MCGUIRE NUCLEAR STATION

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June 7, 2007

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

Subject: Duke Power Company, LLC d.b.a. Duke Energy Carolinas, LLC
McGuire Nuclear Station, Unit 1
Docket Number 50-369
Emergency License Amendment Request for One-Time Limited Duration Extension of Allowed Outage Time for the Unit 1A Emergency Diesel Generator

Pursuant to 10 CFR 50.90, and 10 CFR 50.91(a)(5), Duke Power Company LLC d.b.a. Duke Energy Carolinas, LLC (Duke), the licensee for the William B. McGuire Nuclear Station, proposes a one-time limited duration extension of the Technical Specification Required Action Completion Time associated with the Unit 1 A Emergency Diesel Generator (EDG). The requested extension would allow continued operation of Unit 1 for an additional 168 hours while repairs and related testing of the 1A EDG are completed.

The proposed amendment is being requested on an emergency basis pursuant to 10 CFR 50.91(a)(5). On June 5, 2007, at 1741 hours, the Unit 1A EDG was declared inoperable and the 72 hour action statement of Technical Specification (TS) 3.8.1 Required Action B was entered to perform routine TS surveillance testing. During this testing, the Control Room received an overload alarm. Subsequent troubleshooting determined the cause of the alarm to be an electrical problem with the 1A EDG Jacket/Intercooler Water Pump Motor. The Jacket/Intercooler Water Pump Motor is required for EDG operability.

Both units are currently at 100% power. Completion Times for the applicable TS 3.8.1 Required Actions expire on June 8, 2007 at 1741 hours. The failed motor will have to be shipped to a repair facility and it will take more than the 72 hours allowed by TS 3.8.1. Therefore, in order to avoid the unnecessary shutdown of McGuire Unit 1, Duke requests approval of this license amendment application on a one-time emergency basis by June 8, 2007 at 1741 hours.

Attachment 1 provides a description of the proposed change and the technical justification, an evaluation of significant hazards consideration pursuant to 10 CFR 50.92(c) and an environmental assessment.

Attachment 2 provides the existing TS pages marked-up to show the proposed change.

Attachment 3 contains the retyped (clean) TS pages.

Attachment 4 includes the regulatory commitments documented in this request.

Implementation of this proposed change to the McGuire TS will not impact the McGuire Updated Final Safety Analysis Report (UFSAR).

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

Pursuant to 10 CFR 50.91, a copy of this proposed amendment is being sent to the appropriate North Carolina state official.

Should you have any questions concerning this information, please call K. L. Ashe at (704) 875-4535.

Very truly yours,

Thomas P. Harrall, Jr.
Vice President, Plant Support

Attachments

xc w/ Attachments:

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

Thomas P Harrall Jr

Thomas P. Harrall, Jr., Vice President, Plant Support

Subscribed and sworn to me: June 7, 2007
Date

Kay L Crane, Notary Public

My commission expires: 4-1-2012
Date

KAY L. CRANE
NOTARY
PUBLIC
LINCOLN COUNTY, N.C.

ATTACHMENT 1

DESCRIPTION OF PROPOSED CHANGES, TECHNICAL JUSTIFICATION, SIGNIFICANT HAZARDS CONSIDERATION PURSUANT TO 10CFR50.92c AND ENVIRONMENTAL ASSESSMENT

1. Description:

Pursuant to 10 CFR 50.90, and 10 CFR 50.91(a)(5), Duke Power Company LLC d.b.a. Duke Energy Carolinas, LLC (Duke), the licensee for the William B. McGuire Nuclear Station, proposes a one-time limited duration extension of the Technical Specification Required Action Completion Time associated with the Unit 1 A Emergency Diesel Generator (EDG). The requested extension would allow continued operation of Unit 1 for an additional 168 hours while repairs and related testing of the 1A EDG are completed.

The proposed amendment is being requested on an emergency basis pursuant to 10 CFR 50.91(a)(5). On June 5, 2007, at 1741 hours, the Unit 1A EDG was declared inoperable and the 72 hour action statement of Technical Specification (TS) 3.8.1 Required Action B was entered to perform routine TS surveillance testing. During this testing, the Control Room received an overload alarm. Subsequent troubleshooting determined the cause of the alarm to be an electrical problem with the 1A EDG Jacket/Intercooler Water Pump Motor. The Jacket/Intercooler Water Pump Motor is required for EDG operability.

