February 25, 2005

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: MODIFICATION OF THE CONTROL ROOM EMERGENCY AIR TREATMENT SYSTEM AND CHANGE TO DOSE CALCULATION METHODOLOGY TO ALTERNATE SOURCE TERM (TAC NO. MB9123)

Dear Mrs. Korsnick:

The Commission has issued the enclosed Amendment No. 87 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 May 21, 2003, as supplemented December 1, 2003 (two letters), February 16, March 1 and 8, April 22, May 21, July 8 and 14, August 6 and 18, September 10, October 14 and 18, December 3 and 6, 2004, and January 27, 2005.

The amendment revised the Ginna Technical Specifications (TSs) for the following sections: Definitions (Section 1.1), Control Room Emergency Air Treatment System Actuation Instrumentation (Section 3.3.6), Reactor Coolant System Specific Activity (Section 3.4.16), Containment Spray, Containment Recirculation Fan Cooler, and NaOH Systems (Section 3.6.6), Control Room Emergency Air Treatment System (Section 3.7.9), Programs and Manuals (Section 5.5), and Reporting Requirements (Section 5.6). The TS changes were made to reflect design modifications to the Control Room Emergency Air Treatment System, and elimination of the requirements for the Containment Post Accident Charcoal Filters. A detailed analysis using the alternate source term was used to determine the radiological dose and toxic gas consequences of the proposed design modifications.

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/***

Donna M. Skay, Senior Project Manager, Section 1
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-244

Enclosures: 1. Amendment No. 87 to Renewed License No. DPR-18
2. Safety Evaluation

cc w/encls: See next page

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cc w/encls: See next page

Accession Number: ML050320491 *Input provided by safety evaluation

| OFFICE | PDI-1\PM | PDI-1\LA | EEIB-A\SC | EEIB-B\SC | EMCB-C/SC | EMEB-B\SC |
|---|---|---|---|---|---|---|
| NAME | DSkay | SLittle | EMarinos* | RJenkins* | LLund* | KManoly* |
| DATE | 2/11/05 | 02/07/05 | 01/11/05 | 01/21/05 | 04/07/04 | 01/04/05 |

| OFFICE | SPLB-B/SC | SPSB-C | OGC | PDI-1/SC | | |
|---|---|---|---|---|---|---|
| NAME | SWeerakkody* | RDennig* | Mzobler | RLaufer | | |
| DATE | 09/21/04 | 02/08/05 | 2/23/05 | 2/23/05 | | |

Official Record Copy

DATED: February 25, 2005

AMENDMENT NO. 87 TO RENEWED FACILITY OPERATING LICENSE NO. DPR-18 GINNA NUCLEAR POWER PLANT

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EMarinos
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KManoly
SWeerakkody
RDennig
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OGC
GHill (2)
TBoyce
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GMatakas, RI
RClark
SLittle

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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. 87
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 May 21, 2003, as supplemented December 1, 2003 (two letters), February 16, March 1 and 8, April 22, May 21, July 8 and 14, August 6 and 18, September 10, October 14 and 18, December 3 and 6, 2004, and January 27, 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. 87, 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 upon completion of the installation and testing of the new Control Room Emergency Air Treatment System.

FOR THE NUCLEAR REGULATORY COMMISSION

***/RA/***

Richard J. Laufer, Chief, Section 1
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: February 25, 2005

ATTACHMENT TO LICENSE AMENDMENT NO. 87

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 |
|---|---|
| 1.1-2 | 1.1-2 |
| 3.3.6-1 to 3.3.6-3 | 3.3.6-1 to 3.3.6-3 |
| 3.4.16-1 | 3.4.16-1 |
| 3.6.6-1 to 3.6.6-4 | 3.6.6-1 to 3.6.6-3 |
| 3.7.9-1 to 3.7.9-3 | 3.7.9-1 to 3.7.9-2 |
| 5.5-5 to 5.5-10 | 5.5-5 to 5.5-10 |

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 87 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 May 21, 2003 (ML003150522), as supplemented December 1, 2003, (two letters, ML033430141, ML033420172), February 16, (ML040550267), March 1, (ML040680923),
March 8, (ML040750453), April 22, (ML041211003), May 21, (ML041480294), July 8, (ML041970380), July 14, (ML042020571), August 6, (ML042260356), August 18, (ML042390442), September 10, (ML042650403), October 14, (ML042960082), October 18, (ML043010606), December 3, (ML043450332), December 6, 2004, (ML043480166), and January 27, 2005, (ML050330383), the R. E. Ginna Nuclear Power Plant, Inc. (the licensee) submitted a request for changes to the R. E. Ginna Nuclear Power Plant Technical Specifications (TSs). The requested TS changes were the results of proposed modifications to the control room emergency air treatment system (CREATS) to improve system reliability, performance, and redundancy. The proposed amendment would also revise the Ginna design basis to replace the existing accident radiological source term (TID-14844) by a full implementation of the alternative source term (AST). As part of implementing the AST, Ginna proposed to eliminate the requirements for Containment Post Accident Charcoal Filters from the Ginna TSs. The supplemental 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 July 8, 2003 (68 FR 40718).

### 1.1 Background

The control room ventilation system consists of two separate subsystems; the normal heating, ventilation and air conditioning (HVAC) system and the CREATS, both of which serve the control room emergency zone (CREZ). The CREZ is limited to the top floor of the three-story control building. It includes the control room, bathroom, kitchen, and shift supervisor's office and all CREATS ductwork bounded by the CREZ isolation dampers. These two systems provide three different modes of operation; Normal, Purge and Emergency.

## 1.2 Normal HVAC System

The normal control room HVAC system is located on the bottom floor of the control building and is connected to the CREZ by supply and return ducts. In the Normal mode of operation, this system provides the control room with fresh outside air, exhaust, coarse filtration, and temperature control to provide the operators with a safe and comfortable working environment. In the Purge mode of operation, this system provides the maximum amount of fresh air to purge airborne contaminants from the CREZ. The normal HVAC system's outside air intake duct is equipped with redundant trains of radiation, chlorine, and ammonia monitors, any of which will actuate the emergency mode of operation and provide an alarm in the control room. The normal HVAC system is also equipped with a smoke detector upstream of the normal return air fan to monitor the return air flow from the CREZ and to provide an alarm in the control room. An evaluation of the control room ventilation system fire and smoke protection features is given in Section 3.9.2.

## 1.3 Control Room Emergency Air Treatment System

The CREATS is normally in standby and is configured to provide zone isolation, re-circulation and filtration under accident conditions. The system is not designed to pressurize the CREZ in any mode of operation. It is designed to satisfy General Design Criteria (GDC) 19, "Control room," to provide a protective environment from which the operators can control the plant for 30 days after a design-basis accident (DBA) without exceeding dose limits of 30-rem thyroid or 5-rem whole body. The CREATS is also designed to protect the operators from exposure to toxic gas following an accidental release from sources on or near the Ginna site. Detailed dose analyses are given in Section 3.2 and a detailed toxic gas analysis is given in Section 3.4

The CREATS consists of two seismic category 1, 100% capacity trains that are designed to filter, cool, heat, and recirculate 6000 cfm (± 10%) of control room air. The CREATS fans are powered from Class 1E safeguard buses and will start upon a manual, toxic gas, radiation, or safety injection (SI) signal. The heating and cooling coils installed within the CREATS are designed to maintain the CREZ between a minimum temperature of 50 degrees F and a maximum temperature of 104 degrees F to support human habitability and equipment operation. The heating coils and the condensing units (compressor, fan, electric controls, etc.) are powered by the safeguard buses and are classified as safety-related, seismic category 1. The cooling coils, condensing units and associated piping are part of the Control Room Emergency Cooling System (CRECS) which is an auxiliary support system for the CREATS. The heating and cooling loads associated with these systems are automatically stripped from the safeguard buses by the SI signal. These loads can be manually restored after the SI signal is reset to maintain control room habitability. An evaluation of the cable separation, diesel generator loading analysis, and instrumentation for the above systems is given in Sections 3.6 and 3.7.

The major components of the CREATS are located inside the relay room annex which is east of, and one level below, the control room. The relay room annex is a hardened structure having reinforced concrete walls and roof. Ductwork connects the CREATS to the CREZ via penetrations in the roof of the relay room annex and in the east wall of the control room. This ductwork is a CREZ isolation boundary and, because it is located outside, it is designed to survive tornado-driven missiles from a tornado of up to 132 mph wind speed. An evaluation of the seismic analysis of the external ductwork is given in Section 3.5.

Appendix R analysis, environmental and dynamic effects, control room habitability requirements, and TS changes are evaluated in Sections 3.8, 3.9, 3.10, and 3.11, respectively.

## 2.0 REGULATORY EVALUATION

The Nuclear Regulatory Commission (NRC) staff's evaluation of the proposed design changes to the CREATS is based upon the following regulatory codes, guides, and standards:

Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50

10 CFR Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants"

GDC 2, "Design bases for protection against natural phenomena," requires that the system be capable of withstanding the effects of natural phenomena without loss of capability to perform their safety functions.

GDC 4, "Environmental and dynamic effects design bases" requires that systems, structures, and components important to safety be designed to accommodate the effects of and be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents.

GDC 17, "Electric power systems," requires, in part, that nuclear power plants have onsite and offsite electric power systems to permit the functioning of structures, systems, and components that are important to safety. The onsite system is required to have sufficient independence, redundancy, and testability to perform its safety function, assuming a single failure.

GDC 19, "Control Room," requires that the control room be maintained in a safe, habitable condition under accident conditions.

GDC 20, "Protection system functions," requires, in part, protection systems to sense accident conditions and to initiate the operation of systems and components important to safety.

GDC 21, "Protection system reliability and testability," requires, in part, that no single failure results in loss of the protection function.

GDC 60, "Control of releases of radioactive materials to the environment," requires that the system be able to suitably control release of gaseous radioactive effluents to the environment.

10 CFR Part 50.36, "Technical specifications"
10 CFR Part 50.48, Appendix R, "Fire protection"
10 CFR Part 50.67, "Accident source term"

Regulatory Guides (RGs)

RG 1.23, “Onsite Meteorological Programs”

RG 1.29, “Seismic Design Classification,” position C.1 for safety-related portions of the system and position C.2 for nonsafety-related portions of the system.
RG 1.52, “Design, Inspection, and Testing Criteria for Air Filtration and Adsorption Units of Post-Accident Engineered-Safety-Feature Atmosphere Cleanup Systems in Light-Water-Cooled Nuclear Power Plants”
RG 1.78, “Evaluating the Habitability of a Nuclear Power Plant Control Room during a Postulated Hazardous Chemical Release”
RG 1.105, “Setpoints for Safety-Related Instrumentation”
RG 1.145, “Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants”
RG 1.183, “Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors”
RG 1.194, “Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants”
RG 1.196, “Control Room Habitability at Light-Water Nuclear Power Reactors”
RG 1.197, “Demonstrating Control Room Envelope Integrity at Nuclear Power Reactors”
RG 1.75, “Physical Independence of Electric Systems”
RG 1.9, “Selection, Design, Qualification, and Testing of Emergency Diesel Generators Units Used as Class 1E Onsite Electric Power Systems at Nuclear Power Plants”

Standard Review Plans (SRPs)

SRP Section 2.3.4, “Short-Term Diffusion Estimates for Accidental Atmospheric Releases”
SRP Section 6.4, “Control Room Habitability System”
SPR Section 9.4.1, “Control Room Area Ventilation System”
SRP Section 15.0.1, “Radiological Consequence Analyses Using Alternative Source Terms”

## 3.0 TECHNICAL EVALUATION

### 3.1 Atmospheric Dispersion Estimates

#### 3.1.1 Meteorological Data

The licensee used 5 years of hourly onsite meteorological data collected during calendar years 1999 through 2003 to generate new atmospheric dispersion factors ($\chi/Q$ values) for use in this amendment request. These data were provided for NRC staff review in the form of hourly meteorological data files (for input into the ARCON96 atmospheric dispersion computer code) and a joint frequency distribution (for input to the PAVAN atmospheric dispersion computer code). The data were used to generate control room (CR), exclusion area boundary (EAB), and and low population zone (LPZ) $\chi/Q$ values for all the accidents evaluated in this license amendment request (LAR). The licensee’s data recovery rate exceeded 90 percent during this period of record. The licensee stated that the data was collected by a meteorological monitoring program that meets the criteria in RG 1.23, “Onsite Meteorological Programs.”

All releases were modeled as ground level releases. Hourly data from the 33-foot (10-meter)

and 150-foot (45.7-meter) levels on the onsite meteorological tower were provided as input to ARCON96 whereas the joint frequency distribution used as input to PAVAN was compiled using wind data from the 10-meter level. Stability class was based on delta-temperature measurements made between the 45.7-meter and 10-meter levels on the onsite meteorological tower.

The staff performed a quality review of the ARCON96 hourly meteorological database using the methodology described in NUREG-0917, "Nuclear Regulatory Commission Staff Computer Programs for Use with Meteorological Data." Further review was performed using computer spreadsheets. Examination of the data revealed that wind speed and direction frequency occurrences at both levels were reasonablely similar from year to year. Regarding atmospheric stability, stable and neutral conditions were generally reported to occur at night and unstable and neutral conditions during the day, as expected. There was an appreciable variability in reported unstable (Pasquill stability classes A, B, and C) conditions from year to year, ranging from a low of 8.9% in 1999 to a high of 26.8% in 2001. This phenomenon could be explained by the site's proximity to a large body of water (Lake Ontario) as well as the relative short distance between the delta-temperature sensors (35.7 meters).

However, the occurrence of moderately stable (Pasquill stability F) and extremely stable (Pasquill stability G) conditions remained relatively consistent from year to year, ranging from a combined total low of 16.1% in 2000 to a combined high of 22.0% in 1999. Since the determination of DBA atmospheric dispersion factors for ground level releases is mostly a function of the frequency of occurrence of stable conditions, the variability in the annual frequency of reported unstable conditions should not have a significant impact on the resulting $\chi/Q$ values.

A comparison of the joint frequency distribution derived by the staff from the licensee's hourly ARCON96 database with the joint frequency distribution used by the licencee as input to PAVAN showed reasonably good agreement. A comparison of lower level 1999–2003 wind frequency distribution with the lower level 1966–1967 and 1973–1974 wind direction frequency distribution presented in Ginna updated final safety analysis report (UFSAR) Tables 2.3-9a through 2.3-9g shows good agreement as well.

In summary, the staff has reviewed the available information relative to the onsite meteorological measurements program and the 1999 through 2003 ARCON96 and PAVAN meteorological data input files provided by the licensee. On the basis of this review, the staff concludes that the 1999 through 2003 onsite data provide an acceptable basis for making estimates of atmospheric dispersion for DBA assessments.

