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AUTH.NAME AUTHOR AFFILIATION
HUNTER,R.S. Indiana & Michigan Power Co.
RECIP.NAME RECIPIENT AFFILIATION
DENTON,H.R. Office of Nuclear Reactor Regulation

SUBJECT: Forwards info re potential unreviewed safety question on interaction between nonsafety grade sys & safety grade sys in response to NRC 790917 ltr & IE Info Notice 79-22.

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INDIANA & MICHIGAN POWER COMPANY

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

October 5, 1979
AEP:NRC:00285

Donald C. Cook Nuclear Plant Units 1 and 2
Docket Nos. 50-315 and 50-316
License Nos. DPR-58 and DPR-74
Reference: IE Information Notice 79-22

Mr. Harold R. Denton, Director
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Mr. Denton:

This letter and its attachment respond to your letter of September 17, 1979 which dealt with the existence of a potential unreviewed safety question because of interactions between non-safety grade systems and safety grade systems.

We began a review of this concern as it applied to the Cook Plant just as soon as Westinghouse notified us at a meeting held on September 6, 1979 of potential interactions between non-safety grade control systems and the protective functions performed by safety grade systems. Such a review was necessary because the Westinghouse postulated scenarios are too conservative and do not consider the positve aspects of Cook Plant specific design features.

Our review has specifically addressed the non-safety grade systems listed in IE Information Notice 79-22. For those systems we have not identified any such interaction that could constitute a substantial safety hazard. Our review and findings are contained in the Attachment to this letter.

NUREG-0578 requirements for analyses of potential safety problems envision the type of scenarios identified by Westinghouse and made the subject of IE Information Notice 79-22 (1). This fact makes the scope of IE Information Notice 79-22 consistent with the requirements of NUREG-0578. As part of preparing our response to NUREG-0578 we are investigating the potential for similar interactions involving control systems employed at Cook Plant that are not a part of the Westinghouse investigation.

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Based on our investigation and findings, as well as those of industry - wide studies, we believe that no modification, suspension or revocation our operating license is necessary.

Very truly yours,

R. S. Hunter
Vice President

RSH:em

cc: R. C. Callen
G. Charnoff
R. S. Hunter
R. W. Jurgensen
D. V. Shaller -Bridgman

(1) See Section 3.2, page 17, and page A-45 of NUREG-0578.

STATE OF NEW YORK)
) ss.
COUNTY OF NEW YORK)

R., S. Hunter, being duly sworn, deposes and says that he is the Vice President of Indiana & Michigan Power Company; that he has read the foregoing statements and knows the contents thereof; and that said contents are true to the best of his knowledge and belief.

R S Hunter

Subscribed and sworn to before me this 5th day of October , 1979.

Kathleen Barry
Notary Public

KATHLEEN BARRY
NOTARY PUBLIC, State of New York
No. 41-4606792
Qualified in Queens County
Certificate filed in New York County
Commission expires March 30, 1981

ATTACHMENT TO AEP:NRC:00285

RESPONSE TO IE INFORMATION NOTICE 79-22

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

On September 18, 1979 Westinghouse presented to the NRC Staff a summary of an investigation which led to the identification of four potential scenarios where the effect on control systems of adverse environments resulting from high energy line breaks could lead to consequences more limiting than those implied by the results presented in the FSAR. Table 1 summarizes the scope of Westinghouse's investigation.

The seven control systems selected for the investigation include control systems directly addressed in the Westinghouse functional requirements. The seven events considered included postulated High Energy Line Break (HELB) environments at various locations and for a range of break sizes. Of the forty-nine possible combinations, fifteen scenarios, denoted by an X in Table 1, were identified as having interactions which merited further analysis. The fifteen scenarios are bounded by consideration of the four potential interactions discussed in IE Information Notice 79-22.

Implicit in the four potential interactions identified by Westinghouse are "worst case" assumptions concerning the break size and location, and the type and extent of consequential failures in control systems induced by the adverse environment. The scenarios postulated by Westinghouse did not account for plant specific design features and, therefore, could not be directly applied to Cook Plant.

