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GNRO-2011/00024

April 14, 2011

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555

SUBJECT: Request for Additional Information Regarding
Extended Power Uprate
Grand Gulf Nuclear Station, Unit 1
Docket No. 50-416
License No. NPF-29

REFERENCES:
1. Email to F. Burford dated March 16, 2011 GG EPU Dose Assessment Branch Request for Additional Information (ME4639) (Accession Number ML110750132)
2. License Amendment Request, Extended Power Uprate, dated September 8, 2010 (GNRO-2010/00056, Accession Number ML102660403)

Dear Sir or Madam:

The Nuclear Regulatory Commission (NRC) requested additional information (Reference 1) regarding certain aspects of the Grand Gulf Nuclear Station, Unit 1 (GGNS) Extended Power Uprate (EPU) License Amendment Request (LAR) (Reference 2). Attachment 1 provides responses to the additional information requested by Accident Dose Branch.

No change is needed to the no significant hazards consideration included in the initial LAR (Reference 2) as a result of the additional information provided. There are no new commitments included in this letter.

If you have any questions or require additional information, please contact Jerry Burford at 601-368-5755.

I declare under penalty of perjury that the foregoing is true and correct. Executed on April 14, 2011.

Sincerely,

MAK/FGB/dm

Attachments:
1. Response to Request for Additional Information, Accident Dose Branch

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**Attachment 1**

**GNRO-2011/00024**

**Grand Gulf Nuclear Station Extended Power Uprate**

**Response to Request for Additional Information**

**Accident Dose Branch**

**Response to Request for Additional Information**
**Accident Dose Branch**

By letter dated September 8, 2010, Entergy Operations, Inc. (Entergy) submitted a license amendment request (LAR) for an Extended Power Uprate (EPU) for Grand Gulf Nuclear Station, Unit 1 (GGNS). The U.S. Nuclear Regulatory Commission (NRC) staff, by correspondence dated March 15, 2011 (Accession Number ML110750132), has determined that the following additional information requested by the Accident Dose Branch is needed for the NRC staff to complete their review of the amendment. Entergy's response is also provided below.

**RAI # 1**

In Section 2.9.2 and Section 2.9.3 of Attachment 5A, "Safety Analysis Report for Grand Gulf Nuclear Station Constant Pressure Power Uprate", of your September 8, 2010, submittal, it is stated that all of the design basis accident (DBA) dose consequence analyses were updated to reflect minor changes to design inputs. The NRC staff's initial review of the EPU calculated doses compared to the current licensing basis (CLB) doses indicates more than minimal increase in the consequences of an accident. Because these analyses are a part of the basis for approval of the proposed EPU, the NRC staff needs additional information in order to complete its review. Provide additional information describing all the basic parameters used in for the DBA dose consequence analyses described in Section 2.9.2 and Section 2.9.3 of Attachment 5A of your submittal. For each parameter, please indicate the CLB value, the revised value where applicable, as well as the basis for any changes to the CLB. Also describe any methodologies that may have changed based on the proposed amendment. The NRC staff requests that the information be presented in a tabular form.

**Response**

The primary factor affecting accident radiological doses is the increased source term in the reactor core due to the higher power associated with EPU. If all core isotopes increased exactly in proportion with the reactor power increase, the increase in dose would be ~15%. Due to changes in some fuel design parameters (*e.g.*, enrichments, cycle lengths, and uranium masses), the changes to the core inventories varied significantly; however, the core inventories of the radiologically-significant isotopes of iodine and noble gases increased in the expected range of 13-15%. Tables 1 and 2 provide a comparison of the isotopic changes between the Current Licensed Thermal Power (CLTP) and EPU core-average and peaking bundle source term inventories, respectively.

The CLTP level for GGNS is 3898 MWt. The CLTP dose consequence analyses were for the most part performed at a power level of 3910 MWt (100.3% of the rated power level). The EPU core licensed thermal power is 4408 MWt. The EPU source terms include a 2% margin of uncertainty; i.e., the EPU source terms are based on 4496 MWt

Section 2.9.2 of Attachment 5 to the GGNS EPU submittal lists the GGNS design basis accidents impacted by Alternative Source Terms as the Loss-of-Coolant Accident (LOCA), Fuel Handling Accident (FHA), Control Rod Drop Accident (CRDA), and the Main Steam Line Break (MSLB). These accidents are consistent with those described in Regulatory Guide 1.183, *Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors*.

The evaluation performed to assess the impact of the EPU on the radiological dose consequences of the eleven radiological events described in GGNS USFAR Chapter 15 focused on the assumptions, design inputs, release scenarios and source term inventories utilized in the calculation-of-record (COR), with the intent of identifying those that could be impacted by the EPU. A tabular comparison of the impact of the EPU on key inputs was developed on a per event basis, and these are contained in Tables 3 through 12 at the end of this response. Conservative scaling techniques were used to assess the impact of any EPU related changes in key inputs on dose consequences. It was concluded that the only significant impact to the GGNS dose consequence analyses was the changes to the core source terms associated with EPU.

An accident specific summary of the EPU assessment is provided below.

1. <u>Loss-of-Coolant Accident (LOCA) – Updated Final Safety Analysis Report (UFSAR) Section 15.6.5</u>

   <u>Current Licensing Basis Accident Scenario</u>: The GGNS LOCA was updated in support of the GGNS alternative source term transition (AST) (Attachment 5 to ML003770202). Subsequent to implementation of the AST license amendment, the LOCA analysis was updated in accordance with 10 CFR 50.59, *Changes, tests and experiments*, to reflect minor changes to design inputs.

   This event, which is currently analyzed at a core power level of 3910 MWt, postulates a circumferential break in a recirculation loop pipe resulting in a loss-of-coolant to the core prior to initiation of the Emergency Core Cooling Systems (ECCS). Fission product release fractions from the core are based on the guidance provided in Regulatory Guide 1.183. The primary release pathways are as follows:

   - <u>Containment leakage</u>: Fission products from the core are released to the drywell and then transported to the primary containment. Plate-out of elemental iodine and natural deposition of aerosols is credited in the drywell. The pH of the suppression pool is controlled to a value above 7.0. Therefore, re-evolution of the iodines from the pool is not considered. Airborne activity in the primary containment is removed by sprays and by plate-out. Airborne activity in the primary containment will leak into the secondary containment at a specified rate. Airborne activity in the secondary containment is released to the environment via the Standby Gas Treatment System (SGTS). The secondary containment draw-down time is considered in the COR analysis.

   - <u>Leakage from the Main Steam Isolation Valves (MSIVs)</u>: There is a 250 scfh leak rate for the first 24 hours, and 125 scfh after 24 hours. During the first 20 minutes post-LOCA, the release goes directly to the environment. After 20 minutes, the release goes to the secondary containment.

   - <u>Engineered Safety Features (ESF) liquid leakage outside the primary containment</u>: Only halogens are released. The leak rate is 1.12 gpm (10 min to 30 days). The iodine flash fraction is 10%.

EPU Assessment

The suppression pool pH response is impacted by a modification to the Standby Liquid Control (SLC) System in conjunction with the EPU. This modification increases the boron-10 enrichment of the contents of the SLC tank while reducing the sodium pentaborate concentration. The final SLC system design was determined to deliver sufficient sodium pentaborate to the containment/drywell pools to maintain a pH greater than 7.0 for thirty days post-LOCA taking into consideration increased acid production due to EPU radiation environments. Therefore, consistent with the CLTP case, iodine re-evolution from the suppression pool is precluded for the EPU LOCA radiological analysis.

The isotope-by-isotope comparison between the COR source term and the EPU source term in Table 1 indicates that the total core activity is increased by 6.42%. The releases of importance are the iodines, noble gases, and alkali metals. Among these, the largest impact is on I-135, with an increase of 15.02% due to EPU. This value was used to scale the EPU dose results presented below. See Table 3 for a tabular comparison of the impact of the EPU on key inputs.

The EPU radiological dose consequences of a LOCA are shown in the following table and compared to CLTP doses as recorded in the COR, and the allowable limits per the GGNS licensing basis. Note that the Technical Support Center (TSC) is located within the control room boundary so that the TSC doses are identical to the control room doses. The EPU doses were conservatively estimated based on the source term scaling technique briefly described above and found to be within the regulatory limits stated in 10 CFR 50.67, *Accident source term*.