Completion Times for the applicable TS 3.8.1 Required Actions expire on June 8, 2007 at 1741 hours. Although efforts are currently in progress to perform the necessary repairs, the 1A EDG may not be restored to an operable status by 1741 hours on June 8, 2007. Therefore, in order to avoid the unnecessary shutdown of McGuire Unit 1, Duke requests approval of this license amendment application on a one-time emergency basis by June 8, 2007 at 1741 hours.

2. Proposed Change:

The proposed Unit 1 TS change revises the Completion Time for Required Action 3.8.1 (B.4) from 72 hours to 240 hours on a one-time basis. Marked-up TS pages illustrating the proposed change are provided in Attachment 2.

3. Background:

On June 5, 2007, at 1741 hours, the Unit 1A EDG was declared inoperable and the 72 hour action statement of Technical Specification (TS) 3.8.1 Required Action B was entered to perform routine TS surveillance testing. During this testing, the Control Room received an overload alarm. Subsequent troubleshooting determined the cause of the alarm to be an electrical problem with the 1A EDG Jacket/Intercooler Water Pump

Motor. The Jacket/Intercooler Water Pump Motor is required for EDG operability.

Completion Times for the applicable TS 3.8.1 Required Actions expire on June 8, 2007 at 1741 hours. Although efforts are currently in progress to perform the necessary repairs, the 1A EDG may not be restored to an operable status by 1741 hours on June 8, 2007. In order to avoid the unnecessary shutdown of McGuire Unit 1, Duke proposes a one-time limited duration extension of the Technical Specification Required Action Completion Time associated with the Unit 1A Emergency Diesel Generator (EDG). The requested extension would allow continued operation of Unit 1 for an additional 168 hours while repairs and related testing of the 1A EDG are completed.

4. Current Requirements:

TS 3.8.1 requires that two (2) separate and independent EDGs, capable of supplying the Onsite Essential Auxiliary Power Systems, be operable in Modes 1, 2, 3 and 4. With one EDG inoperable, TS 3.8.1 Condition B, Required Action B.4 requires the restoration of the inoperable EDG to an operable status within 72 hours or the Unit must be in at least hot standby (Mode 3) within the next 6 hours and in cold shutdown (Mode 5) within the following 30 hours.

5. Basis for Current Requirements:

The operability requirements for the alternating current (AC) sources during plant operation ensures that sufficient power will be available to supply safety-related equipment required for 1) the safe shutdown of the facility and 2) the mitigation and control of accident conditions within the facility. The minimum specified independent and redundant alternating power sources satisfy the requirements of 10CFR50, Appendix A, General Design Criteria 17.

The TS Action requirements specified for the levels of degradation of the power sources provide restrictions for continued facility operation commensurate with the level degradation. The operability requirements for the power sources are consistent with the initial condition assumptions of the accident analyses and are based upon maintaining the remaining AC power sources and associated distribution systems operable.

According to Regulatory Guide 1.93, operation may continue with one EDG inoperable for a period that should not exceed 72 hours. In this condition, the remaining operable EDG and offsite

circuits are adequate to supply electrical power to the onsite safety-related electrical distribution system. This 72 hour period takes into account the capability and capacity of the remaining AC sources, a reasonable time for repairs, and the low probability of the Design Basis Accident during this period.

6. Reason for Requesting Emergency Amendment:

Regulation 10 CFR 50.91(a)(5) states that where the NRC finds that an emergency situation exists, in that failure to act in a timely way would result in derating or shutdown of a nuclear power plant, or in prevention or either resumption of operation or increase in power output up to the plant's licensed power level, it may issue a license amendment involving no significant hazards consideration without prior notice and opportunity for a hearing or for public comment. The regulation also states that the NRC will decline to dispense with notice and comment on the no significant hazards if it determines that the licensee has abused the emergency provision by failing to make timely application for the amendment and thus itself creating the emergency. The regulation requires that a licensee requesting an emergency amendment explain why the emergency situation occurred and why the licensee could not avoid the situation. As explained below, an emergency amendment is needed to preclude an unnecessary plant shutdown and cooldown, and Duke could not have reasonably avoided the situation or made timely application for an amendment.

