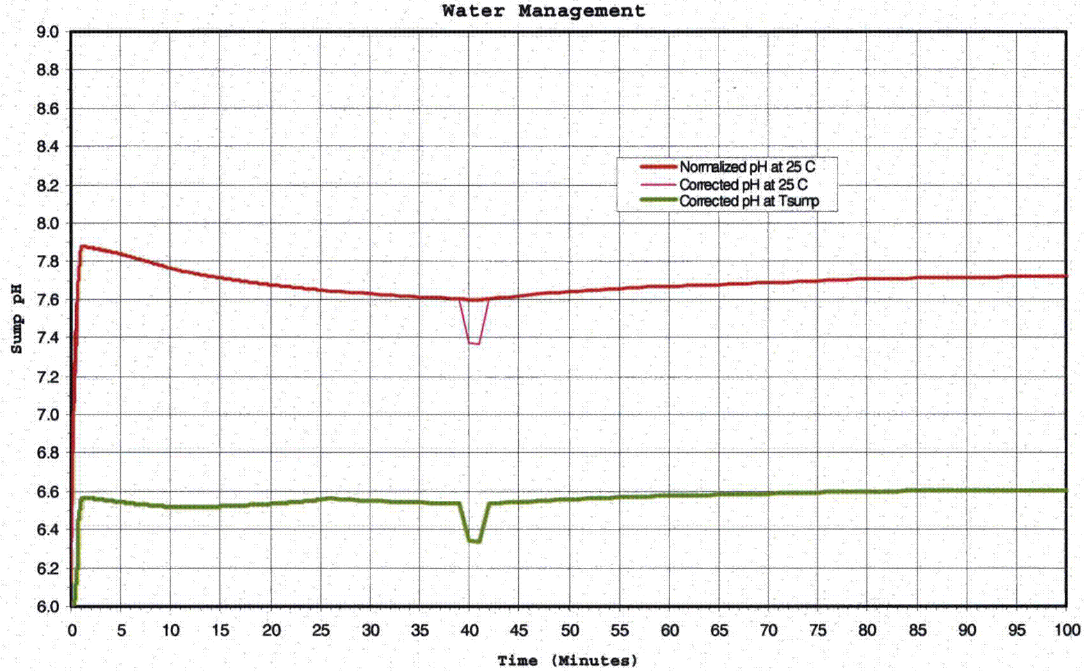
Section 6.3 of the Standard Review Plan also requires that the operator have at least 20 minutes to respond where manual actions are required following a LOCA. The current low level setpoint of 177.15 inches allows sufficient time for the operator to stop all ECCS pumps prior to reaching the RWST no air entrainment level, satisfying this requirement. Since the proposed modifications will reduce the RWST draindown rate and the RWST no air entrainment level, the available operator response time will be increased. Therefore, this criterion continues to be met.

3.2.7 Sump pH

Post-LOCA containment sump pH was calculated with the proposed modifications assumed to be in place. The method of this calculation was unchanged from the current licensing basis (Ref. 1, 2, 5). The water inventory of the RWST was set to its upper bound value adjusted only for the elevation of the outlet piping. In particular, no flow from the Containment Spray System was simulated. Flow from the ECCS was used to simulate the transport of boric acid solution from the RWST to the containment sump. The Containment Spray System will not take suction from the RWST with the proposed modifications. All other input to the calculation of post-LOCA containment sump pH remained unchanged from the current licensing basis (Ref. 1, 2).

The post-LOCA containment sump pH with the proposed modifications is shown in Figure 3. The short term containment sump pH increased somewhat with the proposed modifications. The results showed a very small decrease in the long term post-LOCA containment sump pH. The minimum transient containment sump pH was found to be 7.4 at the reference temperature of 77°F (25°C). The current minimum transient containment sump pH is 7.2 (Ref. 1, 2). The lower bound value for the equilibrium sump pH was found to remain essentially unchanged at 7.7 at 77°F (25°C). With the proposed modifications, the post-LOCA containment sump pH remains above 7 at 77°F (25°C) at all times. Containment sump pH also was calculated at the temperature of the water in the containment sump. This yielded a minimum transient sump pH of 6.3 (up from 6.2) and a lower bound equilibrium pH of 6.6 (not significantly changed from the current licensing basis value).

Figure 3
Catawba LBLOCA Sump pH Response
Water Management



3.2.8 Dose Analysis

The design basis LOCA is the only design basis accident at Catawba for which credit is taken for containment spray to mitigate post-accident radiation doses. Accordingly, the effects of the proposed modifications on post-accident radiation doses were determined with an analysis of the design basis LOCA in which the proposed modifications were simulated. In the current licensing basis analysis of the design basis LOCA (Ref. 1, 2, 4, 6), containment spray operation beginning at 10 minutes with start of the containment air return fans is simulated. The dose analysis also takes credit for auxiliary containment spray from the RHR pumps. Finally, for some design basis LOCA scenarios, credit has been taken for operation of both containment spray pumps. As part of the proposed modifications, the operators will start one containment spray pump only after the RHR system has been aligned to the containment sump for post-LOCA ECCS recirculation. The RHR system will not be credited in the dose analysis for auxiliary containment spray for design basis events. Radiation doses following the design basis LOCA at Catawba were recalculated to simulate these and other effects of the proposed modifications.

Catawba Unit 1 operated with four mixed oxide (MOX) lead fuel assemblies (LFAs) in its core, and these assemblies are currently in the spent fuel pool. Duke does not anticipate reinserting the MOX LFAs into Unit 1. For this reason, this license amendment request does not report a sensitivity study of the effects of the MOX LFAs. Should the proposed modifications be implemented on Unit 1 and the MOX LFAs are reinserted for another cycle of operation, Duke will complete a sensitivity analysis of the effects of Unit 1 operation with the proposed modifications and the MOX LFAs on post-LOCA radiation doses and appropriately update the Catawba UFSAR.

Revisions since the NRC Safety Evaluation of September 30, 2005

The original AST analysis of the LOCA, completed with the Bechtel computer code LOCADOSE (Ref. 7-9) was accepted by the NRC on September 30, 2005 (Ref. 12, cf. Ref. 1, 2, 4, 6). This analysis was revised to incorporate information learned since the original analysis. The revised analysis was used to establish a new baseline before performing the sensitivity study to support the proposed modifications. The revisions are identified as follows:

1) The time constants and decontamination factor (DF) cutoff times for washout of fission products with containment spray were revised. The revised values resolve problems found with some of the assumptions concerning auxiliary containment spray. The revisions also correct and compensate for a code error in the calculation of DF cutoff times. The revised containment spray time constants are presented in Tables 3 and 4 under the heading "Baseline".

2) The partitioning of containment leakage by source, currently by volumes of the lower and upper compartment, was changed to 60% from the lower compartment and 40% from the upper compartment. Originally, the containment leakage was partitioned in proportion to the volumes of the lower and upper compartments. As a complementary measure, containment leakage to the annulus was assumed to mix with half (50%) of the air in the annulus.

3) Increased iodine partition fractions for Engineered Safety Features (ESF) backleakage to the RWST were calculated with the initial water volume set to correspond to the RWST ECCS outlet elevation and vortex allowance. The assumed rate of ESF backleakage to the

RWST was set to 10 gpm (versus 20 gpm taken in the original analysis (Ref. 2)).

4) The calculation of post-LOCA radiation dose accounts for release of activity, transport of effluent to the Control Room Area Ventilation System outside air intakes, and buildup of activity in the control room. Other contributors to the post-LOCA control room radiation doses were assessed. In particular, the direct radiation dose to the control room from fission products outside the control room was calculated consistent with the regulatory positions for full scope implementation of AST methodology (Ref. 10, 11). This direct constituent to the radiation dose in the control room was determined to be 0.75 Rem. This value was added to the TEDE from post-LOCA transport of activity to the Control Room Area Ventilation System outside air intakes to yield the total post-LOCA TEDE in the control room.

The above noted changes yielded a moderate increase in the post-LOCA TEDE at the EAB. The impact of these changes together on the TEDEs at the LPZ is very small (less than 0.1 Rem).

The addition of the direct constituent from external source to the effluent constituent yielded an increase in the post-LOCA control room TEDE by 0.75 Rem. The other changes combined yielded a very small increase in the post-LOCA control room TEDE (less than 0.1 Rem).

The baseline TEDEs at the EAB and LPZ and in the control room for the limiting LOCA scenarios are presented below. The design basis LOCA with failure of cooling water flow through a heat exchanger of either the RHR system or the Containment Spray System was verified to be limiting for TEDEs at the offsite locations (Ref. 2). The design basis LOCA with an initially closed Control Room Area Ventilation System outside air intake was verified to be limiting for the control room TEDE (Ref. 2).

Table 1
Baseline Post-LOCA TEDEs

Type of TEDE	TEDE (Rem)
EAB TEDE	
Post-LOCA containment leakage	3.52
Post-LOCA ESF leakage	3.36
Total	6.88
LPZ TEDE	
Post-LOCA containment leakage	1.70
Post-LOCA ESF leakage	1.56
Total	3.26
Control Room TEDE	
Post-LOCA containment leakage	1.83
Post-LOCA ESF leakage	0.42
Direct radiation doses	0.75
Total	3.00

Dose Analysis for the Proposed Modifications

The proposed modifications were simulated with the following changes to the baseline analysis:

1) The time constants for containment spray washout of fission products were recalculated to account for the proposed modifications. These included delay of start of containment spray for 80 minutes, no simulation of RHR auxiliary spray, and start of only one containment spray pump regardless of the design basis LOCA scenario. The time constants and cutoff times for the baseline analysis and representative of the proposed modifications are compared in Tables 3 and 4.

2) Post-LOCA conditions in the annulus and response of the Annulus Ventilation System were recalculated with containment pressure and compartment temperatures associated with the proposed modifications. The time for the Annulus Ventilation System to draw the annulus pressure to -0.25 inch water gauge everywhere inside the annulus and the Annulus Ventilation System exhaust and recirculation airflow rates were used as inputs to the AST analysis of the design basis LOCA in support of the proposed modifications. The annulus drawdown times and Annulus Ventilation System exhaust and recirculation airflow rates for the design basis LOCA scenarios limiting for offsite and control room radiation doses and with the proposed modifications in place are presented in Table 5.

3) The iodine partition fractions for ESF leakage in the auxiliary building were recalculated based on changes in the post-LOCA containment sump pH with the proposed modifications (Section 3.2.7). In addition, the iodine partition fractions for ESF backleakage to the RWST were recalculated based on changes to the containment sump pH and RWST low-low level setpoint. A conservative approach was taken as follows: For each ESF leakage scenario and interval, the recalculated value was compared to the baseline values. The higher of the two values was taken for the iodine partition fraction. The limiting iodine partition fractions for post-LOCA ESF leakage are shown in Tables 6 and 7. The values associated with the proposed modifications are compared to the baseline values in these tables. The time to begin recirculation with the proposed modifications in place was set to 2160 sec (0.6 hr).

The TEDEs for the limiting design basis LOCA scenarios are listed below. The limiting scenarios remained unchanged with the proposed modifications assumed to be in place. The design basis LOCA with RHR system or Containment Spray System heat exchanger failure was limiting for offsite TEDEs, while the design basis LOCA with an initially closed Control Room Area Ventilation System outside air intake remained limiting for control room TEDE.

Table 2
Limiting Post-LOCA TEDEs
(Proposed Modifications vs Baseline)

Type of TEDE	TEDE (Rem) Baseline	TEDE (Rem) Proposed Modifications
EAB TEDE		
Post-LOCA containment leakage	3.52	5.43
Post-LOCA ESF leakage	3.36	3.36
Total	6.88	8.79
LPZ TEDE		
Post-LOCA containment leakage	1.70	2.11
Post-LOCA ESF leakage	1.56	1.67
Total	3.26	3.78
Control Room TEDE		
Post-LOCA containment leakage	1.83	2.14
Post-LOCA ESF leakage	0.42	0.42
Direct radiation doses	0.75	0.75
Total	3.00	3.31

The acceptance criteria for radiation doses following a design basis LOCA are 25 Rem at the EAB and LPZ and 5 Rem in the control room. The post-LOCA radiation doses for all design basis LOCA scenarios at Catawba remain below these criteria with implementation of the proposed modifications.

Table 3
Time Constants for Containment Spray Removal of Diatomic Iodine

Time Span (sec)		Time Constant (hr^{-1})			
		Baseline	Baseline	Proposed	
Start	End	Min Sfgds	Max Sfgds	Modifications	Notes
0	600	0.0	0.0	0.0	1
600	1,540	20.0	20.0	0.0	2
1,540	1,780	20.0	0.0	0.0	3
1,780	3,000	20.0	0.44	0.0	
3,000	3,240	0.18	0.65	0.0	3
3,240	4,800	0.41	0.65	0.0	
4,800	10,000	0.41	0.65	0.23	4
10,000	20,000	0.41	0.64	0.23	
20,000	25,000	0.41	0.63	0.23	
25,000	30,000	0.41	0.61	0.23	
30,000	40,000	0.40	0.52	0.23	
40,000	45,000	0.37	0.45	0.23	
45,000	50,000	0.37	0.0	0.23	5
50,000	60,000	0.29	0.0	0.23	
60,000	65,000	0.29	0.0	0.23	
65,000	70,000	0.0	0.0	0.23	6
70,000	80,000	0.0	0.0	0.22	
80,000	86,400	0.0	0.0	0.20	
86,400	2,592,000	0.0	0.0	0.0	7

Table 4
Time Constants for Containment Spray Removal of Particulate Fission Products

Time Span (sec)		Time Constant (hr^{-1})			
		Baseline	Baseline	Proposed	
Start	End	Min Sfgds	Max Sfgds	Modifications	Notes
0	600	0.0	0.0	0.0	8
600	1,540	9.51	19.00	0.0	2
1,540	1,780	9.51	0.0	0.0	3
1,780	3,000	9.51	19.00	0.0	
3,000	3,240	7.02	26.00	0.0	3
3,240	4,800	16.50	26.00	0.0	
4,800	6,600	16.50	26.00	9.51	4
6,600	7,000	16.50	2.60	9.51	9
7,000	7,800	1.65	2.60	9.51	10
7,800	86,400	1.65	2.60	0.95	11
86,400	2,592,000	0.0	0.0	0.0	7

Notes on Tables 3 and 4

1) The time constants for containment spray washout of organic iodine and noble gases are set to 0 each.

