Repro. Unit

CF

INDIANA & MICHIGAN POWER COMPANY

P. O. BOX 18
BOWLING GREEN STATION
NEW YORK, N. Y. 10004

July 25, 1979
AEP:NRC:00185B

Donald C. Cook Nuclear Plant Units 1 & 2
Docket Nos. 50-315 and 50-316
License Nos. DPR-58 and DPR-74

Mr. James G. Keppler, Regional Director
U. S. Nuclear Regulatory Commission
Office of Inspection and Enforcement
Region III
799 Roosevelt Road
Glen Ellyn, Illinois 60137

Dear Mr. Keppler:

This letter supplements our letter of May 1, 1979 (AEP:NRC:00185) in response to IE Bulletin 79-06A, Revision 1. On June 20, 1979 we received, by telecopy, a request for additional information as a result of the NRC Staff review of our May 1, 1979 response. The attachment to this letter contains our responses to the requests resulting from your evaluation of our May 1, 1979 submittal. In a phone conversation held Friday, June 22, 1979 with members of the NRC Staff in Bethesda, Md., we were requested to supply responses to some parts of Action Items 4, 7, 8, and 10 before Unit 2 would be allowed to return to power. We provided you with our responses to those action items in our June 25, 1979 letter. These responses are included again, with minor corrections, in the attachment to this letter.

Please note that we are members of a Westinghouse Owner's Group on the Three Mile Island Accident formed at the recommendation of the Commission. Active discussions are now taking place between the Group, Westinghouse, and the NRC, where some of the issues contained in your requests are being evaluated. We therefore reserve the right to modify our responses, actions or commitments once a final position is reached.

Very truly yours,

John E. Dolan

John E. Dolan
Vice President

JED/emc
Attachment

Sworn and subscribed to before me
this 25th day of July, 1979 in
New York County, New York

Gregory M. Gurican

Notary Public

GREGORY M. GURICAN
Notary Public, State of New York
No. 31-4643431
Qualified in New York County
Commission Expires March 30, 1981

A02
60

cc: R. C. Callen
G. Charnoff
R. W. Jurgensen
R. S. Hunter
D. V. Shaller - Bridgman
S. A. Varga - NRC
N. C. Moseley - NRC

ATTACHMENT

ACTION ITEM 2:

Listed below is instrumentation which could be used, under certain circumstances, to recognize void formation in the Reactor Coolant System (RCS):

A) Pressurizer Pressure

B) Core Outlet Temperature
   1) incore thermocouples
   2) wide range hot leg temperatures

C) Pressurizer Level

D) Nuclear Instrumentation

Pressurizer pressure and core outlet temperature would primarily be used to verify subcooled liquid in the RCS. Pressurizer level would be used to verify adequate coolant inventory. Large sudden changes in these parameters might provide indication of potential or actual voiding in the RCS. Data has been collected from the in-core neutron flux system corresponding to many reactor conditions. This data could be used as a basis of comparison with the signals obtained from a core where local bubbles or voids existed.

Operators are instructed to keep system pressure greater than the saturation pressure corresponding to the highest indicated temperature under forced and natural circulation modes. This can be accomplished by increasing pressure or decreasing temperature. One method of control will be by increasing pressure with the pressurizer heaters, if available, or system cooldown by steam release from the Steam Generators.

If a voiding condition develops anyway, there are modes of operation which can be used to remove the void. Our response to Action Item 12, concerning hydrogen removal from the RCS describes these modes of operation and the circumstances under which they would be employed.

Part of the Westinghouse Owners' Group scope includes preparation of generic emergency instructions for natural circulation. These instructions are expected to be available by September 1, 1979. Subsequent to their receipt we will write and issue a new operating procedure whose purpose will be to identify and maintain natural circulation consistent with the Group's generic instructions. In the interim we have put together a review package for all licensed personnel consisting of the following information.

(1) Chronological sequence of the TMI-2 accident.

(2) IE Bulletin 79-06 "Review of operating errors and system malfunctions identified during the Three Mile Island Incident".

