AmerGen

An Exelon/British Energy Company

Clinton Power Station
R.R. 3 Box 228
Clinton, IL 61727-9351
Phone: 217 935-8881

RS-01-225

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U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555-0001

Clinton Power Station, Unit 1
Facility Operating License No. NPF-62
NRC Docket No. 50-461

Subject: Additional Information Supporting the License Amendment Request to Permit Uprated Power Operation at Clinton Power Station

References: (1) Letter from J. M. Heffley (AmerGen Energy Company, LLC) to U.S. NRC, "Request for License Amendment for Extended Power Uprate Operation," dated June 18, 2001

(2) Letter from J. B. Hopkins (U.S. NRC) to O. D. Kingsley (Exelon Generation Company, LLC), "Clinton Power Station, Unit 1 – Request For Additional Information (TAC No. MB2210)," dated October 3, 2001

In Reference 1, AmerGen Energy Company, LLC (i.e., AmerGen) submitted a request for changes to the Facility Operating License No. NPF-62 and Appendix A to the Facility Operating License, Technical Specifications (TS), for Clinton Power Station (CPS) to allow operation at an uprated power level. The proposed changes in Reference 1 would allow CPS to operate at a power level of 3473 megawatts thermal (MWt). This represents an increase of approximately 20 percent rated core thermal power over the current 100 percent power level of 2894 MWt. The NRC, in Reference 2, requested additional information regarding the proposed changes in Reference 1. This letter provides the requested information pertaining to all the NRC questions except 3.1 and 3.2. Responses to NRC Questions 3.1 and 3.2 of Reference 2 will be provided separately.

A001

Should you have any questions related to this information, please contact Mr. T. A. Byam at (630) 657-2804.

Respectfully,

K. A. Ainger

K. A. Ainger
Director – Licensing
Mid-West Regional Operating Group

Attachments:

Affidavit
Attachment A: Additional Reactor Pressure Vessel Fluence Information Supporting the License Amendment Request to Permit Uprated Power Operation at Clinton Power Station
Attachment B: Additional Human Factors Information Supporting the License Amendment Request to Permit Uprated Power Operation at Clinton Power Station
Attachment C: Additional Reactor Systems Information Supporting the License Amendment Request to Permit Uprated Power Operation at Clinton Power Station
Attachment D: Additional Dose Assessment Information Supporting the License Amendment Request to Permit Uprated Power Operation at Clinton Power Station

cc: Regional Administrator – NRC Region III
NRC Senior Resident Inspector – Clinton Power Station
Office of Nuclear Facility Safety – Illinois Department of Nuclear Safety

STATE OF ILLINOIS	)	
COUNTY OF DUPAGE	)	
IN THE MATTER OF	)	
AMERGEN ENERGY COMPANY, LLC	)	**Docket Number**
CLINTON POWER STATION, UNIT 1	)	**50-461**

SUBJECT: Additional Information Supporting the License Amendment Request to Permit Uprated Power Operation at Clinton Power Station

AFFIDAVIT

I affirm that the content of this transmittal is true and correct to the best of my knowledge, information and belief.

K. A. Ainger

K. A. Ainger
Director – Licensing
Mid-West Regional Operating Group

Subscribed and sworn to before me, a Notary Public in and for the State above named, this 17th day of October, 2001.

Timothy Byam

Notary Public

* OFFICIAL SEAL *
Timothy A. Byam
Notary Public, State of Illinois
My Commission Expires 11/24/2001

ATTACHMENT A

Additional Reactor Pressure Vessel Fluence Information Supporting the License Amendment Request to Permit Uprated Power Operation at Clinton Power Station

1.0 Reactor Vessel Fluence

Question
1.1 The 20 percent power increase will result in a neutron source greater than 20 percent. You state in section 3.3.1 of the submittal that the extended power uprate (EPU) fluence is bounded by the (CLTP) fluence. Please explain the analysis which supports these findings. Was the computer Code used in the fluence calculations approved by the Nuclear Regulatory Commission (NRC) staff?

Response
1.1 The current licensed thermal power (CLTP) flux, and consequently fluence, is a conservative estimate that is based on the first cycle dosimeter flux wire measurements. Since the flux wire dosimeter is attached to the surveillance capsule, the measurement is divided by the surveillance capsule lead factor to determine the peak reactor pressure vessel (RPV) inside diameter (ID) surface flux. For Clinton Power Station (CPS) a generic lead factor was used. In addition, the CLTP flux includes a 25% uncertainty factor for the flux wire measurement resulting in an upper bound flux estimate.

The extended power uprate (EPU) flux is calculated using a neutron transport calculation rather than flux wire results. As discussed in Attachment E to Reference 1, Power Uprate Safety Analysis Report (PUSAR) Section 3.3.1, the neutron transport calculation uses methodology presented in Reference 2, which was approved by the Nuclear Regulatory Commission on September 14, 2001. This methodology incorporates the realistic geometric model that removed unnecessary conservatisms in the calculations. In addition, the EPU reload uses GE14 fuel design containing partial length rods that lower the upper core flux, hence shifting the peak flux to a lower elevation. The combination of upper bound flux wire measurements for CLTP and new methodology plus new fuel design for EPU resulted in a lower EPU peak flux than the estimated peak flux for CLTP.

Question
1.2 From Section 3.3.1(e), the staff concludes that Clinton Power Station (CPS) does not have any plant-specific irradiated material testing. Please describe the calculations used to conclude that the upper shelf energy will remain above 50 ft-lb at the end of the current license.

Response
1.2 Certified Material Test Report (CMTR) data for CPS beltline plates and welds were used. Typically there are three Charpy Impact values at ≥ 95% shear that can be used to evaluate upper shelf energy (USE). These three values are averaged to determine the USE at beginning-of-life (BOL). Figure 2 in Regulatory Guide 1.99, "Radiation Embrittlement of Reactor Vessel Materials," Revision 2, May 1988, is used to determine the percent decrease in USE that is predicted for the ¼-thickness fluence experienced by the material. From this, the end-of-life (EOL) USE is determined. From Figure 2 of Regulatory Guide 1.99, Revision 2, no irradiated material test measurements are

required to make the EOL USE prediction. The results of the USE evaluation showed that the USE remains above 50 foot-pounds for both the CLTP and EPU conditions.

Question

1.3 It is stated that "The decrease in the EPU fluence despite the increase in the core thermal power was the result of a more realistic lead factors..." Lead factors are normally associated with capsule irradiation, yet CPS has not removed any surveillance capsules. Please explain the meaning of the lead factors discussed in the submittal.