2) In calculating the spray washout time constants for the current plant configuration, it is assumed that the Containment Spray System does not become effective until 600 sec. It is assumed that the source term appears initially in the lower compartment and is not transported to the upper compartment until the containment air return fans start (simulated at 600 sec).

3) In the baseline analysis, it is assumed that the operators stop the containment spray pumps at 3,000 seconds for the design basis LOCA with Minimum Safeguards and at 1,540 seconds for all other design basis LOCA scenarios. Also, by assumption, for all design basis LOCA scenarios at 3,000 seconds, the operators start auxiliary containment spray from one RHR pump.

4) With the proposed modifications in place, it is assumed that the operators complete the alignment of the Containment Spray System to the containment sump and start only one containment spray pump for post-LOCA recirculation at 4800 sec (80 min). Finally, the operators do not align the RHR system to the auxiliary containment spray headers.

5) For the current plant configuration, the decontamination factor for spray removal of diatomic (elemental) iodine is calculated to reach 200 for the design basis LOCA with both containment spray pumps in operation.

6) For the current plant configuration, the decontamination factor for spray removal of diatomic iodine is calculated to reach 200 for the design basis LOCA with Minimum Safeguards (one containment spray pump in operation).

7) It is assumed that the operators stop containment spray at 1 day (86,400 sec) after event initiation.

8) The containment spray time constants for washout of particulates are applied to spray washout of particulate iodine and all other fission product groups other than iodine (except noble gases).

9) For the current plant configuration, the decontamination factor for spray removal of particulates is calculated to reach 50 for the design basis LOCA with both containment spray pumps in operation.

10) For the current plant configuration, the decontamination factor for spray removal of particulates is calculated to reach 50 for the design basis LOCA with Minimum Safeguards (one containment spray pump in operation).

11) With the proposed modifications in place, the decontamination factor for spray removal of particulates is calculated to reach 50 for all design basis LOCA scenarios.

Table 5
Post-LOCA Annulus Ventilation System Airflow Rates
(Design Basis LOCA with RHR System or Containment Spray System Heat Exchanger Failure and Design Basis LOCA with Initially Closed Control Room Area Ventilation System Outside Air Intake)

Time Interval (sec)		Annulus Ventilation System Airflow Rates (cfm)	
Start	End	Exhaust	Recirculation
0	23	0.0	0.0
23	30.5	16200.0	0.0
30.5	34	16200.0	0.0
34	35	3974.3	12225.7
35	45	5038.5	11161.5
45	60	5578.0	10622.0
60	75	6006.2	10193.8
75	90	6336.6	9863.4
90	105	6580.4	9619.6
105	120	6735.0	9465.0
120	135	6841.9	9358.1
135	150	6920.6	9279.4
150	180	7025.2	9174.8
180	210	7056.1	9143.9
210	300	7092.4	9107.6
300	360	6901.5	9298.5
360	400	6621.1	9578.9
400	500	6387.8	9812.2
500	600	5754.7	10445.3
600	700	5247.8	10952.2
700	800	4865.9	11334.1
800	900	4591.0	11609.0
900	1000	4396.6	11803.4
1000	1800	4261.8	11938.2
1800	3000	3542.8	12657.2
3000	7200	3307.4	12892.6
7200	9000	3274.9	12925.1
9000	12000	3166.9	13033.1
12000	18000	3345.9	12854.1
18000	28800	3189.1	13010.9
28800	54000	3190.1	13009.9
54000	2592000	3176.7	13023.3

Table 6
Limiting Iodine Partition Fractions for Post-LOCA ESF Leakage in the Auxiliary Building

Time Step End (hours)	Iodine Partition Fraction[1]		
	Baseline Values		
	In the ESF Pump Rooms	Outside the ESF Pump Rooms Offsite[2]	Outside the ESF Pump Rooms Control Room[3]
2.9	0.100	0.100	0.013
72.0	0.028	0.024	0.010
720.0	0.010	0.010	0.010
	Values with Proposed Modifications in Place		
	In the ESF Pump Rooms	Outside the ESF Pump Rooms Offsite[2]	Outside the ESF Pump Rooms Control Room[3]
2.9	0.100	0.100	0.013
72.0	0.031	0.027	0.010
720.0	0.010	0.010	0.010

Notes on Table 6

1) The iodine partition fractions for post-LOCA ESF leakage presented here correspond to the LOCA scenarios limiting for radiation doses at offsite locations and in the control room. Separate values are presented for leakage in and outside the ESF pump rooms. The filters of the Auxiliary Building Filtered Ventilation Exhaust System are aligned to the ESF pump rooms. They initially are not aligned to areas outside the pump rooms but are assumed to be aligned to these areas after 72 hours. See Ref. 2 for additional details.

   The values presented under the heading "In the ESF Pump Rooms" are associated with the LOCA scenarios limiting for radiation doses at offsite locations and in the control rooms.

2) The values presented immediately below are associated with the LOCA scenarios limiting for radiation doses at offsite locations (LOCA with failure of cooling water flow through a RHR or Containment Spray Heat Exchanger).

3) The values presented immediately below are associated with the LOCA scenarios limiting for radiation doses in the control room (LOCA with an initially closed control room outside air intake).

Table 7
Iodine Partition Fractions for Post-LOCA Backleakage to the RWST

Table 7a
Baseline Values

Time Span (Hours)	Iodine Partition Fraction: No Heat Exchanger Failure	Iodine Partition Fraction: Heat Exchanger Failure
0.0000 - 0.2198	0.000	0.000
0.2198 - 0.2250	6.482×10^{-10}	6.466×10^{-10}
0.2250 - 0.2500	1.919×10^{-8}	1.916×10^{-8}
0.2500 - 0.3333	1.885×10^{-7}	1.893×10^{-7}
0.3333 - 0.3889	4.601×10^{-7}	4.643×10^{-7}
0.3889 - 0.5000	7.045×10^{-7}	7.138×10^{-7}
0.5000 - 1.0000	1.022×10^{-6}	1.049×10^{-6}
1.0000 - 1.3333	1.071×10^{-6}	1.111×10^{-6}
1.3333 - 1.6667	1.001×10^{-6}	1.044×10^{-6}
1.6667 - 2.0000	9.280×10^{-7}	9.723×10^{-7}
2.0000 - 8.0000	7.701×10^{-7}	8.465×10^{-7}
8.0000 - 10.0000	7.512×10^{-7}	8.989×10^{-7}
10.0000 - 24.0000	3.261×10^{-7}	4.640×10^{-7}
24.0000 - 96.0000	8.632×10^{-7}	3.214×10^{-6}
96.0000 - 720.0000	1.029×10^{-6}	8.347×10^{-6}

Table 7b
Proposed Modifications in Place

Time Span (Hours)	Iodine Partition Fraction: No Heat Exchanger Failure	Iodine Partition Fraction: Heat Exchanger Failure
0.0000 - 0.6000	0.000	0.000
0.6000 - 0.8333	1.022×10^{-6}	1.049×10^{-6}
0.8333 - 1.0000	1.022×10^{-6}	1.049×10^{-6}
1.0000 - 1.1667	1.071×10^{-6}	1.111×10^{-6}
1.1667 - 1.3333	1.071×10^{-6}	1.111×10^{-6}
1.3333 - 1.6667	1.001×10^{-6}	1.044×10^{-6}
1.6667 - 2.0000	9.280×10^{-7}	9.723×10^{-7}
2.0000 - 8.0000	9.139×10^{-7}	1.099×10^{-6}
8.0000 - 10.0000	1.082×10^{-6}	1.586×10^{-6}
10.0000 - 24.0000	6.056×10^{-7}	1.179×10^{-6}
24.0000 - 96.0000	2.154×10^{-6}	8.273×10^{-6}
96.0000 - 720.0000	2.238×10^{-6}	1.230×10^{-5}

3.2.9 Peak Clad Temperature LOCA Analysis

The proposed changes will not impact the LOCA analysis performed to determine the peak clad temperature. This analysis, presented in UFSAR Section 15.6.5, is a relatively short term analysis that terminates during the cold leg injection phase of a LOCA. The proposed changes will extend the duration of cold leg injection. Therefore, the current calculated peak clad temperatures are not affected by the proposed changes.

The proposed change to containment spray will not adversely impact the minimum containment pressure analysis included in the peak clad temperature analysis. The absence of containment spray would be expected to increase the minimum containment pressure as a function of time. However, for ice condenser plants, the increase in containment pressure resulting from the elimination of containment spray would be limited.

3.2.10 Impact to UFSAR Chapter 15 Category III and IV Events

The proposed modifications listed below were evaluated for potential impact to the UFSAR Chapter 15 Category III and IV events as identified in UFSAR Section 15.0.

- Increased initial RWST TS liquid inventory
- Decreased RWST low level alarm setpoint
- Decreased RWST low-low level alarm setpoint
- Elimination of the automatic containment spray actuation
- Revisions to EPs

The UFSAR Chapter 15 events evaluated are listed below:

1. Steam system piping failure (Section 15.1.5)

   Large and small steam line breaks may occur either inside or outside the containment building. Breaks located outside containment do not currently result in a containment spray actuation, and are therefore unaffected by the proposed change. UFSAR Section 15.1.5 is primarily concerned with the core response resulting from the increase in steam flow due to the steam line break. The increase in steam flow causes a decrease in the RCS temperature, which due to a negative moderator temperature coefficient, results in an increase in core thermal power. The core power increase is mitigated by the Reactor Protection System (RPS) and rapid isolation of main feedwater. The evaluation determines the fraction of fuel experiencing a Departure from Nucleate Boiling (DNB), which is translated into a failed fuel fraction. This fraction is input to an analysis to ensure the dose limits are satisfied. The limiting dose analysis assumes the break is located outside containment.

The proposed changes to the RWST and containment spray actuation logic primarily affect the containment response. Variations in containment pressure will not affect the RCS overcooling due to choked conditions being present at the steam line break location. Therefore, the core response calculations are not affected by the proposed modifications.

2. Feedwater system pipe break (Section 15.2.8)

Large and small feedwater system pipe breaks may occur either inside or outside the containment building. Breaks located outside containment do not currently result in a containment spray actuation, and are therefore unaffected by the proposed change. Feedwater system pipe breaks can have a variety of effects. Depending upon the size of the break and the plant operating conditions at the time of the break, the break could cause either a cooldown or a heatup of the RCS. Overcooling of the RCS due to a secondary side pipe rupture is evaluated in Section 15.1.5. UFSAR Section 15.2.8 evaluates the RCS heatup effects due to a secondary side pipe rupture.

A feedwater line rupture reduces the ability to remove heat generated by the core from the RCS. Section 15.2.8 is primarily concerned with establishing that adequate feedwater is available from the Auxiliary Feedwater System to prevent a substantial overpressurization of the RCS and that sufficient liquid is maintained in the RCS to provide adequate decay heat removal. The Section 15.2.8 evaluation focuses entirely on the conditions within the RCS. The containment response resulting from a feedwater line break is not considered in Chapter 15, but is bounded by the LOCA and steam line break analyses presented in Chapter 6.

The proposed modifications involve changes to containment spray actuation and RWST inventory. Therefore, the proposed modifications will not affect the results of this UFSAR section.

3. Complete loss of forced reactor coolant flow (Section 15.3.2)

   The evaluation performed for Section 15.3.2 considers the core response to a loss of forced reactor coolant flow. The primary concern evaluated is the decrease in the heat transferred from the fuel, and the potential for fuel rods to experience DNB. The fraction of fuel experiencing DNB is translated into a failed fuel fraction. This fraction is input to an analysis to ensure the dose limits are satisfied.

   The proposed modifications involve changes to containment spray actuation and RWST inventory. This event does not involve a high energy release into containment and thus does not result in containment spray actuation. The available RWST inventory similarly does not play a role in this event. Therefore, the proposed modifications will not affect the results of this UFSAR section.

4. Rod cluster control assembly misoperation (single rod cluster control assembly withdrawal at full power)(Section 15.4.3)

   There are four Rod Cluster Control Assembly (RCCA) misoperation events described in Section 15.4.3. These are: a) one or more dropped RCCAs within the same group, b) a dropped RCCA bank, c) a statically misaligned RCCA, and d) withdrawal of a single RCCA. Of these events, only the last one is a Category III event.

   The withdrawal of a single RCCA results in a core power increase. This is a relatively short duration event terminated by RPS action. The evaluation determines the fraction of fuel experiencing DNB, which is translated into a failed fuel fraction. This fraction is input to an analysis to ensure the dose limits are satisfied.

   The proposed modifications involve changes to containment spray actuation and RWST inventory. This event does not involve a high energy release into containment and thus does not result in containment spray actuation. The available RWST inventory similarly does not play a role in this event. Therefore, the proposed modifications will not affect the results of this UFSAR section.

5. Inadvertent loading and operation of a fuel assembly in an improper position (Section 15.4.7)

The inadvertent loading and operation of a fuel assembly event evaluation is primarily concerned with the local neutron flux peaks in the fuel pins. There are no radiological consequences associated with the inadvertent loading and operation of a fuel assembly in an improper position, since activity is contained within the fuel rods and the RCS remains within design limits.

The proposed modifications involve changes to containment spray actuation and RWST inventory. This event does not involve a high energy release into containment and thus does not result in containment spray actuation. The available RWST inventory similarly does not play a role in this event. Therefore, the proposed modifications will not affect the results of this UFSAR section.

6. Small break LOCA (Section 15.6.5)

The small break LOCA event is evaluated to ensure compliance with the criterion provided in 10 CFR 50.46. The small break event considered in Section 15.6.5 is defined as a rupture of the RCS pressure boundary with a total cross-sectional area less than 1.0 ft^2 in which the normally operating charging system flow is not sufficient to sustain pressurizer level and pressure. The proposed modifications will affect components that function during the small break LOCA analyses. Each of the modifications is discussed individually below.

