AmerGen

An Exelon/British Energy Company

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

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

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

Subject: Supplemental Information Supporting the License Amendment Request to Permit Extended Power Uprate 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) General Electric Company Licensing Topical Report, "Generic Guidelines for General Electric Boiling Water Reactor Extended Power Uprate," NEDC-32424P-A, Class III, February 1999.

(3) General Electric Company Licensing Topical Report, "Generic Evaluations of General Electric Boiling Water Reactor Extended Power Uprate," NEDC-32523P-A, Class III, February 2000, and Supplement 1, Volumes I and II.

(4) Letter from D. M. Crutchfield (U.S. NRC) to G. L. Sozzi (General Electric), "Staff Position Concerning General Electric Boiling-Water Reactor Extended Power Uprate Program," dated February 8, 1996.

(5) Letter from T. H. Essig (U.S. NRC) to J. F. Quirk (General Electric), "Staff Safety Evaluation of General Electric Boiling Water Reactor (BWR) Extended Power Uprate Generic Analyses," dated September 14, 1998.

(6) 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.

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 uprated power levels. 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

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approximately 20 percent rated core thermal power over the current 100 percent power level of 2894 MWt.

Attachment E of Reference 1 contains detailed plant-specific safety analyses consistent with the generic guidelines for uprating the power of Boiling Water Reactors described in References 2 and 3 and approved by the NRC in References 4 and 5. CPS stated in Section 10.5, "Individual Plant Evaluation," of Attachment E to Reference 1, that a plant-specific probabilistic risk/safety assessment (PRA/PSA) consistent with Individual Plant Evaluation requirements would be performed in support of the Extended Power Uprate (EPU). This section also states that the effect of the EPU on the CPS PRA/PSA would be assessed. A plant-specific PRA/PSA impact assessment, consistent with the guidance of Reference 2, Section 5.11.11, has been completed and concludes that the EPU has negligible impact on plant risk.

In Reference 6, the NRC requested CPS to provide risk information supporting the impact assessment conclusions. A summary supporting this impact assessment for CPS, prepared in accordance with the guidance provided in Reference 6, is provided in Attachment A.

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: Extended Power Uprate PRA/PSA Impact Assessment for 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: Supplemental Information Supporting the License Amendment Request to Permit Extended Power Uprate 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 28th day of

September, 2001.

Timothy Byam

Notary Public

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

ATTACHMENT A

EXTENDED POWER UPRATE PRA/PSA IMPACT ASSESSMENT FOR CLINTON POWER STATION

IMPACT ASSESSMENT SUMMARY

The Extended Power Uprate (EPU) Project for Clinton Power Station (CPS) has been reviewed to determine the net impact on the CPS risk profile. The following information addresses the Nuclear Regulatory Commission's (NRC) request for information related to Item 1 of the enclosure to Reference 1.

The purpose of the Probabilistic Risk Assessment (PRA) impact assessment is to:

(1) Identify any significant change in risk associated with the EPU as measured by the CPS PRA models, and

(2) Provide the basis for the impacts on the risk model associated with the EPU.

The assessment of the power uprate impact on risk has been performed relative to the current CPS Individual Plant Evaluation requirements. The guidelines of Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," were followed to assess the change in risk as characterized by core damage frequency (CDF) and large early release frequency (LERF) and to determine if the change in risk is acceptably small.

The following is a summary of the results from the CPS PRA impact assessment for EPU:

- Detailed thermal hydraulic analyses of the plant response using the EPU configuration indicate slight reductions in the operator action "allowable" times for some actions.
- The reduced operator action "allowable" times resulted in minor increases in the assessed Human Error Probabilities (HEPs) in the PRA model.
- Only small risk increases were identified for the changes associated with the EPU. These small increases were associated with (1) slightly reduced times available for effective operator actions, and (2) changes in initiating event frequency (addressed as sensitivity case).
- The risk impact due to the implementation of the Extended Power Uprate is very low and acceptable. The risk impact has been determined to fall within the "very low" category (i.e., Region III of the Regulatory Guide 1.174 Guidelines) for both ΔCDF and for ΔLERF (i.e., changes from baseline CDF and LERF).

