March 16, 2006

Mrs. Mary G. Korsnick
Vice President R.E. Ginna Nuclear Power Plant
R.E. Ginna Nuclear Power Plant, LLC
1503 Lake Road
Ontario, NY 14519

SUBJECT: R.E. GINNA NUCLEAR POWER PLANT - AMENDMENT RE: MAIN FEEDWATER ISOLATION VALVES (TAC NO. MC8657)

Dear Mrs. Korsnick:

The Commission has issued the enclosed Amendment No. 95 to Renewed Facility Operating License No. DPR-18 for the R.E. Ginna Nuclear Power Plant. This amendment is in response to your application dated April 29, 2005, as supplemented on July 1 and November 21, 2005.

The amendment revises Technical Specification 3.7.3, “Main Feedwater Regulating Valves (MFRVs), Associated Bypass Valves, and Main Feedwater Pump Discharge Valves (MFPDVs),” to allow the use of the main feedwater isolation valves in lieu of the MFPDVs to provide isolation capability to the steam generators in the event of a steam line break.

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

Sincerely,

/RA/

Patrick D. Milano, Senior Project Manager
Plant Licensing Branch I-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-244

Enclosures:
1. Amendment No. 95 to Renewed License No. DPR-18
2. Safety Evaluation

cc w/encls: See next page

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Accession Number: ML060600424

OFFICE	LPLI-1\PM	LPLI-1\LA	SPWB\BC	ACVB\BC	CPTB\BC	OGC	LPLI-1\BC
NAME	PMilano	SLittle	JNakoski	RDennig	SLee	SUttal	RLaufer
DATE	03/14/06	03/15/06	01/26/06	12/22/05	02/14/06	03/13/06	03/16/06

Official Record Copy

DATED: March 16, 2006

AMENDMENT NO.   TO RENEWED FACILITY OPERATING LICENSE NO. DPR-18
R.E. GINNA NUCLEAR POWER PLANT

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P. Milano	RidsNrrPMPMilano
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R. Dennig	RidsNrrDraAcvb
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R.E. GINNA NUCLEAR POWER PLANT, LLC

DOCKET NO. 50-244

R.E. GINNA NUCLEAR POWER PLANT

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 95
Renewed License No. DPR-18

1. The Nuclear Regulatory Commission (the Commission or the NRC) has found that:

   A. The application for amendment filed by the R.E. Ginna Nuclear Power Plant, LLC (the licensee) dated April 29, 2005, as supplemented on July 1 and November 21, 2005, 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 Renewed Facility Operating License No. DPR-18 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 95, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance and shall be implemented prior to startup from the fall 2006 refueling outage.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Richard J. Laufer, Chief
Plant Licensing Branch I-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: March 16, 2006

ATTACHMENT TO LICENSE AMENDMENT NO. 95

RENEWED FACILITY OPERATING LICENSE NO. DPR-18

DOCKET NO. 50-244

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
3.7.3-1	3.7.3-1
3.7.3-2	3.7.3-2

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 95 TO RENEWED FACILITY

OPERATING LICENSE NO. DPR-18

R.E. GINNA NUCLEAR POWER PLANT, INC.

R.E. GINNA NUCLEAR POWER PLANT

DOCKET NO. 50-244

1.0 INTRODUCTION

By letter dated April 29, 2005, as supplemented on July 1 and November 21, 2005 (Agencywide Documents Access and Management System (ADAMS) Accession Nos. ML051260236, ML051920360 and ML053320193, respectively), R.E. Ginna Nuclear Power Plant, LLC (the licensee) submitted a request for changes to the R.E. Ginna Nuclear Power Plant (Ginna) Technical Specifications (TSs). The requested changes would revise TS Section 3.7.3, "Main Feedwater Regulating Valves (MFRVs), Associated Bypass Valves, and Main Feedwater Pump Discharge Valves (MFPDVs)," to allow the use of the main feedwater isolation valves in lieu of the MFPDVs to provide isolation capability to the steam generators (SGs) in the event of a steam line break. The July 1 and November 21, 2005, letters provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on June 7, 2005 (70 FR 33218).

