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RS-01-282

December 5, 2001

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 Risk Information Supporting the License Amendment Request to Permit Uprated Power Operation at Clinton Power Station

Reference: 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

In the referenced letter, AmerGen Energy Company (AmerGen), LLC 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 the referenced letter 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 a conference call, requested additional information regarding the proposed changes in the referenced letter. The attachment to this letter provides the information requested in NRC Questions 11.1, 11.2, 11.3, 11.4, 11.5, 11.6, 11.7, 11.8, 11.9 and 11.10.

A001

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

Respectfully,

Keith R. Jury

K. R. Jury
Director – Licensing
Mid-West Regional Operating Group

Attachments:

Affidavit
Attachment: Additional Risk 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 Risk 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.

Keith R. Jury

K. R. Jury
Director – Licensing
Mid-West Regional Operating Group

Subscribed and sworn to before me, a Notary Public in and for the State above named, this 5 day of December, 2001.

Anese L. Grigsby
Notary Public

OFFICIAL SEAL
ANESE L. GRIGSBY
NOTARY PUBLIC, STATE OF ILLINOIS
MY COMMISSION EXPIRES 3-13-2005

ATTACHMENT

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

Question 11.1

The Licensee has evaluated the impacts of the extended power uprate (EPU) using their current, pre-uprate probabilistic risk assessment (PRA) model and a revised model to reflect the EPU plant conditions. Was the peer review that was performed on the licensee's PRA, conducted by industry personnel, separate from the licensee's/ corporations' organizations, or did it only involve licensee/corporation-related staff? In addition, please provide the overall findings of the review (by element) and discuss any elements that were rated low (e.g., less than a 3 on a scale of 1 to 4) and any findings/ observations that potentially affect the sequences impacted by the licensee's proposed EPU.

Response 11.1

The Clinton Power Station (CPS) Probabilistic Risk Assessment (PRA) Peer Review was performed in August 2000 as part of the Boiling Water Reactor Owners' Group (BWROG) Peer Review/Certification program. This Peer Review was performed by individuals who had no involvement with the development of the CPS PRA model. At the time of the Peer Review none of the individuals on the review team worked for AmerGen Energy Company (AmerGen), LLC (the licensee) or Philadelphia Electric and British Energy the corporations that owned AmerGen.

The peer review was performed using the guidance provided in the draft Nuclear Energy Institute document NEI 00-02, "Probabilistic Risk Assessment (PRA) Peer Review Process Guidance." A summary of the PRA Peer Review Team grades by element is shown in Table 11.1-1. The details of the Peer Review can be found in Facts and Observations (F&O's) written for particular sub-elements of the review process. The significant F&O's were evaluated for their potential impact on the extended power uprate (EPU) risk analysis. In some cases the particular issue was evaluated through additional sensitivity studies. The disposition of the significant F&O's is provided in Table 11.1-2.

Table 11.1-1

SUMMARY OF GRADE ASSIGNMENTS BY PRA ELEMENT

PRA Certification Areas Reviewed	Summary Grade
Initiating Events (IE)	2
Accident Sequences Evaluation (AS)	3
Thermal Hydraulic Analysis (TH)	3
Systems Analysis (SY)	3
Data Analysis (DA)	3
Human Reliability Analysis (HR)	2
Dependency Analysis (DE)	2
Structural Response (ST)	2
Quantification and Results Interpretation (QU)	2
Containment Performance Analysis (L2)	3
Maintenance and Update Process (MU)	3

Table 11.1-2

F&O # (1)	F&O SUMMARY	CPS EPU RESPONSE	IMPACT ON EPU (2)
AS-7	Correct error in assuming no depressurization for anticipated transient without scram (ATWS) if high pressure core spray (HPCS) system is available	*The EPU contribution from failure to actuate the automatic depressurization system (ADS) would have a minor contribution compared to other ATWS mitigation operator actions.*	2
AS-14	Reassess credit for shutdown service water (SX) alignment for an ISLOCA in the shutdown cooling (SDC) "B" compartment	*ISLOCA modeling does not impact EPU delta risk.*	2
QU-11	Consider adverse impacts of all ISLOCA's on SX alignment success	*ISLOCA modeling does not impact EPU delta risk.*	2

Table 11.1-2			
F&O # (1)	**F&O SUMMARY**	**CPS EPU RESPONSE**	**IMPACT ON EPU** (2)
QU-24	Core damage frequency (CDF) increases 30% for a truncation limit change of 8E-10	*Sensitivity cases were performed to evaluate the impact of decreasing the quantification truncation limit on the EPU risk results. Decreasing the truncation limit resulted in increasing the base case and EPU CDF. However, the impact on the delta CDF was minor.*	2
L2-25	Same as F&O #AS-14	*Covered by response to #AS-14.*	2
HR-6	Perform detailed human error probability (HEP) evaluations for risk-significant pre-initiator operator actions	*This F&O is a suggested enhancement that will not significantly impact the EPU risk assessment results. The HEPs are already more refined than simple screening values.*	2
HR-12	When converting median HEP's to mean, do so consistently	*The dominant HEPs impacted by EPU (e.g., initiate ADS, standby liquid control (SLC) system) use the mean failure probabilities.*	2
HR-12	To eliminate non-conservatism, perform more detailed HEP's or ensure all screening HEP's are conservative	*The dominant HEPs impacted by EPU (e.g., initiate ADS, SLC) use detailed evaluations to calculate HEPs. In addition, other risk significant HEPs not directly impacted by EPU (e.g., operator fails to remove internals from 1FP036 check valve, operator fails to align fire protection for reactor pressure vessel (RPV) injection) also use detailed evaluations to calculate HEPs.*	2
HR-14, -20	Perform operator interviews to verify human reliability analysis assumptions, each time an update is done	*This F&O recommends performing operator interviews for each PRA Update – the F&O recognizes that this was done for the 1995 PRA update but not the current update. However, it is also recognized that for risk significant human actions, CPS confirmed that those actions have simple steps and clear indication, and operators are trained on them. Response to this F&O has no significant impact on the EPU risk assessment results or conclusions.*	2
HR-26	Identify dependent operator actions and adjust HEP's, accordingly	*A review was performed for cut sets containing operator actions impacted by EPU. If these cut sets contained credible dependent operator actions, then the secondary operator actions were assumed completely dependent on the EPU-impacted operator action. The sensitivity showed that the impact on CDF was minor. The impact on large early release frequency (LERF) was negligible. Note that the one dependent HEP modeled in the CPS PRA (Basic event FFWOPERSWB - HRA DEPENDENT FAILURE TO RESTORE TRIPPED FEEDWATER SYSTEM), was explicitly re-quantified for the EPU.*	3