**Loss-of-Coolant Accident Radiological Consequences**

| Location | TEDE Dose (REM) | | | | |
|---|---|---|---|---|---|
| | Original AST | CLTP | EPU | Allowable Limit | % Change (CLTP to EPU) |
| CR* | 3.65 | 3.69 | 4.24 | 5 | 15% |
| EAB* | 8.78 | 8.70 | 10.01 | 25 | 15% |
| LPZ* | 5.32 | 5.15 | 5.92 | 25 | 15% |
| TSC | 3.65 | 3.69 | 4.24 | 5 | 15% |

* Control Room (CR); Exclusion Area Boundary (EAB); Low Population Zone (LPZ)

2. Fuel Handling Accident (FHA) - UFSAR Section 15.7.4

Current Licensing Basis Accident Scenario: The GGNS FHA was updated in support of the GGNS alternative source term transition (Attachment 6 to ML003770202). Subsequent to implementation of the AST license amendment, the FHA analysis was updated in accordance with 10 CFR 50.59 to reflect minor changes to design inputs.

This accident is the drop of a fuel assembly onto the reactor core or stored fuel bundles. The hypothesized sequence of events is as follows:

- A fuel assembly is being handled by the fuel handling platform over the spent fuel pool or by the refueling platform over the containment racks or reactor core. When the hoist is at its fully-retracted position, the assembly and the mast drop, striking seated irradiated fuel assemblies.

- All fuel rods in the dropped assembly and a number of rods in the struck assemblies are assumed to fail (a total of 2.03 fuel bundle cladding failures), releasing radioactive gases to the pool water.
- Radioactive gases pass from the water to the air above the drop area.

The FHA is assumed to occur at 24 hours after shutdown. For conservatism, a radial peaking factor is applied to all activities released from the cladding failures. Only halogens and noble gases are assumed to be released, since the alkali metals are completely retained in the pool water.

EPU Assessment

The release scenario is not affected by the EPU. The EPU evaluation confirmed that the fuel failure and peaking factor assumptions in the COR remain valid for EPU. The EPU dose is estimated based on the contribution of each isotope to the calculated dose in the COR and the predicted increase in each isotope for EPU per Table 2. The weighted EPU dose scaling factor applicable to the EAB and CR are 1.18 and 1.12, respectively. See Table 4 for a tabular comparison of the impact of the EPU on key inputs.

The EPU radiological dose consequences of an FHA are shown in the following table and compared to CLTP doses as recorded in the COR, and the allowable limits per the GGNS licensing basis. The EPU doses were estimated based on the source term scaling technique briefly described above and found to be well within (≤ 25% of) the regulatory limits stated in 10 CFR 50.67 for the EAB, and the 10 CFR 50.67 dose limits for the control room.

**Fuel Handling Accident Radiological Consequences**

| Location | TEDE Dose (REM) | | | | |
|---|---|---|---|---|---|
| | Original AST | CLTP | EPU | Allowable Limit | % Change (CLTP to EPU) |
| CR | 2.04 | 2.80 | 3.14 | 5 | 12% |
| EAB | 1.98 | 2.64 | 3.12 | 6.3 | 18% |
| LPZ | NA | NA | NA | NA | NA |

3. Control Rod Drop Accident (CRDA) - UFSAR Section 15.4.9

Current Licensing Basis Accident Scenario: The GGNS CRDA was updated in support of the GGNS AST transition (Attachment 4 to ML003770202). Subsequent to implementation of the AST license amendment, the CRDA analysis was updated in accordance with 10 CFR 50.59 to reflect minor changes to design inputs.

This accident scenario represents the dropping of a control rod out of the reactor core. The accident is currently analyzed at a peak bundle power level of 8.455 MWt, representing a core power level of 3979 MWt with a radial peaking factor of 1.7. In essence, this is a rapid control rod withdrawal from the core, resulting in the failure of 16 fuel bundles (out of 800 bundles in the core) representing the four-bundle cell associated with the dropped control blade and one additional surrounding row. The radionuclide release scenario is as follows:

- Of the failed fuel that does not melt, only gap activity, comprised of 10% of the noble gases, 10% of the halogens, and 12% of the alkali metals, is released to the RCS.
- 0.77% of the failed fuel bundles experience fuel-melt. From that melted fuel, 100% of the noble gases, 50% of the halogens and 25% of the alkali metals are released to the RCS.
- The percentages of RCS activities transported to the turbine and condenser are as follows: 100% of the noble gases, 10% of the halogens, and 1% of the remaining radionuclides.
- The percentages of turbine/condenser activities available for release to the environment are as follows: 100% of the noble gases, 10% of the halogens, and 1% of the remaining radionuclides.
- The leak rate from the condenser to the environment is 1% per day for 24 hours.

EPU Assessment

The release scenario is not affected by the EPU. The EPU evaluation confirmed that the fuel failure / melt assumptions assumed in the COR remain valid for EPU. A comparison was performed of the COR source with the EPU source on an isotope-by-isotope basis. The EPU dose is estimated based on the contribution of each isotope to the calculated dose in the COR and the predicted increase in each isotope for EPU per Table 2. The weighted EPU dose scaling factors applicable to the EAB, LPZ and CR are 0.977, 1.018 and 1.112, respectively. See Table 5 for a tabular comparison of the impact of the EPU on key inputs.

The EPU radiological dose consequences of a CRDA are shown in the following table and compared to CLTP doses as recorded in the COR, and the allowable limits per the GGNS licensing basis. The EPU doses were estimated based on the source term scaling technique briefly described above and found to be well within (≤ 25% of) the regulatory limits stated in 10 CFR 50.67 for the EAB and LPZ, and the 10 CFR 50.67 dose limits for the control room.

**Control Rod Drop Accident Radiological Consequences**

| | TEDE Dose (REM) | | | | |
|---|---|---|---|---|---|
| Location | Original AST | CLTP | EPU | Allowable Limit | % Change (CLTP to EPU) |
| CR | 2.62E-01 | 2.62E-01 | 2.91E-01 | 5.0 | 11% |
| EAB | 1.47E-01 | 1.51E-01 | 1.51E-01 | 6.3 | No Change |
| LPZ | 6.40E-02 | 7.23E-02 | 7.36E-02 | 6.3 | 2% |

4. Main Steam Line Break Outside Containment - UFSAR Section 15.6.4

Current Licensing Basis Accident Scenario: This accident was addressed in the GGNS AST transition (Attachment 1 to ML003770202). Although doses were never quantified in the submittal, it was demonstrated that the doses were within the acceptance criteria due to the change in dose conversion factors. GGNS-specific calculations were subsequently developed.

This accident represents the occurrence of a circumferential break in one of the four main steam lines immediately downstream of the outermost MSIV outside the primary containment. A significant amount of reactor coolant is released to the environment before the MSIVs isolate and the steam header depressurizes.

Iodine and noble gas isotopes are released to the environment as a result of this accident. These isotopes are assumed to be the maximum iodine and noble gas inventories in the reactor coolant and steam allowed by GGNS Technical Specifications 3.4.8, *Reactor Coolant System (RCS) Specific Activity*, and 3.7.5, *Main Condenser Offgas*. The Technical Specification iodine concentrations are based on the following:

- Equilibrium Iodine Case: 0.2 μCi/g Dose Equivalent I-131
- Iodine Spiking Case: 4.0 μCi/g Dose Equivalent I-131

The Technical Specification noble gas release concentrations in steam are based on 380 millicuries/sec release rate after 30 minutes decay.

The reactor coolant is released from the break point to the environment in the form of steam (27,750 lbm) and liquid (112,250 lbm). The resultant doses, both for the equilibrium iodine case and for the iodine spiking case, are within their respective NRC acceptance criteria.

<u>EPU Assessment</u>

The release scenario is not affected by the EPU. The EPU does not change reactor dome pressure, thus the release mass is not affected by EPU. The EPU evaluation confirmed that the assumption of no fuel failure/melt assumed in the COR remains valid for EPU. The iodine and noble gas activity concentrations being released are postulated to be the maximum iodine and noble gas inventories in the reactor coolant and steam allowed by GGNS Technical Specifications 3.4.8 & 3.7.5, and these limiting concentrations are independent of power level. Thus the accident dose estimates will not be affected by EPU. See Table 6 for a tabular comparison of the impact of the EPU on key inputs.

The radiological consequences of a steam system pipe break outside containment as calculated in the COR will not be impacted by the EPU.

