June 8, 2001

Mr. Mark Reddemann
Site Vice President
Kewaunee and Point Beach Nuclear Plants
Nuclear Management Company, LLC
6610 Nuclear Road
Two Rivers, WI 54241

SUBJECT: KEWAUNEE NUCLEAR POWER PLANT - ISSUANCE OF AMENDMENT (TAC NO. MA8017)

Dear Mr. Reddemann:

The U.S. Nuclear Regulatory Commission has issued the enclosed Amendment No. 155 to Facility Operating License No. DPR-43 for the Kewaunee Nuclear Power Plant (KNPP). This amendment revises the Technical Specifications in response to an application dated January 13, 2000, submitted by Wisconsin Public Service Corporation (WPSC), as supplemented March 7, March 30, and May 4, 2001. Subsequent to the submittal, WPSC was succeeded by Nuclear Management Company, LLC (NMC) as the licensed operator of the KNPP. By letter dated October 5, 2000, NMC (the licensee) requested the Nuclear Regulatory Commission (NRC) staff to continue to process and disposition licensing actions previously docketed and requested by WPSC.

The amendment revises the KNPP Technical Specifications (TSs) 3.6, “Containment” to add Limiting Condition for Operation (LCO) and Allowed Outage Times (AOT) for containment isolation devices. In addition, the amendment provides additional information, clarification, and uniformity to the bases of the associated TSs.

A copy of the Safety Evaluation is also enclosed. Notice of Issuance will be included in the Commission's next regular biweekly *Federal Register* notice.

Sincerely,

/RA/
John G. Lamb, Project Manager, Section 1
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-305

Enclosures: 1. Amendment No. 155 to License No. DPR-43
2. Safety Evaluation

cc w/encls: See next page

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Docket No. 50-305
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cc w/encls: See next page
<u>DISTRIBUTION:</u>

PUBLIC	WBeckner	GHill (2)	JLamb
PD III-1 Reading	ACRS	OGC	GGrant, RIII
CCraig	THarris	RGiardina	RGoel
MWohl	*SE input	+see previous concurrence	

OFFICE	PDIII-1/PM	PDIII-1/LA	SPLB/SC*	RTSB/SC*	OGC+	PDIII-1/SC
NAME	JLamb	THarris	GHubbard	RDennig	RHoefling	CCraig
DATE	6/8/01	6/8/01	05/03/01	04/20/01	06/07/01	6/8/01

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Kewaunee Nuclear Power Plant

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NUCLEAR MANAGEMENT COMPANY, LLC

DOCKET NO. 50-305

KEWAUNEE NUCLEAR POWER PLANT

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 155
License No. DPR-43

1. The U.S. Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Nuclear Management Company, LLC (the licensee) dated January 13, 2000, as supplemented March 7, March 30, and May 4, 2001, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. DPR-43 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 155, are hereby incorporated in the license. The licensees shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance, and is to be implemented within 30 days of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/
Claudia M. Craig, Chief, Section 1
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: June 8, 2001

ATTACHMENT TO LICENSE AMENDMENT NO. 155

FACILITY OPERATING LICENSE NO. DPR-43

DOCKET NO. 50-305

Replace the following pages of the Appendix A Technical Specifications with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

REMOVE	INSERT
Table of Contents TS iii	Table of Contents TS iii
Table of Contents TS iv	Table of Contents TS iv
TS 1.0-2	TS 1.0-2
TS 3.6-1 through 3.6-4	TS 3.6-1 through 3.6-4
TS B 3.6-1 through B 3.6-5	TS B 3.6-1 through B 3.6-5
TS 4.4-3 through 4.4-4	TS 4.4-3 through 4.4-4
TS B 4.4-1	TS B 4.4-1
TS B 4.4-3 through 4.4-4	TS B 4.4-3 through 4.4-4