Table 11.1-2			
F&O # (1)	**F&O SUMMARY**	**CPS EPU RESPONSE**	**IMPACT ON EPU (2)**
DE-7	List operator actions used in more than one place, to ensure the commonality is reflected in the model	*This F&O is a suggested documentation enhancement. Response to this F&O has no direct impact on the EPU risk assessment results or conclusions.*	1
ST-4	Provide a discussion of reactor coolant system (RCS) failure pressure and response of the plant to ATWS conditions	*This F&O is a suggested documentation enhancement. Response to this F&O has no direct impact on the EPU risk assessment results or conclusions.*	1
ST-4	Improve documentation of flooding analysis so that the basis for flood frequencies and impacts in each zone are clear	*This F&O is a suggested documentation enhancement. Response to this F&O has no direct impact on the EPU risk assessment results or conclusions.*	1
ST-4	Provide adequate technical basis for not requiring recirculation pump trip (RPT), or add it to ATWS event trees	*The CPS PRA assumes that during ATWS events with the main condenser available, RPT is not required to prevent RCS overpressure failure. Requiring successful RPT for sequences with the main condenser available (e.g., turbine trip events), would increase the base CDF by approximately 6E-9/year (i.e., 2.0/year (turbine trip) * 1E-5 (failure to scram) * 3E-4 (failure of RPT)). EPU has no impact on this additional failure mode.*	2
ST-4	Include containment failures below the water line in Level 1	*Suppression pool failure below the water line is a long-term containment failure sequence. EPU has minimal impact on long-term sequences.*	2
ST-7	Assess value of adding credit for secondary containment	*Existing model is conservative. EPU not impacted by adding credit for secondary containment.*	2
QU-3	For each of the SETS user programs, provide a description of the information flow and quantitative processes being performed	*This F&O is a suggested documentation enhancement. Response to this F&O has no direct impact on the EPU risk assessment results or conclusions.*	1
QU-7	Given limitations of cut set model, identify limits of applicability for online risk (e.g., maximum number of systems that can be removed from service simultaneously)	*This F&O is a suggested enhancement related to the CPS on-line risk cut set model manipulation. Response to this F&O has no direct impact on the EPU risk assessment results or conclusions.*	1
QU-10, -17	Include identified dependent operator action combinations into the PRA	*Covered by response to #HR-26.*	3
QU-10	Include human reliability analysis dependency between containment spray initiation and residual heat removal (RHR) initiation	*The HEPs for long-term operator actions (>1 hour), such as initiation of containment spray and RHR, are not significantly impacted by EPU.*	2

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

Table 11.1-2

F&O # (1)	F&O SUMMARY	CPS EPU RESPONSE	IMPACT ON EPU (2)
L2-11	Revise Level 2 repair credit to be conditional upon failure to repair in the Level 1 model	*Response to this F&O has no significant impact on the EPU risk assessment results or conclusions. The late LPI recovery terms already have high failure probability. More importantly, they apply to loss of decay heat removal (DHR) sequences, which have no impact on LERF.*	2
QU-12	Provide basis for model treatment of asymmetries and identify asymmetries introduced by the model	*This F&O is a suggested documentation enhancement regarding providing explicit discussions concerning model and plant asymmetries. Response to this F&O has no direct impact on the EPU risk assessment results or conclusions.*	1
QU-27, -30	Perform uncertainty analysis of key assumptions and unique features	*This F&O is a recommended enhancement to add uncertainty analyses to the CPS probabilistic safety assessment (PSA). The EPU risk assessment includes a number of sensitivity studies to bound the modeling. No parametric uncertainty analyses were performed, but none are necessary per Nuclear Regulatory Commission (NRC) guidance (this is acknowledged in the F&O, which states "a parametric uncertainty assessment is one step in the process, but is not necessary." There are no unusual or unique features of CPS that have been identified that would change the perception of the uncertainty range associated with the risk spectrum from that evaluated for the Grand Gulf Mark III in NUREG-1150, "Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants."*	1
IE-3	Explain grouping and quantification of initiating events	*This F&O is a suggested enhancement related to PSA documentation for transient initiator groupings. Response to this F&O has no impact on the EPU risk assessment results or conclusions.*	1
IE-10	Systematically evaluate special initiators, including loss of turbine building closed cooling water system (TBCCW)	*This F&O is a suggested enhancement related to the greater PSA documentation. Specifically, this F&O suggests more detailed documentation regarding the identification and analysis of support system initiators. Response to this F&O has no impact on the EPU risk assessment results or conclusions. All dominant support system initiators are included in the CPS PSA.*	1
IE-10	Clarify nature of LOSW, including # of pumps needed	*This F&O is a suggested enhancement to documentation. Response to this F&O has no direct impact on the EPU risk assessment results or conclusions.*	1