**Steam System Piping Break Outside Containment Radiological Consequences**
**Equilibrium Iodine Case**

| Location | TEDE Dose (REM) | | | |
|---|---|---|---|---|
| | CLTP | EPU | Allowable Limit | % Change (CLTP to EPU) |
| CR | 0.153 | 0.153 | 5.0 | No Change |
| EAB | 0.123 | 0.123 | 2.5 | No Change |
| LPZ | NA | NA | 2.5 | NA |

**Steam System Piping Break Outside Containment Radiological Consequences**
**Iodine Spiking Case**

| Location | TEDE Dose (REM) | | | |
|---|---|---|---|---|
| | CLTP | EPU | Allowable Limit | % Change (CLTP to EPU) |
| CR | 3.01 | 3.01 | 5.0 | No Change |
| EAB | 2.39 | 2.39 | 25 | No Change |
| LPZ | NA | NA | 25 | NA |

5. Pressure Controller Failure - UFSAR Section 15.2.1

Current Licensing Basis Accident Scenario: The pressure regulator failure is conservatively modeled with an assumed core wide fuel failure with the gap activity in the fuel being released to the RCS with isolation of the main steam lines following reactor scram. The gap activity in the fuel is released with the steam into the suppression pool via the safety relief valves (SRVs). Ten percent of the iodines and 1% of the alkali metals are assumed to reach the suppression pool. Suppression pool decontamination factors (DFs) of 20 and 35 are applied to particulate and elemental iodines, respectively. All the released noble gases are assumed to reach the containment atmosphere. The activity that evolves into the containment from the suppression pool is released via the containment ventilation system in the high volume purge over a period of 10 seconds before the containment is automatically isolated. Two halogen compositions of iodine species are considered: (a) 97% elemental & 3% organic, and (b) 95% aerosol, 0.15% organic, & 4.85% elemental. Composition (a) yields larger doses and is therefore the limiting case.

EPU Assessment

The release scenario is not affected by the EPU. The EPU evaluation confirmed that the fuel failure assumptions assumed in the COR remain valid for EPU. The pre-EPU vs. EPU isotopic core inventories of isotopes that are present in the gap (i.e., noble gases, halogens, alkali metals), are compared to estimate the increase in activity, per isotope, due to the EPU. The weighted EPU dose scaling factor applicable to the EAB and CR are 1.068 and 1.102, respectively. The EAB scale-up factor is used to estimate the EPU LPZ dose. See Table 7 for a tabular comparison of the impact of the EPU on key inputs.

The EPU radiological dose consequences of a Pressure Controller Failure shown in the table below are compared to CLTP doses as recorded in the COR, and the allowable limits per the GGNS licensing basis. The EPU doses remain within the current licensing basis acceptance criteria of 10% of the 10 CFR 50.67 dose limits for the EAB and LPZ, and the 10 CFR 50.67 dose limits for the control room.

**Pressure Controller Failure Radiological Consequences**

| Location | TEDE Dose (REM) | | | |
|---|---|---|---|---|
| | CLTP | EPU | Allowable Limit | % Change (CLTP to EPU) |
| CR | 3.39 | 3.74 | 5.0 | 10% |
| EAB | 2.28 | 2.43 | 2.5 | 7% |
| LPZ | 0.52 | 0.56 | 2.5 | 7% |

6. MSIV Closure - UFSAR Section 15.2.4

Current Licensing Basis Accident Scenario: The MSIV closure event is an unplanned event in which a MSIV closure may cause an immediate closure of all the other MSIVs depending on reactor conditions. This event is considered to be a moderate frequency event in which no fuel failure occurs, but reactor coolant activities are released into the suppression pool via the SRVs. Radioactivity not scrubbed by the suppression pool water is released into the containment atmosphere and then to the environment.

In accordance with GGNS Technical Specification 3.4.8, the COR postulates that the reactor coolant activity concentrations are at the maximum permitted iodine spiking concentrations with a dose equivalent I-131 specific activity of 4.0 μCi/gram. Only EAB doses are evaluated. The atmospheric dispersion factor utilized bounds that applicable to GGNS.

EPU Assessment

The release scenario is not affected by the EPU. The EPU evaluation confirmed that the assumption of no fuel failure / melt in the COR remains valid for EPU. The Technical Specification based reactor coolant activity concentrations are not expected to be impacted significantly by the EPU. Consequently, the result of the COR analysis is not affected by EPU. See Table 8 for a tabular comparison of the impact of the EPU on key inputs.

The radiological consequence of an MSIV Closure event as calculated in the COR and reported in UFSAR Table 15.2-16 is not impacted by the EPU. Results are summarized in the following table:

**MSIV Closure Radiological Consequences**

| Location | TEDE Dose (millirem) | | | |
|---|---|---|---|---|
| | CLTP | EPU | Allowable Limit | % Change (CLTP to EPU) |
| EAB | 0.083 | 0.083 | 100 | No Change |

7. Misplaced Bundle Accident - UFSAR Section 15.4.7

Current Licensing Basis Accident Scenario: This event (also known as a fuel loading error (FLE) event) is the improper loading of a fuel bundle and subsequent operation of the core.

The pre-EPU radiological consequences of this event are estimated based on a generic analysis performed by GEH (see ML061580108) that conservatively assumed that 5 fuel bundles fail (i.e., the misplaced bundle and the 4 surrounding bundles), releasing the associated gap activity into the RCS. For those plants without a main steam high radiation isolation trip, 100% of the noble gases, 10% of the iodines, and 1% of the alkali metals are estimated to reach the condenser. Only noble gases are released to the environment (via the Offgas System). The source term is based on a core average bundle power of 5.75 MWt, a radial peaking factor of 2.5, and an additional safety factor of 1.4.

GGNS does not have a main steam high radiation isolation trip, thus the release is assumed to occur via the Offgas System. The radiological consequences depend on the site-specific χ/Qs and the offgas system design.

EAB Dose: Based on the Krypton and Xenon holdup times applicable to the GGNS specific Offgas System, and the GGNS EAB 0-2 hour χ/Q, the estimated offsite dose is well below the acceptance criterion of 2.5 Rem TEDE. This dose estimate is based on an average fuel bundle power of 5.75 MWt, which bounds the current licensing-basis average fuel bundle power for GGNS of 3910 MWt / 800 bundles, or 4.89 MWt.

Control Room Dose: The generic FLE analysis demonstrates that the control room dose will be less than the 5.0 Rem TEDE acceptance criterion provided that the applicable control room χ/Q is < 1.25E-2 s/m$^3$. The GGNS specific control room χ/Q is well below the required value. Again, this conclusion is based on an average fuel bundle power of 5.75 MWt, which bounds the current licensing-basis average fuel bundle power for GGNS of 4.89 MWt.

EPU Assessment

The release scenario is not affected by the EPU. The EPU evaluation confirmed that the fuel failure assumptions assumed in the COR remain valid for EPU. The EPU core average bundle power is 4496 MWt / 800 bundles, or 5.62 MWt, which is bounded by the value used in the generic analysis. The Krypton (Kr) and Xenon (Xe) delay times are not affected by EPU because the condenser inleakage is not affected. Hence, EPU will not cause an increase in the current licensing-basis doses and, therefore, will not cause the EAB and control room doses to exceed regulatory limits. See Table 9 for a tabular comparison of the impact of the EPU on key inputs.

The current radiological consequences of an FLE event are based on a generic analysis that bounds operation at the EPU power level.

**Misplaced Bundle Accident Radiological Consequences**

| Location | TEDE Dose (REM) | | | |
|---|---|---|---|---|
| | CLTP | EPU | Allowable Limit | % Change (CLTP to EPU) |
| CR | < 5.0 | < 5.0 | 5.0 | No Change |
| EAB | ≈ 0.02 | ≈ 0.02 | 2.5 | No Change |

8. <u>Offgas System Leak or Failure - UFSAR Section 15.7.1</u>

<u>Current Licensing Basis Accident Scenario</u>: This postulated failure is the rupture of the Offgas System pressure boundary. The failure is assumed to be a break in the charcoal delay line, resulting in releases from (1) charcoal adsorber failure, (2) delay line failure, and (3) continued operation of the steam jet air ejector (SJAE) for 1 hour. The noble gas activity in the Offgas System is based on a continuous release of 399,000 µCi/sec noble gas after 30 minutes decay (which includes a margin for measurement uncertainty). The particulate releases (noble gas daughters) from the failed charcoal adsorbers currently correspond to an analyzed core power level of 3910 MWt.

<u>EPU Assessment</u>

The offgas failure and release paths are not affected by EPU. The delay line transit time is not affected by power level. The charcoal bed hold-up time calculated in the COR is based on a NUREG-0016, *Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Boiling-Water Reactors (BWR-GALE Code)*, Revision 1, dated January 1979, methodology and is not adversely impacted by EPU.

According to Regulatory Guide 1.98, *Assumptions Used for Evaluating the Potential Radiological Consequences of a Radioactive Offgas System Failure in a Boiling Water Reactor*, the noble gas activity release rate is directly proportional to the core power level. For the GGNS EPU, a scaling factor of 1.15 was conservatively used. See Table 10 for a tabular comparison of the impact of the EPU on key inputs.