7. Reason Emergency Situation Has Occurred:

On June 5, 2007, at 1741 hours, the Unit 1A EDG was declared inoperable and the 72 hour action statement of Technical Specification (TS) 3.8.1 Required Action B was entered to perform routine TS surveillance testing. During this testing, the Control Room received an overload alarm. Subsequent troubleshooting determined the cause of the alarm to be an electrical problem with the 1A EDG Jacket/Intercooler Water Pump Motor. The motor failure was evaluated and a determination was made that the motor should be sent to a repair facility. The repair will be beyond the LCO criteria in TS 3.8.1. The Jacket/Intercooler Water Pump Motor is required for EDG operability.

8. Reason the Situation Could Not Have Been Avoided:

The 1A EDG jacket/Intercooler Water Pump Motor is a unique design with a shaft extending through the motor that turns a pump on each end. The preventive maintenance program for these motors is

based upon EPRI Guideline TR-106857-V8, Low Voltage Electrical Motors (600 volt and below) as well as benchmarking of motor maintenance programs at other utilities. These motors receive maintenance [lubrication, external visual inspection, electrical testing (winding resistance, insulation resistance, polarization index), vibration tests, and thermographic checks] on a periodic basis in accordance with our maintenance program. In addition, the review of the 1A motor electrical test data over a 7 year period shows stable data and no degrading trends. Therefore, this failure could not have been foreseen.

The 1A EDG Jacket/Intercooler Water Pump Motor ran normally throughout the period. The failure occurred during TS required testing of the EDG.

Following the normal troubleshooting protocol, additional tests were performed indicating a problem with the electrical motor.

Further diagnostic testing and an internal visual inspection of the motor windings at the vendor's facility will serve to identify the specific cause of the problem with the motor.

The Jacket/Intercooler Water Pump Motor is required for EDG operability. Completion Times for the applicable TS 3.8.1 Required Actions expire on June 8, 2007 at 1741 hours. A spare motor has not been located at this time and no viable alternative cooling options that could be implemented within the timeframe of this extension have been identified; therefore, the 1A EDG will not be restored to an operable status by 1741 hours on June 8, 2007. Alternative cooling options evaluated included:

- Replacement of the single motor with two motors, one for each pump,
- Mounting two motors offset from the pump centerline and coupled to the pumps through belts and pulleys,
- Connecting an external source of water into the system,
- Locating a non-safety related motor with a similar configuration, and
- Installing one motor to drive the pump and installation of a section of pipe in place of the intercooler pump.

The failed motor will have to be shipped to the motor repair vendor for repair and it will take more than the 72 hours allowed by TS 3.8.1. Neither a routine nor an exigent amendment request could have been processed with the 72 hour period. Therefore, an emergency amendment is needed to preclude an unnecessary shutdown.

9. Technical Evaluation:

The proposed amendment to allow a one-time extension of the allowed outage time (AOT) for EDG 1A is based on the following considerations:

Common Cause

The preliminary troubleshooting indicates that a random electrical failure has developed within the motor. This is further supported by the fact that each of the other Jacket/Intercooler Water Pump and Motors were successfully run on June 6, 2007 following the failure of the 1A EDG Jacket/Intercooler Water Pump Motor. Therefore, a potential common cause failure mode has not been identified at this time.

Power Systems

The McGuire onsite electrical power system consists of all sources of electrical power and their associated distribution systems in each of the two generating units. These sources are the main generator, two emergency diesel generators and the batteries. Each unit has two redundant and independent 4160 Volt Essential Auxiliary Power Systems which normally receive power for the normal power distribution system. After verification of a loss of offsite power (LOOP) or a sustained degraded offsite power condition, the normal and alternate incoming feeder circuit breakers automatically trip. During a LOOP condition, power to each of the redundant 4160 Volt Essential Auxiliary Power Systems is provided by a completely independent diesel-electric generating unit. Each of the 4160 Volt Essential Auxiliary Power System (1E) electrical buses is totally capable of fulfilling their design function independently. There are no overlapping electrical loads shared between the 1E buses. A loss of one emergency diesel generator does not increase the demand on any other emergency diesel generator.

Risk Evaluation

Duke Energy has used a risk-informed approach to determine the risk significance of extending the current EDG 1A Technical Specification allowed outage time of 72 hrs by seven days, for a total allowed outage time of 10 days. Considering that the proposed Technical Specification change is temporary, the acceptance guidelines provided in Reg. Guides 1.174 and 1.177 are increased by an order of magnitude, as was reviewed and approved by the NRC in a similar request by DC Cook.