Table 11.1-2

F&O # (1)	F&O SUMMARY	CPS EPU RESPONSE	IMPACT ON EPU (2)
IE-13, -2	Base IE frequency on calendar year	*This F&O recommends switching the IE units to events/calendar year. Response to this F&O has no significant impact on the EPU risk assessment results or conclusions. The F&O recognizes this "...should not significantly affect relative results."*	2
AS-6	ATWS probability appears to be counted twice in IORV	*Certification team did not understand that CPS quantification approach takes care of this issue in the Boolean algebra.*	1
AS-6	Confirm that all critical safety functions are addressed in design of each event tree	*Loss of coolant accident (LOCA) and IORV event trees do not consider requirements for decay heat removal or vapor suppression safety functions. Loss of DHR is a long-term event not significantly impacted by EPU. Loss of vapor suppression is an energetic event where the success criteria are not significantly impacted by EPU.*	2
AS-6	Include vapor suppression in event trees for LOCA-like events	*LOCA initiators with failure of vapor suppression are low frequency sequences. Response to this F&O has no significant impact on the EPU risk assessment results or conclusions.*	2
AS-6	Justify the CPS treatment of pool bypass	*CPS has confirmed, via reference to calculation, that the plant does not need upper pool dump to prevent uncovering horizontal vents when flooding the drywell. Response to this F&O has no direct impact on the EPU risk assessment results or conclusions.*	1
AS-6	Justify reactor core isolation cooling system (RCIC) credit for the bounding small LOCA	*F&O author misunderstood the CPS small LOCA definition size.*	1
AS-6	Add credit for automatic RPT, based on General Electric generic calculations for BWR/6	*Covered by response to 3rd #ST-4.*	2
AS-6	Include effects of RCIC gland seal air compressor failure	*CPS has confirmed that the gland seal compressor is not needed for short-term RCIC success. Response to this F&O has no direct impact on the EPU risk assessment results or conclusions.*	1
AS-6	Reposition the ADS inhibit node in the event tree	*The ATWS event tree will be restructured in the future such that ADS inhibit is considered prior to RPV depressurization. The base ATWS CDF contribution will increase due to the new structure. However, the restructured event tree would not change the conclusions of the EPU evaluation.*	2
AS-15	Remove boron retention credit for SLOCA ATWS below top of active fuel (TAF)	*Removing credit for boron retention would result in a minor change in CDF and LERF for the pre-EPU condition and an identical increase in the post-EPU condition. Therefore, this model change has no impact on the EPU PRA evaluation.*	2

Table 11.1-2			
F&O # (1)	**F&O SUMMARY**	**CPS EPU RESPONSE**	**IMPACT ON EPU** (2)
AS-19	Model correct injection path for low pressure coolant injection (LPCI) for ATWS	*Modeling the correct LPCI injection path for ATWS (i.e., the preferred path is through the SDC return lines) could result in a minor increase in CDF and a negligible change in LERF for the pre-EPU condition and an identical increase in the post-EPU condition. Therefore, this model change has no impact on the EPU PRA evaluation.*	2
AS-21	Include questions for all critical safety functions after recovery to remove need to assign paths to conservative LERF bins	*Existing model is conservative and this issue would not change the conclusions of the EPU evaluation.*	2
TH-7	Reevaluate basis for ISLOCA success criteria with RCIC only	*There is a typographical error in the F&O. Same issue as 5th #AS-6.*	2
TH-8	Document the technical bases for room cooling assumptions, especially for RCIC in station blackout (SBO) and main control room (MCR) in SBO and loss of MCR cooling	*This F&O is a suggested enhancement to documentation. Response to this F&O has no direct impact on the EPU risk assessment results or conclusions.*	1
SY-25	Ensure system notebooks are carefully stored and at least one copy is protected from loss	*This F&O is a suggested enhancement to record keeping and storage of PSA documentation. Response to this F&O has no direct impact on the EPU risk assessment results or conclusions.*	1
SY-25	Ensure Modular Accident Analysis Package (MAAP) results are carefully stored and at least one copy is protected from loss	*This F&O is a suggested enhancement to record keeping and storage of PSA documentation. Response to this F&O has no direct impact on the EPU risk assessment results or conclusions.*	1
SY-25	Create formal tracking system for errors and issues identified between model updates	*This F&O is a suggested enhancement to the tracking of potential model changes for consideration in future PSA updates. Response to this F&O has no direct impact on the EPU risk assessment results or conclusions.*	1
SY-26	Ensure system engineer expertise is used in preparation and review of system notebooks	*This F&O is a suggested enhancement to using system engineers in the preparation and review of system notebooks. CPS PRA staff includes individuals who have served on operating crews or as shift technical advisors (STA). Response to this F&O has no direct impact on the EPU risk assessment results or conclusions.*	1

Table 11.1-2			
F&O # (1)	**F&O SUMMARY**	**CPS EPU RESPONSE**	**IMPACT ON EPU** (2)
ST-5	Several elements dismissed via phenomenological papers should be modeled explicitly	*The two (2) "B" items in this F&O are related to* • *guard pipe and steam tunnel failure modes (pool bypass)* • *suppression pool hydrodynamic loads* *EPU has no quantifiable impact on these failure modes.*	2
ST-5	Examine containment failure sequences to define failure location, size, and impact on equipment	*Covered by response to 4th #ST-4.*	2
QU-6	To resolve truncation issues, develop the model completely in CAFTA-W and use FORTE or NURELMCS for quantification	*Sensitivity cases were performed using Safety Monitor to evaluate the impact of decreasing the quantification truncation limit on the EPU risk results. Decreasing the truncation limit resulted in a minor increase in the delta CDF caused by EPU.*	2
QU-8, -15, -26	Reduce conservatism by adding to the mutually exclusive file all combinations of equipment out-of-service prohibited by Technical Specifications or operating practices	*No change is needed, since experience has shown that these combinations contribute negligibly to results. Such changes will have an insignificant impact on the EPU risk assessment results and no impact on the conclusions.*	2
QU-18	Delete the RCIC FTR recovery term, justify it, or use a time-phased approach	*The number is valid, based on NSAC-161. No change needed. Response to this F&O has no significant impact on the EPU risk assessment results or conclusions.*	2
QU-22	Truncate Level 2 model at a lower value, consistent with sub-element QU-22	*The CPS Level 2 LERF is dominated by ISLOCA failures. Decreasing the Level 2 truncation to lower values would not significantly increase either the pre-EPU or post-EPU LERF.*	2
QU-23	Convergence has not occurred at E-10 truncation	*Sensitivity cases were performed to evaluate the impact of decreasing the quantification truncation limit on the EPU risk results. Decreasing the truncation limit resulted in a minor increase in the risk impact of EPU.*	2
QU-28	Perform sensitivity study that eliminates all credit for hardware repair	*A sensitivity study showed that setting risk significant hardware repair terms to 1.0 in the base PRA model changed the delta CDF for EPU from 4E-7/year to 5E-7/year. A similar sensitivity showed no impact on LERF.*	3
L2-19	Revise the containment failure mode assumed for ATWS	*The F&O suggests that the ATWS containment failures are equally likely to be in the wetwell airspace and wetwell water space (i.e., the base mat to the cylinder joint). The revised failure mode assessment may cause a slight increase to CDF and LERF. However, the impact on EPU would not be significant.*	2

