David A. Heacock
Site Vice President

Dominion Generation
North Anna Power Station
P.O. Box 402, Mineral, VA 23117
Phone: 540-894-2101, Fax: 540-894-2878
David_Heacock@dom.com

Dominion

December 19, 2001

Mr. J. W. Gotzmer
Federal Energy Regulatory Commission
Atlanta Regional Office - Parkridge 85 North Building
3125 Presidential Parkway, Suite 300
Atlanta, Georgia 30340

Serial No. F01-014

Dear Mr. Gotzmer:

PROJECT NO. 6335-VA / NATDAM NO. VA83005
NORTH ANNA HYDROELECTRIC PROJECT

We received your letter dated December 12, 2001, requesting a copy of the "North Anna Tainter Gate Calculation, Calculation No. 26011A-CS-1," dated June 2001. Enclosed is a copy of this calculation.

Should you have any questions concerning this submittal, please contact Mr. Michael P. Whalen at (540) 894-2572.

Very truly yours,

David A. Heacock
Site Vice President

Commitments made in this letter: None

Attachment

cc: U. S. Nuclear Regulatory Commission
Region II
Atlanta Federal Center
61 Forsyth Street, SW, Suite 23T85
Atlanta, Georgia 30303
w/o attachment

Mr. M. J. Morgan w/o attachment
NRC Senior Resident Inspector
North Anna Power Station

Duke proposes to clearly define the current licensing basis associated with non-Category I piping as a through-wall leakage crack for the postulated piping failure. The change is consistent with Standard Review Plan (SRP) Branch Technical Position (BTP) Auxiliary Systems Branch (ASB) 3-1 guidelines that a limited-size through-wall leakage crack is the appropriate postulated piping failure for moderate energy systems. The Fire Protection System and ventilation cooling water system, which were identified as the two sources of potential flooding for the Auxiliary Building in Duke's response (October 24, 1972 submittal) to the September 26, 1972 NRC letter, are moderate energy systems.

Attachment 1 provides the re-typed Oconee UFSAR pages. Attachment 2 provides a mark-up of the affected Oconee UFSAR pages. Duke Energy's evaluation to support the change to the licensing basis for Oconee Nuclear Station is provided in Attachment 3. Attachment 4 documents the determination that the amendment contains No Significant Hazards Considerations pursuant to 10 CFR 50.92. Attachment 5 provides the basis for the categorical exclusion from performing an Environmental Assessment/Impact Statement pursuant to 10 CFR 51.22(c)(9).

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

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

Inquiries on this matter should be directed to Boyd Shingleton at (864) 885-4716.

Very truly yours,

W. R. McCollum, Jr., Site Vice President
Oconee Nuclear Site

Attachments

xc w/attachments:

L. N. Olshan
NRC Senior Project Manager (ONS)
U. S. Nuclear Regulatory Commission
Mail Stop O-14H25
Washington, DC 20555-0001

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

M. C. Shannon
Senior Resident Inspector (ONS)
U. S. Nuclear Regulatory Commission
Oconee Nuclear Site

V. R. Autry, Director
Division of Radioactive Waste Management
Bureau of Land & Waste Management
Department of Health & Environmental Control
2600 Bull Street
Columbia, SC 29201

AFFIDAVIT

W. R. McCollum, Jr., being duly sworn, states that he is Site Vice President of Duke Energy Corporation; that he is authorized on the part of said corporation to sign and file with the Nuclear Regulatory Commission this revision to the Oconee Nuclear Station License Nos. DPR-38, DPR-47, and DPR-55; and that all statements and matters set forth therein are true and correct to the best of his knowledge.

W. R. McCollum, Jr., Site Vice President

Subscribed and sworn to me: 12-20-01
Date

Notary Public: Connie M. Dreazale

My Commission Expires: 2/12/03
Date

SEAL

ATTACHMENT 1

RETYPED OCONEE UFSAR PAGES

A push button in each control room provides capability to close the Condenser Circulating Water (CCW) pump discharge valves to protect against CCW siphoning into the turbine building basement. This flood mitigation station modification has been installed pursuant to the recommendations made in the Oconee Probabilistic Risk Assessment Study.

Valve alignments on the CCW side of the condensate coolers for all three units are designed to limit the backflow from the CCW system during a flood. One condensate cooler on each unit is normally isolated, using manual block valves, to limit backflow from the CCW outlet during a flood. It is desirable, however, to allow a limited amount of backflow during a flood to provide suction for Low Pressure Service Water (LPSW) pumps and the Standby Shutdown Facility Auxiliary Service Water (SSF ASW) pump. Temperature control valves 2CCW-84 and 3CCW-84 have had their air supplies disconnected, effectively failing them in the open position (See Figure 9-9). Operating conditions will sometimes require restoring one or more of the isolated condensate coolers to service. The Probabilistic Risk Assessment study recognized this requirement; the assumption was that the three coolers in question would be isolated ten months out of each year of reactor operation (See Section 3.4.2, reference 4).