(3) Westinghouse's article entitled "Steady State Natural Circulation, Calculation and Verification."

(4) A set of operating charts with written explanation of a Cook Unit 1 trip from 100% power on March 23, 1979. The Unit scram was caused by the tripping of two vital instrument buses. The reactor was in a stable condition on natural circulation for 1 hour and 52 minutes at which time a reactor coolant pump was restarted.

This information has been made a required review for all licensed personnel. Group review sessions have been held in which open discussion was encouraged and received. In this review, natural circulation was discussed in detail and it was pointed out that the major difference between forced and natural circulation is that the Reactor Coolant Pumps are not in operation while on natural circulation. In both cases the system pressure must be maintained higher than hot leg saturation. The pressurizer level must be maintained on scale to indicate adequate RCS water inventory. The steam generator level must be maintained to assure adequate secondary side water inventory for heat removal. No further procedure revisions in this regard are required at this time, as the above key points are adequately covered.

ACTION ITEM 3:

In our May 1, 1979 submittal (AEP:NRC:00185),in response to Item 3 we indicated that,

> "All operations personnel were instructed to manually initiate safety injection during normal operation when the pressurizer pressure indication reaches the actuation setpoint, regardless of pressurizer level, on April 17, 1979".

Our instructions indicated that with an uncontrolled pressure reduction down to the safety injection actuation setpoint, safety injection should be manually initiated, regardless of the water level in the pressurizer. As such, safety injection would be manually initiated by the operator at the setpoint for pressurizer pressure-low. Excerpts from the instructions of April 7, 1979 are as follows:

"On Unit 1 with an uncontrolled reactor coolant system pressure reduction, when the pressure reaches 1800 psig manually initiate safety injection, regardless of pressurizer level."

"On Unit 2 with an uncontrolled reactor coolant system pressure reduction, when the pressure reaches 1900 psig manually initiate safety injection , regardless of pressurizer level."

Since our May 1, 1979 response to Item 3, we have modified the automatic safety injection actuation logic to a 2 out of 3 channel logic on low pressure only. Our submittal of June 6, 1979 (AEP:NRC:00185A) proposed changes to our Technical Specifications and described the new actuation logic. On June 21, 1979 the NRC issued Amendment Nos. 29 and 11 to the operating licenses for Units 1 and 2 and the related safety evaluation report on this matter.

ACTION ITEM 4:

In our May 1, 1979 response to Item 4 we indicated that we were reviewing containment isolation in light of the TMI-2 Incident. This review is completed with the results provided below which supplement our May 1, 1979 submittal.

Our responses to Action Items 7 and 12 indicate that it would be desirable, should an unsafe plant condition develop, to be able to operate certain equipment inside the containment that becomes isolated by a Phase A and or a Phase B signal (automatic safety actuation). Resetting or overriding of the Phase A and/or Phase B containment isolation signals would be required to place this equipment into service. For example, control air inside containment to the Power Operated Relief Valves (PORV) is isolated under Phase A.

The Reactor Coolant Pumps (RCP) can be operated under a Phase A containment isolation. The RCP seal water discharge is isolated by Phase A containment isolation. However the seal water supply line is not isolated and the charging pumps can continue to supply seal water to the RCP's under a Phase A containment isolation. A safety valve set at 150 psi, discharges to the pressurizer relief tank to prevent the supply lines from overpressurizing. Operator action would be required to reset Phase A containment isolation and place the seal water return line back into service by operation of each isolation valve control switch for valves QCM-250 and QCM-350. Phase A containment isolation will not be reset unless an unsafe plant condition develops in accordance with our response to Action Item 7.

Similarly, Component Cooling Water (CCW) to the RCP's is isolated under Phase B. Operation without CCW will result in damage to the pumps. Thus the RCP's should be manually tripped when Phase B is reached. Component Cooling Water to and from the RCP oil coolers and thermal barrier is isolated under Phase B containment isolation. Non-essential Service Water (NESW) to and from the RCP Motor Air Coolers and Air Cooler Vents are isolated under a Phase B containment isolation.