Table 11.1-2			
F&O # (1)	**F&O SUMMARY**	**CPS EPU RESPONSE**	**IMPACT ON EPU** (2)
MU-4	Revise PSA Standard Review instruction to ensure CCF is considered when evaluating design changes	*This F&O is a suggested enhancement related to PSA guidance documentation. Response to this F&O has no direct impact on the EPU risk assessment results or conclusions.*	1
MU-4	Revise the PRA Review Standard to include CW traveling screens in the list of PRA-related systems	*This F&O is a suggested enhancement related to PSA guidance documentation. Response to this F&O has no direct impact on the EPU risk assessment results or conclusions.*	1

Notes

(1) The designators for the F&O's are provided in Table 11.1-1
(2) Description of Impact on EPU
  1 - Documentation issue. No impact on EPU
  2 - Worthy comment. No impact or no significant impact on EPU.
  3 - Worthy comment. Minor, but not significant, impact on EPU.

Question 11.2
Please provide a breakdown, by initiating event, of the current (pre-uprate) and post-uprate core damage frequency (CDF) and large early release frequency (LERF) contribution.

Response 11.2
The core damage frequency (CDF) contribution comparison between pre-EPU and post-EPU conditions is shown in Table 11.2-1. The comparison of large early release frequency (LERF) contributions is shown in Table 11.2-2. The CDF contributions for Sensitivity Case #5 that tested the impact of not installing a motor-driven reactor feedwater pump auto-start design feature, as previously discussed in Reference 1, are shown in Table 11.2-3 and are similar to the contributions presented in Table 11.2-1.

Table 11.2-1

Comparison of EPU to Base CDF Contribution

Initiator	Base Case	EPU Case
Inadvertent open relief valve initiator	6.70%	6.52%
Interfacing system LOCA initiator in feedwater (FW) system	0.40%	0.39%
Interfacing system LOCA initiator in LP system	0.00%	0.00%
Interfacing system LOCA initiator in RHR LPCI system	0.02%	0.02%
Interfacing system LOCA initiator in SDC system	0.47%	0.46%
Large LOCA initiator	0.04%	0.04%
Recovered loss of off-site power	1.40%	1.38%
Loss of off-site power initiator	16.87%	16.48%
Loss of non-safety DC bus initiator	0.51%	0.51%
Loss of non-safety DC bus initiator	0.15%	0.16%
Loss of reserve auxiliary transformer initiator	30.39%	29.89%
Loss of feedwater initiator	5.49%	5.72%
Loss of instrument air initiator	6.88%	7.04%
Loss of plant service water initiator	0.67%	0.66%
Medium LOCA initiator	0.01%	0.01%
Small break LOCA initiator	0.14%	0.14%
Transient with isolation initiator	7.73%	7.16%
Transient without isolation initiator	17.45%	18.64%
Flooding initiators	4.67%	4.80%
TOTAL	**100%**	**100%**

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Table 11.2-2

Comparison of Base to EPU LERF Contribution

Initiator	Base Case	EPU Case
Interfacing system LOCA initiator in FW system	38.20%	36.10%
Interfacing system LOCA initiator in LP system	0.35%	0.33%
Interfacing system LOCA initiator in RHR LPCI system	2.30%	2.18%
Interfacing system LOCA initiator in SDC system	45.10%	42.60%
Loss of reserve auxiliary transformer initiator	6.56%	6.19%
Loss of feedwater initiator	0.22%	0.20%
Transient with isolation initiator	0.81%	0.77%
Transient without isolation initiator	5.29%	10.50%
Flooding initiators	1.15%	1.09%
TOTAL	**100%**	**100%**

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

Table 11.2-3

Comparison of EPU (Sensitivity Case #5) to Base CDF Contribution

Initiator	Base Case	EPU (#5) Case
Inadvertent open relief valve initiator	6.70%	6.54%
Interfacing system LOCA initiator in feedwater (FW) system	0.40%	0.38%
Interfacing system LOCA initiator in LP system	0.00%	0.00%
Interfacing system LOCA initiator in RHR LPCI system	0.02%	0.02%
Interfacing system LOCA initiator in SDC system	0.47%	0.44%
Large LOCA initiator	0.04%	0.04%
Recovered loss of off-site power	1.40%	1.34%
Loss of off-site power initiator	16.87%	15.98%
Loss of non-safety DC bus initiator	0.51%	0.49%
Loss of non-safety DC bus initiator	0.15%	0.15%
Loss of reserve auxiliary transformer initiator	30.39%	28.99%
Loss of feedwater initiator	5.49%	5.54%
Loss of instrument air initiator	6.88%	6.83%
Loss of plant service water initiator	0.67%	0.64%
Medium LOCA initiator	0.01%	0.01%
Small break LOCA initiator	0.14%	0.13%
Transient with isolation initiator	7.73%	8.71%
Transient without isolation initiator	17.45%	19.01%
Flooding initiators	4.67%	4.76%
TOTAL	**100%**	**100%**

ATTACHMENT

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

Question 11.3
Are there any plant modifications being implemented as part of, or in parallel with, the EPU modifications that are associated with equipment actuation or plant scram logic or equipment setpoints that could impact the frequency of reactor scrams? If so, please identify these modifications/impacts and describe how these potential impacts have been considered in determining the change in risk associated with the licensee's proposed EPU.