The proposed modification to remove the automatic containment spray actuation logic will not impact the small break LOCA analysis. During a small break LOCA, RCS pressure remains elevated relative to the containment pressure. The elevated RCS pressure limits the ECCS injected into the RCS and extends the time frame during which break flow exceeds ECCS injection. The break flow characteristics are choked, and therefore independent of the downstream pressure.

The current small break LOCA analysis determines the time at which the RWST low level alarm is expected, and simulates a transition from cold leg injection to sump recirculation. Once in sump recirculation, the high head ECCS pumps are aligned to the RHR pump discharge. Sump liquid is pumped by the RHR pumps, through the RHR heat exchangers where it is cooled and supplied to the high head ECCS pumps. The end result is an increase in the ECCS injection temperature.

The proposed modifications to the RWST will increase the amount of liquid available for ECCS injection. The modification to remove the automatic containment spray actuation will reduce the total flow rate depleting the RWST inventory. Both of these changes will extend the duration of the cold leg injection phase of the event, which would represent a net benefit for the small break LOCA analysis. Therefore, the proposed changes will not adversely impact the results of this UFSAR section.

7. Radioactive gas waste system leak or failure (Section 15.7.1)

The accident is defined as an unexpected and uncontrolled release of radioactive xenon and krypton fission product gases stored in a waste gas decay tank as a consequence of a failure of a single gas decay tank or associated piping. The gas decay tanks and associated piping are not located within the containment building. Therefore, the proposed modifications to containment spray and the RWST will not affect the results of this UFSAR section.

8. Radioactive liquid waste system leak or failure (Section 15.7.2)

The accident is defined as an uncontrolled atmospheric release from the 112,000 gallon recycle holdup tank due to the postulated rupture of the tank. The recycle holdup tank is not located within the containment building. Therefore, the proposed modifications to containment spray and the RWST will not affect the results of this UFSAR section.

9. Postulated radioactive releases due to tank failures (Section 15.7.3)

The accident is defined as an uncontrolled atmospheric release from the 395,000 gallon RWST due to the postulated rupture of the tank. The analysis assumes the entire contents of the RWST are released and ensures that the associated radiological consequences are within acceptable limits. The proposed modifications include changes to the RWST liquid volume specifications. These changes affect the minimum RWST volume that may be credited in the safety analyses. The analysis presented in Section 15.7.3 assumes the full volume of the RWST, which is not affected by the proposed modifications. Therefore, the proposed modifications to containment spray and the RWST will not affect the results of this UFSAR section.

10. Spent fuel cask drop accidents (Section 15.7.5)

    The evaluation presented in Section 15.7.5 concludes that spent fuel casks cannot enter the spent fuel pool due to a postulated dropping or tipping of the cask. The spent fuel pool and associated casks are not located in the containment building. Therefore, the proposed modifications to containment spray and the RWST will not affect the results of this UFSAR section.

11. Reactor coolant pump shaft seizure (locked rotor) (Section 15.3.3)

    The evaluation performed for Section 15.3.3 considers the core response to a reactor coolant pump shaft seizure. This event causes a more severe loss of forced core cooling flow than the complete loss of forced coolant flow event described in Section 15.3.2. The primary concern evaluated is the decrease in the heat transferred from the fuel, and the potential for fuel rods to experience DNB. The fraction of fuel experiencing DNB is translated into a failed fuel fraction. This fraction is input to an analysis to ensure the dose limits are satisfied.

    The proposed modifications involve changes to containment spray actuation and RWST inventory. This event does not involve a high energy release into containment and thus does not result in containment spray actuation. The available RWST inventory similarly does not play a role in this event. Therefore, the proposed modifications will not affect the results of this UFSAR section.

12. Reactor coolant pump shaft break (Section 15.3.4)

    Section 15.3.4 considers the core response to a reactor coolant pump shaft break. This event is a less severe loss of forced core cooling flow than the event described in Section 15.3.3. This event is not specifically evaluated as it is bounded by Section 15.3.3.

The proposed modifications involve changes to containment spray actuation and RWST inventory. This event does not involve a high energy release into containment and thus does not result in containment spray actuation. The available RWST inventory similarly does not play a role in this event. Therefore, the proposed modifications will not affect the results of this UFSAR section.

13. Spectrum of rod cluster control assembly ejection accidents (Section 15.4.8)

    This accident is defined as the mechanical failure of a control rod mechanism pressure housing resulting in the ejection of a RCCA and drive shaft. The consequence of this mechanical failure is a rapid positive reactivity insertion together with an adverse core power distribution, possibly leading to localized fuel rod damage.

    The primary focus of the analysis described in Section 15.4.8 is the mechanical, neutronic, and thermal-hydraulic response to the rapid reactivity insertion. The results of this analysis define a failed fuel fraction which is input to an analysis to ensure the dose limits are satisfied. These calculations are performed for a relatively short duration to capture the fuel rod performance prior to RPS actuation.

    The containment response aspects of this event resulting from the break on the control rod housing are bounded by the Chapter 6 LOCA analysis results.

    The proposed modifications will not affect the core related calculations presented in this UFSAR section. The associated dose analysis does not credit the actuation of containment spray. Therefore, the proposed modifications will not affect the results of this UFSAR section.

14. Steam generator tube failure (Section 15.6.3)

    The accident examined is the complete severance of a single steam generator tube. The accident is assumed to take place at power with the reactor coolant contaminated with fission products corresponding to continuous operation with a limited amount of defective fuel rods. This event does not include a high energy break into containment. Thus, the proposed modification to remove the automatic containment spray actuation logic will not affect the analysis described in this UFSAR section.

The proposed modifications to the RWST will increase the amount of inventory available to mitigate the event. The steam generator tube rupture event is currently mitigated with the available RWST inventory. Therefore, the additional RWST inventory provided by the proposed modifications would represent additional margin. Therefore, the proposed modifications will not affect the results of this UFSAR section.

15. Large break LOCA (Section 15.6.5)

The large break LOCA event is evaluated to ensure compliance with the criterion provided in 10 CFR 50.46. The large break event considered in Section 15.6.5 is defined as a rupture of the RCS pressure boundary with a total cross-sectional area greater than or equal to 1.0 ft^2. The proposed modifications will affect components that function during the large break LOCA analyses.

The proposed changes will not adversely impact the LOCA analysis performed to determine the peak clad temperature. This analysis is a relatively short term analysis that terminates during the cold leg injection phase of a LOCA. The proposed changes to the RWST and containment spray actuation will extend the duration of cold leg injection. Therefore, the current calculated peak clad temperatures will not be adversely impacted by the proposed modifications.

The proposed change to containment spray will not adversely impact the minimum containment pressure analysis included in the peak clad temperature analysis. The absence of containment spray would be expected to increase the minimum containment pressure as a function of time. For ice condenser plants the change in containment pressure by eliminating containment spray would be limited.

16. Design basis fuel handling accidents (Section 15.7.4)

There are two events described in UFSAR Section 15.7.4. The first accident is defined as dropping of a spent fuel assembly, resulting in the rupture of the cladding of all the fuel rods in an assembly. The second accident is the postulated drop of one of two weir gates into the spent fuel pool. Both of these events are postulated to occur during refueling operations.

The major analysis inputs are the fission product inventory, spent fuel pool depth, and internal fuel rod pressure. The analysis determines a scrubbing fraction in the spent fuel pool to determine the atmospheric release. The objective of these analyses is to establish that the radiological consequences are within established limits.

The proposed modifications to the RWST and containment spray actuation logic do not alter assumptions made in this UFSAR section. Containment spray is not assumed to actuate and the RWST liquid volume is not an input to the analysis. Therefore, the proposed modifications will not affect the results of this UFSAR section.

3.2.11 Impact to Anticipated Transient without Scram (ATWS) Events

The proposed changes will not impact the core response analyses associated with ATWS events. The pressurizer relief valves lift during the course of the ATWS event, eventually causing the rupture disk on the pressurizer relief tank to break. The associated mass and energy release due to the blowdown of the pressurizer relief tank will not produce a limiting containment pressure response.

The proposed modification to remove the containment spray actuation logic will be a benefit to the plant response to an ATWS. If the containment pressure were to increase to the high-high containment pressure setpoint, the operation of the containment air return fans will be sufficient to ensure an acceptable containment pressure response. By eliminating the possibility of containment spray pump operation, the amount of liquid available for core cooling will be maximized and the plant response will be simplified.

3.2.12 Impact to Fire Protection

This proposed amendment has no impact on the plant's ability to respond to fire events. Catawba's fire protection systems and the fire protection plan are not adversely impacted by the proposed changes. In addition, Catawba's licensing basis does not require the simultaneous consideration of a design basis accident coupled with a fire event.

3.2.13 Impact to Lower Inlet Door TS

This proposed amendment has no adverse impact on the functioning of the ice condenser lower inlet doors in response to a design basis accident. The functions of the lower inlet doors are to: 1) seal the ice condenser from air leakage during the lifetime of the unit, and 2) open in the event of a design basis accident to direct the hot steam/air mixture from the event into the ice bed, where the ice absorbs energy and limits containment peak pressure and temperature during the accident transient.

In the event of a design basis accident, the lower inlet doors open due to the pressure rise in lower containment. This allows steam and air to flow into the ice condenser. The resulting pressure increase within the ice condenser causes the intermediate deck doors and the top deck doors to open, which allows the air to flow out of the ice condenser into the containment upper compartment. Limiting the pressure and temperature following a design basis accident reduces the release of fission product radioactivity from containment to the environment.

An additional design requirement for the ice condenser door response during a small break accident in which the flow of heated steam and air is not sufficient to fully open the doors is for all of the doors to partially open by approximately the same amount. Thus, the partially opened doors modulate the flow so that each ice bay receives an approximately equal fraction of the total flow. This design feature ensures that the heated steam and air will not flow preferentially to some ice bays and deplete the ice there without utilizing the ice in the other bays.

The lower inlet doors are designed to fully open in response to a 1 psf pressure differential. An automatic containment spray actuation would occur at a 3 psi pressure differential, or at a 432 psf pressure differential. Therefore, it is expected that absent a containment spray actuation, the lower inlet doors will respond as designed in the same manner as they would with the automatic signal in place. As a result, there will be no impact to the lower inlet door TS.

3.2.14 Impact to Early Containment Air Return Fan Operation

The analyses that supported the early containment air return fan operation submittal considered small breaks that did not reach the containment spray actuation setpoint. Therefore, the proposed modification to remove the containment spray actuation logic will not impact the analyses that support the associated Safety Evaluation Report.

3.2.15 Impact to Minimum Containment Sump Level Analysis

The proposed modifications will increase the available RWST liquid between the TS minimum and the RWST low level and RWST low-low level alarms. The proposed modifications will also eliminate the automatic containment spray actuation logic, eliminating any upper containment holdup penalty prior to reaching sump recirculation. These modifications will result in additional liquid inventory in the containment sump, and thus a higher sump level. Therefore, the proposed modifications will not adversely impact the minimum containment sump level analysis.

3.3 Plant Modifications and Procedure Changes

The following plant hardware modifications are associated with the proposed TS changes:

- Deletion or disabling the containment spray automatic actuation circuitry
- Adjustment of the RWST low and low-low level alarm setpoints
- Installation of a new non-safety related dual channel narrow range RWST level instrument loop
- Installation of a new redundant non-safety related wide range RWST level annunciator alarm
- Changing the containment isolation signal for the non-safety related normal containment cooling units from containment high-high pressure to containment high pressure

The proposed modifications also require that several changes be incorporated into the EPs. These changes are summarized below:

- Operator action to transfer high head and intermediate head safety injection pumps is taken upon receipt of the RWST low-low level alarm.
- An operator decision point is added to ensure adequate sump level for RHR pumps prior to RWST low level alarm

based upon new redundant non-safety related wide range RWST level annunciator alarm.

- Following transfer of the RHR pump suction to the containment sump at the RWST low level alarm, one containment spray pump may be manually started after adequate sump level is verified and if containment pressure remains greater than 3 psig.
- Following transfer of the RHR pump suction to the containment sump at the RWST low level alarm, one containment spray pump may be manually started after adequate sump level is verified and if containment spray is required for washout of fission products from the containment atmosphere.
- The operator does not align one RHR pump for auxiliary containment spray based on time. This action is taken based on reaching a containment pressure setpoint. For the supporting analysis, the containment pressure setpoint exceeds the maximum pressure attained. The plant procedures will employ a nominal setpoint.
- The EP Critical Safety Function (CSF) containment status tree will be updated to reflect changes in the requirement for containment spray pump operation. The status tree will be modified to be orange priority if all of the following conditions are met:

    1. Containment pressure is between Phase B containment isolation and RHR auxiliary spray actuation setpoint
    2. Containment spray is off or cooling is lost to the operating containment spray train
    3. Transfer of RHR pump suction to the containment sump has been performed

The new conditions being included in the orange priority for the containment status tree are item 3 and the second part of item 2. These items are consistent with the existing EP and Westinghouse Owners Group/Emergency Response Guidelines guidance that the status tree indicates an orange priority if containment spray is required, but is not operating. The addition of the cooling status to the second item ensures adequate monitoring of the containment spray cooling capability.

4. REGULATORY EVALUATION

4.1 Applicable Regulatory Requirements/Criteria

General Design Criterion 13 - Instrumentation and Control

"Instrumentation shall be provided to monitor variables and systems over their anticipated ranges for normal operation, for anticipated operational occurrences, and for accident conditions as appropriate to assure adequate safety, including those variables and systems that can affect the fission process, the integrity of the reactor core, the reactor coolant pressure boundary, and the containment and its associated systems. Appropriate controls shall be provided to maintain these variables and systems within prescribed operating ranges."

Discussion: The modifications proposed in this amendment request do not compromise the ability to monitor important variables and systems. Deletion of the automatic start function of the Containment Spray System will not result in the inability to monitor important reactor core, reactor coolant, or containment parameters. This criterion will continue to be met. The proposed change to adopt TSTF-493, Rev. 3 on a limited basis revises the TS to enhance the controls used to maintain the variables and systems within the prescribed operating ranges, in order to ensure that automatic protection actions occur as necessary to initiate the operation of systems and components important to safety as assumed in the accident analysis.