2.0 REGULATORY EVALUATION

The steam release resulting from a rupture of a main steam pipe will result in an increase in steam flow, a reduction of reactor coolant system (RCS) temperature and pressure, and an increase in core reactivity. The core reactivity increase may cause a power level increase and a decrease in shutdown margin. The uncontrolled release of steam into the containment as a result of the break also results in a rapid increase in containment pressure and temperature. The predicted peak values of the containment pressure and temperature must remain below the respective containment design limits. Reactor protection and safety systems are actuated to mitigate the transient.

One of the automatic reactor protection system responses to a steam line break is the termination of all main feedwater flow to the SGs. This action tends to limit the cooldown of the core, and the mass and energy releases from the SGs. The Nuclear Regulatory Commission

(NRC) staff reviewed the effect of the proposed change (i.e., to use main feedwater isolation valves (MFIVs) to automatically terminate the main feedwater flow during a steam line break event).

The NRC's acceptance criteria for the main steam line break (MSLB) analyses are based on (1) General Design Criterion (GDC) 27, "Combined reactivity control systems capability," of Appendix A to Part 50 of Title 10 of the *Code of Federal Regulations* (10 CFR), insofar as it requires that the reactivity control systems be designed to have a combined capability, in conjunction with poison addition by the emergency core cooling system (ECCS), of reliably controlling reactivity changes under postulated accident conditions, with appropriate margin for stuck rods, to assure the capability to cool the core is maintained; (2) GDC 28, "Reactivity limits," insofar as it requires that the reactivity control systems be designed to assure that the effects of postulated reactivity accidents can neither result in damage to the reactor coolant pressure boundary (RCPB) greater than limited local yielding, nor disturb the core, its support structures, or other reactor vessel internals so as to significantly impair the capability to cool the core; (3) GDC 31, "Fracture protection of reactor coolant pressure boundary," insofar as it requires that the RCPB be designed with sufficient margin to assure that, under specified conditions, it will behave in a nonbrittle manner and the probability of a rapidly propagating fracture is minimized; and (4) GDC 35, "Emergency core cooling," insofar as it requires the reactor cooling system and associated auxiliaries be designed to provide abundant emergency core cooling. Specific review criteria are contained in NRC Standard Review Plan (SRP) Section 15.1.5 (Reference 1).

Regarding the containment system, the NRC staff based its review on the criteria in GDC 50, "Containment design basis," and GDC 38, "Containment heat removal." GDC 50 states, in part, that the reactor containment structure, including access openings, penetrations, and containment heat removal system shall be designed so that the containment structure can accommodate, without exceeding its design leakage rate and with sufficient margin, the calculated pressure and temperature resulting from a loss-of-coolant accident. The staff has also traditionally applied this GDC to the main steam line break. GDC 38 states, in part, that a system to rapidly remove heat from the reactor containment shall be provided so that containment pressure and temperature are maintained at acceptably low levels.

The NRC staff also reviews the capability of nuclear power plant valves and pumps to perform their safety functions in accordance with 10 CFR Part 50. As part of this review, the staff evaluates the inservice testing (IST) program established by nuclear power plant licensees for valves and pumps within the scope of Section XI of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code) and the ASME Code for Operation and Maintenance of Nuclear Power Plants (OM Code), as applicable. The NRC staff has reviewed the design, qualification, and testing of the MFIVs to be modified at Ginna in support of the proposed TS amendment to rely on the MFIVs to isolate the SGs in the event of a steam line break. As part of this review, the staff evaluated the licensee's consideration of lessons learned from the motor-operated valve (MOV) program at Ginna and other nuclear power plants. The NRC's acceptance criteria are based on the applicable GDC in 10 CFR Part 50, Appendix A, and the IST requirements in 10 CFR 50.55a(f). Specific review criteria are contained in SRP Sections 3.9.3 and 3.9.6.

As noted in the Ginna Updated Final Safety Analysis Report (UFSAR), the design criteria used at Ginna predate Appendix A to 10 CFR Part 50. The Ginna design criteria comprise the

proposed Atomic Industrial Forum (AIF) versions of the criteria issued for comment by the Atomic Energy Commission on July 10, 1967. These criteria define or describe safety objectives and approaches incorporated in the design of Ginna.