For the Auxiliary Building, Duke evaluated the effects of flooding caused by a leak from non-seismic moderate energy piping. A crack size calculated using the Standard Review Plan (SRP) Section 3.6.1, Branch Technical Position (BTP) Auxiliary Systems Branch (ASB) 3-1 guidelines was used. This evaluation concluded that for the bounding case (a crack in the 16-inch HPSW pipe) the effects of flooding can be mitigated without adversely affecting safety-related equipment required for safe shutdown (MODE 3 with average RCS temperature ≥ 525°F). The analysis demonstrates that at least an hour is available from detection for operator action to isolate the flood. The operator would be alerted to a flood based on the level indicating alarms from the Auxiliary Building Waste Tanks (See Section 3.4.2, reference 5).

3.4.1.1.2 Flood Protection Measures Inside Containment

The primary means for detecting leakage in the Reactor Building is the level indication for the normal sump. This indication has a range of 0-to-30 inches, with a statalarm occurring at 15 inches increasing level and a computer alarm at approximately 22 inches. These alarms would alert the operators in the control room such that appropriate actions could be taken. In addition to the alarms, sump level is input to the plant computer and is logged to the alarm log. Level is also recorded on a trend recorder in each control room. Safety related redundant level transmitters with a range of 3 inches to 24 inches are also provided in the normal sump. Both transmitter levels are indicated in the control room on receiver gauges and one train is recorded. Thus, the operators have several methods for monitoring changes in sump level.

The sump fill rate is routinely measured to determine leakage rate. The sump capacity is 15 gallons per inch of height and each graduation on the indicator level indicates 1.5 gallons of leakage into the sump. A 1 gal/min leak would therefore be detectable within less than 10 minutes.

In addition to the normal sump level, indication of the emergency sump level is also provided by redundant safety related systems with a range of 0 to 3 feet. Both trains of instrumentation are indicated on receiver gauges in the control room and one train is recorded. This indication can be used in conjunction with the normal sump level indication to detect abnormal leakage in the Reactor Building. Two additional trains of containment level transmitters are installed in each Reactor Building to provide wide range level indication and recording with a range of 0 to 15 feet.

The normal sump is routinely pumped to the miscellaneous waste holdup tanks whenever the alarm point (15 inches) is reached. Pumping of the sump water is started manually, but terminates automatically when the sump level has dropped to 6 inches (which clears the statalarm). Each time the sump is flow from the normal sump. The flow rate from the sump can be determined using the rate of change in sump level.

In order to provide periodic monitoring of sump levels, the recording of normal and emergency sump levels is done daily. Daily monitoring of level indications is useful in confirming that level instrumentation are operable, while verifying the sump pumps are operable and maintaining the sump level at or below the alarm point. Calibration of the normal and emergency sump indications is performed during refueling.

In the event of increased leakage to the Reactor Building, sampling may be performed to determine the origin of the leakage (e.g., LPSW, feedwater, component cooling, or RC system).

Leakage from the LPSW system in containment can also be detected by the monitoring of other parameters. For example, the inlet and outlet LPSW flows for each Reactor Building Cooling Unit (RBCU) are monitored for any differences which could be indicative of a cooler leak. If a flow difference is detected, an alarm is provided to the control room. The operator can then promptly isolate the affected cooler by closing remote operated valves.

The Reactor Coolant Pump (RCP) motor parameters are also continuously monitored. A leak in the motor stator winding cooler would be alarmed in the control room. A leak in either of the motor bearing oil coolers could be detected by changing motor temperature in conjunction with increasing sump level. The pump could then be stopped and the cooling water isolated from the control room.

The component cooling system is designed to provide cooling water for various inside containment components. In-leakage of reactor coolant is detected by a radiation monitor and an increase in surge tank level which will be annunciated. Out-leakage from the system will result in a decreasing surge tank level which is annunciated. Volume of the surge tank is 50 ft^3 and allows relatively small volumes of in-leakage or out-leakage to be observed.