General Design Criterion 16 - Containment Design

"Reactor containment and associated systems shall be provided to establish an essentially leak-tight barrier against the uncontrolled release of radioactivity to the environment and to assure that the containment design conditions important to safety are not exceeded for as long as postulated accident conditions require."

Discussion: The proposed deletion of the automatic start function of the Containment Spray System will not compromise the overall effectiveness of the containment in serving as a barrier to fission product release following an accident. The safety analyses performed in support of this amendment request demonstrate that acceptable containment performance will be maintained post-accident. In addition, the containment will continue to be inspected and tested as specified in ASME Code, 10 CFR 50, Appendix J, and TS requirements.

General Design Criterion 19 - Control Room

"A control room shall be provided from which actions can be taken to operate the nuclear power unit safely under normal conditions and to maintain it in a safe condition under accident conditions, including loss-of-coolant accidents. Adequate radiation protection shall be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem whole body, or its equivalent to any part of the body, for the duration of the accident. Equipment at appropriate locations outside the control room shall be provided (1) with a design capability for prompt hot shutdown of the reactor, including necessary instrumentation and controls to maintain the unit in a safe condition during hot shutdown, and (2) with a potential capability for subsequent cold shutdown of the reactor through the use of suitable procedures.

Applicants for and holders of construction permits and operating licenses under this part who apply on or after January 10, 1997, applicants for design certifications under part 52 of this chapter who apply on or after January 10, 1997, applicants for and holders of combined licenses under part 52 of this chapter who do not reference a standard design certification, or holders of operating licenses using an alternative source term under § 50.67, shall meet the requirements of this criterion, except that with regard to control room access and occupancy, adequate radiation protection shall be provided to ensure that radiation exposures shall not exceed 0.05 Sv (5 rem) total effective dose equivalent (TEDE) as defined in § 50.2 for the duration of the accident."

Discussion: The proposed modifications do not in any way result in the loss or degradation of control room or alternate shutdown capability. The dose analyses performed in support of this amendment request demonstrate that control room doses remain within regulatory limits. No design changes are being made to the control room or ancillary shutdown equipment that will be detrimental to the ability to shut down the plant and to maintain shutdown conditions in the event of an accident.

General Design Criterion 20 - Protection System Functions

"The protection system shall be designed (1) to initiate automatically the operation of appropriate systems including the reactivity control systems, to assure that specified acceptable fuel design limits are not exceeded as a result of anticipated operational occurrences and (2) to sense accident conditions and to initiate the operation of systems and components important to safety."

Discussion: This proposed amendment, in part, deletes the automatic start function of the Containment Spray System. However, there is no impact on the ability of other protection system functions to be able to automatically start and initiate the operation of systems and components important to safety. Therefore, the ability to meet this criterion is not compromised. The proposed change to adopt TSTF-493, Rev. 3 on a limited basis revises the TS to enhance the controls used to maintain the variables and systems within the prescribed operating ranges, in order to ensure that automatic protection actions occur as necessary to initiate the operation of systems and components important to safety as assumed in the accident analysis.

General Design Criterion 21 - Protection System Reliability and Testability

"The protection system shall be designed for high functional reliability and inservice testability commensurate with the safety functions to be performed. Redundancy and independence designed into the protection system shall be sufficient to assure that (1) no single failure results in loss of the protection function and (2) removal from service of any component or channel does not result in loss of the required minimum redundancy unless the acceptable reliability of operation of the protection system can be otherwise demonstrated. The protection system shall be designed to permit periodic testing of its functioning when the reactor is in operation, including a capability to test channels independently to determine failures and losses of redundancy that may have occurred."

Discussion: The Reactor Trip System and the Engineered

Safety Features Actuation System reliability and testability will not be compromised as a result of the requested amendment. Both systems will retain their ability to perform their accident mitigation functions in the event of a single failure of a protection channel. Minimum redundancy requirements will continue to be met during all phases of plant operation, including testing conditions. Testing of these systems will continue to be governed by TS requirements.

General Design Criterion 38 - Containment Heat Removal

"A system to remove heat from the reactor containment shall be provided. The system safety function shall be to reduce rapidly, consistent with the functioning of other associated systems, the containment pressure and temperature following any loss-of-coolant accident and maintain them at acceptably low levels.

Suitable redundancy in components and features, and suitable interconnections, leak detection, isolation, and containment capabilities shall be provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure."

Discussion: This criterion will continue to be met with the proposed modifications in place. Even though the automatic start function of the Containment Spray System will no longer be required, the system will still be required to be operable by TS as a manually actuated system. The supporting analyses demonstrate that automatic start capability of this system is not required. In addition, the Ice Condenser System will continue to be able to perform its design function in response to accident conditions. No changes are being proposed which will impact the method of operation of the Ice Condenser System. The Ice Condenser System is a passive system which does not rely on the availability of electric power in order to perform its function. Associated systems that are utilized in conjunction with the Ice Condenser System (e.g., the Air Return

System) will continue to perform as designed, both with and without offsite electric power available.

General Design Criterion 39 - Inspection of Containment Heat Removal System

"The containment heat removal system shall be designed to permit appropriate periodic inspection of important components, such as the torus, sumps, spray nozzles, and piping to assure the integrity and capability of the system."

Discussion: The proposed amendment will not compromise the ability to meet this criterion. Although the automatic start function of the Containment Spray System is being deleted by the proposed modifications, this will not impact the ability to inspect the system. These inspections will continue to be performed as required by TS and in accordance with plant procedures.

General Design Criterion 40 - Testing of Containment Heat Removal System

"The containment heat removal system shall be designed to permit appropriate periodic pressure and functional testing to assure (1) the structural and leaktight integrity of its components, (2) the operability and performance of the active components of the system, and (3) the operability of the system as a whole, and under conditions as close to the design as practical the performance of the full operational sequence that brings the system into operation, including operation of applicable portions of the protection system, the transfer between normal and emergency power sources, and the operation of the associated cooling water system."

Discussion: The requested amendment will, in part, delete the automatic start function of the Containment Spray System. However, the system will still be able to be fully actuated by manual operator action. The mechanical portions of the system will retain their ability to be pressure and functionally tested. Applicable TS requirements will still exist to govern testing of the mechanical portions of the system. The proposed modifications will delete the start of the system via the automatic actuation logic and actuation relays and the containment pressure high-high signal (TS Table 3.3.2-1, Functions 2b and 2c, respectively). The system will retain its manual initiation capability (Function 2a). Therefore, the "full operational sequence that brings the system into operation" consists completely of operator actions taken in accordance with EPs (as revised in accordance with the proposed modifications). No "portions of the protection system" will be applicable to the containment heat removal function. The transfer between the normal (offsite) and emergency (onsite) power sources will continue to be verified as part of TS required AC power source testing requirements. Finally, the operation of cooling water support system capability will continue to be tested.

General Design Criterion 41 - Containment Atmosphere Cleanup

"Systems to control fission products, hydrogen, oxygen, and other substances which may be released into the reactor containment shall be provided as necessary to reduce, consistent with the functioning of other associated systems, the concentration and quality of fission products released to the environment following postulated accidents, and to control the concentration of hydrogen or oxygen and other substances in the containment atmosphere following postulated accidents to assure that containment integrity is maintained.

Each system shall have suitable redundancy in components and features, and suitable interconnections, leak detection, isolation, and containment capabilities to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) its safety function can be accomplished, assuming a single failure."

Discussion: The proposed amendment will not compromise the ability of the Containment Spray System to perform its role in containment cleanup. The supporting analyses demonstrate that spray removal of diatomic iodine and particulate fission products remains within acceptable limits. No impact on any systems utilized to control the concentration of hydrogen and oxygen in containment is realized in conjunction with the proposed modifications. These systems will retain their ability to perform their design functions in the event of a single failure.

General Design Criterion 42 - Inspection of Containment Atmosphere Cleanup Systems

"The containment atmosphere cleanup systems shall be designed to permit appropriate periodic inspection of important components, such as filter frames, ducts, and piping to assure the integrity and capability of the systems."

Discussion: The Containment Spray System will retain its ability to undergo all appropriate inspection requirements following implementation of the proposed amendment. These inspection requirements are conducted in accordance with the Catawba Inservice Inspection Program.

General Design Criterion 43 - Testing of Containment Atmosphere Cleanup Systems

"The containment atmosphere cleanup systems shall be designed to permit appropriate periodic pressure and functional testing to assure (1) the structural and leaktight integrity of its components, (2) the operability and performance of the active components of the systems such as fans, filters, dampers, pumps, and valves and (3) the operability of the systems as a whole and, under conditions as close to design as practical, the performance of the full operational sequence that brings the systems into operation, including operation of applicable portions of the protection system, the transfer between normal and emergency power sources, and the operation of associated systems."

Discussion: The Containment Spray System will retain its ability to undergo all appropriate testing requirements following implementation of the proposed amendment. These testing requirements are conducted in accordance with the Catawba Inservice Testing Program and TS.

General Design Criterion 50 - Containment Design Basis

"The reactor containment structure, including access openings, penetrations, and the containment heat removal system shall be designed so that the containment structure and its internal compartments can accommodate, without exceeding the design leakage rate and with sufficient margin, the calculated pressure and temperature conditions resulting from any loss-of-coolant accident. This margin shall reflect consideration of (1) the effects of potential energy sources which have not been included in the determination of the peak conditions, such as energy in steam generators and as required by § 50.44 energy from metal-water and other chemical reactions that may result from degradation but not total failure of emergency core cooling functioning, (2) the limited experience and experimental data available for defining accident phenomena and containment responses, and (3) the conservatism of the calculational model and input parameters."

Discussion: This criterion will continue to be met following implementation of the proposed modifications. The overall function of the containment system will be maintained. Supporting analyses demonstrate that containment performance will remain acceptable following the design basis LOCA. The existing design basis limits regarding post-accident containment pressure and temperature will not be exceeded. In addition, the containment design leakage rate as specified in TS will not be exceeded. The input assumptions inherent in the calculated margin of the overall containment system continue to remain valid.

4.2 Precedent

There have been no precedents established in conjunction with this license amendment request.

4.3 Significant Hazards Consideration

The proposed amendment modifies the Catawba TS to: 1) eliminate Containment Spray System automatic start on a high-high containment pressure signal, 2) raise the minimum RWST volume limit, and 3) lower the RWST low level setpoint. Plant modifications are required to delete the Containment Spray System start function, to install narrow range RWST level indication, and to lower the RWST low level actuation setpoint.

Duke has evaluated whether or not a significant hazard consideration is involved with the proposed changes by analyzing the three standards set forth in 10 CFR 50.92(c) as discussed below:

Criterion 1:

Does the proposed amendment involve a significant increase in the probability or consequences of an accident previously evaluated?

Response: No.

The Containment Spray System and RWST are accident mitigation equipment. As such, changes in operation of these systems cannot have an impact on the probability of an accident.

The RWST will continue to comply with all applicable regulatory requirements and design criteria following approval of the proposed changes (e.g., train separation, redundancy, and single failure). The water level on the containment floor will be higher at the start of transfer to the containment sump but will remain below the maximum design level analyzed for equipment submergence. The change in the sump pH will not result in a significant increase in radiological consequences of a LOCA. Therefore, the design functions performed by the equipment are not changed.

The delay in containment spray operation will result in an increase in containment temperature, containment pressure, offsite dose, and control room dose during a LOCA or high energy line break inside containment. Containment analyses have been performed to demonstrate that containment pressure and temperature remain within the design limits and there is no significant impact on the environmental qualification for equipment inside containment. The impact on piping and supports is acceptable without modification. The reduction in fission product removal due to delayed containment spray operation does not result in exceeding the offsite dose and control room dose limits in 10 CFR 50.67 and 10 CFR 50, Appendix A, GDC 19. The analysis of the change in containment conditions due to a single failure of an operating spray pump and the suspension of containment spray determined that the pressure remained below the design limits.

Regarding the proposed change to adopt TSTF-493, Rev. 3 on a limited basis, the change clarifies the requirements for instrumentation to ensure the instrumentation will actuate as assumed in the safety analysis. Instruments are not an assumed initiator of any accident previously evaluated. As a result, the proposed change will not increase the probability of an accident previously evaluated. The proposed change will ensure that the instruments actuate as assumed to mitigate the accidents previously evaluated. As a result, the proposed change will not increase the consequences of an accident previously evaluated.

Based on this discussion, the proposed amendment does not significantly increase the probability or consequences of an accident previously evaluated.

Criterion 2:

Does the proposed amendment create the possibility of a new or different kind of accident from any accident previously evaluated?

Response: No.

The modifications to install RWST narrow range level indication will be seismically qualified and isolated from the safety related portion of the RWST level indication system. As such, the new level indication will not create the possibility of a new or different kind of accident.

The modification to the low level setpoint will not install any new plant equipment. The setpoint will continue to be included within the engineered safeguards features instrumentation and monitored according to the applicable surveillance requirements. The evaluation of the new level setpoint and the change in the swapover sequence concluded that the equipment aligned to the sump will continue to have sufficient suction pressure prior to containment sump suction swapover. The design of the RWST low level instrumentation complies with all applicable regulatory requirements and design criteria.

The overall function of the Containment Spray System is not changed by this proposed amendment. The proposed change alters the method of controlling the safety system following a design basis event so that manual actions are substituted for automatic actions. Calculations confirm that these actions will be taken within the appropriate scenario sequence timing to provide containment cooling and source term reduction with no significant increase in radiological consequences and without exceeding containment design limits.

Regarding the proposed change to adopt TSTF-493, Rev. 3 on a limited basis, the change does not involve a physical alteration of the plant (i.e., no new or different type of equipment will be installed) or a change in the methods governing normal plant operation. The change does not alter assumptions made in the safety analysis but ensures that the instruments behave as assumed in the accident analysis. The proposed change is consistent with the safety analysis assumptions.

Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

Criterion 3:

Does the proposed amendment involve a significant reduction in a margin of safety?

Response: No.

