January 23, 2001

MEMORANDUM TO: Patrick S. Sekerak, Senior Project Manager
Project Directorate IV
Division of Licensing and Project Management

FROM: Mark A. Caruso, Acting Chief ***/RA/***
Licensing Section
Probabilistic Safety Assessment Branch
Division of Systems Safety and Analysis

SUBJECT: SAFETY EVALUATION FOR PROPOSED CHANGES TO THE GRAND GULF TECHNICAL SPECIFICATIONS IMPLEMENTING ALTERNATIVE SOURCE TERM FOR GRAND GULF NUCLEAR STATION (TAC NO. MA8065)

In response to your request, the Probabilistic Safety Assessment Branch (SPSB) in the Division of Systems Safety and Analysis (DSSA) has completed its review of proposed changes to the Grand Gulf Technical Specifications (TSs) implementing the alternative source term for Grand Gulf Nuclear Station. Our safety evaluation is attached.

On the basis of our review of the licensee's analysis and our own confirmatory assessment of the radiological consequences of the postulated design-basis accidents, we find that the proposed changes to the Grand Gulf TSs are acceptable

This review was performed by Krzysztof Parczewski of Materials and Chemical Engineering Branch in the Division of Engineering (Section 2.1.4) and Jay Lee of SPSB/DSSA (remaining sections). The Oak Ridge National Laboratory, under NRR Technical Assistance Contract JCN J-2604, assisted us in review of post-accident containment water chemistry proposed by the licensee.

Docket No. 50-416

Attachment:
Safety Evaluation

CONTACT: Jay Lee, SPSB/DSSA
415-1080

January 23, 2001

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Attachment: Safety Evaluation

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<u>SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION</u>

FACILITY OPERATING LICENSE NO NPF-29

ENTERGY OPERATIONS, INC.

GRAND GULF NUCLEAR STATION

DOCKET NO. 50-416

1 INTRODUCTION

By letter dated January 21, 2000, supplemented by submittals dated June 29, 2000, September 1, 2000, October 26, 2000, and December 22, 2000, Entergy Operations, Inc. (the licensee), requested an amendment to Facility Operating License No. NPF-29 for the Grand Gulf Nuclear Station. The licensee submitted this license amendment as a full-scope implementation of the alternative source term (AST), as described in Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design-Basis Accidents at Nuclear Power Reactors," and 10 CFR Part 50.67, "Accident Source Term." For this license amendment request implementing the full-scope AST, the licensee reanalyzed and submitted the radiological consequences for three design-basis accidents (DBAs): the loss-of-coolant accident (LOCA), the fuel-handling accident (FHA), and the control rod drop accident (CRDA).

As the lead pilot plant application for implementing an AST at operating nuclear power plants. the licensee submitted this amendment with the endorsement of the Nuclear Energy Institute. In 1998, the staff used the Grand Gulf station as a rebaselining plant for evaluating the impact of implementing the AST at operating nuclear plants as described in SECY 96-242, "Use of the NUREG-1465 Source Term at Operating Reactors." The results and findings from the rebaselining study for implementing the AST at operating reactors (the Grand Gulf, Surry, and

Attachment

Zion stations) were provided in SECY 98-154, "Results of the Revised (NUREG-1465) Source Term Rebaselining for Operating Reactors."

Specifically, the licensee requested that:

1. Technical Specification (TS) 1.1, “Definitions”, be amended to reference new dose conversion factors and to increase the maximum allowable primary containment leakage rate from 0.437 percent to 0.682 percent of primary containment air weight per day.

2. TS Bases 3.3.6.1, 3.3.6.2, 3.6.1.3, 3.6.4.1, 3.6.4.2, and 3.6.4.3 be revised to reflect the new interpretation of the term “recently irradiated fuel assemblies” as fuel which has been irradiated in the reactor within the previous 24 hours instead of within 8 days.

3. TS 3.3.7.1, “Control Room Fresh Air System Instrumentation,” be amended to delete all automatic control room isolation features from the scope of the Grand Gulf TSs. This revision would delete all automatic isolation input signals from Table 3.3.7.1-1, delete Surveillance Requirements 3.3.7.1.1 through 3.3.7.1.5, and revise Limiting Condition for Operation (LCO) 3.3.7.1 and its associated ACTION items. The only safety function required of the control room fresh air (CRFA) system would be manual control room isolation.

4. Surveillance Requirement (SR) 3.6.1.3.8, “Main Steam Isolation Valve (MSIV) Leakage Rate,” be amended to increase the maximum allowable leak rate to less than or equal to 100 standard cubic feet per hour (scfh) per main steam line (MSL) with a total leak rate through all four MSLs of less than or equal to 250 scfh (from less than or equal to 100 scfh through all four MSLs)

5. SR 3.6.4.1.3, “Secondary Containment Drawdown,” be amended to increase the maximum allowable drawdown time from 120 seconds to 180 seconds.
6. TS 3.7.3, “Control Room Fresh Air System,” and its corresponding BASES Section 3.7.3 be amended to address the sole isolation function of the CRFA system and the removal

of radioactive aerosol by high-efficiency particulate air (HEPA) filters in the CRFA system. No credit for iodine removal by the CRFA system charcoal adsorbers would be taken for the purpose of the DBA radiological consequence analyses. This requested amendment would also delete the fuel movement and core alteration periods from the Applicability based on the revised fuel-handling accident analysis. The licensee stated that neither the CRFA system isolation nor filtration by the charcoal adsorbers or HEPA filters would be needed to meet the relevant dose criteria.