In February 1978, the NRC initiated its Systematic Evaluation Program (SEP) for eleven operating plants, including Ginna, that had received construction permits between 1956 and 1967. The SEP consisted of a limited review of the designs of these older plants. The purpose of the SEP was to reconfirm and document the design safety because the safety criteria had changed since the plants were originally licensed. As part of the SEP, the original codes and standards used in the design of structures, systems, and components at Ginna were compared with later licensing criteria. The results for Ginna were documented in NRC Report NUREG-0821, "Integrated Plant Safety Assessment, Systematic Evaluation Program, R.E. Ginna Nuclear Power Plant, Final Report," December 1982 (ADAMS No. 8309200476). The current UFSAR incorporates the SEP review into the current licensing basis at Ginna. The adequacy of the Ginna design relative to the 1972 version of the GDC is described in Section 3.1.2 of the UFSAR. Therefore, the NRC staff reviewed the application and based its acceptance on the design criteria in the GDC noted above rather than the criteria in the 1967 version of the AIF criteria.

3.0 TECHNICAL EVALUATION

3.1 Background

The current TS 3.7.3 addresses main feedwater regulating valves (MFRVs) and associated bypass valves and main feedwater pump discharge valves (MFPDVs). These valves are relied upon for isolation of main feedwater flow for line breaks downstream of these valves such as the main steam line break (MSLB) or main feedwater line break. The MFRV is a fast-closing valve (10-second stroke time), which can terminate feedwater flow to the faulted SG in the case of an MSLB. If the MFRV fails to close, the current licensing basis credits the closure of the MFPDVs. Two MFPDVs must close to isolate feedwater flow from the faulted SG. The MFPDV has a closure time of 80 seconds.

Rather than crediting the MFPDVs as the backup for terminating feedwater flow, the licensee is proposing to credit the MFIVs. The licensee proposes to modify the existing manually-operated SG main feedwater inlet block valves located in the feedwater flow path to the SG in order to convert these valves to MFIVs. The valves will be modified by the addition of air operators that will provide an automatic closure upon receipt of a safety injection signal. The closure time for these valves will be 30 seconds. (32 seconds is the closure time assumed in the safety analyses to account for signal processing time). Since these valves are located closer to the SGs and are faster closing than the MFPDVs, less mass of feedwater will be added to the faulted SG following an MSLB. Only one MFIV will be added to each main feedwater line along with an associated bypass valve. Figure B 3.7.3-1 in the Bases of the Ginna TSs is being revised to show the location of these various valves. The revised figure was included in the licensee's April 29, 2005, application.

In order to accommodate this change, the licensee has reanalyzed the MSLB accident to credit the faster closing time of the MFIV and proposed changes to TS 3.7.3. On July 7, 2005, the licensee submitted an application requesting authorization to increase the steady-state power level from 1520 megawatts thermal (MWt) to 1775 MWt, which is a 16.8% increase and called

an extended power uprate (EPU). The licensee performed the calculations supporting the use of the MFIVs at the EPU conditions.

3.2 Effect on Supporting Accident Analyses

3.2.1 Core and Reactor Systems Response to Steam and Feed Line Breaks

The design basis for the MFIVs and the MFRVs is established by the analyses for the MSLB and feedwater line break (FWLB) accidents. In this regard, the NRC staff's review covered: (1) postulated initial core and reactor conditions; (2) methods of thermal and hydraulic analyses; (3) the sequence of events; (4) assumed responses of the reactor coolant and auxiliary systems; (5) functional and operational characteristics of the reactor protection system; (6) operator actions; (7) core power excursion due to power demand created by excessive steam flow; (8) variables influencing neutronics; and (9) the results of the transient analyses.

The steam release resulting from a rupture of a main steam pipe will result in an increase in steam flow, a reduction of coolant temperature and pressure, and an increase in core reactivity. The core reactivity increase may be enough to overcome the shutdown margin and cause a post-trip return to power. If the location of the MSLB is assumed to be inside containment, then the resulting steam release would also cause an increase in containment temperature and pressure. The steam release would depend upon, among other things, the SG shell-side water inventory and the rate and amount of feedwater flow delivery. The closure time of the MFIVs, therefore, would have a direct effect upon the steam release resulting from an MSLB. Reducing the time it takes to close the MFIVs, reduces the core cooldown and the mass and energy release into containment, as discussed in section 3.2.2 of this safety evaluation.

