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August 30, 2001

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

Subject: McGuire Nuclear Station, Units 1 and 2
Docket Nos. 50-369, 370
Response to Request for Additional Information
Application for Administrative Amendments to Facility Operating Licenses NPF-9 and NPF-17
(TAC Nos. MA9297 and MA9298)

By letter dated July 30, 2001, the NRC staff requested additional information related to Duke Energy Corporation's (DEC) application for administrative amendments to Facility Operating Licenses NPF-9 and NPF-17. Attachment 1 contains Duke's response to the subject request.

Attachment 2 contains administrative changes to include revision numbers on Security related documents per License Condition 2.E for Units 1 and 2. These pages were revised as per telecon with Mr. Robert E. Martin dated August 22, 2001.

Questions regarding this submittal should be directed to Kay Crane, McGuire Regulatory Compliance at (704) 875-4306.

HB Barron

H. B. Barron

A001

cc: Mr. Robert E. Martin, Project Manager
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H. B. Barron, being duly sworn, states that he is Vice President of McGuire Nuclear Station; that he is authorized on the part of Duke Energy Corporation to sign and file with the Nuclear Regulatory Commission this additional information in support of revisions to the McGuire Nuclear Station Facility Operating License Nos. NPF-9 and NPF-17; and, that all statements and matters set forth therein are true and correct to the best of his knowledge.

HB Barron

H. B. Barron, Vice President
McGuire Nuclear Station
Duke Energy Corporation

Subscribed and sworn to before me this 30th day of August, 2001.

Deborah G. Thrap, Deborah G. Thrap
Notary Public

My Commission Expires: 4/6/2002

Attachment 1

McGuire Nuclear Station
Response to Request for Additional Information

LC 2.C (11)(h) and 2.C (10)(e), Hydrogen Control Measures for Units 1 and 2

Restatement of Request

The McGuire Facility Operating Licenses (FOL) each contain a License Condition (LC) on hydrogen control measures. The acceptability of the hydrogen mitigation system for the issuance of the FOL for McGuire Unit 2 is discussed in Supplement Number 7 (SSER-7) to the Safety Evaluation Report (NUREG-0422), issued in May 1983. SSER-7 evaluated a 66 igniter design. Section C.2.3, Igniter Coverage, discusses the need for two additional igniters in lower elevations of the lower compartment and four additional igniters in the lower portion of the upper compartment for a total of 72 igniters, and these requirements were included in LC 2.C.10(e)(1)(a). The staff's conclusions in SSER-7 were as follows:

> Accordingly, subject to meeting the conditions discussed herein dealing with igniter number, locations, system status indication and subject to completing installation, the staff finds the McGuire Units 1 and 2 license conditions dealing with hydrogen control during postulated degraded core accidents to be satisfactorily resolved.

SSER-7 discussed the background and bases for the inclusion of the elements of the McGuire Unit 2 LC on hydrogen control measures. Following a discussion of these issues, the staff indicated in section C.9 of McGuire's SSER-7 that it had identified a number of technical concerns that it would continue to investigate as confirmatory items. These issues were discussed in Supplements 5 and 6 of the Safety Evaluation Report (NUREG-0954) for the Catawba Nuclear Station. By letter dated May 26, 1993, "Closure of the Hydrogen Control Issue Pursuant to 10 CFR 50.44 for Catawba and McGuire Nuclear Stations," the NRC staff reported on the closure of those issues.

The staff finds that reference by Duke Energy Corporation (DEC) in its June 13, 2000 submittal to resolution of these further confirmatory issues in the Catawba SSER is not sufficient to establish the basis for the satisfaction and deletion of the elements of the McGuire Unit 2 LC on hydrogen control measures. Therefore, DEC is requested to provide a detailed discussion of the measures taken to satisfy each of the elements of the McGuire Unit 2 LC on hydrogen control measures.

DEC's response should include information of a specificity comparable to that provided in DEC's report, "An Analysis of Hydrogen Control Measures at McGuire Nuclear Station," as submitted in October 30, 1981, and subsequently revised. This should include reference to the dates of modifications and the number and placement of added igniters.

Clarification should also be provided regarding the total number of igniters in the system. SSER-7 evaluated a 66-igniter system, and the LC required six additional igniters to be installed. However, the system currently described in the Updated Final Safety Analysis Report includes only 70 igniters.

Response:

As noted in the Request for Additional Information, SSER-7 evaluated a 66-igniter design. Section C.2.3, Igniter Coverage, discussed the need for two additional igniters in lower elevations of lower containment and four additional igniters in the lower portion of the upper compartment for a total of 72 igniters. Nuclear Station Modification MG1-1253 (Unit 1) was completed on April 23, 1984. Nuclear Station Modification MG2-0208 (Unit 2) was completed on April 12, 1985. These modifications added the following six igniters to the Hydrogen Mitigation System (EHM) for Units 1 and 2:

Igniter (EHM)	Location	
TB 67	725-91° - 17-BY PR TANK	Lower elevation, Lower containment
TB 68	725-89° - 17-BY PR TANK	Lower elevation, Lower containment
TB 69	824-40° -33-TOP OF SG-C	Lower elevation, Upper Containment
TB 70	824-139° -33-TOP OF SG-B	Lower elevation, Upper Containment
TB 71	824-218° -33-TOP OF SG-A	Lower elevation, Upper Containment
TB 72	824-322° -33-TOP OF SG-D	Lower elevation, Upper Containment

Notification of completion of the modifications was submitted to the NRC staff in the annual 10 CFR 50.59 report by letters dated July 1, 1985 [1](Unit 1) and July 1, 1986[2] (Unit 2). Following completion of these modifications, the number of igniters totaled 72.

