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

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

Subject: Response to Request For Additional Information

References: (1) Letter from J. Hopkins (U.S. NRC) to M. Reandeau (AmerGen), "Request For Additional Information", dated February 15, 2001

(2) Letter from M. Coyle (AmerGen) to U.S. NRC, "Clinton Power Station Application for Amendment of Facility Operating License No. NPF-62 for Extension of Diesel Generator Allowed Outage Time (LA-99-016)," dated December 29, 2000

The purpose of this letter is to transmit the AmerGen Energy Company (AmerGen), LLC response to the NRC's Request For Additional Information (RAI) provided in Reference 1. The RAI contains questions and comments stemming from the NRC review of an AmerGen request to amend the Clinton Power Station (CPS) Technical Specifications (TS) as provided in Reference 2. Based on safety and design reviews, the proposed changes involved a risk-informed TS change that would extend the Division 1 and Division 2 diesel generator Allowed Outage Time from 72 hours to 14 days.

The response to the RAI questions are contained in Attachment A. This submittal does not require a change to the proposed mark-ups provided under Reference 2. Furthermore, there are no additional changes to the TS or the associated Bases as a result of the response to these questions.

AmerGen has reviewed the justification and the Bases for No Significant Hazards Considerations contained in Reference 2. We have concluded that the response to these questions do not alter the bases or conclusions provided in those assessments. In addition, the responses to this RAI do not alter our previous determination that the proposed changes meet the criteria for a categorical exclusion from the requirement for an Environmental Impact Statement.

A001

Should you have any questions concerning this letter, please contact Mr. T. A. Byam at (630)663-7266

Respectfully,

J. M. Heffley
Vice President
Clinton Power Station

Attachments

Affidavit
Attachment A: Response to NRC RAI

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 DEWITT	)	
IN THE MATTER OF	)	
AMERGEN ENERGY COMPANY, LLC	)	Docket Number
CLINTON POWER STATION, UNIT 1	)	50-461

SUBJECT: **Response to Request For Additional Information**

AFFIDAVIT

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

J. M. Heffley
Vice President
Clinton Power Station

Subscribed and sworn to before me, a Notary Public in and

for the State above named, this 22 day of

March, 2001.

Jacqueline S. Matthias
Notary Public

" OFFICIAL SEAL "
Jacqueline S. Matthias
Notary Public, State of Illinois
My Commission Expires 11/24/2001

AmerGen Energy Company (i.e., AmerGen), LLC submitted a proposed amendment to the Clinton Power Station (CPS) Technical Specifications (TS) which would permit a longer Allowed Outage Time (AOT) for the Division 1 and Division 2 Diesel Generators (Reference 1). After a partial review of Reference (1), the NRC issued a request for additional information as provided in Reference (2). The questions were initiated by the Probabilistic Safety Assessment Branch and concerned four areas of the risk analysis basis for the proposed change. Our response to each of the specific NRC questions is provided below.

QUALITY OF PRA

1. **"The submittal indicated that Clinton participated in the Boiling Water Reactor Owners Group (BWROG) probabilistic risk assessment (PRA) Peer Review Certification program. A PRA Certification Team completed an inspection and review of the Clinton PRA. The team found that the Clinton PRA was fully capable of addressing issues associated with the proposed emergency diesel generator (EDG) allowed outage time (AOT) extension with a few enhancements.**

    a. **Did the peer review group specifically address application of the PRA to the EDG AOT extension changes, or was it a general assessment for application to AOT changes?**

    b. **A peer review is one element in a PRA's quality program. Explain what other elements are used to assure quality of the Clinton PRA?**

    c. **What were the few enhancements identified, and how were they addressed in the analysis performed to support the proposed changes?**

    d. **Were the enhancements peer reviewed, and if so, by whom?**

    e. **Who participated in the Clinton PRA peer review, and what were their qualifications?"**

Response:
The following responses address the above questions regarding the Peer Review Process.

a. The BWROG PRA Peer Review Certification program does not specifically evaluate the PRA models for a particular application such as an EDG AOT extension. However, the grading process for the Certification Program is intended to indicate the types of PRA applications for which the attributes of the PRA are suitable. Those certification elements receiving Grade 3 are deemed to be suitable for types of applications such as single TS actions if supported by deterministic evaluations. Not all areas of the PRA have to be assigned Grade 3 or greater to be suitable for TS changes. An important aspect of the certification process is the development of Facts & Observations (F&Os) that describe the issues relevant to particular sub-elements of the

PRA. The impact of these issues on the particular PRA application being developed should be understood and addressed as appropriate. Issues that are pertinent to the risk study in support of the EDG AOT extension are discussed in item c. below.

b. The quality of the PRA is critical to the effective use of the risk insights produced by the PRA. The CPS PRA model for use in the EDG AOT extension has been developed and implemented consistent with the quality guidance provided by Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis." This guidance identifies areas of the PRA and PRA inputs that need to have adequate assurance of quality commensurate with the application. For the EDG AOT application, we have determined that the PRA quality is more than adequate to support the quantitative evaluation of the change in risk.

The BWROG PRA certification peer review was conducted against documented criteria. The results of the peer review along with the actions we have taken to respond to the comments from the peer review provide the basis for the technical acceptability of the PRA model.

The peer review provided comments and recommendations to CPS on specific enhancements to the PRA. These were considered in the EDG AOT extension PRA application and those recommendations of importance were either included or are the subject of specific sensitivity calculations. The results verify that resolutions of the recommendations do not change the conclusions of the risk evaluation. It has been demonstrated that the guidelines from Regulatory Guide 1.174 and Regulatory Guide 1.177, "An Approach For Plant-Specific, Risk-Informed Decisionmaking: Technical Specifications," are met.

Important attributes of the PRA model are that it reflects the as-built and as-operated plant, that important plant behaviors are understood and reflected in the model, and that the model is logically and mathematically correct for its intended application. The CPS PRA model was developed by a PRA staff who have considerable experience with the design and operation of the plant, including its behavior under actual plant events. The in-house staff was augmented with experienced PRA consultants who bring a broader industry perspective on those key attributes that control plant risk, including an understanding of severe accident phenomena. Because the individuals involved are experienced PRA practitioners, they understand the logic and modeling techniques required to accurately account for plant design and operating behavior.

The actual mechanics of model development and use for applications is controlled by CPS procedures that ensure that these products are prepared and reviewed by individuals who are competent in the discipline. The process is intended to produce products that show the reasoning used in modeling decisions and the actual model manipulations performed so the process is repeatable. The review process ensures that the reasoning used in the analysis is correct, the model manipulations are appropriate and have been implemented correctly. The software used for quantification of the PRA model is controlled by the CPS software control process, which ensures that the codes used have been validated.

Additionally, a final check is made to confirm that the results make sense in light of the plant design and operation. The results should point to those plant features that are most important for controlling risk for the particular plant configuration being evaluated. There should be a physical explanation that makes sense why the particular features are important or others are not. Use of experienced plant people again facilitates this reasonableness check.

Since the BWROG PRA peer review/certification, the CPS PRA staff has been assisted extensively by PRA experts from Exelon Corporation and from the Exelon Midwest Regional Operating Group as well as additional PRA experts from ERIN Engineering and Tenera. These personnel have participated in developing the plan for responding to the BWROG certification team comments and recommendations and the preparation of the proposed EDG AOT extension.

c. As described above, a peer review of the updated CPS PRA was performed in August 2000 by the BWROG in accordance with their certification guidelines. Overall, the peer review resulted in the conclusion that most of the elements of the CPS PRA were Grade 3 or suitable for supporting risk-informed applications such as changes to the TS. The review team made 5 F&Os with the significance level of "A" and 91 F&Os with the significance level of "B". The significance levels have the following definitions.

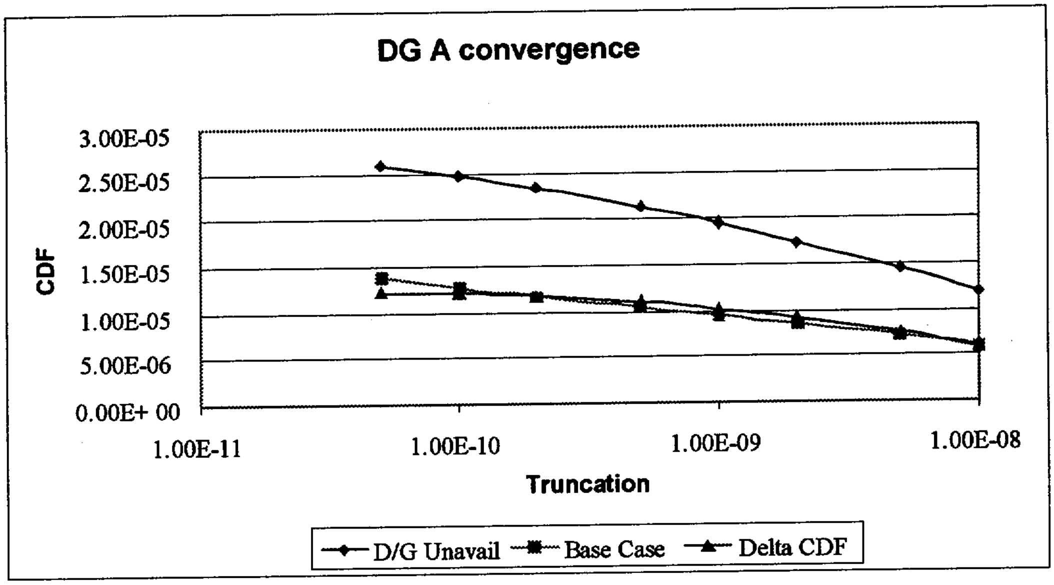
- A - Extremely important and necessary to address for ensuring the technical adequacy of the PRA, the quality of the PRA, or the quality of the PRA update process.
- B - Important and necessary to address, but may be deferred until the next PRA update.

