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AEP INDIANA MICHIGAN POWER

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U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Mail Stop O-P1-17
Washington, DC 20555-001

Donald C. Cook Nuclear Plant Units 1 and 2
LICENSE AMENDMENT REQUEST FOR ONE-TIME EXTENSION OF ESSENTIAL SERVICE WATER SYSTEM ALLOWED OUTAGE TIME – ADDITIONAL INFORMATION

Reference: Letter from J. E. Pollock, Indiana Michigan Power Company, to Nuclear Regulatory Commission Document Control Desk, "License Amendment Request for One-Time Extension of Essential Service Water System Allowed Outage Time," submittal AEP:NRC:2741, dated July 26, 2002

Indiana Michigan Power Company (I&M), submitted a license amendment request for a one-time extension of the 72-hour allowed outage time (AOT) for the Donald C. Cook Nuclear Plant Unit 1 and Unit 2 essential service water (ESW) systems. This request, which was transmitted in the referenced letter, requested a one-time AOT extension to 140 hours to provide for the implementation of a contingency plan during the replacement of the present ESW pumps with pumps having a modified design.

During a telephone conference subsequent to the license amendment request, Nuclear Regulatory Commission (NRC) personnel requested additional information regarding both the probabilistic risk assessment that was performed as part of the request, and the ESW pump testing conducted by the pump's vendor.

The attachment to this letter contains the requested information.

A001

This letter contains no new commitments. Should you have any questions, please contact Mr. Brian A. McIntyre, Manager of Regulatory Affairs, at (269) 697-5553.

Sincerely,

J. E. Pollock
Site Vice President

c: K. D. Curry
J. E. Dyer
MDEQ – DW & RPD
NRC Resident Inspector
R. Whale

AFFIRMATION

I, J. E. Pollock, being duly sworn, state that I am Site Vice President of Indiana Michigan Power Company (I&M), that I am authorized to sign and file this request with the Nuclear Regulatory Commission on behalf of I&M, and that the statements made and the matters set forth herein pertaining to I&M are true and correct to the best of my knowledge, information, and belief.

Indiana Michigan Power Company

J. E. Pollock
Site Vice President

SWORN TO AND SUBSCRIBED BEFORE ME

THIS 23 DAY OF AUGUST, 2002

Jennifer L Kernosky
Notary Public

My Commission Expires 5/26/05

JENNIFER L KERNOSKY
Notary Public, Berrien County, Michigan
My Commission Expires May 26, 2005

ATTACHMENT TO AEP:NRC:2741-01

ADDITIONAL INFORMATION REGARDING LICENSE AMENDMENT REQUEST TO EXTEND THE ESSENTIAL SERVICE WATER SYSTEM ALLOWED OUTAGE TIME

In a letter from J. E. Pollock, Indiana Michigan Power Company (I&M), to Nuclear Regulatory Commission (NRC) Document Control Desk, AEP:NRC:2741, dated July 26, 2002, I&M, the licensee for Donald C. Cook Nuclear Plant (CNP) Unit 1 and Unit 2, requested a one-time license amendment extending the 72-hour essential service water (ESW) allowed outage time (AOT) to 140 hours. This request was made to allow the implementation of a contingency plan during the replacement of the current ESW pumps with pumps having a modified design, should it be necessary.

In an August 9, 2002, telephone conference, NRC personnel requested additional information regarding both the probabilistic risk assessment that was performed to support the license amendment request, and the ESW pump testing conducted by the pump's vendor. This information is provided below.

<u>NRC Request 1</u>

Provide the values for the incremental conditional core damage probability (ICCDP), incremental conditional large early release probability (ICLERP), change in average core damage frequency (ΔCDF), and change in average large early release frequency (ΔLERF).

<u>I&M Response</u>

This information, which has been calculated on the basis of a 140-hour AOT, is provided in Table 1.