Since the most reactive rod cluster control assembly is assumed to be stuck in its fully withdrawn position, the high power peaking factors in the region of the stuck rod would exacerbate a return to power following a steam pipe rupture. The core is shut down by automatic boric acid injection into the RCS by the safety injection system. The rupture of a major steam line is the most-limiting cooldown transient, and it is analyzed assuming the plant is at zero power with no decay heat present. Decay heat would partially offset the RCS cooldown and reduce the possibility of a post-trip return to power.

Although this event is an American Nuclear Society (ANS) Condition IV (Reference 2) event, it was analyzed to meet the more restrictive Condition II acceptance criteria. The Condition II acceptance criteria prohibit the failure of any fuel cladding. Maintenance of fuel clad integrity is inferred from analysis results that indicate the departure from nuclear boiling (DNB) safety limit, expressed as the ratio of adjusted critical heat flux to calculated heat flux (DNBR), is not undermined.

The licensee has proposed to modify the existing manually-operated SG main feedwater inlet block valves by adding air-operated actuators that will automatically close these valves upon receipt of an actuation signal (a safety injection signal). These valves would then serve as MFIVs, in lieu of the MFPDVs, to isolate main feedwater flow. These valves would close in 30 seconds, much faster than the 80-second closure time for the MFPDVs. Since the MFIVs are situated closer to the SGs than the MFPDVs, there would be a smaller water volume resident in the lines downstream of the valves, and this would reduce the mass available for release inside containment.

Since the MFIVs are located in the main feedwater lines leading to each SG, closure of only one valve, the MFIV, is necessary to isolate feedwater to its associated SG. This is an improvement over the current configuration, wherein the MFPDVs are in parallel lines that feed a single common header that splits the flow to each SG. The current configuration requires both MFPDVs to close in order to isolate main feedwater flow in the event that either the MFRV and/or its associated bypass valve in a feedline fails to close.

The licensee evaluated the MSLB event for core response and for containment pressure response to MSLB mass and energy releases. These analyses were performed as part of the July 7, 2005, application (Reference 3) for an EPU, which is currently under review.

The RETRAN computer code (Reference 4), an NRC-approved code, was used to simulate the RCS conditions, particularly the core conditions, that would result from an MSLB. For the core response evaluation, a conservative sample of pertinent transient results (e.g., the core heat flux, core water temperature, and RCS pressure) were input to VIPRE, an NRC-approved code that models the core thermal and hydraulic conditions and calculates the corresponding DNBRs. Maintenance of the minimum transient DNBR at levels that consistently exceed the DNBR safety limit indicates that no fuel clad damage is predicted to occur. For the MSLB, this is the key ANS Condition II acceptance criterion. The MSLB accident analysis results show that the peak heat flux (13.3%) and minimum DNBR (2.58) occur about 54 seconds after the break occurs. Thus, the ANS Condition II acceptance criterion is met because the minimum DNBR is greater than the DNBR safety limit (1.566).

3.2.2 Containment Pressure and Temperature Response to Steam and Feed Line Breaks

The licensee performed MSLB calculations to support the proposed TS changes using NRC-approved methods for the mass and energy release into the containment (Reference 5). The containment pressure response has been reanalyzed using the GOTHIC 7.2 computer code (Reference 6), in accordance with methods previously approved in the NRC staff's safety evaluation (SE) dated September 30, 2003, to the licensee for the Kewaunee Nuclear Power Plant. When used in this way, the heat transfer model is referred to as the diffusion layer model (DLM).

Previously, containment pressure response analyses were performed with the COCO computer code (Reference 7). COCO is the Westinghouse computer code used for the current containment safety analyses in the Ginna UFSAR. The licensee stated that the GOTHIC model was benchmarked against the existing Ginna COCO model in order to keep the implementation of the plant input values on a consistent basis. GOTHIC plant input assumptions are the same as, or slightly more restrictive, than those in the current licensing-basis analysis, performed with the COCO code. A higher setpoint was assumed for the containment pressure high-high signal, which initiates containment spray. This is a conservative assumption because it delays the initiation of containment spray.