The proposed change has the potential to increase the radiological dose at the site boundary and in the control room. However, the calculations demonstrate that the dose consequences at the site boundary, low population zone, and control room remain within regulatory acceptance limits. Additional analysis concluded:

- Peak containment pressure for analyzed design basis accidents will not be significantly increased and containment design limits will not be exceeded.
- Assumptions used in the environmental qualification of equipment exposed to the containment atmosphere remain bounding.
- Pumps aligned to the RWST and to the containment sump will have adequate suction pressure.

Regarding the proposed change to adopt TSTF-493, Rev. 3 on a limited basis, the change clarifies the requirements for instrumentation to ensure the instrumentation will actuate as assumed in the accident analysis. No change is made to the accident analysis assumptions and no margin of safety is reduced as part of this change.

Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Based on the above, Duke concludes that the proposed amendment does not involve a significant hazards consideration under the standards set forth in 10 CFR

50.92(c), and, accordingly, a finding of no significant hazards consideration is justified.

4.4 Conclusions

In conclusion, based on the considerations discussed above, (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

5. ENVIRONMENTAL CONSIDERATION

Duke has determined that the proposed amendment does change requirements with respect to the installation or use of a facility component located within the restricted area, as defined by 10 CFR 20. It also represents a change to surveillance requirements. Duke has evaluated the proposed changes and has determined that they do not involve: (1) a significant hazards consideration, (2) a significant change in the types or a significant increase in the amounts of any effluents that may be released offsite, or (3) a significant increase in individual or cumulative occupational radiation exposures. Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

6. REFERENCES

1. Catawba Nuclear Station Updated Final Safety Analysis, 2006 Update.

2. G.R. Peterson (Duke Energy Corporation) to U.S. Nuclear Regulatory Commission, "Duke Energy Corporation Catawba Nuclear Station, Units 1 and 2 Docket Numbers 50-413 and 50-414 Proposed Technical Specifications Bases Amendment Technical Specification and Bases 3.6.10 Annulus Ventilation System (AVS) Technical Specification Bases 3.6.16 Reactor Building Technical Specification Bases 3.7.10 Control Room Area Ventilation System (CRAVS) Technical Specification Bases 3.7.12 Auxiliary Building Filtered Ventilation Exhaust System (ABFVES) Technical Specification Bases 3.7.13 Fuel Handling Ventilation Exhaust System (FHVES) Technical Specification and Bases 3.9.3 Containment Penetrations Technical Specification 5.5.11 Ventilation Filter Testing Program (VFTP)," November 25, 2002.

3. D.M. Jamil, (Duke Energy Corporation) to U.S. Nuclear Regulatory Commission, "Duke Energy Corporation Catawba Nuclear Station, Units 1 and 2 Docket Numbers 50-413 and 50-414 Proposed Technical Specifications Bases Amendment Technical Specification and Bases 3.6.10 Annulus Ventilation System (AVS) Technical Specification Bases 3.6.16 Reactor Building Technical Specification Bases 3.7.10 Control Room Area Ventilation System (CRAVS) Technical Specification Bases 3.7.12 Auxiliary Building Filtered Ventilation Exhaust System (ABFVES) Technical Specification Bases 3.7.13 Fuel Handling Ventilation Exhaust System (FHVES) Technical Specification and Bases 3.9.3 Containment Penetrations Technical Specification 5.5.11 Ventilation Filter Testing Program (VFTP) TAC Numbers MB7014 and MB7015," November 13, 2003.

4. D.M. Jamil, (Duke Energy Corporation) to U.S. Nuclear Regulatory Commission, "Duke Energy Corporation Catawba Nuclear Station, Units 1 and 2 Docket Numbers 50-413 and 50-414 Proposed Technical Specifications Bases Amendment Technical Specification and Bases 3.6.10 Annulus Ventilation System (AVS) Technical Specification Bases 3.6.16 Reactor Building Technical Specification Bases 3.7.10 Control Room Area Ventilation System (CRAVS) Technical Specification Bases 3.7.12 Auxiliary Building Filtered Ventilation Exhaust System (ABFVES) Technical Specification Bases 3.7.13 Fuel Handling Ventilation Exhaust System (FHVES)

Technical Specification and Bases 3.9.3 Containment Penetrations Technical Specification 5.5.11 Ventilation Filter Testing Program (VFTP) TAC Numbers MB7014 and MB7015," December 16, 2003.

5. D.M. Jamil, (Duke Energy Corporation) to U.S. Nuclear Regulatory Commission, "Duke Energy Corporation Catawba Nuclear Station, Units 1 and 2 Docket Numbers 50-413 and 50-414 Proposed Technical Specifications Bases Amendment Technical Specification and Bases 3.6.10 Annulus Ventilation System (AVS) Technical Specification Bases 3.6.16 Reactor Building Technical Specification Bases 3.7.10 Control Room Area Ventilation System (CRAVS) Technical Specification Bases 3.7.12 Auxiliary Building Filtered Ventilation Exhaust System (ABFVES) Technical Specification Bases 3.7.13 Fuel Handling Ventilation Exhaust System (FHVES) Technical Specification and Bases 3.9.3 Containment Penetrations Technical Specification 5.5.11 Ventilation Filter Testing Program (VFTP) TAC Numbers MB7014 and MB7015," September 22, 2004.

6. D.M. Jamil, (Duke Energy Corporation) to U.S. Nuclear Regulatory Commission, "Duke Energy Corporation Catawba Nuclear Station, Units 1 and 2 Docket Numbers 50-413 and 50-414 Proposed Technical Specifications Bases Amendment Technical Specification and Bases 3.6.10 Annulus Ventilation System (AVS) Technical Specification Bases 3.6.16 Reactor Building Technical Specification Bases 3.7.10 Control Room Area Ventilation System (CRAVS) Technical Specification Bases 3.7.12 Auxiliary Building Filtered Ventilation Exhaust System (ABFVES) Technical Specification Bases 3.7.13 Fuel Handling Ventilation Exhaust System (FHVES) Technical Specification and Bases 3.9.3 Containment Penetrations Technical Specification 5.5.11 Ventilation Filter Testing Program (VFTP) TAC Numbers MB7014 and MB7015," August 17, 2005.

7. Bechtel Corporation, <u>LOCADOSE NE-319 User's Manual</u>, Rev 8), May, 1999.

8. Bechtel Corporation, <u>LOCADOSE NE-319 Theoretical Manual</u>, Rev 8), May, 1999.

9. Bechtel Corporation, <u>LOCADOSE NE-319 Validation Manual</u>, Rev 9), May, 1999.

10. USNRC, Alternative Radiological Source Term for Evaluating Design Basis Accidents at Nuclear Power Reactors, Regulatory Guide 1.183, July 2000.

11. Ibid., Appendix A, "Assumptions for Evaluating the Radiological Assessments of a LWR Loss-of-Coolant Accident."

12. S.E. Peters (USNRC) to D.M. Jamil, "Catawba Nuclear Station, Units 1 and 2 Re: Issuance of Amendments (TAC Nos. MB7014 and MB7015)," September 30, 2005.

Attachment 2

Marked-Up TS and Bases Pages

TS Markup Inserts

INSERT 1: * The requirements of this Function are not applicable for entry into the applicable MODES following implementation of the modifications associated with ECCS Water Management on the respective unit.

INSERT 2: * Following implementation of the modifications associated with ECCS Water Management on the respective unit, the Allowable Value for this Function shall be ≥ 91.9 inches and the Nominal Trip Setpoint for this Function shall be 95 inches.

INSERT 3: * Following implementation of the modifications associated with ECCS Water Management on the respective unit, the RWST borated water volume for this SR shall be ≥ 377,537 gallons.

INSERT 4: * Following implementation of the modifications associated with ECCS Water Management on the respective unit, the requirements of SR 3.6.6.4 shall no longer be applicable.

INSERT TSTF-493 NOTE 1: If the as-found channel setpoint is outside its predefined as-found tolerance, then the channel shall be evaluated to verify that it is functioning as required before returning the channel to service.

INSERT TSTF-493 NOTE 2: The instrument channel setpoint shall be reset to a value that is within the as-left tolerance around the Nominal Trip Setpoint (NTSP) at the completion of the surveillance; otherwise, the channel shall be declared inoperable. Setpoints more conservative than the NTSP are acceptable provided that the as-found and as-left tolerances apply to the actual setpoint implemented in the Surveillance procedures to confirm channel performance. The methodologies used to determine the as-found and the as-left tolerances are specified in the UFSAR.

INSERT for SR 3.3.2.7:

For Functions for which TSTF-493, "Clarify Application of Setpoint Methodology for LSSS Functions" has been implemented, this SR is modified by two Notes as identified in Table 3.3.2-1. The first Note requires evaluation of channel performance for the condition where the as-found setting for the channel setpoint is outside its as-found tolerance but conservative with respect to the Allowable Value. Evaluation of channel performance will verify that the channel will continue to behave in accordance with safety analysis assumptions and the channel performance assumptions in the setpoint methodology. The purpose of the assessment is to ensure confidence in the channel performance prior to returning the channel to service. The performance of these channels will be evaluated under the station's Corrective Action Program. Entry into the Corrective Action Program will ensure required review and documentation of the condition for continued OPERABILITY. The second Note requires that the as-left setting for the channel be returned to within the as-left tolerance of the Nominal Trip Setpoint (NTSP). Where a setpoint more conservative than the NTSP is used in the plant surveillance procedures, the as-left and as-found tolerances, as applicable, will be applied to the surveillance procedure setpoint. This will ensure that sufficient margin to the Safety Limit and/or Analytical Limit is maintained. If the as-left channel setting cannot be returned to a setting within the as-left tolerance of the NTSP, then the channel shall be declared inoperable. The second Note also requires that the methodologies for calculating the as-left and the as-found tolerances be in the UFSAR.

INSERT for SR 3.3.2.9:

For Functions for which TSTF-493, "Clarify Application of Setpoint Methodology for LSSS Functions" has been implemented, this SR is modified by two Notes as identified in Table 3.3.2-1. The first Note requires evaluation of channel performance for the condition where the as-found setting for the channel setpoint is outside its as-found tolerance but conservative with respect to the Allowable Value. Evaluation of channel performance will verify that the channel will continue to behave in accordance with safety analysis assumptions and the channel performance assumptions in the setpoint methodology. The purpose of the assessment is to ensure confidence in the channel performance prior to returning the channel to service. The performance of these channels will be evaluated under the station's Corrective Action Program. Entry into the Corrective Action Program will ensure required review and documentation of the condition for continued OPERABILITY. The second Note requires that the as-left setting for the channel be returned to within the as-left tolerance of the Nominal Trip Setpoint (NTSP). Where a setpoint more conservative than the NTSP is used in the plant surveillance procedures, the as-left and as-found tolerances, as applicable, will be applied to the surveillance procedure setpoint. This will ensure that sufficient margin to the Safety Limit and/or Analytical Limit is maintained. If the as-left channel setting cannot be returned to a setting within the as-left tolerance of the NTSP, then the channel shall be declared inoperable. The second Note also requires that the methodologies for calculating the as-left and the as-found tolerances be in the UFSAR.

Table 3.3.2-1 (page 1 of 5)
Engineered Safety Feature Actuation System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	NOMINAL TRIP SETPOINT
1. Safety Injection[b]						
a. Manual initiation	1,2,3,4	2	B	SR 3.3.2.8	NA	NA
b. Automatic Actuation Logic and Actuation Relays	1,2,3,4	2 trains	C	SR 3.3.2.2 SR 3.3.2.4 SR 3.3.2.6	NA	NA
c. Containment Pressure - High	1,2,3	3	D	SR 3.3.2.1 SR 3.3.2.5 SR 3.3.2.9 SR 3.3.2.10	≤ 1.4 psig	1.2 psig
d. Pressurizer Pressure - Low	1,2,3[a]	4	D	SR 3.3.2.1 SR 3.3.2.5 SR 3.3.2.9 SR 3.3.2.10	≥ 1839 psig	1845 psig
2. Containment Spray						
a. Manual Initiation	1,2,3,4	1 per train, 2 trains	B	SR 3.3.2.8	NA	NA
b. Automatic Actuation Logic and Actuation Relays *	1,2,3,4	2 trains	C	SR 3.3.2.2 SR 3.3.2.4 SR 3.3.2.6	NA	NA
c. Containment Pressure - High High *	1,2,3	4	E	SR 3.3.2.1 SR 3.3.2.5 SR 3.3.2.9 SR 3.3.2.10	≤ 3.2 psig	3.0 psig
3. Containment Isolation[b]						
a. Phase A Isolation						
(1) Manual Initiation	1,2,3,4	2	B	SR 3.3.2.8	NA	NA
(2) Automatic Actuation Logic and Actuation Relays	1,2,3,4	2 trains	C	SR 3.3.2.2 SR 3.3.2.4 SR 3.3.2.6	NA	NA
(3) Safety Injection	Refer to Function 1 (Safety Injection) for all initiation functions and requirements.					

INSERT 1

(continued)

(a) Above the P-11 (Pressurizer Pressure) interlock.

(b) The requirements of this Function are not applicable to Containment Purge Ventilation System and Hydrogen Purge System components, since the system containment isolation valves are sealed closed in MODES 1, 2, 3, and 4.