7. The Applicability Statement of TS 3.7.4, “Control Room Air Conditioning system,” be amended to no longer require the LCO to be met during fuel movement or CORE ALTERATIONS. The Conditions and Required Actions in the ACTIONS would also be modified accordingly.

8. TS 3.8.2, 3.8.5, and 3.8.8 be amended to modify the Applicability Statements of these LCOs to read “when handling *recently* irradiated fuel assemblies” instead of “when handling irradiated fuel assemblies.” For the current fuel cycle, the licensee proposed this term to be defined as those fuel assemblies that have been in a critical reactor core within the previous 24 hours. The Required Actions in the ACTIONS would be also modified accordingly.

In addition, the licensee requested to amend Grand Gulf License Condition 2.C (38), “Control Room Leak Rate,” to increase the maximum allowable control room leak rate during Modes 1, 2, and 3 from 590 to 2010 cubic feet per minute (cfm).

2 EVALUATION

2.1 Loss-of-Coolant Accident

The current radiological consequence analysis for the postulated LOCA using Technical Information Document (TID)-14844 source term is provided in Grand Gulf Updated Final Safety Analysis Report (UFSAR) Section 15.6. To demonstrate that the Grand Gulf engineered safety features (ESFs) designed to mitigate the radiological consequences will remain adequate after this license amendment, the licensee reevaluated the offsite and control room radiological consequences of the postulated LOCA. The licensee has implemented the AST in this reevaluation. The licensee submitted the results of its offsite and control room dose calculations (see Table 1). In addition, the licensee provided a complete dose analysis and described the major assumptions and parameters used in its dose calculations, and the fission product transport, removal, and release models developed by the licensee. As documented In the submittals, the licensee has determined that after the implementation of the AST, the existing ESF systems at Grand Gulf will still provide assurance that the total radiological consequences of the postulated LOCA at the exclusion area boundary (EAB), in the low population zone (LPZ), and in the control room meet the radiation dose criteria specified in 10 CFR Part 50.67.

The staff has reviewed the licensee's analysis and has performed an independent confirmatory radiological consequence dose calculation for the following three potential fission product release pathways after the postulated LOCA:

(1) main steam isolation valve (MSIV) leakage
(2) containment leakage
(3) post-LOCA leakage from ESF systems outside containment

The results of the staff's independent radiological consequence calculations are given in Table 1, alongside the licensee's results. The major parameters and assumptions used by the staff are listed in Tables 2 through 4.

2.1.1 Main Steam Isolation Valve Leakage Pathway

The Grand Gulf MSIV leakage control system (LCS) is composed of independent inboard and outboard systems. The inboard system evacuates the leakage in the volume between the inboard and outboard MSIVs while the outboard system draws the leakage from the volume downstream of the outboard MSIVs. The licensee assumed that the MSIV-LCS is manually actuated within 20 minutes of the postulated LOCA. This assumption is consistent with Regulatory Guide 1.96, "Design of Main Steam Isolation Valve Leakage Control Systems for Boiling Water Reactor Nuclear Power Plants." There are four main steam lines (MSLs) at Grand Gulf. Each MSL has an inboard MSIV and an outboard MSIV. These valves isolate the reactor coolant system in the event of a break in a steam line outside the primary containment, a design basis LOCA, or an other event requiring containment isolation. Although the MSIVs are designed to provide a leak-tight barrier, it is recognized that some leakage occurs through these valves. The current Grand Gulf TS limit for MSIV leakage is less than or equal to 100 scfh through all four MSLs. The licensee requested to increase the maximum allowable leak rate to less than or equal to 100 scfh per MSL with a total leak rate through all four MSLs of less than or equal to 250 scfh

The staff assumed in its analysis that one of the two MSIVs in one MSL fails to close and all allowed leakage (100 scfh) is released directly to the environment through the remaining MSIV in this MSL for the first 20 minutes after the accident. No credit is provided for fission product deposition or holdup for decay in this steam line. Leakage on the remaining three MSLs (150 scfh) is directed into a volume between the closed inboard and outboard valves for the first 20 minutes. After 20 minutes, all MSIV leakage (250 scfh) is collected by the MSIV-

LCS and discharged to the secondary containment before release to the environment through the standby gas treatment system.