The current PRA model (Rev. 3a) was used to perform the risk evaluation. The McGuire PRA is full scope PRA including both internal and external events. The base case non-seismic CDF and LERF are 2.9E-5/yr and 2.4E-6/yr, respectively. (The seismic results typically are not sensitive to unavailabilities of individual components and the seismic impact for this application is judged to be insignificant relative to the non-seismic impacts.)

The results indicate that the incremental conditional large early release probability (ICLERP) is more limiting with respect to the Regulatory Guide acceptance criterion. The LERF and CDF results are both dominated by tornado and loss of offsite power initiated sequences resulting in a station blackout on failure of the redundant diesel generator. Subsequent failure of secondary side heat removal results in core damage. The LERF is sensitive to station blackouts because of the loss of the hydrogen mitigation system. Uncontrolled hydrogen accumulation increases the challenge to containment integrity.

With a one-time 10-day outage on EDG 1A, the non-seismic delta CDF and delta LERF are 1.1E-6/yr and 1.3E-7/yr, respectively and the incremental conditional core damage probability (ICCDP) and ICLERP are 1.1E-6 and 1.3E-7, respectively.

For *permanent* changes, RG 1.177 and RG 1.174 outline acceptance guidance criteria of 5.0E-7 for ICCDP and 5.0E-8 for ICLERP, respectively, based on the baseline CDF being smaller than 1.0E-4/reactor-year.

Consistent with the D. C. Cook Safety Evaluation Report of December 10, 2003, the *temporary* change acceptance guidance criteria are increased by an order of magnitude. Thus, the ICCDP (1.1E-6) and ICLERP (1.3E-7) are within the acceptable values for temporary increases. Using the temporary change criteria, the Technical Specification. allowed outage time could be increased for up to 34 days. In addition, exercising the proposed 7-day extension would avoid additional risk associated with the plant shutdown and transitional risk.

Probabilistic Risk Assessment (PRA) Quality

Duke periodically evaluates changes to the plant with respect to the assumptions and modeling in the McGuire PRA. The original McGuire PRA was initiated in March 1982 by Duke Power Company staff with Technology for Energy Corporation as a contractor. Law Engineering Testing Company and Structural Mechanics Associates provided specific input to the seismic analysis. It

was a full scope Level 3 PRA with internal and external events. A peer review of the draft PRA was conducted by Electric Power Research Institute's Nuclear Safety Analysis Center (NSAC) in May 1983. The final study, which incorporated the comments of the peer review, was completed in July 1984 and resulted in an internal Duke report as Revision 0 to the PRA. In January 1988, Duke Power Company initiated a complete review and update of the original study.

On November 23, 1988, the NRC issued Generic Letter 88-20, which requested that licensees conduct an Individual Plant Examination (IPE) in order to identify potential severe accident vulnerabilities at their plants. The McGuire response to GL 88-20 was provided by letter dated November 4, 1991. McGuire's response included an updated McGuire PRA (Revision 1) study which was the culmination of the review and update which began in January 1988.

The McGuire PRA Revision 1 study and the IPE process resulted in a comprehensive, systematic examination of McGuire with regard to potential severe accidents. The McGuire study was again a full-scope, Level 3 PRA with analysis of both the internal and external events. This examination identified the most likely severe accident sequences, both internally and externally induced, with quantitative perspectives on likelihood and fission product release potential. The results of the study prompted changes in equipment, plant configuration and enhancements in plant procedures to reduce vulnerability of the plant to some accident sequences of concern.

As part of the Generic Letter 88-20 IPE process, the NRC conducted an audit of the human reliability analysis of the McGuire IPE during the period July 28 - 30, 1993. By letter dated June 30, 1994, the NRC provided a Staff Evaluation of the internal events portion of the above McGuire IPE submittal which included the results of the human reliability analysis audit. The conclusion of the NRC letter [page 15] states:
"The staff finds the licensee's IPE submittal for internal events including internal flooding essentially complete, with the level of detail consistent with the information requested in NUREG-1335. Based on the review of the submittal, and audit of "tier 2" supporting information, the staff finds reasonable the licensee's IPE conclusion that no severe accident vulnerabilities exist at McGuire."