Table 3.3.2-1 (page 5 of 5)
Engineered Safety Feature Actuation System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	NOMINAL TRIP SETPOINT
7. Automatic Switchover to Containment Sump						
a. Automatic Actuation Logic and Actuation Relays	1,2,3,4	2 trains	C	SR 3.3.2.2 SR 3.3.2.4 SR 3.3.2.6	NA	NA
b. Refueling Water Storage Tank (RWST) Level – Low	1,2,3,4	4	N	SR 3.3.2.1 SR 3.3.2.7 (a)(b) SR 3.3.2.9 (a)(b) SR 3.3.2.10	≥ 162.4 inches (*)	177.15 inches (*)
Coincident with Safety Injection	Refer to Function 1 (Safety Injection) for all initiation functions and requirements.					
8. ESFAS Interlocks						
a. Reactor Trip, P-4	1,2,3	1 per train, 2 trains	F	SR 3.3.2.8	NA	NA
b. Pressurizer Pressure, P-11	1,2,3	3	O	SR 3.3.2.5 SR 3.3.2.9	≥ 1944 and ≤ 1966 psig	1955 psig
c. T_{avg} - Low Low, P-12	1,2,3	1 per loop	O	SR 3.3.2.5 SR 3.3.2.9	≥ 550°F	553°F
9. Containment Pressure Control System						
a. Start Permissive	1,2,3,4	4 per train	P	SR 3.3.2.1 SR 3.3.2.7 SR 3.3.2.9	≤ <u>1.0</u> psid	<u>0.9</u> psid
b. Termination	1,2,3,4	4 per train	P	SR 3.3.2.1 SR 3.3.2.7 SR 3.3.2.9	≥ 0.25 psid	<u>0.35</u> psid
10. Nuclear Service Water Suction Transfer - Low Pit Level	1,2,3,4	3 per pit	Q,R	SR 3.3.2.1 SR 3.3.2.9 SR 3.3.2.11 SR 3.3.2.12	≥ El. 555.4 ft	El. 557.5 ft

INSERT 2

(a) INSERT TSTF-493 NOTE 1
(b) INSERT TSTF-493 NOTE 2

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.5.4.1	Verify RWST borated water temperature is $\geq$ 70°F and $\leq$ 100°F.	24 hours
SR 3.5.4.2	Verify RWST borated water volume is $\geq$ 363,513 gallons. *	7 days
SR 3.5.4.3	Verify RWST boron concentration is within the limits specified in the COLR.	7 days

INSERT 3

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.6.6.2	Verify each containment spray pump's developed head at the flow test point is greater than or equal to the required developed head.	In accordance with the Inservice Testing Program
SR 3.6.6.3	Verify each automatic containment spray valve in the flow path that is not locked, sealed, or otherwise secured in position, actuates to the correct position on an actual or simulated actuation signal.	18 months
SR 3.6.6.4	Verify each containment spray pump starts automatically on an actual or simulated actuation signal. *	18 months
SR 3.6.6.5	Verify that each spray pump is de-energized and prevented from starting upon receipt of a terminate signal and is allowed to start upon receipt of a start permissive from the Containment Pressure Control System (CPCS).	18 months
SR 3.6.6.6	Verify that each spray pump discharge valve closes or is prevented from opening upon receipt of a terminate signal and is allowed to open upon receipt of a start permissive from the Containment Pressure Control System (CPCS).	18 months
SR 3.6.6.7	Verify each spray nozzle is unobstructed.	10 years

INSERT 4

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

This Function must be OPERABLE in MODES 1, 2, and 3 (above P-11) to mitigate the consequences of an HELB inside containment. This signal may be manually blocked by the operator below the P-11 setpoint. Automatic SI actuation below this pressure setpoint is then performed by the Containment Pressure-High signal.

This Function is not required to be OPERABLE in MODE 3 below the P-11 setpoint. Other ESF functions are used to detect accident conditions and actuate the ESF systems in this MODE. In MODES 4, 5, and 6, this Function is not needed for accident detection and mitigation.

2. Containment Spray

Containment Spray provides two primary functions:

1. Lowers containment pressure and temperature after an HELB in containment; and

2. Reduces the amount of radioactive iodine in the containment atmosphere.

These functions are necessary to:

- Ensure the pressure boundary integrity of the containment structure; and

- Limit the release of radioactive iodine to the environment in the event of a failure of the containment structure.

pump is started manually and

One ~~The~~ containment spray ~~actuation signal starts the containment spray pumps and aligns~~ the discharge of the pump~~s~~ to the containment spray nozzle headers in the upper levels of containment. is aligned ~~Water is initially drawn from the RWST by the containment spray pumps. When the RWST reaches the low low level setpoint, the spray pump suctions are manually shifted to the containment sump if continued containment spray is required. Containment spray is actuated manually or by Containment Pressure-High High.~~

The containment sump is the suction source for the water to the containment spray pumps and one pump is started once adequate level in the containment sump has been established. The second train of containment spray is available in the event of a failure of the first train.

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

a. Containment Spray - Manual Initiation

There are two manual Containment Spray switches, one per train, in the control room. Turning the switch will actuate the associated containment spray train in the same manner as the automatic actuation signal. Two Manual Initiation switches, one per train, are required to be OPERABLE to ensure no single failure disables the Manual Initiation Function. Note that Manual Initiation of containment spray also actuates Phase B containment isolation. Two train actuation requires operation of both Train A and Train B manual containment spray switches.

b. Containment Spray-Automatic Actuation Logic and Actuation Relays

Automatic actuation logic and actuation relays consist of the same features and operate in the same manner as described for ESFAS Function 1.b.

Manual and automatic initiation of containment spray must be OPERABLE in MODES 1, 2, and 3 when there is a potential for an accident to occur, and sufficient energy in the primary or secondary systems to pose a threat to containment integrity due to overpressure conditions. In MODE 4, adequate time is available to manually actuate required components in the event of a DBA. However, because of the large number of components actuated on a containment spray, actuation is simplified by the use of the manual actuation push buttons. Automatic actuation logic and actuation relays must be OPERABLE in MODE 4 to support system level manual initiation. In MODES 5 and 6, there is insufficient energy in the primary and secondary systems to result in containment overpressure. In MODES 5 and 6, there is also adequate time for the operators to evaluate unit conditions and respond, to mitigate the consequences of abnormal conditions by manually starting individual components.

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

c. Containment Spray-Containment Pressure - High High

This signal provides protection against a LOCA or an SLB inside containment.

This is one of the only Functions that requires the bistable output to energize to perform its required action. It is not desirable to have a loss of power actuate containment spray, since the consequences of an inadvertent actuation of containment spray could be serious. Note that this Function also has the inoperable channel placed in bypass rather than trip to decrease the probability of an inadvertent actuation.

Containment Pressure-High High uses four channels in a two-out-of-four logic configuration. Since containment pressure is not used for control, this arrangement exceeds the minimum redundancy requirements. Additional redundancy is warranted because this Function is energize to trip. Containment Pressure-High High must be OPERABLE in MODES 1, 2, and 3 when there is sufficient energy in the primary and secondary sides to pressurize the containment following a pipe break. In MODES 4, 5, and 6, there is insufficient energy in the primary and secondary sides to pressurize the containment and reach the Containment Pressure-High High setpoints.

3. Containment Isolation

Containment Isolation provides isolation of the containment atmosphere, and all process systems that penetrate containment, from the environment. This Function is necessary to prevent or limit the release of radioactivity to the environment in the event of a large break LOCA.

There are two separate Containment Isolation signals, Phase A and Phase B. Phase A isolation isolates all automatically isolable process lines, except component cooling water (CCW) and nuclear service water system (NSWS), at a relatively low containment pressure indicative of primary or secondary system leaks. For

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

Phase B containment isolation is actuated by Containment Pressure-High High, or manually, via the automatic actuation logic, as previously discussed. For containment pressure to reach a value high enough to actuate Containment Pressure—High High, a large break LOCA or SLB must have occurred ~~and containment spray must have been actuated~~. RCP operation will no longer be required and CCW to the RCPs and NSWS to the RCP motor coolers are, therefore, no longer necessary. The RCPs can be operated with seal injection flow alone and without CCW flow to the thermal barrier heat exchanger.

Pushbuttons on the main control board. In addition to manually initiating a Phase B Containment Isolation, the pushbuttons also isolate the containment ventilation system.

Manual Phase B Containment Isolation is accomplished ~~by the same switches that actuate Containment Spray. When either train's switch is turned, Phase B Containment Isolation and Containment Spray will be actuated in its respective train.~~

a. Containment Isolation-Phase A Isolation

(1) Phase A Isolation-Manual Initiation

Manual Phase A Containment Isolation is actuated by either of two switches in the control room. Each switch actuates its respective train.

(2) Phase A Isolation-Automatic Actuation Logic and Actuation Relays

Automatic Actuation Logic and Actuation Relays consist of the same features and operate in the same manner as described for ESFAS Function 1.b.

Manual and automatic initiation of Phase A Containment Isolation must be OPERABLE in MODES 1, 2, and 3, when there is a potential for an accident to occur. In MODE 4, adequate time is available to manually actuate required components in the event of a DBA, but because of the large number of components actuated on a Phase A Containment Isolation, actuation is simplified by the use of the manual actuation push buttons. Automatic actuation logic and actuation relays must be OPERABLE in MODE 4 to support

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

system level manual initiation. In MODES 5 and 6, there is insufficient energy in the primary or secondary systems to pressurize the containment to require Phase A Containment Isolation. There also is adequate time for the operator to evaluate unit conditions and manually actuate individual isolation valves in response to abnormal or accident conditions.

(3) Phase A Isolation-Safety Injection

Phase A Containment Isolation is also initiated by all Functions that initiate SI. The Phase A Containment Isolation requirements for these Functions are the same as the requirements for their SI function. Therefore, the requirements are not repeated in Table 3.3.2-1. Instead, Function 1, SI, is referenced for all initiating Functions and requirements.

b. Containment Isolation-Phase B Isolation

Phase B Containment Isolation is accomplished by manual Initiation, Automatic Actuation Logic and Actuation Relays, and by Containment Pressure channels ~~(the same channels that actuate Containment Spray, Function 2)~~. The Containment Pressure trip of Phase B Containment Isolation is energized to trip in order to minimize the potential of spurious trips that may damage the RCPs.

(1) Phase B Isolation-Manual Initiation

(2) Phase B Isolation-Automatic Actuation Logic and Actuation Relays

Manual and automatic initiation of Phase B containment isolation must be OPERABLE in MODES 1, 2, and 3, when there is a potential for an accident to occur. In MODE 4, adequate time is available to manually actuate required components in the event of a DBA. However, because of the large number of components actuated on a Phase B

ACTIONS (continued)

C.1, C.2.1 and C.2.2

Condition C applies to the automatic actuation logic and actuation relays for the following functions:

- SI;
- ~~Containment Spray;~~
- Phase A Isolation;
- Phase B Isolation; and
- Automatic Switchover to Containment Sump.

This action addresses the train orientation of the SSPS and the master and slave relays. If one train is inoperable, 6 hours are allowed to restore the train to OPERABLE status. The specified Completion Time is reasonable considering that there is another train OPERABLE, and the low probability of an event occurring during this interval. If the train cannot be restored to OPERABLE status, the unit must be placed in a MODE in which the LCO does not apply. This is done by placing the unit in at least MODE 3 within an additional 6 hours (12 hours total time) and in MODE 5 within an additional 30 hours (42 hours total time). The Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

The Required Actions are modified by a Note that allows one train to be bypassed for up to 4 hours for surveillance testing, provided the other train is OPERABLE. The Required Actions are not required to be met during this time, unless the train is discovered inoperable during the testing. This allowance is based on the reliability analysis assumption of WCAP-10271-P-A (Ref. 7) that 4 hours is the average time required to perform channel surveillance.

ACTIONS (continued)

E.1, E.2.1, and E.2.2

Condition E applies to:

- ~~Containment Spray Containment Pressure-High High;~~
- Containment Phase B Isolation Containment Pressure-High High; and
- Steam Line Isolation Containment Pressure - High High.

Neither ~~None~~ of these signals has input to a control function. Thus, two-out-of-three logic is necessary to meet acceptable protective requirements. However, a two-out-of-three design would require tripping a failed channel. This is undesirable because a single failure would then cause spurious isolation ~~containment spray~~ initiation. ~~Spurious spray actuation is undesirable because of the cleanup problems presented.~~ Therefore, these channels are designed with two-out-of-four logic so that a failed channel may be bypassed rather than tripped. Note that one channel may be bypassed and still satisfy the single failure criterion. Furthermore, with one channel bypassed, a single instrumentation channel failure will not spuriously initiate isolation ~~containment spray~~.

To avoid the inadvertent actuation of ~~containment spray and~~ Phase B containment isolation, the inoperable channel should not be placed in the tripped condition. Instead it is bypassed. Restoring the channel to OPERABLE status, or placing the inoperable channel in the bypass condition within 6 hours, is sufficient to assure that the Function remains OPERABLE and minimizes the time that the Function may be in a partial trip condition (assuming the inoperable channel has failed high). The Completion Time is further justified based on the low probability of an event occurring during this interval. Failure to restore the inoperable channel to OPERABLE status, or place it in the bypassed condition within 6 hours, requires the unit be placed in MODE 3 within the following 6 hours and MODE 4 within the next 6 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems. In MODE 4, these Functions are no longer required OPERABLE.

The Required Actions are modified by a Note that allows one additional channel to be bypassed for up to 4 hours for surveillance testing. Placing a second channel in the bypass condition for up to 4 hours for testing purposes is acceptable based on the results of Reference 7.

SURVEILLANCE REQUIREMENTS (continued)

The setpoint shall be left set consistent with the assumptions of the setpoint methodology.

The Frequency of 92 days is justified in Reference 7.

SR 3.3.2.6

SR 3.3.2.6 is the performance of a SLAVE RELAY TEST. The SLAVE RELAY TEST is the energizing of the slave relays. Contact operation is verified in one of two ways. Actuation equipment that may be operated in the design mitigation MODE is either allowed to function, or is placed in a condition where the relay contact operation can be verified without operation of the equipment. Actuation equipment that may not be operated in the design mitigation MODE is prevented from operation by the SLAVE RELAY TEST circuit. For this latter case, contact operation is verified by a continuity check of the circuit containing the slave relay. This test is performed every 92 days. The Frequency is adequate, based on industry operating experience, considering instrument reliability and operating history data.

For slave relays or any auxiliary relays in the ESFAS circuit that are of the type Westinghouse AR or Potter & Brumfield MDR, the SLAVE RELAY TEST is performed every 18 months. This test frequency is based on the relay reliability assessments presented in References 10, 11, and 12. These reliability assessments are relay specific and apply only to the Westinghouse AR and Potter & Brumfield MDR type relays. SSPS slave relays or any auxiliary relays not addressed by Reference 10 do not qualify for extended surveillance intervals and will continue to be tested at a 92 day Frequency.

SR 3.3.2.7

SR 3.3.2.7 is the performance of a COT on the RWST level and Containment Pressure Control Start and Terminate Permissives.

conservative with respect to

A COT is performed on each required channel to ensure the entire channel will perform the intended Function. Setpoints must be found ~~within~~ the Allowable Values specified in Table ~~3.3.1-1~~ 3.3.2-1. This test is performed every 31 days. The Frequency is adequate, based on operating experience, considering instrument reliability and operating history data.