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATING TO AMENDMENT NO. 155 TO FACILITY OPERATING LICENSE NO. DPR-43

NUCLEAR MANAGEMENT COMPANY, LLC

KEWAUNEE NUCLEAR POWER PLANT

DOCKET NO. 50-305

1.0 INTRODUCTION

By letter dated January 13, 2000, as supplemented March 7, March 30, and May 4, 2001, Wisconsin Public Service Corporation (WPSC) submitted a proposed amendment to the Kewaunee Nuclear Power Plant (KNPP) technical specifications (TSs) to revise TS 1.0.g "CONTAINMENT SYSTEM INTEGRITY" and add TS 3.6.b "Containment Isolation Valves" and TS 4.4.f "Containment Isolation Device Position Verification." This revision would establish the remedial measures to be taken for inoperable containment isolation valve(s) (CIV) and appropriate surveillances to assure CIV OPERABILITY. The applicable Kewaunee TS Bases have been revised to document the TS changes and to provide supporting information. These changes, except for the 24-hour allowed outage time for one CIV inoperable in penetration flow paths with two CIVs per penetration, are based on NUREG-1431, "Standard Technical Specifications for Westinghouse Plants" Revision 1, dated April 1995, as modified by Technical Specification Task Force (TSTF)-30 and -45; "Final Policy Statement on Technical Specification Improvements for Nuclear Power Reactors," (Final Policy Statement), published on July 22, 1993 (58 FR 39132); and 10 CFR 50.36, "Technical Specifications," as amended July 19, 1995 (60 FR 36953). In addition, the amendment provides additional information, clarification, and uniformity to the bases of the associated TSs.

Subsequent to the submittal, WPSC was succeeded by Nuclear Management Company, LLC (NMC) as the licensed operator of the KNPP. By letter dated October 5, 2000, NMC (the licensee) requested the Nuclear Regulatory Commission (NRC) staff to continue to process and disposition licensing actions previously docketed and requested by WPSC. By letter dated February 1, 2001, the NRC staff sent the licensee a request for additional information (RAI). By letters dated March 7, March 30, and May 4, 2001, the licensee responded to the RAI.

The March 7, March 30, and May 4, 2001, letters, provided clarifying information that was within the scope of the original application, did not change the NRC staff's initial proposed no significant hazards consideration determination, and did not expand the amendment beyond the scope of the original notice (66 FR 11061).

2.0 BACKGROUND

The KNPP containment system consists of two separate structures: a reactor containment vessel and a shield building.

The reactor containment vessel is a cylindrical steel pressure vessel with a hemispherical dome and an ellipsoidal bottom which houses the reactor pressure vessel, the steam generators, reactor coolant pumps, the reactor coolant loops, the accumulators, the pressurizer, the pressurizer relief tank, and other branch connections of the reactor coolant system.

The reactor containment vessel is completely enclosed by the shield building. The shield building has the shape of a right circular cylinder with a shallow dome roof. A five foot annular space is provided between the reactor containment vessel and the shield building. Clearance at the roof of the shield building is seven feet.

The containment system is designed to provide protection for the public from the consequences of a design-basis accident (DBA). The reactor containment vessel, including penetrations, is designed for low leakage. At the completion of erection, the reactor containment vessel was tested with the penetrations capped. The measured leakage rate was 0.02 percent of the reactor containment vessel's net free volume at a nominal pressure of 46 pounds per square inch gauge. The reactor containment vessel was re-tested with all penetrations installed to assure that the leakage requirements as set forth in the analysis of Section 14 of the updated final safety analysis report, have been met. The principal function of the containment isolation system is to confine the fission products within the reactor containment vessel. Design limits for radiation doses resulting from accidental releases of radioactivity from a power reactor are specified in 10 CFR 100.

Section 182a of the Atomic Energy Act requires that applicants for nuclear power plant operating licenses will state:

> [S]uch technical specifications, including information of the amount, kind, and source of special nuclear material required, the place of the use, the specific characteristics of the facility, and such other information as the Commission may, by rule or regulation, deem necessary in order to enable it to find that the utilization...of special nuclear material will be in accord with the common defense and security and will provide adequate protection to the health and safety of the public. Such technical specifications shall be a part of any license issued.

In 10 CFR 50.36, the Commission established its regulatory requirements related to the content of TS. In doing so, the Commission placed emphasis on those matters related to the prevention of accidents and the mitigation of accident consequences; the Commission noted that applicants were expected to incorporate into their TS "those items that are directly related to maintaining the integrity of the physical barriers designed to contain radioactivity," as set forth in Statement of Consideration, "Technical Specifications for Facility Licenses; Safety Analysis Reports," (33 FR 18610, December 17, 1968). Pursuant to 10 CFR 50.36, TS are required to include items in the following five specific categories related to station operation: (1) safety limits, limiting safety system settings, and limiting control setting; (2) limiting conditions for operation (LCOs); (3) surveillance requirements (SRs); (4) design features; and (5) administrative controls. However, the rule does not specify the particular requirements to be included in a plant's TS.