The EPU is estimated to slightly increase the CPS internal events PRA CDF from the base value of 1.38E-5/yr to 1.42E-5/yr, an increase of 4.0E-7/yr (2.9%). Based on the changes to the Level 1 model as input to the Level 2 model, the LERF increases from the base value of 1.45E-7/yr to 1.53E-7/yr, an increase of 8.0E-9/yr (5.5%). These quantifications are performed using a truncation limit of 1.0E-10/yr. These changes in risk are within Region III acceptance criteria of Figures 3 and 4 specified in Regulatory Guide 1.174. The best estimates for CDF and LERF also meet the Electric Power Research Institute (EPRI) PSA Applications Guide criteria for permanent plant changes (Reference 2).

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IMPACT ASSESSMENT DETAILS

The following information addresses the NRC's request for information related to Item 2 of the enclosure to Reference 1.

Methodology

The PRA impact assessment methodology includes (1) using an analysis approach for the risk assessment; (2) identifying principal elements of the risk assessment that may be affected by the EPU and associated plant changes; (3) identifying the inputs to the risk evaluation; and (4) examining the hardware, procedural, setpoint, and operating condition changes to assess whether there are PRA impacts that need to be considered. The methodology consists of an examination of the important elements of the CPS PRA to assess the impact of the following EPU changes on the PRA elements.

Hardware changes

Hardware changes required to support the EPU were reviewed and determined not to result in new accident types or change the frequency of challenges to plant response. This assessment is based on review of the plant hardware modifications and engineering judgment based on knowledge of the PRA models. The majority of the changes are characterized by either:

- Replacement of components with enhanced like components
- Upgrade of existing components

Procedural changes

Based on Reference 3, Emergency Operating Procedures (EOPs) variables that play a key role in the PRA and which may require adjustment for the EPU include:

- Boron Injection Initiation Temperature
- Pressure Suppression Pressure Limit
- Heat Capacity Temperature Limit

Except for the case of the Heat Capacity Temperature Limit, the specifics of procedural changes associated with EPU were not available prior to completion of this PRA evaluation. It is anticipated, however, that slight adjustments to the CPS EOPs will be made to be consistent with the EPU condition. In almost all respects, the EOPs are expected to remain unchanged because they are symptom-based; however, certain parameter thresholds and graphs are dependent upon power and decay heat levels and will require slight modifications.

As such, no identified or expected EOP changes as part of the EPU will significantly impact scenario timings or operator response times as modeled in the CPS PRA. CPS has implemented the Boiling Water Reactor Owners' Group emergency procedure guideline (EPG) / severe accident guideline (SAG) update to the EPGs. This change has been factored into the PRA. Any EPU related changes to the CPS EOPs or severe accident management guidelines (SAMGs) are considered minor adjustments to the already assessed EPG/SAG changes. Therefore, the EOP/SAMG changes as a result of the EPU will not influence the calculated risk increases due to EPU. However, should any significant changes be identified, by procedure, those changes are reviewed for impact on the CPS PRA models and the associated risk assessment performed in support of the EPU.

Set point changes

None of the planned setpoint changes will result in any quantifiable impact to the PRA. Key setpoints that play a role in the PRA, such as main steam safety/relief valve opening and closing setpoints, reactor pressure vessel (RPV) level setpoints, and RPV pressure setpoints, are planned to remain unchanged. No changes to the PRA are identified as a result of the planned setpoint changes.

Normal Plant Operational Changes

Key plant operational modifications have been evaluated in support of EPU. RPV pressure and temperature will remain unchanged for EPU. Operation of the number of feedwater pumps, condensate pumps, and condensate booster pumps will remain unchanged for EPU. In addition, there are no significant changes in the operating configuration (e.g., number of pumps normally in operation, number of components required to fulfill a required function, etc.) for the major plant safety systems.