INSERT

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.2.8

SR 3.3.2.8 is the performance of a TADOT. This test is a check of the Manual Actuation Functions, AFW pump start on trip of all MFW pumps, AFW low suction pressure, Reactor Trip (P-4) Interlock, and Doghouse Water Level - High High Feedwater Isolation. It is performed every 18 months. Each Manual Actuation Function is tested up to, and including, the master relay coils. In some instances, the test includes actuation of the end device (i.e., pump starts, valve cycles, etc.). The Frequency is adequate, based on industry operating experience and is consistent with the typical refueling cycle. The SR is modified by a Note that excludes verification of setpoints during the TADOT for manual initiation Functions. The manual initiation Functions have no associated setpoints.

SR 3.3.2.9

SR 3.3.2.9 is the performance of a CHANNEL CALIBRATION.

A CHANNEL CALIBRATION is performed every 18 months, or approximately at every refueling. CHANNEL CALIBRATION is a complete check of the instrument loop, including the sensor. The test verifies that the channel responds to measured parameter within the necessary range and accuracy.

CHANNEL CALIBRATIONS must be performed consistent with the assumptions of the unit specific setpoint methodology.

The Frequency of 18 months is based on the assumption of an 18 month calibration interval in the determination of the magnitude of equipment drift in the setpoint methodology.

This SR is modified by a Note stating that this test should include verification that the time constants are adjusted to the prescribed values where applicable. The applicable time constants are shown in Table 3.3.2-1.

INSERT

SR 3.3.2.10

This SR ensures the individual channel ESF RESPONSE TIMES are less than or equal to the maximum values assumed in the accident analysis. Response Time testing acceptance criteria are included in the UFSAR (Ref. 2). Individual component response times are not modeled in the

LCO (continued)

There are three channels of Steam Line Pressure provided for each SG. Two channels per SG are required OPERABLE by the LCO.

19. Refueling Water Storage Tank Level

ECCS

RWST level monitoring is provided to ensure an adequate supply of water to the ~~safety injection and spray~~ pumps during the switchover to cold leg recirculation.

Four channels of RWST level are provided. Only two channels are required OPERABLE by the LCO.

20. Neutron Flux (Wide Range)

Wide Range Neutron Flux indication is provided to verify reactor shutdown.

Neutron flux is used for accident diagnosis, verification of subcriticality, and diagnosis of positive reactivity insertion.

Two channels of wide range neutron flux are required OPERABLE.

21. Steam Generator Water Level (Wide Range)

SG Water Level (Wide Range) is used to verify that the intact SGs are an adequate heat sink for the reactor. One channel per steam generator is required OPERABLE by the LCO. Diverse indication is provided by Steam Generator Water Level (Narrow Range).

APPLICABILITY

The PAM instrumentation LCO is applicable in MODES 1, 2, and 3. These variables are related to the diagnosis and pre-planned actions required to mitigate DBAs. The applicable DBAs are assumed to occur in MODES 1, 2, and 3. In MODES 4, 5, and 6, unit conditions are such that the likelihood of an event that would require PAM instrumentation is low; therefore, the PAM instrumentation is not required to be OPERABLE in these MODES.

B 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

B 3.5.4 Refueling Water Storage Tank (RWST)

BASES

BACKGROUND

The RWST supplies borated water to the Chemical and Volume Control System (CVCS) during abnormal operating conditions, to the refueling pool during refueling and makeup operations, and to the ECCS ~~and the Containment Spray System~~ during accident conditions.

The RWST supplies both trains of the ECCS ~~and the Containment Spray System~~ through separate supply headers during the injection phase of a loss of coolant accident (LOCA) recovery. A motor operated isolation valve is provided in each header to isolate the RWST once the system has been transferred to the recirculation mode. The recirculation mode is entered when pump suction is transferred to the containment sump following receipt of the RWST—Low Level signal. Use of a single RWST to supply both trains of the ECCS ~~and Containment Spray System~~ is acceptable since the RWST is a passive component, and since injection phase passive failures are not required to be assumed to occur coincidentally with Design Basis Events.

The switchover from normal operation to the injection phase of ECCS operation requires changing centrifugal charging pump suction from the CVCS volume control tank (VCT) to the RWST through the use of isolation valves.

During normal operation in MODES 1, 2, and 3, the safety injection (SI) and residual heat removal (RHR) pumps are aligned to take suction from the RWST.

The ECCS pumps are provided with recirculation lines that ensure each pump can maintain minimum flow requirements when operating at or near shutoff head conditions.

When the suction for the ECCS ~~and Containment Spray System~~ pumps is transferred to the containment sump, the RWST flow paths must be isolated to prevent a release of the containment sump contents to the RWST, which could result in a release of contaminants to the atmosphere and the eventual loss of suction head for the ECCS pumps.

This LCO ensures that:

a. The RWST contains sufficient borated water to support the ECCS during the injection phase;

BASES

BACKGROUND (continued)

b. Sufficient water volume exists in the containment sump to support continued operation of the ECCS and Containment Spray System pumps at the time of transfer to the recirculation mode of cooling; and

c. The reactor remains subcritical following a LOCA.

Insufficient water in the RWST could result in insufficient cooling capacity when the transfer to the recirculation mode occurs. Improper boron concentrations could result in a reduction of SDM or excessive boric acid precipitation in the core following the LOCA, as well as excessive caustic stress corrosion of mechanical components and systems inside the containment.

APPLICABLE SAFETY ANALYSES

During accident conditions, the RWST provides a source of borated water to the ECCS and Containment Spray System pumps. As such, it provides containment cooling and depressurization, core cooling, and replacement inventory and is a source of negative reactivity for reactor shutdown (Ref. 1). The design basis transients and applicable safety analyses concerning each of these systems are discussed in the Applicable Safety Analyses section of B 3.5.2, "ECCS—Operating"; B 3.5.3, "ECCS—Shutdown"; and B 3.6.6, "Containment Spray Systems." These analyses are used to assess changes to the RWST in order to evaluate their effects in relation to the acceptance limits in the analyses.

The RWST must also meet volume, boron concentration, and temperature requirements for non-LOCA events. The volume is not an explicit assumption in non-LOCA events since the required volume is a small fraction of the available volume. The deliverable volume limit is set by the LOCA and containment analyses. For the RWST, the deliverable volume is different from the total volume contained due to the location of the piping connection. The ECCS water boron concentration is an explicit assumption in the main steam line break (MSLB) analysis to ensure the required shutdown capability. This assumption is important in ensuring the required shutdown capability. Although the maximum temperature is a conservative assumption in the feedwater line break analysis, SI termination occurs very quickly in this analysis and long before significant RCS heatup occurs. The minimum temperature is an assumption in the MSLB actuation analyses.

377,537

For a large break LOCA analysis, the RWST level setpoint equivalent to the minimum water volume limit of 363,513 gallons and the lower boron concentration limits listed in the COLR are used to compute the post

BASES

APPLICABLE SAFETY ANALYSES (continued)

The RWST temperature limits were originally established with containment spray aligned to the RWST and were not revised when the containment Spray System became a manually actuated system with the initial suction source changed to the containment sump. The RWST temperature limits are contained within additional analyses and remain valid although the basis is historical and no longer relevant. The following paragraph is retained for historical purposes only.

LOCA sump boron concentration necessary to assure subcriticality with all rods in, minus the highest worth rod out (ARI N-1). The large cold leg break LOCA is the limiting case since boron accumulation in the core will be maximized during the cold leg recirculation phase due to core boiling. The accumulation of boron in the core prevents the boron from returning to the sump, which leads to a boron diluted sump condition that may cause the core to become re-critical when switching over to hot leg recirculation. For the post LOCA injection phase, each reload cycle is verified to have all rods out (ARO) critical boron concentrations less than the minimum allowed RWST boron concentration.

The upper limit on boron concentration as listed in the COLR is used to determine the maximum allowable time to switch to hot leg recirculation following a LOCA. The purpose of switching from cold leg to hot leg injection is to avoid boron precipitation in the core following the accident. In addition, this upper limit ensures that the equilibrium pH of the solution in the containment sump following the design basis LOCA is at least 7.5.

was could

In the ECCS analysis, the containment spray temperature is assumed to be equal to the RWST lower temperature limit of 70°F. If the lower temperature limit is violated, the containment spray further reduces containment pressure, which decreases the saturated steam specific volume. This means that each pound of steam generated during core reflood tends to occupy a larger volume, which decreases the rate at which steam can be vented out the break and increases peak clad temperature. The upper temperature limit of 100°F, plus an allowance for temperature measurement uncertainty, is used in the containment OPERABILITY analysis. Exceeding this temperature will result in higher containment pressures due to reduced containment spray cooling capacity. For the containment response following an MSLB, the lower limit on boron concentration and the upper limit on RWST water temperature are used to maximize the total energy release to containment.

The RWST satisfies Criterion 3 of 10 CFR 50.36 (Ref. 2).

LCO

The RWST ensures that an adequate supply of borated water is available to cool and depressurize the containment in the event of a Design Basis Accident (DBA), to cool and cover the core in the event of a LOCA, to maintain the reactor subcritical following a DBA, and to ensure adequate level in the containment sump to support ECCS and Containment Spray System pump operation in the recirculation mode.

BASES

LCO (continued)

To be considered OPERABLE, the RWST must meet the water volume, boron concentration, and temperature limits established in the SRs.

APPLICABILITY

In MODES 1, 2, 3, and 4, RWST OPERABILITY requirements are dictated by ECCS ~~and Containment Spray System~~ OPERABILITY requirements. Since ~~both~~ the ECCS ~~and the Containment Spray System~~ must be OPERABLE in MODES 1, 2, 3, and 4, the RWST must also be OPERABLE to support their operation. Core cooling requirements in MODE 5 are addressed by LCO 3.4.7, "RCS Loops—MODE 5, Loops Filled," and LCO 3.4.8, "RCS Loops—MODE 5, Loops Not Filled." MODE 6 core cooling requirements are addressed by LCO 3.9.4, "Residual Heat Removal (RHR) and Coolant Circulation—High Water Level," and LCO 3.9.5, "Residual Heat Removal (RHR) and Coolant Circulation—Low Water Level."

ACTIONS

A.1

With RWST boron concentration or borated water temperature not within limits, they must be returned to within limits within 8 hours. Under these conditions neither the ECCS nor the Containment Spray System can perform its design function. Therefore, prompt action must be taken to restore the tank to OPERABLE condition. The 8 hour limit to restore the RWST temperature or boron concentration to within limits was developed considering the time required to change either the boron concentration or temperature and the fact that the contents of the tank are still available for injection.

B.1

With the RWST inoperable for reasons other than Condition A (e.g., water volume), it must be restored to OPERABLE status within 1 hour.

In this Condition, ~~neither~~ the ECCS ~~nor the Containment Spray System~~
cannot ~~can~~ perform its design function. Therefore, prompt action must be taken to restore the tank to OPERABLE status or to place the plant in a MODE in which the RWST is not required. The short time limit of 1 hour to restore the RWST to OPERABLE status is based on this condition simultaneously affecting redundant trains.

BASES

ACTIONS (continued)

C.1 and C.2

If the RWST cannot be returned to OPERABLE status within the associated Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.5.4.1

The RWST borated water temperature should be verified every 24 hours to be within the limits assumed in the accident analyses band. This Frequency is sufficient to identify a temperature change that would approach either limit and has been shown to be acceptable through operating experience.

SR 3.5.4.2

plus instrument uncertainty

The RWST water volume should be verified every 7 days to be above the required minimum level in order to ensure that a sufficient initial supply is available for injection and to support continued ECCS and Containment Spray System pump operation on recirculation. Since the RWST volume is normally stable and is protected by an alarm, a 7 day Frequency is appropriate and has been shown to be acceptable through operating experience.

SR 3.5.4.3

The boron concentration of the RWST should be verified every 7 days to be within the required limits. This SR ensures that the reactor will remain subcritical following a LOCA and that the boron content assumed for the injection water in the MSLB analysis is available. Further, it assures that the resulting sump pH will be maintained in an acceptable range so that boron precipitation in the core will not occur and the effect of chloride and caustic stress corrosion on mechanical systems and components will be minimized. Since the RWST volume is normally stable, a 7 day sampling Frequency to verify boron concentration is appropriate and has been shown to be acceptable through operating experience.

B 3.6 CONTAINMENT SYSTEMS

B 3.6.6 Containment Spray System

BASES

BACKGROUND

The Containment Spray System provides containment atmosphere cooling to limit post accident pressure and temperature in containment to less than the design values. Reduction of containment pressure and the iodine removal capability of the spray reduce the release of fission product radioactivity from containment to the environment, in the event of a Design Basis Accident (DBA). The Containment Spray System is designed to meet the requirements of 10 CFR 50, Appendix A, GDC 38, "Containment Heat Removal," GDC 39, "Inspection of Containment Heat Removal Systems," GDC 40, "Testing of Containment Heat Removal Systems," GDC 41, "Containment Atmosphere Cleanup," GDC 42, "Inspection of Containment Atmosphere Cleanup Systems," and GDC 43, "Testing of Containment Atmosphere Cleanup Systems" (Ref. 1).

The containment spray pumps are started manually with the pump suction aligned to the containment sump once the ECCS is aligned to the recirculation mode of operation.

The Containment Spray System consists of two separate trains of equal capacity, each capable of meeting the system design basis spray coverage. Each train includes a containment spray pump, one containment spray heat exchanger, spray headers, nozzles, valves, and piping. Each train is powered from a separate Engineered Safety Feature (ESF) bus. ~~The refueling water storage tank (RWST) supplies borated water to the Containment Spray System during the injection phase of operation. In the recirculation mode of operation, containment spray pump suction is transferred from the RWST to the containment recirculation sump(s).~~

The diversion of a portion of the recirculation flow from each train of the Residual Heat Removal (RHR) System to additional redundant spray headers completes the Containment Spray System heat removal capability. Each RHR train is capable of supplying spray coverage, if ~~required~~ desired to supplement the Containment Spray System.

