SENSITIVE

Duke Power®
A Duke Energy Company

R. A. JONES
Vice President

Duke Power
29672 / Oconee Nuclear Site
7800 Rochester Highway
Seneca, SC 29672

864 885 3158
864 885 3564 fax

November 1, 2002

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

Subject: Oconee Nuclear Station, Units 1, 2, and 3
Docket Numbers 50-269, 50-270, and 50-287
Proposed License Amendment Regarding Revisions to the Licensing Basis for the UFSAR Section on Water Level (Flood) Design (TSC 2002-06)

Pursuant to 10 CFR 50.90, Duke Energy Corporation (Duke) requests an amendment to Facility Operating License Nos. DPR-38, DPR-47, and DPR-55 for Oconee Nuclear Station (ONS) Units 1, 2, and 3, respectively, to revise the licensing basis associated with the failure of non-Category I (non-seismic) piping in the Auxiliary Building. The proposed change would allow deviations from the current licensing basis based on a risk informed justification. The current licensing basis was established by Atomic Energy Commission (AEC) letter dated September 26, 1972, and Duke's response to that request dated October 24, 1972. The licensing basis for non-Category I piping in the Auxiliary Building is that failure of this piping will not result in a flood that will adversely impact safety-related equipment required for safe shutdown.

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 with respect to flooding in the Auxiliary Building. For example, the response assumed the High Pressure Service Water (HPSW) header is dry when it is actually charged. In accordance with 10 CFR 50 Appendix B, Criterion XVI, Duke is resolving the non-conforming condition through a revision to the plant's licensing basis. Additionally, this submittal also resolves any ambiguity relative to the requirements of License Condition 3.D.

This License Amendment Request (LAR) proposes to change the licensing basis to allow HPSW piping and certain portions of the LPSW piping in the Auxiliary Building to remain non-Category I using the risk based approach guidelines of Regulatory Guide

A053

1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis." This risk informed evaluation concludes that the piping is not expected to fail under the maximum hypothetical earthquake (MHE) and that the contribution to core damage frequency (CDF) for this piping being non-seismically designed versus seismically designed is insignificant in relationship to the total CDF.

Duke will modify the plant to limit the scope of non-seismic piping included in the risk informed evaluation as follows: 1) install flow limiting device in the Plant Drinking Water (PDW) system to reduce the flow rate resulting from a postulated total rupture, 2) install and re-size flow limiting devices and check valves in non-seismic Low Pressure Service Water (LPSW) lines to the Auxiliary Building, 3) relocate LPSW piping above the Unit 1 and 2 Control Room to the Turbine Building, and 4) upgrade two fire hose racks to meet seismic standards. For the remaining scope of non-seismic HPSW and LPSW piping, Duke has performed a risk informed evaluation to demonstrate the risk associated with having non-seismic piping in the Auxiliary Building is insignificant.

Duke has determined that this licensing basis revision requires NRC review and approval. Duke requests that the review of this submittal be completed by May 31, 2003.

Attachment 1 provides the re-typed Oconee UFSAR pages. Attachment 2 provides a mark-up of the affected Oconee UFSAR pages. Duke Energy's technical justification, which includes a risk based evaluation to support the change to the ONS licensing basis, 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).

This proposed license amendment has been reviewed and approved by the Plant Operations Review Committee and 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,

Ronald A Jones, Vice President
Oconee Nuclear Site

Attachments

xc w/attachments:

Mr. L. N. Olshan, Project Manager
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

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

Mr. M. C. Shannon
Senior Resident Inspector
Oconee Nuclear Station

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

AFFIDAVIT

Ronald A Jones, being duly sworn, states that he is Vice President, Oconee Nuclear Site, Duke Energy Corporation, that he is authorized on the part of said Company to sign and file with the U. S. Nuclear Regulatory Commission this revision to the Facility Operating License Nos. DPR-38, DPR-47, and DPR-55; and that all statements and matters set forth herein are true and correct to the best of his knowledge.

Ronald A Jones, Vice President
Oconee Nuclear Site

Subscribed and sworn to before me this 1st day of November 2002

Cenice M. Breazeale
Notary Public

My Commission Expires:

2/12/03
Date

SEAL

ATTACHMENT 1

RETYPED OCONEE UFSAR PAGES

AND

UFSAR UPDATE INSTRUCTIONS

UFSAR UPDATE INSTRUCTIONS

Remove Pages	Replace Pages
3.4-2	3.4-2
3.4.3	3.4.3

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.

It is desirable, however, to allow a limited amount of backflow from the CCW discharge through the condensate coolers 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).