Response

1.3 A lead factor relates the flux at the flux wires to the peak vessel flux and is defined as the ratio of surveillance capsule flux to the peak flux at the RPV ID surface. This same lead factor will also relate the flux measurement from the first cycle dosimeter that is attached to the surveillance capsule, as discussed in the response to Question 1.1, to the peak RPV ID surface flux. While the lead factor is a function of core and vessel configurations, it is also dependent on the location of the capsule relative to other reactor internal components. As explained in the response to Question 1.1 above, the lead factor used for CLTP was a generic (i.e., not CPS-specific) lead factor.

The EPU lead factor is derived based on a CPS-specific flux distribution, which is calculated with the recently approved methodology. The EPU lead factor is more realistic because the jet pump components in the downcomer region are correctly modeled in the calculation. Since the capsule azimuth and the azimuth of RPV peak flux are not the same, the effect of jet pump shadowing is not the same for the peak flux and for the capsule. As a result, the calculated lead factor could change if different downcomer models were used in the calculations. For CPS, the EPU lead factor differs from that of CLTP because the CLTP value was not specifically calculated for CPS.

The flux wire measurements for EPU are not used to determine the fluence. Therefore, the lead factor determined for EPU will not be used to determine the EPU fluence, but rather the fluence at the RPV ID surface is determined directly from the approved calculation methodology. The EPU lead factor is important for future surveillance capsule measurements, when the surveillance capsule flux wire measurements are compared to the flux prediction determined using the approved calculation methodology. This practice is consistent with the methodology of Regulatory Guide 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence," March 2001.

ATTACHMENT A

Additional Reactor Pressure Vessel Fluence Information Supporting the License Amendment Request to Permit Uprated Power Operation at Clinton Power Station

REFERENCES

1. Letter from J. M. Heffley (AmerGen Energy Company, LLC) to U.S. NRC, "Request for License Amendment for Extended Power Uprate Operation," dated June 18, 2001

2. Licensing Topical Report, "GE Methodology to RPV Fast Neutron Flux Evaluations," NEDC-32983P, Class III (Proprietary), August 2000

ATTACHMENT B

Additional Human Factors Information Supporting the License Amendment Request to Permit Uprated Power Operation at Clinton Power Station

2.0 Human Factors and Operator Performance

Question
2.1 *Changes in Emergency and Abnormal Operating Procedures*

Describe how the proposed power uprate will change the plant emergency and abnormal procedures.

Response
2.1 The Emergency Operating Procedures (EOPs) remain symptom-based and thus the operator actions remain unchanged. The effect of Extended Power Uprate (EPU) on the EOPs is limited to revisions to previously defined numerical values or inputs to the emergency procedure guideline calculations. These are expected to result in minor changes to EOP figures and limitations.

No significant Abnormal Operating Procedure (AOP) revisions are expected. However, all AOPs are reviewed for EPU conditions and necessary revisions will be completed prior to EPU implementation.

These emergency and abnormal procedure changes will be addressed during operator training sessions prior to operation at EPU conditions.

All normal operating procedures are presently under review for revision. Procedures requiring revision will be presented during operator training sessions prior to EPU power ascension, as required.

Question
2.2 *Changes to Risk-Important Operator Actions Sensitive to Power Uprate*

Describe any new risk-important operator actions required as a result of the proposed power uprate. Describe changes to any current risk-important operator actions that will occur as a result of the power uprate. Explain any changes in plant risk that result from changes in risk-important operator actions.

(i.e., Identify and describe operator actions that will require additional response time or will have reduced time available. Your response should address any operator workarounds that might affect these response times. Identify any operator actions that are being automated as a result of the power uprate. Provide justification for the acceptability of these changes.)

Response
2.2 The response to this question has been included in supplemental information provided in Reference 1 in support of the license amendment request to permit uprated power operation at CPS (i.e., Reference 2).

ATTACHMENT B

Additional Human Factors Information Supporting the License Amendment Request to Permit Uprated Power Operation at Clinton Power Station

Question
2.3 *Changes to Control Room Controls, Displays and Alarms*

> *Describe any changes the proposed power uprate will have on the operator interfaces for control room controls, displays and alarms. For example, what zone markings (e.g., normal, marginal and out-of-tolerance ranges) on meters will change? What set points will change? How will the operators know of the change? Describe any controls, displays and alarms that will be upgraded from analog to digital instruments as a result of the proposed power uprate and how operators were tested to determine they could use the instruments reliably.*

Response
2.3 The primary impacts of EPU on main control room (MCR) operation involve changes to the power-to-flow relationship. There are no major physical changes required to the MCR controls, displays, or alarms as a result of EPU. Some changes are required to MCR panel board indicator spans, alarm settings, and automatic actuation setpoints to accommodate changes in process conditions due to EPU. In addition, the existing zone banding (i.e., green, yellow, and red) on all MCR panel board indications are being reviewed for acceptability and will be revised as necessary prior to EPU operation.

The only required MCR panel board meter changes are on the turbine generator control panel. This panel requires replacement of the intermediate pressure, load, and load set meters with meters that have revised red line scales representative of new limits associated with the EPU.

The following setpoints are also being changed as a result of EPU. These setpoint changes are described in Section 5 of Appendix E to Reference 3.

a) Average Power Range Monitor Flow-Biased Scram and Rod Block Setpoints
b) Main Steam Line High Flow Group 1 Main Steam Isolation Valve Isolation Setpoints
c) Turbine Stop Valve Closure and Turbine Control Valve Fast Closure Scram Bypass and Recirculation Pump Trip Bypass Setpoints
d) Rod Pattern Control System Low Power Setpoint
e) Rod Pattern Control System Rod Withdrawal Limiter High Power Setpoint

Other minor changes to controls, displays or alarms that have been identified or are currently under impact review include the following.

- Increasing the calibrated span for the turbine driven reactor feed pump (TDRFP) suction header temperature to support the increased suction temperature. The suction header temperature is a computer display point only.

- Lowering the TDRFP low suction pressure alarm to accommodate the lower suction pressures associated with EPU.

- Changing the calibration span of the steam chest and intermediate steam pressure transmitters.

Changes to the automatic actuation setpoints, calibration spans and alarm points are implemented as design changes in accordance with approved change control procedures. The change control process includes an impact review by operations and training personnel. Training and implementation requirements are identified and tracked, including simulator impact. Verification of training is required as part of the design change closure process.

There are no planned upgrades of controls, displays or alarms from analog to digital instruments as part of EPU.

Question
2.4 *Changes on the Safety Parameter Display System*

Describe any changes the proposed power uprate will have on the Safety Parameter Display System. How will the operators know of the changes?

Response
2.4 The analog and digital inputs for the Safety Parameter Display System (SPDS) were reviewed to determine the impacts from EPU. The inputs to the SPDS are not affected by EPU. This change will not affect human factors because the display and function of the system are unchanged.

Question
2.5 *Changes to the Operator Training Program and the Control Room Simulator*

Describe any changes the proposed power uprate will have on the operator training program and the plant reference control room simulator, and provide the implementation schedule for making the changes.