In summary, to operate a RCP under Phase B isolation would require resetting Phase B isolation, operating each CCW isolation valve control switch to place CCW into service and then restarting the RCP. It should also be noted that when containment pressure high-high at 3 psig is reached and Phase B isolation is initiated, the containment sprays will also actuate. Operation of the RCP's in a spray environment is uncertain.

Our letter of June 8, 1979 (AEP:NRC:00114A) described actions we have taken regarding the matter of overriding safety actuation signals by use of the reset feature mentioned above. As it applies here a thorough review of the need to use the reset feature is required prior to the Senior Reactor Operator granting approval to reset Phase A or Phase B containment isolation signals. When the reset is used, an alarm will annunciate in the control room. An evaluation of the status of the safety actuation signal inputs will be performed prior to de-energizing this alarm. If an unsafe plant condition develops or becomes aggravated because of placing into operation a RCP or a PORV under Phase A and/or Phase B containment isolation conditions, then the RCP or PORV will be taken out of service and Phase A and/or Phase B will be manually initiated.

The Cook Plant isolation system is designed to limit the leakage of radioactive materials through fluid lines penetrating the containment building. Lines for which isolation is required are designed such that no single failure can prevent isolation. The design criteria for containment isolation is such that it is not reset by the elimination or resetting of the initiating signals, for example by resetting Safety Injection. Containment Isolation can only be reset by manual controls on the main control board as described earlier. Control features are provided for the containment isolation valves such that:

a) The valves will remain in the closed position if the Phase A and or Phase B signal is reset.

b) The containment isolation signals override all other automatic control signals.

c) Each valve can be opened or closed normally after the Phase A and/or Phase B containment isolation signals are reset.

Thus the current design provides for containment isolation and includes provisions such that containment isolation is not degraded by reset of initiating signals. Those lines that provide needed safety features or core cooling capability are not isolated. Should an unsafe plant condition develop requiring the use of some of the lines that are isolated, design features and administrative controls are provided to allow these lines to be placed into service.

The lines automatically isolated by Phase A and Phase B containment isolation are listed in Table 3.6-1 of our Technical Specifications for both Unit 1 and Unit 2. Phase A containment isolation and main feedwater isolation are automatically initiated by a safety injection signal derived from the Reactor Protection System or Containment Pressure - High at 1.2 psi. Phase B containment isolation, main steam isolation and containment spray are automatically initiated by Containment Pressure High-High at 3 psi.

Listed below are all lines penetrating the containment that do not automatically isolate on Phase A containment isolation signal. Similarly, lines penetrating the containment that do not automatically isolate on a Phase B containment isolation are listed below. Some of these lines are not in service or aligned with an operable flow path during power operation and are listed separately.

LINES NORMALLY IN SERVICE OR ALIGNED WITH AN OPERABLE FLOW PATH THAT DO NOT BECOME ISOLATED ON A PHASE "A" SIGNAL OR MAIN FEEDWATER ISOLATION SIGNAL

1. Reactor Coolant Pump Seal Supply
2. Upper Containment Spray Inlet
3. Lower Containment Spray Inlet
4. RHR to Containment Spray
5. RHR Cooldown Suction
6. RHR to RC Hot Legs (LHSI)
7. RHR from Recirculation Sump
8. Safety Injection - (High Head and Intermediate Head)
9. Boron Injection Line
10. Auxiliary Feed Water*
11. Steam Generator Chemical Feed
12. Weld Channel Pressurization Air
13. CCW to Main Steam Penetrations
14. CCW from Main Steam Penetrations
15. CCW to Pressure Equalizing Fans
16. CCW from Pressure Equalizing Fans
17. Air Particulate- Radiogas Sample Return
18. Containment Pressure Transmitters
19. CCW from Reactor Coolant Pump Oil Coolers
20. CCW from Reactor Coolant Pump Thermal Barriers
21. CCW to RCP Oil Coolers and Thermal Barriers
22. Sample to Air Particulate - Radiogas Detector
23. Non-Essential Service Water from Containment Coolers
24. Non-Essential Service Water from Containment Coolers
25. Non-Essential Service Water to Instrument Room Cooler
26. Non-Essential Service Water from Instrument Room Cooler
27. Non-Essential Service Water to RCP Motor Air Coolers
28. Non-Essential Service Water from RCP Motor Air Coolers
29. Main Steam from Steam Generators.