3.4.2 References

1. Elevations taken from Figure 2-2 of FSAR and Oconee FSAR 2.2.6.
2. Response to Question of Effects of Failure of Non-Category I Equipment, Oconee FSAR, Supplement 13 of January 29, 1973, Item No. 7347. Information received from Steam Department.
3. Response to Bulletin 80-24 on Cooling Systems Inside Containment, Attachment to Mr. W. O. Parker, Jr.'s letter of January 6, 1981, Item No. 760. Information received from Steam Department.
4. Letter From Hal B. Tucker (Duke) to Harold R. Denton (NRC) dated April 28, 1986.
5. NRC Safety Evaluation dated xx/xx/xx.

THIS IS THE LAST PAGE OF THE TEXT SECTION 3.4.

only a single pipe break during a seismic event, and the analysis shall determine the effect on the safety-related portion of the system from the most limiting single pipe break. For moderate energy systems a crack is postulated as defined by Standard Review Plan (SRP) Section 3.6.1, Branch Technical Position (BTP) Auxiliary Systems Branch (ASB) 3-1 (See Section 3.7.5, reference 5).

The seismic/non-seismic boundary valve is protected from seismic effects by restraining or anchoring the non-seismic portion of the system downstream of the valve.

3.7.3.10 Seismic Analysis of Reactor Internals

The core support structure is designed as a Class 1 structure, as defined in Section 3.2 to resist the effects of seismic disturbances. The basic design guide for the seismic analysis is AEC publication TID-7024, "Nuclear Reactors and Earthquakes."

Lateral deflection and torsional rotation of the lower end of the core support assembly is limited in order to prevent excessive deformation resulting from seismic disturbance thereby assuring insertion of control rod assemblies (CRA). Core drop in the event of failure of the normal supports is limited by guide lugs so that the CRA do not disengage from the fuel assembly guide tubes. Additional information on design of the Reactor Internals is included in Section 3.9.2.

3.7.3.11 Analysis Procedures for Damping

A 0.5 percent critical damping value is used for vital piping analysis (see Section 3.7.1.3).

3.7.4 Seismic Instrumentation Program

3.7.4.1 Location and Description of Instrumentation

Earthquake instrumentation being provided is a strong motion accelerograph designated SMA-3 and manufactured by Kinemetrics, Inc., of Pasadena, CA. This system consists of a central recording system, control panel, one TS-3 triaxial seismic trigger package, and two force-balance triaxial accelerometer packages.

The operations sequence is as follows:

> The seismic trigger senses the initial earthquake ground motion with a normal setting of 0.01g and actuates the SMA-3 to full operation in less than 0.1 second.
>
> The SMA-3 operates for as long as the trigger detects the earthquake, plus an additional 10 seconds.
>
> The accelerograph can thus record a single earthquake or a sequence of earthquakes and aftershocks lasting as long as 30 minutes.

The output of each triaxial sensor is recorded using frequency modulation on a single four track cassette tape. Three of the tracks on the tape are the data tracks; the fourth is a reference track used for tape speed and amplitude compensation.

The Seismic Trigger and one Force Balance accelerometer of the SMA-3 system are located in the Unit 1 Tendon Gallery. Also, a second Force Balance accelerometer is located directly above at elevation 797' + 6" in the Oconee 1 Reactor Building. The recorder for the system is located in the Unit 1 Cable Room.

Also, a seismic trigger/switch is located in the Unit 1 tendon gallery. The Kinemetrics Model TS-3A has a preset acceleration threshold of 0.05g which activates the statalarm in Units 1 and 3 control rooms, when design conditions occur.

Six 2g peak recording accelorometers, manufactured by Engdahl-Model PAR 400, are also installed at various locations within the Oconee 1 Reactor Building. The instruments will provide post-seismic data for the following locations or items:

1. Adjacent to the strong motion accelerograph located in Tendon Access Gallery.
2. Support of the pressurizer vessel.
3. Support of Core Flood Tank 1A.
4. Main steam line pipe hanger.
5. Feedwater line pipe hanger.
6. Core flood injection line pipe hanger.

The major Class 1 structures, Reactor Building and Auxiliary Buildings, will be founded on a common rock foundation and will have similar base motions. The dynamic structural properties and responses of these structures are generated using similar assumptions and analytical techniques. Therefore, the response of these structures can be determined based upon the instrumentation in one structure.

Top of soil (free field) responses will not provide useful analytical data for the evaluation of major Class 1 structures founded on rock. Therefore, it is felt that free field instrumentation will not contribute to the evaluation of these structures.

3.7.4.2 Comparison of Measured and Predicted Responses

In the event of an earthquake, the data will be analyzed to determine the magnitude of the earthquake. If the design earthquake is exceeded, the units would be shut down and structures, systems, and equipment thoroughly investigated. Responses from instruments located on selected structures, systems and components will be compared to calculated responses for those structures, systems and components at the respective location when subjected to the same base response.