We have prioritized these F&Os for response, and the highest priority F&Os are shown in Table 1. As explained on the last page of that table, the list includes all significance level "A" F&Os, and all significance level "B" F&Os that are related to a sub-element receiving a grade less than "3." Table 1 also summarizes the impact the F&Os would have on the EDG AOT risk study. The majority of the items identified in Table 1 would have minimal impact on the risk study because they do not impact Loss of Offsite Power (LOOP) events or systems used to mitigate LOOPs (e.g. a number deal with Anticipated Transient Without Scram (ATWS) or Interfacing System Loss of Coolant Accident (ISLOCA) events). The current core damage frequency (CDF) estimate for CPS from internal events including flooding (i.e., ~1.3E-5/yr) and from quantified external events like fire (i.e., ~3E-6/yr) place this risk analysis on the left side of Region III as defined in Figure 3 from Regulatory Guide 1.174. Changes in base CDF resulting from resolution of all certification comments in the future is expected to have only a minimal impact on this position. Those F&Os which required model revision or sensitivity studies for the EDG AOT extension are described below.

<u>Convergence (F&Os QU-24)</u>

The Peer Review/Certification Team identified the fact that at the lowest truncation limits used for the SETS computer code (i.e., sequence quantification), the CDF does not yet appear to converge. For the long term, convergence studies will be undertaken using different methods or computer codes. For the EDG AOT risk study, we have performed convergence studies with the current model. As shown by

the graph below, the CDF difference between the base case and the case with a diesel unavailable (i.e., the parameter of interest for the AOT extension risk study), has plateaued at the truncation limit used.



Operator Action Dependence (F&O's HR-26, DE-7)

These issues are resolved by the sensitivity study described in the response to Question 2.

d. The enhancements discussed in item c did not themselves receive a formal industry peer review such as from the BWROG PRA Peer Review process. However, the responses to the F&Os including the sensitivity studies were reviewed by PRA personnel from the Exelon Midwest Regional Operating Group and by PRA staff from ERIN Engineering. This review provided comments that were all satisfactorily incorporated in the EDG AOT base model or in the sensitivity studies referenced in these RAI responses.

e. A list of the individuals who were on the CPS PRA Peer Review Certification Team is provided in Table 2 along with a summary of their work experience.

2. **"The staff safety evaluation report (SER) for the Clinton individual plant examination (IPE) found a few weaknesses for applications other than addressing the intent of generic letter (GL) 88-20. They included the use of generic sources for most test and maintenance unavailability and component reliability data, the credit taken for equipment repairs or restorations, and the issues of hydrogen combustion and ex-vessel steam explosion for the back-end analysis. Explain how these potential weaknesses were addressed in your subsequent PRA updates."**

Response:

The following describes how these issues were addressed.

Plant Specific Data

At the time of the IPE submittal, CPS had only a few years of power operation. Therefore, initiating event and component failure data was largely based on generic sources. The CPS PRA now includes substantial input from plant specific data collected for plant components. The most recent PRA update collected maintenance work record data between 1987 and 1998 for components that had high values for the Fussell-Vesely or Risk Achievement Worth measures of importance. In addition, data was collected from diesel generator logs for the period 1996 to 1999 and maintenance unavailabilities were collected from Maintenance Rule records between 1994 and 1996. The data collection effort included a review of records both at shutdown as well as at power to include failures that would be relevant while the reactor was in operation. At the time of the most recent PRA update, CPS had nearly seven years of power operation. In addition to plant component reliability and availability, this operating data was used to derive plant specific initiating event frequencies as well.

Repair and Recovery Modeling

In Reference (3), the NRC commented that the credit given for local repair of components in the CPS IPE was more optimistic than typically used in other PRAs. Because we had performed a sensitivity analysis that demonstrated that the results were not significantly affected by the repair modeling, the NRC further concluded that the equipment repair model was not a weakness in the IPE. However, in their review of the latest update to the PRA, the BWROG peer review team also commented that the repair modeling in the Clinton PRA was among the more extensive in the industry. Furthermore, in their comments on the Human Reliability, Dependency Evaluation and Accident Sequence Quantification elements, the peer review team noted that multiple operator actions and repairs can occur in the accident sequence cut sets. Specifically, the F&Os recommend performing a sensitivity study and reviewing the method and rules for recovery credit. Given these comments, the EDG AOT evaluation was examined to determine the impact of both multiple human actions and repair assumptions on the results.

The CPS PRA includes a number of recovery events that involve repair of failed equipment following an initiating event. In support of the EDG AOT evaluation, a review of repair and recovery events was performed to identify those that may not be considered typical when compared to other PRAs. For the most part, the failure probabilities for these unique repair events are relatively large (i.e., on the order of 0.4 to 0.5). In addition, these failure probabilities are derived from NSAC-161, "Faulted System Recovery Experience," and would not be expected to influence the results significantly as was demonstrated by the sensitivity performed for the IPE. A possible exception is the value used for restoration of diesel generators having failure probabilities of 0.5 at 1/2 hr. and 0.14 at 4 hours. These diesel generator restoration values are optimistic compared to those of other PRAs such as NUREG/CR-4550, "Analysis of Core Damage Frequency: Internal Events Methodology," (i.e., 0.7 at 5 hours) and could influence the outcome of the PRA particularly as it relates to the evaluation of the EDG AOT.

To address the significance of repair modeling as included in the CPS PRA, as well as the BWROG certification team comments regarding the potential for multiple operator actions, a series of changes were made to the CPS PRA as a sensitivity study on human action modeling and repair. Two changes, in particular, are of significance in addressing these potential non-conservatisms:

- Given its importance to the EDG AOT evaluation, data for diesel generator repair was modified to be the same as that found in the NUREG-1150, "Severe Accident Risks: An Assessment for Five U. S. Nuclear Power Plants," analyses.
- No more than one post-initiator operator action or repair was permitted per cutset. This effectively assumes that any additional operator actions or repairs have a conditional failure probability of 1.0.

In generating these results, conservatisms were found in the modeling and data that had not been important to the original analysis. These conservatisms included normal operator actions that had not been credited such as operator actions to initiate standby systems. Also identified were conservative modeling, generic failure rates, or other data that did not reflect the manner in which the plant is operated, maintained and tested. Therefore, the sensitivity study involved the two changes stated above and the removal of these conservatisms. The increase in CDF produced by taking credit for only a single operator action in each cutset, was offset by removing the conservatisms such that the sensitivity study ΔCDF was actually below the base case. Thus, the sensitivity analysis results remained well within the risk acceptance guideline of R.G. 1.174 (i.e., less than 1E-6/yr ΔCDF) and the acceptance guidance found in R.G. 1.177 (i.e., incremental conditional core damage probability (ICCDP) less than 5E-7).

The additional repair actions included in the CPS PRA do not impact the outcome of the EDG AOT evaluation. This conclusion is reached, including the assumption that only one operator action or repair is credited per cutset, with any other repair/recovery actions being treated as completely failed.

Hydrogen Combustion and Fuel Coolant Interaction Assumptions

In Reference (3), the NRC identified potential weaknesses in the back-end (i.e., Level 2) analysis of the IPE. These weaknesses were particularly concerned with hydrogen ignition after power recovery under station blackout conditions and the treatment of fuel coolant interactions. These issues have been explicitly treated as a part of subsequent updates to the PRA.

In the most recent update to the PRA, Containment Event Tree (CET) branches have been added that represent time-phased challenges to the containment. These challenges are dependent on the time at which offsite power is recovered following a station blackout. The time phases include both early and late recovery of offsite power. Early recovery of offsite power is assumed to ignite hydrogen generated as a part of the early stages of core melt progression causing a pressure spike and possibly failing the containment depending on the pressure rise at the time of the hydrogen burn. These early burns are not expected to fail the drywell due to its strength. As a result, bypass of the suppression pool is not assumed unless other phenomenological events lead to loss

of drywell integrity (i.e., steam explosions or drywell-wetwell vacuum breaker failure). Late recovery of offsite power, however, can permit sufficient buildup of hydrogen to exceed the ultimate capacity of containment if ignited. Buildup of hydrogen to this extent is assumed to lead to failure of both the containment and drywell with certainty on recovery of offsite power. Because it occurs late in time, however, these sequences do not contribute to the large early release frequency.