2.1.1.1 Fission Product Transport in Drywell

The licensee assumed, and the staff agrees, that a large-break LOCA as a result of a double-ended guillotine pipe rupture in a recirculation suction line would be the most limiting LOCA with respect to the offsite and control room radiological consequences. The break releases reactor coolant to the drywell. No water injection from the emergency core cooling system (ECCS) is assumed and the reactor water level drops below the core, exposing the reactor fuel. In Grand Gulf License Amendment No. 143, issued on March 22, 2000, “Implementation of Alternative Source Term Limited Scope Application for the Timing of the Onset of Gap Activity Release,” the staff evaluated the earliest time of fission products release (fuel gap activity release) from perforated fuel rods following a postulated LOCA and concluded that the minimum time would be no earlier than 120 seconds. The staff also assumed that all fission products are released directly to the drywell and leaked into the primary containment and into the main steam lines, bypassing the suppression pool. The staff concludes that these assumptions are appropriate for the large-break LOCA. For small-break LOCAs with operator actuation of an automatic depressurization system (ADS), most of the fission products would be released into the drywell through the pipe break and into the suppression pool through the ADS, where the fission products are removed.

As characterized in NUREG-1465, “Accident Source Terms for Light-Water Nuclear Power Plants,” the gap and early in-vessel fission product releases terminate 2 hours after the postulated LOCA initiation. The staff assumed (as it did for Perry Nuclear Station License Amendment No.103, issued on March 26, 1999, “Main Steam Line Leakage Requirements and Elimination of the Main Steam Isolation Valve Leakage Control system Implementing the Alternative Source Term”) that the fission products are homogeneously distributed between the drywell and the primary containment 2 hours after accident initiation (both Grand Gulf and Perry containments are Mark III type design). This would require reflooding of the reactor vessel. Instead of trying to justify an all encompassing steaming rate due to this reflooding, the staff concludes that a substantial amount of fission products may end up in the primary containment as well as the drywell and that mitigative features such as the standby gas treatment system need to be designed to accommodate a significant portion of the source term. For most of the risk significant cases, such as station blackout and transients, all the fission products are released directly to the primary containment via the safety relief valves. Waiting 2 hours to homogeneously mix the source term is acceptable for achieving an appropriate balance because the worst 2 hours are considered not the first 2 hours used with the TID-14844 source term.

Confirmatory calculations performed by the staff showed that the radiological consequences are dependent upon the drywell bypass leakage prior to the termination of fission product release at 2 hours. Because of this sensitivity, the staff concludes, as it did for Perry, that without relocation to the lower head, the steaming rate of an intact core on the order of 3,000 cfm should be assumed for drywell bypass leakage. The staff's steaming rate prior to 2 hours is conservative in that it does not credit steaming due to relocation, cooling from alternative water sources, or the release of hydrogen gas.

The 3,000 cfm drywell bypass leakage rate is based upon large-break LOCA analyses performed with MELCOR on a Grand Gulf type model. These analyses showed no relocation below the core plate, water level below the core plate, and an average steaming rate of approximately 2,800 cfm prior to quenching of the core at approximately 0.5 hours. Also, alternative water sources, such as the standby liquid control system, would not be available during station blackout sequences which comprise 96% of the core damage frequency for

Grand Gulf. Therefore, the staff concludes the use of 3,000 cfm for the drywell bypass leakage prior to 2 hours is reasonable.

2.1.1.2 Aerosol Deposition Within The Drywell

In its evaluation, the staff used a simplified model developed by the staff's contractor for estimating the fission product aerosol deposition by natural processes in the drywell of BWRs following a postulated LOCA. The model is described in NUREG/CR-6189, "A Simplified Model of Aerosol Removal by Natural Processes in Reactor Containment." This model was derived by correlating the results of Monte Carlo uncertainty sampling analyses and assessing uncertainties in aerosol properties, drywell geometries, accident progression, and aerosol behavior expected to be associated with a postulated LOCA in the drywell.

The staff assumed that the fission product aerosols in the drywell are removed by natural processes (gravitational sedimentation and phoretic phenomena such as diffusiophoresis and thermophoresis). The staff assumed that the drywell is well mixed during the entire duration of the accident. The aerosol removal rates used by the staff represent the 90th percentile of the uncertainty distributions (see Table 2). For the main steam lines, the licensee did not request and the staff has not provided any credit for aerosol deposition.

2.1.2 Containment Leakage Pathway

The primary containment consists of a drywell, a wetwell, and supporting systems to limit fission product leakage during and following the postulated LOCA with rapid isolation of the containment boundary penetrations. The current maximum allowable primary containment leakage rate (L_a), is 0.437 percent of primary containment air weight per day. This value is based on 0.35 percent per day from the containment leak and an additional 100 scfh (0.087 percent per day) through the steam lines. In this amendment request, the licensee proposed (and the staff used in its evaluation) 0.682 percent per day based on 0.385 percent per day from the containment leak and an additional 250 scfh (0.297 percent per day) through the steam lines.

The staff stated in Regulatory Guide 1.183, “Alternative Radiological source Terms for Evaluating Design-Basis Accidents at nuclear Power Reactors,” that these leak rates may be reduced 24 hours into the postulated LOCA, if supported by plant configuration and analyses, to a value not less than 50 percent of the TS leak rate limit. The licensee provided the design- basis drywell and containment pressure profiles for Grand Gulf with adjustments made to the additional hydrogen introduced into the drywell and containment, and the additional heat load associated with increased metal-water reaction and hydrogen ignition. The staff reviewed the licensee’s submittal and accepted the 50 percent reduction in the drywell and containment leak rate after 24 hours.