### 3.1.2 CR Atmospheric Dispersion Factors

The licensee calculated CR air intake $\chi/Q$ values to evaluate containment leakage, containment equipment hatch (roll-up door), atmospheric relief valve, plant vent, auxiliary building leakage, main steam line break, and spent fuel pool releases using the 1999 through 2003 onsite meteorological data and the ARCON96 computer code (NUREG/CR-6331, Revision 1, "Atmospheric Relative Concentrations in Building Wakes"). Staff qualitatively reviewed the inputs to the ARCON96 calculations and found them generally consistent with site configuration drawings and staff practice. Specific areas of note are as follows:

- All postulated releases were considered ground level releases. Both lower level (10-meter) and upper level (45.7-meter) wind data were provided as input to ARCON96. Stability class was determined as a function of delta-temperature measurements between the 45.7-meter and the 10-meter levels on the onsite meteorological tower.

- Containment leakage was modeled as a ground-level vertical area source. Leakage was assumed to be distributed over the containment surface and all penetrations. The source width was the containment outside diameter and the source height was the distance from the ground to the top of the containment dome. The initial diffusion coefficients were determined by dividing the source dimensions by a factor of six in accordance with RG 1.194, "Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants." The distance-to-receptor was set as the shortest horizontal distance from the containment surface to the CR air intake and the source height was the midpoint elevation of the containment. The resulting $\chi/Q$ values were used to model releases for the loss-of-coolant and rod ejection (containment leakage) accidents.

- Containment equipment hatch leakage for a fuel-handling accident (FHA) was also modeled as a ground-level vertical area source. Leakage was assumed to occur through the open hatch via the perimeter seals of a roll-up door. The source dimensions were the face area of the roll-up door. The initial diffusion coefficients were determined by dividing the source dimensions by a factor of six in accordance with RG 1.194. The distance-to-receptor was set as the shortest horizontal distance from the roll-up door perimeter to the CR air intake and the release height was conservatively set to the top of the roll-up door. The resulting $\chi/Q$ values were used to model releases for the FHA inside containment.

- Atmospheric relief valve (ARV) releases were modeled as a ground-level point source. The release height was equal to the height of the ARV riser. The source-to-receptor distance was based on the valve discharge that was closest to the CR air intake. The resulting $\chi/Q$ values were used to model releases for the steam generator tube rupture, locked rotor, and rod ejection (secondary side release) accidents.

- Plant vent releases were modeled as a ground-level horizontal area source. The source dimension was the vent diameter. The initial horizontal diffusion coefficient was determined by dividing the source dimension by a factor of six in accordance with RG 1.194. The release height was set equal to the plant vent height. The resulting $\chi/Q$ values were used to model releases for the FHA in the spent fuel pool.

- Five potential auxiliary building leakage pathways were evaluated (i.e., the north wall surface, the back-draft damper, the building vent intake, and two exterior doors) and the resulting $\chi/Q$ values from the bounding release pathway were used to represent auxiliary building leakage. Each pathway was modeled as a ground-level vertical area source. The initial diffusion coefficients were determined by dividing the source dimensions by a factor of six in accordance with RG 1.194. The resulting $\chi/Q$ values were used to model releases for the loss-of-coolant (emergency core cooling system (ECCS) leakage).

- Main steam line break releases outside containment were modeled as a ground-level point

source. The rupture site was assumed to be in the 36-inch header which is located inside the Turbine Building. The release of steam inside the Turbine Building is assumed to blow out windows and metal siding. The distance-to-receptor was based on the distance from the header midpoint to the CR air intake; the confinement of the plume within the Turbine Building was not considered. The release height was set as the distance to the top of the header as plume rise effects of such a high energy release were conservatively ignored in the licensee's modeling.

- Spent fuel pool releases due to a tornado missile accident were modeled as a ground-level horizontal area source. The tornado missile accident assumes that a utility pole propelled by the wind penetrates the auxiliary building roof and impacts the fuel stored in the spent fuel storage pool. The diffuse horizontal area source was based on the surface area of the spent fuel pool. During the first minute of the accident, the resulting radiological releases were assumed to be dispersed by the "tornado conditions" that caused the accident. These tornado conditions were represented by the highest wind speed recorded at the 10-meter level of the onsite meteorological tower during the period 1999–2003 (stability D with a 22.1 m/s wind speed). For subsequent time periods, dispersion factors were determined using the typical ARCON96 model results.

Note that the CR air intake $\chi/Q$ values from the analyses discussed above were used to analyze both pre-isolated outside air and unfiltered inleakage to the control room. That is, the CR air intake $\chi/Q$ values were considered to be bounding for all potential inleakage pathways to the control room. This assumption is reasonable, given that the CR air intake is located on the Control Room Building roof towards the edge of the roof that is closest to the major potential release pathways. The source-to-inleakage location for all possible inleakage points (besides the air intake) is through other structures first, resulting in torturous paths and longer source-to-receptor distances.

In summary, the staff has reviewed the licensee's assessments of CR post-accident dispersion conditions generated from the licensee's meteorological data and atmospheric diffusion modeling. The resulting CR $\chi/Q$ values are presented in Table 2. On the basis of this review, the staff concludes that these CR $\chi/Q$ values are acceptable for use in DBA CR dose assessments.

### 3.1.3 Exclusion Area Boundary (EAB) and Low Population Zone (LPZ) Atmospheric Dispersion Factors

The licensee calculated EAB and LPZ $\chi/Q$ values for all accident scenarios (except for the initial period of the spent fuel pool tornado missile accident) using the 1999 through 2003 onsite meteorological data and the PAVAN computer code which implements the guidance provided in RG 1.145, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants." The EAB distance ranges from 450 meters to 8000 meters as a function of downwind sector (as presented in Ginna UFSAR Table 2.3-20) and the LPZ distance is 4827 meters in all downwind sectors. All releases were considered to be ground level. A containment building height of 36 m and a minimum containment building facade cross-sectional area of 1850 $m^2$ were used to model building wake effects.

Staff quantitatively reviewed the inputs to the PAVAN computer runs and found the inputs to be generally consistent with site configuration drawings and RG 1.145.

The licensee modeled the first minute of the spent fuel pool tornado missile accident using the CONHAB module of the HABIT computer code (NUREG/CR-6210, Supp. 1, "Computer Codes for Evaluation of Control Room Habitability [HABIT V1.1]"). The χ/Q values during the first minute of this accident were calculated assuming "tornado conditions" that were represented by F stability and 22.1 m/s atmospheric conditions. The staff confirmed the licensee's results by calculating similar dispersion factors using the ARCON96 computer code. For subsequent time periods, the licensee determined χ/Q values for the spent fuel pool tornado missile accident using the PAVAN model results derived for the control room.

In summary, the staff has reviewed the licensee's assessments of EAB and LPZ post-accident dispersion conditions generated from the licensee's meteorological data and atmospheric diffusion modeling. The resulting EAB and LPZ χ/Q values are presented in Table 3. On the basis of this review, the staff concludes that these EAB and LPZ χ/Q values are acceptable for use in DBA, EAB, and LPZ dose assessments.

### 3.2 Radiological Consequences of Design Basis Accidents

In accordance with the guidance in RG 1.183, a licensee is not required to re-analyze all DBAs for the purpose of the AST implementation, only those affected by the proposed changes. However, on approval of this LAR, the AST as described in RG 1.183 and the total effective dose equivalent (TEDE) dose criteria in 10 CFR 50.67 will become the licensing basis for all subsequent radiological consequence analyses intended to show compliance with 10 CFR Part 50 requirements. In keeping with this guidance, the licensee performed an evaluation of previously analyzed DBAs to decide which, if any, were affected by the proposed amendment. The licensee re-analyzed the radiological consequences of the following DBA events:

- Loss-of-coolant accident
- Fuel-handling accident
- Main steamline break accident
- Steam generator tube rupture accident
- Reactor coolant pump locked rotor accident
- Rod ejection accident
- Tornado missile in spent fuel pool accident

In its supplement dated December 3, 2004, the licensee stated that the source term for the gas decay tank rupture was not changed from previous analysis, and therefore, this analysis is not within the scope of the AST implementation and does not require NRC review for that purpose. The staff agrees with the licensee and did not review the gas decay tank rupture event since the staff does not consider the gas decay tank rupture to be a design-basis event, nor is it listed or addressed in RG 1.183 or in SRP 15.0.1 as a DBA.

#### 3.2.1 Loss-of-Coolant Accident (LOCA)

The current radiological consequence analysis for the postulated LOCA is based on the accident source term described in TID-14844 and it is provided in Ginna UFSAR Section 15.6.4.2, "Major Reactor Coolant System Pipe Rupture (Loss-of-Coolant Accident)." To demonstrate that the engineered safety features (ESFs) designed to mitigate the radiological consequences at Ginna will remain adequate after implementing the TS changes requested in this LAR, the licensee re-analyzed the offsite and control room radiological consequences of

the postulated LOCA. The licensee has implemented an AST in this re-analysis.

The licensee calculated and submitted the results of its offsite and control room doses and provided the major assumptions and parameters used in its dose calculations. As documented in its submittals, the licensee has determined that after implementation of the changes requested in this license amendment with use of an AST and installation of the modified CREATS, the existing ESF systems at Ginna will still provide reasonable assurance that the radiological consequences of the postulated LOCA at the EAB, in the LPZ, and in the CR will meet the acceptable radiation dose criteria specified in 10 CFR 50.67(b)(2). As part of the implementation of the AST, the TEDE acceptance criterion of 10 CFR 50.67(b)(2) replaces the previous whole-body and thyroid dose guidelines of 10 CFR 100.11 and GDC 19.

The staff has reviewed the licensee's analyses and has performed independent confirmatory radiological consequence dose calculations for the following two potential fission product release pathways which were previously analyzed and accepted by the staff as design basis leakage pathways in the UFSAR, Chapter 15, "Accident Analyses":

(1) primary containment leakage and
(2) leakage from ECCSs outside containment.

Any potential back leakage into the isolated reactor water storage tank (RWST) during the containment sump water recirculation following a LOCA is included in the leakage pathway from the ECCS outside containment in (2) above.

3.2.1.1 Containment Leakage

The current Ginna design basis containment leak rate specified in the UFSAR is 0.2 percent by volume per day (% per day). For the radiological consequence analysis, this rate is followed by 0.1% per day after 24 hours following a LOCA for the duration of the accident (30 days) consistent with the guideline provided in RG 1.183. The licensee has not proposed to change the design basis containment leak rate.

The fission products in the containment atmosphere following the postulated LOCA at Ginna are mitigated by three processes: (1) natural deposition of fission products in aerosol form, (2) the Containment Spray System (CSS), and (3) by the Containment Recirculation Cooling and Filtration System (CRCFS). The licensee has conservatively neglected natural deposition of fission products in aerosol form in the containment and excluded any credit for the removal of fission products in aerosol form by natural deposition in the radiological consequence re-analyses. The radiological consequence analyses performed by the licensee showed that Ginna would still meet the relevant dose criteria specified in 10 CFR 50.67 without any credit for removing fission products by natural deposition processes in the containment.

The CSS is an ESF system. In conjunction with the CRCFS, it is designed to provide containment cooling and fission product removal in the containment following the postulated LOCA. The CSS consists of two trains. Each train consists of a pump, two spray headers, and associated valves. The CSS is independently capable of delivering 1,300 gpm of borated water from the RWST into 78% of the containment atmosphere. The spray pumps are automatically started whenever the coincidence of two sets of two-out-of-three high containment pressure signals occurs. The licensee assumed that one out of two spray pump starts taking suction

initially from the RWST and initiates building spray through two spray headers until the water in the RWST reaches a pre-set low level at 52 minutes after the accident. The licensee assumed that spray flow is initiated within 80 seconds from the initiation of the postulated LOCA. These CSS design and operational features are not affected by the use of the AST and they were previously accepted by the staff as design bases in the Ginna UFSAR Section 6.2.2.2, "Containment Spray System."

After 52 minutes into the accident, the spray pump suction is transferred manually to the containment sump and the spray water from the containment sump is recirculated until the spray operation is terminated at 30 days. The licensee conservatively did not take any credit for iodine removal by the containment spray during the recirculation phase of containment spray operation. The licensee used the models and guidance provided in RG 1.183 to determine the removal rates (a factor of 20 per hour for elemental iodine and a factor of 3.5 per hour for iodine in particulate form) by the CSS. The staff finds these removal rates acceptable, because they are consistent with the guidance provided in RG 1.183. The major parameters and assumptions used by the licensee, including the spray removal rates, are listed in Table 4.

The CRCFS is designed to remove heat at the design basis rate from the containment atmosphere by depressurizing the containment and to remove fission products following a LOCA. The CRCFS consists of four units, each including, among others, charcoal and high efficiency particulate air (HEPA) filters. Two of the four units are required during the post-accident period. Each unit has 30,000 cfm flow capacity. During normal plant operation, the charcoal filters are by-passed. In the event of a LOCA, the air flow would be directed through the charcoal filters. However, in this LAR, the licensee requested to delete the TS requirement for the containment post-accident charcoal filters and the associated SR. The containment post-accident charcoal filters will not be credited for evaluating potential radiological consequences at the EAB, LPZ, and control room. The revised radiological analyses of the DBAs are performed without taking credit for the containment post-accident charcoal filters and the results show that the EAB, LPZ, and control room doses remain below the guidance provided in RG 1.183. Therefore, the staff finds this request acceptable.

3.2.1.2 Post-LOCA Leakage from Engineered Safety Features Outside Containment

During the initial phases of a LOCA, safety injection and CSS draw borated water from the RWST. At 52 minutes after the start of the event, these systems start to draw water from the containment sump instead. This recirculation flow causes contaminated sump water to be circulated through piping and components outside of the containment where small amounts of system leakage could provide a path for the release of radionuclides to the environment. The licensee conservatively assumed that the leakage rate is two times the expected value (4 gallons per hour) consistent with the guidance provided in RG 1.183. This leakage assumption includes any potential back leakage into the RWST during the containment sump water recirculation following a LOCA

The licensee conservatively assumed that all of the radioiodines released from the fuel are instantaneously moved to the containment sump water and noble gases are assumed to remain in the containment atmosphere. Consistent with the guidelines provided in RG 1.183, the licensee assumed that (1) since the containment sump pH is maintained greater than 7, the radioiodine in the sump solution of nonvolatile iodide or iodate form and, as such, the chemical form of radioiodine in the sump water at the time of recirculation would be 95% aerosol as

cesium iodide, 4.85% elemental and 0.15% organic, and (2) the total iodine in leaked fluid is assumed to become airborne and leak to the environment, via the back-draft damper's louver on the North wall of the auxiliary building, for 30 days after the start of recirculation. This release point has the most conservative atmospheric dispersion factor for the control room.