It is recognized that the timing of containment failure due to hydrogen combustion in the current update may still be optimistic as compared to the NRC results published in NUREG-1150 for Mark III containments. For the purpose of the EDG AOT evaluation, two additional sensitivity studies were performed that are intended to provide bounding assessments of the hydrogen generation and other phenomenological impacts in the CPS Level 2 analysis. For the first sensitivity study, it was assumed that sufficient hydrogen could always be generated to fail the containment and bypass the suppression pool relatively early in the event. On recovery of offsite power following the SBO, ignition of this hydrogen was postulated regardless of operator actions to prohibit operation of the igniters, and a large release was assumed to occur with certainty. Two types of SBO sequences were considered:

Sequence Type	Large Early Release Timing
Long Term Battery depletion	Core damage at 4 hrs leading to large releases at 6 to 10 hours
Reactor injection system failure & Short Term Battery Depletion	Core Damage at 1/2 hr leading to large releases between 2 and 6 hours

The probability for large early releases under these assumptions is then set equal to the probability of recovering an AC power source during the period. Under the assumption that large releases can occur with certainty on recovery of offsite power following core damage during a SBO, the calculated large early release frequency (LERF) is several times the base case and, therefore, would be dominated by SBO-Hydrogen combustion challenges. However, only a fraction of the potential for offsite power recovery occurs precisely during the period between the point at which hydrogen buildup is assumed to be sufficient to fail containment and the time at which offsite protective actions would be expected to be effective. As a result, ΔLERF and incremental conditional large early release probability (ICLERP) meet the acceptance guidelines of Regulatory Guides 1.174 and 1.177.

Fuel coolant interactions have also been explicitly added to the CET quantification with the most recent updates. An early containment challenge heading is quantified that considers phenomenological events such as in-vessel steam explosions, ex-vessel steam explosions, vessel blowdown forces and vapor suppression failure as well as the contribution from containment isolation failure. The concern noted by the NRC in Reference (3) is directed at ex-vessel steam explosions. In the CPS PRA, this challenge principally applies to LOCA sequences where water would be present on the drywell floor at the time of vessel penetration. It is recognized that periodic hydrogen burns in the containment during the course of an SBO could possibly force water over the weir wall into the drywell, but this is expected to be limited due to vacuum breaker operation

on the rise in containment pressure over that in the drywell. As the drywell floor is expected to remain relatively dry at the time of vessel penetration, little challenge to the containment is assumed from ex-vessel steam explosions during station blackout sequences. Other challenges to the containment and drywell are assumed to be possible, however, and are quantified on a best estimate basis as a part of the analysis. These challenges include in-vessel steam explosions, containment isolation failure and, limited to drywell integrity, vacuum breaker failure.

In the second sensitivity study, we have investigated whether adoption of the NUREG-1150 approach to Mark III containment performance analysis would alter the conclusion of the risk assessment performed for the EDG AOT extension. To this end, the critical phenomena identified in NUREG/CR-4551, "Evaluation of Severe Accident Risks: Methodology for the Accident Progression, Source Term, Consequence, and Risk Integration and Uncertainty Analysis," for the Mark III containment and also questioned in Reference (3) have been explicitly evaluated. These phenomena include the following.

- Hydrogen generation in-vessel and ex-vessel
- Combustible gas ignition with and without AC power available
- Detonation or rapid deflagration of combustible gases
- High pressure melt ejection effects
- In-vessel and ex-vessel steam explosion impacts
- Drywell failure or bypass as a result of the severe accident phenomenological effects
- Containment failure

However, not included in this quantification is the impact of a known design difference between the CPS containment and other Mark III containments. The ratio of the containment volume to rated power for CPS is the highest among all the US Mark III plants.

For the NUREG-1150 sensitivity study, the above phenomena are addressed in the same manner as in the NUREG/CR-4551 assessment. This resulted in assuming substantial hydrogen could be generated over a short period of time and that detonable combustible gas mixtures could be achieved in the containment during SBO events. Further, the analysis included the assumption that ignition could occur despite no AC power being available and that such a situation could fail both the drywell and the containment. In addition, both ex-vessel steam explosions and High Pressure Melt Ejection (HPME) phenomena were also quantitatively assessed using the NUREG/CR-4551 probabilistic assessment approach. The results of this second sensitivity study indicate that the ICLERP is substantially larger than that calculated with the base model for CPS. However, despite the larger ICLERP, it remains approximately a factor of two below the acceptance guidelines in RG 1.177 (i.e., less than or equal to 5E-08).

In summary, the NRC, in Reference (3), summarized their concerns regarding the Level 2 part of the IPE. These concerns included the fact that the IPE reported only a 5% chance of containment failure given core damage. As a result of updates to the PRA, additional branches have been added to the CETs explicitly representing phenomenological challenges to the containment post core damage. A new distribution for containment failure has been generated as a part of the PRA updates with this additional detail resulting in a conditional failure of containment on the order of 50%.

Sensitivity studies performed for phenomena important to SBO scenarios indicate that the EDG AOT results remain acceptable from a Level 2 analysis perspective even when using relatively conservative assumptions regarding hydrogen ignition with or without power recovery.

3. **"The submittal indicated that the current PRA has been updated three times since the development of the IPE. How does Clinton assure that the current PRA used for this application represents the as-built and as-operated plant? Have all significant plant operational changes, both hardware and procedural, been appropriately incorporated into the current PRA? List significant plant operational changes and how such changes were incorporated during the updates."**

Response:

Hardware changes and procedural changes are incorporated in the plant model as described under question 4 below. Maintenance and operating practices as they affect the reliability of components are addressed through data updates that take into consideration the plant specific failure history for significant components. Model refinements have been made to better capture actual plant experience. In this regard, the CPS PRA staff has included individuals who have served on operating crews for the Station as Senior Reactor Operators or as Shift Technical Advisors.

The 2000 PRA model update included the addition of an offsite power model for both the 345kv switchyard that supplies the Reserve Auxiliary Transformer (RAT) and the 138 kV supply to the Emergency Reserve Auxiliary Transformer (ERAT). The offsite power model also includes a model for the Static Var Compensators that were added to the plant in 1999 to provide better voltage control for safety related plant buses when connected to offsite sources. Actual plant operating experience was also taken into account with the addition of the loss of RAT initiator as described in the response to question 10 below. Changes to the operating crew actions as a result of implementation of the Severe Accident Management Guidelines (SAMG) has been added to the PRA. Use of the PRA for Maintenance Rule Performance Criteria and for 10 CFR 50.65,"Requirements for monitoring the effectiveness of maintenance at nuclear power plants," paragraph (a)(4) online risk monitoring also ensures consistency between the PRA and actual equipment performance.

Significant operational and hardware changes as they affect AC power supplies and the evaluation of LOOP events have been incorporated in the model. An exception is that the loss of RAT initiating event frequency may be somewhat reduced from what it is in the current model because of design changes and bus alignments that make it less likely to result in a plant shutdown. This operational change was implemented recently and does not adversely impact the conclusions of this risk study.

4. **"Your submittal indicated that you had updated the Clinton PRA to include plant and procedure changes. Please discuss the process for assuring important changes are included in PRA updates in a timely manner."**

Response:

As part of the current hardware change process at CPS there are General Design Review Standards (GDRS) used to identify the need to have particular engineering groups review the design change being developed. GDRS-18, "PRA Review Standard," is used to determine whether the PRA group should review the design change. Review by the PRA group is required if it affects systems modeled in the PRA. If a PRA review is required, a PRA analyst reviews the design change to determine if it would impact the current PRA model. If a model change is determined to be required, the change is typically made during the next full model update unless the change has the potential for significant impact. In this case the design change can be reviewed in a risk analysis that evaluates the risk implications of the change before implementation.

PRA system notebooks have been developed which contain a list of design drawings (e.g. Piping and Instrumentation Drawings and Electrical Schematics) that were used in the development of the fault tree models. During a model update the most recent drawing revisions are reviewed to determine whether the system has changed in a way that requires a system model change. This serves as a second check that relevant design changes are incorporated in the model. The drawing list is updated as part of the system notebook update to show the new drawing revisions that have been considered. The drawing list, with revisions shown, records the design baseline considered for a particular model update.

In a similar way, there is a procedure list contained in each system notebook that shows the procedure baseline considered in development of the system fault tree models. During model updates the latest revisions of these procedures are reviewed and model changes are made as necessary. System notebook revisions for the 2000 PRA update were completed in 1999 and early 2000. Since that time, the only noteworthy change in the plant is a change in bus alignment that may reduce the Loss of RAT frequency. The event trees were developed in consideration of the most recent version of the Emergency Operating Procedures (EOPs) and SAMGs. The PRA is expected to be updated at least once every three years.