The recorded seismic data will be used for comparison and verification of seismic analysis assumptions, damping characteristics, and the analytical model used for the plant seismic design.

3.7.5 References

1. Bechtel Report, "Seismic Analysis Auxiliary Building", January, 1970.
2. Duke Power Engineering Design Report, "Static Method of Seismic Analysis of Piping Systems for Oconee 1, 2 and 3", File OS-27-B, June 6, 1970.
3. AEC Report TID-7024, "Nuclear Reactors and Earthquakes".
4. Newmark, N. M., "Torsion in Symmetrical Buildings".
5. NRC Safety Evaluation dated xx/xx/xx.

THIS IS THE LAST PAGE OF THE TEXT SECTION 3.7.

ATTACHMENT 2

MARK-UP OF OCONEE UFSAR PAGES

Insert for UFSAR Section 3.4.1.1.1

A push button in each control room provides capability to close the Condenser Circulating Water (CCW) pump discharge valves to protect against CCW siphoning into the turbine building basement. This flood mitigation station modification has been installed pursuant to the recommendations made in the Oconee Probabilistic Risk Assessment Study.

Valve alignments on the CCW side of the condensate coolers for all three units are designed to limit the backflow from the CCW system during a flood. One condensate cooler on each unit is normally isolated, using manual block valves, to limit backflow from the CCW outlet during a flood. It is desirable, however, to allow a limited amount of backflow during a flood to provide suction for Low Pressure Service Water (LPSW) pumps and the Standby Shutdown Facility Auxiliary Service Water (SSF ASW) pump. Temperature control valves 2CCW-84 and 3CCW-84 have had their air supplies disconnected, effectively failing them in the open position (See Figure 9-9). Operating conditions will sometimes require restoring one or more of the isolated condensate coolers to service. The Probabilistic Risk Assessment study recognized this requirement; the assumption was that the three coolers in question would be isolated ten months out of each year of reactor operation (See Section 3.4.2, reference 4).

3.4.1.1.2 Flood Protection Measures Inside Containment

The primary means for detecting leakage in the Reactor Building is the level indication for the normal sump. This indication has a range of 0-to-30 inches, with a statalarm occurring at 15 inches increasing level and a computer alarm at approximately 22 inches. These alarms would alert the operators in the control room such that appropriate actions could be taken. In addition to the alarms, sump level is input to the plant computer and is logged to the alarm log. Level is also recorded on a trend recorder in each control room. Safety related redundant level transmitters with a range of 3 inches to 24 inches are also provided in the normal sump. Both transmitter levels are indicated in the control room on receiver gauges and one train is recorded. Thus, the operators have several methods for monitoring changes in sump level.

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In addition to the normal sump level, indication of the emergency sump level is also provided by redundant safety related systems with a range of 0 to 3 feet. Both trains of instrumentation are indicated on receiver gauges in the control room and one train is recorded. This indication can be used in conjunction with the normal sump level indication to detect abnormal leakage in the Reactor Building. Two additional trains of containment level transmitters are installed in each Reactor Building to provide wide range level indication and recording with a range of 0 to 15 feet.

The normal sump is routinely pumped to the miscellaneous waste holdup tanks whenever the alarm point (15 inches) is reached. Pumping of the sump water is started manually, but terminates automatically when the sump level has dropped to 6 inches (which clears the statalarm). Each time the sump is pumped, it is recorded in the Unit Reactor Operator's Log Book. During pumping, a decreasing sump level indication and/or increasing miscellaneous waste holdup tank level indication can be used to verify flow from the normal sump. The flow rate from the sump can be determined using the rate of change in sump level.

In order to provide periodic monitoring of sump levels, the recording of normal and emergency sump levels is done daily. Daily monitoring of level indications is useful in confirming that level instrumentation are operable, while verifying the sump pumps are operable and maintaining the sump level at or below the alarm point. Calibration of the normal and emergency sump indications is performed during refueling.

In the event of increased leakage to the Reactor Building, sampling may be performed to determine the origin of the leakage (e.g., LPSW, feedwater, component cooling, or RC system).

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4. Letter From Hal B. Tucker (Duke) to Harold R. Denton (NRC) dated April 28, 1986.

THIS IS THE LAST PAGE OF THE TEXT SECTION 3.4.