In response to Generic Letter 88-20, Supplement 4, Duke completed an Individual Plant Examination of External Events (IPEEE) for severe accidents. This IPEEE was submitted to the NRC by letter

dated June 1, 1994. The report contained a summary of the methods, results and conclusions of the McGuire IPEEE program. The IPEEE process and supporting McGuire PRA included a comprehensive, systematic examination of severe accident potential resulting from external initiating events. By letter dated February 16, 1999, the NRC provided an evaluation of the IPEEE submittal. The conclusion of the NRC letter [page 6] states:

"On the basis of the overall review findings, the staff concludes that: (1) the licensee's IPEEE is complete with regard to the information requested by Supplement 4 to GL 88-20 (and associated guidance in NUREG-1407), and (2) the IPEEE results are reasonable given the MNS design, operation, and history. Therefore, the staff concludes that the licensee's IPEEE process is capable of identifying the most likely severe accidents and severe accident vulnerabilities, and therefore, that the MNS IPEEE has met the intent of Supplement 4 to GL 88-20 and the resolution of specific generic safety issues discussed in the SER."

In 1997, McGuire initiated Revision 2 of the 1991 IPE and provided the results to the NRC in 1998. Revision 3 of the McGuire PRA was completed in July 2002 and Revision 3a was completed in February 2005. Revision 3 was a comprehensive revision to the PRA models and associated documentation. The objectives of this update were 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 areas for improvement identified in the recent McGuire PRA Peer Review.

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

Revision 3a was a minor change to merge the Containment Air Return and Hydrogen Mitigation fault trees into the simplified LERF fault tree.

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.

Approved workplace procedures address the quality assurance of the PRA. One way the quality assurance of the 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.

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.

In January 2001, an enhanced manual 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. This process was further enhanced in July 2002 with the implementation of an electronic PRA change tracking tool.

Peer Review Process

Between October 23-27, 2000, McGuire participated in the Westinghouse Owners Group (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 (NEI) Risk Applications Task Force. The resulting industry document, NEI-00-02, describes the overall PRA peer review process. The Certification/Peer Review process is also linked to the ASME PRA Standard.

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 McGuire 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 McGuire PRA or IPE.

A summary of some of the McGuire PRA strengths and recommended areas for improvement from the peer review are as follows:

Strengths

- Good Summary Report write-up with insights
- Good system notebooks
- Rigorous Level 2 & 3 PRA Model
- Integrated internal and external events model
- Up-to-date plant database using Maintenance Rule
- Ongoing PRA staff interaction with plant staff, plant staff reviews
- PRA personnel knowledge of plant good

Recommended Areas for Improvement

- Better integration of sequences and recoveries within quantification process needed
- Need to review treatment of events requiring time-phasing in the modeling

- Better approach to closing the loop on PRA update items (tracking of errors/mods) needed
- More thorough, systematic approach to HRA screening values and common cause modeling needed
- Need an approach for reconciling realistic LERF model with NRC expectations from simplistic LERF modeling
- Need to update the PRA model to be more in line with current practices and expectations for state-of-the-art PRA

The significance levels of the 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.

Based on the PRA peer review report, the McGuire PRA received six Fact and Observations (F&O) with the significance level of "A" and 31 F&O with the significance level of "B." All six of the "A" F&O have been resolved and changes have been incorporated into McGuire PRA Revision 3a, the current PRA model. The "B" F&O have been reviewed and prioritized for incorporation into the PRA. Twelve of the "B" F&O have already been incorporated into Revision 3a of the PRA.

It is expected that the remaining F&O will be resolved and incorporated into Revision 4 of the PRA. The 19 remaining "B" F&O were reviewed with respect to the impact on the PRA and were determined to be insignificant with respect to this technical specification change.

PRA Model

The McGuire PRA is a full scope PRA including both internal and external events. The model includes the necessary initiating events (e.g., LOCAs, transients) to evaluate the frequency of accidents. The previous reviews of the McGuire PRA, NRC and peer reviews have not identified deficiencies related to the scope of initiating events considered.

The McGuire PRA includes models for those systems needed to estimate core damage frequency. These include all of the major

support systems (e.g., ac power, service water, component cooling, and instrument air) as well as the mitigating systems (e.g., emergency core cooling). These systems are modeled down to the component level, pumps, valves, and heat exchangers. This level of detail is sufficient for this application.