In 1987, a concern was expressed by the stations (McGuire and Catawba) related to the hydrogen igniters located near the reactor vessel in the incore instrument tunnel. These igniters were an ALARA concern because of their need for testing and maintenance in a high radiation area. In response to this concern, the basis for these igniters was reviewed resulting in removal under the Nuclear Station Modification Process (10 CFR 50.59). The modification packages contain the following basis for deletion:

"Because of advancements in understanding both degraded core accident behavior and hydrogen release and distribution in containment, we believe that the igniters located in the incore instrument tunnel are no longer needed. The following reasons justify this position:

[1] July 1, 1985, Hal B. Tucker to Dr. J. Nelson Grace, McGuire Nuclear Station, Docket Nos. 50-369, 50-370.
[2] July 1, 1986, Hal B. Tucker to Dr. J. Nelson Grace, McGuire Nuclear Station, Docket Nos. 50-369, 50-370.

1. Hydrogen release during the type of degraded core accidents used as design basis for the Hydrogen Mitigation System would not produce burnable concentrations in the incore instrumentation tunnel.

2. Even if hydrogen were to somehow get into the incore tunnel, it would collect near the seal table or near the top of the primary seal wall. These areas are covered by other igniters.

3. Degraded core accidents more severe than the design basis for the igniters could release significant amounts of hydrogen into the incore instrumentation tunnel when the reactor vessel fails. However, the high temperatures and pressures generated by the vessel blowdown would fail any igniters in that area. Additionally, all oxygen in the cavity would be expelled and would prevent hydrogen burns even if the igniters survived."

Based on the information provided above, it was concluded that these igniters were not needed to mitigate the effects of hydrogen generation and release to containment during a degraded core accident.

Specifically: 1EHMTB-1, 1EHMTB-2 for Unit 1 and 2EHMTB-1, 2EHMTB-2 for Unit 2 were removed from service via the Nuclear Station Modification (MEVN-946 completed September 14, 1987 and MEVN-947 completed July 28, 1988) process. The deletion of these two igniters brought the total number of igniters to 70 as stated in the UFSAR.

McGuire Technical Specification Bases 3.6.9, "Hydrogen Mitigation System (HMS)" states in part, "A total of 70 ignitors are distributed throughout the various regions of containment in which hydrogen could be released or to which it could flow in significant quantities. The ignitors are arranged in two independent trains such that each containment region has at least two ignitors, one from each train, controlled and powered redundantly so that ignition would occur in each region even if one train failed to energize."

UFSAR Section 6.2.7, "Supplemental Hydrogen Control System/Hydrogen Mitigation System" provides a description of the system. "Igniter Coverage" states in part "The hydrogen mitigation system consists of 70 igniter assemblies distributed throughout the upper, lower, dead-ended, and ice condenser compartments."

Periodic Test Procedures, "Hydrogen Mitigation Igniter Glow Plug Test" (PT/1&2/A/4350/024) provides a safe and correct method for testing of the Hydrogen Mitigation Igniters, and establishing annual base current values, for meeting the requirements of the Technical Specifications. Enclosure 13.4, "Glow Plug Temperature Measurement," (Attachment 3) contains a list of the 70 igniters as well as their locations.

On September 30, 1998[3], the staff issued License Amendment 184/166. This amendment reflected full conversion to "Standard Technical Specifications – Westinghouse Plants," The Safety Evaluation Report, page 41, item 10, "Hydrogen Mitigation System – Revised Minimum Number of Required Hydrogen Igniters" states in part: "CTS 4.6.4.3.a requires that 32 of 33

[3] September 30, 1998, Frank Rinaldi to Mr. H. B. Barron, Issuance of Amendments – McGuire Nuclear Station, Units 1 and 2 (TAC Nos. M98964 and M98965).

hydrogen igniters be operable on each train. Corresponding ITS SR 3.6.9.1 requires 34 igniters per train to be operable. The actual design contains 35 igniters per train. This change is administrative because it corrects an inadvertent error in the CTS and is consistent with current operation of the system. The correct number of igniters was increased as discussed in MNS SER Supplement 7, Attachment C, after the first refueling outage of each unit. This change corrects the TS with the approved licensing basis as described in the SER supplement. Therefore, this change is acceptable.

LC.2.C (4), Thermal Sleeves for McGuire Unit 2

Restatement of Request:

LC.2.C(4) required that DEC provide a report justifying operation with thermal sleeves removed from selected locations in the reactor coolant system. DEC responded by letter dated May 13, 1983, providing the results of evaluations to support continued operation without the subject thermal sleeves installed. NRC's letter and safety evaluation of December 30, 1986, concluded that continued operation was acceptable with the thermal sleeves permanently removed. The staff's acceptance recognized that the McGuire TS 3/4 3.5.2 required reporting of ECCS actuations and injections, and the usage factor of each nozzle whenever its value exceeds 0.70. DEC is requested to provide a discussion of how the nozzle usage factor is currently monitored and of DEC's planned actions in the event its value exceeds the 0.70 value.

Response

As noted in the Request for Additional Information, the NRC's letter and safety evaluation of December 30, 1986[4], concluded that continued operation with thermal sleeved removed from selected locations was acceptable recognizing that the former McGuire TS 3/4 3.5.2 required reporting of ECCS actuations and injections and the usage factor of each nozzle whenever its value exceeded 0.70.

During Improved Technical Specification (ITS) implementation (approved by License Amendment 184/166 dated September 30, 1998[5]) this reporting requirement was removed. It was concluded that the requirement was redundant to 10 CFR 50.73 (a) (2) (iv) requiring a 30 day report in the event of an ECCS actuation. Since the CFRs are directly enforceable and sufficient regulatory control was provided, the duplication was not necessary within the Technical Specifications. However, the usage factor of 0.70 was not addressed in the justification. A failure to meet this reporting requirement since ITS implementation has not occurred.

This issue has been entered into McGuire's corrective action program (PIP M01-3639). As a corrective action, McGuire commits to revise the Reactor Trip Investigation Procedure to include a requirement to ensure proper NRC notification is accomplished in the event the usage factor of affected safety injection nozzles exceeds the 0.70 value.