Cook Plant specific design characteristics relevant to the four scenarios are:

1. System layout with respect to the postulated HELB.

2. Plant modifications made as a result of the HELB study and presented in Appendix O to the FSAR.

3. Type and qualification of components utilized in the corresponding control systems.

4. Safety grade protective functions which were assumed not to occur in the accident analysis as a conservative measure or for calculational convenience.

With respect to Items 1 and 2 above, the Cook Plant HELB outside containment study considered those effects arising from compartment differential pressures, pipe whip, jet impingement, and exposure to adverse environments which could impact systems required for safe shutdown of the Plant. Control systems, such as the steam generator PORV's, were included in the previous HELB study. Where ever it had been determined that unacceptable damage might occur from a HELB, appropriate plant modifications were made. Depending on the extent of the exposure to the HELB, protective shields or deflectors were added and suitable sealing against environmental conditions was provided.

Pressures and temperatures were calculated for compartments where HELB's had been postulated. The portions of Appendix O which are relevant to the review required by IE Information Notice 79-22 are: Sections 0.7 and 0.11, Tables 0-1, 0-19 and 0-20 and Figures 0-22, 0-26 and 0-27.

Instead of performing a plant specific analysis to address each of the potential interactions involving a feedwater line rupture, Westinghouse has referred to bounding accident analyses that have been submitted to the NRC in WCAP-9600, "Report on Small Break Accidents for Westinghouse NSSS Systems". Section 4.2 of the WCAP provides transient results following a total loss of main and auxiliary feedwater. Westinghouse has performed calculations that, when added to the analysis contained in WCAP-9600, demonstrate that the consequences following the control system interaction scenarios for the steam generator PORV control system, main feedwater control system and pressurizer PORV control system are in fact bounded by the present requirements for operator action.

Presented below is the evaluation for the Cook Plant of the scenarios involving the control systems identified by Westinghouse. Based on previous work, these evaluations and engineering judgement, we believe that the Cook Plant could withstand the failure of any high - energy line inside or outside containment and retain the capability to be safely placed in a cold shutdown condition.

1. MAIN STEAM POWER-OPERATED RELIEF VALVE CONTROL SYSTEM

Summary of Postulated Scenario

Westinghouse has hypothesized an event scenario which is initiated by the non-mechanistic rupture of a main feedwater line outside of reactor containment. Main feedwater discharge at the break location coupled with reverse blowdown from the affected steam generator results in an adverse environment in the vicinity of the break. The adverse environment is assumed in the Westinghouse scenario to result in the failure of the main steam power-operated relief valve control system. Failure of the PORVs in the open position would result in the eventual depressurization of the affected steam lines and could ultimately result in loss of steam supply for the turbine driven auxiliary feedwater pump (TDAFP).

In order for a feedwater line rupture to potentially lead to the total loss of feedwater, the following events must occur concurrently:

(1) The non-mechanistic rupture of the main feedwater line must occur outside of containment between the containment penetration and the containment isolation valve. Rupture of a feedwater line at this location results in loss of auxiliary feedwater capability to the affected loop and allows the reverse blowdown of secondary fluid from the affected steam generator through the break. Hypothetical feedwater line breaks upstream of the containment isolation valve would not result in loss of auxiliary feedwater to the affected steam generator. The NSSS response to postulated feedwater line breaks upstream of the check valves is similar to that of a "Loss of Feedwater" Transient. Conservative analyses of the "Loss of Feedwater" Transient have been performed for both Units of the Donald C. Cook Plant. These analyses are shown in Sections 14.1.9 (Unit No. 1) and 14.1.9 "Yellow Pages" (Unit No. 2) of the FSAR.

(2) The feedwater line rupture would have to occur on one of the two secondary loops which supply steam to the TDAFP. Postulated ruptures on either of the other two secondary loops could not result in the loss of steam supply to the TDAFP at the Cook Plant.