For several years, NRC and industry representatives have sought to develop guidelines for improving the content and quality of nuclear power plant TS. On February 6, 1987, the Commission issued an interim policy statement on TS improvements, "Interim Policy Statement on Technical Specification Improvements for Nuclear Power Reactors" (52 FR 3788). During the period from 1989 to 1992, the utility groups and the NRC staff developed improved Standard Technical Specifications (STS), such as NUREG-1431, that would establish models of the Commission's policy for each primary reactor type. In addition, the NRC staff, licensees, and owners groups developed generic administrative and editorial guidelines in the form of a "Writer's Guide" for preparing TS, which gives greater consideration to human factors principles and was used throughout the development of licensee-specific TS.

In September 1992, the Commission issued NUREG-1431, Revision 0, which as developed using the guidance and criteria contained in the Commission's Interim Policy Statement. The STS in NUREG-1431 were established as a model for developing the improved technical specifications (ITS) for Westinghouse plants in general. The STS reflect the results of a detailed review of the application of the interim policy statement criteria to generic system functions, which were published in a "Split Report" issued to the nuclear steam system supplier owners groups in May 1988. STS also reflect the results of extensive discussions concerning various drafts of STS, so that the configurations and operating characteristics for all reactor designs. As such, the generic Bases presented in NUREG-1431 provides an abundance of information regarding the extent to which the STS present requirements that are necessary to protect public health and safety.

On July 22, 1993, the Commission issued its Final Policy Statement, expressing the view that satisfying the guidance in the policy statement also satisfies Section 182a of the Act and 10 CFR 50.36 (58 FR 39132). The Final Policy Statement described the safety benefits of the STS, and encouraged licensees to use the STS as the basis for plant-specific TS amendments, and for complete conversions to ITS based on the STS. Further, the Final Policy Statement gave guidance for evaluating the required scope of the TS and defined four guidance criteria to be used in determining which of the LCOs and associated SRs should remain in the TS. The Commission noted that, in allowing certain items to be relocated to licensee-controlled documents while requiring that other items be retained in the TS, it was adopting the qualitative standard enunciated by the Atomic Safety and Licensing Appeal Board in *Portland General Electric Co.* (Trojan Nuclear Plant), ALAB-531, 9 NRC 263 (1979). There the Appeal Board observed:

> [T]here is neither a statutory not a regulatory requirement that every operational detail set forth in an applicant's safety analysis report (or equivalent) be subject to a technical specification, to be included in the license as an absolute condition of operation which is legally binding upon the licensee unless and until changed with specific Commission approval. Rather, as best we can discern it, the contemplation of both the Act and the regulations is that technical specifications are to be reserved for those matters as to which the imposition of rigid conditions or limitations upon reactor operation is deemed necessary to obviate the possibility of an abnormal situation or even giving rise to an immediate threat to the public health and safety.

By this approach, existing LCO requirements that fall within or satisfy any of the criteria in the Final Policy Statement should be retained in the TS; those LCO requirements that do not fall within or satisfy these criteria may be relocated to licensee-controlled documents. The Commission codified the four criteria set out in the Federal Policy statement in 10 CFR 50.36 (60 FR 36953, July 19, 1995). The four criteria are as follows:

Criterion 1

Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.

Criterion 2

A process variable, design feature, or operating restriction that is an initial condition of a DBA or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barriers.

Criterion 3

A structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a DBA or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

Criterion 4

A structure, system, or component which operating experience or probabilistic safety assessment has shown to be significant to public health and safety.

Part 3.0 of this safety evaluation (SE) provides the basis for NRC staff's conclusion that the addition of the CIV LCO, remedial measures for inoperable CIVs, and CIV surveillances to the Kewaunee TS are based on the STS, as modified by plant-specific changes, and is consistent with the Kewaunee current licensing basis and the requirements and guidance of the Final Policy Statement and 10 CFR 50.36.

3.0 EVALUATION

3.1 CIV Specifications

By letter dated January 13, 2000, as supplemented by letters dated March 7, March 30, and May 4, 2001, the licensee submitted a request to change the Kewaunee TS to establish remedial measures for inoperable CIVs and surveillances to assure CIV OPERABILITY. The current Kewaunee TS do not contain specific actions to be taken for inoperable CIVs unless CIV inoperability results in loss of CONTAINMENT SYSTEM INTEGRITY as defined in TS 1.0.g "CONTAINMENT SYSTEM INTEGRITY" and TS 3.6.a. If CONTAINMENT SYSTEM INTEGRITY is lost, then the Kewaunee TS require a shutdown in accordance with TS 3.0.c.