Response
2.5 An operator lesson plan will be developed to teach plant changes as a result of the EPU and existing lesson plans will be revised to reflect the changes. The proposed plant changes are described in the Power Uprate Safety Analysis Report (PUSAR). The EPU lesson plan will be presented to licensed/certified operations personnel before startup is initiated for operating at extended power conditions. EPU changes will be incorporated in continuing training lesson plans, as applicable.

Operator training for power uprate conditions will be performed on the simulator prior to operating at uprated conditions. This training will consist of a comparison of the plant conditions between the current maximum power level and the uprated power level, the normal operating procedural actions to achieve the uprated power level, and selected transients and accidents that present the greatest change from previous power levels. In addition, training for the EPU startup testing program will be performed using "just in

time" training of plant operations personnel. This will ensure the continuity of specific startup test training for operations personnel over the duration of the startup test program.

The plant simulator will contain a software module that reflects the major plant systems and reactor changes as a result of EPU. This module will be used for test preparation and operator training conducted prior to EPU implementation. These initial simulator changes will be implemented prior to the licensed-operator requalification training session before power uprate is initiated. Simulator revalidation will be accomplished in two stages. First, the simulator performance will be validated against the EPU expected system response. Second, post-startup data will be collected and compared with simulator performance data, allowing any necessary adjustments to simulator model performance. The simulator performance validation for EPU will be performed in accordance with American National Standards Institute (ANSI)/American Nuclear Society (ANS) 3.5-1993, Section 5.4.1 "Simulator Performance Testing."

ATTACHMENT B

Additional Human Factors Information Supporting the License Amendment Request to Permit Uprated Power Operation at Clinton Power Station

REFERENCES

1. Letter from K. A. Ainger (Exelon Generation Company, LLC) to U.S. NRC, "Supplemental Information Supporting the License Amendment Request to Permit Extended Power Uprate Operation at Clinton Power Station, " dated September 28, 2001

2. Letter from J. B. Hopkins (U.S. NRC) to O. D. Kingsley (Exelon Generation Company, LLC), "Clinton Power Station, Unit 1 – Extended Power Uprate (TAC No. MB2210)," dated July 30, 2001

3. Letter from J. M. Heffley (AmerGen Energy Company, LLC) to U.S. NRC, "Request for License Amendment for Extended Power Uprate Operation," dated June 18, 2001

ATTACHMENT C

Additional Reactor Systems Information Supporting the License Amendment Request to Permit Uprated Power Operation at Clinton Power Station

Question

3.3 Confirm (Explain) that AmerGen performed technical /quality assurance audits of General Electric Company support for CPS EPU application.

Response

The evaluations and reviews performed in support of the extended power uprate (EPU) and the corresponding safety analysis report (i.e., Appendix E to Reference 1) were documented in specific EPU project Task Reports. Each EPU Task Report, as it was completed, was reviewed in accordance with Clinton Power Station (CPS) Engineering Instruction DE-32, "Instruction for the Review of Task Reports for the Extended Power Uprate (EPU) Project." This process ensured a thorough, documented technical review. Subsequently, each EPU Task Report was reviewed by CPS personnel for impact assessment to identify documents, procedures or hardware that may be impacted by the methods, conclusions, or results.

In April 2001, CPS performed a technical review audit of selected General Electric (GE) Company Design Record Files (DRFs) specific to 10 EPU Task Reports. The audit resulted in generic as well as task specific findings that were identified to GE for corrective action. An input to this audit included the results of a previous audit by personnel from the Duane Arnold Energy Center of the GE DRFs. In addition, Exelon's Nuclear Fuel Management personnel performed an audit of GE during the Dresden Nuclear Power Station and Quad Cities Nuclear Power Station EPU projects.

GE is a "Qualified Supplier" in accordance with10 CFR 50, "Domestic Licensing of Production and Utilization Facilities," Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," requirements. Periodic vendor audits by the CPS Quality Assurance organization are conducted to verify compliance with applicable requirements.

REFERENCE

1. Letter from J. M. Heffley (AmerGen Energy Company, LLC) to U.S. NRC, "Request for License Amendment for Extended Power Uprate Operation," dated June 18, 2001

ATTACHMENT D

Additional Dose Assessment Information Supporting the License Amendment Request to Permit Uprated Power Operation at Clinton Power Station

4.0 Radiological Consequences

Question

4.1 In order to make a finding regarding the acceptability of the proposed EPU, the staff must make a finding in regard to offsite doses (10 CFR 100.11) and control room doses (10 CFR 50 Appendix A, GDC-19). The submittal only addresses control room doses for the design-basis accident (DBA) loss-of-coolant accident (LOCA). No discussion of the EPU impact on the control room doses for the other DBAs is provided.

4.1a Since GDC-19 requires that adequate radiological protection be provided for all accidents, including the LOCA, please provide a statement regarding the acceptability of the EPU impact on control room habitability for all DBA's currently analyzed for CPS (i.e., that GDC-19 will be met for all accidents in the CPS design basis).

4.1b If your position is that the control room dose for a LOCA is bounding, that statement should be made in your docketed response. In support of this position, please provide a basis for this conclusion. Include in your justification (1) the impact of different release modes (e.g., ground level vs. elevated releases) for the various accidents, (2) the impact of release point location in relation to the control room intake for the various accidents, (3) differences in release pathway filtration and other mitigation, and (4) differences in the means and timing of the actuation of control room isolation/filtration. The NRC staff's experience in reviewing license amendment requests indicates that these considerations can often make other accidents more limiting with regard to control room habitability.

Response

4.1a The extended power uprate (EPU) impact on control room habitability is only analyzed for the loss of coolant accident (LOCA) for Clinton Power Station (CPS) because the LOCA is the bounding accident.

4.1b Control room doses are only addressed for the design basis accident (DBA) LOCA because it is the bounding accident for control room habitability. Since the same control room dose limits apply to all accidents, to demonstrate the LOCA is bounding, the activity released to the environment following the LOCA is compared to the activity released to the environment following the other DBAs. The use of activity released to the environment to compare the different accidents eliminates the need to evaluate differences in release path filtration and other mitigation. At CPS, all releases are considered ground level releases, so the release mode is the same for all accidents. In addition, the control room ventilation emergency filtration system is initiated on high radiation readings at the control room intake, so the actuation of this system will be the same for all accidents. The only differences between the LOCA and the other accidents will therefore be caused by differences in activity released and any difference in atmospheric dispersion caused by a difference in the release point. Table 4.1-1 contains the activity released to the environment for each accident. Note that this is the activity released under current design conditions (i.e., not EPU), but the following qualitative assessment of the relative impact of other design basis accidents compared to the LOCA is valid for EPU.