Duke's Auxiliary Building flooding licensing basis was established by requirements provided in an AEC letter dated September 26, 1972. This letter required Duke to evaluate the effects of flooding in the Auxiliary Building caused by the failure of non-seismic piping. Except for HPSW piping and certain portions of LPSW piping, at least 45 minutes is available from detection for operator action to isolate the source of flooding prior to adversely affecting the function of safety-related equipment required for safe shutdown. Only those portions of LPSW piping located in the Unit 1 and 2 LPI/RBS pump rooms have the potential to adversely affect the function of safety-related equipment prior to operator action. Probabilistic risk assessment (PRA) concluded that the HPSW piping and this portion of the LPSW piping is not expected to fail under the maximum hypothetical earthquake (MHE). The contribution to core damage frequency (CDF) for this piping being non-seismic versus seismic is insignificant in relationship to the total CDF. Therefore, the deviation from the requirements of the AEC's September 26, 1972, letter for the HPSW piping and the specific portion of the LPSW piping in the Auxiliary Building is acceptable. (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 pumped, it is recorded in the Unit Reactor Operator's Log Book. During pumping, a decreasing sump 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.

ATTACHMENT 2

MARKUP OF 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.

It is desirable to allow a limited amount of backflow from the CCW discharge through the condensate coolers 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).

Insert for UFSAR Section 3.4.1.1.1

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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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).

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.

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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. Deleted Per 2001 Update.

$\boxed{\text{Insert for UFSAR Section 3.4.2}}$

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ATTACHMENT 3

TECHNICAL JUSTIFICATION

ATTACHMENT 3

TECHNICAL JUSTIFICATION

OVERVIEW

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 with respect to flooding in the Auxiliary Building.

Pursuant to Regulatory Guide (RG) 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," this risk informed License Amendment Request (LAR) submittal provides a method for obtaining NRC review and approval of a proposed revision to the current licensing basis associated with the non-Category I Low Pressure Service Water (LPSW) and High Pressure Service Water (HPSW) piping in the Auxiliary Building. Currently, the Oconee Nuclear Station (ONS) licensing basis is based on information provided to the NRC during the early 1970's. The current licensing basis for HPSW piping in the Auxiliary Building assumes that the piping is empty and dry except when manually energized to fight a fire. This is contrary to the current plant configuration which maintains the piping full for fire protection reasons and to provide backup water to the HPI pump motor bearing coolers. The current licensing basis for LPSW piping in the Auxiliary Building assumes that adequate time is available to mitigate flooding from a total rupture. A recent Duke evaluation has concluded that adequate time is not available to mitigate flooding associated with a double-ended break of this piping prior to adversely affecting the function of safety-related equipment required for safe shutdown.

Duke's risk informed evaluation concludes that LPSW and HPSW piping capable of flooding and adversely affecting the function of safety-related equipment in the Auxiliary Building is not expected to fail under the maximum hypothetical earthquake (MHE) and that the contribution to core damage frequency (CDF) for this piping being non-seismically qualified is insignificant in relationship to the total CDF. Piping included in the scope of this evaluation is all HPSW piping located in the Auxiliary Building and LPSW piping located in the Unit 1 and 2 LPI/RBS pump rooms. If this piping were to rupture it could affect safety related equipment within the assumed 45 minute isolation time.

Results of the assessment demonstrate that the changes requested by this LAR, which allows these portions of the LPSW and HPSW piping in the Auxiliary Building to remain non-seismic USAS B31.1 design, are acceptable. Upgrading the piping to meet seismic design requirements would result in an insignificant decrease of 3E-07/yr in the total CDF. This provides the necessary justification to revise the licensing basis to retain the current piping design requirements for these portions of the LPSW and HPSW piping.

As part of Duke's reconstitution of the 1972 AEC letter requirements, Duke confirmed that other utility responses were focused on the effects of flooding on safety-related equipment and were limited to a review of non-seismic piping. This confirmation is based on utility responses to the September 26, 1972 letter and an NRC Safety Evaluation Report.

DESCRIPTION OF THE CURRENT LICENSING BASIS

The ONS licensing basis for LPSW and HPSW piping in the Auxiliary Building was established in the early 1970's. The only AEC requirements for ONS in this area were imposed by an AEC letter dated September 26, 1972. 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 facility or to limit the consequences of an accident. The basis for the 1972 letter was an 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 Units 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. The response indicated that the fire protection system or HPSW System header is empty and dry when it is actually charged. Also, the response states that adequate time is available to mitigate flooding from a total rupture of the ventilation cooling water system or LPSW System. Duke has evaluated this scenario and concluded that adequate time is not available to mitigate flooding associated with a double-ended break of LPSW 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 to allow HPSW and certain portions of the LPSW piping to remain non-Category I. Piping included are those in which a rupture would result in flooding of safety-related equipment before operator action can be taken to isolate the break. Credit for operator action is taken if the break can be isolated in < 45 minutes without adversely affecting the function of safety-related equipment required for safe shutdown.