See Insert for UFSAR Section 3.4.2

Insert for UFSAR Section 3.4.1.1.1

For the Auxiliary Building, Duke evaluated the effects of flooding caused by a leak from non-seismic moderate energy piping. A crack size calculated using the Standard Review Plan (SRP) Section 3.6.1, Branch Technical Position (BTP) Auxiliary Systems Branch (ASB) 3-1 guidelines was used. This evaluation concluded that for the bounding case (a crack in the 16-inch HPSW pipe) the effects of flooding can be mitigated without adversely affecting safety-related equipment required for safe shutdown (MODE 3 with average RCS temperature ≥ 525ºF). The analysis demonstrates that at least an hour is available from detection for operator action to isolate the flood. The operator would be alerted to a flood based on the level indicating alarms from the Auxiliary Building Waste Tanks (See Section 3.4.2, reference 5).

Insert for UFSAR Section 3.4.2

5. NRC Safety Evaluation dated xx/xx/xx.

Insert for Oconee UFSAR Section 3.7.3.9

only a single pipe break during a seismic event, and the analysis shall determine the effect on the safety-related portion of the system from the most limiting single pipe break.

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The operations sequence is as follows:

> The seismic trigger senses the initial earthquake ground motion with a normal setting of 0.01g and actuates the SMA-3 to full operation in less than 0.1 second.
>
> The SMA-3 operates for as long as the trigger detects the earthquake, plus an additional 10 seconds.
>
> The accelerograph can thus record a single earthquake or a sequence of earthquakes and aftershocks lasting as long as 30 minutes.

The output of each triaxial sensor is recorded using frequency modulation on a single four track cassette tape. Three of the tracks on the tape are the data tracks; the fourth is a reference track used for tape speed and amplitude compensation.

The Seismic Trigger and one Force Balance accelerometer of the SMA-3 system are located in the Unit 1 Tendon Gallery. Also, a second Force Balance accelerometer is located directly above at elevation 797' + 6" in the Oconee 1 Reactor Building. The recorder for the system is located in the Unit 1 Cable Room.

Also, a seismic trigger/switch is located in the Unit 1 tendon gallery. The Kinemetrics Model TS-3A has a preset acceleration threshold of 0.05g which activates the statalarm in Units 1 and 3 control rooms, when design conditions occur.

Six 2g peak recording accelorometers, manufactured by Engdahl-Model PAR 400, are also installed at various locations within the Oconee 1 Reactor Building. The instruments will provide post-seismic data for the following locations or items:

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Top of soil (free field) responses will not provide useful analytical data for the evaluation of major Class 1 structures founded on rock. Therefore, it is felt that free field instrumentation will not contribute to the evaluation of these structures.

3.7.4.2 Comparison of Measured and Predicted Responses

In the event of an earthquake, the data will be analyzed to determine the magnitude of the earthquake. If the design earthquake is exceeded, the units would be shut down and structures, systems, and equipment thoroughly investigated. Responses from instruments located on selected structures, systems and components will be compared to calculated responses for those structures, systems and components at the respective location when subjected to the same base response.

The recorded seismic data will be used for comparison and verification of seismic analysis assumptions, damping characteristics, and the analytical model used for the plant seismic design.

3.7.5 References

1. Bechtel Report, "Seismic Analysis Auxiliary Building", January, 1970.
2. Duke Power Engineering Design Report, "Static Method of Seismic Analysis of Piping Systems for Oconee 1, 2 and 3", File OS-27-B, June 6, 1970.
3. AEC Report TID-7024, "Nuclear Reactors and Earthquakes".
4. Newmark, N. M., "Torsion in Symmetrical Buildings".
5. Deleted Per 2000 Update

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See Insert for UFSAR Section 3.7.5

Insert for Oconee UFSAR Section 3.7.3.9

For moderate energy systems a crack is postulated as defined by Standard Review Plan (SRP) Section 3.6.1, Branch Technical Position (BTP) Auxiliary Systems Branch (ASB) 3-1 (See Section 3.7.5, reference 5).