The Containment Spray System ~~and RHR System~~ provide(s) a spray of cold or subcooled borated water into the upper containment volume to limit the containment pressure and temperature during a DBA. ~~The RWST solution temperature is an important factor in determining the heat removal capability of the Containment Spray System during the~~

BASES

BACKGROUND (continued)

injection phase. In the recirculation mode of operation, heat is removed from the containment sump water by the Containment Spray System and RHR heat exchangers. Each train of the Containment Spray System, supplemented by a train of RHR spray, provides adequate spray coverage to meet the system design requirements for containment heat removal.

For the hypothetical double-ended rupture of a Reactor Coolant System pipe, the pH of the sump solution (and, consequently, the spray solution) is raised to at least 8.0 within one hour of the onset of the LOCA. The resultant pH of the sump solution is based on the mixing of the RCS fluids, ECCS injection fluid, and the melted ice which are combined in the sump. The alkaline pH of the containment sump water minimizes the evolution of iodine and the occurrence of chloride and caustic stress corrosion on mechanical systems and components exposed to the fluid.

by actuation of separate train related switches on the main control board. Once a train is started,

The Containment Spray System is actuated either automatically by a containment pressure high-high signal or manually. An automatic actuation opens the containment spray pump discharge valves, starts the two containment spray pumps, and begins the injection phase. A manual actuation of the Containment Spray System requires the operator to actuate two separate train related switches on the main control board to begin the same sequence of two train actuation. The injection phase continues until an RWST level Low-Low alarm is received. The Low-Low alarm for the RWST signals the operator to manually align the system to the recirculation mode. The Containment Spray System in the recirculation mode maintains an equilibrium temperature between the containment atmosphere and the recalculated sump water. Operation of the Containment Spray System in the recirculation mode is controlled by the operator in accordance with the emergency operation procedures.

and

ECCS

train related

recirculated

The Containment Spray System is not actuated until

The RHR spray operation is initiated manually, when required by the emergency operating procedures, after the Emergency Core Cooling System (ECCS) is operating in the recirculation mode. The RHR sprays are available to supplement the Containment Spray System, if required, in limiting containment pressure. This additional spray capacity would typically be used after the ice bed has been depleted and in the event that containment pressure rises above a predetermined limit. The Containment Spray System is an ESF system. It is designed to ensure that the heat removal capability required during the post accident period can be attained.

desired

and the ECCS has been realigned in the recirculation mode of operation

and to manually initiate the Containment Spray System

BASES

BACKGROUND (continued)

The operation of the Containment Spray System, together with the ice condenser, is adequate to assure pressure suppression subsequent to the initial blowdown of steam and water from a DBA. During the post blowdown period, the Air Return System (ARS) is automatically started. The ARS returns upper compartment air through the divider barrier to the lower compartment. This serves to equalize pressures in containment and to continue circulating heated air and steam through the ice condenser, where heat is removed by the remaining ice.

The Containment Spray System limits the temperature and pressure that could be expected following a DBA. Protection of containment integrity limits leakage of fission product radioactivity from containment to the environment.

APPLICABLE SAFETY ANALYSES

The limiting DBAs considered relative to containment OPERABILITY are the loss of coolant accident (LOCA) and the steam line break (SLB). The DBA LOCA and SLB are analyzed using computer codes designed to predict the resultant containment pressure and temperature transients. No two DBAs are assumed to occur simultaneously or consecutively. The postulated DBAs are analyzed, in regard to containment ESF systems, assuming the loss of one ESF bus, which is the worst case single active failure, resulting in one train of the Containment Spray System, the RHR System, and the ARS being rendered inoperable (Ref. 2).

The DBA analyses show that the maximum peak containment pressure results from the LOCA analysis, and is calculated to be less than the containment design pressure. The maximum peak containment atmosphere temperature results from the SLB analysis and was calculated to be within the containment environmental qualification temperature during the DBA SLB. The basis of the containment environmental qualification temperature is to ensure the OPERABILITY of safety related equipment inside containment (Ref. 3).

reaching the RWST low level setpoint prior to

the

modeled in

The ~~modeled~~ Containment Spray System actuation ~~from~~ the containment analysis is based on ~~a response~~ time associated with ~~exceeding the containment pressure high-high signal setpoint to~~ achieving full flow through the containment spray nozzles. A delayed response time initiation provides conservative analyses of peak calculated containment temperature and pressure responses. The Containment Spray System total response time ~~of 45 seconds~~ is composed of ~~signal delay, diesel generator startup,~~ and system startup time.

operator action delay

BASES

APPLICABLE SAFETY ANALYSES (continued)

For certain aspects of transient accident analyses, maximizing the calculated containment pressure is not conservative. In particular, the ECCS cooling effectiveness during the core reflood phase of a LOCA analysis increases with increasing containment backpressure. For these calculations, the containment backpressure is calculated in a manner designed to conservatively minimize, rather than maximize, the calculated transient containment pressures in accordance with 10 CFR 50, Appendix K (Ref. 4).

Inadvertent actuation is precluded by a design feature consisting of an additional set of containment pressure sensors which prevents operation when the containment pressure is below the containment pressure control system permissive.

The Containment Spray System satisfies Criterion 3 of 10 CFR 50.36 (Ref. 5).

LCO

During a DBA, one train of Containment Spray System is required to provide the heat removal capability assumed in the safety analyses. To ensure that this requirement is met, two containment spray trains must be OPERABLE with power from two safety related, independent power supplies. Therefore, in the event of an accident, at least one train operates.

being manually initiated to take suction from the containment sump and delivering it to the containment spray rings.

Each Containment Spray System includes a spray pump, headers, valves, heat exchangers, nozzles, piping, instruments, and controls to ensure an OPERABLE flow path capable of ~~taking suction from the RWST upon an ESF actuation signal and manually transferring suction to the containment sump.~~

APPLICABILITY

In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material to containment and an increase in containment pressure and temperature requiring the operation of the Containment Spray System.

In MODES 5 and 6, the probability and consequences of these events are reduced because of the pressure and temperature limitations of these MODES. Thus, the Containment Spray System is not required to be OPERABLE in MODE 5 or 6.

BASES

SURVEILLANCE REQUIREMENTS (continued)

ensures that spray pump performance has not degraded during the cycle. Flow and differential head are normal tests of centrifugal pump performance required by Section XI of the ASME Code (Ref. 6). Since the containment spray pumps cannot be tested with flow through the spray headers, they are tested on bypass flow. This test confirms one point on the pump design curve and is indicative of overall performance. Such inservice inspections confirm component OPERABILITY, trend performance, and detect incipient failures by indicating abnormal performance. The Frequency of this SR is in accordance with the Inservice Testing Program.

SR 3.6.6.3 ~~and SR 3.6.6.4~~

This SR

~~These SRs~~ require 5 verification that each automatic containment spray valve actuates to its correct position ~~and each containment spray pump starts upon receipt of an actual or simulated Containment Phase B Isolation signal~~. This Surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls. The 18 month Frequency is based on the need to perform ~~these~~ Surveillance~~s~~ under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance~~s~~ were performed with the reactor at power. Operating experience has shown these components usually pass the Surveillance~~s~~ when performed at the 18 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

upon manual containment spray initiation

this

The surveillance of containment sump isolation valves is also required by SR 3.6.6.3. A single surveillance may be used to satisfy both requirements.

SR 3.6.6.4 Not used.

SR 3.6.6.5 and SR 3.6.6.6

These SRs require verification of proper interaction between the CPCS system and the Containment Spray System.

SR 3.6.6.5 deals solely with the containment spray pumps. It must be shown through testing that: (1) the containment spray pumps are prevented from starting in the absence of a CPCS permissive, (2) the containment spray pumps start when given a CPCS permissive, and (3) when running, the containment spray pumps stop when the CPCS permissive is removed. The "inhibit", "permit", and "terminate" parts of the CPCS interface with the containment spray pumps are verified by

Attachment 3

NRC Commitments

The following NRC commitments are being made in support of this amendment request:

1. The approved amendments will be implemented within 60 days from the date of NRC approval. "Implemented" means that the approved amendments will have been placed into the control room copies of the TS. However, the provisions afforded by the approved amendments will not actually be utilized until such time that the associated plant modifications are in place.

2. Prior to actually utilizing the provisions afforded by the approved amendments, Catawba will have in place all required design, document, and process changes necessary to support these provisions.

3. Within 180 days of the implementation of the associated modifications for the final unit, Catawba will submit a follow-up administrative license amendment request to delete the superceded TS and Bases requirements.

Overview of proposed changes

- Change table of contents (pages vii and viii)
- Change number of References section from 3.6 to 3.7
- Insert new section 3.6 to describe manual containment spray actuation analyses

3.6 Manual Containment Spray Actuation

In order to optimize the LOCA containment response, the removal of the automatic actuation logic that initiates containment spray flow on high containment pressure has been considered. To support the removal of the containment spray automatic actuation logic, several changes to the ECCS alignment are required. These realignments and associated operator actions involve station and procedural modifications. This section described the modeling changes required to perform mass and energy release calculations with the containment spray automatic actuation logic removed.

3.6.1 ECCS Alignment Changes

To support the removal of the containment spray automatic actuation logic, several changes to the ECCS alignment are required. These changes are described below.

- Removal of the containment spray automatic actuation logic precludes containment spray flow until operator action is taken
- Operator action to align safety injection and centrifugal charging pump suction to RHR pump discharge is delayed from RWST low level until RWST low-low level
- Operator action to align RHR pump to auxiliary containment spray header is changed from 50 minutes to be based upon a containment pressure setpoint plus a delay for operator action and the RHR pumps taking suction from the sump

Overview of ECCS operation

Following a large break LOCA, the RHR, safety injection and centrifugal charging pumps will automatically actuate per the plant TS to mitigate the event. Containment spray will not automatically actuate. The operating ECCS pumps will deplete the RWST to the low level setpoint. At RWST low level, the RHR pumps will automatically transfer from taking suction from the RWST to taking suction from the containment sump. The safety injection and centrifugal charging pumps will continue depleting the RWST until the low-low level setpoint is reached. Operator action is then taken to align the safety injection and centrifugal charging pump suction to the discharge of the RHR pump.

Operator action is taken to initiate containment spray flow taking suction from the containment sump after reaching RWST low level. For most maximum and minimum safeguards scenarios, the operator will initiate containment spray flow before reaching the RWST low-low level alarm. For scenarios that include a single failure affecting the automatic transfer of RHR suction to the sump, the operator will initiate containment spray flow after reaching the RWST low-low level alarm.

For the analyses that support the removal of the containment spray automatic actuation logic, the initiation of RHR auxiliary spray is affected. The original plant licensing basis analyses aligned RHR flow to the auxiliary spray header based on elapsed time (50 minutes). In the revised analyses, RHR auxiliary spray is assumed to be aligned based on reaching a containment pressure setpoint plus a time delay to allow for operator action.

3.6.2 RELAP5 Analysis Model

The analyses to support the removal of the containment spray automatic actuation logic include a model change not previously described in this report. The model change is the inclusion of non-condensable gas (assumed to be nitrogen) in the containment atmosphere to address the reverse break flow phenomenon described in Section 3.6.3.1, below. The containment boundary condition described in Section 3.3 is typically specified as saturated steam.

There are two approaches taken to including nitrogen in the mass and energy release calculation. The first approach, and preferred option, is to specify nitrogen as part of the containment boundary volume conditions. The second approach is to inject nitrogen in a node upstream of the break based upon a pressure differential indicating that reverse break flow is occurring. For both approaches, a conservative boundary volume temperature is assumed.

3.6.3 Manual Containment Spray Actuation Analysis Results

3.6.3.1 Discussion of Phenomena

The phenomena observed in the analyses to support the removal of the containment spray automatic actuation logic are generally the same as those previously described.

The modeling approach presented in Section 3.3 has on occasion resulted in a reverse break flow. This is steam predicted to be drawn from containment into the RCS by steam condensed on subcooled ECCS injection. Since the containment boundary volume is generally assumed to be saturated steam, reverse break flow results in additional steam being added to the RCS which adds to the mass and energy release calculation. Mass and energy is not conserved due to reverse break steam flow for the containment response methodology, as this flow in the RELAP5 mass and energy release calculation is not subtracted from the associated GOTHIC calculations. Generally, reverse break flow is an embedded conservatism in the mass and energy release calculation as it artificially increases the mass and energy release. However, it is desirable to include modeling to mitigate the effects of reverse break steam flow when the magnitude is significant. The inclusion of nitrogen in the containment atmosphere will reduce the associated reverse break flow by reducing the steam condensation rate on the subcooled ECCS injection, which ultimately causes the reverse break flow. With a delayed alignment of RHR auxiliary spray, RHR flow to the cold legs continues for an extended period of time, leading to a significant increase in reverse break steam flow. The inclusion of nitrogen is used to conservatively mitigate this phenomenon and reduce the magnitude of the reverse break flow penalty.

A second phenomenon not previously encountered is the refilling of the intact reactor coolant loop pump seals for cold leg breaks. This phenomenon has been previously observed for hot leg breaks and is not consequential to the analytical results. For cold leg breaks, this phenomenon has the potential to directly affect the peak containment pressure response. In the results presented in Section 3.4, RHR pump flow is realigned from the cold legs and core cooling, to the auxiliary containment spray header at 50 minutes. The ECCS injection flow after 50 minutes is insufficient to cause the intact loop seal to refill. The analysis supporting the removal of the containment spray automatic actuation logic keeps RHR flow aligned to the cold legs if containment pressure remains below the setpoint for aligning auxiliary containment spray. In these analyses, refilling of the intact loop seals is observed. For a cold leg break, the core steaming rate is proportional to the decay heat, which decreases with time. A fraction of the steam generated in the core is drawn through the intact loops by steam condensed on subcooled ECCS injection. Eventually the steam velocity in the intact legs decreases to the point where liquid spills over the reactor coolant pump weirs and refills the intact loop seals. After this point in time, most of the steam generated in the core exits the broken loop into containment. The increased steaming rate to containment affects the resulting containment pressure, potentially affecting the peak containment pressure result.