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[4] December 30, 1986, Darl Hood to Mr. H. B. Tucker, Reactor Coolant System Thermal Sleeves – McGuire Nuclear Station, Units 1 and 2.

[5] September 30, 1998, Frank Rinaldi to Mr. H. B. Barron, Issuance of Amendments – McGuire Nuclear Station, Units 1 and 2 (TAC Nos. M98964 and M98965).

Thermal fatigue is addressed in the McGuire Nuclear Station Fatigue Management Program by analyzing components with a postulated number of bounding transients that are commonly called design transients. A continual counting of the occurrences of the transients and a comparison of that count to the number used in design and analysis of the components confirm the bounding fatigue life for those components. The Safety Injection nozzles at McGuire Nuclear Station have been analyzed to the ASME Boiler and Pressure Vessel Code, Section III, Nuclear Power Plant Components, including all addenda through Winter, 1971. Using the ASME Code acceptance criteria, cumulative usage factors (CUF) must be less than 1.0. All McGuire Nuclear Station safety nozzles stresses and usage factors are less than 1.0 and are documented in the pertinent piping analysis calculations and stress reports. The number of occurrences of transients is monitored by Operations and Engineering and is documented and managed by the Thermal Fatigue Management Program. This program is a prevention program in that it seeks to preclude cracking due to low-cycle thermal fatigue. It is accomplished by continually showing that the severity and number of occurrences of the transients actually occurring are enveloped by the severity and number of occurrences of the analyzed transients.

McGuire Nuclear Station determines the usage factor by monitoring the number of occurrences of operating transient cycles and comparing this number to the allowed design cycles. The responsible engineer will identify plant transient conditions that require assessment. Should the thermal and pressure profile for a specific transient be outside the parameters defined for that transient set or should an allowable cycle count limit for a transient cycle set be approached or exceeded, the program requires that the responsible engineer identify the issue to the appropriate engineering groups for resolution within a manageable time period. A manageable time period is the time needed to complete actions to ensure the affected components stay within acceptable cycle count limits. Some of the corrective actions consist of reanalysis, procedure changes, modifications, and replacement of components.

Thermal fatigue transients have been tracked since operation began at McGuire Nuclear Station. Operating experience associated with the Thermal Fatigue Management Program has demonstrated that the program continues to monitor plant transients and track the accumulation of these transients consistent with the requirements in Technical Specification 5.5.6, "Component Cyclic or Transient Limit." McGuire's Problem Identification Program (PIP) documents the discovery of transients and corrective actions. It records appropriate corrective actions and tracks completion of those actions.

LC.2.C (7), Fire Protection for McGuire Unit 2

Restatement of Request:

DEC stated that LC 2.C(7), item (c) had been met on the basis that its letter of May 13, 1983, provided a schedule for installation of reactor coolant system cold-leg temperature (T-cold) monitors as part of the Standby Shutdown System and that a second letter provided a schedule for installation of source-range neutron flux instrumentation. However, these letters provide only a provisional schedule for the installation of this instrumentation and fail to document that the instrumentation was actually installed. Further, Selected Licensee Commitments, section

16.9.7, Table 16.9.7-1, does not include this instrumentation. DEC is requested to provide further justification for the proposed deletion of this item.

Response:

The instrumentation described in this license condition is physically located in the plant. The Standby Shutdown System reactor coolant loops A & D cold leg wide range temperature instruments, 1(2)NCDR5860 and 5940, have been installed since the 1980's. Wide Range Neutron Flux instrumentation (1(2)ENBPI9511) have been installed since approximately 1983.

The Design Basis Specification (DBD) for the Standby Shutdown System (MCS-1223.SS-00-0001) describes the instrumentation. Section 20.4.4.1.2, "Reactor Coolant Loop A and D Cold Leg Wide Range Temperature (1(2)NCRD5860 and 5930)" states: "These instruments shall provide a means of monitoring the Reactor Coolant System (NC) loop A and D cold leg water temperature per requirement found in Reference 20.5.1.3.3. Power supplies and output shall be swapped from the control room to the SSF (swapped in Auxiliary Bldg., A SWGR Room). This wide range temperature indication has a range of 0 - 700°F." Reference 20.5.1.3.3 refers the reader to McGuire Units 1 and 2 SER Supplement No. 6, Appendix C. DBD Section 20.4.4.5.1, "Wide Range Neutron Flux (1(2)ENBPI9511)" states: This indicator shall provide source range neutron flux indication on the SSFCP. This indication shows that the control rods have been inserted into the core (decreasing flux) and that hot standby is being maintained using a borated water source for primary side makeup. If an unborated makeup source was supplied to the primary coolant system, a boron dilution accident would occur (increasing flux). The display band range is 10^0 - 10^5 CPS.

The installation of this instrumentation meets the requirements of the subject license condition. With regards to SLC 16.9.7, the subject cold leg wide range temperature instrumentation and wide range neutron flux instrumentation will be added to Table 16.9.7-1 of SLC 16.9.7. These additions will be tracked via the McGuire corrective action program by PIP M01-3466.

LC.2.C (10) (c), Inadequate Core Cooling Instrumentation (ICCI) for McGuire Unit 2

Restatement of Request:

LC 2.C(10)(c), part (2) required that prior to startup following the first refueling outage, the licensee shall upgrade the in-containment portion of the incore thermocouple system (ITS) and provide a schedule for update of the remainder of the system. The NRC staff issued a safety evaluation on the system on September 17, 1984 that concluded that McGuire's proposed final ICCI was in compliance with the Item II.F.2 requirements and would be acceptable for Unit 2 upon upgrading the existing ICCI, implementation of the revised emergency operating procedures, and installation of and calibration of the reactor vessel level instrumentation system. DEC's letter of June 25, 1985, provided a status of the ICCI implementation of the installation, functional testing, and calibration of the ITS, as well as the implementation of the revised EOPs. DEC's June 25, 1985 letter stated that NRC approval of plant specific installation was "to be determined" for Unit 2. It also said a further DEC status report would be submitted. DEC is requested to provide further updating of the status of this information which demonstrates that the requirements of the LC have been met.