ATTACHMENT D

Additional Dose Assessment Information Supporting the License Amendment Request to Permit Uprated Power Operation at Clinton Power Station

Comparison of the activity released from the control rod drop accident (CRDA) over the first 24 hours with the same time period for the LOCA (considering individual nuclides and the total) indicates the iodine activity released following a LOCA is a factor of 400 higher than the CRDA. A similar difference exists in the noble gas activity. Since the control room ventilation (CRV) emergency filtration system is initiated by a high radiation signal sensed at the CRV intakes, the only mechanism that would cause the CRDA dose to be higher than the LOCA would be less atmospheric dispersion. Following a LOCA, activity released from both the containment and the standby gas treatment system is modeled as a ground level release into the wake created by the containment. Following a CRDA, the activity is released from the condenser in the turbine building, which is also a ground level release but is farther from the CRV intakes than the containment. Therefore, the factor of 400 difference in the activity released is more than adequate to compensate for any change in the dispersion factor. As a result, the LOCA dose will bound the CRDA dose.

The activity release following a main steam line break (MSLB) occurs over a relatively short time period. A comparison of the iodine activity released following the MSLB to the iodine activity released during the first two hours following a LOCA indicates the LOCA activity is a factor of 180 larger than the MSLB. The LOCA noble gas activity is more than a factor of 10,000 larger than the MSLB. Since the CRV emergency filtration system will start automatically on a high radiation at the CRV intake, a significant decrease in atmospheric dispersion is the only condition that would cause the MSLB impact to exceed the LOCA. The main steam tunnel is adjacent to containment, so releases following the MSLB will be ground level releases into the wake of the containment, creating dispersion conditions that are similar to the LOCA. The dispersion following the MSLB is likely to be better than a LOCA if the rapidly expanding nature of the steam is also taken into account. The factor of 180 will compensate for differences due to the different release point. Also, the iodine activity released during the first two hours of the LOCA is less than 25% of the total iodine activity released during the LOCA, which will further increase the LOCA dose over the MSLB dose. Therefore, the LOCA control room doses will bound the MSLB.

The iodine activity released following a fuel handling accident (FHA) is a factor of 10 lower than the iodine activity released following the MSLB, so it follows that the LOCA is bounding for the thyroid dose. The noble gas activity released during the FHA is only a factor of 6 lower than the noble gas activity released during the first two hours following a LOCA. However, the total noble gas release following the LOCA is more than a factor of 160 larger than the total noble gas release following the FHA. This indicates the whole body and beta skin doses following the LOCA will bound the same doses following the FHA. The release pathway for the FHA is the same as the LOCA so the atmospheric dispersion will be comparable, and the radiation monitors will activate the CRV emergency filtration.

For the main steam line isolation valve (MSLIV) closure event and for the gaseous and liquid radwaste system failures, the total activity released is about a factor of 1000

smaller than the LOCA. The one exception is the noble gas releases following a gaseous radwaste system failure, which are more than a factor of 60 less than comparable releases for a LOCA. This difference is large enough to compensate for differences in atmospheric dispersion. Therefore, the control room impact from these accidents will be bounded by the LOCA.

ATTACHMENT D

Additional Dose Assessment Information Supporting the License Amendment Request to Permit Uprated Power Operation at Clinton Power Station

Table 4.1-1
Activity Released to the Environment Following Design Basis Accidents (Ci)

Isotope	LOCA 2 hour	LOCA 24 hour	LOCA 30 day	CRDA 24 hour	MSLB 2 hour	FHA 2 hour	MSLIV Closure 24 hour	Gaseous Radwaste 2 hour	Liquid Radwaste 2 hour
I-131	2.16E+3	5.59E+3	4.12E+4	1.4E+1	1.80E+0	2.6E+0	8.68E-2	1.0E-2	1.6E+1
I-132	2.68E+3	3.11E+3	3.11E+3	3.4E+0	1.80E+1	3.2E+0	3.55E-4	1.3E-1	5.8E-2
I-133	4.44E+3	9.43E+3	1.40E+4	1.5E+1	1.20E+1	1.7E+0	2.15E-2	7.6E-2	4.3E-1
I-134	3.33E+3	3.43E+3	3.43E+3	2.3E+0	3.86E+1	1.7E-6	1.17E-4	2.3E-1	3.2E-4
I-135	4.02E+3	6.26E+3	6.51E+3	1.3E+1	1.80E+1	4.7E-1	2.34E-3	1.2E-1	3.4E-1
Total I	1.66E+4	2.78E+4	6.82E+4	4.8E+1	8.83E+1	8.0E+0	1.11E-1	5.7E-1	1.7E+1
Kr-83m	8.34E+3	1.24E+4	1.24E+4	2.8E+1	6.95E-2	1.5E+1	1.71E+0	3.1E+1	
Kr-85m	2.07E+4	5.20E+4	5.31E+4	1.6E+2	1.22E-1	2.5E+2	4.11E+0	7.0E+1	
Kr-85	1.04E+3	7.75E+3	1.96E+5	1.2E+1	4.75E-4	6.8E+2	1.70E-1	1.3E+0	
Kr-87	3.08E+4	3.88E+4	3.88E+4	9.7E+1	3.79E-1	4.0E-2	7.12E+0	1.4E+2	
Kr-88	5.29E+4	9.84E+4	9.86E+4	3.1E+2	3.89E-1	7.7E+1	1.12E+1	1.9E+2	
Kr-89	4.53E+3	4.53E+3	4.53E+3	7.4E+0	1.62E+0	1.2E-3	2.94E+0	1.3E+3	
Xe-131m	5.44E+2	3.96E+3	5.08E+4	1.0E+1	3.88E-4	1.9E+2	1.30E-1	1.4E+1	
Xe-133m	7.86E+3	5.14E+4	1.80E+5	3.3E+2	5.81E-3	6.2E+3	6.46E-1	2.2E+1	
Xe-133	1.90E+5	1.34E+6	9.62E+6	2.2E+3	1.63E-1	4.3E+4	1.56E+1	1.5E+3	
Xe-135m	9.59E+3	9.60E+3	9.60E+3	1.0E+1	4.76E-1	2.9E+2	3.05E+1	1.8E+2	
Xe-135	2.33E+4	9.01E+4	1.06E+5	8.8E+1	4.39E-1	1.2E+4	2.40E+1	2.3E+2	
Xe-137	1.17E+4	1.17E+4	1.17E+4	1.3E+1	2.14E+0	2.4E-3	4.03E+0	1.5E+3	
Xe-138	3.96E+4	3.96E+4	3.96E+4	5.1E+1	1.62E+0	6.6E-3	8.11E+0	7.4E+2	
Total NG	4.00E+5	1.76E+6	1.04E+7	3.3E+3	7.42E+0	6.2E+4	1.10E+2	5.9E+3	

ATTACHMENT D

Additional Dose Assessment Information Supporting the License Amendment Request to Permit Uprated Power Operation at Clinton Power Station

Question

4.2 Table 9-2 provides the dose results for the LOCA. The doses identified in this table as "current" are different from the values documented in Table 15.6.5-6 of the CPS USAR.