* Headered together with main feedwater downstream of the main feedwater isolation valve.

LINES NORMALLY IN SERVICE OR ALIGNED WITH AN OPERABLE FLOW PATH THAT DO NOT BECOME ISOLATED ON A PHASE 'B' OR MAIN STEAM ISOLATION SIGNAL (OR PHASE 'A' SIGNAL)

1. Reactor Coolant Pump Seal Supply
2. Upper Containment Spray Inlet
3. Lower Containment Spray Inlet
4. RHR to Containment Spray
5. RHR Cooldown Suction
6. RHR to RC Hot Legs (LHSI)
7. RHR from Recirculation Sump
8. Safety Injection - (High Head and Intermediate Head)
9. Boron Injection Line
10. Auxiliary Feed Water
11. Steam Generator Chemical Feed
12. Weld Channel Pressurization Air
13. CCW to Main Steam Penetrations
14. CCW from Main Steam Penetrations
15. CCW to Pressure Equalizing Fans
16. CCW from Pressure Equalizing Fans
17. Containment Pressure Transmitters

LINES NOT NORMALLY IN SERVICE WHICH DO NOT RECEIVE PHASE A AND B SIGNALS

LINE	MEANS OF ISOLATION
1. S.I. And Accumulator Test Line	2 Manual Valves - Locked Closed
2. Fuel Transfer Tube	Blind Flange
3. Service Air to Containment	Manual Valve-Locked Closed and Blind Flange
4. Ice Loading Supply Line	Blind Flange
5. Containment Pressurization Test Line	Blind Flange
6. Ice Loading Return Line	Blind Flange
7. Refueling Water Supply	2 Manual Valves - Locked Closed
8. Demineralized Water Supply	2 Manual Valves - Locked Closed
9. Refueling Cavity Drain	2 Manual Valves - Locked Closed
10.Dead Weight Test Connection	Manual Valve - Closed
11.Lower Containment Radiation Sample	2 Manual Valves - Closed
12.Upper Containment Radiation Sample	2 Manual Valves - Closed
13.Instrument Room Radiation Sample	2 Manual Valves - Closed
14.Incore Flux Detection System Access	Blind Flange

ACTION ITEM 7

Operating procedures and training instructions have been reviewed and in no case is the operator instructed to reset or override any automatic ESF actuation signal. The procedures have been changed to include the statement "unless it would result in unsafe plant conditions". This requirement does not apply to spurious or inadvertent actuation when the cause is known.

After 50°F of sub-cooling has been achieved, termination of Safety Injection operation prior to 20 minutes is only permissible if it has been determined that continued operation would result in an unsafe plant condition. This requirement does not apply to spurious or inadvertent actuation when the cause is known.

The following procedural steps exist requiring that a minimum of 2 Reactor Coolant Pumps remain in operation as long as the pump is providing forced flow and continued operation shall not result in any unsafe plant condition.

PROCEDURE STEPS	KEY POINTS
4.2.7 Maintain a minimum of 2 Reactor Coolant pumps running for primary coolant circulation.	1. It is essential that a minimum of two R.C.P.'s be maintained in service in order to provide forced cooling of the Reactor core. 2. Pump operation must be continued even though the normal minimum operating requirements may not be met. 3. If the continued operation of the R.C.P.'s threatens to cause an increase in the severity of the accident condition, the pump or pumps should be removed from service. 4. If component cooling water flow to the R.C.P.'s is lost bearing failures will occur very rapidly and therefore the pump or pumps must be removed from service. It is estimated that bearing failure may occur within 5 minute of loss of component cooling water.

ACTION ITEM 8

A complete valve lineup walk around of all safety related valves including locked valves has been performed prior to the startup following the May 19, 1979 Unit 2 outage and following the April 6th outage of Unit 1.