Following the guidance in Regulatory Guide 1.177, Section 2.4, and Regulatory Guide 1.174, Section 2.2.4, I&M has concluded that the Technical Specification AOT change has only a small quantitative impact on plant risk. Table 1 of this attachment illustrates that the guideline for ICCDP, 5.0E-07, is exceeded by a maximum of 3.9E-07 and the guideline for ICLERP, 5.0E-08, is satisfied. The average CDF is about 4.9E-05/year and the average LERF is about 5.6E-06/year. Based on these values, the average ΔCDF and ΔLERF are within the Regulatory Guide 1.174 guidelines when determined on a one-year basis.

Although the ICCDP guideline value of 5.0E-07 is exceeded, the following items provide justification for exceeding this value on a one-time basis.

Improvements in Reliability and Availability

The benefit of the license amendment request is to improve the material condition of the ESW pumps. It is anticipated that the upgrades to the pumps will improve performance, availability and reliability. These activities directly support the goals and objectives of the Maintenance Rule and are indicative of a philosophy that supports good preventive maintenance practices. The long-term benefits are expected to more than compensate for the small short-term increase in risk.

Planned Activities

Except for routine surveillances of short duration, entry into an action statement is typically a result of an unanticipated component malfunction (unplanned corrective maintenance action). However, these maintenance activities have been planned in anticipation of ESW pump degradation. As a result, the plant will be put into a stable and safe configuration. Pre-job briefs will have been performed prior to removing the pump from service. Technical expertise will be present. Parts and tools will be pre-staged. Contingencies and compensatory measures will have been considered. Lessons learned from the early pump upgrades will be used to improve the installation of subsequent pump upgrades.

Risk of Shutting Down

I&M has made a comparison of the risk of shutting down the plant pursuant to the existing Technical Specification AOT to the increase in risk of continued operation. The risk of shutting down a unit has been determined to be greater than the risk of continued operation while performing these maintenance activities. The ICCDP of shutting down Unit 1 is approximately 4.62E-06 and the ICCDP of shutting down Unit 2 is approximately 4.69E-06. For each unit, this value was conservatively determined by setting the initiating event frequency for transients with power conversion available (IE-TRA) to zero and solving the Probabilistic Risk Assessment (PRA) model using Safety Monitor™ software to obtain a new result. The difference between the base case in core damage frequency (CDF) and this value is the contribution to CDF from IE-TRA. The IE-TRA CDF contribution was then divided by the IE-TRA frequency of 1.54 to estimate the ICCDP attributed to shutting down a unit. Transients without power conversion were not included in this estimate. The ICCDP values for continued operation are provided in Table 1. This approach has previously been accepted by the NRC for notice of enforcement discretion submittals by I&M.

<u>NRC Request 2</u>

Provide a discussion of the PRA quality. What audits and checks have been performed? What were the findings, and what actions have been taken regarding the findings?

I&M Response

Background

I&M submitted the initial CNP individual plant examination (IPE) to the NRC staff for review on May 1, 1992. The IPE data analysis was revised in a model completed in June 1994. In response to the NRC's requests for additional information, the Human Reliability Analysis (HRA) was revised in June 1994 and October 1995. The NRC staff evaluation report was sent to I&M on September 6, 1996, and concluded that the IPE satisfied the requirements of Generic Letter 88-20, "Individual Plant Examination for Severe Accident Vulnerabilities – 10 CFR 50.54(f)," dated November 23, 1988, and the guidance given in NUREG-1335, "Individual Plant Examination: Submittal Guidance, Final Report," dated August 1, 1989.

Revisions to the CNP IPE not considered in the staff evaluation report included updating plant specific data in May 1996, changing from a linked fault tree model to a top logic model in August 1997, and a major revision in 2001. The table below provides a summary of these changes.

PRA Version	**Rev. 0**	**Rev. 1**	**Rev. H/H1**	**PA-96-03**	**Rev. HH**	**2001 Update**
Reason for revision	Submittal of Initial IPE	Updated Data Analysis	Revised HRA Methodology	Updated Data Analysis	Converted to Top Logic Model	Major Revision (described below)
Date of revision	May-92	Jun-94	Oct-95	May-96	Aug-97	Jun-01

Summary of the 2001 PRA Update

The following general changes were made to the PRA model:

The existing Computer Aided Fault Tree Analysis (CAFTA) model was converted to a WinNUPRA™ model to better support implementation of Safety Monitor™ for on-line and shutdown risk evaluation.