The Grand Gulf secondary containment (which surrounds the primary containment) will collect and retain any fission product leakage from the primary containment and will release fission products to the environment through the standby gas treatment system (SGTS) following the postulated LOCA. During normal plant operation, the containment is maintained at a slight negative pressure at a vacuum of 0.4-inch water gauge. The Grand Gulf UFSAR states that the secondary containment pressure is expected to remain negative following the postulated LOCA. However, for a short period (3 minutes), it may not be maintained below the negative pressure of 0.25-inch water gauge. Therefore, the licensee assumed, and the staff agrees, that the entire primary containment leakage is released directly to the environment during the first 3 minutes of the postulated LOCA. After 3 minutes, the SGTS draws 4000 cfm of secondary containment atmosphere air through a HEPA filter with a 99 percent aerosol removal efficiency and a charcoal adsorber with a 99 percent iodine removal efficiency before release to the environment.

2.1.2.1 <u>Standby Gas Treatment System</u>

The SGTS is an engineered safety features system and is designed to collect, process, and release the fission product leakage from the primary containment into the shield building. The SGTS is a redundant system consisting of two 100 percent capacity subsystems. Each subsystem has a design capacity of 4000 cfm and consists of, among other things, a pre-HEPA filter, a 4-inch deep charcoal adsorber, and a post-HEPA filter. The system is designed to Seismic Category 1 standards and is located in a Seismic Category 1 structure.

2.1.2.2 <u>Containment Spray</u>

The containment spray system (CSS) is an engineered safety features system and is designed to provide containment cooling and fission product removal in the containment following the postulated LOCA. The CSS consists of two redundant and independent loops. Each loop has a design spray water flow capacity of 5650 gpm. The system is designed to Seismic Category 1 standards and is located in a Seismic Category 1 structure. No chemical additives are used in the CSS (see Section 2.1.4).

Before the CSS is activated (30 minutes after the accident), the licensee assumed that a mixing rate of 2 unsprayed volumes per hour between the sprayed and unsprayed portions of the containment atmosphere consistent with the guidance provided in Standard Review Plan (SRP) Section 6.5.2. During the CSS operation, the licensee assumed a mixing rate of 7.5 unsprayed volumes per hour between the sprayed and unsprayed portions of the containment atmosphere with 70,000 cfm of dry air flow. The proposed mixing rate is higher than that specified in SRP Section 6.5.2. The staff accepted the proposed higher mixing rate during the CSS operation on the basis of its review of the licensee's calculation that demonstrated that an adequate mixing flow existed between unsprayed and sprayed regions by natural convection. The spray condenses steam in the containment atmosphere and the movement of the condensation creates additional mixing of the containment air.

In its evaluation, the staff used a simplified model for estimating the fission product aerosol removal by containment sprays following a postulated LOCA. The model is described in NUREG/CR-5966, "A Simplified Model of Aerosol Removal by Containment Sprays". This model was derived by correlating the results of Monte Carlo uncertainty sampling analyses assessing the uncertainties in aerosol properties, aerosol behavior, spray droplet behavior, and the initial and boundary conditions expected to be associated with a postulated LOCA in the containment. Two parameters used in this evaluation are not treated as uncertainty distributions for Grand Gulf: spray water flux and spray fall height. These parameters are plant specific and their values are listed in Table 3. The staff used 90th percentile uncertainty distributions for fission product in aerosol form in calculating the radiological consequences. The major parameters used, including spray removal rates for elemental iodine in the sprayed region, are listed in Table 3.

2.1.3 Post-LOCA Leakage Pathway From Engineered Safety Features Outside Containment

Any leakage water from ESF components located outside the primary containment releases fission products during the recirculating phase of long-term core cooling following a postulated LOCA. The licensee calculated this leakage to be less than 1.02 gallons per minute (gpm) and assumed ESF leakage to begin 10 minutes into the postulated LOCA through the entire duration of the accident (30 days). The staff finds the leakage value calculated by the licensee and the leakage initiation time to be reasonable The licensee also assumed that ten percent of all forms of iodine contained in the leakage is released directly to the environment consistent with Regulatory Guide 1.183.

2.1.4 <u>Post-Accident Containment Water Chemistry Management</u>

In NUREG-1465, the staff concluded that iodine entering the containment from the reactor coolant system during an accident would be composed of at least 95 percent cesium iodide (CsI), with no more than 5 percent of iodine (I) and hydriodic acid (HI). The licensee conservatively assumed that all 5 percent of this release is in the form of HI in order to maximize the acid generation. Once in the containment, highly soluble cesium iodide will readily dissolve in water pools forming iodide (I^-) in solution. The radiation-induced conversion of iodide in water into elemental iodine (I_2) is strongly dependent on the pH. The staff stated in the NUREG that without pH control, a large fraction of iodine dissolved in water pools in ionic form will be converted to elemental iodine and will be reevolved into the containment atmosphere if the pH is less than 7. On the other hand, if the pH is maintained above 7, very little (less than 1 percent) of the dissolved iodine will converted to elemental iodine.