NSED Project Instruction EP-19, "Design Change and Procedure Revision Review for Impact on the CPS PRA," provides instruction for reviewing these changes. NSED Instruction EP-10, "Probabilistic Risk Assessment Files," covers the development and revision of PRA System Notebooks. NSED Instruction EP-12, "PRA System Modeling and Quantification," provides guidance in modeling conventions and the quantification process.

RISK IMPACT DUE TO EXTERNAL INITIATING EVENTS

5. **"Your submittal indicated that the risk impact from fire scenarios would be minimally small because the number of scenarios is few and the associated fire ignition frequencies collectively are small in comparison to the loss of offsite power (LOOP) initiator, used for the internal events PRA. However, certain fire scenarios could not only cause a loss of offsite power initiator but also fail systems needed to mitigate the initiator (e.g., a train or part of a train of emergency core cooling system (ECCS). Evaluate your fire areas for such scenarios to assess the potential risk impact due to the proposed change. Provide the fire ignition frequencies used for the related fire areas. For each fire area, the conditional core damage probability (with an EDG out of service) could also be useful to demonstrate the fire risk significance. Clinton should justify whether or not the fire risk impact would clearly meet the acceptable guidelines in regulatory guides (RGs) 1.174 and 1.177. Further, explain how your programs or analyses employed for the Tier 2 and 3 aspects of RG 1.177 would address these potentially risk significant configurations."**

Response:

CPS developed a Fire PRA to address the fire portion of the Individual Plant Examination for External Events (IPEEE). The basic approach used was to find a target set of equipment associated with a particular fire scenario. These are components that may be directly impacted by the fire scenario or may be impacted by fires affecting cables that power or control the components. Based upon the fire scenario, existing initiators from the plant full power internal events PRA were selected to represent the type of plant shutdown that could occur. The list of initiating events and basic events representing the components lost were input as failures into the full power PRA model to derive conditional core damage probabilities (CCDPs) given a fire. This CCDP was typically multiplied by the fire ignition frequency to derive an estimated core damage frequency for a particular fire scenario.

Because the diesel generators are only required to mitigate loss of offsite power events in the PRA analysis, the only fire scenarios that could increase in risk due to the EDG AOT extension are those that would lead to the LOOP initiator. Random occurrences of LOOPs concurrent with internal fire events are believed to be probabilistically insignificant.

The files that contain lists of basic events and initiators representing the individual fire scenarios for the Fire PRA were reviewed to identify those that involve the LOOP initiator. There are five individually modeled in-plant (i.e., not in the Main Control Room) fire scenarios that were identified as leading to a LOOP initiator. These events are shown in Table 3 with their CDF contributions from the Fire PRA. These CDF contributions were corrected for by the full-zone sprinkler system existing in fire zone CB-3a. The three transient cases for zone CB-3a already have failures that would disable both the Division 1 and 2 buses. Therefore, having a diesel generator out of service on Divisions 1 and 2 would not have any additional effect on these three scenarios. The fixed fire scenario for the electrical panel 1PL89JA involves a loss of feedwater components combined with a LOOP event. Because LOOP events by definition result in a loss of feedwater in the station, this fire scenario is a LOOP event

with no additional loss of equipment. This has a similar impact as the internal events LOOP initiator except that it is much smaller in frequency. The target set for fire scenario R-1t area 2 includes feedwater and other BOP components. Again, because balance of plant equipment would be lost due to a LOOP initiator there is no additional effect beyond the LOOP initiator due to the affected balance of plant components. This scenario also has a similar impact as the internal events LOOP initiator except that it is much smaller in frequency.

For the Main Control Room (MCR) fire analysis portion of the fire PRA, conditional core damage probabilities were calculated based upon the target sets located within a particular control room panel. This approach was similar to that used for the fire scenarios described above, except that the target sets used in the MCR analysis often were assumed to be whole divisions of safety related equipment corresponding to the electrical divisions present in the panel. Only one panel (i.e., panel H13-P870) was identified as having the LOOP initiator as the applicable plant initiator. This panel also has Division 1 and 2 components associated with it, although the fire PRA documentation notes that the divisional components are no more than containment isolation valves which should not affect the core damage results. Thus, the impact of a fire within this panel on the level 1 PRA results is essentially equivalent to the LOOP initiator alone. The fire ignition frequency for this panel is, however, several orders of magnitude smaller than the LOOP initiator frequency used in the internal events PRA.

As can be seen from Table 3, the total LOOP frequency due to fire is much smaller than the contribution from internal events. Having a diesel out of service for maintenance either does not impact these scenarios or does not impact them in a way that is significantly different than a LOOP from the internal events PRA.

Based upon the foregoing discussion, these fire scenarios would have very little impact on the calculated risk increase for the EDG AOT extension. This is true, even given consideration that LOOP events from fire may be more difficult to recover from than LOOP events from other sources. Fire-induced LOOP sequences progress in a manner similar to LOOPs with failure of offsite power recovery at 4 hours. The PRA base model gives little credit for recovery of off-site power after 4 hours. Therefore, if the fire ignition frequency for those zones is small compared to the frequency of a LOOP which is unrecovered after 4 hours, then fire risk can make no significant difference in the ICCDP of the EDG AOT extension. Table 3 provides the fire ignition frequency for fire zones causing LOOP scenarios. Fires in the first three zones in that table disable both the Division 1 and Division 2 diesel generators. Therefore, the risk from those zones is unaffected by the proposed AOT extension. For the remaining fire zones, the fire has no impact on safety-related core cooling equipment beyond that caused by the LOOP initiator. Adding the fire ignition frequencies for the remaining three fire zones yields a number that is a few percent of the product of the internal events LOOP initiator and the 4-hr. offsite power non-recovery probability. Therefore, even considering a bounding assumption that offsite power recovery was impossible for all such fires, the contribution to CDF from such fires is an insignificant addition to the contribution already included in the ICCDP calculation from similar non-fire losses of offsite power. No additional risk controls beyond that provided by the existing fire protection program are believed to be necessary.

6. "What is the seismic initiating event frequency for causing a loss of offsite power at your plant?"

Response:

We did not perform a seismic PRA analysis for the IPEEE, so we have not previously determined the seismic LOOP initiator frequency. Ceramic insulators for offsite power transformers tend to be the most vulnerable components in the offsite power system during a seismic event. NUREG/CR-4550, Vol. 4, Rev. 1, Part 3, "Analysis of Core Damage Frequency, Peach Bottom Unit 2 External Events," estimates the median peak ground acceleration at which these ceramic insulators are lost to be approximately 0.25 g. Using this value and EPRI report RP 101-53, "Probabilistic Seismic Hazard Evaluation for Clinton Power Station," the conclusion can be reached that the seismic LOOP initiator is over an order of magnitude less than the LOOP initiating event frequency times the 4 hour non-recovery probability for AC power used in the base PRA model.

Industry experience also supports this conclusion. At least in recent history, seismic events appear to be a relatively minor contributor to the industry LOOP frequency. Evidence of this is provided in EPRI Report TR-110398, "Losses of Offsite Power at U.S. Nuclear Plants – Through 1997." This report records no LOOP events caused by seismic events, even though the database includes over a thousand years of unit operating experience and includes a period of time that had noteworthy earthquakes.

RISK IMPACT DUE TO INTERNAL INITIATING EVENTS

7. "What is the percentage and absolute core damage frequency (CDF) contributions due to the LOOP/station blackout (SBO) initiator?"

Response:

In the current base internal events model, which includes flooding, the total LOOP Initiator contributes to about 18 % of the core damage frequency (i.e., Fussell-Vesely value of 0.181). The absolute CDF contribution from LOOP initiators in the same model is 2.32E-6/yr.

With the Division 1 EDG (i.e., the more risk significant of the Division 1 and 2 EDGs) out of service for maintenance, using the re-quantified zero maintenance CDF case, including internal flooding, the LOOP initiator contributes to about 62% of CDF (i.e., Fussell-Vesely value of 0.615). The absolute CDF contribution from LOOP initiating events in the same Division 1 case model is 1.53E-5/yr.

8. "What are the top five dominant LOOP/SBO sequences? Describe the sequences."

Response:

The top five LOOP/SBO sequences in the base PRA model are as follows.

1. **Sequence TLUL4DD:** This sequence involves a LOOP with failure of the Division 1 and 2 EDGs (i.e., SBO) leading to battery depletion with the additional failure of High Pressure Core Spray (HPCS). HPCS failure may be due to failures of the Division 3 EDG, including common cause events with Divisions 1 and 2. Reactor Core Isolation Cooling (RCIC) which is a steam and DC powered system is successful in maintaining reactor water level for a while. Core damage occurs because of failure to recover an AC power source (e.g., either an EDG or offsite power source). Under these circumstances RCIC eventually will fail because of its long-term dependency on AC power (i.e., power is required for battery chargers to maintain the DC system). This sequence has a frequency of 1.38E-6/yr.