The PRA was updated to include new plant specific data, procedure and/or design changes, revision of the treatment of common cause failures to comply with the latest methodology, and removal of conservative assumptions and simplifications.

The IPE was a single unit model and applied only to an operating unit. The 2001 update created a dual unit model including inter-unit dependencies and spanned all modes of operation (operating and shutdown). This effort included the development of Safety Monitor™ full power models based on the updated PRA and development of and inclusion of a shutdown risk model, which can be used to support assessment and management of shutdown risk.

The following specific changes were made:

A. Initiating Events

Large-break and medium-break loss of coolant accidents (LOCAs), steam generator tube ruptures, and steam line breaks were subdivided into the individual contributions from each loop and four separate initiating events were evaluated for each of these categories.

Initiators for loss of a single direct current train were added for each train separately.

The loss of offsite power initiator was divided into loss of offsite power to a single unit and loss of offsite power to both units (dual unit loss of offsite power) to improve modeling of the unit crossties.

Similarly, loss of ESW was split to consider the loss of a single unit's ESW separately from a total (dual unit) loss of ESW to improve modeling of the unit crossties.

Initiating event frequencies were reassessed based on updated plant-specific data and new generic data. In addition, a number of the frequencies were obtained from models built into the overall PRA as transfers from other initiators. The initiators included:

1. Consequential medium-break and small-break LOCAs resulting from a reactor coolant system power operated relief valve or safety relief valve failing to reclose.
2. Station blackouts.
3. Anticipated Transients Without Scram (ATWS) events.

Also, several initiating event frequencies were obtained from detailed system models:

1. Loss of ESW to a single unit.
2. Loss of ESW to both units.
3. Loss of component cooling water (CCW).
4. Loss of 250 volt direct current busses.

B. Fault Trees

The fault tree models were revised to incorporate design changes and operational changes.

Individual component common cause groups were identified for Multiple Greek Letter method common cause analysis.

The models were revised to support the implementation of Safety Monitor™.

The heat removal function was removed from the recirculation model, and this function was included in a separate long term cooling model.

Extensive changes were made in the ESW system model to properly account for interactions between units for this shared system.

The 4160 volts alternating current system model was changed to address a reconfiguration of the reserve auxiliary transformers.

C. Reliability and Unavailability Data

Revision of component failure data analysis included collecting and analyzing more recent CNP failure data for the time period since the previous update and the enhancement of common cause failure data for all components.

D. Human Reliability Analysis

Evaluation of human error probabilities were limited to those affected by changes in procedures or were new to the updated model. The principal re-evaluation involved the revised Emergency Operating Procedure for switching to cold leg recirculation.

The revised procedure for a loss of CCW was also used to update the associated human error probabilities.

The net result was to add or revise 30 human error probabilities (20% of the total human interaction events).

Results

The CDF is less than that from the 1995 update of 7.14E-05 per year. This can be attributed to a number of factors including a reduction in LOCA-initiating event frequencies, the removal of conservative assumptions, and the more detailed and complete modeling of ESW crossties between units.

The Unit 2 results are almost identical to those for Unit 1 with the differences being due to minor differences in power supply arrangements to support systems and ATWS unfavorable exposure times.

The distribution of the contributions to the results has changed from the 1995 update. The station blackout contribution is now 36% of the total CDF and is higher than the 1995 result.

Sequences related to a loss of all ESW contribute approximately 24% of the total CDF. The most significant contributors are loss of ESW either as the initiator or following a normal transient initiator with subsequent loss of ESW combined with failure to recover ESW.

Small-break LOCA is still an important contributor (17%) to CDF. The importance of small-break LOCA has decreased from the 1995 evaluation due to the reduced initiator frequency. The contribution to the total of steam generator tube ruptures has been reduced due to more detailed modeling while the contribution from steamline breaks has gone up because of an increase in assessed secondary side pipe break frequency.

The dominant contributors to LERF are loss of off site power initiated sequences that make up approximately 50% of the total. Steam generator tube ruptures, loss of ESW, and small-break LOCAs each contribute about 10% to the total LERF.

