

## STAFF EVALUATION REPORT

### PALISADES PLANT INDIVIDUAL PLANT EXAMINATION SUBMITTAL

DOCKET NO. 50-255

#### I. INTRODUCTION

On January 29, 1993, Consumers Power Company submitted the Palisades Nuclear Plant Individual Plant Examination (IPE) in response to Generic Letter (GL) 88-20 and associated supplements. On April 22, 1994, the staff sent questions to the licensee requesting additional information. The licensee responded in a letter dated July 22, 1994.

A "Step 1" review of the Palisades IPE submittal was performed and involved the efforts of Science & Engineering Associates, Inc., Scientech, Inc./Energy Research, Inc., and Concord Associates in the front-end, back-end, and human reliability analysis (HRA), respectively. The Step 1 review focused on whether the licensee's method was capable of identifying vulnerabilities. Therefore, the review considered (1) the completeness of the information and (2) the reasonableness of the results given the Palisades design, operation, and history. A more detailed review, a "Step 2" review, was not performed for this IPE submittal. A summary of contractors' findings is provided below. Details of the contractors' findings are in the attached technical evaluation reports (Appendices A, B, and C) of this staff evaluation report (SER).

In accordance with GL 88-20, Consumers Power Company proposed to resolve Unresolved Safety Issue (USI) A-45, "Shutdown Decay Heat Removal Requirements." No other specific USIs or generic safety issues were proposed for resolution as part of the Palisades IPE.

#### II. EVALUATION

Palisades is a two-loop pressurized water reactor (PWR) with a large, dry containment. The Palisades IPE has estimated a core damage frequency (CDF) of  $5.1E-05$ /reactor-year from internally initiated events, including the contribution from internal floods. The Palisades CDF compares reasonably with that of other PWR plants. Transients (including loss of offsite power) contribute 36 percent, loss of coolant accidents (LOCAs) 28 percent, station blackout 18 percent, anticipated transients without scram 9 percent, steam generator tube rupture (SGTR) 5 percent. Internal flooding contributes less than one percent to overall CDF. The important system/equipment contributors to the estimated CDF that appear in the top sequences are: failure of secondary cooling and once through cooling in transient events, failure of high pressure injection in the injection or recirculation phases in transient events, and unavailability of PORVs causing failure of once through cooling, also in transients. The licensee's Level 1 analysis appears to have examined the significant initiating events and dominant accident sequences.

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Based on the licensee's IPE process used to search for decay heat removal (DHR) vulnerabilities, and review of Palisades plant-specific features, the staff finds the licensee's DHR evaluation consistent with the intent of the USI A-45 (Decay Heat Removal Reliability) resolution.

The licensee performed an HRA to document and quantify potential failures in human-system interactions and to quantify human-initiated recovery of failure events. The licensee identified the following operator actions as important in the estimate of the CDF: operator fails to align makeup to the condensate storage tank, safety injection raw water tank switch miscalibrated high, miscalibration of all auxiliary feedwater (AFW) low suction pressure switches, miscalibration of all flow instruments in the AFW headers, operator fails to open power operated relief valves (PORVs) and associated motor operated valves, and operator fails to align alternate AFW pump suction source.

The staff concluded, however, that there were limitations, described below, in the HRA approach used by the licensee:

1. The treatment of pre-initiator and post-initiator errors using the "quasi-generic" Technique for Human Error Rate Prediction (THERP) models limits the degree of in-depth insights about plant-specific factors influencing human performance. Such plant-specific insights are valuable products of the HRA.
2. The licensee used screening values for post-initiator human actions that are significantly lower than typically used for post-initiator actions and did not include dependencies in the initial quantification. These facts together raise a concern about the potential for screening out risk-significant actions/sequences. The licensee believes that it is unlikely risk-significant actions/sequences were screened out because (a) sequences with very low frequencies were retained in the model even though they were not reported, and (b) dependencies were treated in the subsequent model quantification. While the licensee's argument is plausible, the documentation of the licensee's approach and justification is very limited. The staff is not able to confirm whether or not the approach provided reasonable assurance that actions/sequences were not inappropriately screened out.
3. The treatment of diagnosis for post-initiator actions is not consistent with most nuclear plant HRA approaches. Most post-initiator human actions in the Palisades model are quantified using THERP. Based on our limited review of several sample "generic THERP" calculations, it appears that the licensee did not use the THERP diagnostic model but instead treated all post-initiator responses as rule-based responses to alarms/annunciators. However, most HRA approaches treat the "cognitive" actions associated with diagnosis, detection, decision-making, etc. distinctly different, often using a time-based probability. None of the Palisades documentation reviewed discusses the rationale for excluding diagnosis. In our view,

the limited discussion/treatment of the diagnosis portion of the response represents a limitation in the documentation of the HRA, and may reflect a weakness in the licensee's understanding of human behavior in severe accidents.