Response

McGuire is in compliance with this license condition. The Inadequate Core Cooling Instrumentation (ICCM) System is in place on both Units 1 and 2. The system monitor in the Control Room provides the Operators with a visual display and receives input from three subsystems: the Reactor Vessel Level Instrumentation System (RVLIS), the Core Exit Thermocouple Monitor (CETM), and the Subcooled Margin Monitor (SSM).

The final upgrade of the system was completed for Unit 1 on October 2, 1986 via Nuclear Station Modification (NSM) MG-1-1629 and for Unit 2 on July 21, 1986 via NSM MG-2-0491. This included upgrading the Incore Thermocouple monitoring to Class 1E, added Class 1E Core Subcooling Margin monitoring, deleted the RVLIS analog display with graphic displays, added a redundant Class 1E microprocessor, and added a complete system display monitor.

The ICCM system for both units is described in the McGuire UFSAR, Sections 1.8.27, "Inadequate Core Cooling Instruments (II.F.2)," and 7.5.4, "Inadequate Core Cooling Instrumentation." The ICCM system is also described in the Reactor Coolant System Design Basis Document.

The present ICCM system was fully described to the NRC by Duke letter dated November 24, 1986[6]. Additionally, the ICCM system was audited by NRC contractors on April 28, 1988 (see NRC letter dated May 27, 1988[7]).

All Emergency Operating Procedures that address Inadequate Core Cooling have been upgraded to the Westinghouse ERG revision 1C. Any EOP procedure revisions are accompanied by a deviation document describing any deviation and justification for deviating from ERG rev. 1C. The deviation documents are available upon request.

LC 2.C (10)(d), Anticipatory Reactor Trip for McGuire Unit 2

Restatement of Request:

The McGuire Unit 2 FOL includes an LC on Anticipatory Reactor Trip. Please provide test information that confirms that his test was completed as required by the LC and that the test acceptance criteria were met.

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[6] November 24, 1986, Hal B. Tucker to Mr. Harold R. Denton, McGuire Nuclear Station, Catawba Nuclear Station, Docket Nos. 50-369, -370, -413, -414, Reactor Coolant Pump Trip.

[7] May 27, 1988, Darl Hood to Duke Power Company, Summary of April 28, 1988 Meeting on Inadequate Core Cooling Instrumentation Systems (TACS 67755 and 67756).

Response:

By letter dated February 8, 1984[8], McGuire submitted the Unit 2 Startup Report. This report was prepared in accordance with the requirements of the Technical Specifications (formally Tech. Spec. 6.9.1) and addressed the results of startup testing from initial fuel loading through testing at the 90% full power level.

The Unit 2 Startup Report, Section 7.5, page 7.5-1, "Unit Loss of Electrical Load Test - TP/2/A/2650/06" (Attachment 4) documents successful completion of the subject testing. This section states in part, "The purpose of this test was to demonstrate the ability of the primary plant, secondary plant, and automatic reactor control system to withstand a total loss of Electric Generator Load - thereby verifying the ability of the unit to handle a load rejection transient. Also, it was performed to acquire data to evaluate the interaction between control systems in response to the load rejection. Another purpose was the verification of the proper response of the Steam Dump Control System. Finally this test was performed (at 50% power only) to demonstrate that the Pressurizer Power Operated Relief Valves (PORV's) will not open during a load rejection from the highest power level at which there is no anticipatory reactor trip upon turbine trip." Note that the anticipatory reactor trip upon turbine trip is unblocked at > 48% power. Consequently, performing the test at 50% power enveloped the highest power level at which there is no anticipatory reactor trip upon turbine trip.

The acceptance criteria was documented as follows:

(1) Safety Injection is not initiated.
(2) Main steam and pressurizer safety valves do not lift.
(3) No operator action is required until reactor power is less than approximately 20% (except to swap feedwater to lower nozzle).
(4) No safety limits are exceeded.
(5) Pressurizer PORV's do not open (50% test only).

Neither the turbine nor reactor tripped during the 50% power test. The pressurizer PORV's, safety valves and the steam generator safety valves did not open. Safety injection did not occur. The steam dump system functioned satisfactorily. All acceptance criteria were met for the 50% power test.

LC.2.C (11)(d), Control Room Design (1.D.1)

Restatement of Request:

DEC stated that LC 2.C(11)(d) had been met on the basis that SSER-6 reported the issue to be complete. The staff reviewed this issue as reported in the DEC letters dated February 18, 1983, and August 15, 1980, and in SSER-4, dated January 1981. The NRC's technical review of the issue was complete as reported in SSER-6. However, SSER-6 also noted that verification of

[8] February 8, 1984, Hal B. Tucker to Mr. James P. O'Reilly, McGuire Nuclear Station Unit 2, Docket No. 50-370, Startup Report.

implementation remained to be performed. DEC is requested to provide further information demonstrating that these issues have been resolved as required by the LC.

Response:

The five McGuire Control Room issues identified by License Condition I.D.1 have been resolved.

The TMI related Control Room human factors issues, with the exception of the Control Room lighting issue, were resolved by Duke's Detailed Control Room Design Review (DCRDR) project mandated by NUREG-0737. The DCRDR project generated over 700 issues referred to as HEDs (human engineered discrepancies). The NRC reviewed the DCRDR Summary Report for McGuire Units 1 and 2 and concluded that it met the requirements of NUREG-0737, Supplement 1 (see NRC letter dated August 6, 1985[9]).