DESCRIPTION OF THE CHANGE

This change will be reflected in the Updated Final Safety Analysis Report (UFSAR) following approval of this amendment. UFSAR Section 3.4.1.1.1, Current Flood Protection Measures for the Turbine and Auxiliary Buildings, will be revised, as follows, to indicate the proposed licensing basis:

"Duke's Auxiliary Building flooding licensing basis was established by requirements provided in an AEC letter dated September 26, 1972. This letter required Duke to evaluate the effects of flooding in the Auxiliary Building caused by the failure of non-seismic piping. Except for HPSW piping and certain portions of LPSW piping, at least 45 minutes is available from detection for operator action to isolate the source of flooding prior to adversely affecting the function of safety-related equipment required for safe shutdown. Only those portions of LPSW piping located in the Unit 1 and 2 LPI/RBS pump rooms have the potential to adversely affect the function of safety-related equipment prior to operator action. Probabilistic risk assessment (PRA) concluded that the HPSW piping and this portion of the LPSW piping is not expected to fail under the maximum hypothetical earthquake (MHE). The contribution to core damage frequency (CDF) for this piping being non-seismic versus seismic is insignificant in relationship to the total CDF. Therefore, the deviation from the requirements of the AEC's September 26, 1972, letter for the HPSW piping and the specific portion of the LPSW piping in the Auxiliary Building is acceptable. (Reference 5)"

UFSAR Section 3.4.2, References, will be revised to reference the NRC Safety Evaluation associated with this License Amendment Request (LAR).

JUSTIFICATION FOR THE PROPOSED CHANGES

Duke performed a PRA that justifies allowing portions of the LPSW and HPSW piping in the Auxiliary Building to remain non-seismically qualified. The PRA calculation demonstrates that the increased risks associated with the current piping design (USAS B31.1) are insignificant as compared to seismic designed piping. The SSF will remain available to mitigate the effects of an Auxiliary Building flood resulting from the total rupture of the piping.

LPSW and HPSW piping included in the scope of this change is piping that if it were to rupture could not be isolated prior to adversely affecting the function of safety related equipment required for safe shutdown. The specific piping is as follows:

- All HPSW piping in the Auxiliary Building, and
- LPSW piping to Air Handling Units (AHUs) 6, 7, and 8 physically located in the Unit 1 and 2 LPI/RBS pump rooms.

The Auxiliary Building flood resulting from a pipe crack (size defined by Standard Review Plan Section 3.6.1) does not adversely affect the function of safety-related equipment needed to place the plant in safe shutdown if isolated within 45 minutes.

The large bore HPSW and LPSW piping is Duke Class G welded carbon or stainless steel piping and was constructed in accordance with

USAS B31.1 piping design criteria. The smaller threaded piping was also constructed using USAS B31.1 piping design criteria. Industry experience has shown that steel piping (including those supported in accordance with USAS B31.1) is extremely resistant to damage by earthquakes of a magnitude several times larger than the Oconee Maximum Hypothetical Earthquake (MHE). Therefore, Duke's evaluation of this piping has concluded that its failure during an MHE is not credible. 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 large break of the 16-inch HPSW piping would have to be assumed. This assumption is considered overly conservative. The Service Water Piping Inspection Program assesses and manages the aging of raw water piping systems susceptible to general corrosion. Ultrasonic examinations performed by this program on selected locations in the LPSW and HPSW systems conclude no significant wall thinning has occurred due to general corrosion. The measured pipe wall thickness exceeds the established minimum acceptable values, and structural integrity of the pipe remains acceptable. Based on the above reasons, high confidence exists that this piping will remain intact after an MHE.

ENHANCEMENTS TO REDUCE FLOOD RISK

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 at appropriate intervals to properly monitor the piping wall thickness integrity. An Operations Auxiliary Building flood procedure was developed and implemented to direct response to a flood caused by failure of a non-seismic flooding source.

Curbs have been installed on the first and second floors to prevent water from entering the Low Pressure Injection (LPI) hatch area to address License Condition 3.D, Fire Protection. This will prevent water from pipe cracks from entering the LPI/ Reactor Building Spray (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.

Although not required for safe-shutdown, defense in depth measures are being taken to protect safety-related equipment in the LPI/RBS pump rooms from a moderate size flood. Dividing wall penetrations between the individual LPI/RBS pump rooms as well as the LPI/RBS and HPI pump rooms have been sealed to prevent flood water migration from one pump room to another pump room.

EFFECTS ON SAFETY

Deterministic Evaluation

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. These other systems are closed loop or of limited capacity and do not present the flooding challenges of HPSW.

Duke will make several changes to minimize the amount of piping included in the risk informed scope. These NRC commitments are as follows:

- Install flow limiting devices in the Plant Drinking Water (PDW) system to reduce the flow rate resulting from a postulated total rupture,
- Install and re-size flow limiting devices and check valves in non-seismic LPSW lines to the Auxiliary Building,
- Relocate LPSW piping above the Unit 1 and 2 Control Room to the Turbine Building, and
- Upgrade two fire hose racks to meet seismic standards.