In Section 15.6.4.1 of the Ginna UFSAR, it is stated that 4% of the radioiodines contained in the ECCS leakage would be released to the auxiliary building atmosphere. The staff previously accepted this value as a Ginna design basis. Subsequent to the staff's approval of this value, the staff issued Appendix A to RG 1.183 stating that the amount of iodine that becomes airborne should be assumed to be 10% of the total iodine activity in the leaked fluid, unless a smaller amount can be justified based on the actual sump pH history and ventilation rate. The licensee calculated and proposed the amount of iodine that becomes airborne to range from 7% at the start of the containment sump water recirculation, gradually decreasing to 1%, as a function of time for the duration of 30-day accident period. For this estimate, the licensee used a constant enthalpy equation based on actual Ginna specific sump water temperature and pressure circulating outside the containment following a LOCA. The staff independently confirmed the licensee's calculation. Based on the discussion below, the staff concluded that a range of 7 to 2% for the amount of iodine that becomes airborne is acceptable, as opposed to the range of 7 to 1% proposed by the licensee. The staff believes that a range of 7 to 2% is more representative of actual iodine behavior and transport at Ginna following the LOCA than a constant 4% value previously accepted by the staff as a design basis.

The staff's acceptance of the amount of iodine that becomes airborne is based on the licensee's constant enthalpy calculation based on Ginna sump water temperature and pressure, the actual Ginna sump water pH history ranging from 7.9 to 9.7 (See Section 3.4, "Containment Sump Water Chemistry" of this safety evaluation (SE)), and the auxiliary building ventilation system (ABVS) design. The Ginna ABVS consists of, among others, a single 100% capacity bank of HEPA filters, a single charcoal filter bank, and redundant 100% capacity fans discharging to the environment through the plant vent. On the receipt of a high radiation alarm, the auxiliary building air supply fans and all exhaust fans are tripped except those exhausting to the vent through the charcoal filters. The charcoal filters are tested in accordance with the Ginna TS Section 5.5.10, "Ventilation Filter Testing Program," for iodine removal efficiencies of 90% and 70% for elemental iodine and organic iodine, respectively. The licensee did not take any credit for the removal of iodine through the ABVS filters because the HEPA and charcoal filter units are single train.

In concluding that a range of 7 to 2% is acceptable, the NRC staff considered the possible uncertainties associated with the following inputs: (1) potential availability of the HEPA and charcoal filters provided in the ABVS during and following a LOCA, (2) actual sump water pH history, (3) the ABVS design for dilution and holdup of iodine that becomes airborne, and (4) the staff's previous acceptance of the amount of iodine that becomes airborne (4%) as a design basis. The major parameters and assumptions used by the licensee are listed in Table 4.

#### 3.2.1.3 Radiological Consequence of Loss-of-Coolant Accident

The licensee reevaluated the radiological consequences resulting from the postulated LOCA using the AST and concluded that the radiological consequences at the EAB, LPZ, and control

room are within the dose criteria specified in 10 CFR 50.67. The results of the licensee's radiological consequence calculation are provided in Table 1 and the major parameters and assumptions used by the licensee and found acceptable by the staff are listed in Table 4.

The licensee assumed a control room isolation delay of 60 seconds to account for damper positioning and instrumentation delays and an additional 10 seconds for the new CREATS to be operational following a LOCA. Following isolation, there will be no outside air makeup and the filtered recirculation flow of 5400 cfm (6000 cfm nominal less 10%) is initiated. The licensee assumed an unfiltered inleakage rate of 300 cfm and recirculation filter efficiencies of 90%, 70%, and 98% for elemental, organic, and particulate, respectively (see Section 3.10, "Control Room Habitability" of this SE).

The radiological consequences of the LOCA at the EAB and at the LPZ calculated by the licensee are within the dose criteria specified in 10 CFR 50.67 and the control room dose is within the limit established by GDC 19. The radiological consequences of the LOCA at the EAB and at the LPZ calculated by the licensee are within the dose criteria specified in 10 CFR 50.67 and the control room dose is within the limit established by GDC 19. The staff performed independent radiological consequence dose calculations for the EAB, LPZ and control room using NRC computer code, "HABIT, Version 1.1," and confirmed the licensee's conclusions.

### 3.2.2 Fuel-Handling Accident (FHA)

The FHA assumes the dropping of a spent fuel assembly during refueling. This event could occur inside the containment or in the fuel storage building. The affected assembly is assumed to be that with the highest inventory of radionuclides of the fuel assemblies in the core. All of the fuel rods in the assembly are conservatively assumed to rupture. Volatile constituents of the core fission product inventory migrate from the fuel pellets to the gap between the pellets and the fuel rod clad. The radionuclide inventory in the fuel rod gap of the damaged fuel rods is assumed to be instantaneously released. Fission products released from the damaged fuel are decontaminated by passage through the overlaying water in the reactor cavity or spent fuel pool depending on their physical and chemical form. Appendix B of RG 1.183 identifies acceptable radiological analysis assumptions for an FHA.

The licensee assumed no decontamination for noble gases, an effective decontamination factor of 200 for radioiodines, and retention of all aerosol and particulate radionuclides within the spent fuel pool water. The licensee assumed that 100% of the radionuclides released from the reactor cavity are released to the environment in 2 hours without any credit for filtration, holdup, or dilution. For an FHA in the spent fuel pool, the licensee assumed iodine removal by the ABVS charcoal filters (90% for elemental iodine and 70% for organic iodine). All of above assumptions are consistent with the guidance provided in RG 1.183 with no exception. The Ginna TSs require operation of the ABVS during irradiated fuel movement within the auxiliary building when one or more fuel assemblies in the auxiliary building has decayed less than 60 days since being irradiated. The charcoal filters are tested in accordance with the Ginna TS Section 5.5.10, "Ventilation Filter Testing Program."

A decay time of 100 hours prior to moving irradiated fuel was assumed for both FHAs in the containment and in the spent fuel pool. To ensure that the analysis would be bounding for both release cases, the licensee did the analysis using the atmospheric dispersion factors for the most limiting combination of release point and receptor (See Section 3.1, "Atmospheric Dispersion Estimates" of this SE).

The licensee assumed a control room isolation delay of 60 seconds to account for damper positioning and instrumentation delays and an additional 10 seconds for the new CREATS to be operational following an FHA. Following isolation, there will be no outside air makeup and the filtered recirculation flow of 5400 cfm (6000 cfm nominal less 10%) is initiated. The licensee assumed an unfiltered inleakage rate of 300 cfm and recirculation filter efficiencies of 90%, 70%, and 98% for elemental, organic, and particulate, respectively (see Section 3.10, "Control Room Habitability" of this SE).

The NRC staff found that the licensee used analysis assumptions and inputs consistent with applicable regulatory guidance identified in Section 2.0 of this SE and with those stated in the Ginna UFSAR as design bases. The assumptions found acceptable to the staff are presented in Table 5. The EAB, LPZ, and control room doses estimated by the licensee for the FHA were found to meet the applicable accident dose criteria and are therefore acceptable. The radiological consequences of the LOCA at the EAB and at the LPZ calculated by the licensee are within the dose criteria specified in 10 CFR 50.67 and the control room dose is within the limit established by GDC 19. The staff did independent radiological consequence dose calculations for the EAB, LPZ and control room using NRC computer code, "HABIT, Version 1.1," and confirmed the licensee's conclusions.

3.2.3 Main Steamline Break (MSLB)

The MSLB accident considered is the complete severance of the 36-inch main steam header outside containment inside the turbine building. This is the largest MSLB outside containment. The radiological consequences of a break outside containment will bound the consequences of a break inside containment. Thus, only the MSLB outside of containment is considered with regard to the radiological consequences. The single failure is assumed to be a failure of the main steam isolation valve on the faulted steam generator. The faulted steam generator will rapidly depressurize and release the initial contents of the steam generator to the environment. A reactor trip occurs, main steam isolation occurs, safety injection actuates, and a loss of offsite power (LOOP) occurs concurrently with the reactor trip. As this LOOP renders the main condenser unavailable, the plant is cooled down by releases of steam to the environment. The MSLB accident is described in the Ginna UFSAR Section 15.1.5, "Spectrum of Steam System Piping Failure Inside and Outside of Containment." Appendix E of RG 1.183 identifies acceptable radiological analysis assumptions for an MSLB.

The licensee stated that no fuel damage is postulated to occur as the result of an MSLB. Two radioiodine spiking cases are considered. The first assumes that a pre-incident radioiodine spike occurred just before the event and the reactor coolant system (RCS) radioiodine inventory is at the maximum value (for 100 percent power) permitted by TSs. The second case assumes the event initiates a co-incident radioiodine spike. Radioiodine is released from the fuel to the RCS at a rate 500 times the normal radioiodine appearance rate for 8 hours. At approximately 10 minutes, the faulted steam generator is isolated by operator action. The intact steam generator is then used for cooldown, where steam is released to the atmosphere through the intact steam generator ARV. The licensee assumed that the faulted steam generator boils dry within 10 minutes, releasing the entire liquid inventory and entrained radionuclides through the faulted steam line to the environment.

Leakage from the RCS to the steam generators is assumed to be the maximum value permitted

by TSs. Primary-to-secondary leakage is assumed to be 1 gpm each to the faulted and intact steam generators.

The leakage to the ruptured steam generator is assumed immediately to flash to steam and be released to the environment without holdup or dilution. The leakage in the unaffected steam generator mixes with the bulk water and is released at the assumed steaming rate. This steaming from the unaffected steam generator is assumed to continue for 8 hours. The licensee determined that the tubes in the unaffected steam generator would remain covered by the bulk water. The licensee assumed that the radionuclide concentration in the steam generator is partitioned such that 1 percent of the radionuclides in the bulk water of the unaffected steam generator enters the vapor space and is released to the environment. No partitioning is assumed in the faulted steam generator.

The licensee assumed a control room isolation delay of 60 seconds to account for damper positioning and instrumentation delays and an additional 10 seconds for the new CREATS to be operational following an MSLB accident. Following isolation, there will be no outside air makeup and the filtered recirculation flow of 5400 cfm (6000 cfm nominal less 10%) is initiated. The licensee assumed an unfiltered inleakage rate of 300 cfm and recirculation filter efficiencies of 90%, 70%, and 98% for elemental, organic, and particulate, respectively (see Section 3.10, “Control Room Habitability” of this SE).

The NRC staff found that the licensee used analysis assumptions and inputs consistent with applicable regulatory guidance identified in Section 2.0 of this SE and with those stated in the Ginna UFSAR as design bases. The assumptions found acceptable to the staff are presented in Table 6. The EAB, LPZ, and CR doses estimated by the licensee for the MSLB were found to meet the applicable accident dose criteria and are therefore acceptable. The radiological consequences of the LOCA at the EAB and at the LPZ calculated by the licensee are within the dose criteria specified in 10 CFR 50.67 and the control room dose is within the limit established by GDC 19. The staff did independent radiological consequence dose calculations for the EAB, LPZ and control room using NRC computer code, “HABIT, Version 1.1,” and confirmed the licensee’s conclusions.

### 3.2.4 Steam Generator Tube Rupture (SGTR)

The accident considered is the complete severance of a single tube in one of the steam generators resulting in the transfer of RCS water to the ruptured steam generator. The primary-to-secondary break flow through the ruptured tube following an SGTR results in radioactive contamination of the secondary system. A reactor trip occurs, safety injection actuates, and a LOOP occurs concurrently with the reactor trip. As this LOOP renders the main condenser unavailable, the plant is cooled down by releases of steam to the environment. The licensee determined that the most limiting single failure is a single ARV on the intact steam generator to fail open providing a continuous release path. The failed ARV is assumed to be closed by manual operator action within 25 minutes after failing open. Appendix F of RG 1.183 identifies acceptable radiological analysis assumptions for an SGTR.

Two radioiodine spiking cases are considered. The first assumes that a pre-incident radioiodine spike occurred just before the event and the RCS radioiodine inventory is at the maximum value (for 100 percent power) permitted by TSs. The second case assumes the event initiates a coincident radioiodine spike. Radioiodine is released from the fuel to the RCS

at a rate 335 times the normal radioiodine appearance rate for 8 hours. The licensee assumed that a portion of the break flow flashes to vapor, rises through the bulk water, enters the steam space, and is immediately released to the environment with no mitigation or holdup. The flashing fraction ranges from 0 to 0.15 averaging approximately 0.07. The portion of the break flow that does not flash is assumed to mix with the bulk water of the steam generator. In addition to the break flow, the licensee assumed there is primary-to-secondary leakage at the maximum value permitted by TSs. Primary-to-secondary leakage is assumed to be 150 gpd into the bulk water of the ruptured steam generator and 300 gpd total into the bulk water of the unaffected steam generator.

The radionuclides in the bulk water are assumed to become vapor at a rate that is a function of the steaming rate for the steam generators and the partition coefficient. The licensee determined that tubes in the unaffected steam generator would remain covered by the bulk water. The licensee assumed that the radionuclide concentration in the steam generator is partitioned such that 1% of the radionuclides in the unaffected steam generator bulk water enter the vapor space and are released to the environment. The partition coefficient does not apply to the flashed break flow. The steam release from the ruptured and unaffected steam generators continues until the residual heat removal (RHR) system can be used to complete the cooldown at approximately 8 hours.

The licensee assumed a control room isolation delay of 60 seconds to account for damper positioning and instrumentation delays and an additional 10 seconds for the new CREATS to be operational following an SGTR accident. Following isolation, there will be no outside air makeup and the filtered recirculation flow of 5400 cfm (6000 cfm nominal less 10%) is initiated. The licensee assumed an unfiltered inleakage rate of 300 cfm and recirculation filter efficiencies of 90%, 70%, and 98% for elemental, organic, and particulate, respectively (see Section 3.10, “Control Room Habitability” of this SE).

The NRC staff found that the licensee used analysis assumptions and inputs consistent with applicable regulatory guidance identified in Section 2.0 of this SE and with those stated in the Ginna UFSAR as design bases. The assumptions found acceptable to the staff are presented in Table 7. The EAB, LPZ, and control room doses estimated by the licensee for the SGTR were found to meet the applicable accident dose criteria and are therefore acceptable. The radiological consequences of the LOCA at the EAB and at the LPZ calculated by the licensee are within the dose criteria specified in 10 CFR 50.67 and the control room dose is within the limit established by GDC 19. The staff did independent radiological consequence dose calculations for the EAB, LPZ and control room using NRC computer code, “HABIT, Version 1.1,” and confirmed the licensee’s conclusions.

### 3.2.5 Primary Coolant Pump Locked Rotor Accident (LRA)

The accident considered is the instantaneous seizure of a reactor coolant pump rotor (i.e., a locked rotor accident) which causes a rapid reduction in the flow through the affected RCS loop. A reactor trip occurs, safety injection actuates, and a LOOP occurs concurrently with the reactor trip. The flow imbalance creates localized temperature and pressure changes in the core. If severe enough, these differences may lead to localized boiling and fuel damage. As the LOOP renders the main condenser unavailable, the plant is cooled down by releases of

steam to the environment. Appendix G of RG 1.183 identifies acceptable radiological analysis assumptions for an LRA.