Insert for UFSAR Section 3.7.5

5. NRC Safety Evaluation dated xx/xx/xx.

ATTACHMENT 3

DESCRIPTION OF PROPOSED CHANGE
AND
TECHNICAL JUSTIFICATION

ATTACHMENT 3

DESCRIPTION OF PROPOSED CHANGE
AND
TECHNICAL JUSTIFICATION

ATTACHMENT 3

DESCRIPTION OF PROPOSED CHANGE
AND
TECHNICAL JUSTIFICATION

Proposed Change

Duke Energy Corporation (Duke) proposes to change the licensing basis associated with the failure of non-Category I (non-seismic) piping. The ONS licensing basis does not include moderate energy line breaks. The current licensing basis was established by an Atomic Energy Commission (AEC) letter dated September 26, 1972, and Duke's response to that request dated October 24, 1972. This licensing basis addresses the failure of non-Category I piping and concludes that failure will not adversely affect safety-related equipment required for safe shutdown. However, the docketed correspondence does not clearly define the failure mechanism in terms of being a crack or a break. The September 26, 1972, AEC letter was issued based on a recent failure of an expansion bellows in the circulating water line that occurred at Quad Cities. Duke is unaware of any clear guidance in place at the time the AEC request was made as to the type of failure to be considered. The basis for Duke's proposed change to the licensing basis is Standard Review Plan (SRP) Section 3.6.1 Branch Technical Position (BTP) Auxiliary Systems Branch (ASB) 3-1 guidelines. These guidelines state a limited-size through-wall leakage crack is the appropriate postulated piping failure for moderate energy systems. The Fire Protection System and ventilation cooling water system, which were identified as the two sources of potential flooding for the Auxiliary Building in Duke's response (October 24, 1972 submittal) to the September 26, 1972 AEC letter, are moderate energy systems.

The proposed change will be documented as part of the plant's licensing basis by revising the Updated Final Safety Analysis Report (UFSAR), as shown in Attachments 1 and 2 following approval of this amendment. The justification for the proposed change is provided below.

Background

Consistent with other plants that were licensed during the same time period, Oconee Nuclear Station (ONS) is not licensed for moderate energy line breaks. The only AEC requirements for ONS in this area were imposed by an AEC letter dated September 26, 1972. This letter did not provide moderate energy line break requirements. The September 26, 1972, letter requested Duke to review ONS to determine whether the failure of any non-Category I equipment, particularly in the circulating water system and fire protection system, could result in a condition, such as flooding or the release of chemicals, that might potentially adversely affect the performance of safety-related equipment required for safe shutdown of the facilities or to limit the consequences of an accident. This letter did not clearly stipulate the type of failure to be assumed. However, the basis for the 1972 letter was a recent event at Quad Cities Nuclear Station where an expansion bellows in the circulating water line that serves the main condenser had failed.

The September 26, 1972, AEC letter was as follows:

"A failure of an expansion bellows in the circulating water line which serves the main condenser recently occurred at Quad-Cities, Unit 1. The resultant flooding caused degradation of some safety-related equipment.

You are requested to review Oconee Nuclear Station, Units 1, 2 and 3 to determine whether the failure of any non-Category I (seismic) equipment, particularly in the circulating water system and fire protection system, could result in a condition, such as flooding or the release of chemicals, that might potentially adversely affect the performance of safety-related equipment required for safe shutdown of the facilities or to limit the consequences of an accident.

The integrity of barriers to protect critical equipment from potentially damaging conditions should be assumed only when the barrier has been specifically designed for such conditions, If your review determines that safety-related equipment could be adversely affected, provide your plans and schedules for corrective action."

Duke's response to the 1972, AEC letter indicated that there was a remote possibility of flooding in the Turbine Building at the basement level due to failure of expansion joints in the condenser water box inlet or outlet nozzles. Duke provided an evaluation of potential flooding and measures to mitigate the flood. With regard to the Auxiliary Building, Duke provided the following response:

"The auxiliary building could be subject to flooding from two sources: the fire protection system and the ventilation cooling water system. The fire protection system does not constitute a threat due to the fact that the headers inside the auxiliary building will be empty and dry except when manually energized to fight a fire. The possibility for flooding from the ventilation cooling water system is reduced by flow limiting valves installed in all non-category I supply lines entering the auxiliary building larger than 3" in diameter. The maximum flow which can flood the building from a single rupture is 1140 gpm. Without taking credit for auxiliary building sumps, over 10 minutes is available for corrective action before safety-related equipment would be affected. Flooding by this source will be detected by high level alarm sensors in the auxiliary building sumps and necessary action taken by the operator to isolate the line rupture."

The AEC accepted Duke's response as noted in the Unit 2 and 3 Safety Evaluation Report (SER) dated July 7, 1973. Duke's design basis review effort has identified aspects of plant configuration and operation that are not in conformance with the October 24, 1972 Duke response. For example, the response indicated that the fire protection system header is empty and dry when it is actually charged. Also, the response implies that adequate time is available to mitigate flooding from a total rupture of the ventilation cooling water system. Duke has evaluated this scenario and concluded that adequate time is not available to mitigate flooding associated with a double-ended break of this piping prior to affecting safe shutdown equipment.

The original licensing basis for Turbine Building flooding was modified by installation of the Standby Shutdown Facility (SSF). The Nuclear Regulatory Commission (NRC) documented their acceptance of the SSF design to resolve Turbine Building flooding concerns by NRC SER dated April 28, 1983. As stated in the SSF SER, the SSF is designed to satisfy the safe shutdown requirements for fire protection, Turbine Building flooding and physical security. Safe shutdown at Oconee is defined as MODE 3 with average RCS temperature ≥ 525°F. As such, the technical justification for this licensing basis change request focuses on Auxiliary Building flooding.