Duke evaluated the effects of the total rupture of the other non-seismic piping in the Auxiliary Building and determined that 45 minutes is available from detection for operator action to isolate the flood prior to adversely affecting the function of safety-related equipment required for safe shutdown. The operator would be alerted to a flood based on level indicating alarms from level instruments for the Auxiliary Building Waste Tanks. This level instrumentation is included in the periodic maintenance program. Work request history shows the level instrumentation to be very reliable. Duke also evaluated the effects of a flood resulting from the bounding case crack (HPSW) and determined that flooding can be mitigated without adversely affecting the function of safety-related equipment required for safe shutdown (MODE 3 with average RCS temperature ≥ 525ºF).

Duke's evaluation of the HPSW pipe rupture (bounding case) concluded that the function of safety-related equipment required for safe shutdown would be adversely affected by the resulting flood. However, the units can still be placed in the safe shutdown condition using the SSF and the EFW system.

Using the conservative assumption that the Auxiliary Building HPSW header failure will be a total rupture, the resulting flood will cause the loss of HPI pumps. As demonstrated by risk assessment, this would result in an insignificant increase in CDF and LERF. Auxiliary Building flooding does not affect the ability to achieve hot shutdown using the SSF Reactor Coolant (RC) Makeup for seal cooling and EFW for secondary side cooling. The HPSW System piping is maintained full for fire protection reasons and to provide backup water to the HPI pump motor bearing coolers.

Probabilistic Risk Assessment

The probabilistic risk assessment used to assess the impact of the proposed change is based on Regulatory Guide 1.174 guidance (Reference 1). The risk impact of allowing the HPSW and LPSW lines to remain non-seismically mounted is compared to the risk that would be present if the lines were seismically mounted as intended by the guidance of the 1972 correspondence.

Scope of Review

Oconee is a three unit plant with inter-connecting Auxiliary Buildings and Turbine Buildings. Safety-related equipment is located in both of these locations. Turbine Building floods have been analyzed previously and a licensing basis has been reviewed and accepted by the NRC. Therefore the scope of this evaluation will only look at the possible impact of flooding on safety-related equipment in the Auxiliary Building. Even though possible floods could impact other units, we have assumed that all the water that spills from the postulated break stays in the unit where the break occurred. This maximizes the flood level and minimizes the time available to take operator action.

Identification of Critical Areas

A critical area is defined as an area where a flood could cause an initiating event, fail the related mitigating systems, or cause both with a high frequency relative to non-flood contributions. This implies that there is a high potential for damage and a credible source of flooding.

The initiating events can be divided into two groups: LOCAs and transients (e.g., a reactor trip). The LOCAs can occur through four failure modes: RCS pipe rupture, inadvertent opening of isolation valves, a relief valve opening and failing to re-close, and reactor coolant pump (RCP) seal failure. The first two are very unlikely to result from a flood. No mechanism has been identified where flooding can cause a pipe rupture. Furthermore, by design, a LOCA leading to containment flooding will not lead to the failure of the LOCA-mitigating functions. It is possible that motor-operated valves (MOVs) may open when water sprays hit an electrical cabinet. However, Duke determined that more than one cabinet has to be affected to lead to a LOCA through an isolation valve opening (redundant valves with separate power supplies). These cabinets are in the electrical equipment room, and a simultaneous failure of two cabinets is less likely than other adverse events. In addition, there are no normally pressurized fluid system pipes in this room.

The failure of a pressurizer relief valve to re-close is unlikely to be affected by a flood. This event has been included in the PRA model whenever a transient initiating event is followed by failures resulting in an increase in RCS pressure to the relief valve setpoint. The RCP seal failure can occur when the high pressure injection (HPI) pumps, the Component Cooling System and the SSF reactor coolant makeup pump are unavailable. Auxiliary Building floods do have the potential of failing HPI and CC. However, it would take an extremely large flood (in excess of 500,000 gallons) to impact CC. It would have to be large enough to fill all LPI and HPI rooms. No floods close to this size have been experienced in the industry. Both the relief valve failure to re-close and an RCP seal failure result in small LOCAs.

A transient initiating event may result when any equipment needed to sustain power operation is affected. Based on engineering judgment, the likelihood of a manually initiated reactor trip becomes very high in some cases where the failure does not cause a direct trip.

Based on a review of plant systems, interdependencies, and the physical layout of the plant, the following areas were identified as potentially critical flooding areas:

1. Control room,
2. Cable-spreading room,
3. Cable shaft,
4. Equipment room,
5. Component cooling pump room,
6. HPI pump room, and
7. Other Auxiliary Building pump rooms.

The control room is manned at all times during operation. In addition, the potential flood sources in the control room area cannot create a significant flow. Therefore, the operators would have more than adequate time to take mitigating actions.

The cable spreading room contains limited flood sources. A fire hose rack is located in the room. However, it will be seismically supported to minimize the likelihood of breaking. The air handling units are cooled by the Chilled Water System. This system has a limited volume; pipe breaks are not expected to cause a significant enough flood to impact any equipment in the room. Therefore, the cable spreading room is not considered to be a critical area.

Cables in the cable shaft area are not spliced, and all terminate outside the area. These cables would not be affected even if they became submerged.