The licensee conservatively assumed that 50% of the fuel rods will experience departure from nucleate boiling (DNB) and are therefore assumed to release their gap activity into the RCS. A radial peaking factor of 1.65 was applied. The radionuclides released from the fuel are assumed to be instantaneously and homogeneously mixed in the RCS and transported to the secondary side via primary-to-secondary leakage of 500 gpd for each steam generator for 8 hours. The licensee assumed that this leakage mixes with the bulk water of the steam generators and that the radionuclides in the bulk water become vapor at a rate that is a function of the steaming rate for the steam generators and the partition coefficient. The tubes in the unaffected steam generator would remain covered by the bulk water. The licensee assumed that the radionuclide concentration in the steam generator is partitioned such that 1% of the radionuclides in the bulk water of the unaffected steam generator enter the vapor space and are released to the environment. The steam releases from the steam generators continue until the RHR system can be used to complete the cooldown at approximately 8 hours.

The licensee assumed a control room isolation delay of 60 seconds to account for damper positioning and instrumentation delays and an additional 10 seconds for the new CREATS to be operational following a LRA. Following isolation, there will be no outside air makeup and the filtered recirculation flow of 5400 cfm (6000 cfm nominal less 10%) is initiated. The licensee assumed an unfiltered inleakage rate of 300 cfm and recirculation filter efficiencies of 90%, 70%, and 98% for elemental, organic, and particulate, respectively (see Section 3.10, “Control Room Habitability” of this SE).

The staff found that the licensee used analysis assumptions and inputs consistent with applicable regulatory guidance identified in Section 2.0 of this SE and with those stated in the Ginna UFSAR as design bases. The assumptions found acceptable to the staff are presented in Table 8. The EAB, LPZ, and control room doses estimated by the licensee for the LRA were found to meet the applicable accident dose criteria and are therefore acceptable. The radiological consequences of the LOCA at the EAB and at the LPZ calculated by the licensee are within the dose criteria specified in 10 CFR 50.67 and the control room dose is within the limit established by GDC 19. The staff performed independent radiological consequence dose calculations for the EAB, LPZ and control room using NRC computer code, “HABIT, Version 1.1,” and confirmed the licensee’s conclusions.

### 3.2.6 Rod Ejection Accident (REA)

The accident considered is the mechanical failure of a control rod drive mechanism pressure housing that results in the ejection of a rod cluster control assembly and drive shaft. Localized damage to fuel cladding and a limited amount of fuel melt are projected due to the reactivity spike. This failure breeches the reactor pressure vessel head resulting in a LOCA to the containment. A reactor trip occurs, safety injection actuates, and a LOOP occurs concurrently with the reactor trip. As this LOOP renders the main condenser unavailable, the plant is cooled down by releases of steam to the environment. The release to the environment is assumed to

occur through two separate pathways:

- Release of containment atmosphere (i.e., design leakage)
- Release of RCS inventory via primary-to-secondary leakage through steam generators.

While the actual doses from an REA would be a composite of the two pathways, an acceptable dose from each pathway, modeled as if it were the only pathway, would show that the composite dose would also be acceptable. Appendix H of RG 1.183 identifies acceptable radiological analysis assumptions for an REA.

The licensee assumed that 10% of the fuel rods fail, releasing the radionuclide inventory in the fuel rod gap. The design basis REA in Section 15.4.5.3.5 of the Ginna UFSAR stated that less than 10% of the fuel rod enters DNB based on a detailed Ginna specific three-dimensional THINC analysis. The licensee further assumed that 10% of the core inventory of radioiodines and noble gases is in the fuel rod gap. A radial peaking factor of 1.75 was applied. In addition, localized heating is assumed to cause 0.375% of the fuel to melt, releasing 100% of the noble gases and 25% of the radioiodines contained in the melted fuel to the containment. For the secondary release case, 100% of the noble gases and 50% of the radioiodines contained in the melted fuel are released to the secondary.

For the containment leakage case, the radionuclides released from the fuel are assumed to be instantaneously and homogeneously mixed in the containment free volume. The licensee assumed that the containment leaks at the TS value of 0.2% volume per day for the first 24 hours and 0.1% volume per day for days 2 through 30 consistent with the guideline provided in RG 1.183. The licensee has taken credit for removal of iodine in particulate form by HEPA in the containment recirculation and filtration system (CRFS) but not iodine in elemental and organic forms. The CRFS is a safety-related system and its operational requirements are specified in the Ginna TSs. The licensee requested to remove charcoal filters from the CRFS in this license amendment.

The licensee does not credit containment spray operation as a radionuclide removal mechanism. However, the licensee does assumes that natural deposition processes result in a removal of aerosols at a rate of 0.023 $hr^{-1}$ based on the methodology of NUREG/CR-6189, “A Simplified Model of Aerosol Removal by Natural Processes in Reactor Containments.” The staff compared the 0.023 $hr^{-1}$ removal rate proposed by the licensee against the data in Table 2.2.2.1-3 of NUREG/CR-6604, “RADTRAD: A Simplified Model for Radionuclide Transport and Removal and Dose Estimation,” and determined it to be more conservative. As such, the staff finds the value of 0.023 $hr^{-1}$ to be acceptable.

For the secondary release case, the radionuclides released from the fuel are assumed to be instantaneously and homogeneously mixed in the RCS and transported to the secondary side via primary-to-secondary leakage at 500 gpd for each steam generator for 8 hours which bounds the current TS value of 144 gpd. The licensee assumed that this leakage mixes with the bulk water of the steam generators and that the radionuclides in the bulk water become vapor at a rate that is a function of the steaming rate for the steam generators and the partition coefficient. The licensee conservatively assumed that the chemical form of the radioiodine released to the environment would be 97% elemental and 3% organic consistent with the guideline provided in RG 1.183. The licensee assumed that the aerosol and iodine radionuclide concentration in the steam generator is partitioned such that the 1% of the radionuclides that

enter the unaffected steam generator from the RCS enter the vapor space and are released to the environment. This assumption is also consistent with the guideline provided in RG 1.183. The steam releases from the steam generators continue until the RHR system can be used to complete the cooldown at approximately 8 hours.

The licensee assumed a control room isolation delay of 60 seconds to account for damper positioning and instrumentation delays and an additional 10 seconds for the new CREATS to be operational following an REA. Following isolation, there will be no outside air makeup and the filtered recirculation flow of 5400 cfm (6000 cfm nominal less 10%) is initiated. The licensee assumed an unfiltered inleakage rate of 300 cfm and recirculation filter efficiencies of 90%, 70%, and 98% for elemental, organic, and particulate, respectively (see Section 3.10, "Control Room Habitability" of this SE).

The NRC staff found that the licensee used analysis assumptions and inputs consistent with applicable regulatory guidance identified in Section 2.0 of this SE and with those stated in the Ginna UFSAR as design bases. The assumptions found acceptable to the staff are presented in Table 9. The EAB, LPZ, and control room doses estimated by the licensee for both cases of the REA were found to meet the applicable accident dose criteria and are therefore acceptable. The radiological consequences of the LOCA at the EAB and at the LPZ calculated by the licensee are within the dose criteria specified in 10 CFR 50.67 and the control room dose is within the limit established by GDC 19. The staff performed independent radiological consequence dose calculations for the EAB, LPZ and control room using NRC computer code, "HABIT, Version 1.1," and confirmed the licensee's conclusions.

### 3.2.7 Tornado Missile in Spent Fuel Pool Accident (TMA)

The NRC staff does not consider this event as a DBA and it is neither listed nor addressed in RG 1.183 or in SRP 15.0.1. Nevertheless, the licensee performed a radiological evaluation of this event because it was previously analyzed in Section 9.1, "Fuel Storage and Handling," of the Ginna UFSAR.

The licensee analyzed this event to determine the radiological consequence resulting from damage to stored spent fuel from the impact of a tornado missile. The licensee assumed that the hypothetical tornado missile propelled by the wind penetrates the auxiliary building roof and impacts 9 fuel assemblies in the spent fuel storage pool (5 fuel assemblies decayed for 100 hours and 4 fuel assemblies decayed for 60 days). In its analysis, the licensee further assumed the tornado missile to be a 1490 pound wooden pole, 35 feet in length and 13.5 inches in diameter. These assumptions are the current design bases as documented in the Ginna UFSAR Section 9.1. All other assumptions and parameters used in the radiological consequence analysis of this event are the same as those used in the FHA above.

The licensee assumed a control room isolation delay of 60 seconds to account for damper positioning and instrumentation delays and an additional 10 seconds for the new CREATS to be operational following a TMA. Following isolation, there will be no outside air makeup and no filtered air recirculation flow. The licensee assumed an unfiltered inleakage rate of 300 cfm.

The staff found that the licensee used analysis assumptions and inputs consistent with those used for the FHA, which was found acceptable by the staff in Section 3.2.2 of this SE. The assumptions found acceptable to the staff are presented in Table 10. The EAB, LPZ, and control room doses estimated by the licensee for this event were found to meet the FHA dose

criteria and are therefore acceptable. The radiological consequences of the LOCA at the EAB and at the LPZ calculated by the licensee are within the dose criteria specified in 10 CFR 50.67 and the control room dose is within the limit established by GDC 19. The staff performed independent radiological consequence dose calculations for the EAB, LPZ and control room using NRC computer code, "HABIT, Version 1.1," and confirmed the licensee's conclusions.

### 3.2.8 AST Summary

The NRC staff finds that the licensee used analysis methods and assumptions consistent with the conservative regulatory requirements and guidance identified in Section 2.0 above in its re-analysis of the affected DBAs. The staff compared the doses estimated by the licensee to the applicable criteria identified in RG 1.183. The staff finds, with reasonable assurance, that the licensee's estimates of the EAB, LPZ, and control room doses will comply with these criteria. The staff finds reasonable assurance that the Ginna design bases, as modified by this license amendment, will continue to provide sufficient safety margins with adequate defense-in-depth to address unanticipated events and to compensate for uncertainties in accident progression and analysis assumptions and parameters.

This licensing action is considered a full implementation of the AST. With this approval, the previous accident source term in the Ginna design basis is superseded by the AST proposed by the licensee. The previous offsite and control room accident dose criteria expressed in terms of whole body, thyroid, and skin doses are superseded by the TEDE criteria of 10 CFR 50.67 or fractions thereof, as defined in Regulatory Position 4.4 of RG 1.183. All future radiological accident analyses performed to show compliance with regulatory requirements shall address all characteristics of the AST and the TEDE criteria as defined the the Ginna design bases, as modified by the present amendment.

## 3.3 Containment Sump Water Chemistry

The licensee proposed to revise the TS SR to reflect the actual amount of sodium hydroxide (NaOH) required to meet the pH requirement, consistent with the new analysis. The licensee uses NAOH solution to control and maintain alkaline pH (above 7) in the containment spray water during the injection phase and in the containment sump water during the ECCS recirculation phase. The volume of sodium hydroxide solution and the concentration of sodium hydroxide are specified in the Ginna TS Surveillance Requirements 3.6.6.7 and 3.6.6.8, respectively. In its amendment request, the licensee proposed to decrease the volume of the solution from 4500 gallons to 3000 gallons and specified an upper limit of less than or equal to 35 weight percent (wt %) for the sodium hydroxide concentration in the spray additive tank.

The licensee performed an analysis which indicated that the change in the volume and concentration of sodium hydroxide will not produce unacceptable values for pH in the spray and containment sump solutions. The original CSS design for Ginna required that the pH of the sprayed liquid be maintained in the range of 9.0 to 10.0. The analysis performed by the licensee has indicated that these values could be achieved by mixing 30 wt % to 32 wt % sodium hydroxide solution with borated water from the RWST having boron concentration in the range of 2300 to 2600 ppm. Use of the maximum concentration of sodium hydroxide of 35 wt % instead of 32 wt % will produce slightly higher value for pH, which will still provide an acceptable operational margin. The minimum value pH of 7.0 for the containment sump water was specified in the SRP. The licensee demonstrated that, when a volume of between 1048 and 2977 gallons of the 30 wt % to 35 wt % sodium hydroxide solution from the spray additive

tank is mixed with the borated water accumulated in the containment sump, the resulting pH will be in the range of 7.9 to 9.7. This exceeds the minimum pH specified in the SRP. Therefore, the staff concludes that the proposed changes by the licensee are acceptable.

3.4 Control Room Toxic Gas Analysis

GDC 19, "Control Room," requires that a control room be provided from which actions can be taken to operate the plant safely under normal conditions and to maintain it in a safe condition under accident conditions. Releases of hazardous chemicals can result in the control room becoming uninhabitable. RG 1.78, "Assumptions for Evaluating the Habitability of a Nuclear Power Plant Control Room during a Postulated Hazardous Chemical Release" provides guidance in assessing the habitability of the control room during and after a postulated external release of chemicals. Consistent with RG 1.78, the licensee performed an analysis to evaluate the impact of the new CREATS on toxins that could threaten CR habitability, and calculated the worst case substance concentrations in the CR assuming an in-leakage of 300 cfm, consistent with the designed in-leakage rate of the new CREATS.

RG 1.78 provides that the concentration of toxic substances introduced into the control room not exceed their toxic limits for at least 2 minutes after being detected. The 2-minute requirement was justified by the time needed for the control room operators to don their respirators and protective clothing. The toxic limits for the most common chemicals are specified in RG 1.78 and NUREG/CR-6624.

In its analyses, the licensee considered toxic chemicals which, under certain conditions, could produce a safety hazard. The following chemicals were evaluated: ammonia (in ammonium hydroxide solution), chlorine, sodium hypochlorite, Halon, Refrigerant R-22, and carbon dioxide.

The licensee has determined that the concentrations of Halon and Refrigerant R-22 in the control room, after their accidental release, are very low because they become diluted by atmospheric air before reaching the control room intake. In addition, their toxic limits are high; between 20,000 and 50,000 ppm for Halon and between 3000 and 5000 ppm for R-22. They could not, therefore, endanger the habitability of the control room. Carbon dioxide also does not pose any significant safety problem because its toxic limit is high (40,000 ppm) and because the only source of carbon dioxide in the control room is the occupants' respiration, requiring a very long time to produce enough carbon dioxide to reach its toxic limit. However, the licensee chose to address this safety issue by assuming that in some cases, the control room may stay isolated from the outside for an extended period of time and the internal buildup of carbon dioxide could eventually reach high enough levels to be of a concern. The evaluation by the licensee has indicated that in the Ginna control room, no such buildup of carbon dioxide could occur because the inleakage of the outside air would preclude buildup of carbon dioxide to an unacceptable level.

The plant has a 2500 gallon tank containing a 16% sodium hypochlorite solution. The tank is located 310 ft from the control room. It is situated below the control room intake and is protected by a 16 ft concrete dike. The licensee calculated, using the HABIT computer code, that under the worst-accident conditions the concentration of sodium hypochlorite outside of the control room will be less than 0.0004 $mg/m^3$. This value is much too low to cause any safety concerns even if some of the chemical leaks into the control room.

The two chemicals which, because of their quantities stored and their chemical nature, could

pose significant hazards are: ammonia and chlorine. The licensee has determined their effects on control room habitability using the HABIT code. The projected buildup of ammonia in the control room was higher than that for chlorine, mainly because of the proximity of the storage tank containing ammonium hydroxide solution to the control room inlet. However, the analyses for both ammonia and chlorine showed that the 2-minute criterion was met for both these chemicals.