The Ginna GOTHIC containment evaluation model consists of a single lumped-parameter node. It uses the diffusion layer model for heat transfer to all structures in the containment. As previously stated, this heat transfer option was conditionally approved by the NRC (Reference 8) for use in Kewaunee containment analyses. The GOTHIC containment model for Ginna complies with the conditions of acceptance that were specified for the Kewaunee model. Kewaunee and Ginna are both enclosed in large, dry containments with comparable

containment volumes and active heat removal capabilities. Therefore, the use of the Ginna model is acceptable.

In its April 29, 2005, application, the licensee described the Ginna GOTHIC containment model and the assumptions used in the analyses. These include the assumed power levels at which the MSLB may occur, single failure assumptions, and containment description used in the analyses. In Table 1 of Enclosure 5 of the application, the licensee listed the initial conditions and parameters describing the containment spray and fan cooler systems. The NRC staff has reviewed the licensee's input and assumptions and found them acceptable since they covered a spectrum of power levels and single failures and conservatively model the containment for MSLBs.

The licensee also provided the results of the MSLB accident calculations for the most limiting conditions. The limiting conditions include: the power level of 1817 MWt analyzed in the EPU application, a shutdown margin of 1.3% (reduced from the current analysis), and the failure of a vital bus (the worst single failure), which results in the loss of one train of containment cooling and one train of safety injection). An initial power level of 70% gave the most limiting results.

The containment pressure response analysis credits feedwater isolation due to the automatically actuated MFIV with a stroke time of 30 seconds, assuming the faulted loop MFRV fails open. Although the EPU conditions result in higher mass and energy releases early in the transient, the faster MFIV closure reduces delivery of main feedwater to the faulted SG, which results in a lower peak containment pressure (50.8 psig), when compared to 59.8 psig from the previous analysis, which credited feedwater isolation due to closure of the MFPDVs with an 80-second stroke time.

Crediting the automatic operation of the MFIVs shifts the worst single failure from an MFRV failure to a vital bus failure, and the most conservative initial power level from 100% to 70% of the EPU level. The results for this MSLB containment pressure response case yield a peak containment pressure of 59.4 psig, just below the containment design pressure of 60.0 psig.

The NRC staff's review of the licensee's MSLB analyses, for core and containment pressure responses, indicates that the analyses were performed using acceptable analytical models, and that the acceptance criteria, with respect to the DNB and containment pressure safety limits have been met.

During its review, the NRC staff noted that the proposed modification adds automatic air-operated actuators to the manual block valves that isolate main feedwater and that these actuators would be designed to fail closed to shut the valves. Therefore, failure of this added equipment could cause a loss of feedwater (LOFW) event. The NRC staff asked the licensee to explain how the proposed change would not increase the probability of an LOFW event. The licensee's response dated November 21, 2005, stated that, although the added equipment could result in an increase in the frequency of occurrence of an LOFW event, this increase would not be great enough to affect the event's classification as an ANS Condition II event. In other words, after the proposed change is made, the LOFW would remain an event of Moderate Frequency. Therefore, there would be no effect upon the licensing basis or the accident analyses thereof. The NRC staff finds this rationale acceptable and notes that, after the proposed change is implemented, the main feedwater system at Ginna will be similar to the

main feedwater system designs at most Westinghouse plants, which also treat the loss of main feedwater event as a Condition II event.

The NRC staff reviewed the licensee's evaluations, analyses and proposed TS changes to support changes to the Ginna TSs to allow use of the MFIVs in lieu of the MFPDVs to terminate main feedwater flow to the SGs in the event of an MSLB. The NRC staff found that the licensee's calculations supporting the proposed TS changes were done with acceptable methods and that conservative input and assumptions were used. The proposed TS changes, as discussed in this evaluation, adequately reflect the results of the supporting analyses. Therefore, the staff concludes that the proposed TSs changes are acceptable for implementation at Ginna.

3.3 Design, Qualification, and Testing of the MFIVs

The NRC staff conducted a technical evaluation of the proposed modification of the MFIVs at Ginna in support of the application dated April 29, 2005, that will rely on the MFIVs to isolate the SGs in the event of a steam line break. In its letters dated November 21, 2005, and January 25, 2006, the licensee provided additional information in response to questions from the staff.