(3) As reported in WCAP-9226 Revision 1, "Reactor Core Response to Excessive Secondary Side Steam Releases," if the highest worth control rod assembly is <u>not</u> assumed to be stuck out of the core, the core response to an excessive secondary side release is insignificant from a reactivity point of view.

(4) The failure to start one of the Motor Driven Auxiliary Feedwater Pumps (MDAFP), must occur in the train simultaneously supplying cooling water to two out of the three intact steam generators.

If the single failure is taken on the MDAFP train supplying water to the broken secondary loop, then the remaining MDAFP will supply adequate flow to two intact steam generators to assure an orderly, safe shut down of the reactor. This postulated scenario, single failure on the MDAFP train feeding the break, is similar to the "Loss of Feedwater" Transient referenced above.

(5) The assumption made in the Westinghouse scenario that an adverse environment would result in the opening, and subsequent failure to reseat, of a Main Steam PORV is highly improbable for the Cook Plant as presented below.

Thus it can be seen that the Westinghouse scenario represents the combination of a series of very low probability events (items (1) through (5) above) coupled with an initiating design basis event, Feedwater Line Break, which is in and of itself a low probability event. Overall the Westinghouse scenario seems to be a highly unlikely event.

<u>Cook Plant Status:</u>

The effects of a high energy line break (HELB), and the resulting adverse environment, on the main steam PORV control system is not an unreviewed safety question for the Cook Plant. A design study was performed in 1973 to review potential interactions due to HELBs outside containment. This study evaluated the potential effects of an adverse environment on control systems in the vicinity of high energy lines, including the main steam and main feedwater lines. As a result of this study, modifications were made at the Cook Plant to assure that a high energy line rupture on one secondary loop could not lead to an adverse failure on the adjacent loop.

Main feedwater piping outside of containment up to and including the containment isolation valve is designed to Seismic Category I standards. As shown in Appendix 'O' to the FSAR, the feedwater isolation valves are on the order of one to two feet upstream of their respective containment penetration. Stress calculations were performed on this and other main feedwater piping runs.

The calculated stresses at various points along the main feedwater line are shown in Table O-3 of Appendix 'O' to the FSAR. The maximum combined stress at the containment penetration of any feedwater line never exceeds one third (33%) of the code allowable stress.

The stress values shown in Table 0.3 of Appendix 'O' are for the Unit No. 1 main feedwater lines. The orientation of and layout of the main feedwater lines in Unit No. 2 are similar to Unit No. 1. The stresses in the Unit No. 2 feedwater lines are of similar magnitude to those reported in Table 0.3 for Unit No. 1.

Rupture of a main feedwater pipe at its containment penetration was postulated as part of the Appendix 'O' study. The environmental conditions in the main steam enclosure housing the affected secondary loop were calculated using the Westinghouse CONTEMPT-PS and TMD Codes.

The calculational model used for the analysis is extremely conservative in that:

(1) No credit was taken for the steam flow restrictors inside and outside containment. These restrictors limit the effective area of a steam line rupture at the penetration to 1.4 square feet. The Appendix 'O' analysis conservatively assumed a break area of 4.27 square feet. The flow restrictors outside containment in the intact secondary lopps would limit backflow to the break.

(2) The mass and energy assumed to be released into the affected main steam enclosure is based on continuous forward flow from the affected generator and ten seconds of backflow from the intact steam lines. The ten second time limit assumed for the termination of backflow was based on an assumed ten second closure time of the main steam isolation valves (MSIVs). Subsequent to the submittal of Appendix 'O' to the Commission in September 1973, the closure time of the MSIVs has been reduced to $\leq$5 seconds. Thus, the backflow assumed in the Appendix 'O' analysis is very conservative with respect to the current Cook Plant valve response times.

