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AUTH.NAME AUTHOR AFFILIATION
FITZPATRICK,E. American Electric Power Co., Inc.
RECIP.NAME RECIPIENT AFFILIATION
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SUBJECT: Forwards response to request for addl info re power update & related changes.Request primarily dealt w/training & simulator issues.

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Indiana Michigan
Power Company
500 Circle Drive
Buchanan, MI 49107 1395

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September 2, 1997

AEP:NRC:1223J

Docket Nos.: 50-315
50-316

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

Gentlemen:

Donald C. Cook Nuclear Plant Unit 2
RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION
REGARDING POWER UPRATE AND RELATED CHANGES

This letter and its attachment constitute a response to the July 30, 1997, NRC request for additional information regarding our July 11, 1996, 5% thermal power uprate submittal AEP:NRC:1223. The request primarily dealt with training and simulator issues.

This letter is submitted pursuant to 10 CFR 50.30(b) and, as such, includes an oath statement.

Sincerely,

S. Brewer for E.E.F.

E. E. Fitzpatrick
Vice President

SWORN TO AND SUBSCRIBED BEFORE ME

THIS 2nd DAY OF Sept., 1997

Linda L. Boelcke
Notary Public

My Commission Expires 1-21-2001

LINDA L. BOELCKE
Notary Public, Berrien County, MI
My Commission Expires January 21, 2001

vlb

Attachments

c: A. A. Blind
A. B. Beach
MDEQ - DW & RPD
NRC Resident Inspector
J. R. Padgett

A001

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ATTACHMENT TO AEP:NRC:1223J

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION
REGARDING POWER UPRATE AND RELATED CHANGES

NRC Topic 1

"Discuss whether the power uprate will change the type and scope of plant emergency and abnormal procedures. Will the power uprate change the type, scope, and nature of operator actions needed for accident mitigation and will it require any new operator actions?"

Response to Topic 1

The power uprate will not result in any changes to the type of emergency and abnormal operating procedures that are being used at the plant, and will not require any new operator actions. The uprate did result in the hot leg swapover time changing from 12 hours to 8.5 hours. This is noted as a long-term recovery action and, therefore, is not considered a concern with regard to operator response.

Emergency operating procedure (EOP) setpoint values that were considered to be potentially impacted as a result of the uprate include the following:

(a) Saturation pressure corresponding to full power reactor coolant system (RCS) hot leg temperature.

(b) RCS pressure value to prevent accumulator nitrogen injection.

(c) Minimum safety injection (SI) pump flow to remove decay heat.

(d) Reactor coolant pump trip parameter.

(e) Values showing steam generator levels sufficiently low so as to cause entry into FR-H.1, "Response to Loss of Secondary Heat Sink."

(e) Values showing steam generator levels requiring initiation of bleed-and-feed in FR-H.1, "Response to Loss of Secondary Heat Sink."

Most of the EOP setpoints remain unchanged. For those that are potentially impacted; the changes will result in operator response to slightly different values than were previously considered. Final setpoints associated with the above values have not been calculated. However, an evaluation has indicated that many would remain the same as their current value, and with one exception, for those that do change, their impact on operator action will be almost transparent.

Of the above changes, the one exception is the minimum emergency core cooling system pump flow required to remove decay heat. This information is used in ECA-1.1, "Loss of Emergency Coolant Recirculation." The value changed by 5%, raising the requirement for pumped flow needed to remove decay heat for this event by a like amount. However, this event is beyond the plant design basis that assumes emergency coolant recirculation to be available when required. We consider this change not to significantly affect the ability of the operator to recover from the postulated occurrence.

In summary, although the uprate did have a minor impact, it did not have any significant impact on the type, scope, or recovery actions associated with accident mitigation.

NRC Topic No. 2

"Provide examples of operator actions potentially sensitive to power uprate and address whether the power uprate will have any effect on operator reliability or performance. Identify operator actions that would necessitate reduced response times associated with a power uprate. Please specify the expected response times before the power uprate and the reduced response times. What have simulator observations shown relative to operator response times for operator actions that are potentially sensitive to power uprate? Please state why reduced operator response times are needed. Please state whether reduced time available to the operator due to the power uprate will significantly affect the operator's ability to complete manual actions in the times required."

Response to Topic No. 2

For a Westinghouse 4-loop pressurized water reactor, examples of operator actions potentially sensitive to the power uprate would be operator actions required to respond to anticipated transient without scram (ATWS) events to reduce core power, and operator actions sensitive to the short-term restoration and maintenance of decay heat removal such as certain steam generator tube rupture actions and bleed-and-feed. For an increase in power of 5%, the effects on long-term events, such as long-term cooling actions, would be negligible due to the relatively long response times and the dependency of human error on other actions and peripheral factors. This is also because, as time available for operator response increases, the human error due to diagnosis decreases and the overall human error probability (HEP) becomes dominated by errors of omission and commission, which are not time-dependent or power-dependent errors.