Results of Reviews with Respect to this LAR

A review of the analyses (cut sets and pertinent accident sequences) was made for accuracy and completeness. Specifically, cut sets generated for the solutions were screened and invalid cut sets were removed and appropriate recovery events applied. This process was documented in a Duke calculation. The review verified that the calculation adequately modeled the effects of the extended EDG 1A unavailability. Consistent with the work place procedures governing PRA analysis, this calculation has undergone independent checking by a qualified reviewer.

Tier 2 Assessment: Avoidance of Risk-significant Plant Equipment Outage Configurations

Tier 2 provides reasonable assurance that risk-significant plant equipment outage configurations will not occur when specific plant equipment is out of service consistent with the proposed TS change. Specific mitigating actions to be taken as a result of the proposed Technical Specification change are discussed in the "Operation and Maintenance Restrictions" portion of this submittal.

Duke has several Work Process Manual procedures and Nuclear System Directives that are in place at McGuire Nuclear Station to ensure that risk-significant plant configurations are avoided. The key documents are as follows:

- Nuclear System Directive 415, "Operational Risk Management (Modes 1-3) per 10 CFR 50.65 (a.4)," Revision 4, May 30, 2007.

- Nuclear System Directive 403, "Shutdown Risk Management (Modes 4, 5, 6, and No-Mode) per 10 CFR 50.65 (a.4)," Revision 16, November 1, 2006.

- Work Process Manual, WPM-609, "Innage Risk Assessment Utilizing ORAM-SENTINEL," Revision 8, June 2004.

- Work Process Manual, WPM-608, "Outage Risk Assessment Utilizing ORAM-SENTINEL," Revision 7, June 2004.

The program uses a blended approach of quantitative and qualitative evaluation of each configuration assessed. The McGuire on-line computerized risk tool, ORAM-SENTINEL, considers both internal and external initiating events with the exception of seismic events. Thus, the overall change in plant risk during maintenance activities is expected to be addressed adequately in accordance with RG 1.177 considering the proposed Technical Specifications.

Tier 3 Assessment: Maintenance Rule Configuration Control

10 CFR 50.65(a)(4), RG 1.182, and NUMARC 93-01 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 requires utilities to assess and manage the risks that occur during the performance of outages.

As stated above, Duke has approved procedures and directives in place at McGuire to ensure the requirements of the Maintenance Rule are implemented. These documents are used to address the Maintenance Rule requirements, including 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 McGuire are based on a "blended" approach of probabilistic 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.

Impact of PRA Analysis on Fire and Flooding Events

None of the fire initiating events that result in a loss of offsite power impact the diesel generators. The frequencies for these fires are much smaller than the loss of offsite power and tornado initiating event frequencies. The fire initiating events are not significant contributors to the increase in CDF associated with the DG 1A unavailability. The internal flooding events do not result in a loss of offsite power and are not important to this application.

Grid Reliability

The System Operating Center/Transmission Control Center (grid operator) has been notified to put actions in place to limit work activities that could affect the McGuire Switchyard for the duration of the 1A EDG repairs. This includes work activities outside the McGuire Switchyard.

In actions taken in response to Generic Letter 2006-02, "Grid Reliability and the Impact on Plant Risk and the Operability of Offsite Power," protocols have been put in place to improve communications between grid operators and McGuire operating staff.

Adverse weather procedures are in place for meteorological conditions which could potentially affect offsite power availability.

Operation and Maintenance Restrictions for the duration of the extension

Currently, the A Train of Control Room Ventilation and Chilled Water system (VC/YC) is out of service for scheduled work. Once the A Train of VC/YC is returned to operable status, it will

remain aligned to Unit 2 power to preclude adding further risk to the 1A Train Essential Switchgear.

The Switchyard is a controlled access area. All elective work has been suspended and will not resume until the 1A EDG has been returned to operable status.

As a further enhancement to the communications protocols implemented as part of GL 2006-02 response, daily communications will take place between McGuire Operations and the Grid Operator.

Routine essential equipment rotations during the duration of the extension on both units will not occur as scheduled due to the problem with the 1A EDG. This action will prevent any challenges to the offsite power source to 1ETA by not placing additional loads on the normal incoming breaker. Elective maintenance and testing during the allowed outage time extension will be rescheduled for both Units as warranted minimizing the risk of Unit transients.