2. **Sequence TLU1U3:** This sequence involves a LOOP with failure of the Division 1 and 2 EDGs (i.e., SBO) with failure of HPCS and failure of RCIC. HPCS failure may be due to failures of the Division 3 EDG including common cause events with Divisions 1 and 2. RCIC fails due to short term problems such as the RCIC turbine failing to start. This sequence has a frequency of 5.4E-7/yr.

3. **Sequence TPU2UV:** In this sequence, a LOOP occurs with at least one of the Division 1 or 2 EDGs successful (i.e., no SBO). Failure of the RCIC system occurs, including causes from its long-term dependence on the Division 1 EDG. HPCS fails (e.g., from its dependence on the Division 3 EDG). Depressurization is successful using Safety Relief Valves (SRVs). Low pressure systems fail at least in part due to hardware failures within the systems or within their immediate support systems (e.g., failure of room cooling). If failure of injection systems is delayed (e.g., ECCS fail due to loss of room cooling) time is available for core cooling using alternate systems (e.g., fire protection). In these circumstances the alternate system failures must also occur. This sequence has a frequency of 2.64E-7/yr.

4. **Sequence TLIU1U3:** In this sequence, a LOOP occurs with failure of the Division 1 and 2 EDGs (i.e., SBO). In addition, it includes at least one open SRV that does not re-close (i.e., Inadvertent Open Relief Valve or Stuck Open Relief Valve). HPCS fails, potentially from the failure of the Division 3 EDG including common cause events with Divisions 1 and 2. RCIC fails in the short term either due to hardware failures or due to loss of motive steam if several SRVs are open. This sequence has a frequency of 6.41E-8/yr.

5. **Sequence TLIU1V:** In this sequence, a LOOP occurs with failure of the Division 1 and 2 EDG (i.e., SBO) with the failure of HPCS. In addition, it includes at least one open SRV that does not re-close (i.e., Inadvertent Open Relief Valve or Stuck Open Relief Valve). RCIC runs in the short term, since

there is no more than 1 SRV open, which would provide an opportunity for EDG recovery. However, RCIC is lost on depressurization of the reactor vessel. In addition, non-diesel hardware failures eliminate the possibility of core cooling. This sequence has a frequency of 3.73E-8/yr.

9. **"Given an EDG out of service, what are the top five cutsets with respect to CDF and large early release frequency (LERF)? What are the sources and values for basic events used in those cutsets? How do the values compare with the plant experience?"**

Response:

The top five cutsets for core damage with the Division 1 EDG out of service are shown in Table 4. The top five cutsets for LERF with the Division 1 EDG out of service are shown in Table 5. The results for the Division 2 EDG are similar except the CDF with the Division 2 EDG out of service is slightly smaller because it does not support all the same mitigation functions as Division 1 (e.g., RCIC and modeled containment venting systems are supported by Division 1). As can be seen from these leading cutsets, even with the Division 1 EDG out of service, total loss of power events do not dominate the leading cutsets.

The values used for the basic events are shown in the two tables. The Loss of RAT initiator and LOOP initiator are discussed under question 10. The ISLOCA initiator values were derived from pipe and valve failure data. The flooding initiator shown was derived given consideration of failure of piping components and maintenance activities that can lead to a flood. CPS has experienced only one of these at-power PRA initiators. A loss of RAT event occurred on April 9, 1996 due to mishandling of switchyard breaker work.

The common cause events shown in the tables are based upon the Multiple Greek Letter (MGL) method of common cause modeling using the data and software (CCFWIN) provided in NUREG/CR-6268, "Common-Cause Failure Database and Analysis System." These common cause events include the air operated ECCS Room Cooler (VY) valves failing to open and Plant Service Water to Shutdown Service Water motor operated valves failing to close. CPS has never experienced these particular common-cause failures.

The individual motor operated valve failure-to-open events were derived through a Bayesian update process that takes into account actual plant experience with this particular failure mode. The value derived through the Bayesian update process is lower than the generic value for MOVs failing to open because plant performance in this area was better than the generic data suggested.

The valve plugging events shown are based upon a surveillance test interval model using generic data with a very long assumed surveillance interval. CPS has not experienced this failure mechanism in ECCS systems.

Failure of the human actions to align the fire protection system and remove internals from fire protection check valve 1FP036 are based on Detailed Human Reliability

Assessments of these particular actions. Both of these actions are required to provide fire protection system flow to the reactor. CPS has never actually had to align the fire protection system for injection to the reactor vessel, but has practiced the evolution in mock-ups during emergency drills.

The basic event representing low Makeup Condensate (MC) storage tank level represents a fraction of time the MC tank has insufficient inventory to supply the reactor with water for the 24 hour mission time. This is a conservative estimate based upon normal inventories maintained in this tank.

The offsite power non-recovery probabilities shown were derived from the EPRI data used to calculate the LOOP frequency discussed in question 10. They were based upon the recovery times for industry LOOP events that were deemed to be applicable to CPS.

Beyond these top five cutsets, leading cutsets that involve both a LOOP as well as the Division 1 EDG out of service are dominated by those from the TLUL4DD sequence described above. These involve the LOOP initiator with the Division 1 EDG out of service for maintenance and failure of the Divisions 2 and 3 EDGs (e.g., EDG failure to run and EDG failure to run common cause). RCIC initially runs providing core cooling. Early battery depletion and consequential RCIC failure, within 1 hour, is prevented by DC load shedding. Early operation of RCIC until battery depletion occurs (i.e., approximately 4 hours) provides time for offsite power recovery. Run time failures of EDGs may provide additional time for offsite power recovery. Core damage occurs because power recovery efforts are unsuccessful.

The top five cutsets for LERF with the Division 1 EDG out of service are shown in Table 5. Note that since LERF is dominated by ISLOCA, LERF is not increased by the extension of the EDG AOT. In fact, the restrictions on unavailability of other equipment while an EDG is out of service cause a reduction in LERF during that time, compared to average unavailabilities. Sensitivity studies were described in the response to question 2 above that included bounding assumptions regarding hydrogen combustion during SBO conditions. Given the assumption that hydrogen combustion can lead to large early releases, with or without recovery of offsite power following core damage during SBO, calculated LERF may be several times the base case. Under the conditions postulated by the sensitivity studies, LERF would be dominated by SBO-Hydrogen combustion challenges.

10. "What is the LOOP initiating event frequency used? What is the basis for the value?"

Response:

The CPS PRA utilizes two initiators that involve a LOOP. One initiator involves a loss of the RAT. This transformer supplies all the balance of plant loads after a turbine generator trip, and is the normal supply for the safety-related AC buses. If the RAT is lost, the safety related buses automatically transfer to the ERAT. If the circumstances surrounding the loss of RAT result in a generator trip, a plant shutdown will occur and all balance of plant loads will be lost. Loss of the ERAT will not result in a plant trip because there normally are no plant buses aligned to the ERAT. The loss of RAT event,

although it represents a partial LOOP, can result in a demand on the EDGs if there are random (i.e., mission time) failures in the ERAT supply to the safety-related buses. These failures are modeled in the CPS PRA. However, the loss of RAT initiator contributes relatively little to the importance of an EDG, because of the additional failure required to cause a demand of the diesel. The loss of RAT initiator frequency used was 4.1E-2/yr. This value was derived using a Bayesian update process that takes into account a loss of RAT event that occurred at CPS in 1996.

The other initiator, referred to as the LOOP initiator, represents a loss of all offsite power (i.e., loss of both the RAT and ERAT sources). Many of the events classified as LOOP events in published EPRI and NRC studies involve losses of an individual source or switchyard, and these would not lead to a LOOP event at CPS. LOOP events for CPS are dominated by weather and wide-reaching transmission system problems. The LOOP frequency and recovery used in the CPS PRA model correlate in many cases to those shown for grid and weather-related events in published studies. Because there is a large degree of independence between the RAT, which is fed from the 345 kV switchyard, and the ERAT, which is fed from a separate 138 kV transmission line coming to the site via another route, the mechanisms that can cause loss of both sources are low in frequency. The LOOP initiator frequency used was 0.97E-2/yr. This number was derived through a review of industry LOOP experience presented in EPRI TR-110398. We have examined the events one-by-one for CPS applicability. Based on the events involving losses of separate offsite power sources, we have calculated a LOOP initiating event frequency and recovery curve. LOOP non-recovery probabilities at any given time are higher for the CPS PRA than for a plant using average values.

11. "What are the common cause failure rates used for EDGs? What is the basis for the values?"

Response:

The following EDG failure to start common cause events have been modeled with the following failure probabilities.

COMMON CAUSE EVENT	FAILURE PROBABILITY
Common cause failure of all three EDGs to start	3.28E-4
Common cause failure of Division 1 and 2 EDGs to start	1.50E-4
Common cause failure of Division 2 and 3 EDGs to start	1.50E-4
Common cause failure of Division 1 and 3 EDGs to start	1.50E-4

The following EDG failure to run common cause events have been modeled with the following failure probabilities.