The licensee developed and submitted the methodologies used to calculate the post-accident suppression pool water pH transient for determining long-term iodine reevolution from the pool water into the containment atmosphere. In addition, the licensee provided a complete post-accident suppression pool water pH analysis. The staff finds that the licensee followed closely the models and methodologies described in NUREG/CR-5950, “Iodine Evolution and pH Control,” NUREG/CR-5732, “Iodine Chemical Forms in LWR Severe Accidents,” and NUREG-1465 for determining (1) the formation of hydrochloric and nitric acids, (2) the suppression pool pH transient, and (3) long-term iodine reevolution. The staff previously used and has accepted the methods in NUREG/CR-5950 and NUREG/CR-5732. The staff finds the licensee’s methods used to calculate post-accident suppression pool water pH transient for determining long-term iodine reevolution to be acceptable.

Using its methodologies, the licensee has determined that the suppression pool water pH rises steadily during fuel gap and in-vessel releases from the initial pH value of 5.3 to above 7 due to the introduction of cesium hydroxide (CsOH) into the pool. The pH then begins to decrease after the in-vessel release terminates due to the continued formation of nitric acid in the suppression pool and the formation of hydrochloric acid from radiolytic decomposition of the electrical cable jacketing. After approximately 4 days, the suppression pool water pH value decreases to less than 7.

The staff believes that, for the first 24 hours into the postulated LOCA, the fission product source term behavior, its transport, and release to the environment will be entirely dominated by thermal hydraulic conditions in the drywell and in the containment (drywell leakage, steam production and condensation, and mixing), and by aerosol removal mechanisms (containment spray and aerosol deposition) independent of suppression pool water pH and iodine reevolution from the suppression pool to the containment atmosphere. Consequently, any postulated radiological consequences at any point on the boundary of the exclusion area for a 24 hour period will not be affected by iodine reevolution and pH control.

The licensee stated that in the event of an unmitigated LOCA such as the staff postulated to occur, the Grand Gulf Severe Accident Procedures direct the plant operator to inject the standby liquid control system(SLCS) solution into the reactor vessel in the early stages of the accident for both vessel inventory and re-criticality protection when the core is re-flooded. The SLCS is a safety-related system and designed to Seismic Category 1 standards. It is designed as a reactivity control system and provides backup capability so as to be able to shut down the reactor if the normal control becomes inoperable. The Grand Gulf TS requires the system to be maintained in an operable status whenever the reactor is critical. The system is manually initiated from the main control room to add a boron neutron absorber solution

(sodium

pentaborate) to the reactor vessel. The SLCS contains at least 5800 pounds of sodium pentaborate. Sodium pentaborate dissolves in water, producing boric acid and sodium borate:

$$Na_2B_{10}O_{16} + 15\ H_2O \rightarrow 2\ Na^+ + 2\ H_2BO_3^- + 8\ H_3BO_3$$

Since boric acid is a relatively weak acid and sodium hydroxide is a strong base, their solution has a buffering effect and will maintain the pH of the suppression pool water at the pH values higher than 7. The sodium pentaborate solution will be well mixed with the suppression pool water by the end of 24 hour period as a result of reflooding the reactor vessel. The licensee stated that as a backup, 5000 pounds each of anhydrous borax and boric acid would be mixed in the condensate storage tank in accordance with Grand Gulf Emergency Procedure. This will produce approximately additional 10,000 pounds of sodium pentaborate. The licensee stated, and the staff agrees, that sodium pentaborate from the SLCS is capable of controlling and maintaining long-term suppression pool water pH levels at 7 or above from the first 24 hours through the entire 30-day period of the accident.

2.1.5 Control Room Habitability

The Grand gulf control room is normally maintained at a slightly positive pressure to prevent the introduction of air into the control room from sources other than the 2000 cfm outdoor air makeup flow. The licensee proposed to manually isolate the control room air intakes no later than 20 minutes after the initiation of the postulated LOCA. The normal control room air intake rate of 2000 cfm (without filtration through the control room fresh air (CRFA) system) is assumed for the first 20 minutes, in addition to 10 cfm from ingress/egress into the control room. Once the air intakes are isolated, the control room atmosphere is recirculated through the control room air conditioning units and 4000 cfm of this flow is processed through the CRFA system. The CRFA system is a redundant system. Each subsystem consists of, among other things, a pre-filter, a high-efficiency particulate air filter, a charcoal adsorber, and a post-HEPA filter. The staff assumed a removal efficiency of 99 percent for fission products in particulate form for the HEPA filter. The licensee has not requested and the staff has not provided any iodine removal credit for the charcoal adsorber. After 3 days, the CRFA system draws outside air at 4000 cfm through HEPA filters into the control room envelope and the recirculation flow terminates. The staff assumed an unfitered air inleakage rate of 2010 cfm into the control room during the entire 30-day accident period while the control room is isolated.

The licensee reevaluated the control room habitability with the application of the AST and concluded that the radiological consequences to the control room operator resulting from the postulated LOCA are within the 5 rem total effective dose equivalent (TEDE) criterion specified in 10 CFR Part 50.67. The licensee reached this conclusion.

(1) using the revised atmospheric relative concentrations at the control room air intake,
(2) with manual isolation of control room at 20 minutes after the initiation of the postulated LOCA,
(3) with an unfiltered air inleakage of up to 2010 cfm into the control room during the entire period of the accident while the control room is isolated, and
(4) taking no credit for iodine removal by the CRFA system.

