Duke Energy®

REGIS T. REPKO
Vice President
McGuire Nuclear Station

Duke Energy
MG01VP / 12700 Hagers Ferry Rd.
Huntersville, NC 28078

980-875-4111
980-875-4809 fax
regis.repko@duke-energy.com

April 5, 2011

U. S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555-0001

Subject: Duke Energy Carolinas, LLC (Duke Energy)
McGuire Nuclear Station, Units 1 and 2
Docket Numbers 50-369 and 50-370
Summary Report of Evaluations Performed Pursuant to 10 CFR 50.59
Changes, Tests, and Experiments

Pursuant to 10 CFR 50.59(d)(2), attached is a summary report of evaluations performed at McGuire Nuclear Station for the period from January 1, 2010 to December 31, 2010. These evaluations demonstrate that the associated changes do not meet the criteria for license amendments as defined by 10 CFR 50.59(c)(2).

This submittal document contains no regulatory commitments.

If there are any questions or if additional information is needed, please contact M. K. Leisure at (980) 875-5171.

Sincerely,

Regis T. Repko

Regis T. Repko

Attachment

IE47
NRR

xc:

V. M. McCree
Regional Administrator, Region II
U.S. Nuclear Regulatory Commission
Marquis One Tower
245 Peachtree Center Ave. NE, Suite 1200
Atlanta, GA 30303-1257

J. H. Thompson (addressee only)
NRC Project Manager
U.S. Nuclear Regulatory Commission
Mail Stop O-8 G9A
11555 Rockville Pike
Rockville, MD 20852-2738

J. B. Brady
NRC Senior Resident Inspector
McGuire Nuclear Station

McGuire Nuclear Station (MNS)
Changes Evaluated Under 10 CFR 50.59

Main Generator Protective Relaying Upgrade
Modification MD100165 (Unit 1)
(Action Request No. 00243703)

The scope of this modification was to improve the reliability, security and monitoring capabilities of the protective relaying for the main generators by replacing the existing protective relays with a set of multifunction microprocessor-based relays. In addition to replacing the protective functions for the existing relays, the new microprocessor-based relays include protective functions that are not provided by the existing relaying scheme.

An evaluation of the proposed protective relaying modification demonstrated that the change had no adverse impact on any systems, structures, or components (SSCs) with accident mitigation functions, and that the American Nuclear Society (ANS) Condition II transients that may be initiated by the protective relaying remain bounded by the present UFSAR analysis.

New potential failure modes and increased susceptibility to existing failures were evaluated. These include common mode software failure, failures resulting from changing the coincident trip logic, and electromagnetic interference-induced or radio frequency interference-induced failures. When considering the features of redundant trains, independent channels, diversity, self-diagnostics, and improved human-machine interfaces, the proposed relaying scheme proves to be as reliable and more secure than the existing system.

The proposed protective relaying scheme contains protective functions that are not provided with the existing scheme. These include inadvertent energization, reverse power breaker failure and out-of-step protection. Each of the new protective features was evaluated to ensure the proposed changes would not lead to additional challenges or otherwise adversely impact SSCs that serve to mitigate transients initiated by protective relay action.

The evaluation demonstrates that the main generator protective relaying modification preserves the current licensing basis. The activity does not create more than a minimal increase in the frequency or consequences of accidents or malfunctions of SSCs important to safety. The activity does not create the potential for a new type of unanalyzed event, has no impact on the fission product barriers, and does not affect evaluation methodology. Therefore under 10 CFR 50.59, it is permissible to implement this modification without prior approval from the NRC.

McGuire Unit 1 Cycle 21 (M1C21) Reload Design
(Action Request No. 00302161)

This activity installed the core designed for McGuire Nuclear Station (MNS) Unit 1 Cycle 21 (M1C21). The M1C21 Reload Design Safety Analysis Review (REDSAR), performed in accordance with Engineering Directives Manual EDM-501, "Engineering Change Program for Nuclear Fuel", and the M1C21 Reload Safety Evaluation confirm that the MNS Updated Safety Analysis Report (UFSAR) accident analyses remain bounding with respect to predicted M1C21 safety analysis physics parameters (SAPP), and fuel thermal and mechanical performance limits. The SAPP method is described in topical report DPC-NE-3001-PA, "Multidimensional Reactor Transients and Safety Analysis Physics Parameters Methodology."

The M1C21 core reload is similar to past cycle core designs, with a design generated using approved methods. The M1C21 Core Operating Limits Report (COLR) was prepared in accordance with Technical Specification 5.6.5. Additionally, applicable Technical Specifications and the UFSAR have been reviewed and no changes are required for the operation of M1C21. This 10CFR50.59 evaluation concluded that no prior NRC approval is necessary for M1C21 operation.