ACTION ITEM 10

There is an administrative requirement which has been in effect over one year, since July 7, 1978, which instructs the Shift Supervisor, at the start of each shift, to place in his log all Technical Specification items that are inoperable. The requirement also includes instructions to log any equipment that becomes inoperable during the shift and also any equipment that is returned to operable status during the shift. These logs must be reviewed by the incoming Shift Supervisor and verified by signoff. In this manner, the status of safety related systems is known at a shift change.

ACTION ITEM 12

Our response to Item 12, contained in our May 1, 1979 letter responding to IE Bulletin 79-06A, addressed hydrogen control in the containment. This additional response to Item 12 addresses hydrogen control in the RCS and supplements our May 1, 1979 submittal.

The engineered safeguards at Cook Plant are designed to meet the limits of 10 CFR 50.46 which require that the hydrogen generation from clad water reaction in a LOCA be limited to less than 1% of the clad metal, and no where exceed 17% of the clad thickness.

Hydrogen removal from the Reactor Coolant System (RCS) can be accomplished by operation of RCS letdown, Reactor Coolant Pumps, power operated relief valves and/or normal pressurizer pressure control. Systems required to support these functions such as offsite power, component cooling water, compressed air and pressurizer heaters and sprays may not be available inside of the containment. Their use is dependent upon the circumstances surrounding an event where their use may be desirable. These systems are not essential for a safe shutdown of the plant during a design basis accident and are isolated on Containment Isolation (Phase A or Phase B). Our supplemental response to Action Item 4 provides further information on containment isolation.

Containment Isolation Phase A is automatically initiated by a Safety Injection signal derived from the Reactor Protection System or Containment Pressure High at 1.2 psi. Containment Isolation Phase B is automatically initiated from Containment Pressure - High-High at 3.0 psi. If the circumstances surrounding an event are such that the use of the above mentioned non-essential systems inside of the containment is desirable and the containment is isolated then deliberate operator action is required. After careful review of the need to break containment isolation the operator would have to reset and override the Containment Isolation signal (Phase A or B) and manually place the desired system or component into service. Also, operating these systems during a station blackout would require operator action. The modes for hydrogen removal described below can be utilized from the main control room.

The modes for removing hydrogen from the reactor coolant system are:

1. Hydrogen can be stripped to the pressurizer vapor space by pressurizer spray operation if loop number 3 reactor coolant pump is operating and control air inside containment is available.

ACTION ITEM 12 (CONT'D.)

2. Hydrogen in the pressurizer vapor space can be vented and discharged to the pressurizer relief tank by operating the power operated relief valves if control air inside containment is available.

3. Hydrogen can be removed from the reactor coolant system by the letdown line and stripped in the volume control tank where it enters the waste gas system (if the letdown line is available).

4. In the event of a LOCA, hydrogen would vent with the steam to the containment.

If for some reason a non-condensible gas bubble becomes situated somewhere in the primary coolant system, there are several options for continued core cooling and removing the bubble.

If the gas bubble becomes located in the upper head, the following methods of core cooling are unaffected. Steam generators can be used to remove decay heat using reactor coolant pump forced flow (if RCP is available) or natural circulation. The safety injection system can be used to cool the core while venting through the pressurizer power operated relief valve if this method is available. Core cooling by any of these methods can proceed indefinitely if the primary coolant pressure is held constant. If the bubble were to grow to the top of the hot leg, it would slide across the hot leg and up into the steam generators. As depressurization continues, the gas bubbles grow in the steam generators and upper head but the core remains covered and cooled by safety injection water. If there is enough gas, the pressurizer surge line would eventually be "uncovered". Some of the gas would burp into the pressurizer and out the valve. This burping process would continue until the system were at the desired pressure. At that time, the current cooling mode could be continued or the system could be placed in the RHR cooling mode if at the appropriate pressure for RHR operation.

Note that a gas bubble cannot be located in the steam generator with the reactor coolant pumps running. If a gas bubble forms in the the steam generator during natural circulation, the reactor coolant pumps could be turned back on for degassing (if available) or safety injection flow could be initiated with the power operated relief valve open (if control air is available).