The environmental conditions in a main steam enclosure are more severe following steam line rupture than they would be following a feedwater line rupture. The more limiting environmental conditions of the steam line break were used in the determination of minimum environmental qualification requirements for equipment in the enclosure required to function during and after the event.

Appendix 'O' shows that the peak calculated temperature in the west main steam enclosure following a steam line break is 230^{o}F and the peak pressure is approximately 12 psig. The enclosure temperature decreases rapidly once steam line isolation has been completed (five seconds after the initiation of the event), and would eventually stabilize at the ambient outside temperature and atmospheric pressure.

Operation of the PORV on a given steam line is controlled by one of the three pressure transmitters on each loop used for the "Steamline Pressure" function in the Solid-State Protection System (SSPS). The transmitter output signal is processed through the SSPS and back to the Electro Pneumatic Transducer (EPT) device on the PORV. The EPT output controls the PORV position.

The transmitters utilized for PORV control are Foxboro Model E11GM Gauge Pressure Transmitters. The transmitters are contained in leak tight watertight housings and are located such that a HELB in one loop will not have an adverse impact on PORV control system equipment on the adjacent loop. The electrical cable in the enclosure running from the transmitter to the SSPS has been protected from HELB on the adjacent loop. Similar protection was provided for cable from the SSPS to the EPT device. The cable is qualified for post LOCA service inside containment. Hence the transmitters and their associated cabling would not be affected by the adverse environment generated by the HELB in the enclosure. The EPT devices are manufactured by the Fisher Company and are also qualified for the pressure and temperature environment for the post-LOCA use inside containment. Actuation of the PORV positioner would only occur if an "action" signal were received from the EPT device. The pressure transmitters, the associated cable, and the EPT device are qualified for operation following a HELB, thus the possibility of environmentally induced PORV opening is eliminated. Furthermore the duration of the temperature "spike" resulting from the feedwater line rupture is so short that the thermal capacity of the equipment coupled with the protective features mentioned above and distance from the break dampens the environmental condition to the point where equipment in the PORV control system will perform its intended function. Thus the postulated scenario is not credible at the Cook Plant.

2. PRESSURIZER PORV CONTROL SYSTEM:

Summary of Postulated Scenario

Westinghouse has hypothesized that following a feedwater line rupture inside containment, the pressurizer PORV control system malfunctions in such a manner that the PORVs fail in the open position. Thus, in addition to a feedwater line rupture between the steam generator nozzle and the containment penetration, a break of the reactor coolant system boundary occurs in the pressurizer vapor space.

Westinghouse has evaluated the consequences resulting from such an accident in the following manner. Section 4.2 of WCAP-9600 describes transient analyses for a postulated loss of all main and auxiliary feedwater (no pipe rupture). The results indicate that, in the event that the operator cannot restore auxiliary feedwater to the steam generators, the operator is required to open the pressurizer PORVs within 2,500 seconds to maintain adequate core cooling.

The interaction scenario postulated above is similar to that presented in Section 4.2 of WCAP-9600. The additional assumptions made are the following:

a. A feedline rupture is assumed to occur between the steam generator nozzle and the containment penetration.

b. Auxiliary feedwater is injected into the intact steam generators following the feedline rupture.

Conservatively assuming that all liquid inventory in the steam generator associated with the ruptured feedline is lost via the rupture without removing any heat (i.e., liquid blowdown), the loss of heat sink due to the liquid inventory blowdown of the ruptured steam generator is more than counterbalanced by the auxiliary feedwater being injected into the intact steam generators following reactor trip. Therefore, the results of the analyses presented in WCAP-9600, Section 4.2, which illustrated that the operator is not required to take corrective action for at least 2,500 seconds following the loss of feedwater, also apply to this scenario.