		USAR Table 15.6.5-6	*Table 9-2 Current*	*Table 9-2 EPU*
EAB				
	Whole Body	*4.4*	*11*	*13.5*
	Thyroid	*163*	*225*	*267*
LPZ				
	Whole Body	*1.7*	*3.5*	*4.5*
	Thyroid	*156*	*86*	*102*
Control Room				
	Whole Body	*2*	*3*	*3.5*
	Thyroid	*27*	*25*	*29*

4.2a Please explain the source of the values identified as "current." Please explain why the exclusion area boundary (EAB) thyroid dose shows an increase of about 40 percent, when the LPZ (low population zone) dose shows a decrease of 45 percent and the control room thyroid dose shows a decrease of about 7 percent. Your submittal did not identify any changes to the design basis that would account for the observed differences in the reported doses.

4.2b Please provide a tabulation of all EPU analysis inputs and assumptions for the LOCA in sufficient detail to enable the staff to evaluate the acceptability of these assumptions, and as necessary, perform confirming EAB, LPZ, and control room dose calculations.

4.2c Please identify any changes to prior design basis inputs, assumptions, and methodologies, including offsite and control room atmospheric dispersion coefficients, incorporated in these re-analyses.

4.2d If the atmospheric dispersion coefficients documented in the USAR were revised, please identify the methodology used and all inputs and assumptions, and provide a computer readable file of the hourly meteorology data or joint frequency data (as appropriate) used in your reanalysis.

4.2e Please provide a justification for control room unfiltered inleakage assumptions that have not been substantiated by appropriate integrated boundary leakage testing.

Response

4.2a The "current" values of Table 9-2 of Appendix E to Reference 1 are supported by the values contained in Revision 9 of the Updated Safety Analysis Report (USAR). Revision 9 of the USAR was previously submitted to the NRC in Reference 2. Revision 9 of the USAR revised Table 15.6.5-6 values based upon the NRC approval and

subsequent implementation of the Feedwater Leakage Control System (FWLCS) modification. The change to the FWLCS was approved by Amendment Number 127 to the CPS Operating License in a Safety Evaluation Report dated April 25, 2000 (Reference 3). The addition of the FWLCS created an additional pathway to the environment following the LOCA, which increased the amount of noble gas activity released to the environment at the beginning of the LOCA. This resulted in an increase in the whole body doses, as indicated by the difference in the "USAR" and "Current" values referenced in the question above. In the re-analysis for the addition of the FWLCS, credit was taken for suppression pool scrubbing and the dose conversion factors for iodine were based on the International Commission on Radiological Protection, "Limits for Intakes of Radionuclides by Workers," ICRP-30. These changes were made based on regulatory guidance developed since the initial approval of the CPS Safety Analysis Report. The effect of these changes, as indicated by the difference in the "USAR" and "Current" values referenced in the question above, was a net decrease in the thyroid dose for the low population zone (LPZ) and the control room. For the exclusion area boundary (EAB) dose, which occurs during the first two hours, the thyroid dose increased because the addition of the FWLCS has the biggest dose impact at the beginning of the accident.

4.2b The assumptions and data used to evaluate the radiological consequences of a LOCA are summarized in Tables 4.2-1 and 4.2-2 below. Note that these assumptions and data are the same as those used to support the addition of the FWLCS with the exception of the initial core inventory, which increases for EPU because of the increase in core power.

4.2c The only change to the prior design basis inputs, assumptions and methodology required for EPU was the change in source term. Table 4.2-2 below summarizes the "prior to" EPU and the EPU source term, which increases due to the increase in core power. There were no changes in offsite and control room atmospheric dispersion coefficients.

4.2d The atmospheric dispersion coefficients documented in the USAR were not revised.

4.2e The control room ventilation system is designed for zone isolation with filtered recirculation and zone pressurization (i.e., a type 3 system as described in Section 6.4, "Control Room Habitability System," Standard Review Plan, NUREG-0800). The zone pressurization precludes unfiltered inleakage into the control room emergency zone. Although integrated boundary leakage testing has not been conducted at CPS, a combination of design and surveillance testing assures that the inleakage into the control room emergency zone will not exceed the assumptions used in the dose assessment.

The assessment of the dose to the control room operators assumes two sources of outside air inleakage. The major source of inleakage is 650 cubic feet per minute (cfm) into the ductwork upstream of the return fan. This ductwork is outside the control room envelope and is at a negative pressure relative to the outside atmosphere. As required in Technical Specification 3.7.3, "Control Room Ventilation System," this section of

ductwork is tested on a regular basis to assure the inleakage does not exceed 650 cfm. This inleakage is into the return duct and passes through the recirculation filter. Therefore, this is filtered inleakage.

The remaining ductwork and the control room emergency zone is maintained at a positive pressure relative to all adjacent areas. This is verified by periodic testing to assure the pressure differential with all adjacent areas is greater than 1/8-inch water gauge. The control room also contains double vestibule doors to prevent inleakage during access and egress. Although no inleakage is expected, 10 cfm of unidentified, unfiltered inleakage is assumed in the dose assessment to assure the assessment is conservative.

Additional Dose Assessment Information Supporting the License Amendment Request to Permit Uprated Power Operation at Clinton Power Station

Table 4.2-1
Dose Assessment Assumptions

1.	100% of the noble gases and 25% of the iodines of the core inventory are instantaneously airborne in the drywell and are available for release to the environment. When credit is taken for suppression pool scrubbing (Decontamination Factor = 6.667, 0.0158 pool bypass fraction), the iodine is reduced to 15% of this 25% (3.75% of total iodine inventory)
2.	No mixing in the secondary containment.
3.	Containment and main steam isolation valve (MSIV) source terms are uniformly mixed in the drywell and primary containment free volumes.
4.	During the draw down time (300 seconds) all the leakage is released unfiltered, i.e., 100% standby gas treatment system (SGTS) bypass.
5.	MSIV leakage starts two (2) hours after shutdown and 100% of the MSIV leakage is filtered by the SGTS.
6.	After draw down time is reached (300 seconds), 92% of the containment leakage is filtered by the SGTS and 8% of the containment leakage bypasses the secondary containment and the SGTS.
7.	Because of the height of the SGTS stack with respect to the containment building, releases from this pathway are at ground level.
8.	Containment atmosphere leaks at a constant rate through the feedwater isolation valves equal to the tested leak rate for the feedwater check valves until the feedwater line is filled (1 hour).
9.	The leakage through the feedwater valves is released unfiltered from the plant stack, which is collocated with the SGTS stack. Therefore, the atmospheric dispersion factors for the SGTS are used for these releases.
10.	Suppression pool water is conservatively assumed to leak through the feedwater check valves from the beginning of the accident for 30 days.
11.	The suppression pool source is 50% of the core inventory of iodines uniformly distributed in a volume of water equal to the minimum suppression pool volume (146,400 cubic feet).