The following table describes each of these issues, the applicable HEDs, and the McGuire Nuclear Station Modification (NSM) that resolved the issue. Each HED applies to both units unless otherwise stated.

LC I.D.1 ISSUE	Applicable HED	NSM number (completion date)
1. Controllers with reversed scales shall be replaced or signal reversing relays installed where applicable.	M-1-0075, M-1-0363	01295 (9/17/84), 20407 (3/5/85), 01585 (7/11/86), 20464 (6/18/87)
2. All applicable meter scales shall be permanently marked.	M-1-0111, M-1-0198, M-1-0177, M-1-0274, M-1-0281, M-1-0285, M-1-0287, M-1-0288, M-1-0282, M-1-0370, M-1-0653	01516 (8/13/86), 20403 (6/9/86)
3. The licensee shall re-scale or replace circular displays.	M-1-0188	Not performed. Meters determined to be acceptable. See Duke letter dated April 28, 1989 and NRC acceptance letter dated January 23, 1990 (attached).
4. Selector switches that can be placed in an intermediate (no selection) position shall be replaced.	M-1-0184, M-1-0671, M-1-0337, M-1-0343, M-1-0344	01535 (6/21/85), 20423 (5/4/85), 01515 (5/30/85), 20406 (4/2/85), 01538 (7/11/86), 20453 (6/9/86)

[9] August 6, 1985, Thomas M. Novak to Mr. H. B. Tucker, Detailed Control Room Design Review Supplemental Safety Evaluation Report #2 for McGuire Nuclear Station, Units 1 and 2.

I.D.1 (5): "Appropriate modifications to the normal and emergency lighting systems shall be made to ensure adequate illumination of the Control Room under all operating conditions." Per SER Supplement 4, the NRC stated that "Emergency DC lighting resulted in glare on some panels, and makes displays difficult to read."

To resolve the Control Room Lighting issue, Duke retained a consultant, Gibbs & Hill, Inc., to perform a detailed survey of the lighting in the McGuire Control Room area. The purpose of the survey was to determine the adequacy of both normal and emergency illumination under credible operating conditions, and, if appropriate, to recommend modifications that would enhance visual performance. The survey took into account NRC requirements and concerns. The report was completed and presented to Duke on February 19, 1981.

Several recommendations were made by the consultant. The recommendations resolving the NRC concerns from LC I.D.1 (5) are as follows:

1. To improve display, CRT, and label readability under emergency lighting conditions, minor wiring and circuiting changes should be made between the normal and emergency lighting. CRT screens should also have hoods added and be tilted forward.

2. To eliminate glare caused by the lighting system, a combination of diffusing panels and different louvers should be installed to re-direct the light in precise patterns.

3. To maintain the improved lighting quality, a periodic program should be implemented to test, re-lamp, and clean the louvers and diffusing panels.

In March 1981, Duke Design Engineering reviewed and accepted the Gibbs & Hill recommendations and preceded with implementation. The recommended modifications were completed by Duke's Construction Department under special project SPM-110 and McGuire Nuclear Station Modification MG-00215. The addition of the diffusing panels eliminated the glare on the CRTs such that the hoods were not required. Duke drawing MC-1846-03 reflects the changes. Follow-up illumination surveys were made by Duke Design Engineering as verification.

A Preventative Maintenance (PM) activity to test, re-lamp, and clean fixtures has been in place since 1981. This PM is on a 18 month frequency, scheduled under the Work Management System (WMS) model work order 85059748, and is implemented by procedure IP/0/B/3190/011. The procedure specifies the exact replacement lamp to maintain the proper illumination level.

LC.2.D, Exemptions to Appendix G for Units 1 and 2

Restatement of Request:

The McGuire FOLs each contain an LC on Appendix G exemptions. Multiple exemptions were discussed in SSER-2 and SSER-4. The licensee stated that the need for some of these exemptions was obviated by later revisions to the requirements of the Commission's regulations in Appendix G. However, DEC's proposed justification is not sufficiently detailed to permit a staff review of this issue. DEC is requested to submit specific information identifying (a) each of the exemptions addressed by SSER-2 and 4, (b) those exemptions DEC proposes are no longer needed, and (c) the specific change in the regulations that obviates the continued need for the exemption.

Response:

NRC required exemptions addressed in SSER-2 and 4

During the initial licensing of McGuire, the NRC staff determined that exemptions to 10 CFR 50 Appendices G and H were required. The staff also determined that the exemptions were justified. A complete discussion of the exemptions is provided in SSER-2, Appendix B, "Safety Evaluation Report, McGuire Nuclear Station, Unit Nos. 1 and 2, Exemptions from Certain Requirements of appendices G and H of 10 CFR Part 50," March 1979. In SSER-2, Appendix B, Section III.A, the NRC indicated that exemptions were necessary to enable the substitution of an alternative method of compliance with the following 10 CFR 50 Appendix G requirements:

Item 1: For Unit No. 1, Section III.C of Appendix G is not complied with to the extent that the unirradiated impact tests did not include all the required tests from separate weldment specimens taken from excess material from each of the six beltline shell plates with the corresponding heat of filler material used in the fabrication of the vessel.

Item 2: For Unit No. 1, Section IV.A.4 of Appendix G is not complied with to the extent that the Charpy V-notch test for the reactor vessel bolting material was not conducted in terms of the lateral expansion of the specimens and the test was not conducted at the lower of the preload temperature or at the lowest service temperature.

Item 3: For Unit No. 2, Section IV.A.4 of Appendix G is not complied with to the extent that the Charpy V-notch test for the reactor vessel bolting was not conducted in terms of lateral expansion of the specimens and the test was not conducted at the lower of the preload temperature or at the lowest service temperature.

In McGuire SSER-2, Appendix B, Section III.B, the NRC indicated that exemptions were necessary to enable the substitution of an alternative method of compliance with the following 10 CFR 50 Appendix H requirements:

Item 4: For both Unit Nos. 1 and 2, Section II.C.2. of Appendix H is not complied with to the extent that the calculated neutron flux lead factor for four of the six surveillance capsules is 3.6 instead of the maximum allowable factor of 3.0.