Cook Plant Status

Our review of the type and qualification of the components utilized in the pressurizer PORV control system indicates that the control system will operate correctly. Specifically, the transmitters are protection grade instruments qualified for the anticipated environment expected inside containment as are the cables associated with the PORV control system. The solenoids and relief valves themselves are located inside the enclosure near the top of the pressurizer. This equipment is protected from a 100% humidity condition and from jet impingement. Under a feedwater line rupture inside containment, the ice condenser would function, thus, environmental conditions in the top of the pressurizer enclosure would not be as severe as the lower containment. The solenoids for the PORVs will function as designed up to an ambient temperature of 176°F. The duration of the temperature "spike" resulting from a feedwater line rupture inside containment is so short that when the thermal capacity of the equipment is coupled with the protective features mentioned above and when the distance from the break is taken into account, the environmental condition will be damped to the point where the solenoids will perform their intended function. Thus, the Westinghouse scenario is not credible at the Cook Plant.

3. MAIN FEEDWATER CONTROL SYSTEM

Summary of Postulated Scenario

Westinghouse has hypothesized an event scenario initiated by a small feedwater line rupture, break area $\leq$ 0.2 sq. ft., between the containment penetration and the containment isolation valve. The resulting adverse environment would then have an adverse effect on the main feedwater regulating valves on the intact loops. Due to the adverse environment, the feedwater regulating valves on all four loops would then go to the closed position in the Westinghouse scenario. Such a sequence of events could conceivably lead to the situation where all four steam generators would be at the low-low water level at the time of reactor trip. For reasons cited below, the scenario postulated by Westinghouse is not possible at the Cook Plant.

Cook Plant Status

The feedwater regulating valves for secondary loops 1 and 4 are located in the annulus outside of the east main steam enclosure. The feedwater regulating valves for secondary loops 2 and 3 are located in main steam/feedwater access tunnel. A small feedwater line rupture near the containment penetration could, at most, have an adverse effect on the feedwater regulating valves on two secondary loops. The electrical equipment used to control the feedwater regulating valves in the vicinity of the subject high energy lines is, in fact, fully qualified for use in a post-LOCA environment inside containment. Thus, the event scenario postulated by Westinghouse is not possible at the Cook Plant.

4. SECONDARY SIDE BREAKS CAUSING MALFUNCTION OF THE AUTOMATIC ROD CONTROL SYSTEM

Summary of Postulated Scenario:

Westinghouse has postulated an event scenario consisting of an intermediate secondary side break occurring at a time when the automatic rod control system (ARCS) is in operation. Westinghouse has hypothesized that the post-break containment environment could cause the ARCS to malfunction in such a manner that the control rods are stepped out of the core before a reactor trip is generated.

The increased power generation coupled with the RCS conditions produced by the excessive cooldown could then cause the DNBR to decrease below established safety limits. Westinghouse assumes that a reactor trip will occur about two minutes after the break due to the overpower ΔT trip. The scoping calculations were done for a three loop RESAR-3S plant.

Cook Plant Status:

Although this postulated event is not specifically analyzed for Cook Plant, there are numerous FSAR, reload and other analyses which are relevant to this event. Some of these are: Steam Line Break (SLB) Accident Analysis, (FSAR Section 14.2.5 - Units 1 and 2; Appendix Q 022.9, 212.7 and .12; XN-76-35, XN-NF-70-17(1); and generic LOTIC-III intermediate SLB analyses) and Uncontrolled Rod Withdrawal Accident Analysis (FSAR Section 14.1.2 - Units 1 and 2, XN-76-35, and SN-NF-79-17(1)). All of these analyses show that the Cook responses to these events are well within established acceptance criteria.

There are, in addition, major differences between the four loop ice condenser Cook Plant and the reference three loop RESAR 3S plant from a dynamic response viewpoint. In particular, the containment design of the Cook Plant calls for much lower "containment high-pressure trip" setpoints (1.2 psig). The Cook Plant has four excore detectors, all of which must fail for the above scenario to be at all applicable. The NSSS response of a four loop plant is not the same as the three loop plant.

In order for the Westinghouse scenario to become a problem at the Cook Plant, the NSSS must remain unchecked long enough for a DNB condition to develop. This implies both the lack of a reactor trip and the addition of a certain amount of reactivity.