Table 4.2-1
Dose Assessment Assumptions

12.	A partition factor of 73.4 is used for the iodine in the suppression pool water, i.e., 1.36% of the activity in the suppression pool that leaks through the feedwater isolation valves becomes airborne and is available for release from the plant.
13.	The containment pressure and temperature are assumed to be 9 pounds per square inch gauge (psig) and 200°F, respectively, for the entire period of the accident.
14.	The control room filtered inleakage is 650 cfm. This is consistent with Section 15.6.5.5.3 of the USAR.
15.	The duration of the accident is 30 days.
16.	The suppression pool water temperature is assumed to be 80°C (176°F). This is consistent with Figure 2.8 of the USAR.
17.	The control room emergency ventilation system is assumed to have a single failure and manual start 20 minutes after the onset of the accident.
18.	There is a constant 10 cfm unfiltered inleakage into the control room
19.	The filtered and unfiltered inleakage enter the control room at the makeup inlet, i.e., the same atmospheric dispersion factor as the makeup flow is applicable.
20.	Control room ventilation data same as USAR with exception of a 20-minute delay to start.
21.	0.65% per day containment leak rate directly to the environment for the first 300 seconds, and 0.052% per day thereafter.
22.	Containment free volume = 1.512E+6 cubic feet.
23.	Drywell free volume = 241,000 cubic feet.
24.	MSIV leak rate = 28 standard cubic feet per hour per line.
25.	Suppression pool water volume = 146,400 cubic feet (Technical Specification minimum).

Additional Dose Assessment Information Supporting the License Amendment Request to Permit Uprated Power Operation at Clinton Power Station

Table 4.2-2
Shutdown Core Inventory for Dose Assessment

Isotope	Core Inventory Prior to EPU (Ci)	EPU Core Inventory (Ci)
Kr-83m	9.53E+06	1.16E+07
Kr-85	9.16E+05	1.12E+06
Kr-85m	2.05E+07	2.43E+07
Kr-87	3.93E+07	4.64E+07
Kr-88	5.56E+07	6.55E+07
Kr-89	6.92E+07	7.97E+07
Xe-131m	4.81E+05	1.07E+06
Xe-133	1.68E+08	1.90E+08
Xe-133m	7.00E+06	6.09E+06
Xe-135	2.17E+07	6.20E+07
Xe-135m	3.17E+07	3.83E+07
Xe-137	1.47E+08	1.69E+08
Xe-138	1.40E+08	1.59E+08
I-131	8.00E+07	9.56E+07
I-132	1.17E+08	1.39E+08
I-133	1.67E+08	1.94E+08
I-134	1.84E+08	2.13E+08
I-135	1.58E+08	1.82E+08

Question
4.3 In the NEDC evaluations previously submitted for other boiling-water reactors, core inventories were recalculated using ORIGEN2 to address current fuel design, burnup and enrichment. These discussions note that use of the earlier Ci/MWt values based on TID 14844 do not properly account for the difference in U-235 and Pu-239 fission product yields associated with higher burnup fuels. The NRC staff believes this to be a valid concern. The TID 14844 values, issued in 1962, reflected the low enrichment, low burnup fuels in use at that time. The staff notes that an NEDC analysis for a 17 percent EPU used a thyroid scaling factor of 26 percent, which is about 30 percent greater than the 20 percent used in the CPS analysis. Please provide an explanation of why the CPS submittal does not address this consideration, and why you believe the revised core inventory is adequately conservative.

Response
4.3 The core inventories in the USAR were based on calculations performed by General Electric and were not based on TID-14844. The core inventories used in the re-evaluation of the LOCA doses for EPU were calculated using the ORIGEN2 code. The source terms in curies/megawatt-thermal (Ci/MWt) following EPU are provided below in

Table 4.3-1 and are compared to the pre-EPU source terms. The changes in the Ci/MWt for the iodines (the only contributors to the thyroid dose) are very small.

Table 4.3-1
ORIGEN2 Source Terms

Nuclide	Current Ci/MWt	EPU Ci/MWt	EPU/Current
I-131	2.70E+4	2.70E+4	1.00
I-132	3.80E+4	3.91E+4	1.03
I-133	5.50E+4	5.49E+4	1.00
I-134	5.90E+4	6.02E+4	1.02
I-135	5.10E+4	5.14E+4	0.99
Kr-85	2.90E+2	3.17E+2	1.09
Kr-85m	7.20E+3	6.87E+3	0.95
Kr-87	1.20E+4	1.31E+4	1.09
Kr-88	1.80E+4	1.85E+4	1.03
Kr-89	2.20E+4	2.25E+4	1.02
Kr-90	2.50E+4	2.22E+4	0.89
Xe-133	5.50E+4	5.36E+4	0.97
Xe-135	7.40E+3	1.75E+4	2.36
Xe-135m	9.70E+3	1.08E+4	1.11
Xe-137	4.90E+4	4.78E+4	0.98
Xe-138	4.70E+4	4.50E+5	0.96
Xe-139	3.90E+4	3.51E+4	0.90

Since the source terms for the iodines change very little with EPU, the core inventories (and the resulting thyroid doses) will be directly related to the increase in the thermal power, which is 20% for CPS. Considering the fact that the original USAR had a 5% margin added, compared to 2% for EPU, the net increase in core inventories for the iodines is less than 20% and the approach is conservative. The variation in the noble gas activities is much greater than the variation in the iodine activities. In particular, the Xe-135 activity increases by a factor of 2.36, which has a significant impact on the whole body dose. This change is believed to be related to differences between the methods and data incorporated in ORIGEN2 and the methods and data previously used by General Electric.

Question

4.4 Section 11.4.2.8 of the NEDC report states that the LOCA was reanalyzed and that the increase in the iodine release is nearly proportional to the increase in power level, but the noble gas releases are slightly higher. The document then states that the observed differences for the LOCA were used to scale the remaining DBA doses. The staff believes that this approach, as described, is inappropriate. By basing the scaling factor on the LOCA release rates, you are in effect crediting release mitigation features (e.g., plate out, filtration, etc.) which are appropriate for the LOCA, but may not be

appropriate for the remaining accidents to which the scaling factors are applied. The staff believes that the scaling factors need to be developed by multiplying the pre-EPU and post-EPU core inventories by the corresponding dose conversion factors, isotope-by-isotope, summing the isotopic results for each inventory and then calculating the scaling factor from the two sums. This may explain, in part, why the CPS thyroid scaling factor appears low in comparison to those used by other applicants. Please provide a justification for the approach used, or correct the submittal.