The feedwater/condensate flow rates will be increased to support the EPU. Despite the increase in flow, these operational changes (or the associated hardware modifications) are not expected to significantly impact component failure rates or initiating event frequencies.

PRA Changes Related to EPU Changes

The PRA impact assessment includes the complete risk contribution associated with the EPU conditions. Risk impacts due to internal events are assessed using the current CPS Level 1 and Level 2 PRA. External events are evaluated using the analyses of the CPS Individual Plant Examination of External Events (IPEEE) Submittal (Reference 4). The impacts on shutdown risk contributions are evaluated on a qualitative basis. The PRA impact assessment summarizes the risk impacts of the EPU implementation on the following areas:

- Level 1 Internal Events PRA (includes internal flooding)
- Fire Induced Risk
- Seismic Induced Risk
- Other External Hazards Risk
- Shutdown Risk
- Level 2 PRA

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Level 1 Internal Events PRA

Qualitative engineering insights regarding the adequacy of procedures and systems to prevent postulated core damage scenarios are among the principal results of the Level 1 portion of the PRA. The Level 1 PRA model explicitly incorporates internal flooding initiating events and flooding effects. These insights deal with the adequacy of, or improvements to, CPS procedures or systems, frontline or support, to accomplish their safety mission of preventing core damage. The severe accident scenarios that have been identified in the Level 1 PRA have been reviewed and the relatively small perturbations due to power uprate do not affect the scenario development or the qualitative insights.

Based on CPS EPU Task Report analyses and Modular Accident Analysis Package (MAAP) runs performed in support of this analysis, no changes in systemic success criteria for the Level 1 CPS PRA due to the EPU are identified for this risk assessment. The MAAP analysis considered the increased heat inputs resulting from the upgrade in power (i.e., from 2894 megawatts thermal (MWt) to 3473 MWt). The thermal power increase is the dominant input into the MAAP analysis.

The proposed increase in power level reduces the time available for some operator actions by small increments. The reduction in the available operator response time is generally small compared with the total time required to detect, diagnose, and perform the actions. Operator actions were identified and reviewed based on (1) a Fussell-Vesely importance measure greater than 5.0E-3, and (2) a time critical action of less than 30 minutes available for operator action. Twenty-eight (28) operator actions of highest importance in the PRA (Fussell-Vesely Importance greater than 5.0E-3) were identified; and an additional 17 time critical (i.e., <30 minutes available) human error probability (HEPs) were identified. Table 1 provides additional information regarding changes to operator response times.

Table 1 provides a summary of the PRA model changes incorporated as a result of the power uprate evaluation. The changes in timing are estimated to result in minor changes in the HEPs. Of the total operator actions identified, only eight actions were identified as warranting HEP re-calculation. A non-HEP related item has also been identified. Specifically, a change was made to the stuck open relief valve (SORV) probability for the anticipated transient without scram (ATWS) scenarios. This change is necessary since there is a small potential that the probability would increase due to the reduction of margin for certain transient challenges between the operating pressure and the setpoints given the increased thermal energy in the RPV and the core. The likelihood of an increase in the SORV probability is greater for ATWS scenarios considering the greater number of SRV demands. The CPS PRA probability for an SORV during an ATWS, 1.60E-2, is increased by 20% (judgment, based on the EPU power level increase) to a nominal value of 1.90E-2.

Equipment reliability of plant components was evaluated as part of the EPU. Although equipment reliability as reflected in failure rates can be theoretically postulated to behave as a "bathtub" curve (i.e., the beginning and end of life phases being associated with higher failure rates than the steady-state period), no significant impact on the long-term average of initiating event frequencies, or equipment reliability during the 24-hour mission time is expected. However, an examination of the impact on initiating event frequency was performed as a sensitivity case with minimal impact.

The EPU increases the CPS internal events PRA CDF from the base value of 1.38E-5/yr to 1.42E-5/yr, an increase of 4.0E-7 (2.9%). The majority of the change in risk is from loss of coolant inventory control accident scenarios, due to the increase in the HEP for RPV emergency depressurization, and the remainder is due primarily to ATWS scenarios due to the increase in the standby liquid control (SLC) system initiation HEPs.