To verify the licensee's radiological consequence assessments, the staff performed confirmatory radiological consequence dose calculations for the control room operator. The results are given in Table 1, alongside the licensee's results. The staff calculated a TEDE of 3.7 rem to the control room operator from the postulated LOCA. This is within the 5 rem TEDE dose criterion specified in 10 CFR Part 50.67. In addition, the staff's sensitivity study indicates that the Grand Gulf control room can take in approximately 3500 cfm of unfitered air inleakge and still meet the relevant dose criterion.

Therefore, the staff concludes that adequate radiation protection is provided to the control room operator to permit access to and occupy the control room under accident conditions without personnel receiving radiation exposures exceeding a TEDE of 5 rem dose criterion specified in 10 CFR Part 50.67. Therefore, the staff finds that the control room habitability assessment performed by the licensee is acceptable.

2.1.6 Resulting radiological consequences from the postulated LOCA

The staff calculated a TEDE of 9.0 rem at the EAB, a TEDE of 3.8 rem in the LPZ, and a TEDE of 3.7 rem in the control room resulting from the postulated LOCA. As shown in Table 1, these calculated dose values are all within the dose criteria specified in 10 CFR Part 50.69. Therefore, the staff concludes that the requested license amendment for the postulated LOCA is acceptable.

2.2 Fuel-Handling Accident

The current radiological consequence analysis of the design basis fuel-handling accident (FHA) using the TID-14844 source term is provided in the Grand Gulf UFSAR Section 15.4.9. In this license amendment request, the licensee reanalyzed the FHA with the implementation of the AST and presented its results in Attachment 6 of the submittal.

In Grand Gulf License Amendment No. 139, issued on October 20, 1999, "Operational Conditions for Handling Irradiated fuel in the Primary and Secondary Containments," the licensee defined the new term "recently irradiated fuel" as irradiated fuel assemblies that contain sufficient fission products to require the operability of ESF systems to meet the relevant offsite and control room operator dose criteria. This term is a plant-specific parameter and is evaluated for each fuel cycle by the licensee. In Amendment No. 139, the licensee and the staff determined that, using the TID-14844 source term, an 8-day decay of irradiated fuel assemblies is sufficient to assure that the radiological consequences of a fuel-handling accident will be within the relevant dose acceptance criteria specified in the SRP Section15.4 and General Design Criterion 19.

In this proposed amendment, the licensee requested that this new term be defined as 24 hours decay period for the current fuel cycle with the application of the AST. The licensee concluded that this decay period will be sufficient to meet the TEDE dose criterion in the SRP Section 15.0.1 and the criterion specified in 10 CFR Part 50.67. The licensee reached this conclusion assuming that:

(1) after 24 hours of fission product decay, irradiated fuel assemblies are moved without secondary containment integrity,
(2) core alterations are performed without secondary containment integrity,
(3) the control room is not isolated, and
(4) the CRFA system is not available to remove airborne fission products.

To verify the licensee's radiological consequence assessments, the staff performed confirmatory radiological consequence dose calculations for the following most limiting fuel drop scenario that will produce the most radiological consequence. The scenario involves the drop of an irradiated fuel assembly onto the core without secondary containment after 24 hours of fission product decay in the fuel. The results of the staff's independent radiological consequence calculation are provided in Table 1, along with those results calculated and provided by the licensee. The major parameters and assumptions used by the staff in the radiological consequence calculations are listed in Table 5. The staff calculated a TEDE of 1.8 rem at the EAB and a TEDE of 1.9 rem in the control room. The radiological consequences at the EAB and in the control room calculated by the staff and the licensee are well within the dose criterion specified in 10 CFR Part 50.67 (5 rem TEDE in the control room), and meet the dose acceptance criterion specified in the SRP Section 15.0.1 (a 6.3 rem TEDE at EAB). Therefore, the staff finds the license amendment requested by the licensee to change Grand Gulf TS Sections 3.7.4, 3.8.2, 3.8.5, and 3.8.8 to be acceptable.

2.3 Control Rod Drop Accident

The current radiological consequence analysis of the design-basis control rod drop accident using the TID-14844 source term is provided in Grand Gulf UFSAR Section 15.4.9. In this license amendment request, the licensee reanalyzed the CRDA with the application of the AST and presented its results in Attachment 4 of the submittal. The staff reviewed the licensee's accident source term, fission product transport, and removal models, the release pathways, and the major assumptions and parameters used to calculate the radiological consequences. The staff finds that they are acceptable.

To verify the licensee's radiological consequence calculations, the staff performed a confirmatory radiological consequence dose calculation. The results are provided in Table 1, alongside with the licensee's results. The major parameters and assumptions used by the staff in the radiological consequence calculations are listed in Table 6. The staff calculated a TEDE of all less than 1 rem each at the EAB, in the LPZ, and in the control room. The radiological consequences calculated by the staff and the licensee are well within the dose criterion specified in 10 CFR Part 50.67 (5 rem TEDE in the control room), and meet the dose acceptance criterion specified in the SRP Section 15.0.1(6.3 rem TEDE at EAB and in the LPZ). Therefore, the staff finds the license's reanalysis using the AST to be acceptable.