To investigate the effects of the proposed power uprate on procedures and operator response time, representative simulator runs were made using the ATWS and loss of secondary heat sink events to evaluate some of the thermal-hydraulic timing parameters and the ability of the operators to respond to events with a power uprated core. Based on human error analysis, further discussed below, these events contain the fundamental operator actions most sensitive to power uprate conditions. For an ATWS, the simulator model predicted the time to reach peak pressure would decrease by 21 seconds to 1 minute, 12.7 seconds for the uprated core case. A scenario involving a loss of secondary heat removal leading to "bleed-and-feed" conditions was run for the pre- and post-uprated core. The simulator model predicted the boil-off rate for the steam generators would increase slightly for the uprated core case. The simulated time to boil from 50% level to 11% level (the bleed-and-feed initiation setpoint) decreased by about 6.5 minutes to a total time of 45.6 minutes for the uprated core case. During the simulator runs, the operator continued to be able to complete the required actions in response to normal operation, transients, or accidents during simulator training utilizing simulated power uprate conditions.

In addition to the simulator scenarios, an analysis of the uprate on the previously calculated HEPs in the probabilistic risk assessment was performed. This involved a re-evaluation of applicable HEP, including the effect of those HEPs on the core damage probability. If operator actions and the associated HEPs are affected, the total effect on core damage frequency (CDF) must also be considered. Even for cases in which the HEP increases several times, the percentage increase in CDF is much smaller than the percentage increase in any one HEP.

As previously stated, the effect of the power uprate on operator error is to lower the diagnosis time and increase the operator diagnosis error. The uprate would also affect the operator stress, dependency of operator actions, and the time available for recovery actions. The diagnosis error associated with Cook Nuclear Plant HEPs has been characterized using the cognitive error model. Within this model, there is an allowance for what is called a failure mechanism b, or $p_c b$ that calculates an error due to the "data of an event not being attended to". Part of this calculation is an error factor that characterizes the workload of the operators responding to a given event. This variable is the closest to being a time-dependent variable in our calculation of diagnostic error. In performing the assessment of the uprate's impact, these factors were adjusted to increase the workload as appropriate, thereby effectively reducing the modeled response time. Where appropriate, we have also increased the factors for stress and/or dependency on operator actions considered potentially sensitive to the power uprate in order to arrive at an increased estimate for HEP for these events. Additionally, we modeled reduced recovery time for certain actions where appropriate.

The following table provides the operator actions that are potentially sensitive to power uprate, and shows the impact of power uprate on the resulting HEPs and the corresponding change in CDF. The current mean HEPs correspond to the updated individual plant examination human reliability assessment, and the mean HEP with the power uprate reflects changes in the uprated HEPs. The operator actions considered include:

OLI	Depressurization to Allow Low Pressure Injection
OL21.22	RCS Cooldown (CCW, ESW)
OA1	Operator Actions to cooldown and Depressurize the RCS and Terminate Safety Injection Before the Ruptured S/G Fills
OA2	Operator Actions to Cooldown and Depressurize the RCS and Terminate Safety Injection After the Ruptured S/G Fills
MF1	Initiate MFW or U2 AFW Cross-tie
PBT	Primary Feed and Bleed Without SI.
MRI	Manual Rod Insertion (for ATWS)
PPR	Primary Pressure Relief (for ATWS)
LTS	Long Term Shutdown
SGI	Steam Generator Isolation
DNMV	Failure to Reopen AFW Valves to SG

HEPs were recalculated using the adjustments discussed above and yielded the following results.

Operator Action	Current Mean HEP	Mean HEP W/Uprate	Percent HEP Change
OLI	3.4 E-02	4.5 E-02	32.3%
OL21	1.7 E-03	2.5 E-03	47%
OL22	2.8 E-02	3.3 E-02	17.8%
OA1	9.2 E-03	1.0 E-02	9%
OA2	4.7 E-04	5.0 E-04	6%
MF1	1.4 E-02	1.8 E-02	28.6%
PBT	2.1 E-02	2.1 E-02	0%
MRI	3 E-04	3 E-04	0%
PPR	6.9 E-03	6.9 E-03	0%
LTS	6.0 E-02	6.0 E-02	0%
SGI	4.2 E-03	5.7 E-03	36%
DNMV	5.4 E-03	5.5 E-02	2%

The conservatisms in our previous assumptions for the operator actions PBT, MRI, PPR and LTS were such that any adjustments for uprated conditions produced no measurable change to these HEPs. These increased HEPs were all simultaneously input in Cook Nuclear Plant's probabilistic risk assessment model, and resulted in an increase of approximately 3.5% in the base CDF from 7.09 E-5 to 7.34 E-5. The CDF remains below 1 E-04 and experiences a very small annual increase. It is noted that this calculated increase in CDF is considered to be a bounding estimate of the impact of the uprate because we used conservative dependency, recovery, and stress conditions in our evaluation.