Fire Induced Risk

The plant risk due to internal fires was evaluated in 1995 as part of the CPS IPEEE (Reference 4). EPRI Fire Induced Vulnerability Evaluation Methodology and Fire PRA Implementation Guide screening approaches and data were used to perform the CPS IPEEE fire PRA study. The CDF contribution due to internal fires was calculated at 3.26E-6/yr. For the EPU PRA impact assessment, the IPEEE documentation for the fire induced core damage scenarios and the associated frequency results were reviewed.

Based on the results of the internal events PRA evaluation for EPU and a review of the CPS IPEEE, it is concluded that the increase in risk contribution associated with fire induced sequences is minimal (i.e., <3% increase in CDF).

Seismic Induced Risk

The CPS seismic risk analysis was performed as part of the CPS IPEEE. CPS performed a seismic margins assessment (SMA) following the guidance of NUREG-1407, "Procedural and Submittal Guidance for the Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities," and EPRI NP-6041, "A Methodology for Assessment of Nuclear Power Plant Seismic Margin." The SMA is a deterministic evaluation process that does not calculate risk on a probabilistic basis. No core damage frequency sequences were quantified as part of the seismic risk evaluation. There were no vulnerabilities identified as part of the SMA. Based on a review of the CPS IPEEE, the conclusions of the SMA are unaffected by the EPU. The power uprate has little or no impact on the seismic qualifications of the systems, structures and components.

Other External Hazards Risk

In addition to internal fires and seismic events, the CPS IPEEE submittal analyzed a variety of other external hazards.

- High Winds/Tornadoes
- External Flooding
- Transportation and Nearby Facility Accidents
- Other External Hazards

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Based on a review of the Clinton IPEEE, the EPU has no significant impact on the plant risk profile associated with tornadoes, hurricanes, transportation accidents, or other external hazards.

Shutdown Risk

The impact of the EPU on shutdown risk is similar to the impact on the at-power Level 1 PRA. Shutdown risk is affected by the increase in decay heat power. However, the lower power operating conditions during shutdown (i.e., lower decay heat level, lower RPV pressure) allow for additional margin for mitigation systems and operator actions. Based on a review of the potential impacts on initiating events, success criteria, and human reliability analysis (HRA), the EPU impact on shutdown risk is minimal (i.e., <1% increase in CDF). Shutdown risk is assessed on an ongoing basis during outages using the deterministic Outage Risk Assessment and Management (ORAM) model. These models are based on defense-in-depth for key shutdown safety functions and are not affected by equipment unavailability values in the PRA model. Regardless, increased on-line maintenance reduces the need for equipment out-of-service during maintenance and refueling outages, thus reducing risk of those outages.

Level 2 PRA

No changes in success criteria have been identified with regard to the Level 2 containment failure evaluation. The slight changes in accident progression timing and decay heat load have only minor or negligible impacts on Level 2 PRA safety functions, such as containment isolation, ex-vessel debris coolability, and challenges to the ultimate containment strength. The systems that perform these functions continue to maintain this capability and event times for operator actions remain long even with an increase in decay heat because of the Mark III containment design.

The Level 2 PRA calculates the containment response under postulated severe accident conditions and provides an assessment of the containment adequacy. The EPU change in power represents a relatively small change to the containment failure frequency under severe accident conditions.

The impact of the EPU on the CPS Level 2 model, independent of the Level 1 input analysis, is minor. Based on the changes to the Level 1 model as input to the Level 2 model, the at-power internal events LERF increased from the base value of 1.45E-7/yr to 1.53E-7/yr, an increase of 8.0E-9 (5.5%).

PRA Quality

The following manifests the quality of the CPS PRA models used in performing the risk assessment for the CPS EPU.

- Sufficient scope and level of detail in PRA,
- Active maintenance of the PRA models and inputs, and
- Comprehensive Critical Reviews.