The equipment room contains load centers X8 and X9, motor control centers XS1, XS2, XS3, XO, and XP, and other vital equipment in the AC and DC Power Systems. However, a review of Oconee equipment rooms shows that none of these rooms contain normally pressurized fluid system piping.

The component cooling pumps are located in an area that is open to corridors that run the length of the three Auxiliary Buildings. Any flood in this room would dissipate throughout these corridors.

The three HPI pumps are located at elevation 758 ft., in the lowest part of the Auxiliary Building basement. They share the room with one low-activity and one high-activity waste tank. The low-activity waste tank holds up to 2850 gallons of water, and can be emptied at 200 gpm by two transfer pumps. The high-activity waste tank holds up to 1870 gallons of water, and can be emptied at 100 gpm by its two transfer pumps. These tanks are significant because they receive runoff from the Auxiliary Building floor drain network, which means any water spilled in the building will eventually end up in the HPI pump room. The three HPI pumps are of the vertical type, with the motors located about 5 feet off the floor.

The Reactor Building Spray pumps and the Low Pressure Injection pumps are located in a separate set of rooms at elevation 758 in the Auxiliary Building basement. Flooding of the HPI and LPI pump rooms is independent: there are no direct paths for floods to propagate from one room to another. There are no waste tanks or other connections to the building drain network in the LPI room. Penetrations between the LPI room and the HPI room have been sealed. Floods could occur on upper levels of the Auxiliary Building and drain down into the LPI and RBS pump room. However, the probability of a flood at the same time as a transient which requires LPI or RBS is remote. LPI is important during shutdown periods; however, these are typically short in duration.

Based on this survey of critical areas, Duke concluded that the only area of concern in the Auxiliary Building flooding analysis is the HPI pump room. The sources available to flood this area, along with the potential resulting initiating event, point to the need for further analysis.

Current Plant Condition

To determine one input of the Regulatory Guide 1.174 evaluation, the ability of the non-seismic piping to survive an earthquake must be determined. Walk downs of the piping in question were conducted by ABS Consultants (formerly EQE). Conformance with USAS B31.1 piping code was evaluated. Pipe supports, hangers, material of construction, etc. were reviewed to ensure that the Oconee HPSW and LPSW could be evaluated by the industry experience data. A few discrepancies identified through walkdowns are being resolved through the corrective action program. For example, in the HPI pump rooms, U-bolts are being removed from LPSW piping near the air handling units and hanger discrepancies were corrected. Once the critical characteristics of the piping were validated, then the experience data was used to develop a fragility value for the piping. In addition to the experience data, deterministic calculations were performed on selected bounding locations (hangers) to validate the conclusions of the experience data. This combination of seismic experience and analytical calculations was used to conclude that the applicable median ground acceleration for the existing piping is 0.85 g (with uncertainties of Br = 0.3 and Bu = 0.46)(Reference 2). This controlling case is for small threaded piping and was conservatively applied to all non-seismic piping.

Current State Core Damage Frequency

The additional scenario to be evaluated is the bounding rupture of a service water pipe in the Auxiliary Building. If it were to rupture, the resulting flood would incapacitate the HPI pumps. This eliminates the primary means of providing Reactor Coolant Pump seal cooling. If we assume no credit for operator action, eventually the flood would reach the level where the Component Cooling pump motor control centers are located on the floor above. This would eliminate the backup means of providing RCP seal cooling (via the pump thermal barriers). The only remaining method for providing seal cooling is the Standby Shutdown Facility. The SSF has a Reactor Coolant Makeup Pump that can supply seal cooling to the Reactor Coolant Pumps, and it will be unaffected by the flood. Therefore, there are two additional cut sets to be added to the list of seismic cut sets: seismic pipe failure with seismic SSF failure, or seismic pipe failure with random SSF failure.

The Oconee seismic model was run and a list of the top cut sets was generated. The two new cut sets were added into this listing. SEISM was then used to combine the ONS earthquake curve with these cut sets. The seismic hazard curve for Oconee comes from an EPRI study that evaluated nuclear power plant sites in the Central and Eastern United States (Reference 3). The SEISM program combines the seismic hazard curve with the fragility curve for the piping and supports. This program is the same one that was used in the IPE and IPEEE Reports. It uses the "Zion method" as described in the PRA Procedures Guide (NUREG-CR-2300) with the exception of using the Monte-Carlo simulation in propagating uncertainties.

Results of the evaluation show that the total seismic core damage probability is 3.89E-05/year. The contribution to CDF from failure of the non-seismic service water piping comes from earthquakes above 0.3 g. There is a negligible chance that an MHE will result in core damage.

Core damage Risk with Upgraded (seismically designed) Piping

The second input to the Regulatory Guide 1.174 evaluation is an evaluation of the core damage risk if the non-seismic piping were upgraded to Oconee seismic piping criteria. ABS Consultants determined the fragility values that would be applied to this piping if it were upgraded to seismic design standards. They calculate that the mean acceleration would be 1.95 g. The associated uncertainty factors are Br = 0.33 and Bu = 0.59. When these values are used in the SEISM analysis, the core damage frequency is calculated to be 3.86E-05.