Response 11.3
There have been no modifications to actuation or plant scram logic as a part of the EPU process that could affect the CPS scram frequency. Instrument setpoint changes are identified in Appendix E to Reference 2. The instrument setpoint adjustments continue to preserve the existing operating margin from the trip setpoints, therefore, are not anticipated to impact the frequency of reactor scrams.

Independent of the EPU tasks there has been an effort to reduce the CPS scram frequency by identifying scram-likely situations and equipment configurations and reducing or eliminating these scram potentials. The effects of these changes are captured in future PRA updates as the transient initiator frequencies are adjusted based upon actual plant experience. A reduction in transient scram frequencies would produce a reduction in plant risk both in the base and post-EPU risk cases and a reduction in the difference as well. No attempt was made to account for the reductions in risk provided by the scram reduction efforts in this risk study. In this respect the results are somewhat conservative.

Question 11.4
During plant normal or expected conditions (e.g., following a turbine trip) for the EPU plant configuration is there any equipment that may be operated beyond its name plate specifications (e.g., main transformer), operating ranges, or limits? If so, please identify the equipment that may be operated beyond its design limits, etc. and describe how these potential impacts have been considered in determining the change in risk associated with the licensee's proposed EPU.

Response 11.4
The EPU team performed extensive evaluations of the capabilities of systems and components that will need to run at higher capacities. Replacement or modification of components is being made to improve the capability and or reliability of components as needed. Examples of systems or components that are being replaced or modified include the following.

1. The main power transformers are being replaced with transformers that can accommodate the station's increased power output.

2. The isophase bus duct cooling system is being modified to provide additional cooling for the bus ducts.

3. The main generator hydrogen cooling system is being upgraded to accommodate higher power output.

4. The main turbine rotors and blades are being replaced to reliably accommodate increased power output.

5. The last stage buckets of the turbines for the turbine driven reactor feedwater pumps (TDRFPs) are being modified to improve the reliability of the TDRFPs for continuous operation at increased flow.

A list of planned modifications was previously provided in Attachment G to Reference 2. The systems that are not being modified as part of EPU were shown to be adequate for EPU operation as described in Attachment E to Reference 2. This includes the emergency core cooling systems, reactor core isolation cooling, condensate, condensate booster, auxiliary power and cooling water systems.

The design review for EPU ensured that systems and components maintain operation within their design limits. A system and component review summary is provided in Attachment E to Reference 2. The long-term reliability of the systems and components is anticipated to be comparable to the existing reliability.

Because of the replacement and modification of a significant number of components, especially in those systems related to power conversion, there may be a temporary reduction in equipment reliability during the "infant mortality stage" for the new equipment. This has been accounted for by the sensitivity case that evaluates an increased transient initiator frequency at the beginning of EPU operation. The reliability of components used to provide core cooling in the post scram condition should be unaffected because systems credited for core cooling were largely unchanged.

Question 11.5
Appendix A of Regulatory Guide (RG) 1.174 refers to the need for the use of importance measures (e.g., Fussell-Vesely (F-V)) to be a function of the base case CDF and LERF rather than being a fixed value for all plants and states further that "...the licensee should demonstrate how the chosen criteria are related to, and conform with, the acceptance guidelines described in this document [RG 1.174]." The licensee's submittal indicates that important operator actions are defined as those that have a F-V importance measure greater than 5E-3 and a time available of less than 30 minutes. How do these criteria relate to the acceptance guidelines of RG 1.174? Are there any operator actions that have not been evaluated in the licensee's submittal, that if assumed failed, would increase the CDF by more than 1E-6/year or LERF by more than 1E-7/year? If so, please identify and address these additional operator actions.

Response 11.5
The following criteria were used in the CPS EPU risk assessment to identify operator actions to be explicitly considered for potential impact by the EPU.

- Operator action Fussell-Vesely (F-V) importance measure ≥ 5E-3

- Time critical action (time available ≤ 30 minutes)

HEP changes, if any, due to the EPU for any action not identified by the above criteria would result in a negligible increase in the CPS calculated CDF and LERF values. The F-V criterion above is a common risk significance criterion used in many industry guidelines and programs. This criterion results in identifying for explicit analysis those operator actions that contribute 0.5% or more to the CDF. The second criterion makes sure to identify for explicit analysis short-term operator actions, that is, actions with HEPs with the potential to be impacted at least marginally by the increase in decay heat load. Individual actions not identified by the above two criteria represent less than 0.5% of the CDF and are longer term actions with HEPs that are non-significantly impacted by the increase in decay heat load due to the EPU. Even if each individual screened operator action HEP were conservatively assumed to have a F-V equal to 4.99E-3 and conservatively assumed to increase by a factor of 1.2 (i.e., ratio representing the EPU power increase), each such action would result in a delta CDF increase of less than 1E-7/year. This is below the Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," guidance of 1E-6/year delta CDF increase for "very small" changes in risk. It also results in a LERF increase below the Regulatory Guide 1.174 guidance of 1E-7/year delta LERF increase for "very small" changes in risk.

The Risk Achievement Worth (RAW) screening criterion described by this question was applied to the HEP screening process used in the CPS EPU risk assessment. The results of this evaluation are discussed below.

Based on the CPS pre-EPU CDF of 1.38E-5/year, the RAW value that results in an *"increase [in] the CDF by more than 1E-6/year"* is 1.06. Similarly, based on the CPS pre-EPU LERF of 1.45E-7/year, the RAW value that results in an *"increase…[in] LERF by more than 1E-7/year"* is 1.7. Applying these additional screening criteria, the following operator actions in Table 11.5-1 are identified for explicit consideration in the CPS EPU risk assessment.

Table 11.5-1

Basic Event ID	Action Description
Y2SC2HPXXH	Operator fails to start SLC injection in time to avoid pool depletion, two trains
QVROPERTRH	Operator fails to vent containment
YALTBINSWH	Failure to take necessary actions for alternate boron injection
Y2SC2HPCXH	Operator fails to start SLC injection in time to avoid pool depletion after containment failure
MVACPMPSYH	Failure to line up vacuum pumps
RSPCOOLSWH	Failure to initiate RHR suppression pool cooling

Basic Event ID	Action Description
RXSWINJMVH	Operator fails to initiate SX injection through RHR discharge line "B"

The above operator actions were assessed for possible impact on calculated HEPs due to reductions in allowable time caused by the increased power and decay heat load of the EPU. This HEP assessment is summarized in Table 11.5-2. As shown in Table 11.5-2, the EPU does not result in any changes to the HEPs of these seven operator actions.

