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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: Response to Request for Additional Information Regarding Risk-Informed Inservice Inspection Program

Reference: Letter from AmerGen Energy Company, LLC to U.S. NRC, "Alternative to the ASME Boiler and Pressure Vessel Code Section XI Requirements for Class 1 and Class 2 Piping Welds Risk Informed Inservice Inspection Program," dated October 15, 2001

In the above referenced letter, AmerGen Energy Company (AmerGen), LLC, requested approval of an alternative to the existing edition of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," requirements for the selection and examination of Class 1 and 2 piping welds. This alternative utilizes the Risk-Informed Inservice Inspection (RI-ISI) program methodology discussed in Electric Power Research Institute (EPRI) Topical Report 112657, "Revised Risk-Informed Inservice Inspection Evaluation Procedure," Revision B-A, December 1999. The NRC verbally requested additional information in support of their review of the referenced letter. The attachment to this letter provides the requested information.

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

Respectfully,

Keith R. Jury

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

Attachment: Response to Request for Additional Information Regarding Risk-Informed Inservice Inspection Program

cc: Regional Administrator – NRC Region III
NRC Senior Resident Inspector – Clinton Power Station

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ATTACHMENT

Response to Request for Additional Information Regarding Risk-Informed Inservice Inspection Program

Question 1

What is the date and version number of the Clinton Power Station (CPS) PRA used for their Risk-Informed Inservice Inspection (RI-ISI) analysis? In its submittal, the licensee only states on page 3 that the "PRA used for the risk determinations for this regulatory application is a recent upgrade to the Clinton Power Station Individual Plant Examination (IPE)."

Response 1

The Probabilistic Risk Assessment (PRA) model used for the CPS RI-ISI analysis is the CPS Revision 3a PRA model, which was approved on December 29, 2000.

Question 2

The staff found that the IPE satisfied the intent of Generic Letter 88-20, however, the staff states in the conclusion of their Safety Evaluation Report (SER) of the CPS IPE:

"The staff, however, believes that the licensee did not provide an adequate technical justification for: (1) the extensive use of generic data, (2) the credit taken for equipment repair (other than diesel generator), (3) the credit taken for containment performance under hydrogen combustion upon power recovery (negligible failure probability) and (4) the credit taken for containment performance under ex-vessel steam-explosion conditions (no containment failure). These limitations may limit the IPE's usefulness for other regulatory applications."

How have these issues been addressed in updated versions of the PRA?

Response 2

The following provides a discussion of how NRC concerns identified in the CPS IPE SER have been addressed in the context of the RI-ISI evaluation.

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 and Maintenance Rule records between 1994 and 1999. 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.

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Repair and Recovery Modeling

In Reference (1), 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 CPS 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. A more recent sensitivity study using the current Revision 3a PRA model was performed for the Emergency Diesel Generator allowed out of service time extension license amendment request. The conclusions of the risk analysis were found to remain valid even after evaluating the impact of the repair and recovery modeling.

For the RI-ISI analysis the PRA results are used to group, or bin, pipe failures into High, Medium, and Low consequence categories. These bins are each at least two orders of magnitude wide (i.e., 100 times). Because of the relatively broad categories used, the impact of CPS repair and recovery modeling on the RI-ISI program is believed to be minimal.

Containment Performance Under Hydrogen Combustion

In Reference (1), the NRC identified potential weaknesses in the Level 2 analysis of the CPS IPE. These weaknesses were particularly concerned with hydrogen ignition after power recovery under station blackout (SBO) conditions and with the treatment of fuel coolant interactions. These issues have been explicitly addressed as a part of subsequent updates to the PRA.

In the most recent update to the CPS 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 SBO. The time phases utilized in the analyses 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. This ignition of hydrogen is assumed to cause a pressure spike and possible failure of 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 to occur unless other phenomenological events lead to loss of drywell integrity (e.g., steam explosions or drywell-to-wetwell vacuum breaker failure). Late recovery of offsite power however, can permit sufficient buildup of hydrogen that, if ignited, could exceed the ultimate capacity of containment. 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 these failures occur late in the event, however, these sequences do not contribute to the large early release frequency (LERF).

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, "Severe Accident Risks: An Assessment of Five U.S. Nuclear Power Plants, Final Report," for Mark III containments. However, the impact on the RI-ISI results is expected to be minimal since the hydrogen can only build up to the point that it presents a containment challenge in events in which the hydrogen ignitors are

unavailable (e.g., SBO events). For many of the RI-ISI events the initiator involves a LOCA through a particular piping segment. Since these failures would not make hydrogen ignitors unavailable, hydrogen buildup to the point that containment structural integrity could be compromised would not be likely. For piping systems which would only cause failure of systems when unrelated initiators have occurred, SBO sequences that could cause large hydrogen accumulations are only a fraction of the core damage frequency. As long as the LERF fraction remains below 0.1 of the core damage frequency, the LERF results would not be controlling for the RI-ISI evaluation. Based upon the results of a sensitivity study performed for the Emergency Diesel Generator allowed out of service time extension license amendment request, which used assumptions more consistent with NUREG 1150, the conditional probability of a large early release given core damage due to an SBO is less than 0.1. As a result, binning would be based upon the core damage frequency results.

Containment Performance Under Ex-Vessel Steam Explosion Conditions (No Containment Failure)

The concern noted by the NRC in Reference (1) is directed at ex-vessel steam explosions. 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. Thus, the issue of ex-vessel steam explosions has been addressed.

Reference:

(1) 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