Duke proposes to change the current licensing basis for non-Category I piping in moderate energy systems to assume a through-wall leakage crack as the postulated piping failure. The proposed change is consistent with Standard Review Plan (SRP) Branch Technical Position (BTP) Auxiliary Systems Branch

(ASB) 3-1 guidelines that allow through-wall leakage cracks to be assumed for moderate energy systems.

Justification for Change

There are no components or systems affected by the proposed change to the licensing basis. The large bore HPSW piping and LPSW piping is Duke Class G welded carbon or stainless steel piping and meets USAS B31.1 piping design criteria. Industry experience has shown that welded steel piping (including those supported in accordance with B31.1) is extremely resistant to damage by earthquakes of a magnitude several times larger than the Oconee Safe Shutdown Earthquake (SSE). Therefore, complete rupture of this piping during an SSE is not likely. The non-seismic HPSW System, specifically the 16-inch piping, represents the bounding non-seismic flooding source inside the Auxiliary Building. In order to affect components or systems in the Auxiliary Building a total rupture of the 16-inch HPSW piping would have to be assumed. This assumption is considered overly conservative. Walkdowns performed on the 16-inch HPSW piping confirmed it is adequately supported per USAS B31.1. Ultrasonic testing performed on selected locations concluded the HPSW piping is in good condition with respect to wall loss from erosion/corrosion. Based on the above reasons, good confidence exists that this piping will remain intact after an SSE.

Duke believes the Standard Review Plan (SRP) provides the appropriate guidelines for moderate energy line breaks. SRP Section 3.6.1, Appendix C (BTP ASB 3-1) specifies that a crack, rather than a break, be postulated in moderate energy piping. The Fire Protection System (HPSW) and ventilation cooling water system (LPSW), which were identified as the two sources of potential flooding for the Auxiliary Building in Duke's response (October 24, 1972 submittal) to the September 26, 1972 AEC letter, are moderate energy systems. The BTP states that the crack size to be assumed for moderate energy piping is one-half the pipe diameter in length and one-half the pipe wall thickness in width.

From a deterministic perspective, the HPSW system is the bounding non-seismic flood source. Besides the HPSW and LPSW systems, there are other non-seismic piping systems in the Auxiliary Building. However, they are closed loop or of limited capacity and do not present the flooding challenges of HPSW. Duke evaluated the effects of flooding caused by a leak, from a crack size calculated using the SRP guidelines, in the 16-inch

HPSW header. This evaluation concluded that for the bounding case, the effects of flooding can be mitigated without adversely affecting safety-related equipment required for safe shutdown (MODE 3 with average RCS temperature ≥ 525ºF). The analysis demonstrates that at least an hour is available from detection for operator action to isolate the flood. The operator would be alerted to a flood based on the level indicating alarms from the Auxiliary Building Waste Tanks. The level instrumentation is included in the periodic maintenance program.

Duke has taken action to reduce the risk of flooding in the Auxiliary Building. Ultrasonic testing was initially performed along the 16″ HPSW pipe where it is most susceptible to corrosion and where the Auxiliary Building would be most vulnerable from a flood standpoint should a break occur in the pipe. Evaluation of the results concluded wall thicknesses are acceptable. In addition, a requirement to conduct periodic inspections of HPSW and LPSW piping has been incorporated into Oconee's service water inspection program. Ultrasonic testing will be performed on appropriate intervals to properly monitor the piping wall thickness integrity. An Operations Auxiliary Building flood procedure has been created to direct isolation of Auxiliary Building flooding sources.

Although not required for safe-shutdown, defense in depth measures are being taken to protect safety-related equipment in the Low Pressure Injection (LPI)/Reactor Building Spray (RBS) pump rooms from a moderate size flood (SRP 3.6.1 pipe crack). Curbs are being installed on the first and second floors to prevent water from entering the LPI hatch area. This will prevent water from entering the LPI/RBS pump rooms from the spiral stair openings as well as the other smaller openings, and it will prevent water from affecting the Motor Control Centers (MCC's) for the Component Cooling (CC) and Spent Fuel Cooling (SFC) pumps. In addition to providing curbing, dividing walls penetrations between the individual LPI/RBS pump rooms as well as the LPI/RBS and HPI pump rooms will be sealed to prevent flood water migration.