2.4 Atmospheric Relative Concentration Estimates

The licensee used five years of onsite meteorological data collected during calendar years 1995 through 1999 to estimate the atmospheric relative concentration (X/Q) values used in the dose assessments described above. These X/Q values are listed in Tables 5 through 7.

Meteorological data were measured at 10 and 50 meters above grade. Joint recovery of wind speed, wind direction and atmospheric stability data during the entire five year period met the recommended minimum of 90 percent cited in Regulatory Guide 1.23, “Onsite Meteorological Programs,” although data recovery was slightly less than this recommended recovery rate during 1995 and 1998. The licensee examined the data and performed year-to-year comparisons to confirm the overall quality of the data.

The licensee used the PAVAN computer code to calculate X/Q values for the EAB and LPZ. This methodology is described in Regulatory Guide 1.145, “Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants.” As input to this calculation, the licensee sorted the five years of meteorological data to approximate a single year of data in joint frequency distribution form. Staff made confirmatory estimates using the five years of data in hourly form. Resultant staff X/Q estimates were somewhat higher for the EAB and LPZ for the 0-2 hour time period and lower for the LPZ for the longer time periods. The licensee performed additional calculations to demonstrate that the X/Q values used in their dose assessment were bounded by their prior X/Q calculations. The staff has concluded that the difference between the licensee and staff X/Q estimates does not affect the staff’s conclusion that the resultant dose estimates in support of this amendment meet regulatory requirements.

The licensee used the ARCON96 methodology to calculate X/Q values for control room dose assessment. ARCON96 is described in NUREG/CR-6331, Revision 1, “Atmospheric Relative Concentrations in Building Wake.” Estimates were based on the five-year period of meteorological data in hourly form. All postulated releases were calculated as ground level point releases assuming no release flow. Where appropriate, estimates were made from each postulated release location assuming possible intake at either the Safeguard Switchgear and Battery Ventilation Intake or the Control Room Roof Intake. The higher X/Q value for each time period was then used in the dose assessment. The fuel handling accident control room X/Q value was calculated as an effective X/Q by estimating a weighted average of the Containment Vent and Standby Gas Treatment System X/Q values. The staff concludes that the licensee's control room X/Q estimates are acceptable.

3 CONCLUSION

The staff has reviewed the licensee's analysis and performed a confirmatory assessment of the radiological consequence of the postulated LOCA, FHA, and CRDA. The doses calculated by the staff and the licensee are listed in Table 1. The doses are all within relevant dose criteria specified in 10 CFR Part 50.67 and the SRP. Therefore, the staff concludes that the radiological consequences analyzed and submitted by the licensee are acceptable. On the basis of this evaluation, the staff further concludes that the license amendment requested by the licensee is acceptable.

Table 1

Radiological Consequences Expressed as TEDE

(rem)

Design-Basis Accidents	EAB[1]		LPZ[2]		Control Room	
	NRC	Grand Gulf	NRC	Grand Gulf	NRC	Grand Gulf
LOCA	9.0	8.779	3.8	4.601	3.7	3.647
Dose criteria[3]	25		25		5.0	
Fuel handing accident	1.8	1.982	NA[4]	NA[4]	1.9	2.035
Dose criteria	6.3[5]				5.0[3]	
Control rod drop accident	0.1	0.147	<0.1	0.064	0.26	0.26
Dose criteria	6.3[5]		6.3[5]		5.0[3]	

(1) Exclusion area boundary,
(2) Low population zone
(3) 10 CFR 50.67
(4) Not Applicable
(5) Dose acceptance criteria in SRP Section 15.0.1

Table 2
Parameters and Assumptions Used in
Radiological Consequence Calculations
Main Steam Isolation Valve Leakage Pathway

Parameter	Value
Reactor power	3910 MWt
Drywell volume	$2.7x10^5$ ft^3
Leakage rate for intact main steam lines	
0 to 24 hours	150 ft^3/hr
1 to 30 days	75 ft^3/hr
Leakage rates for ruptured main steam line	
0 to 24 hours	100 ft^3/hr
1 to 30 days	50 ft^3/hr
Release pathways	
0 to 20 minutes	To environment
20 minutes to 30 days	To secondary containment
Aerosol removal rate in drywell (per hour)	
Hours	Rates
O to 0.5	0.7474
0.5 to 2.0	0.2983
2.0 to 5.0	1.0550
5.0 to 8.3	0.6390
8.3 to 12	0.5571
12 to 19.4	0.5236
19.4 to 24	0.5068
Aerosol removal rate in main steam lines	0
Elemental iodine removal rate in drywell	0.866 per hour

Table 3
Parameters and Assumptions
used in
Radiological Consequence Calculations
Containment Leakage Pathway