The staff reviewed the licensee's analyses and assumptions. The staff confirmed that the assumptions are consistent with the guidance in RG 1.78 and the methodology is consistent with that specified in NUREG/CR-6624. The concentrations of toxic gases are within the limits specified in RG 1.78. Therefore, the staff concludes that the licensee has demonstrated that chemicals released inside the plant or in its proximity will pose no hazard to the control room operators.

3.5 Seismic Analysis

The staff reviewed Rochester Gas & Electric (RG&E) Specification ME-326, Revision 1, dated July 30, 2003. This specification defined the technical requirements for the CREATS and CRECS to include; design, fabrication, shop testing, documentation, delivery, installation, startup, and field testing. The licensee stated, in Section 4.0, Design Requirements that, unless noted otherwise, all new components installed by this modification shall be safety related and shall be Seismic Category I, per NRC RG 1.29, and that 10 CFR Part 50, Appendix B Quality Assurance requirements shall be applied. Loads and load combinations are listed in Section 12.0 of ME-326, where structural load criteria, and loading combinations and stress limits for piping and duct supports were extracted from Attachment A of the Design Criteria, PCR 2000-0024, Revision 0.

The licensee provided the following information:

a. A list of mechanical and electrical instrumentation and control equipment installed in the modified CREATS and CRECS systems requiring seismic qualification.

b. The location (elevation in a building or structure) of the equipment and the source of the required response spectra, consistent with the plant licensing basis, to be used for seismic qualification of the equipment.

c. The seismic qualification method to be used for the equipment.

d. The results of seismic qualification of the equipment.

In its July 8, 2004, letter the licensee provided a list of equipment items with associated equipment location and seismic qualification method. The licensee indicated that the majority of equipment are to be qualified using IEEE Standard 344-1987 for testing and analysis criteria guidelines.

By letter dated December 3, 2004, the licensee clarified some equipment seismic qualification criteria and provided the status of the seismic qualification of all equipment listed for the CREATS and CRECS systems. Attachment 1 to the December 3, 2004, letter, indicates that the documentation of the analysis qualification for 5 equipment items are either in progress or in review. The licensee has stated that the remaining 5 reports will be completed prior to

implementation of the modification and is being tracked by the licensee as part of the CREATS modification program. The NRC staff finds that reasonable controls for the completion of these reports can be provided by the licensee's administrative processes.

Based on the staff's review of the licensee's response to the RAIs related to the seismic qualification of equipment installed in the modified CREATS and CRECS systems, the NRC staff finds that the licensee has completed the seismic qualification, including documentation, of a majority of equipment listed in Attachment 1 to the December 3, 2004, letter. The staff has concluded that the licensee is using acceptable methodology and criteria to seismically qualify the equipment in the new CREATS and CRECS systems. In addition, the staff finds that the licensee has adequate administrative controls to ensure that the results of the seismic qualification are properly documented prior to implementation of the new systems. The staff finds the information provided reasonable and acceptable.

In response to the staff's request regarding "Interconnecting Duct/Pipe Between Control Room and Relay Room Annex," the licensee provided an analysis summary for this outside piping in a letter dated April 22, 2004. The submittal addressed the following three areas of interest:

3.5.1 Impact of Relay Room Annex on the Control Building Response Spectra

The licensee stated that the floor response spectra of the Control Building and other site buildings are based on a STARDYNE finite element model produced under the Seismic Upgrade Program. The Control Building response spectra was developed using RGs 1.60 and 1.61, under the Ginna Station Seismic Upgrade Program, Auxiliary Structures Seismic Analysis, Addendum I, Additional Floor Response Spectra, dated March 12, 1981, by Gilbert Associates. In the late 1980's, the Control Building Relay Room East wall was modified as part of the Ginna Structural Upgrade Program. A separate enclosure (Relay Room Annex) was installed to address the Systematic Evaluation Program (SEP) topics of Severe Weather Phenomenon, Wind and Tornado, Tornado Missile and Flooding. As a result of adding the Relay Room Annex enclosure, ABS Consulting, Inc. was contracted to:

- Investigate the impact of Relay Room Annex modification on the floor response spectra of the Control Building.
- Develop floor response spectra for the Relay Room Annex.
- Develop seismic differential displacements between the Control Building and the Relay Room Annex.

The licensee further stated that the ABS Consulting developed calculation #1292405-C-001 using SAP2000, the STARDYNE inputs for the existing Control Building response spectra model, drawings of the Relay Room Annex, and masses of the equipment to be installed in the building. The licensee indicated that analyses of the Control Building, with and without the Relay Room Annex and with varying soil properties beneath the Annex, showed that the Relay Room Annex adds mass and stiffness to the Control Building and that the existing response spectra for the Control Building structure remains bounding, and also bounds the Relay Room Annex response spectra. Additionally, the calculation provides seismic differential

displacements that were used to analyze the duct/pipe between the Relay Room Annex and the Control Building.

### 3.5.2 Summary of Seismic Qualification for Duct/Pipe between the Relay Room Annex and the Control Building

The licensee stated that the supply and return air ducts running between the Relay Room Annex and the Control Building for the CREATS were seismically analyzed as piping in accordance with PCR 2000-0024, Design Criteria, Section 9.6. The computer program PS+CAEPIPE along with the seismic differential displacements from ABS Consulting calculation #1292405-C-001 were used to determine the piping loads and stresses for all piping runs. The piping was modeled as fixed at both anchor points and the analysis considered deadweight, thermal, Operating Basis Earthquake (OBE), and Safe Shutdown Earthquake (SSE). Friction loads were not considered since there are no movements that cause the pipe to slide over any member support. The outputs from the computer model were combined to determine the maximum ANSI B31.1 stress due to sustained loads, stress due to occasional loads, and thermal expansion stress range. The licensee concluded that the maximum piping stresses for both supply air pipe and return air pipe are all well below the defined allowable stresses.

The licensee further stated that the computer program PS+CAEPIPE also defined the loads and moments at the connections to the Relay Room Annex Roof and the Control Building East Wall. Analysis of the loads and moments applied to the building structure connections were performed. The connection at the Relay Room Annex Roof was designed using the return air duct loads since they were bounding relative to the supply air duct. A summary of design calculations for Relay Room Annex Roof Baseplate indicates that all design values for anchor bolt and baseplate are below allowable values.

The licensee stated that the connection at the Control Building East Wall was designed using the return air duct loads because they were bounding over that of the supply air duct. A summary of design calculations for Control Building East Wall Connection also indicates that design values for plate thickness, weld sizes, and armor plate moment are all below allowable values.

The NRC staff has reviewed the results of the licensee's seismic analysis for duct/pipe between the Relay Room Annex and the Control Building. The staff has concluded that the assumptions and loads used in the modeling are acceptable and the resulting stresses are within allowable limits. Therefore, the staff finds that the duct/pipe between the Relay Room Annex and the Control is seismically qualified.

### 3.5.3 Structural Qualification for Duct/Pipe between the Relay Room Annex and the Control Building

The licensee stated that the supply and return air ducts running between the Relay Room Annex and the Control Building for the CREATS were structurally analyzed in accordance with PCR 2000-0024, Design Criteria, Attachment A. The structural analysis included dead loads, live loads, normal wind loads, and design tornado loads. The licensee indicated that lateral earth pressure and buoyant force were not considered since they are not applicable to this

application. Additionally, snow loads were not considered since the majority of the piping is vertical and the horizontal piping will not allow for a significant buildup due to its geometry. The licensee stated that a conservative maximum wind speed of 188 mph was used in the analysis for wind loading. The two specific design basis tornado missiles considered per PCR 2000-0024, Design Criteria, are:

- 8 lb 1-inch diameter steel rod, 36 inches long traveling at 116 ft/sec.
- 1490 lb-13.5 inch diameter utility pole, 35 feet long traveling at 77 ft/sec.

The penetration analysis compared the kinetic energy of the two tornado missiles to the energy required to penetrate the pipe/duct. The acceptance criteria was that the total kinetic energy of the missile prior to impact is less than the energy required for penetration. Based on the analysis results, the licensee concluded that the CREATS pipe/duct would not be penetrated by either design basis missiles, the 13.5 inch diameter utility pole, or the 1-inch diameter steel rod. The licensee found the safety factor for missile penetration by the 13.5 inch-1490 lb utility pole to be 7.23, and the safety factor for missile penetration by the 1-inch 8 lb steel rod to be 11.96.

The impact analysis modeled the pipe/duct as an elastic-plastic spring using the approach specified in Civil Engineering and Nuclear Power, Vol. V, “Report of the ASCE Committee on Impactive and Impulsive Loads," which also defines the forcing function of the utility pole missile. That forcing function was factored to consider the site specific design basis missile. Because a forcing function is not available for the case of the steel rod missile impact, the analysis for this case was based on the principles of conservation of energy and momentum. The acceptance criteria for the impact analysis of both missiles was that the pipe/duct remains in the elastic range and is therefore ductile enough to absorb the energy transmitted to it during the impact. The analysis results show that the CREATS pipe/duct remains in the elastic range for the impact of the design basis steel rod and the utility pole. The licensee further indicated that, in performing this analysis, it has conservatively assumed that the pipe/duct was unsupported creating a longer piping span. Thus, the licensee concluded that the pipe/duct is ductile enough such that the impact from either missile would not cause CREATS pipe/duct failure.

The licensee utilized ANSI A58.1-1982, and a maximum wind speed of 188 mph to determine the velocity pressure and total wind pressure. The return air duct was used in determining the velocity pressure and wind pressure because it has a higher surface area over the supply duct. The results from the total wind pressure were used in conjunction with the results from the impact analysis for the load combinations. The results indicate that the velocity pressure is 119.4 lb/square-ft. Using the design criteria-case 2 load combination of deadweight, live load and tornado wind load, the licensee found the combined stress of 4,098 psi to be well below the acceptance criteria of 23,100 psi.

The licensee further stated that the results from the impact analysis and tornado wind analysis were combined along with the deadweight and live loads to determine the maximum stress on the pipe/duct. The combined loads were compared to the allowable stress to determine its acceptability. A combined load stress lower than the allowable stress results in sufficient ductility that the pipe/duct can withstand the combined deadweight, live, tornado wind, and missile impact loads. The pipe/duct was determined to have sufficient ductility to withstand the impact of both missiles. Using the design criteria-case 8 load combination of normal operating load, tornado wind load and tornado missile load, the licensee found the combined stress of

33,848 psi to be below the acceptance criteria of 36,960 psi.

The NRC staff has reviewed the results of the licensee's structural analysis for duct/pipe between the Relay Room Annex and the Control Building. The staff has concluded that the assumptions and loads used in the modeling are acceptable and the resulting stresses are within allowable limits. Therefore, the staff finds that the duct/pipe between the Relay Room Annex and the Control is structurally qualified to withstand the combined deadweight, live, tornado wind, and missile impact loads.

Summary

Based on its review as described above, the staff finds the licensee's design information for the proposed CREATS modification as discussed in Section 3.10 of this SE, including the design information related to the piping interconnection between the relay room annex and control building east wall, to be reasonable and acceptable.

## 3.6 Cable Separation and Diesel Generator Loading Analysis

### 3.6.1 Electrical Modification Description

Train A - New equipment will include a new 480V MCC-N to be fed from the existing MCC-C (which in-turn is fed from 480V bus "14" which has an emergency feed from DG A). The MCC-N will feed CREATS fan A, cooling unit A, heating unit A, and a 480V-120V transformer PXCB008 (which will in-turn feed a 120V Panel ACPDPCB11 with supplies to the actuators of Train A dampers).

Train B - New equipment will include a new 480V MCC-P to be fed from the existing MCC-D (which in-turn is fed from 480V bus "16" which has an emergency feed from DG B). The MCC-P will feed CREATS fan B, cooling unit B, heating unit B, and a 480V-120V transformer PXCB009 (which will in-turn feed a 120V Panel ACPDPCB12 with supplies to the actuators of Train B dampers).

All new safety-related motor control centers, breakers, distribution panels, cables and other electrical equipment are being purchased Class 1E through vendors with a qualified Appendix B QA program and qualified to IEEE 323-1974.

The equipment will have Seismic Qualification per IEEE 344-1987.

### 3.6.2 Diesel Generator Load Analysis

The licensee has provided the following Emergency Diesel Generator (EDG) Loading Analysis:

The new CREATS and Control Room Emergency Heating/Cooling loads are:

Fan - 25 HP,
Heating Unit- 27.8 kVA
Cooling Unit - 16 kVA
120V loads - 2 kW (480V-120V, 7.5 kVA transformer loaded to about 30% of its capacity)

During the injection phase of an accident, a timer will start the new 25 HP fan motor approximately 50 seconds after a safety injection. Both the heater and cooling units are controlled manually and will not be turned on until after the injection phase is complete.

The 25 HP fan motor is started approximately 9 seconds after all other large ECCS loads are started. The simulation results show that frequency and voltage variations, and maximum peak transient kW loading associated with turning on the 25 HP motor are less than what are experienced earlier in the loading sequence (associated with starting of large loads). The voltage and frequency variations remain within the guideline limits of RG 1.9.

The rating of each EDG is as follows:

| | |
|---|---|
| Continuous | 1950 kW |
| 2 Hours | 2250 kW |
| 30 minutes | 2300 kW |

The CREATS modification increases the maximum steady state loading from 1982 kW to 2003 kW. Although the 102.7% loading of the continuous duty rating is the maximum duty expected, the injection phase lasts less than 2 hours for a worst-case loading (LOCA) scenario, and then the loading on the diesel generator is reduced thereafter (maximum 1713 kW during the recirculation phase after LOCA). This new duty on the EDG is well within the 2-hour rating for the equipment.

Based on the above analysis, the impact of CREATS modification on the EDGs will be minimal.

3.6.3 Breaker Sizing Review for the New CREATS Loads

In Attachment 2 of the letter dated December 3, 2004, the licensee stated:

“The feeder breakers to MCC N (P) from existing 480 volt MCC C (D) are 100 amp thermal magnetic breakers, sized to provide over 125% of expected capacity. The breakers were reviewed to ensure cable protection to the 4/0 feeder cable, including all cable deratings. Coordination curves were created versus upstream MCC and bus breakers to ensure adequate coordination.”

“The load breakers in the new MCCs N & P are sized to carry a minimum of 125% of expected load and have an instantaneous setting of 173% of the motor’s locked rotor current, while providing adequate cable protection. The three breakers supplying the Fan, Cooler, and Heater are each fed through a 50 amp magnetic-only breaker via a contactor with a built in thermal overload device appropriately sized for the load. The transformer is fed by a 20 amp thermal breaker sized for full transformer rating.”

Because the breaker sizes and settings exceed the required capacity, the staff finds that they are acceptable.