NRC required exemptions that DEC considers no longer needed

For the reasons stated below, DEC concludes that the NRC required exemptions described in Items 1 through 4 above are no longer needed.

Specific change in the regulations that obviates the continued need for these NRC required exemptions

Item 1: McGuire SSER-4, was published in January 1981. In Section 5.2.3 of this supplement, the NRC staff stated that the exemption granted from Appendix G, paragraph III.C concerning weld and heat-affected zone material fracture toughness testing, for Unit 1 is still required since the affected unit did not fully comply with the then current regulation. However, a general revision to Appendices G and H was in the process of being issued back in 1981. The staff indicated in SSER-4 that the exemption presently being granted in 1981 from this requirement would no longer be necessary when the proposed revision of the regulations became effective. That revision to Appendix G, Section III, which is effective, currently provides requirements for fracture toughness tests. Paragraph III.A specifically allows for reactor vessels constructed to an ASME Code earlier than the Summer 1972 Addenda of the 1971 Edition, the fracture toughness data and data analyses must be supplemented in a manner approved by the Director, Office of Nuclear Reactor Regulation, to demonstrate equivalence with the fracture toughness requirements of Appendix G. As per SSER-4, with regard to this issue, the NRC staff indicated that Unit 1 had a level of quality equivalent to that required by Appendix G. Consequently, the subject exemption is no longer required.

Items 2 and 3: In Section 5.2.3 of McGuire SSER-4, the NRC staff explained that Appendices G and H had been partially revised such that some of the exemptions discussed in SSER-2 were no longer needed. Specifically, the NRC staff stated that the exemptions granted from Appendix G, paragraph IV. A.4 (Items 2 and 3 above), concerning fracture toughness requirements for bolting material, for Units 1 and 2 are no longer necessary since the affected units now comply with the present regulations.

Item 4 In Section 5.2.3 of McGuire SSER-4, the NRC staff explained that Appendices G and H had been partially revised such that some of the exemptions discussed in SSER-2 were no longer needed. Specifically, the NRC staff stated that the exemptions granted from Appendix H, paragraph II.C.2 (Item 4 above), concerning neutron flux lead factor, for Units 1 and 2, are no longer necessary since the affected units now comply with the present regulations.

Attachment 2

2.E. Duke Energy Corporation shall fully implement and maintain in effect all provisions of the Commission-approved ~~physical~~ Nuclear security and Contingency, guard training and qualification plans ~~and safeguards contingency plans~~ including amendments made pursuant to provisions of the Miscellaneous Amendments and Search Requirements revisions to 10 CFR 73.55 (51 FR 27817 and 27822) and to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The plan~~s~~, which contains safeguards information protected under 10 CFR 73.21, is ~~are~~ entitled: ~~"McGuire Nuclear Station Physical Security Plan," with revisions submitted through September 25, 1987; "McGuire Nuclear Station Training and Qualification Plan," with revisions submitted through July 3, 1986; and "McGuire Nuclear Station Safeguards Contingency Plan," with revisions submitted through March 21, 1986.~~ "Nuclear Security and Contingency Plan," Revision 12, as revised per 10 CFR 50.54(p). The plan which does not contain Safeguards information is entitled "Nuclear Security Training and Qualification Plan," Revision 6, as revised per 10 CFR 50.54(p). Changes made in accordance with 10 CFR 73.55 shall be implemented in accordance with the schedule set forth therein.

2.F. (Deleted)

D. The facility requires an exemption from certain requirements of Appendix G to 10 CFR Part 50. This exemption is described in the Office of Nuclear Reactor Regulation's Safety Evaluation Report, Supplement No. 2 and in Supplement No. 4 (Section 5.2.3). This exemption is authorized by law and will not endanger life or property or the common defense and security and is otherwise in the public interest. The exemption is, therefore, hereby granted pursuant to 10 CFR 50.12. With the granting of this exemption, the facility will operate, to the extent authorized therein, in conformity with the application, as amended, the provisions of the Act, and the rules and regulations of the Commission;

E. Duke Energy Corporation shall fully implement and maintain in effect all provisions of the Commission-approved ~~physical~~ Nuclear security and Contingency, guard training and qualification plans ~~and safeguards contingency plans~~ including amendments made pursuant to provisions of the Miscellaneous Amendments and Search Requirements revisions to 10 CFR 73.55 (51 FR 27817 and 27822) and to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The plans, which contain safeguards information protected under 10 CFR 73.21, is ~~are~~ entitled: ~~"McGuire Nuclear Station Physical Security Plan," with revisions submitted through September 25, 1987; "McGuire Nuclear Station Training and Qualification Plan," with revisions submitted through July 3, 1986; and McGuire Nuclear Station Safeguards Contingency Plan," with revisions submitted through March 21, 1986~~. Changes made in accordance with 10 CFR 73.55 shall be implemented in accordance with the schedule set forth therein.

F. The licensee shall report any violations of the requirements contained in Section 2 Items C.(1), C.(4) through C.(11), and E of this license within 24 hours by telephone and confirm by telegram, mailgram, or facsimile transmission to the NRC Regional Administrator, Reg. II, or his designate, no later than the first working day following the violation, with a written followup report within 14 days;

"Nuclear Security and Contingency Plan," Revision 12, as revised per 10 CFR 50.54. The plan, which does not contain Safeguards Information is entitled "Nuclear Security Training and Qualification Plan," Revision 6, as revised per 10 CFR 50.54(p).

(13) Additional Conditions

The Additional Conditions contained in Appendix C, as revised through Amendment No. , are hereby incorporated into this license. Duke Energy Corporation shall operate the facility in accordance with the Additional Conditions.