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Table 11.5-2

SUMMARY OF HEPs IDENTIFIED USING RAW CRITERION

Basic Event ID	Action Description	Basis of Importance	Action Time Available		HEP		Comment
			Pre-EPU Power	EPU Power	Pre-EPU Power	EPU Power	
Y2SC2HPXXH	Operator fails to start SLC injection in time to avoid pool depletion, two trains	RAW_{CDF}=1.77	3.25 hrs	2.75 hrs	4.40E-05	<Same>	Allowable time based on suppression pool (SP) inventory depletion due to boil-off due to high power discharge into the pool. Reduction in time frame results in a negligible HEP change.
QVROPERTRH	Operator fails to vent containment	RAW_{CDF}=1.57	>24 hrs	>24 hrs	1.00E-03	<Same>	Allowable time close to 2 days. The timing impact due to EPU is minor at this time frame and results in no HEP change.
YALTBINSWH	Failure to take necessary actions for alternate boron injection	RAW_{CDF}=1.31	>>1 hr.	>>1 hr.	1.00E-03	<Same>	This action is used when condenser is available, and is on the order of hours. The allowable time is still on the order of hours for the EPU, and results in no HEP change.
Y2SC2HPCXH	Operator fails to start SLC injection in time to avoid pool depletion after containment failure	RAW_{CDF}=1.30	3.25 hrs	2.75 hrs	1.40E-04	<Same>	This is a conditional HEP given early SLC failed. Allowable time based on SP depletion due to boil-off due to ATWS discharge to pool. Reduction in time frame results in no increase in calculated HEP.

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Table 11.5-2

SUMMARY OF HEPs IDENTIFIED USING RAW CRITERION

Basic Event ID	Action Description	Basis of Importance	Action Time Available		HEP		Comment
			Pre-EPU Power	EPU Power	Pre-EPU Power	EPU Power	
MVACPMPSYH	Failure to line up vacuum pumps	RAW_{CDF}=1.11	>45 min.	~1hr.	2.50E-02	<Same>	Allowable time is based upon the time that sufficient steam is still available to the steam jet air ejectors (SJAE). The EPU will not reduce this time frame and may in fact extend it.
RSPCOOLSWH	Failure to initiate RHR suppression pool cooling	RAW_{CDF}=1.07	610 min. (10.2 hrs)	~10 hrs.	1.63E-03	<Same>	Base time allowable is time to SP/T=185F (647 minutes) - time to SP/T= 95F (28 minutes). Minor change, if any, in time allowable results in no calculated HEP change.
RXSWINJMVH	Operator fails to initiate SX injection through RHR discharge line "B"	RAW_{CDF}=1.06	~1.5 hrs	~1.5 hrs	1.00E-03	<Same>	Action applies to ISLOCAs when emergency core cooling system (ECCS) used for core cooling. Allowable time based on pool depletion due to ECCS use. Time is unchanged by EPU.

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Question 11.6

On page 4 of the September 28, 2001 supplemental submittal, the licensee indicates that they identified 28 operator actions of highest importance in the PRA and an additional 17 time-critical human error probabilities (HEPs). However, Table 1 of this supplemental submittal only describes the 8 operator actions that were actually changed in the model to reflect the EPU conditions. Some operator actions that the staff expects to be impacted by EPU includes anticipated transient without scram (ATWS) scenarios in which the operators perform power/level control and ATWS scenarios in which the operators need to inhibit the automatic depressurization system (ADS) when high-pressure systems are available. Specifically how are these two operator actions impacted by the proposed EPU? In addition, please provide the current and EPU HEPs, the supporting basis for these values (i.e., the times available to perform these actions and if the current and EPU HEPs are the same though the available times are reduced, include an explanation for not increasing the EPU HEP value) and a description (i.e., the plant information that triggers the action and the specific manual action performed) for each of the operator actions that were identified as important (either due to F-V value or timing).

Response 11.6

The information requested is provided in Table 11.6-1.

Table 11.6-1
Summary of Level/Power Control and ADS Inhibit HEPs

Basic Event ID	Action Description	Procedural Trigger and Action Performed	Action Time Available		HEP		Comment
			Pre-EPU Power	EPU Power	Pre-EPU Power	EPU Power	
YATWSLCLHH	Operator fails to control level using high pressure systems (ATWS)	Governing procedures are EOP-1A, ATWS RPV Control, and 4411.03 Injection Using Preferred ATWS Makeup Systems. Compelling signals are scram failure, SP/T increasing rapidly, SLC injection. The tasks in performing this action are, in general terms: • Lower RPV level by terminating various injection systems • Monitor RPV level drop • Inject SLC (modeled under a separate basic event) • Gradually increase and control RPV level using Preferred ATWS HP systems.	n/a	n/a	6.43E-3	<Same>	The time dependent diagnosis portion of this HEP is modeled as part of the SLC initiation HEP. The execution error modeled by this HEP is not impacted by any decrease in time available due to the EPU.

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Table 11.6-1
Summary of Level/Power Control and ADS Inhibit HEPs

Basic Event ID	Action Description	Procedural Trigger and Action Performed	Action Time Available		HEP		Comment
			Pre-EPU Power	EPU Power	Pre-EPU Power	EPU Power	
YATWSLCLLH	Operator Fails To Control Level Using Low Pressure Systems (ATWS)	The discussions above apply to this action, as well, except that Preferred ATWS LP systems are used to increase and control RPV level.	n/a	n/a	4.0E-2	<Same>	The time dependent diagnosis portion of this HEP is modeled as part of the SLC initiation HEP. The execution error modeled by this HEP is not impacted by any decrease in time available due to the EPU.
YMSSRVXRVH Note (1)	Failure to Inhibit ADS	Governing procedures are the EOPs Compelling signals are scram failure, ADS timer start. The tasks in performing this action are simple, involving turning selector switches on the Main Control boards.	20 mins.	16 mins.	2.8E-3	4.0E-3	Base allowable timing is estimated based on engineering judgment. The EPU allowable time is estimated by applying a ratio reflective of the EPU percentage power increase.