COMMON CAUSE EVENT	FAILURE PROBABILITY
Common cause failure of all three EDGs to run	1.57E-3 probability for 24 hrs
Common cause failure of Division 1 and 2 EDGs to run	7.08E-4 probability for 24 hrs
Common cause failure of Division 2 and 3 EDGs to run	7.08E-4 probability for 24 hrs
Common cause failure of Division 1 and 3 EDGs to run	7.08E-4 probability for 24 hrs

Common cause values for the CPS PRA model were developed using the MGL approach for common cause modeling. The MGL factors were derived from data and software (i.e., CCFWIN) provided in NUREG/CR-6268. The result is the following Beta and Gamma factors which were applied to the single EDG failure events to calculate the above shown common cause values.

For fails to start events:

Beta = 6.82E-2
Gamma = 5.23E-1
Single Diesel = 9.19E-3

For fails to run events:

Beta = 1.10E-1
Gamma = 5.26E-1
Single Diesel = 2.71E-2 probability for 24 hrs

The single EDG failure probabilities were derived using a Bayesian update of generic EDG failure rates using plant specific EDG experience.

12. **"The proposed changes would allow, if approved, Clinton to perform a corrective maintenance for a failed EDG using the 14-day AOT. For corrective maintenance, a typical PRA assumes that the remaining EDG would be subject to a potential common cause failure. The corresponding incremental conditional core damage probability (ICCDP)/ICLERP can be significantly higher than that calculated for a preventive, planned, maintenance. Provide the ICCDP/ICLERP for a corrective maintenance and demonstrate that it meets the acceptable guidelines set forth in RG 1.177."**

Response:

Corrective maintenance is performed when equipment is failed or is degraded sufficiently that action should be taken to improve future reliability. Common cause failure modeling in PRA analysis presumes that given a failure of individual components the potential exists that the same failure mechanism could exist on like components in the plant due to similar circumstances such as design, maintenance or operating practices. Thus, given an absence of knowledge about the particular failure mechanism

involved, when one failure occurs, a higher likelihood of failure could be assigned to the like components.

In actual practice, if a failure of an EDG does occur, an investigation of the failure mechanism is performed to understand its common cause implications. When a diesel is found to be inoperable, this is a requirement of TS Section 3.8.1, "AC Sources – Operating," Action B.3.1. Alternatively, in accordance with Action B.3.2, the remaining EDG can be tested through a surveillance run to demonstrate that it remains operable. After these particular activities, the state of knowledge about the failure mechanism has been improved to the point that it can be determined with confidence that the particular failure mechanism is not a common cause mechanism that would cause the remaining EDGs to be unable to perform their design functions. If the remaining diesels are found to be inoperable, the 14-day AOT will not be allowed in accordance with TS Section 3.8.1, Action E.1 which requires restoration of one EDG to operable status in 2 hours or 24 hours if the Division 3 EDG is inoperable. Failure to restore one of the EDGs to operable in the required completion time requires that the plant be in Mode 3 in 12 hours and Mode 4 in 36 hours.

Presuming that this operability determination was not made in error, the best estimate of the remaining potential for common cause failures would be represented by the common cause failures and values in the base PRA model. Therefore, once the common cause implications have been investigated and discounted the risk increase due to corrective maintenance activities on a diesel is estimated to be approximately that of preventative maintenance activities. The conclusions of the diesel generator risk study are believed to be valid regardless of whether the unavailability is incurred due to corrective or preventative maintenance.

TIER 2

13. **"With an EDG out of service, what are the most risk significant equipment, or basic events, based on your PRA? Have you performed a systematic search for such equipment? What are the restrictions currently placed on such risk significant equipment? Are there any additional restrictions, in terms of enhancement in technical specifications (TSs) or procedures, needed to avoid risk-significant configurations?"**

Response:

Through the EDG AOT evaluation performed in support of Reference 1, importance measures were examined to generate insights regarding what was driving risk. The following summarizes the conclusions with respect to Division 1 EDG out of service.

The largest contributors to CDF with an EDG removed from service are recovery of offsite power and repair of EDG events, which affect well over half of the CDF.

The importance rankings shown below are based on the Fussell-Vesely measure of importance which shows those failures that control risk at their estimated failure rates. The Risk Achievement Worth (RAW) importance measure shows the relative importance of a particular failure if it were to occur. Like the baseline PRA results, those failures that

have the most impact on risk, if they were to occur, are common cause events. Common cause failures of DC supplies dominate as they are important to the operation of all balance of plant and safety systems. Failure to scram, which would require failure of redundant hardware, is also high in RAW due to its impact on reactivity control and its being backed up by the manually actuated Standby Liquid Control (SLC) System. Common cause failures of the Shutdown Service Water System follow due to the dependence of all engineered safeguards on the component and room cooling supported by this system.

Basic events that do not represent common cause failures that have the highest RAWs include those system trains that would cause the largest risk increases if they were taken out of service. For the case of the Division 1 EDG being taken out of service, the system trains with the highest RAW importance are Division 2 and 3 support systems (e.g. Shutdown Service Water, EDGs, AC power systems and DC power systems), HPCS (the sole injection system in Division 3) and offsite power systems (i.e., the RAT and ERAT). In a similar manner when the Division 2 EDG is taken out of service the Division 1 and 3 systems have the greatest RAWs. With either EDG out of service, the steam driven DC controlled RCIC system is important because of its role in mitigating SBO Events.

The systems identified above are the systems that are most important in preventing and mitigating LOOP events. TS and/or the on-line risk assessment process already limit the removal of these systems from service when the Division 1 EDG is out of service.

No additional procedures or TS changes are suggested beyond the need to monitor plant configuration while an EDG is out of service. This monitoring already occurs as a part of compliance with 10CFR50.65 paragraph (a)(4).

The internal events core damage frequency with Division 1 EDG out for maintenance, including internal flooding, is 2.49E-5/yr, or roughly twice the baseline core damage frequency. Offsite power sources and the remaining two divisions of emergency AC power keep the contribution to risk relatively low for removal of a single EDG from service.

Initiating events that dominate CDF with an EDG removed from service are as follows.

INITIATING EVENT	% CDF
LOOP	62%
Loss of RAT	16%
Transient w/o isolation	7%
Inadvertently Open SRV	3%

LOOP and loss of RAT remain the two dominant contributors to CDF. They are reversed in terms of their contribution from the baseline PRA including internal flooding. In the baseline PRA, loss of RAT contributes 33% and LOOP contributes 18%. That loss of RAT is not as affected by the removal of an EDG is a result of the availability of the ERAT to provide offsite power to safety-related buses.

The largest failures that contribute to CDF are all related to recovery of an AC power source.

FAILURE	% CDF
Recovery of offsite power in 1/2 hr	63%
Recovery of offsite power in 4 hrs	49%
Recovery of Div 2 EDG in 4 hrs	16%
Recovery of EDG in 1/2 hr	11%

The first hardware failures encountered are as follows.

EQUIPMENT FAILURE	%CDF
Common cause failure of ESF room cooling air operated valves	12%
Failure of Div 3 EDG circuit breaker to close	7%
Failure of Div 3 EDG to run	6%
Failure of Div 3 feed breaker from RAT to open	5%
Failure of Div 2 EDG circuit breaker to close	4%
Failure of Div 2 EDG to run	4%

The room cooling control valves provide flow to coolers in all the ESF rooms (i.e., HPCS, RCIC, RHR and LPCS). The breaker failures and EDG failures shown disable whole divisions of core cooling equipment under LOOP conditions.

REFERENCES

1. Letter from M. Coyle (AmerGen) to U.S. NRC, "Clinton Power Station Application for Amendment of Facility Operating License No. NPF-62 for Extension of Diesel Generator Allowed Outage Time (LA-99-016)," dated December 29, 2000.
2. Letter from J. Hopkins (U.S. NRC) to M. Rendeau (AmerGen), "Clinton Power Station, Unit 1 – Request For Additional Information (TAC No. MB0861)," dated February 15, 2001.
3. Letter from D. Pickett (U. S. NRC) to P. Telthorst (Illinois Power Company), "Staff Evaluation of Clinton Power Station Individual Plant Examination – Internal Events (TAC No M74396)," dated March 27, 1997.