Response

4.4 As discussed in the response to Question 4.3 above, the core inventory developed for EPU will increase in proportion to the increase in power level, except for the inventory of Xe-135, which increases by a factor of 2.36. The use of dose conversion factor (DCF) -weighted core inventories to estimate the effect of an increase in activity on the resulting doses includes the implicit assumption that doses are directly proportional to the initial core inventory. Although this is true for a single isotope, a mix of isotopes that has a varying increase in the activity for individual isotopes will produce a time varying scale factor because the isotopes have different decay rates. This is illustrated on Table 4.4-2 below. For the initial core inventory of noble gases, the DCF-weighted activity is about 17% higher than the current core inventory. At one hour, the DCF-weighted activity for the EPU source term is 24% higher than the current source term, and at one day, the DCF-weighted activity is about 31% higher than the current activity. The ratio between the DCF-weighted EPU and current activity then begins to decline, and at 100 hours the difference is 13%, which corresponds to the difference of the long-lived isotope, Xe-133. This variation occurs because early in the accident the dose is dominated by short lived isotopes with little dose significance. Later in the accident the activity is dominated by the more dose significant isotopes, such as Xe-135, which subsequently give way to the less dose-significant, longer-lived isotopes. The iodine isotopes also show some variation with time, but the variation is much less because the increase in iodine activity for EPU is more uniform.

The strong time dependence of the scaling factor based on DCF-weighted activity indicates a more appropriate approach would be to weight the total activity released by the DCF, rather than the initial core inventory. A reasonable proxy for the total activity released is the dose. The calculation of offsite and control room doses takes credit for removal processes that can affect the total activity released. However, since these removal processes are applied to each isotope in a chemical species in the same manner (e.g., all iodines are reduced the same amount by the SGTS filters), the change in dose is determined primarily by the change in the amount of activity released, which is directly proportional to the change in the initial core inventory.

The radiological consequences for the current design basis and the EPU core inventories are summarized in Table 4.4-1 below. These are actual calculated values, and are generally less than the values provided in the USAR.

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Additional Dose Assessment Information Supporting the License Amendment Request to Permit Uprated Power Operation at Clinton Power Station

Table 4.4-1
Radiological Consequences

	Current	EPU	Ratio
Whole Body Doses (rem)			
Control Room	2.425	3.094	1.28
EAB	9.948	12.05	1.21
LPZ	3.022	3.733	1.24
Thyroid Doses (rem)			
Control Room	21.87	25.96	1.19
EAB	204.4	242.5	1.19
LPZ	77.56	92.17	1.19

The ratios for the whole body dose reflect the variability of the noble gas activity illustrated in Table 4.4-2, whereas the thyroid doses, which are due to iodines only, are much more consistent. Based on these results, and understanding that the only sources that will change are the core inventory, scaling factors of 1.30 for the whole body doses and 1.20 for the thyroid doses were used. These scaling factors, in addition to being more conservative than those based on DCF-weighted activity, bound the increase in doses demonstrated using actual release calculations. The use of this approach, rather than the DCF-weighted activity ratio, was required because of the unusual increase in the Xe-135 core inventory.

The increase in doses due to the other accidents that use core inventories (i.e., the CRDA and the FHA) will be bounded by these scale factors. The CRDA is based on shutdown core inventory released over one day, which is bounded by the LOCA releases over 30 days. The FHA uses a decayed source term, but an inspection of the attached table indicates the most significant long lived noble gas (Xe-133) and iodine (I-131) have increases that are equal to or less than 30% and 20%, respectively. Therefore, the use of the LOCA scaling factors is conservative for these accidents.

Additional Dose Assessment Information Supporting the License Amendment Request to Permit Uprated Power Operation at Clinton Power Station

Table 4.4-2
Comparison of the Pre and Post EPU DCF Weighted Core Inventory at Various Times Post LOCA

Isotope	Core Inventory (Ci)		DCF*	Decay Constant /hr	Initial Core Inventory			One Hour Core Inventory			One Day Core Inventory		
	Pre EPU	EPU			Pre EPU * DCF	EPU * DCF	Change	Pre EPU * DCF	EPU * DCF	Change	Pre EPU * DCF	EPU * DCF	Change
Noble Gas Activity													
Kr-83m	9.53E+06	1.16E+07	1.59E-16	3.79E-01	1.51E-09	1.84E-09	0.22	1.04E-09	1.26E-09	0.22	1.71E-13	2.08E-13	0.22
Kr-85	9.16E+05	1.12E+06	1.37E-16	7.38E-06	1.26E-10	1.54E-10	0.22	1.26E-10	1.54E-10	0.22	1.26E-10	1.54E-10	0.22
Kr-85m	2.05E+07	2.43E+07	9.72E-15	1.55E-01	1.99E-07	2.36E-07	0.19	1.71E-07	2.02E-07	0.19	4.86E-09	5.77E-09	0.19
Kr-87	3.93E+07	4.64E+07	4.89E-14	5.45E-01	1.92E-06	2.27E-06	0.18	1.11E-06	1.31E-06	0.18	4.00E-12	4.72E-12	0.18
Kr-88	5.56E+07	6.55E+07	1.20E-13	2.44E-01	6.70E-06	7.89E-06	0.18	5.25E-06	6.18E-06	0.18	1.91E-08	2.25E-08	0.18
Kr-89	6.92E+07	7.97E+07	1.13E-13	1.32E+01	7.82E-06	9.00E-06	0.15	1.46E-11	1.68E-11	0.15	2.59E-143	0.00E+00	0.15
Xe-131m	4.81E+05	1.07E+06	1.24E-15	2.43E-03	5.96E-10	1.32E-09	1.22	5.94E-10	1.32E-09	1.22	5.62E-10	1.25E-09	1.22
Xe-133	1.68E+08	1.90E+08	2.79E-15	5.51E-03	4.69E-07	5.30E-07	0.13	4.66E-07	5.27E-07	0.13	4.11E-07	4.65E-07	0.13
Xe-133m	7.00E+06	6.09E+06	2.56E-15	1.32E-02	1.79E-08	1.56E-08	-0.13	1.77E-08	1.54E-08	-0.13	1.30E-08	1.13E-08	-0.13
Xe-135	2.17E+07	6.20E+07	1.53E-14	7.61E-02	3.31E-07	9.47E-07	1.86	3.07E-07	8.77E-07	1.86	5.34E-08	1.52E-07	1.86
Xe-135m	3.17E+07	3.83E+07	2.65E-14	2.71E+00	8.41E-07	1.02E-06	0.21	5.61E-08	6.77E-08	0.21	5.00E-35	6.04E-35	0.21
Xe-137	1.47E+08	1.69E+08	1.16E-14	1.09E+01	1.70E-06	1.95E-06	0.15	3.27E-11	3.75E-11	0.15	1.09E-119	0.00E+00	0.15
Xe-138	1.40E+08	1.59E+08	6.94E-14	2.94E+00	9.71E-06	1.10E-05	0.14	5.12E-07	5.81E-07	0.14	2.05E-36	2.33E-36	0.14
				Weighted Sum	**2.97E-05**	**3.49E-05**	**0.17**	**7.89E-06**	**9.77E-06**	**0.24**	**5.02E-07**	**6.58E-07**	**0.31**
Iodine Activity													
I-131	8.00E+07	9.56E+07	1.08E+06	3.59E-03	8.64E+13	1.03E+14	0.20	8.61E+13	1.03E+14	0.20	7.93E+13	9.47E+13	0.20
I-132	1.17E+08	1.39E+08	6.44E+03	3.01E-01	7.53E+11	8.95E+11	0.19	5.57E+11	6.62E+11	0.19	5.44E+08	6.46E+08	0.19
I-133	1.67E+08	1.94E+08	1.80E+05	3.33E-02	3.01E+13	3.49E+13	0.16	2.91E+13	3.38E+13	0.16	1.35E+13	1.57E+13	0.16
I-134	1.84E+08	2.13E+08	1.07E+03	7.91E-01	1.97E+11	2.28E+11	0.16	8.93E+10	1.03E+11	0.16	1.13E+03	1.31E+03	0.16
I-135	1.58E+08	1.82E+08	3.13E+04	1.05E-01	4.95E+12	5.70E+12	0.15	4.45E+12	5.13E+12	0.15	4.02E+11	4.63E+11	0.15
				Weighted Sum	**1.22E+14**	**1.45E+14**	**0.18**	**1.20E+14**	**1.43E+14**	**0.19**	**9.32E+13**	**1.11E+14**	**0.19**