PRA Results and Conclusions

The current core damage frequency for Oconee Nuclear Station due to seismic events is 3.89E-05/yr. The contribution to this risk from failures of piping in the Auxiliary Building is 4.28E-07/yr. If the non-seismic piping in the Auxiliary Building were upgraded to seismic mounting standards the decrease in core damage frequency is very small. The total seismic risk would decrease to 3.86E-05/yr.

Therefore, piping failure in the Auxiliary Building at Oconee is a very small contributor to the total seismic risk of the plant. In addition, upgrading non-seismic piping to seismic standards would only result in a delta decrease in the risk of 3E-07. Per Regulatory Guide 1.174, this falls within Region III of Figure 3, "Acceptance Guidelines of Core Damage Frequency (CDF)." Therefore, this LAR (to not upgrade the pipe to seismic standards) is considered an acceptable change to the licensing basis.

Large Early Release Frequency (LERF)

The calculation of LERF is unaffected by this change. Review of the PRA model reveals that practically all the contribution to LERF comes from interfacing system LOCA sequences. A postulated pipe break in the service water piping in the Auxiliary Building has no impact on the likelihood of the occurrence of an interfacing system LOCA. Therefore there is no change to the LERF value.

Shutdown

In addition to the at power risk calculated above, shutdown risk was evaluated. The LPI pumps are important during shutdown due to their decay heat removal function. Postulated Auxiliary Building leaks/floods would be routed to the Auxiliary Building basement and could potentially flood the LPI pumps. The LPI pumps are located in the Auxiliary Building basement similar to the HPI pumps. However, as shown above, the service water piping in the Auxiliary Building piping is very robust. It will not be damaged by a design basis earthquake. In fact, a review of the IPE shows that the Auxiliary Building piping and supports have a higher fragility value than the LPI cooler supports. As such, if an earthquake were to occur that is large enough to damage the Auxiliary Building piping, the decay heat removal function will likely also be lost from the seismic event. Therefore, the consequences of flooding are irrelevant. Further, since the period of time that LPI is relied upon as the only means of DHR is short, the chance of a large earthquake occurring during that window of operation is small. SEISM results show that pipe failures will start to occur when earthquakes reach a level of approximately 0.3 g. Reference 4 shows that the annual probability of exceeding 0.3 g at Oconee is 3E-05/yr. This converts to a daily probability of 8E-08/day. During a typical refueling, Oconee is in the LPI cooling mode for 2-3 weeks. Therefore the probability of an earthquake of sufficient size to damage LPI while it is required for decay heat removal is approximately 2E-06.

Component	A	Br	Bu
AB Piping	0.85	0.3	0.46
LPI Cooler Supports	0.71	0.27	0.48

PRA QUALITY

PRA Updates

Duke's Severe Accident Analysis Group (SAAG) periodically evaluates changes to the plant with respect to the assumptions and modeling in the Oconee PRA. The original 1984 Oconee NSAC-60 PRA (Reference 5) was a Level 3 PRA with internal and external events sponsored by the Electric Power Research Institute (EPRI) and Duke. The NRC contractor, Brookhaven National Laboratory (BNL), reviewed NSAC-60 and published its findings in NUREG/CR-4374 Vol. 1-3 (Reference 6). In 1990, a large-scale review and update of the PRA resulted in the Individual Plant Examination Report (IPE) submitted to the NRC as part of Generic Letter 88-20 response (Reference 4). The NRC reviewed the IPE submittal and documented its review in a Staff Evaluation (Reference 7).

In 1995, Oconee initiated Revision 2 of the 1990 Individual Plant Examination, and provided the results to the NRC staff in 1997 (Reference 8). Currently, Revision 3 of the Oconee PRA is underway. This update is a comprehensive revision to the PRA models and associated documentation. The objectives of this update are as follows:

- To ensure the models comprising the PRA accurately reflect the current plant, including its physical configurations, operating procedures, maintenance practices, etc.

- To review recent operating experience with respect to updating the frequency of plant transients, failure rates, and maintenance unavailability data.

- To correct items identified as errors and implement PRA enhancements as needed.

- To address weaknesses identified in the recent Oconee PRA Peer Review.

- To utilize updated Common Cause Analysis data and Human Reliability Analysis data.

PRA maintenance encompasses the identification and evaluation of new information into the PRA and typically involves minor modifications to the plant model. PRA maintenance and updates as well as guidance for developing PRA data and evaluation of plant modifications, are governed by Workplace Procedures. In January 2001, an enhanced configuration control process was implemented to more effectively track, evaluate, and implement PRA changes to better ensure the PRA reflects the as-built, as-operated plant.

Peer Review Process

Between May 7-11, 2001, Oconee participated in the B&W Owners Group (B&WOG) PRA Certification Program. This review followed a process that was originally developed and used by the Boiling Water Reactor Owners Group (BWROG) and subsequently broadened to be an industry-applicable process through the Nuclear Energy Institute Risk Applications Task Force. The resulting industry document, NEI-00-02 (Reference 9), describes the overall PRA peer review process. The Certification /Peer Review process is also linked to a recently approved ASME standard (Reference 10).