Based on its review of the proposed TS amendment, the NRC staff requested the licensee to discuss the qualification of the structural capability of the MFIVs to perform their new safety function. In its RAI response dated November 21, 2005, the licensee reported that the MFIVs are Seismic Category I and that the new actuators have been procured as Seismic Category I. The licensee re-evaluated the piping system for the new loads associated with the actuators. The load combinations used for the design of the MFIV modification are consistent with the Ginna licensing basis as previously reviewed by the NRC staff.

The NRC staff requested the licensee to discuss the calculation of the thrust necessary to operate the MFIVs under the applicable pressure and flow conditions for their new safety function. In its RAI response dated November 21, 2005, the licensee reported that the thrust necessary to operate the MFIVs had been calculated by the globe valve manufacturer based on worst-case differential pressure and flow conditions provided in the design specification of 450 psid and 8,000,000 lbs/hr. The manufacturer determined the required thrust to overcome valve friction based on industry experience, and to overcome the pressure and forces due to flow acting on the valve plug and stem. In its January 25, 2006, letter, the licensee stated that the analysis of the forces required to overcome the pressures due to flow acting on the valve plug and stem was consistent with the methods recommended in the Electric Power Research Institute (EPRI) Air-Operated Valve (AOV) User's Guide.

The NRC staff requested the licensee to discuss the qualification of the actuators to be installed on the MFIVs. In its November 2005 RAI response, the licensee stated that the valve actuators will be certified according to 10 CFR Part 50, Appendix B, by the actuator manufacturer. The actuators were being procured as safety-related components and selected based on historical performance. In its January 25, 2006, letter, the licensee described the qualification of the actuators in more detail. In particular, the manufacturer will conduct a pneumatic pressure test and a breakaway pressure test to quantify actuator internal friction forces. The thrust capability of the actuator will be determined by analysis from the thrust available based on the actuator air pressure and area, less the actuator internal forces. The results of these analyses and tests

will be used to demonstrate the output thrust capability of the actuator and to assure that it meets the design specification.

The NRC staff requested the licensee to discuss the monitoring and surveillance of the performance of the MFIVs as part of the IST Program at Ginna. In its November 2005 RAI response, the licensee reported that the valves will be stroke-time tested and position-indication verified during cold shutdown in accordance with the IST Program. The valves will also be included in the Ginna AOV Program, and baselined and monitored for degradation to ensure reliable performance. The licensee provided additional information regarding the surveillance and monitoring of the MFIVs in its January 25, 2006, letter. In particular, the licensee reported that the Ginna AOV Program is based on guidance from the Joint Owners' Group (JOG) on AOVs for verification of valve functionality at design conditions and long-term periodic verification. The JOG AOV program includes lessons learned from the MOV programs at nuclear power plants. Further, the licensee reviewed NRC Regulatory Issue Summary (RIS) 2000-003, "Resolution of Generic Safety Issue 158: Performance of Safety-Related Power-Operated Valves Under Design Basis Conditions," in developing the Ginna AOV program. The licensee also evaluated the lessons learned from the Joint Owners' Group Program on MOV Periodic Verification, and found no necessary actions related to degradation of the globe valves used in the MFIVs.

Based on the information provided by the licensee, the NRC staff finds the licensee's design, qualification, and testing of the MFIVs at Ginna in support of the proposed TS amendment to rely on the MFIVs to isolate the SGs in the event of a steam or feedwater line break to be acceptable.

3.4 Changes to TS 3.7.3

In its April 29, 2005, application, the licensee proposed a number of changes to the TSs, associated with the proposed modification. Basically, these changes are necessitated by the addition of the automatic closure function to the MFIVs and its role in accident mitigation. In this regard, the licensee proposed several changes to TS Section 3.7.3 as follows:

1. Changes to reflect the fact that the MFIVs are to be credited as the backup isolation valves rather than the MFPDVs.

    a. Revise the page headers for TS 3.7.3 that currently state "MFRVs, Associated Bypass Valves, and MFPDVs" to state "MFIVs, MFRVs, and Associated Bypass Valves."

    b. Revise the title for TS 3.7.3 from the current "Main Feedwater Regulating Valves (MFRVs), Associated Bypass Valves, and Main Feedwater Pump Discharge Valves (MFPDVs)" to "Main Feedwater Isolation Valves (MFIVs), Main Feedwater Regulating Valves (MFRVs), and Associated Bypass Valves."

    c. Revise TS 3.7.3 Limiting Condition for Operation (LCO) which currently states "Two MFRVs, associated bypass valves, and two MFPDVs shall be OPERABLE" to state "Two MFIVs, two MFRVs, and associated bypass valves shall be OPERABLE."

d. Revise TS 3.7.3 LCO Condition A, which currently states “One or more MFPDV(s) inoperable,” to state “One or more MFIV(s) inoperable.”