Peer Review

The 2001 update was the model provided to the Westinghouse Owners Group PRA Peer Review Team for review. The PRA peer review was performed in September 2001.

The summary of strengths and areas for improvement has been extracted from the draft report and is provided below.

Strengths

- PRA Notebooks are mostly well constructed and useful.
- Good interaction with plant personnel/functions, good input into Human Reliability Analysis.
- Use of a Risk-Informed Steering Committee.
- Broad scope PRA and Information Tools (e.g., On-Line Safety Monitor™, shutdown model, external events models).

- Strong attention to detail in the modeling and quantification process and documentation.
- Highly sophisticated single fault tree model able to be used for PRA or Safety Monitor™ quantification.

Areas for Improvement

- Better estimate success criteria analyses to remove conservatisms.
- Should re-create some of the analytical bases for IPE success criteria.
- Internal flooding analysis should be updated.
- Common cause process could be improved, and plant-specific common cause screening should be considered.
- Highly sophisticated single fault tree model able to be used for PRA or Safety Monitor™ quantification requires high degree of attention to quantification process.

All elements of the PRA Peer Review received a grade of contingent 3. An aspect of a grade 3 is that the PRA can be used in licensing submittals to the NRC to support positions concerning absolute levels of safety significance if supported by deterministic evaluations. The contingent designation indicates that the peer review team identified Facts and Observations (F&O) that either require resolution immediately (Level A) or at the next scheduled update (Level B) for an element. Once the necessary F&O(s) for an element are resolved, the grade is no longer considered to be contingent. I&M has undertaken an update to the PRA model to address all of the F&Os leading to the contingent designation.

Facts and Observations

The peer review team identified 4 Level A F&Os, the most significant, and 24 Level B F&Os. Many of the F&Os identified by the peer review team were resolved shortly after the review team completed their evaluation by providing additional information or explanation to support the analysis (i.e., no changes to the model were required). For example, to resolve many of the common cause failure (CCF) issues, CNP obtained clarification from the author of NUREG/CR-5485. The clarification provided by the author resolved the issue with no model changes required. The Level A F&Os, which are the relevant F&Os for this request, are discussed below.

There were four Level A F&Os identified by the peer review team. Of the four F&Os, an F&O concerning the emergency diesel generator was resolved shortly after the completion of the peer review by providing additional information. No model changes were required to resolve the F&O. As a result, this F&O is not discussed below. The remaining open Level A F&Os are addressed below and include a discussion as to why these F&Os do not affect the quantitative results of this submittal.

Observation AS-7

The reviewers identified some event tree transfers that did not retain the dependencies of the initiating events. However, as indicated in the F&O, this practice may not have an effect on baseline CDF. Additionally, for applications, the effects of this practice can be assessed through screening of the dominant cutsets. Safety Monitor™ accident sequence cutsets were reviewed to identify any potential dominant sequences in which the lack of initiating event dependencies would be important to subsequent sequence progression. No such cutsets were identified to sequence frequencies of 1E-08/year. Based on this review, this F&O is judged to not significantly affect the results presented herein for the ESW AOT.

Observation SY-11

The reviewers believed that passive failures, specifically a single boundary failure resulting from a leak or rupture, were not well represented in the fault trees.

PRA modifications to add passive failures to the system are in progress. Specifically, heat exchanger ruptures, system leaks and heat exchanger plugging are being added to the CCW fault trees. The ESW initiator fault trees are being modified to include system leaks and to treat common cause failure of ESW system strainers consistently for all initiators. It should be noted that a number of other fault tree changes are also being addressed. These fault tree changes include removing some simplifying conservative assumptions that were made during the IPE. The quantification results shown below reflect all of the CCW and ESW fault tree modifications being made since the changes specifically made to address F&O SY-11 are not easily quantified separately.

Preliminary results indicate that the total CDF will decrease by about 10% from the base model value of 4.85E-05 with average test and maintenance uncertainty. The contribution to CDF from the loss of CCW initiating event will increase from 4.6% to about 12.7%. The contribution from the ESW4 (Loss of ESW to both units) initiating event will decrease from 12.9% to about 2.6% and the ESW2 (Loss of ESW to one unit) initiating event contribution will decrease from 5.0% to about 2.6%.