*The dose conversion factors for iodine are rem/Ci. The dose conversion factors for noble gases are (Sv/sec)/(Bq/m^3) and are based on the Regulatory Guide 1.3, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Boiling Water Reactors," semi-infinite cloud model.

ATTACHMENT D

Additional Dose Assessment Information Supporting the License Amendment Request to Permit Uprated Power Operation at Clinton Power Station

Question
4.5 The submittal addressed the EPU impact on the LOCA, main steamline break accident, fuel handling accident, and control rod drop accident. The CPS USAR Chapter 15 addresses a much larger spectrum of accidents with regard to radiological consequences. While the majority of these analyses conclude there are no radiological consequences, there are analyses of accidents other than those addressed in the application for which radiation doses were calculated. Please address the impact of the EPU (source term and transport considerations) on the following CPS-specific analysis results:

15.2.4.5, Main steamline isolation valve closures (cross-referenced by many Chapter 15 sections)
15.6.6.5, Feedwater line break consequences
15.7.1.1, Main condenser offgas treatment system failure

Response
4.5 The consequences of accidents that do not result in failed fuel or the depressurization of the primary system are bounded by the evaluation of the MSLIV closure event. In this event it is assumed all four MSLIVs close. Blowdown to the suppression pool is used to remove decay heat. The activity in the steam that blows down to the suppression pool is released to the containment, where it is purged to the atmosphere through either the SGTS or the containment building heating ventilating and air conditioning (HVAC) for 3.5 hours following the event. The reactor coolant activity is the Technical Specification iodine spiking limit of 4 micro-curies per gram (µCi/g) dose equivalent I-131. The noble gas activity release rate is 289,000 µCi/sec at 30 minutes.

The two inputs to this analysis that could change due to EPU are the reactor coolant activity and the safety relief valve mass blowdown. Since the activity is based on the Technical Specification limits, it will not change due to EPU. Because of the higher power level in the core, the amount of heat to be removed will increase. For this analysis, it is assumed the SRV blowdown increases in proportion to the power level, or 20%. Table 4.5-1 below summarizes the worst case doses (containment building HVAC release) and compares the doses to the 10 CFR 20 annual exposure to the general public from activity releases (50 mrem). The EPU values are the current values increased by 20%. The resulting doses are less than 2% of the limits.

As noted in USAR Section 15.6.6.5.1, the feedwater line break is not considered a design basis accident and no design basis accident results are presented in the USAR. The realistic analysis presented in the USAR is based on an offgas release rate of 100,000 µCi/sec at 30 minutes delay. Offgas release rates for current operations, as noted in Section 8.4 of Attachment E to Reference 1, are well below this design basis. Therefore, this same design basis is used for EPU, and there are no changes in the activity released or the resulting doses from the realistic analysis of the feedwater line break.

The source term assumptions for the main condenser offgas system failure are that the offgas system operated at 100,000 µCi/sec noble gas after 30 minutes for 11 months,

followed by 1 month at 350,000 μCi/sec after 30 minutes. The 100,000 μCi/sec noble gas after 30 minutes is the same as the design basis release rate, and, as indicated above, this assumption is conservative and is applicable to EPU. The larger release rate for the last month bounds the Technical Specification Section 3.7.5, "Main Condenser Offgas," limit of 289,000 μCi/sec noble gas after 30 minutes. Therefore there is no need to change these assumptions and the dose consequences will remain unchanged under EPU. The current USAR exclusion area boundary (EAB) and low population zone (LPZ) doses for the Total Body dose, which is bounding, are summarized below in Table 4.5-2.

Table 4.5-1
MSLIV Closure Event Radiological Consequences

Location	Current	EPU	Limit
Restricted Area Boundary:			
Thyroid Dose, mrem	0.60	0.72	≤ 50
Gamma Air Dose, mrem	0.28	0.336	≤ 50

Table 4.5-2
Main Condenser Offgas System Failure Radiological Consequences

Location	Current	EPU	Limit
Exclusion Area:			
Whole Body Dose, rem	0.19	No change	≤ 0.5
Low Population Zone:			
Whole Body Dose, rem	0.045	No change	≤ 0.5

ATTACHMENT D

Additional Dose Assessment Information Supporting the License Amendment Request to Permit Uprated Power Operation at Clinton Power Station

REFERENCES

1. Letter from J. M. Heffley (AmerGen Energy Company, LLC) to U.S. NRC, "Request for License Amendment for Extended Power Uprate Operation," dated June 18, 2001

2. Letter from J. M. Heffley (AmerGen Energy Company, LLC) to U.S. NRC, "Submittal of the Updated Safety Analysis Report, Revision 9," dated May 4, 2001

3. Letter from J. B. Hopkins (NRC) to M. Reandeau (AmerGen Energy Company, LLC), "Issuance of Amendment – Clinton Power Station, Unit 1 (TAC No. MA3888)," dated April 25, 2000