The objective of the PRA Peer Review process is to provide a method for establishing the technical quality and adequacy of a PRA for a range of potential risk-informed plant applications for which the PRA may be used. The PRA Peer Review process employs a team of PRA and system analysts, who possess significant expertise in PRA development and PRA applications. The team uses checklists to evaluate the scope, comprehensiveness, completeness, and fidelity of the PRA being reviewed. One of the key parts of the review is an assessment of the maintenance and update process to ensure the PRA reflects the as-built plant.

The review team for the Oconee PRA Peer Review consisted of six members. Three of the members were PRA personnel from other utilities. The remaining three were industry consultants. Reviewer independence was maintained by assuring that none of the six individuals had any involvement in the development of the Oconee PRA or IPE.

The Peer Review team noted that the Oconee PRA had a strong foundation laid in NSAC-60 and the IPE and that the full scope Level 3 PRA with external events should support a wide range of applications. A summary of some of the Oconee PRA strengths and areas for enhancement from the peer review are as follows:

Strengths

- Results summary and insights
- Uncertainty/sensitivity analysis
- Time dependent RCP seal LOCA/SBO treatment
- Delineation of small LOCA contributors
- Bayes' update of failure data validated
- Detailed analysis of hydroelectric plant
- Strong maintenance and update process
- Thorough system notebooks with good detail, separate quantification, clear boundaries, and tie to service experience

Areas for Enhancement

- Improved basis for identifying and screening support system initiators
- Enhanced documentation of dependencies
- Enhanced guidance and documentation for event sequence quantification
- Enhanced completeness and accountability of common cause failures
- Enhanced treatment of dependencies and time basis for human reliability
- Improved justification for assumptions and calculations impacting LERF
- Enhanced documentation of screening of containment isolation and bypass pathways
- Enhanced documentation of standby test intervals

The significance levels of the B&WOG Peer Review Certification process have the following definitions:

A. Extremely important and necessary to address to ensure the technical adequacy of the PRA, the quality of the PRA, or the quality of the PRA update process.

B. Important and necessary to address but may be deferred until the next PRA update.

The Oconee PRA received 4 "A" and 35 "B" fact and observation findings during its peer review. All four of the "A" findings have been addressed and are being incorporated into Oconee PRA Rev. 3 update that is nearing completion. Many of the "B" findings have been incorporated as well. Any remaining "B" findings will be incorporated at the next PRA update.

PRA Quality Assurance Methods

Approved workplace procedures address the quality assurance of the PRA. One way the quality assurance of the Oconee PRA is ensured is by maintaining a set of system notebooks on each of the PRA systems. Each system PRA analyst is responsible for updating a specific system model. This update consists of a comprehensive review of the system including drawings and plant modifications made since the last update as well as implementation of any PRA change notices that may exist on the system. The analyst's primary focal point is with the system engineer at the site. The system engineer provides information for the update as needed. The analyst will review the PRA model with the system engineer and as necessary, conduct a system walkdown with the system engineer. This interaction is documented in a memorandum.

The system notebooks contain, but are not limited to, documentation on system design, testing and maintenance practices, success criteria, assumptions, descriptions of the reliability data, as well as the results of the quantification. The system notebooks are reviewed and signed off by a second independent person and are approved by the manager of the group.

When any change to the PRA is identified, the same three-signature process of identification, review, and approval is utilized to ensure that the change is valid and that it receives the proper priority.

<u>Maintenance Rule Configuration Control</u>

10 CFR 50.65 (a)(4), Regulatory Guide 1.182 (Reference 11), and NUMARC 93-01 (Reference 12) require that prior to performing maintenance activities, risk assessments shall be performed to assess and manage the increase in risk that may result from proposed maintenance activities. These requirements are applicable for all plant modes. NUMARC 91-06 (Reference 13) requires utilities to assess and manage the risks that occur during the performance of outages.

Duke has several Work Process Manual procedures and Nuclear System Directives that are in place at the Oconee Nuclear Station to ensure the requirements of the Maintenance Rule are implemented. The key documents are as follows:

- Nuclear System Directive 415, "Operational Risk Management (Modes 1-3) per 10 CFR 50.65 (a.4)," Revision 0, October 2000.
- Nuclear System Directive 403, "Shutdown Risk Management (Modes 4, 5, 6, and No-Mode) per 10 CFR 50.65 (a.4)," Revision 7, April 2001.
- Work Process Manual, WPM-609, "Innage Risk Assessment Utilizing ORAM-SENTINEL," Revision 3, November 2000.
- Work Process Manual, WPM-608, "Outage Risk Assessment Utilizing ORAM-SENTINEL," Revision 3, November 2000.