D. Deleted

E. Duke Energy Corporation shall fully implement and maintain in effect all provisions of the Commission-approved nuclear security and contingency, guard training and qualification plans including amendments made pursuant to provisions of the Miscellaneous Amendments and Search Requirements revisions to 10 CFR 73.55 (51 FR 27817 and 27822) and to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The plan, which contains safeguards information protected under 10 CFR 73.21, is entitled, "Nuclear Security and Contingency Plan," Revision 12, as revised per 10 CFR 50.54(p). The plan which does not contain safeguard information is entitled, "Nuclear Security Training and Qualification Plan," Revision 6, as revised per 10 CFR 50.54(p). Changes made in accordance with 10 CFR 73.55 shall be implemented in accordance with the schedule set forth therein.

F. Deleted

G. Deleted

H. Deleted

I. The licensee shall have and maintain financial protection of such type and in such amounts as the Commission shall require in accordance with Section 170 of the Atomic Energy Act of 1954, as amended, to cover public liability claims.

J. Pursuant to an Order by the Atomic Safety and Licensing Board, dated April 23, 1975, the Nuclear Regulatory Commission incorporates herein the following antitrust conditions:

a. The licensee makes the commitments contained herein, recognizing that bulk power supply arrangements between neighboring entities normally tend to serve the public interest. In addition, where there are net benefits to all participants such arrangements also serve the best interests of each of the participants. Among the benefits of such transactions are increased electric system reliability, a reduction in the cost of electric power, and minimization of the environmental effects of the production and sale of electricity.

Any particular bulk power supply transaction may afford greater benefits to one participant than to another. The benefits realized by a small system may be proportionately greater than those realized by a larger system. The relative benefits to be derived by the parties from a proposed transaction, however, should not be controlling upon a decision with respect to the desirability of participating in the transaction. Accordingly, the licensee will enter into proposed bulk power transactions of the types hereinafter described which, on balance, provide net benefits to the licensee. There are net benefits in a transaction if the licensee

(12) Deleted

(13) Additional Conditions

The Additional Conditions contained in Appendix D, as revised through Amendment No. , are hereby incorporated into this license. Duke Energy Corporation shall operate the facility in accordance with the Additional Conditions.

D. Deleted

E. Duke Energy Corporation shall fully implement and maintain in effect all provisions of the Commission-approved nuclear security and contingency, guard training and qualification plans including amendments made pursuant to provisions of the Miscellaneous Amendments and Search Requirements revisions to 10 CFR 73.55 (51 FR 27817 and 27822) and to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The plan, which contains safeguards information protected under 10 CFR 73.21, is entitled: "Nuclear Security and Contingency Plan," Revision 12, as revised per 10 CFR 50.54 (p). The plan which does not contain safeguards information is entitled, "Nuclear Security Training and Qualification Plan," Revision 6, as revised per 10 CFR 50.54 (p). Changes made in accordance with 10 CFR 73.55 shall be implemented in accordance with the schedule set forth therein.

F. Deleted

G. Deleted

H. The licensee shall have and maintain financial protection of such type and in such amounts as the Commission shall require in accordance with Section 170 of the Atomic Energy Act of 1954, as amended, to cover public liability claims;

I. In accordance with the Commission's direction in its Statement of Policy, Licensing and Regulatory Policy and Procedures for Environmental Protection; Uranium Fuel Cycle Impacts, October 29, 1982, this license is subject to the final resolution of the pending litigation involving Table S-3. See, Natural Resources Defense Council v. NRC, No. 74-1586 (D.C. cir. April 27, 1982); and

J. The licensee is authorized to receive from the Oconee Nuclear Station, Units 1, 2 and 3, possess, and store irradiated Oconee fuel assemblies containing special nuclear material, enriched to not more than 3.24% by weight U-235 subject to the following conditions:

   a. Oconee fuel assemblies may not be placed in the McGuire Nuclear Station, Units 1 and 2, reactors.

   b. Irradiated fuel shipped to McGuire Nuclear Station, Units 1 and 2, from Oconee shall have been removed from the Oconee reactor no less than 270 days prior to shipment.

   c. No more than 300 Oconee irradiated fuel assemblies shall be received for storage at McGuire Nuclear Station.

Attachment 3

Enclosure 13.4

Glow Plug Temperature Measurement

IGNITER (EHM)	LOCATION	GLOW PLUG TEMP (°F)
TB 3	736-6°-46-PIPE CHASE	
TB 4	736-6°-46-PIPE CHASE	
TB 5	736-92°-48-PC DUCK WALL	
TB 6	736-92°-48-PC DUCK WALL	
TB 7	736-182°-47-PIPE CHASE	
TB 8	736-182°-47-PIPE CHASE	
TB 9	736-273°-45-PIPE CHASE	
TB 10	736-273°-45-PIPE CHASE	
TB 11	764-2°-51-VENT ROOMS	
TB 12	764-2°-51-VENT ROOMS	
TB 13	764-50°-45-ACCUM ROOM-A	
TB 14	764-50°-45-ACCUM ROOM-A	
TB 15	764-91°-53-SEAL TABLE	
TB 16	764-91°-53-SEAL TABLE	
TB 17	764-144°-53-ACCUM ROOM-B	
TB 18	764-144°-53-ACCUM ROOM-B	
TB 19	764-178°-54-VENT ROOM	
TB 20	764-176°-54-VENT ROOM	
TB 21	764-216°-54-ACCUM ROOM-C	
TB 22	764-216°-54-ACCUM ROOM-C	
TB 23	764-245°-46-ACCUM ROOM-C	
TB 24	764-245°-46-ACCUM ROOM-C	
TB 25	764-321°-50-ACCUM ROOM-D	
TB 26	764-321°-50-ACCUM ROOM-D	
TB 27	775-55°-23-TOP OF RCP-A	
TB 28	775-55°-23-TOP OF RCP-A	
TB 29	778-121°-2-UNDER M SHIELD	