Although some of the ESW unavailabilities will be increased in response to the F&Os, the significant reduction in total CDF and ESW initiating event contributions indicate that the next PRA model will consider the ESW pumps to be less risk significant. Therefore, with regards to the F&Os, the PRA analysis presented for extending the ESW AOT is conservatively bounding.

Observation ST-5

The reviewers identified several weaknesses in the current flooding analysis and suggested that an updated flooding analysis be performed. I&M concurs with this observation and has initiated a complete revision to the flooding analysis in an effort to accomplish that objective.

Although the influence on the PRA results from revising the flooding analysis cannot be specifically quantified at this time, the effects of extending the ESW AOT on flooding can be qualitatively assessed. Any additional influence on flooding resulting from maintenance on an ESW pump will be small due to the physical location of the ESW pumps.

Since the ESW pumps reside in neither the turbine building nor the auxiliary building, maintenance on the ESW pumps cannot cause flooding of other vital equipment. Additionally, if flooding were to occur in the vicinity of the ESW pump, within the confines of the pump room, the result would be water running into the forebay with no effect on equipment. Similarly, water spray in the pump room would also be confined within the pump room.

NRC Request 3

Provide information regarding both the avoidance of risk-significant plant configurations and risk-informed configuration management.

I&M Response

The Tier 2 recommendations of Regulatory Guide 1.177 state that the licensee should provide reasonable assurance that risk-significant plant equipment outage configurations will not occur when specific plant equipment is out of service consistent with the proposed technical specification change. The Tier 3 recommendations of Regulatory Guide 1.177 state that the licensee should develop a program that ensures that the risk impact of out-of-service equipment is appropriately evaluated prior to performing any maintenance activity.

The Tier 2 and Tier 3 recommendations of Regulatory Guide 1.177 have been addressed at CNP. CNP currently has in place a risk-informed on-line and shutdown risk management process to support the requirements of 10 CFR 50.65(a)(4). Safety Monitor™ software is currently used for on-line risk assessment (Modes 1, 2, and 3) while ORAM is currently used for shutdown risk assessment (Modes 4, 5, and 6). This risk-informed process is implemented and governed by plant procedures. These procedures assure that the risk associated with the various plant configurations planned during at-power or shutdown conditions are assessed prior to entry into these configurations and appropriately managed while the plant is in these various configurations.

NRC Request 4

Discuss External Events

I&M Response

Seismic Event

The effects of the one-time extension for each ESW pump's AOT on the results of the CNP Seismic PRA (SPRA) have been qualitatively evaluated. Although the CNP SPRA model has not been re-analyzed, the seismic modeling and results were reviewed to determine the potential impact of increasing an individual ESW pump's test and maintenance unavailability. In the base SPRA model, the ESW pumps are not significant contributors to the seismic PRA risk results on an individual basis. This is because pump failures due to seismic events are assumed to simultaneously affect both ESW pumps. For example, seismic-induced loss of ESW initiating events is caused by a catastrophic failure of the screen-house due to a seismic event. Such events are assumed to cause the ESW pumps to fail simultaneously and in an unrecoverable manner. Assuming this type of failure precludes any potential effect of an extended AOT for one ESW pump on the loss-of-ESW initiator due to seismic events. A similar bounding treatment was applied to seismic-induced failures of the ESW pumps (not due to screen-house failure). That is, ESW pump failure due to a seismic event is assumed to result in both pumps failing simultaneously. This implies, once again, that an extended ESW pump AOT does not impact sequences with ESW pump failure due to seismic events. Finally, review of all of the listed accident cutsets in the CNP SPRA reports shows that there are no random ESW pump unavailablilities in any of these cutsets. Since these accident sequence cutsets are reported to frequencies of 1E-11/year, it is reasonable to conclude that the increased maintenance unavailability associated with an extended ESW pump AOT would not have a significant impact on the reported Seismic PRA contribution of 3.17E-06/year.