NOTE (1) - It was discovered during preparation of this response that the ADS Inhibit operator action was inadvertently not discussed in the CPS EPU Submittal; however, the HEP change described above was incorporated into the risk assessment documented in the submittal.

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Question 11.7
In Table 1 of the September 28, 2001 supplemental submittal, the licensee indicates that the initiation of rapid depressurization is slightly impacted, which is represented by a single event, GADSMANSYH. Does this event address all conditions including, ATWS, transients, small loss of coolant accidents (LOCAs), and medium LOCAs? Based on other boiling water reactors (BWRs) that have been reviewed, there is typically considerably less time available for this action for ATWS (about 10 minutes) and medium LOCAs (about 25 minutes) than for the other events (about 60 minutes). Please provide the available times and associated HEPs for this action under current and EPU conditions for each of these initiating events or explain why there is no difference in timing.

Response 11.7
The GADSMANSYH HEP basic event for RPV emergency depressurization is applied in the CPS PRA to all the conditions listed above. The available action time and associated HEP for this action are provided in Table 11.7-1.

Table 11.7-1

Basic Event ID	Action Description	Time Available		HEP		Comment
		Pre-EPU Power	EPU Power	Pre-EPU Power	EPU Power	
GADSMANSYH	Operator fails to manually initiate rapid RPV depressurization	31.8 minutes	27.8 minutes	5.0E-4	7.0E-4	The CPS PRA conservatively uses 31.8 minutes for the HEP calculations for RPV depressurization based on a time of 27.8 minutes for RPV level to drop to 2/3 core height plus an additional 4 minutes for assumed core damage after reaching 2/3 core height given no injection at t=0. MAAP runs CPS1d and CPS1e indicate that the time allowable for the EPU case is reduced approximately 4 minutes

A sensitivity study was conducted to determine the impact of having separate HEPs for the operator action to depressurize for transients, small LOCAs, medium LOCAs, and ATWS.

The sensitivity study utilized the original CPS value for transients, but substituted values as used for medium LOCA and for ATWS in the Quad Cities Nuclear Power Station (QCNPS) EPU submittal for the other events of interest. These details are provided in Table 11.7-2.

Table 11.7-2

	Pre-EPU		Post-EPU	
	Time Available	HEP	Time Available	HEP
Medium LOCA (1)	25 minutes	7E-4	20 minutes	1.1E-3
Small LOCA (1)	See above	7E-4	See above	1.1E-3
ATWS (1)	10 minutes	1.7E-2	8.6 minutes	2.2E-2
Other initiators (2)	31.8 minutes	5E-4	27.8 minutes	7E-4

Notes:
(1) Values obtained from QCNPS
(2) Values obtained from CPS

The substitution of these values increased the contribution of cut sets involving manual ADS for medium LOCA, for small LOCA, and for ATWS. However, their contributions are still sufficiently small to make no significant difference to the total pre-EPU CDF of 1.38E-5/year or post-EPU CDF of 1.42E-5/year or to the delta CDF of 4E-7/year.

Question 11.8
In Table 1 of the September 28, 2001 supplemental submittal, the licensee differentiates between operator actions involving 1 standby liquid control (SLC) pump and 2 SLC pumps (e.g., compare event Y1SC20CXXH with event Y2SC20CXXH and event Y1SC30CXXH with event Y2SC30CXXH), with the 2 SLC pump actions having more time available, and thus a lower HEP, than the 1 SLC pump actions. Please describe the differences in these actions and the conditions that result in there being additional time available for the 2 SLC pump action, which results in a lower HEP for this action.

Response 11.8
The only differences in the operator actions for the cases of single-pump and dual-pump operation is that with boron being introduced at a faster rate, the boron injection can be started later. This provides more available time for operator action when both pumps operate, and thus, reduces the error probability for the operator action as compared with singe-pump operating.

Question 11.9
It is indicated in the individual plant examination of external events (IPEEE) safety evaluation report (SER) that cables that were previously routed from the Division 2 inverter through the Division 1 cable spreading room and then through the Division 3 switchgear room were to be rerouted and that the licensee took credit for this rerouting in their IPEEE fire PRA, which reduced the fire CDF by about 76%. Has this rerouting of cabling been performed or will it be performed prior to implementing the proposed EPU? If so, does the rerouted cabling meet the assumptions that were used and credited in the IPEEE fire PRA (e.g., the actual routing, affects of these other routes) or what is the revised fire risk associated with the actual routing? If the cabling has not been rerouted,

please provide the schedule for rerouting this cabling and the risk implications of the existing conditions (i.e., the base CDF and LERF and the associated change in CDF and LERF due to EPU for the current cabling arrangement).

Response 11.9
The rerouting of the subject cables was completed per CPS modification FP-091. In this modification, cables associated with the Division 2 nuclear safety protection system inverter were rerouted such that they no longer pass through the Division 1 cable spreading area (fire zone CB-4) and the Division 3 switchgear area (fire zone CB-5a). This rerouting satisfies the assumptions credited in the Fire PRA regarding these zones. The cable rerouting did not result in the cables passing through any new fire zones with the exception that some now pass though the Main Control Room envelope. The risk contributions from the new cable routings are expected to be much less than the original installation, because of better separation. The Main Control Room has adequate fire suppression features. Even if Main Control Room fires affect the rerouted cable, the remote shutdown capability utilizes Division 1 for safe shutdown at the remote shutdown panel, and, therefore, does not rely on Division 2 cabling.