The licensee proposes to revise the Kewaunee TS to add TS 3.6.b which specifies that all CIVs and blind flanges shall be OPERABLE when CONTAINMENT SYSTEM INTEGRITY is required and specifies the remedial measures to be taken when CIVs are inoperable. In addition, TS 4.4.f is also added which specifies various CIV surveillances to assure CIV OPERABILITY to maintain CONTAINMENT SYSTEM INTEGRITY. As a result of these two changes, TS 1.0.g is revised to reflect these additional requirements of CIV OPERABILITY and the other specifications in TS 3.6 and 4.4 have been administratively relabeled to support this change. Corresponding Bases are also added to support this change.

Containment isolation valves form a part of the containment boundary. The containment isolation valves' safety function is related to minimizing the loss of reactor coolant inventory and establishing the containment boundary during a DBA. Thus, the CIVs satisfy Criterion 3 of 10 CFR 50.36.

Except for the 24-hour allowed outage time for one CIV inoperable in penetration flow paths with two CIVs per penetration discussed in Section 3.2 below, the proposed modifications are bases on the LCOs, Actions, and Surveillances contained in NUREG-1431, "Standard Technical Specifications for Westinghouse Plants," Revision 1, as modified by TSTF-30 and 45. The proposed LCO (TS 3.6.b.1) along with the surveillances specified in TS 4.4.f.1 through 4.4.f.3 provide assurance that the containment isolation valves and purge valves will perform their designed safety functions to minimize the loss of reactor coolant inventory and establish the containment boundary during accidents. In the event of CIV inoperability, the proposed Actions isolate the affected penetration flow paths. The method of isolation include the use of at least one isolation barrier that cannot be adversely affected by a single active failure. These Actions and the associated completion times, except for the change from the NUREG-1431 completion time of 4 hours to 24 hours for an inoperable CIV in a two valve penetration, are reasonable considering the time required to isolate the penetration, the means used to isolate the penetration and the relative importance of supporting containment OPERABILITY and the leak tightness during plant operations. Furthermore, TS 3.6.b and 4.4.f impose additional restrictions on plant operations that are not contained in the current Kewaunee TS, which make this change a more restrictive change. Thus, the staff finds these changes to be acceptable.

3.2 <u>24-Hour Completion Time for Inoperable CIV</u>

The licensee proposed modifications to TS Section 1.0.g, "Containment System Integrity," and TS Section 3.6.b, "Containment Isolation Valves". TS 1.0.g is being modified so that TS 1.0.g.1 and TS 1.0.g.4 allow exceptions delineated in TS 3.6.b for LCOs and AOTs associated with the containment isolation valves and other such devices used for containment isolation.

TS 1.0.g required all manual valves and blind flanges to be closed as required. This amendment adds TS Section 3.6.b, which will state that these components shall be operable and delineate their associated LCOs. The allowed outage time (AOT) requested is 24 hours for one inoperable valve verses the current TS, which does not have an AOT.

TS 1.0.g required all automatic containment system isolation valves to be operable, or deactivated in the closed position, or at least one closed valve in each line having an inoperable valve. The proposed amendment adds TS Section 3.6.b, which will specify an AOT of 24 hours. The licensee indicated that other proposed changes to TS Section 3.6 and bases are as per STS.

TS 3.6 is being modified to add explicit guidance for closed isolation valve (CIV) LCOs, AOTs, and required compensatory measures. For open systems, a 24-hour AOT for a single inoperable CIV is being proposed. A 72-hour AOT has been approved for penetration isolation devices in closed systems to incorporate the approval of TSTF Traveler, TSTF-30.

The principal function of the containment isolation system is to confine the fission products within the primary containment system boundary. Design limits for radiation doses resulting from accidental releases of radioactivity from a reactor plant are specified in 10 CFR 100. The off-site dose consequences for the loss-of-coolant accident (LOCA) are contained in the updated safety analysis report (USAR), Section 14.3.5, "Off-site Dose Consequences."