As further justification for this change, Duke has also evaluated the risk associated with the total rupture of the 16-inch HPSW piping. The 16-inch HPSW piping presents the greatest non-seismic moderate energy flooding source in the Auxiliary Building. The Oconee PRA was not used to make this assessment. Instead, risk assessment tools were used to make a simplified, bounding quantification of the risk. The risk assessment concluded that the total core damage frequency (CDF) from the

Auxiliary Building flooding event resulting from a seismically induced total rupture of the 16-inch HPSW line is 1.8E-08/yr. The Oconee baseline CDF is 8.9E-05/yr according to Revision 2 of the Oconee PRA. The Large Early Release Frequency (LERF) is unaffected.

In addition to the risk during unit operation, the risks associated with an HPSW line break during unit shutdown were also evaluated. The evaluation concluded that an HPSW line break caused by an earthquake that results in Auxiliary Building flooding causing the loss of the LPI pumps (loss of decay heat removal) is not as likely as the direct loss of decay heat removal capability due to an earthquake.

In conclusion, the impact of flooding from a seismically induced crack in non-seismic moderate energy piping has been evaluated. Adequate time exists for operator action to isolate flooding sources prior to adversely affecting safety-related equipment required for safe shutdown. Duke's proposal to postulate cracks is consistent with SRP Section 3.6.1 BTP ASB 3-1 guidelines.

ATTACHMENT 4

NO SIGNIFICANT HAZARDS CONSIDERATION

Attachment 4
No Significant Hazards Consideration

Duke proposes to clearly define the current licensing basis associated with non-Category I piping as a through-wall leakage crack for the postulated piping failure. The basis for this change is Standard Review Plan (SRP) Branch Technical Position (BTP) Auxiliary Systems Branch (ASB) 3-1 guidelines that allow through-wall leakage cracks to be assumed in moderate energy systems. The Fire Protection System and ventilation cooling water system, which were identified as the two sources of potential flooding for the Auxiliary Building in Duke's response (October 24, 1972, submittal) to the September 26, 1972, AEC letter, are moderate energy systems. The fact that the Fire Protection System piping is charged does not conform with the current licensing basis for non-seismic piping failures.

Pursuant to 10 CFR 50.91, Duke Power Company (Duke) has made the determination that this amendment request involves a No Significant Hazards Consideration by applying the standards established by the NRC regulations in 10 CFR 50.92. This ensures that operation of the facility in accordance with the proposed amendment would not:

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

   No. The License Amendment Request (LAR) proposes to change the licensing basis for non-Category I (non-seismic) piping to assume a through-wall crack as the postulated piping failure. The proposed change does not involve any physical alteration of plant systems, structures or components, changes in parameters governing normal plant operation, or methods of operation. The proposed change does not affect any Chapter 15 accident analyses. Duke evaluated the effects of flooding caused by a leak from a crack size calculated using the SRP guidelines in the 16-inch HPSW header. This evaluation concluded that for the bounding case, the effects of flooding can be mitigated without adversely affecting safety-related equipment. At least an hour is available from detection for operator action to isolate the leak. Therefore, the probability or consequences of an accident previously evaluated is not significantly increased.

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

   No. The License Amendment Request (LAR) changes the licensing basis associated with non-seismic moderate energy line breaks. The proposed change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. Therefore, the possibility of a new or different kind of accident from any kind of accident previously evaluated is not created.

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

   No. The License Amendment Request (LAR) changes the licensing basis associated with non-seismic moderate energy line breaks. The impact of flooding from a seismically induced crack in non-seismic moderate energy piping has been evaluated. Adequate time exists for operator action to isolate flooding sources prior to adversely affecting safety-related equipment required for safe shutdown. As such, the proposed change does not involve a significant reduction in a margin of safety.

ATTACHMENT 5

ENVIRONMENTAL IMPACT ANALYSIS

ATTACHMENT 5

Environmental Impact Analysis

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

1. A significant hazards consideration.

   This conclusion is supported by the determination of no significant hazards contained in Attachment 4.

2. A significant change in the types or significant increase in the amounts of any effluents that may be released offsite.

   This LAR involves changing the licensing associated with non-seismic moderate energy line breaks at Oconee Nuclear Station. The plant will continue to operate as before. Therefore, this LAR will not change the types or amounts of any effluents that may be released offsite.

3. A significant increase in the individual or cumulative occupational radiation exposure.

   This LAR involves changing the licensing basis associated with non-seismic moderate energy line breaks at Oconee Nuclear Station. The plant will continue to operate as before. Therefore, this LAR will not increase the individual or cumulative occupational radiation exposure.

In summary, this LAR meets the criteria set forth in 10 CFR 51.22 (c)9 of the regulations for categorical exclusion from an environmental impact statement.