As discussed in Section 3.2 of this SE, the NRC staff found the proposed change from the MFPDVs to the MFIVs acceptable. Thus, these changes are consistent with the supporting analyses.

2. The licensee proposed the following changes to the Required Actions in TS 3.7.3:

a. Change Required Action A.1 from “Close MFPDV(s)” to “Close or isolate MFIV(s).”

b. Change Required Action A.2 from “Verify MFPDV(s) is closed” to “Verify MFIV(s) is closed or isolated.”

The NRC staff finds the changes acceptable since either closing the inoperable valve or isolating the valve restores the safety function. The alternative to isolate or to close is consistent with the NUREG-1431, “Westinghouse Improved Standard Technical Specifications,” Revision 3. In response to the staff’s request that the licensee clarify how the isolation requirement of Required Action A.1 would be implemented, the licensee stated in its November 21, 2005, letter that:

> An orderly shutdown will commence, and once the steam generator on the affected train is being supplied Auxiliary Feedwater sufficient to maintain level, the valve will be isolated.

The Bases for NUREG-1431 does not provide explicit guidance on implementing the isolation. The licensee also stated in the same letter that there will be no impact on the function of the Auxiliary Feedwater System.

3. The licensee proposed to revise the Completion Time for Required Actions A.1, B.1, and C.1 from the currently stated “24 hours” to “72 hours.”

The NRC staff finds the proposed extension of the completion time to be consistent with the time in NUREG-1431 Bases, which justifies 72 hours based upon the redundancy afforded by the remaining OPERABLE valves and the low probability of an event occurring during this time period that would require isolation of the main feedwater flow paths. Since the main feedwater system configuration of NUREG-1431, Revision 3, is similar to that proposed by the licensee, the staff finds the revision of the completion time from 24 to 72 hours to be acceptable.

4. The licensee proposed to change the entry requirements for Condition D, which currently read “Required Action and associated Completion Time for Conditions A, B, and C not met,” to “Required Action and associated Completion Time not met.”

The NRC staff finds that this wording is consistent with NUREG-1431, Revision 3, and is acceptable since the main feedwater isolation system configuration of NUREG-1431 is similar to that at Ginna.

5. The licensee proposed to revise the requirements for Condition E as follows:

   a. The entry requirements currently state:

      "One or more MFPDV(s) and one or more MFRV(s) inoperable.

      <u>OR</u>

      One or more MFPDV(s) and one or more MFRV bypass valve(s) inoperable."

      Condition E is being revised to state "Two valves in the same flowpath inoperable."

   b. Required Action E.1, which currently states "Enter LCO 3.0.3," is being revised to state "Isolate affected flowpath."

   c. The completion time for Required Action E.1 currently requires the action to be carried out "immediately." The completion time is being revised to require the action to be carried out in "8 hours."

   The NRC staff finds that, if both an MFRV and an MFIV in the same flowpath were inoperable, there may be no redundant system to operate automatically to perform the required main feedwater isolation safety function. The Bases in NUREG-1431 also point out that such a failure of two valves in the same line may indicate a common cause failure. Under these conditions, the proposed change would require that the affected valves in each flow path must be restored to OPERABLE status within 8 hours. This completion time is consistent with NUREG-1431, Revision 3. The elimination of the required entry into LCO 3.0.3 is acceptable because, by the licensee's description of isolation, above, the plant would still be shutdown. Since LCO 3.0.3 is prescriptive as to the time for shutdown and the isolation requirement is not (a time for isolation is specified but not a time for shutdown), a slightly longer shutdown may occur. Since the proposed changes are the same as the corresponding requirements of NUREG-1431, Revision 3, and since the main feedwater isolation system configuration of NUREG-1431, Revision 3 is similar to that at Ginna, the NRC staff finds these proposed changes to be acceptable.