The conclusion of these evaluations is that the power uprate will not significantly affect operator actions, reliability, or performance.

NRC Topic No. 3

"Discuss any changes the power uprate will have on control room instruments, alarms, and displays. Are zone markings on meters changed (e.g., normal range, marginal range, and out-of-tolerance range)?"

Response to Topic No. 3

As part of the uprate program for unit 2, the planned value of T_{avg} at 100% reactor power is being raised from 574° F to 576° F. This increase affects the value of the high auctioneered T_{avg} alarm that is defined to be 3° F above the nominal T_{avg} at full load. Thus, the new value for the alarm will be 579° F.

The new value for the high auctioneered T_{avg} alarm affects the zone coding of the four T_{avg} indicating meters (one for each reactor

coolant loop) 2-NTI-13, 23, 33, and 43. Yellow zone coding will begin at 579° F on the meter scale and continue to the upper end of the scale (630° F).

As part of the uprate program, the range of the steam flow and feedwater flow instrument loops is being increased from 4 million #/hr. to 4.5 million #/hr. This increase requires replacing scales on eight steam flow indicating meters and eight feedwater flow indicating meters on the main control panel, and also replacing scales on four feedwater flow indicating meters on the unit 2 hot shutdown panel.

In addition to the above scale changes for panel meters, scales for steam flow and feedwater flow on four chart recorders (one for each steam/feedwater flow loop) will be replaced.

As part of the uprate program, the control range for the feedwater pump speed control is increasing from 45 - 173 psig to 45 - 189 psig. This change will require replacing the scale on chart recorder 2-MR-24, and will require new chart paper.

As part of the uprate program, the low steam line pressure alarm/low steam line pressure - steam line isolation setpoints are being lowered from 650/600 psig to 550/500 psig. This change will affect the zone coding on twelve steam line pressure indicators (three per steam loop) on the main control panel. Yellow zone coding will begin at 550 psig and continue to 500 psig. Red zone coding will begin at 500 psig and will continue to 0 psig.

The setpoint for low pressurizer pressure - SI is being lowered from 1900 psig to 1815 psig. This change does not affect any control room displays, zone coding, etc.

<u>NRC Topic No. 4</u>

"Discuss any changes the power uprate will have on the Safety Parameter Display System (SPDS)."

<u>Response to Topic No. 4</u>

We have determined that there will be no major changes to the SPDS displays on the plant process computer (PPC). Currently, the operations group is analyzing a few minor setpoint changes that will have to be made. During the outage, these setpoints will be changed and tested on the PPC. As the new setpoints become available, they will be incorporated and tested on the simulator PPC.

<u>NRC Topic No. 5</u>

"Describe any changes the power uprate will have on the operator training program and the plant simulator. Provide a copy of the post-modification test report (or test abstracts) to document and support the effectiveness of simulator changes as required by ANSI/ANS 3.5-1985, Section 5.4.1.

Specifically, please propose a license condition and/or commitments that address the following:

(a) Provide classroom and simulator training on the power uprate modification.

(b) Complete simulator changes that are consistent with ANSI/ANS 3.5-1985. Simulator fidelity will be re-validated in accordance with ANSI/ANS 3.5-1985, Section 5.4.1, "Simulator Performance Testing." Simulator revalidation will include comparison of individual simulated systems and components and simulated integrated plant steady state and transient performance with reference plant responses using similar startup test procedures.

(c) Complete control room and plant process computer system changes as a result of the power uprate.

(d) Modify training and plant simulator relative to issues and discrepancies identified during the startup testing program."

Response to Topic No. 5

We propose the following commitments:

(a) We will conduct neutral/positive moderator temperature coefficient (MTC) training on the simulator to support the unit 2 refueling utilizing a simulator core load based on the uprated design. The training will be supplemented by reactor theory classroom training at a scope determined by a training needs assessment. The training will be attended by licensed operators and shift technical advisors. This training will be completed prior to attaining mode 2 on unit 2 cycle 12.

The training needs analysis for the classroom portion of the training was completed on March 31, 1997. The 1996 generic fundamentals examination standards exam was administered to three of the five shifts of licensed operators to serve as a basis for this needs analysis. The results of the exams were then quantified and presented to the operations superintendent for approval on March 31, 1997. The objectives for the reactor theory portion of the analysis were approved at this meeting.