The documents listed above are used to address the Maintenance Rule requirement and the on-line (and off-line) Maintenance Policy requirement to control the safety impact of combinations of equipment removed from service. More specifically, the Nuclear System Directives address the process; define the program; and state individual group responsibilities to ensure compliance with the Maintenance Rule. The Work Process Manual procedures provide a consistent process for utilizing the computerized software assessment tool, ORAM-SENTINEL, which manages the risk associated with equipment inoperability.

ORAM-SENTINEL is a Windows-based computer program designed by the Electric Power Research Institute as a tool for plant personnel to use to analyze and manage the risk associated with all risk significant work activities including assessment of combinations of equipment removed from service. It is independent of the requirements of Technical Specifications and Selected Licensee Commitments.

The ORAM-SENTINEL models for Oconee are based on a "blended" approach of probabilistic (the full Oconee Revision 2 PRA model is utilized) and traditional deterministic approaches. The results of the risk assessment include a prioritized listing of equipment to return to service, a prioritized listing of equipment to remain in service, and potential contingency considerations.

Additionally, prior to the release of work for execution, Operations personnel must consider the effects of severe weather and grid instabilities on plant operations. This qualitative evaluation is inherent of the duties of the Work Control Center Senior Reactor Operator (SRO). Responses to actual plant risk due to severe weather or grid instabilities are programmatically incorporated into applicable plant emergency or response procedures.

REFERENCES

1. Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions On Plant-Specific Changes to the Licensing Basis," July 1998.
2. ABS Calculation #1095211 C-001, Rev. 0, "Seismic Fragilities, Service Water Piping," 8-28-02.
3. Probabilistic Seismic Hazard Evaluation for Oconee Nuclear Station, RP101-53, Electric Power Research Institute, April, 1989.
4. Oconee Nuclear Station Unit s 1, 2, and 3, IPE Submittal Report, Duke Power Company, November 30, 1990.
5. "A Probabilistic Risk Assessment of Oconee Unit 3," NSAC-60, June 1984.
6. NUREG/CR-4374, Vol. 1-3, "A Review of the Oconee-3 Probabilistic Risk Assessment," Brookhaven National Laboratory, March 1986.
7. NRC Letter to Duke Power Company, "Examination of the Oconee, Units 1,2 and 3 Individual Plant examination for examination (IPE) - Internal Events Submittal," April 1, 1993.
8. "Probabilistic Risk Assessment Individual Plant Examination," Oconee Nuclear Site letter to NRC, February 13, 1997.
9. NEI-00-02, "Industry PRA Peer Review Process," Nuclear Energy Institute, January 2000.

10. "Standard For Probabilistic Risk Assessment for Nuclear Power Plant Applications," ASME RA-S-2002, April 5, 2002.
11. NRC Regulatory Guide 1.182, "Assessing and Managing Risk Before Maintenance Activities at Nuclear Power Plants," May 2000.
12. NUMARC 93-01, "Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," March 2000.
13. NUMARC 91-06, "Guidelines for Industry Actions to Assess Shutdown Management," December 1991.

ATTACHMENT 4

NO SIGNIFICANT HAZARDS CONSIDERATION

Attachment 4

No Significant Hazards Consideration

Pursuant to 10 CFR 50.91, Duke Energy Corporation (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 associated with HPSW piping and portions of LPSW piping located in the Auxiliary Building to remain non-Category I using risk based approach guidelines of Regulatory Guide 1.174. The proposed change does not involve changes in parameters governing normal plant operation, or methods in operation. The proposed change does not affect any Chapter 15 accident analyses. Duke evaluated the effects of flooding caused by a leak in the 16-inch HPSW header and a total rupture of the HPSW header. This evaluation concluded that for the leak, the effects of flooding can be mitigated without adversely affecting the function of safety-related equipment. The evaluation concluded that the total rupture of HPSW piping would result in a flood that would adversely affect the function of safety-related equipment required for shutdown. However, the Standby Shutdown Facility (SSF) can provide reactor coolant makeup and the Emergency Feedwater (EFW) System can provide secondary side cooling in place of those safety-related systems whose function would be adversely affected during this flood. Duke's risk informed evaluation concludes that the LPSW and HPSW piping located in the Auxiliary Building is not expected to fail under the maximum hypothetical earthquake (MHE) and that the contribution to core damage frequency (CDF) for this piping being non-seismically qualified is insignificant in relationship to the total CDF. 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 LAR changes the licensing basis associated with non-seismic HPSW piping and portions of non-seismic LPSW piping located in the Auxiliary Building. The proposed change does not necessitate a change 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 of HPSW piping and certain portions of LPSW Piping in the Auxiliary Building. The requested change to the licensing basis does not affect any Chapter 15 analyses. Duke's PRA concludes that the piping is not expected to fail under the maximum hypothetical earthquake (MHE), that the contribution to core damage frequency (CDF) for this piping being non-seismic versus seismic is insignificant in relationship to the total CDF, and that the SSF can provide reactor coolant makeup and the EFW can provide secondary side cooling in place of safety-related systems that are assumed unavailable due to Auxiliary Building flooding. 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 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 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.