The licensee indicated that the proposed amendment maintains the current defense-in-depth of the containment isolation system. Current TSs prevent plant heatup greater than 200 °F (Cold Shutdown) until containment integrity is established. If containment integrity is lost while above 200 °F the operators are directed to restore containment integrity within one hour. To ensure at least one valve is closed when needed, two isolation barriers are provided per containment penetration thus, ensuring single-failure criteria is met. If there were a failure of an operable containment isolation valve to close, current plant emergency operating procedures directs the operators to manually close the valve. This proposed amendment does not change the current defense-in-depth concept. Thus, if an inoperable automatic isolation valve were open, the redundant operable automatic isolation valve would close or the operators would close it.

The licensee stated that the purpose of LCOs and AOTs is to permit temporary outages of redundant components and are specified for specific time intervals that are consistent with maintenance as stated in KNPP TS Bases Section 3.3. Inoperability of a single component does not negate the ability of the system to perform its function, but it reduces the redundancy provided in the plant design and thereby limits the ability to tolerate additional equipment failures. However, the equipment AOTs specified in the LCOs are a temporary relaxation of the single-failure criterion, which, consistent with overall system reliability considerations, provides a limited time to restore equipment to the operable condition.

Experience at the Kewaunee Plant shows that 4 hours, the STS AOT, is insufficient to perform minor maintenance on containment isolation devices. The licensee states that, depending on the failure mechanism, repairs may take 10-12 hours and post-maintenance testing may take another 6-10 hours. Therefore, the requested 24-hour AOT is consistent with previous maintenance time requirements.

3.2.1 Probabilistic Risk Assessment (PRA) Insights

The licensee performed a PRA which included, but was not limited to, a large break loss-of-coolant accident (LBLOCA), a medium break loss-of-coolant accident (MBLOCA), and a small-break loss-of-coolant accident (SBLOCA). This analysis included failures associated with the containment isolation (CI) signal, mechanical failures, operator errors, and common cause failures.

Additionally, the change in large early release frequency (ΔLERF) and incremental conditional large early release probability (ICLERP) was calculated using an AOT of 24 hours. This calculation is independent of the above LOCA calculation and includes all

16 initiating events in the Kewaunee internal events PRA. As the CI function is not a contributor to the Core Damage Frequency (CDF), a change in CDF and incremental core damage probability was not calculated by the licensee. The large early release frequency (LERF) was recalculated assuming one train of CI was out of service. The baseline LERF was subtracted from this conditional LERF and the difference multiplied by 24-hours to obtain the ICLERP. The ΔLERF and ICLERP calculated are 5.4 E-08 per year and 1.5 E-10 respectively. The ΔLERF is below 1.0 E-07, so it is characterized by RG 1.174 as very small. The ICLERP is below the 5.0 E-08 guideline value of RG 1.177 as having a small impact on plant risk.

The proposed AOT change was also reviewed for its impact on fire, flooding, and external events. As with internal events, there would be no change in CDF. The total ΔLERF and ICLERP for internal fire, internal flooding, and seismic initiators were calculated by the licensee to be 3.9 E-07 per year and 1.1 E-09 respectively. Summed together, the total internal and external ΔLERF and ICLERP are 4.4 E-07 per year and 1.2 E-09 respectively. The ΔLERF is in the 1.0 E-07 per year to 1.0 E-06 per year range (Figure 4 of RG 1.174). The licensee's combined internal and external events LERF is 7.6 E-06 per year, below the RG 1.174 guideline for a change of this magnitude. The total ICLERP is well below the RG 1.177 value. Other external event initiators were examined by the licensee (high winds, external fires, external flooding, transportation accidents, and hazardous materials) and found to be unaffected by this change.

Containment penetrations of 5-inch diameter or greater, based upon a containment release rate of one containment volume per hour, were included in the ICLERP calculation. This list of included penetrations was further reduced by including a penetration only if 1) The line penetrating containment is a containment sump or reactor cavity sump drain line, or 2a) The line penetrating containment directly communicates with either the containment atmosphere or the reactor coolant system, and 2b) The line penetrating containment is not part of a closed system outside of containment, capable of withstanding severe accident conditions.

This left two types of penetrations: (1) those that are administratively closed during the entire cycle and, (2) those that are isolated by two check valves. For this calculation, the licensee assumed that one of the two check valves was out of service for the two remaining penetrations.