In accordance with current licensing basis, the acceptance dose criteria for this accident are based on being well within 10 CFR 100, *Reactor site criteria*, limits for EAB and LPZ doses. The acceptance criterion for the control room dose is defined in 10 CFR 50.67.

The EPU radiological dose consequences of an Offgas System Leak or Failure are shown in the following table and compared to CLTP doses as recorded in the COR, and the allowable limits per the GGNS licensing basis. The EPU doses were conservatively estimated based on the source term scaling technique briefly described above and found to be well within (≤ 25% of) the GGNS current licensing basis limits stated in 10 CFR 100 for the EAB and LPZ, and the 10 CFR 50.67 dose limits for the control room.

**Offgas System Leak or Failure**

| Location | Dose (REM) | | | |
|---|---|---|---|---|
| | CLTP | EPU | Allowable Limit | % Change (CLTP to EPU) |
| EAB-Thyroid | negligible | negligible | 75 | No Change |
| LPZ-Thyroid | negligible | negligible | 75 | No Change |
| CR-TEDE | 0.124 | 0.143 | 5 | 15% |
| EAB-Whole Body | 1.68 | 1.93 | 6 | 15% |
| LPZ-Whole Body | 0.384 | 0.442 | 6 | 15% |

9. Radioactive Liquid Waste System Leak or Failure (release to atmosphere) - UFSAR Section 15.7.2

Current Licensing Basis Accident Scenario: This accident is the failure of the limiting radwaste system vessel, with a resulting release of iodine isotopes to the atmosphere. It has been determined that the limiting radwaste system vessel is the Equipment Drain Collection Tank. The source term is based on primary coolant activity concentrations based on a reactor thermal power of 3833 MWt.

In the evaluation of the radiological consequences of this accident, the following assumptions are made:

- Radioisotope inventory in liquid radwaste system is based on normal system operation.
- Only radioiodine isotopes are released since noble gases are not present and particulate radioisotopes will not become airborne.
- The entire airborne iodine inventory is assumed to be in the elemental chemical species.
- No operator mitigation is assumed.
- Instantaneous release is assumed.
- No credit is taken for partition, filtration, holdup, or dilution of iodine once it is released from the failed tank.

The acceptance dose criteria for this accident are based on a small fraction of 10 CFR 100 limits for EAB and LPZ doses. The acceptance criterion for the control room dose is defined in GDC-19, *Control room*.

EPU Assessment

The release scenario is not affected by the EPU. Since only iodine isotopes are postulated to be released, the impact of the change in fuel cycle length is minimal. The scaling factor for this event is conservatively determined by dividing EPU rated power, including uncertainty, by original licensed thermal power (OLTP) power; this results in a scaling factor of 1.17 (i.e., 4496 / 3833.) See Table 11 for a tabular comparison of the impact of the EPU on key inputs.

The impact of the EPU on the radiological dose consequences of a Radioactive Liquid Waste System Leak or Failure are shown in the following table and compared to CLTP doses as recorded in the COR and the allowable limits per the GGNS licensing basis. The EPU doses were conservatively estimated based on the source term scaling technique briefly described above and found to be a small fraction (<10%) of the regulatory limits stated in 10 CFR 100 for the EAB and LPZ, and GDC 19 for the control room. Note that UFSAR section 15.7.2 addresses a failure of the evaporator bottoms tank. The UFSAR acknowledges that this system is no longer used but since the tank may be used to hold other radwaste liquids, the previously analyzed bounding EAB dose values of 1.25 Rem thyroid and negligible whole body presented in the UFSAR have been retained as the licensing basis values. As demonstrated above, the UFSAR doses will also bound the EPU dose consequences of a Radioactive Liquid Waste System Leak or Failure.

**Radioactive Liquid Waste System Leak or Failure Radiological Consequences**

| Location | Dose (REM) | | | |
|---|---|---|---|---|
| | CLTP | EPU | Allowable Limit | % Change (CLTP to EPU) |
| CR-Thyroid | 2.51E-01 | 2.94E-01 | 30 | 17% |
| EAB-Thyroid | 2.47E-01 | 2.90E-01 | 30 | 17% |
| LPZ-Thyroid | 5.47E-02 | 6.42E-02 | 30 | 17% |
| CR-Whole Body | 1.63E-04 | 1.91E-04 | 5 | 17% |
| EAB-Whole Body | 4.74E-03 | 5.56E-03 | 2.5 | 17% |
| LPZ-Whole Body | 1.05E-03 | 1.23E-03 | 2.5 | 17% |

10. Liquid Radwaste Tank Failure (release to groundwater) - UFSAR Section 15.7.3

Current Licensing Basis Accident Scenario: This event is a failure of a liquid radwaste system tank, resulting in the largest release to groundwater of significant radionuclides in the liquid radwaste system. That tank has been determined to be the reactor water cleanup system (RWCU) phase separator decay tank in the radwaste building, and its failure is considered to be a limiting fault.

The fuel cycle length in the COR is assumed to be 12 months. The nuclides considered are strontium-90 (Sr-90) and cesium-137 (Cs-137) because they comprise the greatest potential health hazard in the event of an accidental spill. The pre-EPU radiological consequences, as presented in the UFSAR, are based on the OLTP of 3833 MWt.

The COR shows that the concentrations of Sr-90 and Cs-137 are reduced to below maximum permissible concentration (MPC) (Sr-90 = 3.0E-07 μCi/cc, Cs-137 = 2.0E-05 μCi/cc) at a distance of about 57 ft. from the plant. The concentration of the contaminants at the Mississippi River after the estimated ground water travel time of 12.5 years to reach the river would be essentially zero ($<10^{-20}$ μCi/cc).

EPU Assessment

The release scenario is not affected by the EPU. Considering that the post-EPU fuel cycle will be 24 months (in contrast with the initial fuel cycle of 12 months), the EPU scale-up ratio is determined by taking the product of the power scale-up ratio and the fuel cycle scale-up ratio. Applying the resultant scale-up ratio to the pre-EPU radiological consequences would yield a post-EPU result that would remain negligible. See Table 12 for a tabular comparison of the impact of the EPU on key inputs.

The post-EPU radiological release to the river due to a Liquid Radwaste Tank Failure (release to groundwater) is estimated to remain negligible.

11. Recirculation Pump Seizure Accident - UFSAR Section 15.3.3

Current Licensing Basis Accident Scenario: The pump seizure event is a postulated accident in which the operating recirculation pump suddenly stops rotating, causing a rapid diminution in core flow, heat transfer from fuel rods, and critical power ratio. A calculation performed for the previous cycle was used as the basis for the CLTP data presented below. However, as reported in UFSAR Section 15.3.3.5, the current GGNS fuel vendor does not predict any fuel failures associated with the recirculation pump seizure event. Therefore, the radiological consequences of this event are bounded by those calculated for the MSIV closure event.

EPU Assessment

For the EPU assessment of this event, as reported in LAR Attachment 5, the early CLTP dose consequences were scaled up by 15%. The comparison of the CLTP and EPU radiological dose consequences of a Recirculation Pump Seizure Accident are shown in the following table. However, as noted above, since there are no longer any fuel failures associated with this event, the radiological dose consequences are actually bounded by those for the MSIV closure event. Note, a tabular comparison of the impact of the EPU on key inputs is not provided for this event.

**Recirculation Pump Seizure Accident Radiological Consequences**

| | TEDE Dose (REM) | | | |
|---|---|---|---|---|
| Location | CLTP | EPU | Allowable Limit | % Change (CLTP to EPU) |
| CR | 3.72 | 4.28 | 5.0 | 15% |
| EAB | 1.886 | 2.17 | 2.5 | 15% |
| LPZ | 0.957 | 1.10 | 2.5 | 15% |