Fire

The effects of the one-time extension for each ESW pump's AOT on the results of the Fire Analysis for the CNP PRA (Fire Analysis) have also been qualitatively evaluated. The present revision of the Fire Analysis was updated in February 1995 to address concerns raised by the NRC during review of CNP's submittal in response to Supplement 4 of Generic Letter 88-20, which required utilities to perform an Individual Plant Examination External Event for internal fire events. Along with the seismic analysis, the Fire Analysis results are not included in the total core damage quantification. Fires in the control room dominate the core damage frequency for internal fire events with a contribution of 1.81E-06/year. The contribution to CDF for a fire in an ESW pump room was estimated to be 1.07E-07/year.

As stated in the CNP Fire Analysis, the ESW pump rooms are not very susceptible to fires. The main reason for this is that the ESW pump rooms are essentially concrete and steel and contain minimal combustibles. Furthermore, the maintenance activities associated with upgrading each ESW pump are not expected to change that low susceptibility to a fire. Any heat producing activities such as welding or grinding are controlled at CNP by plant procedures and processes, including activity-specific controls such as welding permits. Extension of the ESW system AOT does not affect the conclusion of the evaluation in the fire analysis that a fire in the screenhouse motor control center disabling the ESW pumps is not credible and could be screened from further evaluation.

The Fire Analysis evaluation of control room cabinet fires that could cause a loss of ESW pumps determined that a single panel fire would cause the loss of both ESW pumps in a unit. Given this panel configuration, the fire analysis then estimated the frequency of loss of all ESW due to a fire in a single control room. No resulting frequencies were greater than 1E-08/year. Extension of the ESW AOT does not affect these results since both ESW pumps would fail due to the panel fire.

Flooding

Flooding was discussed in the discussion of peer review observation ST-5.

NRC Request 5

Provide information regarding the pump testing conducted by the pump's vendor.

I&M Response

The pump's vendor will conduct performance tests at their facility. The data to be taken include flow, pump discharge head, operating speed, and power input. The tests will be run at three impeller settings to determine optimum performance. Additionally, data will be taken for axial pump shaft position, axial motor shaft position, and radial pump shaft vibration.

Table 1

ESW Pump AOT Extension Results

ESW Pump Being Replaced	ICCDP[1]	ICLERP[1]
1-PP-7W being replaced		
Unit 1:	6.81E-07	3.53E-08
Unit 2:	2.54E-07	1.42E-08
1-PP-7E being replaced		
Unit 1:	8.90E-07	4.19E-08
Unit 2:	7.80E-07	4.44E-08
2-PP-7W being replaced		
Unit 1:	2.48E-07	1.37E-08
Unit 2:	6.92E-07	3.58E-08
2-PP-7E being replaced		
Unit 1:	7.59E-07	4.33E-08
Unit 2:	8.77E-07	4.09E-08
ΔCDF[2]	**Unit 1**	**Unit 2**
Based on one-year interval	2.58E-6/year	2.60E-06/year
ΔLERF[2]		
Based on one-year interval	1.34E-07/year	1.35E-07/year

Note 1: ICCDP and ICLERP are based on a duration of 140 hours

Note 2 The average change in CDF (ΔCDF) and LERF (ΔLERF) were obtained by summing the ICCDP and ICLERP respectively and dividing by the time interval assumed.

The results provided in Table 1 are based on a specific set of plant alignments and the following conditions:

1. Effect of burn-up on unfavorable exposure time is included in the evaluation for both units.
2. Solid state protection system and engineered safety features actuation system logic testing assumed for the unit with the pump being replaced.
3. Prior to entering the 140-hour AOT period, balance of plant equipment relied upon for continued operation is capable of performing its intended function.
4. No switchyard work is in progress or initiated following the entry into the 140-hour action statement.
5. At the start of the replacement activity, no severe weather is forecasted during the 140-hour the AOT period.
6. During the normal work week planning and risk evaluation, the planned plant alignment is evaluated and changed as necessary to maintain ICCDP values and ICLERP values below 1E-06 and 1E-07, respectively.
7. No biocide treatment is performed during the entire pump replacement period.
8. No manipulation of valve 12-WMO-30, circulating water intake tunnel shut-off valve, is allowed during the entire pump replacement period.