TABLE 1
SUMMARY OF MOST SIGNIFICANT BWROG PEER REVIEW/CERTIFICATION FACTS AND OBSERVATIONS

#	Fact/Observation Applicable to the Base PSA Model	Level of Signif.	Priority	Response for D/G Completion Time
AS-7	Correct error in assuming no depressurization for ATWS if HPCS available	A	1	ATWS modeling does not affect Completion Time delta risk.
AS-14	Reassess credit for SX alignment for an ISLOCA in the SDC "B" compartment	A	1	ISLOCA modeling does not affect Completion Time delta risk.
QU-11	Consider adverse impacts of all ISLOCA's on SX alignment success	A	1	ISLOCA modeling does not affect Completion Time delta risk.
QU-24	CDF increases 30% for a truncation limit change of 8E-10	A	1	Convergence plot within response to RAI Question #1 shows Completion Time delta risk converging.
L2-25	Same as F&O #AS-14	A	1	
HR-6	Perform detailed HEP evaluations for risk-significant pre-initiator operator actions	B	1	Current values come from NUREG-1278 (ASEP), and they are already more refined than simple screening values. For the Completion Time calculations, further refinement would reduce the calculated ICCDP and ICLERP.
HR-12	When converting median HEP's to mean, do so consistently	B	1	The dominant HEP's used in this risk study use the mean failure probabilities.
HR-12	To eliminate non-conservatism, perform more detailed HEP's or ensure all screening HEP's are conservative	B	1	Operator actions important for this submittal already have detailed HEP's, plus a sensitivity study shows that other actions are not important for the submittal.
HR-14, -20	Perform operator interviews to verify HRA assumptions, each time an update is done	B	1	For the submittal, Clinton reviewed DC load shed and RCIC steam tunnel isolation with operators. For other human actions important in the submittal, Clinton confirmed that those actions have simple steps and clear indication, and operators are known to be trained on them.

TABLE 1

SUMMARY OF MOST SIGNIFICANT BWROG PEER REVIEW/CERTIFICATION FACTS AND OBSERVATIONS

#	Fact/Observation Applicable to the Base PSA Model	Level of Signif.	Priority	Response for D/G Completion Time
HR-26	Identify dependent operator actions and adjust HEP's, accordingly	B	1	For response to RAI Question #2, Clinton examined cutsets for which d/g was OOS and op. actions were dependent, assumed complete dependency, and showed that the impact was small i.e., meets the guidelines of RG 1.174 and 1.177.
DE-7	List operator actions used in more than one place, to ensure the commonality is reflected in the model	B	1	Covered by response to #HR-26.
ST-4	Provide a discussion of RCS failure pressure and response of the plant to ATWS conditions	B	1	ATWS modeling does not affect Completion Time delta risk.
ST-4	Improve documentation of flooding analysis so that the basis for flood frequencies and impacts in each zone are clear	B	1	Having a D/G OOS has no impact on the flood initiators, and the effect on plant response is included in the Completion Time analysis.
ST-4	Provide adequate technical basis for not requiring RPT, or add it to ATWS event trees	B	1	Covered by response to 1st #ST-4.
ST-4	Include containment failures below the water line in Level 1	B	1	For the Completion Time analysis, the issue is credit for HPCS when the suppression pool could fail. Review of HPCS success sequences, crediting containment heat removal, containment venting and containment failure location probability, demonstrates that incorporating the necessary critical safety function results in an increase in ICCDP (but not ICLERP) but that the increase results in an ICCDP that is within the RG 1.177 Guidelines.
ST-7	Assess value of adding credit for 2ndry containment	B	1	Existing modeling is conservative.

TABLE 1
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#	Fact/Observation Applicable to the Base PSA Model	Level of Signif.	Priority	Response for D/G Completion Time
QU-3	For each of the SETS user programs, provide a description of the information flow and quantitative processes being performed.	B	1	None, since it is a documentation issue, only.
QU-7	Given limitations of cutset model, identify limits of applicability for online risk--e.g. max number of systems that can be removed from service simultaneously	B	1	Online risk model is not applicable.
QU-10, -17	Include identified dependent operator action combinations into the PRA	B	1	Same issue as #HR-26.
QU-10	Include HRA dependency between containment spray initiation and RHR initiation	B	1	Covered by response to #HR-26.
L2-11	Revise Lvl 2 repair credit to be conditional upon failure to repair in the Lvl 1 model	B	1	The late LPI recovery terms already have high failure probability. More importantly, they apply to loss of DHR sequences, which have no impact on LERF.
QU-12	Provide basis for model treatment of asymmetries and identify asymmetries introduced by the model	C	1	None, since it is a documentation issue, only.
QU-27, -30	Perform uncertainty analysis of key assumptions and unique features	C	1	Consistent with the guidance provided by Reg Guide 1.174 several sensitivity studies were performed that examined key modeling assumptions of the PRA relative to the submittal. There are no unusual or unique features of the Clinton Power Station 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.

TABLE 1
SUMMARY OF MOST SIGNIFICANT BWROG PEER REVIEW/CERTIFICATION FACTS AND OBSERVATIONS

#	Fact/Observation Applicable to the Base PSA Model	Level of Signif.	Priority	Response for D/G Completion Time
IE-3	Explain grouping and quantification of initiating events	B	2	None, since it is a documentation issue, only.
IE-10	Systematically evaluate special initiators, including loss of TBCCW	B	2	These initiators do not affect Completion Time delta risk.
IE-10	Clarify nature of LOSW, including # of pumps needed	B	2	This initiator does not affect Completion Time delta risk.
IE-13, -2	Base IE frequency on calendar year	B	2	Numerical impact is negligible.
AS-6	ATWS probability appears to be counted twice in IORV	B	2	Certification team did not understand that the Clinton quantification approach takes care of this issue in the Boolean algebra.
AS-6	Confirm that all critical safety functions are addressed in design of each event tree	B	2	Response to 4th #ST-4 covers the ones that apply to AOT.
AS-6	Include vapor suppression in event trees for LOCA-like events	B	2	LOCA modeling does not affect Completion Time delta risk.
AS-6	Justify the Clinton treatment of pool bypass	B	2	Clinton has confirmed, via reference to calculation, that Clinton does not need upper pool dump to prevent uncovering horizontal vents when flooding the drywell.
AS-6	Justify RCIC credit for the bounding small LOCA	B	2	F&O author misunderstood the Clinton small LOCA definition size. It was based on bounds for RCIC success in the 2000 update.
AS-6	Add credit for auto RPT, based on GE generic calculations for BWR/6	B	2	Covered by response to 3rd #ST-4.
AS-6	Include effects of RCIC gland seal air compressor failure	B	2	Clinton confirmed that it is not needed for short-term RCIC success.
AS-6	Reposition the ADS inhibit node in the event tree	B	2	ATWS modeling does not affect Completion Time delta risk.
AS-15	Remove boron retention credit for SLOCA ATWS below TAF	B	2	ATWS modeling does not affect Completion Time delta risk.

TABLE 1
SUMMARY OF MOST SIGNIFICANT BWROG PEER REVIEW/CERTIFICATION FACTS AND OBSERVATIONS

#	Fact/Observation Applicable to the Base PSA Model	Level of Signif.	Priority	Response for D/G Completion Time
AS-19	Model correct injection path for LPCI for ATWS	B	2	ATWS modeling does not affect Completion Time delta risk.
AS-21	Include questions for all critical safety functions after recovery to remove need to assign paths to conservative LERF bins	B	2	Existing modeling is conservative.
TH-7	Reevaluate basis for ISLOCA success criteria with RCIC only	B	2	There is a typographical error in the F&O. Same issue as 5th #AS-6.
TH-8	Document the technical bases for room cooling assumptions, especially for RCIC in SBO and MCR in SBO and loss of MCR cooling	B	2	Analyses exist for Main Control Room and for RCIC.
SY-25	Ensure system notebooks are carefully stored and at least one copy is protected from loss	B	2	Documentation issue, only.
SY-25	Ensure MAAP results are carefully stored and at least one copy is protected from loss	B	2	Documentation issue, only.
SY-25	Create formal tracking system for errors and issues identified between model updates	B	2	Response to RAI Question #3 describes why the PRA represents as-built, as-operated plant.
SY-26	Ensure system engineer expertise is used in preparation and review of system notebooks	B	2	Response to RAI Question #3 describes why the PRA represents as-built, as-operated plant.
ST-5	Several elements dismissed via phenomenological papers should be modeled explicitly	B	2	For the Completion time application, the 2 "B" items in this F&O are not applicable

TABLE 1
SUMMARY OF MOST SIGNIFICANT BWROG PEER REVIEW/CERTIFICATION FACTS AND OBSERVATIONS

#	Fact/Observation Applicable to the Base PSA Model	Level of Signif.	Priority	Response for D/G Completion Time
ST-5	Examine containment failure sequences to define failure location, size, and impact on equipment	B	2	Same issue as 4th #ST-4.
QU-6	To resolve truncation issues, develop the model completely in CAFTA-W and use FORTE or NURELMCS for quantification	B	2	Covered by response to #QU-24.
QU-8, -15, -26	Reduce conservatism by adding to the mutually exclusive file all combinations of equipment OOS prohibited by Tech Spec or operating practices	B	2	No change is needed, since experience has shown that these combinations contribute negligibly to results.
QU-18	Delete the RCIC FTR recovery term, justify it, or use a time-phased approach	B	2	The number is valid, based on NSAC-161. No change needed.
QU-22	Truncate Lvl 2 model at a lower value, consistent with sub-element QU-22	B	2	The Completion Time analysis is quantified in such a way as to include maintenance unavailabilities related to the AOT in cutsets several orders of magnitude lower than in previous base models.
QU-23	Convergence has not occurred at E-10 truncation	B	2	Covered by response to #QU-24.
QU-28	Perform sensitivity study that eliminates all credit for hardware repair	B	2	Covered by response to #HR-26.
L2-19	Revise the containment failure mode assumed for ATWS	B	2	ATWS modeling does not affect Completion Time delta risk.
MU-4	Revise PSA Standard Review instruction to ensure CCF is considered when evaluating design changes	B	2	Not applicable to Completion Time analysis.
MU-4	Revise the PRA Review Standard to include CW traveling screens in the list of PRA-related systems	B	2	Not applicable to Completion Time analysis.