3.6.4 Separation and Isolation of Redundant Class 1E Circuits

The licensee provided the following information on separation and isolation of redundant class 1E circuits:

There are no CREATS 480 volt power cables which run in cable trays. The Train A and Train B 480V cables are routed in separate conduits.

CREATS 120V AC or 125V DC cables are routed in conduits and cable trays. Separate conduits and trays are used for Train A and Train B cables. However, where common cable tray is utilized for both trains, physical separation in the cable tray will be maintained so that A and B train cables are routed in separate sections of the tray.

CREATS pre-modification cables that run in cable trays have fault protective devices installed (fuses or breakers) to protect the cables from sustaining or propagating a fault across both CREATS trains.

In the Auxiliary Benchboard (ABB), the control and internal wiring for each train have been separated by the distinct sections of the ABB. The ABB center section will house train A components, the ABB right section will house train B and non-divisional (normal HVAC) components. The cabinet panels themselves provide the separation between the devices. Per IEEE 384-1981, wiring of redundant trains installed inside of the cabinets must be maintained at minimum separation distances or have suitable barriers between redundant trains. That minimum separation will be maintained for all train specific wiring associated with this modification, or suitable barrier installed between circuits when minimum separation distances cannot be maintained. However, the licensee took exception to IEEE 384-1981 as it applies to the wiring for the logic to damper isolation relays where wires connect A train relays to B train relays. This exception has been justified by the licensee based on the review of the logic wiring which demonstrates that any fault in the cabinet that causes failure of a wire will result in an opening of the associated circuit. This opening of the circuit will only impact the affected train. There is no credible failure mode that would result in a condition in which faulted or failed wires in the ABB would prevent the safety system from performing its function if an actuation signal is present.

All non-safety-related cables and loads fed from a safety-related source of power are electrically isolated from that power source with qualified isolation devices in accordance with the requirements of IEEE 384-1981.

The licensee's design philosophy is to provide protection schemes that assure that the fuses and breakers selected for a circuit are sized to protect the cable from damage. The fuse/breaker clearing curves are evaluated to ensure that a fault or overload clears before cable damage is reached. The fault protection of all cables under this modification has been analyzed to show that the protective device for each cable will clear all faults before the cable damage curve is reached. This feature ensures that the fault will clear before cable damage can cause a tray fire that could spread to the redundant train.

Based upon the above, the cable separation and isolation justification provided by the licensee (which is consistent with the design criteria used in the past) is acceptable.

3.6.5 Summary

The NRC staff has reviewed the EDG load analysis, breaker sizes, and separation and isolation criteria of redundant class 1E circuits, associated with the proposed CREATS modification at Ginna Nuclear Power Plant. As discussed above, the staff finds the modification acceptable.

3.7 CREATS Instrumentation

The licensee committed to evaluate and revise the control room radiation monitor analytical limit (AL) to reflect the use of the AST. The staff reviewed design calculation DA-EE-2001-013 to verify that the analysis for the AL was appropriate for the most limiting DBA. The staff determined that the calculation for the AL was conservative and that the limiting safety system setpoint (LSSS) was appropriately derived from the AL based on the Ginna setpoint calculation DA-EE-2000-09.

The requested TS changes with regards to operability are consistent with the Performance-Based Technical Specifications (PBTS) approach approved for Ginna by the NRC (see SE dated September 22, 2004, Accession Number ML041180293). The PBTS approach verifies channel operability by measuring the amount the trip setpoint varies over the surveillance interval and then comparing the results with the design values given in the Ginna setpoint calculations. If the channel is found to be behaving in accordance with the design expectations, the channel is declared operable. Otherwise it is declared inoperable and appropriate actions are initiated.

On the bases of this review, the staff concludes that the LSSS and its operability requirements for the control room radiation monitor as defined in TS function table 3.3.6-1 provide adequate assurance that there is at least 95% probability with 95% confidence level that the ALs, and hence the Safety Limits, will not be violated. This success and probability level are consistent with the guidelines of RG 1.105 and are, therefore, acceptable.

3.8 Appendix R Analysis

The licensee has identified four plant changes associated with this modification that may have an adverse impact on either the plant's fire protection program or Appendix R compliance. These four plant changes are: 1) supply air and return air ducts within the control room emergency zone, 2) supply air and return air ducts/pipes between the control building emergency zone and the relay room annex, 3) electric power, controls, power distribution equipment, and 4) two new trains of CREATS mechanical HVAC components and ductwork, including charcoal filter units. The licensee concluded than no Appendix R/Fire Protection evaluation is required for plant changes 1 and 2, and has performed a conformance verification for plant changes 3 and 4.

A new fire detection system will be installed in the relay room annex. The system will include the installation of smoke detection in each beam pocket and will be designed and installed in accordance with the requirements of National Fire Protection Association (NFPA) 72. The new detection system will alarm in the control room, which is a constantly manned location. Manual fire protection for the relay room is provided via hose reels from outside the relay room. In addition to wide area smoke detection, high temperature conditions within the charcoal filters

will be monitored and annunciated to the auxiliary benchboard in the control room. A high temperature alarm from the charcoal filters will initiate the use of an alarm response procedure that will direct operators to verify the high temperature and, if required, initiate the deluge system within the charcoal filter to extinguish the fire.

The charcoal filter unit is equipped with a deluge system that provides a flow rate of 3.2 gpm per cubic foot of carbon in the filter units. This design is consistent with American Society of Mechanical Engineers (ASME) AG-1, 1997. Although the deluge system is not hard piped to the plant fire water system, upon verification of high temperature in the charcoal filter bank, a hose will be attached to provide fire water to the deluge system. If suppression within the charcoal filter bank is delayed, leakage through the isolation dampers is unlikely due to the heavy construction of the dampers (10 gage or thicker steel) and the ASME AG-1 leakage class 1 rating. The dampers are unlikely to be damaged by the fire since the distance between the charcoal and the dampers is at least 10 feet.

The licensee performed calculations to determine the heat-up of the CREATS charcoal filters due to iodine desorption. Their calculations estimate the charcoal filter heat-up rate to be approximately 5.20E-7 deg F/sec. Based on the results of this calculation, it is unlikely that the CREATS filters will heat up from normal room temperature to the ignition temperature of the charcoal (680 deg F) due to iodine desorption.

All new penetrations in rated barriers will be sealed in accordance with plant procedures to maintain the fire integrity of the barrier. Non-fire rated barriers that are part of gaseous fire suppressant system enclosures will be sealed in accordance with plant procedures. Any structural steel fire proofing removed for this modification will be reinstalled in accordance with plant procedures.

All new cables will be IEEE-383 qualified. Additional combustible loading will be considered in the fire hazards analysis of affected areas. These additions will be considered in the plant's fire combustible loading program and fire hazards analysis where appropriate. No concentrations of cables or fixed combustibles, other than charcoal, are being installed that would create the possibility of different fire scenarios that have not been previously evaluated.

The NRC staff has reviewed the application and the May 21 and August 18, 2004, supplemental letters. The staff concludes, based on the technical content of the submittals,

that plant operations with the proposed CREATS modification will have an insignificant impact on the licensee's compliance with their Fire Protection licensing basis, 10 CFR 50.48, or applicable portions of 10 CFR Part 50, Appendix R.

### 3.9 Environmental and Dynamic Effects

GDC 4, "Environmental and Missile Design Basis," requires that the modifications to the CREATS and CREZ be protected from HELBs and/or missiles. The CREZ occupies the top floor of the three-story control building and has only two adjacent indoor spaces: the turbine building operating deck to the north, and the relay room annex located one floor below the control room. The control room north and east walls have 0.25-inch steel plates to protect the control room from HELB and/or missiles. Because the proposed modification affects the east wall of the control building, the staff limited its review to the potential effects the modification

would have on the CREZ boundary due to seismic, tornado missiles and external smoke and fire. The potential effects due to external smoke and fire were obtained from the licensee's proposed changes to Ginna UFSAR Section 6.4.

3.9.1 Seismic and HELB Effects

The supply and return ducts for the CREATS are located outside the control building. Therefore, the CREZ integrity could be compromised due to the loads generated by seismic activity, tornadoes, or missiles. The staff reviewed the licensee's structural analysis for the CREATS supply and return ducts for all applicable loads. As indicated in Section 3.5 of this SE, the staff concluded that the new supply and return ducts are seismically qualified for both OBE and SSE. The staff also reviewed the licensee's impact analysis for two specific design basis tornado missiles:

- 8 lb 1-inch diameter steel rod, 36 inches long traveling at 116 ft/sec.
- 1490 lb-13.5 inch diameter utility pole, 35 feet long traveling at 77 ft/sec.

The analysis for both missiles indicated that the pipe/duct remains in the elastic range and is, therefore, ductile enough to absorb the energy transmitted during impact and will not fail when impacted by a tornado generated missile.

On the basis of this review, the staff concludes that neither a seismic event nor the impact from design basis tornado missiles would cause the supply and return ducts for the CREATS or the control building east wall to fail and, thereby, compromise CREZ integrity.

3.9.2 External Sources of Smoke and Fire

The licensee stated that a fire in any space adjacent to the control room is unlikely to spread to the CREZ and require control room evacuation because:

a. The control room's north wall is protected, on the turbine building side, by a water curtain fire suppression system.

b. The relay room annex, located below the control room, is protected by both a halon and a manually actuated water suppression system.

c. The west and south walls of the control room are adjacent to the main, auxiliary, and two offsite transformers. These transformers present a significant fire load, but all are equipped with fire detection and water suppression systems. The west and south walls are also 3-hour fire rated, 20-inch thick reinforced concrete, with minimum of penetrations.

d. The roof of the control room is 20-inch thick reinforced concrete, and the east wall is shielded with .25-inch thick armor plate. There is no significant combustible load located outside on the roof or the east wall of the control building.

Smoke from a fire in areas adjacent to the control room is not expected to affect control room habitability because of the limited inleakage across CREZ boundaries. If a fire occurred in adjacent spaces, especially the transformer yard, actuating the emergency mode of operation

would isolate the control room to prevent smoke from outside being brought into the control room by the normal HVAC system. Redundant dampers make this isolation function single failure proof but if, by some extreme circumstances, excessive smoke entered the control room from fire in an adjacent area, there are three paths available for operators to exit the Control Building and from there proceed to the alternative shutdown panels:

1. through the main control room door to the Turbine Building operating level.

2. through the control room's back door to the stairwell, relay room, and then to the mezzanine level of the Turbine Building.

3. through the control room's back door to the stairwell, relay room, and then through the relay room annex to the outdoors.

It is not credible for a single fire event to simultaneously admit excessive smoke to the control room and obstruct all three of these pathways.

Since all ductwork and housing for the CREATS will meet the design requirements for leakage of Class I equipment as described in Appendix SA-B of ASME AG-1-2003, the licensee's assumption that inleakage across the CREZ boundaries will be limited is reasonable. In addition, smoke detectors located in the normal return air duct will provide a control room alarm which will allow the operator adequate time to manually actuate the CREATS and isolate the CREZ.

3.10 Control Room Habitability

The licensee has undertaken an initiative to upgrade the CREATS. The upgraded CREATS will have redundant trains in place of the current single train system. In addition, it includes the following new features:

- two new filter trains complete with fans, HEPA and charcoal filters, and coolers
- filtered control room air recirculation
- safeguard power source with diesel generator backup

The licensee stated that the new upgraded CREATS will provide redundancy and improve its reliability and performance reducing control room unfiltered air inleakage. The licensee further stated that the current existing CREATS will remain in place and will serve as normal HVAC, except for the existing control room emergency fan and filter unit which will be either removed or abandoned in place.

Each train will provide a minimum of 5400 cfm (6000 cfm nominal minus 10%) of control room air recirculation flow through HEPA and charcoal filters. A new actuation signal will be installed to automatically shift the CREATS to the emergency mode upon receipt of a safety injection signal in addition to the already existing manual, high radiation, and toxic gas initiation signals. During normal operation, fresh air is admitted to the control room air ventilation system through an intake louver located on the outside wall of the turbine building with 2200 cfm makeup air flow (2000 cfm nominal plus 10%).

In the event of an accident, the CREATS actuation instrumentation will automatically close the redundant dampers in the fresh air intake duct and the dampers in the return air duct to the

turbine building, and open the dampers to the CREATS to allow a minimum of 5600 cfm air. The licensee assumed a control room isolation delay of 60 seconds to account for damper positioning and instrumentation delays and an additional 10 seconds for the new CREATS to be operational. The licensee assumed an unfiltered air inleakage rate of 300 cfm during the entire period of each DBA.

On June 12, 2003, the NRC staff issued Generic Letter (GL) 2003-01, "Control Room Habitability." This GL identifies NRC staff concerns regarding the reliability of current surveillance testing to identify and quantify control room inleakage, and requested licensees to confirm the most limiting unfiltered inleakage into their control room envelope. On August 4, 2003, the licensee submitted a response to this GL for Ginna. In its response, the licensee stated that it will perform a tracer gas in-leakage test to verify the 300 cfm unfiltered air inleakage rate assumed in this LAR.

Separately, the licensee also committed to perform a tracer gas inleakage test of the control room envelope after completion of the CREAFS modification in Attachment 5, “List of Regulatory Commitments,” of its submittal dated May 21, 2003, and in its letter dated June 30, 2004 utilizing methods from ASTM E741, “Standard Test Method for Determining Air Change in a Single Zone by Means of a Tracer Dilution,” upon completion of the CREATS modification. The test was performed on February 8 and 9, 2005. In a conference call with the NRC staff on February 11, 2005, the licensee stated that the results of the tracer gas testing showed an inleakage of between 4 cfm ($\pm$ 9 cfm) and 15 cfm ($\pm$ 6 cfm) for various configurations of fans running and dampers closed. The final report of the test will be completed in March 2005. The licensee has stated that the final report will be completed prior to implementation of the modification and is being tracked by the licensee as part of the CREATS modification program. The NRC staff finds that reasonable controls for the completion of this report can be provided by the licensee's administrative processes. Because the test results are significantly below the assumption of 300 cfm used in the dose analyses, the staff finds that 300 cfm in-leakage value is acceptable for use as an assumption to the accident analyses.

Using assumed unfiltered air inleakage rate of 300 cfm, the licensee re-evaluated the radiological consequences resulting from the postulated DBAs and concluded that the radiological consequences in the control room are within the dose criteria specified in 10 CFR 50.67. The staff reviewed these analyses as discussed in Section 3.2 of this SE and found them acceptable.

The licensee conservatively assumed recirculation filter efficiencies of 90%, 70%, and 98% for iodine in elemental, organic, and particulate forms, respectively. The staff finds that the recirculation filter efficiencies assumed are more conservative than the test criteria specified in Ginna TS Section 5.5.10, “Ventilation Filter Testing Program,” and the guidelines provided in RG 1.52. Therefore, the staff finds that the recirculation filter efficiencies used by the licensee are acceptable.