In addition, the following equipment will be protected by plant procedure:

4160 Essential Bus (1ETA)	1A/2A Busline
6900/4160 Auxiliary Transformer (1ATC)	6900/4160 Auxiliary Transformer (SATA)
4160 Essential Bus (1ETB)	1B/2B Busline
6900/4160 Auxiliary Transformer (1ATD)	6900/4160 Auxiliary Transformer (SATB)
Unit 1 Transformer Yard	Unit 2 Transformer Yard
Switchyard	Standby Shutdown Facility (SSF)
Unit 1 Auxiliary Feedwater (CA) Pumps	Unit 1 Nuclear Service Water (RN) Pumps
1B Chemical & Volume Control (NV) Pump	1B Residual Heat Removal (ND) Pump
1B Safety Injection (NI) Pump	1B Containment Spray (NS) Pump
Unit 1 Component Cooling (KC) Pumps	B Train Control Area HVAC/Chilled Water (VC/YC)
1B Emergency Diesel Generator (EDG)	Instrument Air Compressors (VI)

To minimize the risk of losing offsite power to the 1A 4.16 kV Essential Bus, Technical Specification Surveillances for 3.3.5.1 will not be performed for the 1A undervoltage and degraded voltage relaying. As a result, the TS Surveillance for 3.3.5.1 "Loss of Power EDG Start Instrumentation" will expire during this period of 1a EDG unavailability. This surveillance will be performed once 1A EDG is returned to an available status.

10. REGULATORY SAFETY ANALYSIS:

10.1 No Significant Hazards Consideration:

Duke Energy Carolinas, LLC (Duke) has concluded that operation of McGuire Nuclear Station Unit 1 in accordance with the proposed change to the Technical Specifications (TS) does not involve a significant hazards consideration. Duke's conclusion is based on its evaluation, in accordance with 10 CFR 50.91(a)(1), of the three standards set forth in 10 CFR 50.92(c).

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

Response: No.

The 1A EDG functions as an accident mitigator and is not required unless an accident occurs. The 1A EDG does not affect any accident initiators or precursors. The proposed extension of the allowed outage time (AOT) does not affect the 1A EDG's interaction with any system whose failure or malfunction could initiate an accident. Therefore, the probability of an accident previously evaluated is not significantly increased.

The 1A EDG functions to mitigate a loss of offsite power to vital components. The risk evaluation performed in support of this amendment request demonstrates that the consequences of an accident are not significantly increased. As such, the proposed change do not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

Response: No.

The proposed amendment does not involve the addition, removal or modification of any plant system, structure or component. The proposed change

will not affect the operation of any plant system, structure or component as directed in plant procedures. Operation of the facility in accordance with this amendment does not create the possibility of a new or different kind of accident from those previously evaluated.

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

Response: No.

Based upon the availability of redundant systems, the mitigating actions that have been taken and the low probability of an accident, McGuire concludes that the reduction in availability of the 1A EDG does not result in a significant reduction in the margin of safety.

The margin of safety is related to the confidence in the ability of the fission product barriers to perform their design functions during and following an accident situation. These barriers include the fuel cladding, the reactor coolant system, and the containment system. The performance of the fuel cladding, containment and the reactor coolant system will not be significantly impacted by the proposed change.

Thus, it can be concluded that the proposed change does not involve a significant reduction in the margin of safety.

10.2 Applicable Regulatory Requirements/Criteria:

The analysis presented in this LAR demonstrates that McGuire will remain in compliance with the applicable regulations and requirements. These are: 10CFR50, Appendix A, General Design Criterion (GDC) 17, This LAR is being submitted in accordance with 10 CFR 50.90 and 10 CFR 50.91(a)(5).

11. Environmental Consideration:

The proposed change does not involve a significant hazards consideration, a significant change in the types of or significant increase in the amounts of any effluents that may be

released offsite, or a significant increase in individual or cumulative occupational radiation exposure. Therefore, the proposed changes meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), an environmental assessment of the proposed changes is not required.

12. Precedent:

NRC approval of D. C. Cook Nuclear Plant, Unit 2 license amendment request of December 9, 2003 requesting a one-time limited duration exception from the allowed outage time (AOT) for the Unit 2AB EDG, Amendment 264, dated December 10, 2003, and the conclusions of the associated NRC Safety Evaluation Report.

NRC approval of Browns Ferry Nuclear Plant, Unit 3 license amendment request of April 6, 2007 requesting a one-time limited duration exception from the allowed outage time (AOT) for the Unit 3D EDG, Amendment 257, dated April 6, 2007, and the conclusions of the associated NRC Safety Evaluation.