Parameter	Value
Reactor power	3910 MWt
Volume of sprayed region	$8.4x10^5$ ft^3
Volume of unsprayed region	$5.6x10^5$ ft^3
Flow rate from drywell to unsprayed region	
0 - 2 hours	3000 ft^3/min
2 hours - 30 days	Well mixed
Flow rate from unsprayed region to drywell	
0 - 2 hours	0 ft^3/min
2 hours - 30 days	Well mixed
Flow rate between drywell and sprayed region	0 ft^3/min
Flow rate from sprayed region to unsprayed region	
0 - 0.5 hours	18,667 ft^3/min
0.5 - 24 hours	70,000 ft^3/min
Flow rate from unsprayed region to sprayed region	
0 - 0.5 hours	18,667 ft^3/min
0.5 - 24 hours	70,000 ft^3/min
Containment leak rate to environment	
0 - 24 hours	0.682% per day
1 - 30 days	0.341% per day
Spray removal rates	
Aerosols	9.51 per hour
Time to reach DF(1) of 50	3 hours
Elemental iodine	20 per hour
Time to reach DF(1) of 200	6.1 hours
Spray water flux	0.03177 (cm)/(cm sec)
Spray fall height	64 ft

(1) Decontamination factor

Table 4
Parameters and Assumptions
Used in
Radiological Consequence Calculations
Emergency Core Cooling system Leakage Pathway

ECCS Leakage Model

Parameter	Value
Plant power	3910 MWt
Release fractions and timing	As specified for BWR in NUREG-1465 (gap and early in-vessel iodine releases only)
Release location	Directly to suppression pool
Suppression pool water volume	$1.706x10^5$ ft^3
Secondary containment volume	$3x10^5$ ft^3
ECCS leak rate	
10 minutes to 30 days	0.15 ft^3/min
Partition factor	10
Standby gas treatment system	
Flow rate	4000 cfm
Iodine removal efficiency	99%
Aerosol removal efficiency	99%

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

Parameter	Value
Reactor power	3910 MWt
Radial peaking factor	1.7
Fission product decay period	24 hours
Number of fuel rods damaged	142
Fuel pool water depth	23 ft
Fuel gap fission product inventory	
Noble gases excluding Kr-85	5%
Kr-85	10%
I-131	8%
Alkali metals	12%
Fuel pool decontamination factors	
Iodine	200
Noble gases	1
Secondary containment requirements	None
Control room	
Isolation	No
Fresh air system	No credit
Unfiltered infiltration	Not applicable
Recirculation flow	Not applicable
Iodine Protection factor	1
Atmospheric relative concentrations (χ/Q values)	
Exclusion area boundary (0 to 2 hours)	6.0E-4 sec/m^3
Control room	8.5E-4 sec/m^3
Duration of accident	2 hours
Computer code used in dose calculation	Habitt 1.1
Dose conversion factors	Federal Guidance Reports 11 and 12

Table 6
Parameters and Assumptions
Used in
Radiological Consequence Calculations
Control Rod Drop Accident

Parameters	Values
Source term	NUREG-1465
Reactor power	3910 MWt
Core radial peaking factor	1.7
Condense airborne volume	2.27E+5 ft^3
Turbine airborne volume	5.65E+4 ft^3
Gap release fractions	
Noble gases	10 percent
Halogens	10 percent
Alkali metals	12 percent
Core melt fractions	
Noble gases	100 percent
Halogens	50 percent
Alkali metals	25 percent
Fraction of fuel melted in each fuel assembly	0.77 percent
Number of fuel assemblies affected	16
Transfer rate from condenser to environment	1 percent per day
Fission product release fractions from each fuel assembly	
Noble gases	10.308 percent
Halogens	10.693 percent
Alkali metals	12.1 percent
Release fractions to turbine/condenser	
Noble gases	100 percent
Iodine	10 percent
Particulate	1 percent
Atmospheric dispersion factors (sec/ m^3)	
0 to 2 hours EAB	6.0E-4
0 to 2 hours LPZ	1.25E-4
2 to 8 hours LPZ	6.0E-5
8 to 24 hours LPZ	4.5E-5

Table 6
Parameters and Assumptions
Used in
Radiological Consequence Calculations
Control Rod Drop Accident
(Continued)

Control Room

Volume	2.53E+5 ft^3
Manual isolation	30 minutes
Automatic isolation	No
Fresh air supply system fission product removal efficiencies	
HEPA filter	99 percent
Charcoal adsobers	0 percent
Recirculation rate	4,000 cfm (after 18 minutes)
Unfitered inleakage rate	2010 cfm (from 0 to 30 days)
Control room atmospheric dispersion factors (sec/ m^3)	
0 to 2 hours	8.0E-4
2 to 8 hours	7.0E-4
8 to 24 hours	3.0E-4
Source term release point	Turbine building vent
Air intake point into control room	Control building roof

Table 7

**Meteorological Data
for
Loss-of-Coolant Accident**

Exclusion Area Boundary

Time (hr)	X/Q (sec/m^3)
0-2	6.0E-4

Low Population Zone

Time (hr)	X/Q (sec/m^3)
0-2	1.25E-4
2-8	6.0E-5
8-24	4.5E-5
24-96	2.0E-5
96-720	7.0E-6

Control Room

Time (hr)	X/Q (sec/m^3)
0-2	8.0E-4
2-8	5.0E-4
8-24	2.5E-4
24-96	1.6E-4
96-720	1.3E-4