Finally, CIVs connected to the safety injection line check valve leakage path are included in the model. Other lines/valves included in the containment isolation fault tree, but not considered for early releases, since their diameters are less than 5 inches are (1) the reactor coolant pump bleedoff line, (2) CIVs in penetrations used to support reactor coolant system (RCS) inventory control safety functions under accident conditions, and (3) CIVs in penetrations used to support the containment heat removal function using containment sprays.

The primary area of focus in this application pertains to the containment isolation model, which the licensee states is complete in that it includes all penetrations of interest. The cutoff for which penetrations contribute to the ICLERP is based on Westinghouse Owners Group recommendations. The additional cutsets generated by the licensee in this analysis were examined by the licensee and determined to be reasonable.

3.2.2 PRA Quality

The PRA model used in the submittal reflects the as-built, as-operated plant as of April 1998. There have been no changes, according to the licensee, to the plant since then that would affect the analysis in the license's submittal. A peer review of the Kewaunee PRA was conducted. This consisted of an independent internal review, involving plant staff who had not worked on the PRA, and an independent external review involving contractors who had not worked on the PRA.

The internal review was conducted by five people with Senior Reactor Operator Licenses and four past or present Shift Technical Advisors. These reviewers were drawn from operations, reactor engineering, maintenance, and operator training. The external review was led by Sargent and Lundy Engineers. The reviewers were experts in a variety of PRA areas, including human reliability, structural engineering, systems modeling, data analysis, and accident sequences. Reviewers were from Sargent and Lundy, Battelle National Laboratory, Safety Management Incorporated, and Wisconsin Electric Power Company. The combined group determined the more major findings to be:

-The non safety-related valves at the outlet of the component cooling heat exchangers should be credited.

-Several motor operated valves (MOVs) were modeled as spuriously transferring when there was no credible mechanism for them to do so.

-Several MOV power supplies were mis-identified.

-There is no procedural way to get to primary system feed and bleed operation during a steam generator tube rupture.

-Many reactor trips were mis-categorized in the initiating event calculation.

Other major findings were as follows:

-System initiators were modeled separately from the event tree models, potentially missing some dependencies.

-Fault tree truncation levels were inconsistent.

-Some support systems could be lost because of an inappropriate failure probability.

-Human error dependency was not addressed in all cases.

-Common cause was inappropriately applied.

-Human error probabilities appeared to be overly optimistic.

-Some initiating event frequencies appeared to be low.

In an RAI response, the licensee states that all of the points delineated above were properly addressed and corrections made. Other findings were examined for validity after the review and addressed if necessary.

Subsequent to the review, numerous changes were made to the Kewaunee PRA. The major changes were as follows:

-Addressed Staff RAI including:

-Adding a new initiating event: Loss of a 4160 V ac bus.
-Completely revising the human reliability analysis to address several staff concerns.
-Changed containment fan coil unit success criteria to reflect equipment survivability concerns.

-Removed operator action to stop residual heat removal (RHR) pumps running on miniflow.

-Credit taken for refueling water storage tank (RWST) refill.

-Credit taken for air accumulators on certain air-operated valves (AOVs).

-Modeled alternate means of cooling air compressors.

-Reactor cavity changed from dry to wet due to design change.

-Test and maintenance modeled for both trains instead of only one.

-Loss of dc bus modeled for each train instead of most conservative.

-Loss of ac bus modeled for each train instead of most conservative.

-Component cooling modeled so each train has a 0.5 probability of being in standby.

-LOCAs, steam generator tube rupture (SGTRs), and steamline break (SLBs) modeled so each loop has a 0.5 probability of being the broken loop.

-Charging pump relief valve model corrected.

-Service water strainers removed based on analysis.

-A probability that pressurizer power-operated relief valve (PORV) block valves are open was added; previously, 1 of 2 was assumed to be open.

The licensee states that each PRA revision, whether to the models or the documentation, is accomplished through a proceduralized form which is signed by two members of the PRA group, one as the author, and one as the reviewer.

Based upon consistency with Kewaunee's current licensing basis, its AOT requirements for completion of at-power maintenance, and the licensee's PRA-determined ICLERP being substantially below the RG 1.177 guideline value of 5.0 E-08, as well as the ΔLERF being acceptably low (RG 1.174), the staff concludes that the requested 24-hour AOT for containment isolation devices is acceptable.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Wisconsin State official was notified of the proposed issuance of the amendment. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration and there has been no public comment on such finding (66 *FR* 11061). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment for the above items.

6.0 CONCLUSION

The NRC staff has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

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