The relevant automatic reactor trips at the Cook Plant are:

1. High neutron flux
2. High positive flux rate
3. Overtemperature ΔT
4. Overpower ΔT
5. Steam flow/feed flow mismatch coincident with low steam generator level
6. Low-Low steam generator level
7. Pressurizer Pressure-high
8. Reactor Trip from safety injection signal generated by:
   a) containment pressure high
   b) differential steam line pressure
   c) low pressurizer pressure
   d) steam flow in two steam lines high coincident with Tavg low-low or steam line pressure low (Unit 1)
   e) low steam line pressure (Unit 2)

There are a number of these trips which are independent of the reactivity insertion rate postulated. The steam/feed mismatch coincident with low steam generator level trip and the low-low steam generator level trip would occur during such an event. The reactor trip from a safety injection signal generated by differential steam line pressure should also occur. An estimate of this trip time, using the RESAR 3S results for a 0.22 ft^2 SLB and the Cook setpoints, is about 80 seconds. The final and most significant trip is high containment pressure generating a safety injection signal and a subsequent reactor trip at 1.2 psig. A generic ice condenser analysis for a 0.1 ft^2 SLB, at 30% power, shows that the high pressure setpoint is reached in 20 seconds.

The reactivity insertion rates depend on the core life and the withdrawal rate and rod worths of the control assemblies. Considering the cooldown effect of an intermediate SLB and possible range of control rod withdrawal speeds, the possible insertion range is bounded by 6 pcm/sec. and 40 pcm/sec. The results of the FSAR analysis of uncontrolled rod withdrawals are as follows:

Insertion Rate	Unit 1	Unit 2
80 pcm/sec (Unit 1) 70 pcm/sec (Unit 2)	trip in 5 sec. on high flux. MDNBR = 1.65	trip in 5 sec. on high flux. MDNBR = 2.2
2 pcm/sec	trip in 55 sec. on overtemperature ΔT MDNBR = 1.4 (this is independent of the excore detectors)	trip in 80 sec. on overtemperature ΔT MDNBR = 2.0 (this is independent of the excore detectors)

The FSAR analyses are not completely applicable to the Westinghouse scenario because the coolant conditions are different for the two events. However, considering that the most likely reactivity insertion rate is around 30 pcm/sec., the high flux trip would occur in approximately 10 seconds. Thus, the excore detectors would have to operate for only 10 seconds after the RCS cooldown begins.

The excore detectors are located inside the reactor cavity in individual wells. They are protected from a 100% humidity condition and against jet impingement. The duration of the temperature "spike" resulting from a secondary line rupture inside containment is very short. This coupled with the protective features mentioned above, will dampen the environmental condition to the point where the excore detectors will most likely perform their intended function.

This scenario is not credible at Cook Plant since the reactor will be tripped on high containment pressure in approximately 20 seconds which is much less than the two minutes assumed by Westinghouse.

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(1) XN-76-35 "Plant Transient Analysis for the Donald C. Cook Unit 1 Nuclear Plant"

XN-NF-79-17 "Plant Transient Analysis for the Donald C. Cook Unit 1 Nuclear Power Plant"

Control System / Accident	Reactor Control	Pressurizer Pressure Control	Pressurizer Level Control	Feedwater Control	Steam Generator Pressure Control	Steam Dump System	Turbine Control
Small Steamline Rupture	X	X			X		
Large Steamline Rupture		X			X		
Small Feedline Rupture	X	X		X	X		
Large Feedline Rupture	X	X			X		
Small LOCA	X	X		X			
Large LOCA							
Rod Ejection							

TABLE 1

PROTECTION SYSTEM-CONTROL SYSTEM POTENTIAL ENVIRONMENTAL INTERACTION

X - POTENTIAL INTERACTION IDENTIFIED THAT COULD DEGRADE ACCIDENT ANALYSIS

☐ - NO SUCH INTERACTION MECHANISM IDENTIFIED