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The CPS PRA model and documentation is routinely updated to reflect the current plant configuration following refueling outages and to reflect the accumulation of additional plant operating history and component failure data. The Level 1 and Level 2 CPS PRA analyses were originally developed and submitted to the NRC in Reference 5 to support the CPS Individual Plant Examination (IPE). The CPS PRA has been updated several times since the original IPE submittal. A summary of the CPS PRA history is as follows.

- CPS IPE (September 1992)
- Revision 1 (April 1994)
- Revision 2 (January 1995)
- Revision 3 (June 2000)
- Revision 3a (December 2000)

The CPS PRA model has also benefited from comprehensive technical reviews. A comprehensive self-assessment of the CPS at-power Level 1 and Level 2 PRA models was performed in July 2000. CPS identified both the strengths of the PRA model, and areas where potential enhancements to the PRA model could improve the traceability of the PRA documentation and improve its use for risk informed applications. The self-assessment review was performed using the Nuclear Energy Institute (NEI) checklists of the 11 technical elements that were also used during the peer review certification. The results of the review were documented for each technical element using NEI guidance.

Additionally, a peer review of the CPS PRA was performed in August 2000. The peer review was performed using Reference 6 of this attachment as the basis for the review. The PRA peer review included the review and evaluation of the eleven main technical elements and sub-elements for an at-power PRA, including containment analysis. The peer review provided comments and recommendations to CPS on specific enhancements (e.g., Certification Facts and Observations (F&Os)) to the PRA. The prominent results of the peer review were evaluated for impact on EPU.

CPS PRA F&Os with a significance category of "A" and "B" were reviewed for their impact on the EPU risk study results. The majority of these F&Os have no impact or a minor impact on the EPU risk study results, because they either did not affect the PRA model (e.g., documentation issues) or affected aspects of the model that were not strongly impacted by the EPU changes evaluated. The F&Os that could potentially have significant impacts on the EPU results were further evaluated through sensitivity analysis and confirmed to have non-significant impacts. The F&Os that were evaluated include the following.

- The certification team noted that the evaluation of dependent HEPs had not been updated since the previous revision of the PRA. A review was performed for core damage combinations containing operator actions affected by EPU to test the impact of potentially dependent operator actions. The review assumed that if these combinations contained credible dependent operator actions, then the secondary operator actions would be completely dependent on the EPU impacted operator action. The sensitivity showed that the impact on ΔCDF is not significant and meets the guidelines of Regulatory Guide 1.174.

- The CPS PRA model includes several hardware repair recoveries. This F&Os suggested a sensitivity study to evaluate the overall impact of these recoveries in the model. A sensitivity case was performed for the EPU study by setting risk significant hardware repair terms to 1.0 in the base and EPU models to evaluate the change in ΔCDF. The sensitivity showed that the impact on ΔCDF is not significant and meets the guidelines of Regulatory Guide 1.174.

- The certification team identified that the CPS model did not converge quickly to a set CDF value with lowered truncation levels. This is because of the balanced plant design at CPS in which core damage risk is controlled by multiple components (i.e., the CPS PRA results have many low probability core damage combinations that contribute to risk). A sensitivity case evaluated the rate of change of the base case results, the EPU results and the ΔCDF results as a function of truncation level. Although the base and EPU results increased noticeably with decreasing truncation level, the ΔCDF, which is the parameter evaluated in Regulatory Guide 1.174, varied much less with truncation level. The conclusion reached from this sensitivity case is that the ΔCDF value, when fully converged meets the guidelines of Regulatory Guide 1.174.

The results of these reviews concluded that each of the outstanding potential enhancements have no significant impact on EPU.

<u>Quantitative Bounds on Risk Change</u>

The base EPU model estimates a risk increase of 2.9% in CDF and 5.5% increase in LERF. CPS performed sensitivity calculations for five situations with different assumed conditions to provide input into the decision-making process. The following sensitivity studies investigate the impact on the at-power internal events CDF and LERF.