ATTACHMENT 2

MARKED UP McGUIRE UNIT 1 TECHNICAL SPECIFICATION

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. (continued)	B.4 Restore DG to OPERABLE status.	72 hours * AND 6 days from discovery of failure to meet LCO *
C. Two offsite circuits inoperable.	C.1 Declare required feature(s) inoperable when its redundant required feature(s) is inoperable.	12 hours from discovery of Condition C concurrent with inoperability of redundant required feature(s)
	AND	
	C.2 Restore one offsite circuit to OPERABLE status.	24 hours

(continued)

* For Unit 1 only, the Completion Time that the 1A EDG can be inoperable as specified by Required Action B.4 may be extended beyond the "72 hours and 6 days from discovery of failure to meet the LCO" up to a total of 10 days as part of the 1A EDG Jacket/Intercooler Water Pump Motor repair. Upon completion of the repair and restoration, this footnote is no longer applicable and will expire at 1741 hours on June 15, 2007.

ATTACHMENT 3

RE-TYPED McGUIRE UNIT 1 TECHNICAL SPECIFICATION

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. (continued)	B.4 Restore DG to OPERABLE status.	72 hours * AND 6 days from discovery of failure to meet LCO *
C. Two offsite circuits inoperable.	C.1 Declare required feature(s) inoperable when its redundant required feature(s) is inoperable.	12 hours from discovery of Condition C concurrent with inoperability of redundant required feature(s)
	AND	
	C.2 Restore one offsite circuit to OPERABLE status.	24 hours

(continued)

* For Unit 1 only, the Completion Time that the 1A EDG can be inoperable as specified by Required Action B.4 may be extended beyond the "72 hours and 6 days from discovery of failure to meet the LCO" up to a total of 10 days as part of the 1A EDG Jacket/Intercooler Water Pump Motor repair. Upon completion of the repair and restoration, this footnote is no longer applicable and will expire at 1741 hours on June 15, 2007

ATTACHMENT 4

REGULATORY COMMITMENTS

LIST OF REGULATORY COMMITMENTS:

The following table identifies those actions committed to by McGuire in this document, for the duration of the extension. Any other statements made in this licensing submittal are provided for informational purposes only and are not considered to be regulatory commitments. Please direct any questions you may have in this matter to K. L. Ashe at (704) 875-4535.

Regulatory Commitment	**Due Date**
Once the A Train of the Control Room Ventilation and Chilled Water systems (VC/YC) is returned to operable status, it will remain aligned to Unit 2 power to preclude adding further risk to the 1A Train Essential Switchgear.	June 15, 2007
The Switchyard is a controlled access area. All elective work has been suspended and will not resume until the 1A EDG has been returned to operable status.	June 15, 2007
As a further enhancement to the communications protocols implemented as part of GL 2006-02 response, during the duration of the extension, daily communications will take place between McGuire Operations and the Grid Operator.	June 15, 2007

Regulatory Commitment	Due Date
Routine essential equipment rotations during the duration of the extension on both units will not occur as scheduled due to the problem with the 1A EDG. This action will prevent any challenges to the offsite power source to 1ETA by not placing additional loads on the normal incoming breaker. Elective maintenance and testing during the allowed outage time extension will be rescheduled for both Units as warranted minimizing the risk of Unit transients.	June 15, 2007
The following equipment will be protected: 1ETA, 1ETB, 1A/2A Busline, 1B/2B Busline, 1ATC, 1ATD, SATA, SATB, U1 Transformer Yard, U2 Transformer Yard, Switchyard, Standby Shutdown Facility, U1 CA Pumps, Unit 1 RN Pumps, 1B NV Pump, 1B ND Pump, 1B NI Pump, 1B NS Pump, Unit 1 KC Pumps, B Train VC/YC, 1B EDG, and VI Compressors.	June 15, 2007
To minimize the risk of losing offsite power to the 1A 4.16 kV Essential Bus Technical Specification Surveillances for 3.3.5.1 will not be performed for the 1A undervoltage and degraded voltage relaying.As a result, the TS Surveillance for 3.3.5 "Loss of Power EDG Start Instrumentation" will expire during this period of 1a EDG unavailability. This surveillance will be performed once 1A EDG is returned to an available status.	June 15, 2007