Training related to this commitment commenced on July 14, 1997. It includes two phases. The phase currently being conducted focuses on application of positive ITC/MTC conditions, power distribution, and reactivity control. This training includes classroom fundamentals and simulator, and is planned to be completed for licensed operators and shift technical advisors on September 5, 1997. This training utilizes a new simulator core model based on unit 2 cycle 12.

The second phase of the training is planned to occur during the unit 2 refueling pre-outage/pre-startup training. This training is scheduled to commence on September 22, 1997, and be completed on September 26, 1997. The licensed operators and shift technical advisors are scheduled to attend this training. This training will be a refresher session on the simulator that will focus on low power (<30%) operation with a positive MTC. Included will be competency assessments of the operating crews relative to operating with the elevated positive MTC expected with the uprated core. This training will also include low power physics testing on the simulator with a positive MTC, and a discussion of the cycle 12 core

physics characteristics. Nuclear engineering will participate in this portion of the training.

We will conduct training on significant control room hardware changes and plant system changes resulting from the uprated design. The scope and setting for this training will be determined utilizing a systematic approach to training methodology. This training (as needed) will be completed prior to reaching mode 2 on unit 2 cycle 12.

(b) The unit 2 reference plant simulator will be updated in accordance with ANSI/ANS 3.5-1985, section 5.2, "Simulator Update Design Data."

Reference plant uprate design and process control changes incorporated on the simulator are verified at the module level and validated at the system level. These changes are installed and tested in accordance with training administrative manual 6.05. This includes the installation of the unit 2 cycle 12 high energy core.

The unit 2 cycle 12 core at 3 GWD/MTU (maximum positive ITC/MTC) installation and testing were completed to support positive ITC/MTC conditions, power distribution, and reactivity control training that commenced on July 14, 1997. The 0 GWD/MTU and 8 GWD/MTU core stages will be installed and tested to support pre-outage/pre-startup training by September 22, 1997. Other core stages required per ANSI/ANS 3.5-1985, section 3.4.1, "Initial Conditions", will be installed and tested as required to support the training program with an expected completion date no later than December 1, 1997.

System specific uprate design and process control changes will be incorporated and tested to support pre-outage/pre-startup training by September 22, 1997.

Specific testing in accordance with ANSI/ANS 3.5-1985, Section 5.4.1., "Simulator Testing", is as follows.

Steady state testing will be completed at four separate power levels with results compared to predictive heat balance data for the reference plant uprate. In addition, the full power stability test will be performed. This testing will be completed by September 22, 1997.

Transient tests identified in ANSI/ANS 3.5-1985, appendix B, section B.2.2, "Transient Performance", will be completed and reviewed by a team of subject matter experts using the criteria established in section 4.2.1 of the same standard.

Malfunctions that have been impacted by system changes for the uprated conditions will be tested. This will be in conjunction with the routine 25% per year malfunction test schedule.

Normal plant evolutions will be tested in accordance with section 3.1.1 of ANSI/ANS 3.5-1985, with the exception of operator surveillance testing on safety

related equipment or systems that have not been impacted by simulator uprate changes. These tests will be performed using approved plant procedures. This testing will be completed by September 22, 1997.

(c) The unit 2 reference plant simulator control room hardware will be updated to reflect the uprated plant configuration. This will include meter scale changes and zone coding. These changes will be completed as required to support the training program.

Currently, all PPC uprate changes are being installed on the simulator PPC. This will allow better testing of changes and help identify any additional changes that may be required. During the outage, all changes will be implemented and tested on the PPC.

The changes to the PPC include:

1) Rescaling of the steam flow and feed flow analog input points.
2) Increasing full power reactor thermal output from 3411 to 3588.
3) Changing steam pressure low alarms.
4) Change T_{ave} from 573.8 to 576.
5) Changing delta temperature conversion factors.
6) Changing 150% delta temperature values.
7) Changes to safety parameter display system as identified by operations group.
8) Changing pressurizer pressure low alarms.
9) Changing steam flow and feed flow mismatch alarms.

(d) Training resulting from needs identified during startup will be determined and scheduled utilizing a systematic approach to training methodology.

Steady state simulator tests, performed in accordance with section 4.1 of ANSI/ANS 3.5-1985, will be compared to actual plant data obtained during the power escalation and power operation of the uprated plant configuration. Any discrepancies identified will be corrected and incorporated into the reference simulator. Additionally, the obtained data will be utilized as the new baseline data for steady state tests performed in accordance with section 4.1 of the standard.