**EPU IMPACT ON GGNS DESIGN BASIS ACCIDENT DOSE CONSEQUENCES**

| **Table 1 - GGNS Core Source Term Isotopic Comparison** | | | |
|---|---|---|---|
| GGNS UFSAR Isotopes | | Bounding EPU Core | % Difference from |
| Table 15.6-9 | | Ci/Core | Licensing basis |
| Nuclide | Ci/Core at t=0 | t=0 | |
| CO58 | 5.978E+05 | 1.135E+06 | 89.90% |
| CO60 | 7.155E+05 | 1.965E+06 | 174.61% |
| BR82 | 7.625E+05 | 7.838E+05 | 2.79% |
| BR83 | 1.377E+07 | 1.510E+07 | 9.69% |
| BR84 | 2.424E+07 | 2.625E+07 | 8.28% |
| KR85 | 1.517E+06 | 1.574E+06 | 3.73% |
| **KR85M** | **3.562E+07** | **3.205E+07** | **-10.03%** |
| **KR87** | **6.479E+07** | **6.156E+07** | **-4.99%** |
| **KR88** | **8.743E+07** | **8.664E+07** | **-0.90%** |
| RB86 | 2.884E+05 | 3.032E+05 | 5.13% |
| SR89 | 1.093E+08 | 1.169E+08 | 6.94% |
| SR90 | 1.232E+07 | 1.250E+07 | 1.49% |
| SR91 | 1.409E+08 | 1.460E+08 | 3.62% |
| SR92 | 1.472E+08 | 1.579E+08 | 7.28% |
| Y90 | 1.271E+07 | 1.332E+07 | 4.78% |
| Y91 | 1.392E+08 | 1.505E+08 | 8.12% |
| Y92 | 1.478E+08 | 1.586E+08 | 7.32% |
| Y93 | 1.681E+08 | 1.833E+08 | 9.03% |
| ZR95 | 1.822E+08 | 2.162E+08 | 18.66% |
| ZR97 | 1.794E+08 | 2.229E+08 | 24.22% |
| NB95 | 1.828E+08 | 2.172E+08 | 18.83% |
| MO99 | 2.009E+08 | 2.305E+08 | 14.73% |
| TC99M | 1.759E+08 | 2.010E+08 | 14.25% |
| RU103 | 1.767E+08 | 1.930E+08 | 9.20% |
| RU105 | 1.292E+08 | 1.362E+08 | 5.45% |
| **RU106** | **7.695E+07** | **7.459E+07** | **-3.06%** |
| RH105 | 1.207E+08 | 1.276E+08 | 5.72% |
| SB125 | 2.172E+06 | 2.563E+06 | 17.99% |
| **SB127** | **9.087E+07** | **1.353E+07** | **-85.11%** |
| SB129 | 3.640E+07 | 3.998E+07 | 9.82% |
| TE127 | 1.259E+07 | 1.343E+07 | 6.69% |
| TE127M | 1.680E+06 | 1.805E+06 | 7.43% |
| TE129 | 3.582E+07 | 3.933E+07 | 9.79% |
| **TE129M** | **7.781E+06** | **5.848E+06** | **-24.84%** |
| TE131M | 1.595E+07 | 1.782E+07 | 11.70% |
| TE132 | 1.528E+08 | 1.731E+08 | 13.30% |

| Table 1 - GGNS Core Source Term Isotopic Comparison | | | |
|---|---|---|---|
| GGNS UFSAR Isotopes Table 15.6-9 | | Bounding EPU Core Ci/Core | % Difference from Licensing basis |
| Nuclide | Ci/Core at t=0 | t=0 | |
| TE133M | 8.086E+07 | 9.072E+07 | 12.19% |
| TE134 | 1.833E+08 | 2.058E+08 | 12.25% |
| I131 | 1.078E+08 | 1.220E+08 | 13.17% |
| I132 | 1.555E+08 | 1.760E+08 | 13.18% |
| I133 | 2.156E+08 | 2.476E+08 | 14.84% |
| I134 | 2.375E+08 | 2.721E+08 | 14.56% |
| I135 | 2.015E+08 | 2.318E+08 | 15.02% |
| XE133 | 2.121E+08 | 2.384E+08 | 12.40% |
| XE135 | 8.422E+07 | 8.440E+07 | 0.21% |
| **CS134** | **3.204E+07** | **2.806E+07** | **-12.43%** |
| **CS136** | **9.400E+06** | **9.056E+06** | **-3.66%** |
| CS137 | 1.641E+07 | 1.666E+07 | 1.55% |
| CS138 | 1.995E+08 | 2.267E+08 | 13.64% |
| BA139 | 1.953E+08 | 2.214E+08 | 13.34% |
| BA140 | 1.926E+08 | 2.139E+08 | 11.07% |
| LA140 | 1.982E+08 | 2.274E+08 | 14.71% |
| LA141 | 1.815E+08 | 2.018E+08 | 11.21% |
| LA142 | 1.746E+08 | 1.951E+08 | 11.75% |
| CE141 | 1.786E+08 | 2.029E+08 | 13.59% |
| CE143 | 1.703E+08 | 1.876E+08 | 10.16% |
| CE144 | 1.398E+08 | 1.659E+08 | 18.68% |
| PR143 | 1.667E+08 | 1.815E+08 | 8.89% |
| ND147 | 7.449E+07 | 8.112E+07 | 8.90% |
| NP238 | 6.178E+07 | 6.332E+07 | 2.49% |
| **NP239** | **2.569E+09** | **2.520E+09** | **-1.91%** |
| **PU238** | **7.409E+05** | **5.102E+05** | **-31.14%** |
| **PU239** | **5.341E+04** | **5.164E+04** | **-3.31%** |
| **PU240** | **8.090E+04** | **6.956E+04** | **-14.02%** |
| PU241 | 2.170E+07 | 2.283E+07 | 5.22% |
| **AM241** | **2.788E+04** | **2.552E+04** | **-8.46%** |
| **CM242** | **8.481E+06** | **6.769E+06** | **-20.19%** |
| **CM244** | **1.790E+06** | **4.282E+05** | **-76.08%** |
| | | | |
| Total | 8.960E+09 | 9.535E+09 | 6.42% |

Bolded isotopes indicate the licensing basis value bounds the EPU calculated value.

| **Table 2 - GGNS Bundle Source Term FHA and CRDA Isotopic Comparison<br>Fuel Handling Accident** | | | |
|---|---|---|---|
| GGNS Licensing Basis FHA Isotopes | | Bounding EPU | % Difference |
| Calc XC-Q1J11-98018 R3 | | Bundle Source Term | from licensing basis |
| Nuclide | Ci/bundle, t=0 | Ci/bundle, t=0 | |
| | | With 2.2 assumed RPF | |
| BR82 | 3.538E+03 | 3.918E+03 | 10.75% |
| **BR83** | **4.941E+04** | **4.657E+04** | **-5.74%** |
| **BR84** | **9.246E+04** | **8.360E+04** | **-9.58%** |
| I128 | 1.107E+04 | 1.182E+04 | 6.74% |
| I130 | 2.600E+04 | 2.792E+04 | 7.38% |
| I131 | 3.117E+05 | 3.483E+05 | 11.73% |
| I132 | 4.451E+05 | 4.996E+05 | 12.25% |
| I133 | 6.222E+05 | 6.785E+05 | 9.05% |
| I134 | 6.984E+05 | 7.511E+05 | 7.54% |
| I135 | 5.797E+05 | 6.382E+05 | 10.09% |
| CS132 | 1.477E+02 | 1.656E+02 | 12.13% |
| CS134 | 1.069E+05 | 1.881E+05 | 75.92% |
| CS134M | 3.156E+04 | 3.531E+04 | 11.88% |
| **CS135M** | **3.679E+04** | **3.098E+04** | **-15.80%** |
| CS136 | 3.035E+04 | 4.701E+04 | 54.91% |
| CS137 | 3.579E+04 | 7.698E+04 | 115.08% |
| CS138 | 6.137E+05 | 6.395E+05 | 4.21% |
| RB86 | 1.345E+03 | 1.557E+03 | 15.79% |
| **RB88** | **3.313E+05** | **2.900E+05** | **-12.48%** |
| **RB89** | **4.323E+05** | **3.753E+05** | **-13.18%** |
| **KR83M** | **4.939E+04** | **4.657E+04** | **-5.70%** |
| KR85 | 3.193E+03 | 6.160E+03 | 92.92% |
| **KR85M** | **1.150E+05** | **1.033E+05** | **-10.18%** |
| **KR87** | **2.323E+05** | **2.028E+05** | **-12.69%** |
| **KR88** | **3.285E+05** | **2.864E+05** | **-12.80%** |
| **KR89** | **4.172E+05** | **3.564E+05** | **-14.57%** |
| XE129M | 1.230E+01 | 1.261E+01 | 2.52% |
| XE131M | 3.534E+03 | 3.920E+03 | 10.93% |
| XE133 | 5.843E+05 | 6.785E+05 | 16.12% |
| XE133M | 1.815E+04 | 2.144E+04 | 18.15% |
| XE135 | 1.706E+05 | 2.625E+05 | 53.85% |
| XE135M | 1.261E+05 | 1.402E+05 | 11.19% |
| XE137 | 5.480E+05 | 5.940E+05 | 8.39% |
| XE138 | 5.698E+05 | 5.826E+05 | 2.24% |

Bolded isotopes indicate the licensing basis value bounds the EPU calculated value.