Upgrade W-7300 Process Control System with Ovation DCS
Modifications MD100242 (Unit 1)
(Action Request Nos. 00253414, 00304908, and 00305014)

The control (non-safety related) portion of the 7300 Process Instrumentation and Control System was replaced with a modern digital Distributed Control System (DCS) using Emerson/Westinghouse Ovation equipment. Ovation is commercial off-the-shelf (COTS) technology that provides a secure open-systems architecture. The Ovation platform is extendable, and incorporates interfaces with most widely adopted communications bus standards, allowing it to interface with "smart" technologies, e.g. Highway-Accessible Remote Transducer (HART).

The new DCS is capable of performing the same control and monitoring functions as the existing system. In addition, the DCS provides advanced functions commonly found in a modern DCS, such as graphical displays, trending, logging, soft controls, more sophisticated control algorithms, and control algorithm detuning. The DCS provides current and historical plant data to the operator aid computer (OAC) and to the MNS site local area network (LAN) via a secure data server that meets the appropriate industry standards and regulatory guidance for cyber security.

The new DCS is more fault-tolerant than the legacy 7300 system being replaced. The design includes redundant controller pairs, redundant communication buses, and redundant power supplies. In key control loops, additional field devices are added to increase the level of redundancy among the sensor inputs. An automatic signal selection of process variables (e.g., median select) validates input signals and alarms when a faulty input is detected. Diagnostic capabilities for both internal and external field devices add improved reliability. Instrument channels installed in the reactor protection cabinets that no longer perform a protection function are relocated to the DCS. The potential for undesirable control and protection system interaction has been reduced.

The new DCS automates several tasks currently performed manually in the control room. Manual channel selector switches are replaced with automatic algorithms, relieving the control room operator of manual operator actions when a control loop fails. Paper chart recorders were replaced with computer-based trending and graphic user interfaces, reducing the clutter on the control board and the maintenance expenditure on chart recorders. The risk of operator errors causing an event has been minimized in the DCS design by following the appropriate human factors engineering standards and regulatory guidance in the development of work station screens and displays.

The new DCS has been evaluated for potential failure modes with new or different consequences. The failures considered include those unique to digital systems and microprocessors, e.g. electro-magnetic and radio frequency interference, electro-static discharges, software configuration management, cyber security threats, and other software common cause failure mechanisms. The evaluations concluded that the risk of such failures is minimal and the potential consequences are acceptable.

Revision of UFSAR Section 6.2.1.2 and Table 6-53
Engineering Change 103500
(Action Request No. 00307710)

This evaluation addresses revisions to the evaluation methodology described in UFSAR Section 6.2.1.2. The Transient Mass Distribution (TMD) code was utilized to determine if the reactor cavity walls could maintain their structural integrity during the short pressure pulse which accompanies a high-energy line rupture within the sub-compartment. Although the TMD code was utilized for the initial evaluation that was done during the initial licensing of MNS, the code and model had to be rebuilt since the original code and analysis are obsolete and cannot be re-run. This is considered a change to a method of evaluation described in the UFSAR.

The changes associated with the TMD code do not result in a departure from a method of evaluation since the revised method provides essentially the same results as the UFSAR described evaluation methodology. The results from the Westinghouse assessment in which the UFSAR results of the peak pressure in the reactor cavity sub-compartment with no covers in place compare to the analysis of the same configuration with the rebuilt code are provided in Table 1, below. Case 1 represents the results from the MNS UFSAR, as determined at the initial licensing of MNS. Case 2 gives the peak pressures of the rebuilt TMD model based on what the analyst believed were the inputs and model that were used in support of the initial Licensing effort for MNS. Case 3 represents the UFSAR rebuilt model and inputs based on how the analyst would perform the analysis for MNS today.

In the event that changes to a method of evaluation are such that the results move in the non-conservative direction, the change can be made without prior NRC approval, provided the revised result is "essentially the same" as the previous result. The results, as documented in Table 1, are "essentially the same" since the analysis performed are within the constraints and limitations associated with the topical report and NRC SER for the TMD computer program. The analytical solution for the TMD mathematical model used for the revised analysis is developed by considering the conservation equations of mass, momentum, and energy and the equation of state, together with the control volume technique for simulating spatial variation. The governing equations for TMD are a two-phase (liquid water droplets and steam – air vapor) two-component (air-water) system which is solved in determining local properties.

Table 1
Reactor Cavity Peak Pressures

Node in TMD Model	Design Pressure (psig)	Peak Pressures (psig)		
		Case 1	Case 2	Case 3
Upper Reactor Cavity (47)	10.6	5.3	5.098	5.364
Lower Reactor Cavity (2)	32	3.2	3.044	3.53
Reactor Annulus (5-20 and 36-44)	140	55.8	55.723	55.713
Reactor Pipe Sleeve (54)	1120	221.2	220.383	220.349
Inspection Shaft (53)	400	226.1	225.251	225.220
Inspection Cavity (1)	400	256.7	256.116	256.077