TABLE 1
SUMMARY OF MOST SIGNIFICANT BWROG PEER REVIEW/CERTIFICATION FACTS AND OBSERVATIONS

#	Fact/Observation Applicable to the Base PSA Model	Level of Signif.	Priority	Response for D/G Completion Time
	The designators for the F&O's are as follows.			
	IE - Initiating Event			
	AS - Accident Sequence Analysis			
	TH - Thermal Hydraulic Analysis			
	SY - System Analysis			
	DA - Data Analysis			
	HR - Human Reliability Analysis			
	DE - Dependency Analysis			
	QU – Quantification			
	MU – Maintenance & Update			
	Bases for Assignments of Priorities			
	High Priority (1)	Peer Review/Certification "A" level of significance; or		
		"B" level of significance, and associated with a Sub-element receiving a Grade of "1" or "N/A."		
	Medium-High Priority (2)	"B" level of significance, and associated with a Sub-element receiving a Grade of "2."		
	Medium Priority (3)	"B" level of significance, and associated with a Sub-element receiving a Grade of "3."		
	Low Priority (4)	"C" level of significance.		

TABLE 2

PRA PEER REVIEW CERTIFICATION TEAM EXPERIENCE

TEAM MEMBER	EXPERIENCE SUMMARY			
	Degree	Years Experience	Years of PRA Experience	Selected PRA Projects
E. T. Burns	BS – Engineering Science – RPI MS – Nuclear Engineering – RPI Ph.D., Nuclear Engineering , RPI	26	21	• Technical reviewer of Level 1 IPEs for twenty one BWR plants • Manager, technical advisor, or lead engineer on many IPEs/PRAs for BWR plants • Lead engineer on several containment safety studies
Ed Vezey	B.S. Mechanical Engineering – Texas A & M Univ.	45+	30	• 17 years of BWR experience with GENE Division • PSA application to Tech Specs for TPC • Managed PSA for a BWR 6
L.K. Lee	B.S. Mechanical – U.C. Berkeley	8	8	• Technical reviewer or modeler for over 10 Level 1 and Level 2 IPEs. • Lead engineer on over 10 Probabilistic Shutdown Safety Assessments • Technical engineer for applying risk informed evaluations to support plant ISI programs
Bruce Logan	BS, Electrical Engineering – Auburn University MS, Electrical Engineering – Auburn University	24	24	• INPO PRA Coordinator • Harris PRA Review • Manager, Duke Power PRA Group • Oconee, McGuire and Catawba PRAs

TABLE 2

PRA PEER REVIEW CERTIFICATION TEAM EXPERIENCE

TEAM MEMBER	EXPERIENCE SUMMARY			
	Degree	Years Experience	Years of PRA Experience	Selected PRA Projects
Xavier Polanski	B.A., Physics, Ripon College	23	13	• Team Leader for ComEd Zion, Byron, and Braidwood IPEs and subsequent revisions • Chairman Westinghouse Owners' Group Risk Based Technology Working Group • Current Project Manager, Quad Cities PSA Update and CAFTA Conversion
Gerry Kindred	BS, Technology/ Health Physics Specialty, Univ. of State of New York AS – Nuclear Engineering Technology Chattanooga State	20	1	• On-line PRA Evaluations • Project Manager – Perry Safety Monitor

TABLE 3

INDIVIDUALLY MODELED FIRE SCENARIOS
FROM THE FIRE PRA INVOLVING THE LOOP INITIATOR

Individual In-Plant Scenarios					
Fire Zone	Scenario	Ignition Freq.	CCDP	Reduction Factor for Sprinklers	CDF From Fire PRA
CB-3a	Transient Area 6	3.65E-06 per year	1.07E-01	20	1.95E-08 per year
CB-3a	Transient Area 8	4.23E-06 per year	1.00E-02	20	2.12E-09 per year
CB-3a	Transient Area 9	4.39E-06 per year	9.91E-03	20	2.18E-09 per year
CB-3a	1PL89JA	6.20E-05 per year	1.87E-05	NA	1.16E-09 per year
R-1t	Transient Area 2	9.98E-05 per year	3.51E-05	NA	3.50E-09 per year

Main Control Room Scenarios				
Scenario	Ignition Freq	CCDP	RSP Recovery	CDF
1H13-P870	3.17E-05 per year	2.44E-04	1	7.73E-09 per year

Total Ignition Frequency = 2.06E-04/yr

TABLE 4

CORE DAMAGE CUTSETS WITH
DIVISION 1 EDG OUT OF SERVICE

Total Core Damage Frequency Including Flooding = 2.49E-05/yr

Cutset No.	Inputs	Description	Event Probability	Cutset Freq (per year)
1	XCDVYCCAVO	COMMON CAUSE FAILURE FOR VY COOLER DISCHARGE VALVE	7.27E-05	3.07E-07
	E1FP036CVH	OPERATORS FAIL TO REMOVE INTERNALS FROM CHECK VALVE 1FP036	1.03E-01	
	IEYLOSRATI	LOSS OF RESERVE AUX TRANSFORMER INITIATOR	4.10E-02	
2	XCDVYCCAVO	COMMON CAUSE FAILURE FOR VY COOLER DISCHARGE VALVE	7.27E-05	2.52E-07
	ADG01KADGM	DG01KA OUT OF SERVICE – PREVENTIVE MAINTENANCE	(TRUE)	
	YOS0TO4SWH	OFF-SITE POWER NOT RECOVERED WITHIN 4 HOURS	5.00E-01	
	YOSHALFSWH	FAILURE TO RECOVER OFFSITE POWER IN ONE HALF HOUR	7.14E-01	
	IEYLOOPXXI	LOSS OF OFF-SITE POWER INITIATOR	9.70E-03	
3	XCDVYCCAVO	COMMON CAUSE FAILURE FOR VY COOLER DISCHARGE VALVE	7.27E-05	2.12E-07
	EALIGNVSYH	OPERATOR FAILS TO ALIGN FIRE PROTECTION SYSTEM FOR INJECTION	7.10E-02	
	IEYLOSRATI	LOSS OF RESERVE AUX TRANSFORMER INITIATOR	4.10E-02	
4	XCDVYCCAVO	COMMON CAUSE FAILURE FOR VY COOLER DISCHARGE VALVE	7.27E-05	1.82E-07
	FMCTANKTKL	INVENTORY IN THE MC TANK INSUFFICIENT FOR MISSION TIME	2.50E-01	
	FLOOD0034	Flood Initiator in Zone	1.00E-02	
5	XSX14CCMVC	COM CAUSE WS TO SX A B&C VALVES FAIL TO CLOSE	3.96E-05	1.67E-07
	E1FP036CVH	OPERATORS FAIL TO REMOVE INTERNALS FROM CHECK VALVE 1FP036	1.03E-01	
	IEYLOSRATI	LOSS OF RESERVE AUX TRANSFORMER INITIATOR	4.10E-02	

TABLE 5

LARGE EARLY RELEASE FREQUENCY (LERF)
CUTSETS WITH THE DIVISION 1
EDG OUT OF SERVICE

TOTAL LERF = 1.41E-07/YR

Cutset #	Inputs	Description	Event Probability	Cutset Freq. (per yr)
1	R2F039BXVP IEYISLOCFI	INJECTION LINE MANUAL VALVE F039B PLUGGED INTERFACING SYSTEM LOCA INITIATOR IN SDC SYSTEM	1.75E-02 2.54E-06	4.45E-08
2	R2F039BXVP IEYISLOCBI	INJECTION LINE MANUAL VALVE F039B PLUGGED INTERFACING SYSTEM LOCA INITIATOR IN FW SYSTEM	1.75E-02 2.28E-06	3.99E-08
3	IEYISLOCFI ERHF094MVO	INTERFACING SYSTEM LOCA INITIATOR IN SDC SYSTEM SX TO RHR F094 MOV FAILS TO OPEN	2.54E-06 1.72E-03	4.37E-09
4	IEYISLOCFI ERHF096MVO	INTERFACING SYSTEM LOCA INITIATOR IN SDC SYSTEM SX TO RHR F096 MOV FAILS TO OPEN	2.54E-06 1.72E-03	4.37E-09
5	R2F042BMVO IEYISLOCFI	FAILURE OF RHR INJ MOV F042B TO OPEN INTERFACING SYSTEM LOCA INITIATOR IN SDC SYSTEM	1.72E-03 2.54E-06	4.37E-09