Both evaluations considered a radial peaking factor of 2.2.

| Table 2 - GGNS Bundle Source Term FHA and CRDA Isotopic Comparison<br>Control Rod Drop Accident | | | |
|---|---|---|---|
| GGNS Licensing Basis CRDA Isotopes | | Bounding EPU | % Diff |
| Calc XC-Q1J11-98018 R2 | | Bundle Source Term | from licensing |
| Ci, t=0 | | Ci/bundle, t=0 | basis |
| | | With 1.7 assumed RPF | |
| BR82 | 2.717E+03 | 3.028E+03 | 11.44% |
| **BR83** | **3.798E+04** | **3.599E+04** | **-5.24%** |
| **BR84** | **7.107E+04** | **6.460E+04** | **-9.10%** |
| I128 | 8.534E+03 | 9.131E+03 | 6.99% |
| I130 | 2.002E+04 | 2.157E+04 | 7.76% |
| I131 | 2.394E+05 | 2.691E+05 | 12.41% |
| I132 | 3.415E+05 | 3.861E+05 | 13.05% |
| I133 | 4.781E+05 | 5.243E+05 | 9.66% |
| I134 | 5.369E+05 | 5.804E+05 | 8.10% |
| I135 | 4.455E+05 | 4.932E+05 | 10.70% |
| CS132 | 1.140E+02 | 1.280E+02 | 12.26% |
| CS134 | 1.002E+05 | 1.453E+05 | 45.03% |
| CS134M | 2.432E+04 | 2.729E+04 | 12.19% |
| CS135M | | 2.394E+04 | N/A |
| CS136 | 2.653E+04 | 3.633E+04 | 36.94% |
| CS137 | 3.544E+04 | 5.948E+04 | 67.84% |
| CS138 | 4.718E+05 | 4.942E+05 | 4.75% |
| RB86 | 1.048E+03 | 1.203E+03 | 14.83% |
| **RB88** | **2.546E+05** | **2.241E+05** | **-12.00%** |
| **RB89** | **3.323E+05** | **2.900E+05** | **-12.72%** |
| **KR83M** | **3.796E+04** | **3.599E+04** | **-5.19%** |
| KR85 | 3.102E+03 | 4.760E+03 | 53.45% |
| **KR85M** | **8.844E+04** | **7.982E+04** | **-9.75%** |
| **KR87** | **1.786E+05** | **1.567E+05** | **-12.25%** |
| **KR88** | **2.525E+05** | **2.213E+05** | **-12.34%** |
| **KR89** | **3.207E+05** | **2.754E+05** | **-14.13%** |
| XE129M | 9.566E+00 | 9.744E+00 | 1.87% |
| XE131M | 2.711E+03 | 3.029E+03 | 11.74% |
| XE133 | 4.502E+05 | 5.243E+05 | 16.45% |
| XE133M | 1.490E+04 | 1.657E+04 | 11.21% |
| XE135 | 1.573E+05 | 2.028E+05 | 28.93% |
| XE135M | 9.684E+04 | 1.083E+05 | 11.88% |
| XE137 | 4.213E+05 | 4.590E+05 | 8.95% |
| XE138 | 4.380E+05 | 4.502E+05 | 2.78% |

Bolded isotopes indicate the licensing basis value bounds the EPU calculated value.

Both evaluations considered a radial peaking factor of 1.7.

| **Table 3 - Loss-of-Coolant Accident<br>UFSAR Sect. 15.6.5** | |
|---|---|
| **COR: Bases / Assumptions / Inputs** | **EPU Impact** |
| Core Power: 3910 MWt (100.3% of the current rated 3898 MWt) | EPU Core Power: 4408 MWt (115%of OLTP).<br><br>The EPU source terms include a 2% margin of uncertainty; i.e., the EPU source terms are based on 4496 MWt.<br><br>The fuel cycle length is 24 months. |
| Source Term Basis: The source term used in the COR is in accordance with Reg. Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," July 2000. | The source term basis (Reg. Guide 1.183) is not affected by EPU. |
| The COR analyses consider several release pathways:<br><br>Containment Leakage: The core source is released to the drywell and then transported to the primary containment. Elemental iodine plate-out and aerosol natural deposition in the drywell is credited. Activities in the primary containment leak to the secondary containment at a specified rate. Activities in the primary containment are removed by spray and by plate-out. The pH of the suppression pool is controlled to a value above 7. Therefore, re-evolution is not considered. Activities in the secondary containment are released via the SGTS with an effective filter efficiency of 98.975% (4001 cfm total flow rate). The secondary containment draw-down time is considered in the COR analysis.<br><br>MSIV Leakage: There is a 250 scfh leak rate for the first 24 hours, and 125 scfh after 24 hours. During the first 20 minutes post-LOCA, the release goes directly to the environment. After 20 minutes, the release goes to the secondary containment.<br><br>ESF Liquid Leakage outside Containment: Only halogens are released. The leak rate is 1.12 gpm (10 min to 30 days), and the minimum suppression pool volume is 170,954 $ft^3$. The iodine flash fraction is 10%. | EPU Impact on Each Pathway:<br><br>Containment Leakage: Not affected by EPU, except for SLCS. While the SLC system is being modified, it is still designed to maintain a suppression pool pH of greater than 7.0 taking into consideration the increased acid production due to EPU radiation environments.<br><br>MSIV Leakage: Not affected by EPU<br><br>ESF Liquid Leakage outside Containment: Not affected by EPU |

| Table 3 - Loss-of-Coolant Accident<br>UFSAR Sect. 15.6.5 | |
|---|---|
| **COR: Bases / Assumptions / Inputs** | **EPU Impact** |
| <u>Control Room Model</u>: The control room volume is $2.53 \times 10^5$ ft$^3$. The control room normal intake rate is 2000 cfm, which is taken in through the intake duct on the Control Building roof. There is an additional 10 cfm of inleakage into the control room due to ingress & egress. No additional inleakage is assumed during the first 20 minutes post-LOCA. Following isolation of intakes at 20 minutes post-LOCA (accomplished manually), the control room atmosphere is re-circulated through the control room air conditioning while 4000 cfm of this flow is drawn into the Control Room Fresh Air Supply (CRFAS) system and is passed through a 99% efficient HEPA filter before being discharged back into the HVAC system and returned to the control room. The post-isolation inleakage is 2010 cfm. | Not affected by EPU |

| Table 4 - Fuel Handling Accident<br>UFSAR Sect. 15.7.4 | |
|---|---|
| **COR: Bases / Assumptions / Inputs** | **EPU Impact** |
| <u>Core Power</u>: 3910 MWt (100.3% of the current rated 3898 MWt) | <u>EPU Core Power</u>: 4408 MWt (115%of OLTP).<br><br>The EPU source terms include a 2% margin of uncertainty; i.e., the EPU source terms are based on 4496 MWt.<br><br>The fuel cycle length is 24 months. |
| <u>Source Term</u>: The fuel bundle activities with radial peaking factor of 2.2 and decayed for 24 hours. | GEH has confirmed that the radial peaking factor is not affected by the EPU. Decay time prior to fuel movement is also not impacted by EPU. |
| <u>Worst Drop Scenario Considered</u>: Drop of an irradiated fuel assembly over the core, resulting in 142 fuel rod cladding failures, or 2.03 failed bundles for a 70-rod bundle (consistent with the fuel vendor's prediction). Although fuel pool water DF is credited, containment retention and SGTS filtration is not. | Increase of reactor core power does not affect the number of fuel bundles being damaged. |
| <u>Pool DF</u>: 200 | Not affected by EPU |
| <u>Control Room Data</u>: No control room isolation occurs and no control room fresh air system is actuated. The control room dose is independent of the control room volume and intake rate. | Not affected by EPU |
| <u>Fuel Bundle Activities Available for Release</u>:<br><br>The activity released reflects the noble gases and iodines in the fuel gap of the impacted fuel bundles with a radial peaking factor of 2.2 (a conservatively assumed value that bounds the peaking factor given in Reg. Guide 1.25, *Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors (Safety Guide 25)*) and decayed for 24 hours (the minimum time at which the accident is assumed to occur). Note that the alkali metals are excluded from the list because they are completely retained in the pool water. | Corresponding EPU gap inventory in the fuel bundle. |
| <u>Gap Fractions</u>: (RG 1.183, Table 3)<br>• I-131: 8%<br>• Kr-85: 10%<br>• Other NG & Halogens: 5%<br>• Alkali Metals: 12%<br>The continued applicability of the RG 1.183 gap fractions are confirmed by cycle-specific evaluations. | The gap fractions are from Table 3 of Reg. Guide 1.183 and subject to limits of burn-up and linear heat generation rate.<br>As per the COR, the continued applicability of the RG 1.183 gap fractions are confirmed by cycle-specific evaluations. |