<u>Sensitivity #1</u>

This sensitivity case addresses the issue regarding whether or not the changes to the balance of plant (BOP) side of the plant in support of the EPU will have a significant impact on plant trip frequency. The base quantification assumes no impact. In this sensitivity case the base "Transient Without Isolation" initiating event frequency is increased by 10%.

<u>Sensitivity #2</u>

Many operator response times are modeled where the time available is less than 30 minutes (e.g., 20 minutes) in the Base PRA model. The actual time available to perform operator actions may be longer, even for the EPU configuration. This sensitivity conservatively reduces the available time for selected operator actions by 20% (i.e., equal to the increase in power level) to evaluate the sensitivity of short time frame operator actions. The affected HEPs were then recalculated using the same methodology as the base model.

Sensitivity #3

The CPS PRA models a number of repair and recovery actions (e.g., repair or recovery of pumps or valves). This sensitivity case models the risk impact if selected repair and recovery terms are set to guaranteed failure. In addition, this sensitivity includes the HEP modeling modifications identified in Sensitivity #2.

Sensitivity #4

This sensitivity case addresses the impact on the plant risk profile of the combined modeling modifications of Sensitivity cases #1, #2, and #3.

Sensitivity #5

The sensitivity case addresses the impact of not adding an auto-start feature to the standby motor driven feedwater pump (MDFWP) following a trip of an operating turbine driven feedwater pump as this modification may not be necessary to support EPU (Section 7.4 of Attachment E to Reference 7).

Sensitivity Results Summary

The key result of the PRA evaluation for the sensitivity cases showed that small risk increases were calculated for both CDF and LERF (Table 2). For the base case (Table 1), the risk increase is primarily associated with reduced times available for RPV emergency depressurization during transient and SLC initiation during ATWS operator action scenarios. The risk increase for at-power internal events due to the EPU is a ΔCDF of 4.0E-7, an increase of 2.9% over the base CDF of 1.38E-5/yr. The at-power internal events LERF increase due to the EPU is a ΔLERF of 8.0E-9, an increase of 5.5% over the base LERF of 1.45E-7/yr.

For the sensitivity case of increasing the "Transient Without Isolation" initiating event frequency by 10% (i.e., Sensitivity #1), the EPU would increase the CPS internal events CDF and LERF. The risk increase for at-power internal events due to the EPU for this sensitivity case is a ΔCDF of 7.0E-7, an increase of 5.1% over the base CDF of 1.38E-5/yr. The at-power internal events LERF increase due to the EPU for this sensitivity case is a ΔLERF of 1.1E-8, an increase of 7.6% over the base LERF of 1.45E-7/yr.

For the sensitivity case of reducing the available time for selected operator actions by 20% (i.e., Sensitivity #2), the EPU would increase the CPS internal events CDF and LERF. The risk increase for at-power internal events due to the EPU for this sensitivity case is a ΔCDF of 6.0E-7, an increase of 4.3% over the base CDF of 1.38E-5/yr. The at-power internal events LERF increase due to the EPU for this sensitivity case is a ΔLERF of 8.0E-9, an increase of 5.5% over the base LERF of 1.45E-7/yr.

For the sensitivity case of reducing the available time for selected operator actions by 20% and setting selected hardware recovery actions to a "FAILED" value of 1.0 (i.e., Sensitivity #3), the EPU would increase the CPS internal events CDF and LERF. The risk increase for at-power internal events due to the EPU for this sensitivity case is a ΔCDF of 2.9E-6, an increase of 21.0% over the base CDF of 1.38E-5/yr. The at-power internal events LERF increase due to the EPU for this sensitivity case is a ΔLERF of 8.0E-9, an increase of 5.5% over the base LERF of 1.45E-7/yr.