4. In general, quantitative results for post-initiator actions are consistent with nominal values reported in other PRAs, and consistent with the range typically generated using THERP. However, in one of the two exceptions in which Accident Sequence Evaluation Program (ASEP) models were used instead of THERP, the calculated human error probability (HEP) was the lowest value of all post-initiator HEPs, and it was also among the most important human actions. Typically, ASEP would be expected to produce higher values.

Regardless of these limitations, however, it appears that the licensee in its systemic examination gained an understanding of the quantitative impact of human performance on core damage and radioactive material release frequencies such that a potential vulnerability was not overlooked.

The licensee evaluated and quantified the results of the severe accident progression through the use of a containment event tree and considered uncertainties in containment response through the use of sensitivity analyses. The licensee's back-end analysis appeared to have considered important severe accident phenomena. Among the Palisades conditional containment failure probabilities: early containment failure is 2 percent with direct containment heating the primary contributor; late containment failure is 15 percent with overpressurization due to steam and/or noncondensable gas being the primary contributor, and bypass is 6 percent with SGTR the primary contributor. The containment remains intact 77 percent of the time after taking into account the committed containment improvement discussed below to plug the cavity drain lines in the containment sump. (Without the improvement, early containment failures are an additional 31 percent of CDF.) Early radiological releases are dominated by SGTR and late releases are dominated by either station blackout or loss of secondary cooling transient coincident with loss of high pressure injection in the recirculation mode (i.e., loss of feed and bleed) sequences. The licensee's response to containment performance improvement program recommendations is consistent with the intent of GL 88-20 and associated Supplement 3.

Some insights and unique plant safety features identified at Palisades are:

1. Increased PORV capability based on new valves installed in 1990, and operation with PORV block valves closed.
2. Automatic switchover of emergency core cooling systems from injection to recirculation.
3. Nitrogen back-up system behind the normal air supply for selected plant equipment.
4. Multiple sources of offsite power, via 345 kV lines, to the plant switchyard.
5. Independence of high pressure safety injection (HPSI) pumps from low pressure safety injection pumps (LPSI) during recirculation without the need to "piggy-back" suction from the LPSI pumps during recirculation.

6. Potential single failure of HPSI system due to inadvertent closure of pump minimum flow recirculation valves.
7. Four containment air coolers provide separate cooling apart from containment sprays.

The licensee used a set of three questions to define a vulnerability:

1. Do the IPE results meet the NRC's safety goal for core damage?
2. Are the IPE results consistent with other probabilistic risk analyses (PRAs)?
3. Are large releases more than 10 percent of CDF?

Based on this definition, the licensee did not identify any vulnerabilities.

Plant improvements, however, were identified. Based on the IPE, the licensee has decided to install a new switchyard transformer to help reduce the frequency of loss of offsite power transients at the plant. This improvement has been credited in the IPE and implemented.

In addition, the licensee has committed to plug two, one-inch, reactor cavity drain lines during the 1997 refueling outage. These drain lines form a direct pathway for radiological release from the reactor cavity to the engineered safety features (ESF) sump. This will help delay the relocation of core debris from the reactor cavity to the ESF sump.

To enhance the availability of the containment cavity flooding system, the licensee completed procedural and inspection changes before the end of the 1995 refueling outage.

The licensee is also evaluating additional means of providing makeup to the safety injection refueling water tank following an interfacing system LOCA or SGTR, but no formal commitment has been made regarding this issue.

### III. CONCLUSION

Based on the above findings, the staff notes that: (1) the licensee's IPE is complete with regard to the information requested by GL 88-20 (and associated guidance NUREG-1335), and (2) the IPE results are reasonable given the Palisades design, operation, and history. As a result, the staff concludes that the licensee's IPE process is capable of identifying the most likely severe accidents and severe accident vulnerabilities, and therefore, that the Palisades IPE has met the intent of GL 88-20.

It should be noted that the staff's review primarily focused on the licensee's ability to examine Palisades for severe accident vulnerabilities. Although certain aspects of the IPE were explored in more detail than others, the review is not intended to validate the accuracy of the licensee's detailed findings (or quantification estimates) that stemmed from the examination. Therefore, this SER does not constitute NRC approval or endorsement of any IPE material for purposes other than those associated with meeting the intent of GL 88-20. The licensee has not indicated its intent to use the IPE as a "living PRA." Regardless, the staff encourages the licensee to improve the Palisades IPE in order to make it

a valuable tool for other applications. Without the HRA improvements discussed above the staff believes that the Palisades IPE will be limited in regard to future regulatory uses.

**Date:** February 7, 1996

APPENDIX A  
PALISADES NUCLEAR PLANT INDIVIDUAL PLANT EVALUATION  
TECHNICAL EVALUATION REPORT  
(FRONT-END)

Enclosure 2