Question 10.10
On page 6 of the September 28, 2001 supplemental submittal a discussion is provided on shutdown risk that is very brief. Does the licensee have a shutdown PRA that has been used to determine the change in shutdown risk associated with the EPU conditions? If so, please describe how this model was changed and evaluated and the results of this evaluation (i.e., change in risk from current, pre-uprate shutdown risk). This discussion will also need to address the quality of this shutdown PRA model to assure that the model reflects the shutdown conditions. If a shutdown PRA is not used, please describe the licensee's shutdown risk management philosophies/processes that are relied upon to ensure that the impact of EPU on shutdown risk is non-significant. Specifically, the licensee needs to address those aspects of shutdown risk that are impacted by the EPU conditions (e.g., greater decay heat removal, longer times to shutdown, longer times before alternative decay heat removal systems can be used, shorter times to boiling, and shorter times for operator responses).

Response 10.10
CPS does not have a shutdown PRA model. CPS uses the standard safety-function-based, defense-in-depth approach to shutdown risk.

The functional impacts of the EPU on shutdown risk are similar to the impacts on the at-power Level 1 PRA, with the exception that reactivity additions have a different nature in the shutdown condition compared with the at-power condition.

The shutdown risk contributors include the following.

- loss of shutdown cooling
- reactor pressure vessel (RPV) water makeup/injection failures
- reactivity control failures

The reactivity control functional impact at shutdown is related to mis-loaded fuel or mislocated fuel, as opposed to failure to scram issues for the at-power evaluation. The shutdown reactivity control issues are not a function of EPU and, therefore their contribution to changes in CDF or LERF is assessed as zero. The first two functional challenges are similar in nature to the at-power risk assessment.

The following qualitative discussion applies to the shutdown conditions of Hot Shutdown (Mode 3), Cold Shutdown (Mode 4), and Refueling (Mode 5). The EPU risk impact during the transitional periods such as at-power (Mode 1) to Hot Shutdown and Startup (Mode 2) to at-power are subsumed by the at-power Level 1 PRA.

Important initiating events for shutdown include RPV draindown and loss of shutdown cooling, however, no new initiating events or increased potential for initiating events during shutdown (e.g., loss of decay heat removal (DHR) train) have been identified based on the EPU configuration.

The impact of the EPU on the success criteria during shutdown is similar to the Level 1 PRA. The increased power level decreases the time to boildown. However, because the reactor is already shutdown, the boildown times are relatively long compared to the at-power PRA. The boildown time is approximately 3 hours at 2 hours after shutdown (e.g., time of Hot Shutdown) and approximately 5-6 hours at 12-24 hours after shutdown (e.g., time of Cold Shutdown). The changes in the boildown time when comparing the pre-EPU cases with the EPU cases are small fractions of the total boildown time. These small changes in timing have a negligible effect on the calculated HEPs, which are found to be dominated by diagnosis errors rather than errors related to completing tasks.

The increased decay heat levels presented by EPU do not affect the success criteria for those regular systems used to remove decay heat. A single train of shutdown cooling (SDC) is still capable of bringing the reactor to cold shutdown. A single train of fuel pool cooling and cleanup (FPCC) is capable of accommodating the decay heat removal needs of the spent fuel pool even considering a full core offload. The increased decay heat loads associated with the EPU impacts the time when low capacity DHR systems such as FPCC and reactor water cleanup (RWCU) can be considered successful alternate reactor DHR systems. The EPU condition delays the time after shutdown when FPCC or RWCU may be used as an alternative to SDC. However, shutdown risk is dominated during the early time frame soon after shutdown when the decay heat level is high and FPCC and RWCU would not be a viable DHR systems for either pre-EPU or EPU conditions. CPS assesses the time in each outage when various DHR systems are viable. The RWCU and FPCC systems would not be included in the defense-in-depth evaluation until the EPU decay heat level was sufficiently low for these systems to be successful. Therefore, the impact of the EPU on the FPCC and RWCU success criteria has a negligible risk impact.

It is recognized in the shutdown risk quantifications that the SDC equipment is operating continuously for a significant portion of the outage. Therefore, for the post-EPU case, SDC would be required to run for a longer time than in the pre-EPU case before other systems with lower heat removal capacity are adequate for decay heat removal. These generally are very low risk periods during the outage. Therefore, for those low risk situations when FPCC or RWCU could provide a backup in the pre-EPU case, they would become marginal in the post-EPU case for some short period of time. Because the shutdown risk profile is generally dominated by the risk at early times in the outage (e.g., 0 to 10 days), increasing the time when shutdown cooling is the only adequate DHR system (during which the risk is low due to low decay heat) has a minor impact on the overall shutdown risk. With CPS outages moving toward lasting less than 20 days, this change in success criteria has no impact on the integrated shutdown risk.

Other success criteria are marginally impacted by the EPU. The EPU has a minor impact on shutdown RPV inventory makeup requirements because of the low makeup requirements associated with the low decay heat level. The heat load to the suppression pool is also lower than at power because of the low decay heat level, such that the margins for suppression pool cooling capacity are adequate for the EPU condition.

The EPU impact on the success criteria for blowdown loads, RPV overpressure margin, and safety relief valve (SRV) actuation is estimated to be minor because of the low RPV pressure and low decay heat level during shutdown.

Similar to the at-power Level 1 PRA, the decreased boildown time decreases the time available for operator actions. The risk significant operator actions during shutdown conditions include recovering a failed DHR system or initiating alternate DHR systems. However, the longer boildown times during shutdown results in the EPU having a minor impact on the shutdown HEPs associated with recovering or initiating DHR systems because the available time is relatively long and the HEPs are dominated by diagnosis errors.

Based on a review of the potential impacts on initiating events, success criteria, and HRA, the EPU configuration will have a minor impact on shutdown risk.

Any qualitative impact on the EPU on shutdown risk is performed using the ORAM software. ORAM evaluates the planned plant configuration including systems available, RPV water level, RPV and containment status, and decay heat level. ORAM evaluates the planned outage schedule to ensure that adequate defense in depth is maintained throughout the outage. With respect to the EPU, based on the increased decay heat level, ORAM will be able to identify how much longer SDC needs to operate (e.g., 12 days longer) before alternate DHR systems (e.g., FPCC and RWCU) could be placed in service.

References

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

2. 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