For the sensitivity case of increasing the "Transient Without Isolation" initiating event frequency by 10%, reducing the available time for selected operator actions by 20% and setting selected hardware recovery actions to 1.0 (i.e., Sensitivity #4), the EPU would increase the CPS internal events CDF and LERF. The risk increase for at-power internal events due to the EPU for this sensitivity case is a ΔCDF of 3.2E-6, an increase of 23.2% over the base CDF of 1.38E-5/yr. The at-power internal events LERF increase due to the EPU for this sensitivity case is a ΔLERF of 1.1E-8, an increase of 7.6% over the base LERF of 1.45E-7/yr.

For the sensitivity case of not adding an auto-start feature to the MDFWP (i.e., Sensitivity #5), the EPU would increase the CPS internal events CDF and LERF. The risk increase for at-power internal events due to the EPU for this sensitivity case is a ΔCDF of 9.0E-7, an increase of 6.5% over the base CDF of 1.38E-5/yr. The at-power internal events LERF increase due to the EPU for this sensitivity case is a ΔLERF of 8.0E-9, an increase of 5.5% over the base LERF of 1.45E-7/yr. As part of the CPS design process, EPU modifications are evaluated for PRA impact.

Using the NRC guidelines established in Regulatory Guide 1.174 and the calculated results from the Level 1 and 2 PRA, the CDF risk increase for the base EPU model (i.e., 4.0E-7/yr) is well within Region III (i.e., changes that represent very small risk changes). The LERF increase for the base EPU model (i.e., 8.0E-9/yr) is also well within Region III. The guidance in Regulatory Guide 1.174 is also consistent with the EPRI PSA Applications Guide criteria for permanent plant changes.

No modifications associated with the EPU significantly impact the PRA Impact Assessment. The design program at CPS ensures that future modifications are reviewed for associated PRA impact.

REFERENCES

(1) 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.

(2) EPRI, "PSA Applications Guide," EPRI TR-105396, Final Report, August 1995.

(3) Licensing Topical Report, "Generic Evaluations of General Electric Boiling Water Reactor Extended Power Uprate," NEDC-32523P-A, Class III, February 2000, and Supplement 1, Volumes I and II.

(4) Illinois Power Company, "Clinton Power Station Individual Plant Examination for External Events (IPEEE) Submittal," Final Report, September 1995.

(5) Letter from J. S. Perry (Illinois Power Company) to U.S. NRC, "Response to Generic Letter 88-20, Supplement 1," dated September 23, 1992.

(6) NEI, "Probabilistic Risk Assessment (PRA) Peer Review Process Guidance," Revision A-3, June 2, 2000.

(7) 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.

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Table 1

CPS PRA MODEL CHANGES TO REFLECT EPU PLANT CONFIGURATION (BASE CASE)

Description	Basic Event ID	Base Probability	EPU Probability	Contribution to CDF Impact	Comment
Operator Fails to Manually Initiate Rapid Depressurization	GADSMANSYH	5.0E-04	7.0E-04	+3.0%	Reduced available action timing due to the EPU increasing HEP. Time available decreases from 31.8 to 27.8 minutes.
Operator Fails to Start SLC to Avoid Hotwell Depletion With 20% Steam Flow to Suppression Pool (1 Pump)	Y1SC20CXXH	3.1E-01	5.2E-01	+0.4%	Reduced available action timing due to the EPU increasing HEP. Time available decreases from 9 to 6 minutes.
Operator Fails to Start SLC to Avoid Hotwell Depletion With 20% Steam Flow to Suppression Pool (2 Pumps)	Y2SC20CXXH	2.1E-01	3.1E-01	+1.1%	Reduced available action timing due to the EPU increasing HEP. Time available decreases from 12 to 9 minutes.
Operator Fails to Start SLC to Avoid Containment Overpressure With 20% Steam Flow to Suppression Pool (1 Pump)	Y1SC30CXXH	9.9E-03	2.8E-02	0%	Reduced available action timing due to the EPU increasing HEP. Time available decreases from 15 to 10 minutes.
Operator Fails to Start SLC to Avoid Containment Overpressure With 20% Steam Flow to Suppression Pool (2 Pumps)	Y2SC30CXXH	4.8E-03	9.9E-03	0%	Reduced available action timing due to the EPU increasing HEP. Time available decreases from 22 to 17 minutes.