6. The licensee proposed to renumber Conditions D and E as follows:

   a. Condition D and Required Actions D.1 and D.2 would be renumbered as E, E.1 and E.2, respectively.

   b. Conditions E and Required Action E.1 are being renumbered as D and D.1, respectively.

   The NRC staff finds that exchanging the placement of Conditions D and E and their associated Required Actions and Completion Times, as modified by other proposed changes, has no effect on safety and is acceptable.

7. The licensee proposed to revise the valve closure time requirements in TS Surveillance Requirement (SR) 3.7.3.1 by changing the SR from "Verify the closure time of each MFPDV is # 80 seconds on an actual or simulated actuation signal," to "Verify the closure time of each MFIV is # 30 seconds on an actual or simulated actuation signal."

   The NRC staff finds that the change from MFPDVs to MFIVs as the backup main feedwater isolation valves is acceptable from a safety analysis perspective since:

   a. Crediting the MFIVs as the backup isolation valves to the MFRVs requires only one valve per line to close. Previously, the MFPDVs were credited with this backup function. This configuration requires both MFPDVs to close to provide feedwater isolation in the event of a single failure of either MFRV and/or the associated bypass valve in one flowpath to an individual steam generator.

   b. The MFIV closure time is significantly shorter than that for the MFPDV (30 seconds versus 80 seconds). Therefore, the water volume downstream of the valves is reduced. This results in less mass and energy released inside the containment.

   c. The MFIVs are closer to the steam generators. Therefore, the water volume downstream of the valves is reduced. This also results in less mass and energy released inside the containment.

   d. Because of the above, the RCS cooldown and containment conditions (pressure, temperature) at the proposed EPU conditions are less than the respective design limits.

   Therefore, the NRC staff finds the proposal to use the MFIVs rather than the MFPDVs as backup to the MFRVs for main feedwater isolation and the revision of the SR for closure time to be acceptable. The TS change relates directly to accident analyses is TS 3.7.3, and the specified closure time is shown to provide adequate protection by the results of accident analyses, which were performed using accepted methods. The licensee assumes a closure time of the MFIVs of 32 seconds (30 second closure plus 2 seconds for instrumentation response).

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the New York 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 and changes surveillance requirements. 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 (70 FR 33218). 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.

6.0 CONCLUSION

The Commission 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 the amendment will not be inimical to the common defense and security or to the health and safety of the public.

7.0 REFERENCES

1. NUREG-0800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants, LWR Edition,” July 1981.

2. ANS Standard 51.1, “Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants,” 1983 (replaces American National Standards Institute (ANSI) Standard ANSI N18.2).

3. Ginna LLC letter, M. Kornick to NRC, “License Amendment Request Regarding Extended Power Uprate,” dated July 7, 2005.

4. Huegel, D. S., et al., Westinghouse Report WCAP-14882-P-A (Proprietary), “RETRAN-02 Modeling and Qualification for Westinghouse Pressurized Water Reactor Non-LOCA Safety Analyses,” April 1999.

5. Burnett, T. W., et al.,Westinghouse Report WCAP 7907-P-A (Proprietary), “LOFTRAN Code Description,” April 1984.

6. Numerical Applications, Inc. (NAI) Report 8907-02, Revision 14, "GOTHIC Containment Analysis Package User Manual," Version 7.1, January 2003; NAI 8907-06, Revision 13, "GOTHIC Containment Analysis Package Technical Manual," Version 7.1, January 2003; and NAI 8907-09, Revision 7, "GOTHIC Containment Analysis Package Qualification Report," Version 7.1, January 2003.

7. Bordelon, F. M., and E. T. Murphy, Westinghouse Report WCAP-8326 (Non-Proprietary), "Containment Pressure Analysis Code (COCO)," and WCAP- 8327 (Proprietary), July 1974.

8. NRC Letter, A. C. McMurtray to T. Coutu (NMC), “Kewaunee Nuclear Plant - Issuance of Amendment,” Enclosure 2, Safety Evaluation, September 29, 2003.

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