The function of the CREATS is to satisfy GDC-19 by providing a protected environment from which control room operators can control the plant for 30 days after a DBA without exceeding a dose limit of 5 rem TEDE. The CREATS also protects control room operators from exposure to chlorine or ammonia following an accidental release from sources on or near the Ginna site (see Section 3.4 of this SE). The CREATS is designed for continuous operation for a minimum period of 30 days in the accident environment. Both 100% capacity trains of CREATS will normally be in standby, while a separate existing system provides heating,

cooling, and ventilation during normal modes of operation. Both trains of the CRECS will also normally be secured, in standby. If either CREATS train is actuated, its associated CRECS will maintain control room temperatures conducive to continuous occupancy during normal or accident conditions.

The licensee provided the following information regarding the qualifications of the components used in the new CREATS system:

- the new CREATS system will be located in the Relay Room Annex, which is not part of the Control Room Emergency Zone (CREZ); thus the pressure boundary of each component (ductwork, fan housing, filter housing, etc) functions as a boundary of the CREZ.

- Ducts and housing will be designed and fabricated in accordance with sections SA and HA of ASME AG-1-2003, "Code of Nuclear Air and Gas Treatment." Both will meet the requirements of Leakage Class I described in Appendix SA-B of AG-1-2003.

- Unless otherwise noted, all new components installed by this modification will be safety related, and will be Seismic Category I.

- 10 CFR Part 50 Appendix B Quality Assurance (QA) requirements will be applied.

The staff reviewed the licensee's proposed design to assure that control room operators are adequately protected against the effects of accidental releases of toxic and radioactive gases and that the control room can be maintained as the backup center from which technical support center personnel can safely operate in the case of an accident.

Because this is a modification to an existing CREATS, the staff also reviewed its layout and functional design to determine flow rates and filter efficiencies used for input into the radiological consequence and toxic gas analyses and found them to be acceptable as discussed in Sections 3.2 and 3.4 of this SE.

## 3.11 Technical Specification Changes

### 3.11.1 Section 1.1, "Definition - Dose Equivalent I-131"

This proposed change would revise the definition of dose-equivalent I-131 in TS Section 1.10 by replacing the current reference to RG 1.109 with a reference to International Conference on Radiation Protection (ICRP) 30 for the thyroid dose conversion factors. The intent of the TSs on specific activity is to ensure that assumptions made in the DBA radiological consequence analyses remain bounding. As such, the specification should have a basis consistent with the basis of the dose analyses. Historically, licensees have calculated the dose equivalent I-131 using thyroid dose conversion factors, since the limiting analysis result was the thyroid dose. The AST analyses, however, determine the TEDE rather than the whole-body dose and thyroid dose as done previously.

The NRC staff finds that the use of the thyroid dose conversion factors in the definition of dose

equivalent I-131 results in higher radioiodine inventory in the primary coolant than would be obtained using the effective dose conversion factors (DCFs), resulting in potentially higher doses at the EAB, LPZ, and CR. However, the staff notes that the thyroid DCFs would result in lower concentrations of radioiodines in the RCS (by approximately 3%), allowing slightly higher dose-equivalent I-131 limit in the Ginna TSs. Nevertheless, the thyroid dose conversion factors in the definition of dose equivalent I-131 would ultimately result in higher doses at the EAB, LPZ, and CR. Therefore, given the conservative nature of the dose equivalent I-131 inventory based on thyroid dose, the staff finds the licensee's proposed definition of dose-equivalent I-131 is acceptable.

3.11.2 Section 3.3.6, "Control Room Emergency Air Treatment System (CREATS) Actuation Instrumentation"

Technical justification for changes to the LSSS settings for the control room radiation monitors is given in Section 3.7 of this SE.

3.11.3 Section 3.4.16, "RCS Specific Activity"

The licensee proposed to eliminate TS Figure 3.4.16-1, "Reactor Coolant DOSE EQUIVALENT I-131 Specific Activity Limits Versus % of Rated Thermal Power," and instead provide a single limit of 60 micro curies per gram in TS Section 3.4.16, "RCS Specific Activity," Action A.1 and Condition B. The surveillance requirement (SR) 3.4.16.2, verifying reactor coolant DOSE EQUIVALENT I-131 specific activity to be less than 1.0 micro curie per gram of reactor coolant is not affected and remains the same. The staff finds that this request is acceptable because there is no change in technical requirements and it is consistent with Westinghouse Standard Technical Specifications (NUREG-1431, Revision 2).

3.11.4 Section 3.6.6, "Containment Spray (CS), Containment Recirculation Fan Cooler (CRFC), NaOH, and Containment Post Accident Charcoal Systems"

The licensee proposed to eliminate the requirement for containment post-accident charcoal filters and associated SR. The containment post-accident charcoal filters will not be credited for evaluating potential radiological consequences at the EAB, LPZ, and control room. This change results in an operational efficiency that is achievable from implementing the AST. The revised radiological analyses of the DBAs are performed without taking credit for the containment post-accident charcoal filters and the results show that the EAB, LPZ, and control room doses remain below the guidance provided in RG 1.183. Therefore, the staff finds that this request acceptable.

3.11.5 Surveillance Requirements 3.6.6.8 and 3.6.6.9 - Verification of NaOH System Volume and Concentration

The role of sodium hydroxide is to maintain alkaline pH in the containment sprays during the injection phase of an accident and in the sump during the recirculation phase to provide for effective iodine removal. The volume of sodium hydroxide solution and the concentration of sodium hydroxide are specified in revised TS SRs 3.6.6.7 and 3.6.6.8, respectively. The licensee requested to change these specifications by decreasing the volume of the solution from 4500 gallons to 3000 gallons and specifying an upper limit of# 35 wt % for the sodium hydroxide concentration in the spray additive tank.

The licensee performed an analysis which indicated that the change in the volume and concentration of sodium hydroxide will not produce unacceptable values for pH in the spray and containment sump solutions. The licensee demonstrated that when a volume of between 1048 and 2977 gallons of the 30 wt % to 35 wt % sodium hydroxide solution from the spray additive tank is mixed with the borated water accumulated in the containment sump, the resulting pH will be in the range of 7.9 to 9.7. This exceeds the minimum pH of 7.0 specified in the SRP. The staff reviewed the results of the licensee's analyses described in the Ginna UFSAR and independently verified the resulting pH. The staff concludes that the proposed changes to the sodium hydroxide spray additive TS meet the requirement of GDC-41 as it applies to containment atmosphere cleanup and, therefore, are acceptable.

3.11.6 TS Sections 3.7.9, "Control Room Emergency Air Filtration System (CREATS)"

The replacement of the existing CREATS necessitates that the TSs and the ventilation filter testing program be modified to reflect the characteristics of the new upgraded CREATS. Accordingly, the licensee proposed to modify TS 3.7.9, "Control Room Emergency Air Treatment System (CREATS)."

The proposed changes reflect the change in design of the CREATS to a two train system. Previously, the CREATS consisted of a single train. The implementation of the two train design resulted in the following changes to the TSs.

1. If one train is inoperable, the allowable outage time to restore the inoperable train is 7 days.

2. Since the two train design has been incorporated, it is no longer necessary for Ginna to configure the control room ventilation systems such that outside makeup air is isolated and control room ventilation is operating in a recirculation mode of operation.

3. Since the isolation configuration described in item 2 above is no longer necessary as a compensating mode of operation when the CREATS train is inoperable, CONDITION B, which addressed a single inoperable isolation damper, was deleted. Also deleted were all references to that CONDITION and all REQUIRED ACTIONS involving an inoperable isolation damper. Also deleted were CONDITIONS which addressed two inoperable isolation dampers in MODES 1-4, during movement of irradiated fuel assemblies or during CORE ALTERATIONS.

The NRC staff finds that the above proposed TS changes are consistent with the requirements for a two train system and are acceptable.

3.11.7 TS Section 5.5.10, "Ventilation Filter Testing Program (VFTP)"

With the new upgraded CREATS, certain changes were necessary to the Ventilation Filter Testing Program to reflect the new CREATS design. The proposed changes also reflect increased quality of the new design including lesser allowable penetration of the HEPA filter and charcoal absorber during an in-place test and greater methyl and elemental iodide removal capability. The changes included:

1. Demonstration of an acceptable pressure drop across the pre-filter, HEPA filter,

charcoal absorber, and post-filter rather than the demonstration of acceptable pressure drops separately across the HEPA filter bank and the charcoal absorber bank.

2. Change in the acceptance criteria for the in-place tests of the HEPA filter and charcoal absorber from 1% to 0.05% penetration and system bypass.

3. Change in the acceptance criteria for the laboratory test of the charcoal absorber to #1.5% methyl iodine penetration to reflect the assumed absorber efficiency of 97% based upon a 0.33 second residence time for the 4-inch absorber bed depth.

4. Designation of the performance of the laboratory test of charcoal at a face velocity of 61 ft/min based actual design.

The NRC staff has reviewed these proposed TS and ventilation filter testing program changes. The staff finds these TS changes consistent with those for systems designed to RG 1.52 except with respect to charcoal absorber residence time. The existing CREATS does not possess a 0.25 second residence time per 2-inch bed depth of charcoal. The new CREATS design incorporates 4-inch bed and a flow rate which equates to a face velocity of 61 ft/min and a 33 second residence time. Because this new design is not based on the standard 40 ft/min face velocity, it was necessary to specify the face velocity for which the laboratory should conduct the test to determine methyl iodine penetration. The licensee indicated that the actual face velocity is 61 ft/min. The licensee has proposed that the laboratory test be conducted at a face velocity of 61 ft/min.

The staff notes that at 61 ft/min an absorber efficiency of approximately 97% for elemental and organic forms of iodide could be assumed, provided the laboratory test of charcoal is conducted at a face velocity of 61 ft/min and the acceptance criteria for the laboratory test is 1.5% penetration. However, the licensee conservatively used 90% for elemental and 70% organic forms of iodine in its control room habitability dose calculations for DBAs. Therefore, the staff also used 90% for elemental and 70% organic forms of iodine in its confirmatory calculations.

The staff finds that the above proposed acceptance criteria for the ventilation filter testing program are consistent with the assumptions used in the dose calculations for the DBAs. Therefore, the proposed changes to TS Section 5.5.10 are acceptable.

### 3.12 Summary

As described above, the staff reviewed the assumptions, inputs, and methods used by the licensee to assess the radiological impacts of the proposed full implementation of an AST, proposed modifications to the CREATS, and its associated TS change requests.

Based on the review of the applicant's proposed design criteria, the design bases, and safety classification for the CREATS, and the requirements for system performance to maintain a suitable environment during normal, abnormal, and accident conditions, the staff concludes that the design of the CREATS and its auxiliary supporting system is in conformance with the Commission's regulations as set forth in GDC 2, 4, 5, 19, and 60. This conclusion is based on the following:

1. The licensee has met the requirements of GDC 2, “Design Bases for Protection Against Natural Phenomena,” with respect to the system being capable of withstanding the effects of earthquakes by meeting the guidelines of RG 1.29, “Seismic Design Classification,” position C.1 for safety-related portions of the system and position C.2 for nonsafety-related portions of the system. The licensee stated in its letter dated February 16, 2004, that “unless noted otherwise, all new components installed by this modification shall be safety related, and will be seismic Category I, per Section C.1.n of RG 1.29, ‘Seismic Design Classification’.” The staff concluded in Section 3.5 of this SE that the licensee’s seismic analysis results for the external duct/piping between the Relay Room Annex and the Control Building are reasonable and acceptable.

2. The licensee has met the requirements of GDC 4, “Environmental and Dynamic Effects Design Basis,” by maintaining environmental conditions in the control room compatible with the design limits of the essential equipment located therein during normal, transient, and accident conditions. The staff concluded in Section 3.9 of this SE that neither external fire, smoke, seismic or impact from design basis tornado missiles would cause the external supply and return ducts for the CREATS or the control building east wall to fail and, thereby, compromise CREZ integrity. In Section 3.8 of this SE, the staff also concluded that plant operations with the proposed CREATS modification will have an insignificant impact on licensee’s compliance with their Fire Protection licensing basis, 10 CFR 50.48, or applicable portions of 10 CFR Part 50, Appendix R.

3. The licensee has met the requirements of GDC 17, “Electric Power Systems,” by ensuring that the electrical modifications associated with new CREATS have sufficient independence, redundancy, and testability to perform its safety function.

4. The licensee has met the requirements of GDC 19, “Control Room,” as it relates to maintaining the control room in a safe, habitable condition under accident conditions by providing adequate protection against radiation. Sections 3.1 and 3.4 of this SE verified that the proposed modifications to the CREATS would provide the reactor operators adequate protection against the effects of accidental releases of radioactive gases in accordance with GDC 19.

5. The licensee has met the requirements of GDC 20, “Protection system functions,” by ensuring that the setpoint and its operability requirements for the control room radiation monitor provide adequate assurance that the safety limits will not be violated.

6. The licensee has met the requirements of GDC 60, “Control of Releases of Radioactive Materials to the Environment,” by ensuring that the radiological consequences of the design basis accidents at the EAB and at the LPZ are within the dose criteria specified in 10 CFR 50.67.

## 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 (68 FR 40718). 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.