| Table 5 - Control Rod Drop Accident<br>UFSAR Sect. 15.4.9 | |
|---|---|
| **COR: Bases / Assumptions / Inputs** | **EPU Impact** |
| <u>Core Power</u>: 3833 MWt (plus 3.8% margin for core power and power distribution uncertainty). | <u>EPU Core Power</u>: 4408 MWt (115% of OLTP).<br><br>The EPU source terms include a 2% margin of uncertainty; i.e., the EPU source terms are based on 4496 MWt.<br><br>The fuel cycle length is 24 months. |
| <u>No. of Fuel Bundles That Fail</u>: 16 (out of 800 fuel bundles in the core)<br><br>This is a conservative value that bounds the fuel vendor's calculated number of failed bundles. | GEH has confirmed that fuel failure assumptions in the COR remain valid for EPU. |
| <u>Fraction of Failed Fuel That Melts</u>: 0.0077<br>(Maximum fraction of damaged fuel that reaches melting temperature, per GEH's NEDO-31400A, *Safety Evaluation for Eliminating the Boiling Water Reactor Main Steamline Isolation Valve Closure Function and Scram Function of the Main Steamline Radiation Monitor*, October 1992.) | GEH has confirmed that fuel melt assumptions in the COR remain valid for EPU. |
| <u>Radial Peaking Factor</u>: 1.7<br>(A conservative value that bounds the peaking factor of 1.5 given in Appendix A, *Radiological Consequences of Control Rod Drop Accident (BWR),* of Standard Review Plan (SRP) 15.4.9, *Spectrum of Rod Drop Accidents (BWR)*, Rev. 2) | GEH has confirmed that the peaking factor in the COR remains valid for EPU. |
| <u>Activity Release Path</u>: noble gases, halogens and alkali metals from failed/melted fuel to RCS, transported to turbine/condensers via steam lines, partitioned to condenser steam space and available for release, leaked from condenser to atmosphere as a ground-level release | Not affected by EPU |
| <u>Fraction of Activity Released from Failed Fuel to RCS (Gap fraction)</u>:<br>• Noble gases – 10%<br>• Halogens – 10%<br>• Alkali metals – 12% | Remains valid since these fractions of activity released from failed fuel to the RCS are consistent with Reg. Guide 1.183 (Rev.0) values |
| <u>Fraction of Activity Released from Melted Fuel to RCS</u>:<br>• Noble gases – 100%<br>• Halogens – 50%<br>• Alkali metals – 25%<br>(Other group nuclides are ignored since their contribution is negligible) | Remains valid since these fractions of activity released from failed fuel to the RCS are consistent with Reg. Guide 1.183 (Rev.0) values |

| Table 5 - Control Rod Drop Accident<br>UFSAR Sect. 15.4.9 | |
|---|---|
| **COR: Bases / Assumptions / Inputs** | **EPU Impact** |
| Fraction of RCS Activity Transported to Turbine And Condensers: 100 % for noble gas, 10 % for halogen, 1% for the Remaining radionuclides | Not affected by EPU |
| Fraction of Turbine/Condenser Activity Available for Environmental Release: 100 % for noble gas, 10 % for halogen, 1% for the Remaining radionuclides | Not affected by EPU |
| Leak Rate from Condenser to Environment: 1% per day for 24 hours | Not affected, consistent with Reg. Guide 1.183 |
| Control Room Model:<br>• CR Volume: 2.53E+05 $ft^3$<br>• CRFA credited at t = 20 minutes<br>• During t < 20 minutes, unfiltered intake flow rate is 2010 cfm.<br>• During t ≥ 20 minutes, inleakage is 2010 cfm, recirculation rate is 4000 cfm, and HEPA efficiency is 99% for aerosols. | Not affected by EPU |

| Table 6 - Main Steam Line Break Outside Containment<br>UFSAR Sect. 15.6.4 | |
|---|---|
| **COR: Bases / Assumptions / Inputs** | **EPU Impact** |
| Fuel damage: None | GEH has confirmed that there is no fuel damage following EPU. |
| Activity Released: Technical Specification iodine & noble gas activities in RCS | Not affected by EPU. |
| Technical Specification iodine concentration limits:<br>• Steady state – 0.2 μCi/g Dose Equivalent I-131<br>• Spiking case – 4 μCi/g Dose Equivalent I-131<br>Technical Specification noble gas release rate limit:<br>• Gross noble gas activity release rate of less than or equal to 380 millicuries/sec release rate after 30 minutes decay. | These limits are Technical Specification limits. They are unaffected by the EPU. |
| Coolant Source Terms: Isotopic concentrations of noble gases and iodines in the reactor coolant and steam based on the maximum values allowed by GGNS Technical Specifications 3.4.8 & 3.7.5 (listed above). | EPU may change the relative isotopic compositions slightly. However, the small variation in the relative isotopic compositions, if any, will have an insignificant impact on the calculated doses. |
| Primary Coolant Released: It is assumed, based on SRP 15.6.4, *Radiological Consequences of Main Steam Line Failure Outside Containment (BWR),* that 27,750 lbm steam and 112,250 lbm liquid leave the break. These values have a conservative margin of 4.3% over the GEH-calculated values for GGNS. | The EPU does not change reactor dome pressure and steam line pressure & temperature. The release masses per the COR are not affected by EPU. |
| Control Room Model: See Table 3 (LOCA). | Not affected by EPU |

| Table 7 - Pressure Controller Failure<br>UFSAR Sect. 15.2.1 | |
|---|---|
| **COR: Bases / Assumptions / Inputs** | **EPU Impact** |
| Core Power: 3910 MWt (100.3% of the current rated 3898 MWt) | EPU Core Power: 4408 MWt (115%of OLTP).<br><br>The EPU source terms include a 2% margin of uncertainty; i.e., the EPU source terms are based on 4496 MWt.<br><br>The fuel cycle length is 24 months. |
| Source Term Basis: The source term used in the COR is in accordance with Reg. Guide 1.183. | The source term basis (Reg. Guide 1.183) is not affected by EPU. |
| Applicable Source Term: The source term considered in the Pressure Controller Failure accident analysis is comprised of those reactor core isotopes that reside in the fuel cladding gap, specifically, isotopes of bromine, krypton, rubidium, iodine, xenon, and cesium. The radiological consequence of a Pressure Controller Failure is based on an assumed core-wide gap failure. | Fuel failure assumption retained for EPU. |
| Control Room Model:<br>Control Room Volume is 2.53E+05 $ft^3$.<br><br>The CRFA System is credited at 20 minutes post-accident. Prior to 20 minutes, the total inflow is 4010 cfm (2000 cfm from normal intake, 2000 cfm from inleakage, and 10 cfm from ingress/egress). After 20 minutes, inleakage is 2010 cfm, recirculation is 4000 cfm, and HEPA filter efficiency is 99% for particulates. | Not affected by EPU |

| Table 8 - MSIV Closure<br>UFSAR Sect. 15.2.4 | |
|---|---|
| **COR: Bases / Assumptions / Inputs** | **EPU Impact** |
| Source Term: Consistent with GGNS Technical Specification 3.4.8, the reactor coolant and steam activity (μCi/g per isotope) used in the COR is based on the maximum permitted iodine spiking with a dose equivalent I-131 specific activity of 4.0 μCi/gram in the reactor coolant. | Since the Technical Specification limits are not affected, the only effect the EPU could have is a small impact on the relative isotopic compositions. This small variation in the relative isotopic compositions, if any, will have an insignificant impact on the calculated doses. |
| Reactor Vessel Fluid Mass: 6.815E+05 lbs. | A comparison of the pre-EPU reactor water mass with the EPU reactor water mass indicates that there is no significant change. |
| Steam Mass in Reactor & Steam Lines: 34,000 lbs. | The pre-EPU steam mass was calculated based on a reactor water level of 36-inches. The EPU reactor water level is between 32.7 inches and 40.7 inches. The pre-EPU water level falls within the EPU water level range. Therefore, a 36-inch water level remains applicable for EPU. |
| All activity in reactor water and steam is released to the suppression pool via the SRVs.<br>• Aerosols (95%) released via SRVs: DF=100<br>• Elemental Iodine (4.85%) released via SRVs: DF=10 | Not affected by EPU |
| Containment Ventilation: High volume containment purge is assumed to be operating (6000 cfm, both trains) when event occurs. The containment charcoal trains are credited to remove 99% of the elemental and organic iodine, and the HEPA filters are credited to remove 99% of the aerosols. | Not affected by EPU |
| Bounding χ/Q Value: $2.0 \times 10^{-5}$ sec/m$^3$ | Not affected by EPU |