Table 1

CPS PRA MODEL CHANGES TO REFLECT EPU PLANT CONFIGURATION (BASE CASE)

Description	Basic Event ID	Base Probability	EPU Probability	Contribution to CDF Impact	Comment
Operator Fails to Manually Start a Diesel Generator if Auto-Start Fails	BDGMANINIT	1.22E-02	1.25E-02	0%	Reduced available action timing due to the EPU increasing HEP. Time available decreases from 30 to 26 minutes.
Any Safety/Relief Valve Fails to Reclose (ATWS Conditions)	YMSSRVMRVC	1.60E-02	1.90E-02	0%	Increase in probability to account for increase in number of SRV demands during ATWS events.
Operator Fails to Bypass Main Steam Isolation Valve Isolation to Maintain Steam Path	MMSIVISSYH	3.0E-02	1.38E-01	0%	Reduced available action timing due to the EPU increasing HEP. Time available decreases from 30 to 26 minutes.
Operator Fails to Place a Feedwater Pump Back in Service (After Tripping on High Level)	FFWOPERSWH	6.0E-03	6.0E-04	-0.2%	This HEP modification (decreased) is performed as a surrogate to logic model re-structuring to account for the auto-start feature of the MDFWP. Random failure of MDFWP is conservatively estimated to be 0.1. Therefore, failure to place turbine driven feedwater pump back in service (HEP = 6.0E-3) AND random failure of MDFWP (probability = 0.1) is 6.0E-4.

ATTACHMENT A

EXTENDED POWER UPRATE PRA/PSA IMPACT ASSESSMENT FOR CLINTON POWER STATION

Table 1

CPS PRA MODEL CHANGES TO REFLECT EPU PLANT CONFIGURATION (BASE CASE)

Description	Basic Event ID	Base Probability	EPU Probability	Contribution to CDF Impact	Comment
DEPENDENT HEP: Operator Fails to Place a Feedwater Pump Back in Service (After Tripping on High Level) Given Operator Failure to Initiate RPV Depressurization	FFWOPERSWB	1.23E-01	2.3E-02	-1.4%	Dependent HEP decreased as a surrogate to model addition of auto-start logic for MDFWP. Random failure of MDFWP is conservatively estimated to be 0.1. Therefore, dependent failure to place turbine driven FW back in service (dependent HEP = 2.3E-1) AND random failure of MDFWP (probability = 0.1) is 2.3E-2.

ATTACHMENT A

EXTENDED POWER UPRATE PRA/PSA IMPACT ASSESSMENT FOR CLINTON POWER STATION

Table 2

CPS PRA SENSITIVITY CASES IN SUPPORT OF EPU

Case	Description	CDF(1) (% increase over Base EPU)	LERF(1) (% increase over Base EPU)
Base	Base Level 1 Model (pre-EPU)	1.38E-05 (N/A)	1.45E-07 (N/A)
Base EPU	Base Level 1 EPU Model	1.42E-05 (2.9%)	1.53E-07 (5.5%)
Sensitivity #1	Base EPU with "Transient Without Isolation" IE increased by 10%	1.45E-05 (5.1%)	1.56E-07 (7.6%)
Sensitivity #2	Base EPU with selected HEPs increased to account for conservatively reduced operator action times.	1.44E-05 (4.3%)	1.53E-07 (5.5%)
Sensitivity #3	Base EPU with Sensitivity #2 and selected hardware recovery actions set to 1.0.	1.67E-05 (21.0%)	1.53E-07 (5.5%)
Sensitivity #4	Base EPU with Sensitivity #1, #2, and #3	1.70E-05 (23.2%)	1.56E-07 (7.6%)
Sensitivity #5	Base EPU without MDFWP auto-start feature.	1.47E-05 (6.5%)	1.53E-07 (5.5%)

(1) The Level 1 and Level 2 PRA truncation limit used was 1.0E-10/yr