Unit 1

IGNITER (EHM)	LOCATION	GLOW PLUG TEMP (°F)
TB 30	778-121°-2-UNDER M SHIELD	
TB 31	775-21°-28-TOP OF RCP-B	
TB 32	775-121°-28-TOP OF RCP-B	
TB 33	775-215°-24-RCP-C & S/G-C	
TB 34	775-215°-24-RCP-C & S/G-C	
TB 35	775-326°-21-RCP-D & S/G-D	
TB 36	775-326°-21-RCP-D & S/G-D	
TB 37	820-18°-31-TOP OF S/G A	
TB 38	820-18°-3-TOP OF S/G A	
TB 39	815-114°-34-PRESSURIZER	
TB 40	815-114°-34-PRESSURIZER	
TB 41	820-161°-35-TOP OF S/G-B	
TB 42	820-161°-35-TOP OF S/G-B	
TB 43	820-205°-30-TOP OF S/G-C	
TB 44	820-205°-30-TOP OF S/G-C	
TB 45	820-336°-32-TOP OF S/G-D	
TB 46	820-336°-32-TOP OF S/G-D	
TB 47	840-59°-46-ICE COND	
TB 48	840-34°-56-ICE COND	
TB 49	840-157°-46-ICE COND	
TB 50	840-133°-56-ICE COND	
TB 51	840-206°-46-ICE COND	
TB 52	840-232°-56-ICE COND	
TB 53	840-321°-46-ICE COND	
TB 54	840-344°-56-ICE COND	
TB 55	DOME-53°-21	
TB 56	DOME-53°-21	

Unit 1

IGNITER (EHM)	LOCATION	GLOW PLUG TEMP (°F)
TB 57	DOME-136°-23	
TB 58	DOME-136°-23	
TB 59	DOME-222°-24	
TB 60	DOME-222°-24	
TB 61	DOME-314°-22	
TB 62	DOME-314°-22	
TB 63	840-108°-46-ICE COND	
TB 64	840-183°-56-ICE COND	
TB 65	840-11°-46-ICE COND	
TB 66	840-84°-56-ICE COND	
TB 67	725-91°-17-BY PR TANK	
TB 68	725-89°-17-BY PR TANK	
TB 69	824-40°-33-TOP OF S/G-C	
TB 70	824-139°-33-TOP OF S/G-B	
TB 71	824-218°-33-TOP OF S/G-A	
TB 72	824-322°-33-TOP OF S/G-D	

NOTE: Replacement of glow plugs or transformers is necessary for components that do **NOT** meet acceptance criteria. (Replacement glow plugs must be preconditioned.)

Attachment 4

7.5 Unit Loss of Electrical Load Test - TP/2/A/2650/06

The purpose of this test was to demonstrate the ability of the primary plant, secondary plant, and automatic reactor control systems to withstand a total loss of Electric Generator Load - thereby verifying the ability of the unit to handle a load rejection transient. Also, it was performed to acquire data to evaluate the interaction between control systems in response to the load rejection. Another purpose was the verification of the proper response of the Steam Dump Control System. Finally, this test was performed (at 50% power only) to demonstrate that the Pressurizer Power Operated Relief Valves (PORV's) will not open during a load rejection from the highest power level at which there is no anticipatory reactor trip upon turbine trip.

Prior to load rejection, the following control systems were verified to be functioning properly in automatic mode: Reactor Rod Control System, Pressurizer Level Control System, Pressurizer Pressure Control System, Steam Generator Level Control System, Steam Dump Control System, and Main Feedwater Pump Speed Control System. All pressurizer and main steam relief and safety valves were verified to be operable. For a five minute period of time prior to the load rejection, the following unit parameters were determined to be stable: Reactor Power, Reactor Coolant System Average Temperature, Reactor Coolant System Pressure, Steam Generator Levels, and Feedwater Flows. Reactor Power was approximately 50% full power, the turbine-generator was at 513 MWe, and Control Bank D was at 215 steps withdrawn.

The acceptance criteria for this test is as follows:

(1) Safety Injection is not initiated.
(2) Main steam and pressurizer safety valves do not lift.
(3) No operator action is required until reactor power is less than approximately 20 percent (except to swap feedwater to lower nozzle).
(4) No safety limits are exceeded.
(5) Pressurizer PORV's do not open (50% test only).

The data gathering devices used in this test were the Unit Operator Aid Computer (OAC) Transient Monitor Program and the plant valve timing program. The test was initiated on August 26, 1983 at 2043:59 hours by manually placing both main generator breakers in their trip position. The load on the turbine dropped to zero megawatts, and the turbine was kept at about 1800 rpm in Speed Control. The reactor ranback in automatic control to approximately 6% full power. Control Rods were at their maximum speed for 136 seconds. During that time Control Bank D stepped in from 215 to 97 steps withdrawn. Seven out of the nine Main Steam Bypass Valves to Condenser were completely open 11 seconds into the transient. All nine steam dump valves were open 33 seconds into the transient. The first Main Steam Bypass Valve to Condenser closed 84 seconds into the transient.

During the reactor runback, no operator action was made above 15% excore power (except to swap from main to auxiliary feedwater nozzles). During the initial phase of the transient, the reactor coolant average temperature increased by 3.0°F (from 573°F to 576°F). The Pressurizer Pressure Control System allowed a reactor coolant pressure increase of 67 psig (from 2230 psig to 2297 psig).

Neither the turbine nor reactor tripped. The pressurizer PORV's, safety valves and the steam generator safety valves did not open. Safety injection did not occur. The steam dump system functioned satisfactorily. The test was very smooth and no problems were encountered. All acceptance criteria were met for the 50% power test. Figures 7.5-1 to 7.5-8 show the response of various plant parameters during this transient.

This test is scheduled to be run again at the 100% full power level.