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**Table 1**

**Radiological Consequences Expressed as TEDE** (1)
**(rem)**

| Design Basis Accidents | EAB (2) | LPZ (3) | Control Room |
|---|---|---|---|
| LOCA | 2.69 | 1.02 | 4.3 |
| Dose criteria(4) | 25 | 25 | 5.0 |
| Fuel-handing accident in containment | 5.07E-1 | 5.87E-2 | 1.16 |
| Dose criteria | 6.3 | 6.3 | 5.0 |
| Fuel-handing accident in auxiliary building | 1.38E-1 | 1.59E-2 | 9.85E-2 |
| Dose criteria | 6.3 | 6.3 | 5.0 |

| | | | |
|---|---|---|---|
| Main steamline break accident [4] | 4.76E-1 | 1.27E-1 | 6.32E-1 |
| Dose criteria | 2.5 | 2.5 | 5.0 |
| Main steamline break accident [5] | 6.96E-2 | 2.80E-2 | 1.74E-1 |
| Dose criteria | 25 | 25 | 5.0 |
| Steam generator tube rupture [4] | 9.70E-2 | 1.40E-2 | 1.40E-1 |
| Dose criteria | 2.5 | 2.5 | 5.0 |
| Steam generator tube rupture [5] | 3.20E-1 | 4.30E-2 | 8.90E-1 |
| Dose criteria | 25 | 25 | 5.0 |
| Locked rotor accident | 1.0 | 3.03E-1 | 1.88 |
| Dose criteria | 2.5 | 2.5 | 5.0 |
| Rod ejection accident | 6.64E-1 | 2.03E-1 | 1.06 |
| Dose criteria | 6.3 | 6.3 | 5.0 |
| Tornado missile accident | 2.16E-2 | 4.79E-3 | 3.55 [6] |
| Dose criteria | 6.3 | 6.3 | 5.0 |

[1] Total effective dose equivalent
[2] Exclusion area boundary
[3] Low population zone
[4] Accident initiated iodine spike
[5] Pre-accident iodine spike
[6] Maximum dose without CR recirculation filtration

**TABLE 2**

**Control Room Atmospheric Dispersion Factors**

**Release Pathway**
Containment Leakage

**Accidents**
Loss-of-coolant
Rod Ejection

| Time Interval | χ/Q Value (sec/m$^3$) |
|---|---|
| **0–2 hours** | $1.77\times10^{!3}$ |
| **2–8 hours** | $1.25\times10^{!3}$ |
| **8–24 hours** | $4.80\times10^{!4}$ |

| | |
|---|---|
| **1–4 days** | $4.24\times10^{!4}$ |
| **4–30 days** | $3.66\times10^{!4}$ |

**Release Pathway**
Containment Equipment Hatch

**Accidents**
Fuel Handling Accident (inside containment)

| Time Interval | χ/Q Value (sec/m$^3$) |
|---|---|
| **0–2 hours** | $5.58\times10^{!3}$ |
| **2–8 hours** | $4.66\times10^{!3}$ |
| **8–24 hours** | $1.65\times10^{!3}$ |
| **1–4 days** | $1.58\times10^{!3}$ |
| **4–30 days** | $1.32\times10^{!3}$ |

**Table 2 (cont'd)**

**CR Atmospheric Dispersion Factors**

**Release Pathway**
Atmospheric Relief Valves

**Accidents**
Locked Rotor
Rod Ejection (secondary side release)
Steam Generator Tube Rupture

| Time Interval | χ/Q Value (sec/m$^3$) |
|---|---|
| **0–2 hours** | $3.72\times10^{!3}$ |
| **2–8 hours** | $2.51\times10^{!3}$ |
| **8–24 hours** | $1.15\times10^{!3}$ |
| **1–4 days** | $8.35\times10^{!4}$ |
| **4–30 days** | $6.88\times10^{!4}$ |

**Release Pathway**
Plant Vent

**Accidents**
Fuel Handling Accident (in the spent fuel pool)

| Time Interval | χ/Q Value (sec/m$^3$) |
|---|---|
| **0–2 hours** | $1.99\times10^{!3}$ |
| **2–8 hours** | $1.46\times10^{!3}$ |
| **8–24 hours** | $6.35\times10^{!4}$ |
| **1–4 days** | $5.01\times10^{!4}$ |
| **4–30 days** | $4.47\times10^{!4}$ |

**Table 2 (cont'd)**

**CR Atmospheric Dispersion Factors**

**Release Pathway**
Auxiliary Building Leakage

**Accidents**
Loss-of-coolant (ECCS leakage)

| Time Interval | χ/Q Value (sec/m$^3$) |
|---|---|
| **0–2 hours** | $4.69\times10^{!3}$ |
| **2–8 hours** | $3.97\times10^{!3}$ |
| **8–24 hours** | $1.40\times10^{!3}$ |
| **1–4 days** | $1.32\times10^{!3}$ |
| **4–30 days** | $1.11\times10^{!3}$ |

**Release Pathway**
Turbine Building Main Steam Header

**Accidents**
Main Steam Line Break (outside containment)

| Time Interval | χ/Q Value (sec/m$^3$) |
|---|---|
| **0–2 hours** | $2.59\times10^{!3}$ |
| **2–8 hours** | $1.88\times10^{!3}$ |
| **8–24 hours** | $8.28\times10^{!4}$ |
| **1–4 days** | $5.90\times10^{!4}$ |
| **4–30 days** | $4.77\times10^{!4}$ |

**Table 2 (cont'd)**

## CR Atmospheric Dispersion Factors

**Release Pathway**
Spent Fuel Pool

**Accidents**
Tornado Missile Accident

| Time Interval | χ/Q Value (sec/m$^3$) |
|---|---|
| **0–1 minute** | $5.14\times10^{!5}$ |
| **1 minute – 2 hours** | $1.44\times10^{!3}$ |
| **2–8 hours** | $1.22\times10^{!3}$ |
| **8–24 hours** | $4.54\times10^{!4}$ |
| **1–-4 days** | $4.17\times10^{!4}$ |
| **4–30 days** | $3.38\times10^{!4}$ |

**Table 3**

**EAB and LPZ Atmospheric Dispersion Factors**

**Exclusion Area Boundary (EAB) (except for the Tornado Missile Accident)**

| Time Interval | χ/Q Value (sec/m$^3$) |
|---|---|
| 0–2 hours | $2.17\times10^{!4}$ |

**Exclusion Area Boundary (EAB) (Tornado Missile Accident only)[(a)]**

| Time Interval | χ/Q Value (sec/m$^3$) |
|---|---|
| 0–1 minute | $2.17\times10^{!6}$ |
| 1 minute – 2 hours | $2.17\times10^{!4}$ |

**Low Population Zone (LPZ) (except for the Tornado Missile Accident)**

| Time Interval | χ/Q Value (sec/m$^3$) |
|---|---|
| 0–8 hours | $2.51\times10^{!5}$ |
| 8–24 hours | $1.78\times10^{!5}$ |
| 1–4 days | $8.50\times10^{!6}$ |
| 4–30 days | $2.93\times10^{!6}$ |

**Low Population Zone (LPZ) (Tornado Missile Accident only)**

| Time Interval | χ/Q Value (sec/m$^3$) |
|---|---|
| 0–1 minute | $4.14\times10^{!7}$ |
| 1 minute – 8 hours | $2.51\times10^{!5}$ |
| 8–24 hours | $1.78\times10^{!5}$ |
| 1–4 days | $8.50\times10^{!6}$ |
| 4–30 days | $2.93\times10^{!6}$ |

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[(a)]The licensee provided the 0–1 minute EAB tornado missile accident χ/Q value of $2.17\times10^{!6}$ sec/m$^3$ in its Response to Request for Additional Information letter dated December 3, 2004 (ADAMS Accession Number ML043450332).

**Table 4**
**Parameters and Assumptions Used in**
**Radiological Consequence Calculations**
**for**
**Loss-of-Coolant Accident**

| Parameter | Value |
|---|---|
| Reactor power | 1550 MWt |
| Containment volume | 1.0E+6 $ft^3$ |
| Sprayed area | 7.8E+5 $ft^3$ |
| Unsprayed area | 2.2E+5 $ft^3$ |
| Containment leak rates | |
| 0 to 1 hour | 0.2% per day |
| 1 to 720 hours | 0.1% per day |
| Containment mixing rates | |
| Sprayed to unsprayed | 4.5E+4 $ft^3$/hr |
| Unsprayed to sprayed | 4.5E+4 $ft^3$/hr |
| Aerosol removal rates by containment spray (per hour) | |
| Time | Rates |
| 0 to 80 seconds | 0 |
| 80 seconds to 52 minutes | 3.5 |
| 52 minutes to 3 days | 0 |
| Elemental iodine removal rates by spray (per hour) | |
| Time | Rates |
| 0 to 80 seconds | 0 |
| 80 seconds to 52 minutes | 20 |
| 52 minutes to 30 days | 0 |
| Containment sump volume | 2.647E+4 $ft^3$ |
| ECCS leak rates | |
| Time | Rates |
| 0 to 52 minutes | 0 |
| 52 minutes to 30 days | 4.0 gph |
| Iodine partition factors | 2 to 7% |
| Release points | |
| Containment leakage | containment |
| ECCS leakage | auxiliary building |

**Table 5**
**Parameters and Assumptions**
**Used in**
**Radiological Consequence Calculations**
**for**
**Fuel Handling Accident**

| Parameter | Value |
|---|---|
| Reactor power | 1550 MWt |
| Radial peaking factor | 1.75 |
| Fission product decay period | 100 hours |
| Number of fuel assembly damaged | 1 |
| Fuel pool water depth | 23 ft |
| Fuel gap fission product inventory (%) | |
| Noble gases excluding Kr-85 | 5 |
| Kr-85 | 10 |
| I-131 | 8 |
| Other halogens | 5 |
| Alkali metals | 12 |
| Fuel pool decontamination factors | |
| Iodine | 200 |
| Noble gases | 1 |
| Fission product inventory and activity released | Submittal Table 6.3 |
| Duration of accident | 2 hours |
| Release filtration or holdup | None credited |
| Release points | |
| FHA inside containment | containment equipment hatch |
| FHA in spent fuel pool | plant vent |

**Table 6**
**Parameters and Assumptions**
**Used in**
**Radiological Consequence Calculations**
**for**
**Main Steamline Break Accident**

| Parameter | Value |
|---|---|
| Pre-incident iodine spike activity (60 µCi/gm dose equivalent I-131) | Submittal Table 3.1 |
| Coincident spike appearance rate, based on | Submittal Table 3.1 |
| RCS letdown flow rate, gpm | 60+10% |
| RCS letdown demineralizer efficiency | 100 |
| RCS leakage, gpm | 11 |
| Co-incident spike multiplier | 500 |
| Iodine spike duration, hrs | 8 |
| Chemical form release fractions | |
| Elemental | 0.97 |
| Organic | 0.03 |
| Primary-to-secondary leakage per SG, gpm | 1 |
| Duration, hours | 8 |
| Liquid Masses, gm | |
| RCS | 1.247E+8 |
| SG (each) | 5.817E+7 |
| Steam release from faulted SG, lbm | |
| 0 to 610 seconds | 128,237 |
| 610 seconds to 8 hours | 0 |
| Steam release from intact SG, lbm | |
| 0 to 610 seconds | 37,780 |
| 610 seconds to 8 hours | 755,097 |
| Steam iodine partition coefficient in SGs | |
| Faulted SG (elemental and organic) | 1.0 |
| Unaffected SG | |
| Elemental | 0.01 |
| organic | 1.0 |
| Release points | Turbine building main steam header |

**Table 7**
**Parameters and Assumptions**
**Used in**
**Radiological Consequence Calculations**
**for**
**Steam Generator Tube Rupture Accident**

| Parameter | Value |
|---|---|
| Pre-incident iodine spike activity (60 µCi/gm dose equivalent I-131) | Submittal Table 3.1 |
| Coincident spike appearance rate, based on | Submittal Table 3.1 |
| RCS letdown flow rate, gpm | 60+10% |
| RCS letdown demineralizer efficiency | 100 |
| RCS leakage, gpm | 11 |
| Co-incident spike multiplier | 335 |
| Iodine spike duration, hrs | 8 |
| Chemical form release fractions | |
| Elemental | 0.97 |
| Organic | 0.03 |
| Primary-to-secondary leakage to intact SG, gpd | 150 |
| Duration, hours | 8 |
| Liquid Masses, gm | |
| RCS | 1.247E+8 |
| SG (each) | 5.817E+7 |
| Steam release from faulted SG, lbm | |
| 0 to 49 seconds (to condenser) | 45,500 |
| 49 to 3492 seconds (to atmosphere) | 62,400 |
| Steam generator rupture flow, lbm | |
| 0 to 49 seconds | 2,900 |
| 49 to 3492 seconds | 107,400 |
| Steam release from intact SG, lbm | |
| 0 to 49 seconds (to condenser) | 45,200 |
| 49 to 3492 seconds (to atmosphere) | 60,000 |
| 3492 seconds 2 hours (to atmosphere) | 147,500 |
| 2 to 8 hours (to atmosphere) | 459,900 |
| Steam generator iodine partition coefficients | |
| elemental and organic | 1.0 |
| Release points | Atmosphere relief valves |

**Table 8**
**Parameters and Assumptions**
**Used in**
**Radiological Consequence Calculations**
**for**
**Locked Rotor Accident**

| Parameter | Value | |
|---|---|---|
| Fraction of failed fuel | 0.50 | |
| Fraction of Core Inventory in Gap | | |
| Kr-85 | 0.10 | |
| I-131 | 0.08 | |
| Alkali metals | 0.12 | |
| Other noble gases / iodines | 0.05 | |
| Iodine species fraction | Containment | SG |
| Aerosol | 0.95 | 0 |
| Elemental | 0.0485 | 0.97 |
| Organic | 0.0015 | 0.03 |
| Primary to secondary leakage per SG, gpd | 500 | |
| Primary to secondary leakage duration, hours | 8 | |
| Steam generator mass in 2 SGs, gm | 8.5E+7 | |
| Steam partition coefficient in SGs | 0.01 | |
| Steam release from 2 SGs, lbm | | |
| 0-10 minutes | 54,620 | |
| 10 to 30 minutes | 14,446 | |
| 0.5 to8 hours | 685,229 | |
| Release point | Atmospheric relief valves | |

**Table 9**
**Parameters and Assumptions**
**Used in**
**Radiological Consequence Calculations**
**for**
**Control Rod Ejection Accident**

| Parameter | Value | |
|---|---|---|
| Radial peaking factor | 1.65 | |
| Fraction of rods that exceed DNB | 0.15 | |
| Gap fraction, all nuclide groups | 0.10 | |
| Fraction of rods in core that exceed DNB | 0.00375 | |
| Melt isotopic composition | | |
| | CNMT | SG |
| Noble gases | 1.0 | 1.0 |
| Iodine | 0.25 | 0.5 |
| Iodine species fraction | CNMT | SG |
| Particulate/aerosol | 0.95 | 0 |
| Elemental | 0.0485 | 0.97 |
| Organic | 0.0015 | 0.03 |
| Containment free volume, $ft^3$ | 1.0E6 | |
| Containment Sprays | Not credited | |
| Containment release | | |
| 0-24 hours, %/day | 0.2 | |
| 24-720 hours, %/day | 0.1 | |
| Containment Particulate deposition 1/hr | 0.023 | |
| Duration of release, days | 30 | |
| Containment fan cooler iodine removal efficiencies | | |
| Aerosols | 95 | |
| Elemental/organic | 0 | |
| Primary to secondary leakage per SG, gpd | 1.0 | |
| Primary to secondary leakage duration, hours | 8 | |
| Steam generator mass for 2 Sgs, lbm | 8.5E+7 | |
| Steam partition coefficient in SGs | 0.01 | |
| Steam release from 2 SGs, lbm | | |
| 0-10 minutes | 2.478E+6 | |
| 10 to 30 minutes | 3.276E+5 | |
| 0.5 to8 hours | 6.907E5 | |
| Release point | | |
| Containment leakage | Containment | |
| Secondary | Atmospheric relief valves | |

**Table 10**
**Parameters and Assumptions**
**Used in**
**Radiological Consequence Calculations**
**for**
**Tornado Missile in Spent Fuel Pool**

| Parameter | Value |
|---|---|
| Number of damaged fuel assemblies | |
| Hot | 5 |
| Cold | 4 |
| Decay times | |
| Hot | 100 hours |
| Cold | 60 days |
| Fraction of Core Inventory in Gap | |
| Kr-85 | 0.10 |
| I-131 | 0.08 |
| Alkali metals | 0.12 |
| Other noble gases / iodines | 0.05 |
| Iodine species above water | |
| Elemental | 0.57 |
| Organic | 0.43 |
| Overall pool DF | 200 |
| Iodine removal filter efficiencies for all forms | 0 |