| Table 9 - Misplaced Bundle Accident<br>UFSAR Sect. 15.4.7 | |
|---|---|
| **COR: Bases / Assumptions / Inputs** | **EPU Impact** |
| Core Power: 3910 MWt (100.3% of the current rated 3898 MWt) | EPU Core Power: 4408 MWt (115% of OLTP).<br><br>The EPU source terms include a 2% margin of uncertainty; i.e., the EPU source terms are based on 4496 MWt.<br><br>The fuel cycle length is 24 months. |
| Source Term Used in the Generic Misplaced Bundle Accident (a.k.a. FLE Accident) Analysis:<br>• 5 fuel bundles (the misplaced bundle plus the 4 surrounding bundles) are conservatively assumed to fail, releasing all gap activity to the RCS.<br>• Core average bundle power is 5.75 MWt.<br>• There is a safety factor of 1.4 to account for variations in fission product inventory over the operational cycle, and a safety factor of 2.5 to account for variations in cycle-dependent bundle power as ratio to the end of cycle average bundle power. | • Not affected by EPU.<br>• EPU core average bundle power is 5.62 MWt (based on total core power of 4496 MWt and 800 fuel bundles in the core)<br>• The 1.4 safety factor is retained for the EPU. GEH has confirmed that the 2.5 radial peaking factor assumption in the generic analysis bounds EPU operation. GEH has confirmed that the fuel failure assumption in the generic analysis bounds EPU operation. |
| Gap Activity Fraction:<br>• Noble gas – 10%<br>• Iodine – 10%<br>• Alkali metals – 20% | Not affected by EPU. |
| As described in Section S.5.3 of GESTAR II (NEDE-24011-P-A), the generic analysis demonstrates that the offsite and control room dose consequences of the FLE event have been confirmed to result in dose consequences that are bounded by the 10 CFR 50.67 acceptance criteria. The applicability of the generic analyses depends on plant-specific confirmations. For the offsite radiological analysis, GESTAR Scenario 2 (plants that do not have a main steam high radiation isolation trip) applies to GGNS. The radiological consequences depend on the 0-2 hour EAB $\chi/Q$ and the offgas system design. Using the applicable plant specific GGNS offgas system Krypton and Xenon holdup times would result in an offsite dose of approximately 2.0E-2 Rem TEDE, which is a small fraction of the acceptance criterion of 2.5 Rem TEDE.<br><br>For the control room, the generic FLE analysis demonstrates that the control room dose will be less than the 5.0 Rem TEDE acceptance criterion provided that the applicable control room $\chi/Q$ is < 1.25E-2 s/m$^3$. The GGNS specific control room $\chi/Q$ is well below the required value. | The generic analysis is based on an average fuel bundle power that exceeds the target EPU power level. The Kr and Xe holdup times in the Offgas System are not affected by EPU because the condenser air inleakage is not affected. Therefore, EPU will not change the conclusion of the COR and the EAB and control room doses will not exceed regulatory limits. |

| Table 10 - Offgas System Leak or Failure<br>UFSAR Section 15.7.1 | |
|---|---|
| **COR: Bases / Assumptions / Inputs** | **EPU Impact** |
| Core Power: 3910 MWt (100.3% of current rated 3898 MWt) | EPU Core Power: 4408 MWt (115%of OLTP).<br><br>The EPU source terms include a 2% margin of uncertainty; i.e., the EPU source terms are based on 4496 MWt.<br><br>The fuel cycle length is 24 months. |
| Assumed Event: Offgas System Failure, the resulting doses are due to combined releases from (1) charcoal adsorber failure, (2) delay line failure, and (3) continued operation of the steam jet air ejector (SJAE) | The offgas failure and release paths are not affected by EPU. The delay line transit time is not affected by power level. The charcoal bed hold-up time calculated in the COR is based on NUREG-0016 methodology and is not adversely affected by EPU. |
| Particulate Releases: The following particulates are released to the environment: rubidium (Rb-88, Rb-89), yttrium (Y-89m), Rb-90, barium (Ba-137m), cesium (Cs-138, and Cs-139). Particulates released are based on a power level of 3910 MWt. | The magnitudes of particular isotopes that are released to the environment are proportional to power level. The EPU scaling factor for particulates is 1.15 (4496 ÷ 3910). |
| Noble Gas Release Rate: The noble gas activity release rate is based on the maximum allowable offgas activity release rate reported in GGNS Technical Specification 3.7.5, which requires that the noble gas offgas release rate be no more than 380 mCi/sec after 30 minutes of decay. The total release rate used in the COR is 399 mCi/sec, which corresponds to a power level of 3990 MWt. | According to Reg. Guide 1.98, the activity release rate is directly proportional to power level. Given the noble gas release rate used in the COR and the target EPU power level, the current activity release rates would need to be scaled up by a factor of 1.13 (4496 ÷ 3990) for the noble gas source. Note that the impact of the change in fuel cycle length from 18 months to 24 months on long lived isotope Kr-85 is deemed insignificant since the release inventory of this isotope is orders of magnitude less than that of the remaining isotopes. A conservative EPU scaling factor of 1.15 will be used to be consistent with that applicable to the particulates. |
| Control Room Ventilation Parameters: See Table 3 (LOCA). | Not affected by EPU |

| Table 11 - Radioactive Liquid Waste System Leak or Failure (Release to Atmosphere)<br>UFSAR Section 15.7.2 | |
|---|---|
| **COR: Bases / Assumptions / Inputs** | **EPU Impact** |
| <u>Core Power</u>: 3833 MWt (OLTP)<br><br>The activities released are the radionuclides, in the Equipment Drain Collection Tank, listed below. The source term is calculated based on primary coolant activity concentrations based on a reactor thermal power of 3833 MWt.<br><br>I-131: 0.4243 Ci<br>I-132: 4.243 Ci<br>I-133: 2.857 Ci<br>I-134: 8.882 Ci<br>I-135: 4.243 Ci | <u>EPU Core Power</u>: 4408 MWt (115%of OLTP).<br><br>The EPU source terms include a 2% margin of uncertainty; i.e., the EPU source terms are based on 4496 MWt.<br><br>The fuel cycle length is 24 months.<br><br>Since only iodine isotopes are postulated to be released, the impact of the change in fuel cycle length is minimal |
| The current design-basis analysis is based on the failure of the limiting radwaste system vessel (the Equipment Drain Collection Tank). Since the analysis of record is not based on AST, the exposure guidelines are provided by 10 CFR 100 for EAB and LPZ, and 10 CFR 50, GDC-19 for the control room. The operative EAB and LPZ limits for this analysis are “a small fraction” (i.e., 10%) of 10 CFR 100 limits; i.e., 30 Rem to the thyroid and 2.5 Rem to the whole body. For the control room, the limits are 5 Rem to the whole body and 30 Rem to the thyroid. | Not affected by EPU |
| <u>Analysis Assumptions / Bases</u>:<br><br>• Radioisotope inventory in liquid radwaste system is based on normal system operation.<br>• Only radioiodine isotopes are released since noble gases are not present and particulate radioisotopes will not become airborne.<br>• The entire airborne iodine inventory is assumed to be in the elemental chemical species.<br>• No operator mitigation is assumed.<br>• Instantaneous release is assumed.<br>• No credit is taken for partition, filtration, holdup, or dilution of iodine once it is released from the failed tank. | Not affected by EPU |
| <u>Control Room Model</u>:<br><br>CR normal intake of 2000 cfm (+10% uncertainty) during the entire duration of the event, with no control room isolation or initiation of CRFAS. | Not affected by EPU |

| Table 12 - Liquid Radwaste Tank Failure (Release to Groundwater)<br>UFSAR Section 15.7.3 & 2.4.13.3 | |
|---|---|
| **COR: Bases / Assumptions / Inputs** | **EPU Impact** |
| <u>Core Power</u>: 3833 MWt (OLTP)<br><br>Fuel cycle length is assumed to be 12 months. This was the original fuel cycle basis upon which the GGNS operating license was initially granted. | <u>EPU Core Power</u>: 4408 MWt (115%of OLTP).<br><br>The EPU source terms include a 2% margin of uncertainty; i.e., the EPU source terms are based on 4496 MWt.<br><br>The fuel cycle length is 24 months. |
| Nuclides considered are Sr-90 and Cs-137 because they comprise the greatest potential health hazard in the event of an accidental spill. | These same two nuclides remain the nuclides of importance following EPU. |
| The maximum concentrations for Sr-90 and Cs-137 are 3.36E+01 µCi/cc and 3.56E+01 µCi/cc, respectively. These two concentrations are contained in two RWCU phase separator tanks. | The conservative EPU source scale-up factor for those two long-lived isotopes is (4496 ÷ 3833) × (24÷12), or 2.35. |