INDIANA & MICHIGAN ELECTRIC COMPANY

P.O. BOX 16631
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August 3, 1984
AEP:NRC:0775M

Donald C. Cook Nuclear Plant Unit Nos. 1 and 2
Docket Nos. 50-315 and 50-316
License Nos. DPR-58 and DPR-74
SUPERHEATED STEAM RELEASE FOLLOWING A MAIN STEAM LINE BREAK

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

Dear Mr. Denton:

This letter responds to your staff's verbal request of July 20, 1984, that information be provided regarding the effects of superheated steam following a postulated Main Steam Line Break (MSLB) at the Donald C. Cook Nuclear Plant. The requested information is contained in the attachment to this letter.

The enclosed information summarizes our preliminary review findings on this topic, including the identification of those physical plant areas where a postulated superheated steam release could occur, and the electrical equipment items within those areas which must function to mitigate or control the accident. Our ongoing reviews of cable routings and equipment operability requirements have been taken into account and, where applicable, the availability of alternate equipment to perform required functions has been considered. We have also considered the effect of increased temperatures on the steel and concrete in the physical plant areas. Our preliminary review has indicated that the identified areas would remain structurally sound. It is noted that this information has not yet undergone our corporate quality assurance review.

Furthermore, from a preliminary mechanistic pipe break evaluation performed by Westinghouse Electric Corporation, it appears very unlikely that a steam pipe rupture of the size necessary to generate superheated steam will occur. Indeed, Westinghouse has determined that unstable crack extension will not occur when a postulated flaw is subjected to a faulted load, and that the crack opening area resulting from the load would be less than that at which superheating of steam may occur, i.e., approximately 0.1 ft^2.

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We note that the Westinghouse Owners Group (WOG) is currently considering generic actions pertaining to the issue of superheated steam release following a MSLB. We will participate in WOG activities in this area, and carefully consider any WOG findings for impact on the Donald C. Cook Nuclear Plant. Our involvement with the WOG in this area is expected to supplement our ongoing internal review.

This letter and its attachment have been reviewed and approved by the Nuclear Safety Design Review Committee (NSDRC) and the Plant Nuclear Safety Review Committee (PNSRC). Based on the preliminary review findings, both committees believe that the Donald C. Cook Nuclear Plant can be safely shut down in the event of a Main Steam Line Break in either the East or West Main Steam Enclosure.

This letter has been prepared following Corporate procedures which incorporate a reasonable set of controls to ensure its accuracy and completeness prior to signature by the undersigned.

Very truly yours,

M. P. Alexich
Vice President

MPA/dam

cc: John E. Dolan
W. G. Smith, Jr. - Bridgman
R. C. Callen
G. Charnoff
E. R. Swanson - NRC Resident Inspector, Bridgman

ATTACHMENT TO AEP:NRC:0775M
SUPERHEATED STEAM RELEASE FOLLOWING A MAIN STEAM LINE BREAK
DONALD C. COOK NUCLEAR PLANT UNIT NOS. 1 AND 2

Introduction

Westinghouse Electric Corporation (W) letter No. NS-PL-12153 [W. J. Johnson, W, to M. P. Alexich, American Electric Power Service Corporation (AEPSC)], dated January 3, 1984, indicated that W had reanalyzed the blowdown from a Main Steam Line Break (MSLB) in response to an NRC question posed in 1982. This reanalysis involved the modeling of steam superheating which may occur when the steam generator U-tubes become uncovered, and considered the effects this superheated steam could have on containment response. As noted in W letter No. NS-PL-12153, the results showed ". . .that for standard ice condenser containments, the impact is an increase in the peak temperatures reached in the lower compartment following a steamline break. . . .For the D. C. Cook units, this effect will not occur. This is because the lower compartment sprays will be actuated before the tube bundle uncovers and the spray will de-superheat the containment atmosphere. . . ."

W letter No. AEP-84-611 [W. J. Johnson, W, to W. G. Smith, Jr., Indiana & Michigan Electric Company (IMECo)], dated June 6, 1984, extended the concern with regard to superheated steam to the environmental qualification of outside containment equipment. More specifically, this letter indicated that the worst case temperature versus time qualification profiles might not be adequately reflected in the Final Safety Analysis Report (FSAR) if saturated steam blowdowns were previously used. The NRC was verbally notified by AEPSC staff that this issue was potentially reportable under 10 CFR 50.49(h) for the Donald C. Cook Nuclear Plant.

This attachment provides preliminary review findings with regard to the superheated steam issue for the Donald C. Cook Nuclear Plant. The sections below consider the physical plant areas involved, the identification of 10 CFR 50.49 equipment within those areas, the operability requirements for the equipment items of concern, the availability of alternate equipment to fulfill required safety functions, and the effects of superheated steam on steel and concrete.

Identification of Physical Plant Areas

The Donald C. Cook Nuclear Plant FSAR Appendix O identifies five (5) high energy piping systems, i.e., main steam system, feedwater system, steam generator blowdown system, Chemical and Volume Control System (CVCS), and the steam supply to the Turbine Driven Auxiliary Feedwater Pumps (TDAFPs). The

CVCS is excluded from the superheated steam issue, however, because a break in that system does not result in loss of secondary side inventory. Postulated breaks in the 4" nominal diameter lines which supply steam to the TDAFPs, or in the 2" and 3" lines which comprise the steam generator blowdown system, are also excluded from consideration since such breaks will not result in break flow areas greater than 0.1 ft^2 which, according to W, is the lower break size limit at which superheat may be expected to occur. Additionally, W has informed us that superheated steam releases are not a concern for feedwater line ruptures. Therefore, only postulated main steam system ruptures are considered.

W letter No. AEP-84-611 indicates that U-tube uncovery and subsequent superheated steam generation may occur as late as about 9 minutes for small breaks on the order of 0.1 ft^2, or as early as about 1.5 minutes for a large double ended MSLB. According to the FSAR Appendix O, however, the main steam stop valves, i.e., MRV-210, -220, -230, and -240 will close within 10 seconds after the occurrence of a MSLB, limiting the flow of superheated steam outside containment. For this reason, it is believed that the only break location outside containment which may involve the release of superheated steam is in that length of main steam piping between the containment wall and the main steam stop valves.

The breaks to be considered may occur either in the East Main Steam Enclosure (lead Nos. 1 and 4) or in the West Main Steam Enclosure (lead Nos. 2 and 3). As indicated in the Donald C. Cook Nuclear Plant FSAR Appendix O Figure O-17, the steam introduced into the West Main Steam Enclosure may vent either to the atmosphere or to the Main Steam Accessway. The steam entering the Main Steam Accessway may ultimately pass into the turbine building where its effects upon equipment should have no impact on plant nuclear safety. The steam entering the East Main Steam Enclosure, on the other hand, is vented directly to the atmosphere. A break in one enclosure will not result in a steam environment in the other.

Based on the above, it is believed that the physical plant areas of concern are the East Main Steam Enclosure, the West Main Steam Enclosure, and the Main Steam Accessway.

Identification of 10 CFR 50.49 Equipment

The Donald C. Cook Nuclear Plant 10 CFR 50.49 equipment list, provided in Attachment 2 to IMECo letter No. AEP:NRC:0775L [M. P. Alexich, IMECo, to H. R. Denton, NRC], dated July 6, 1984, was reviewed to identify those equipment items located in the East and West Main Steam Enclosures and the Main Steam Accessway. These equipment items, exclusive of their auxiliary electrical installations (e.g., cables, terminations, etc.), are identified below by plant identification number:

- East Main Steam Enclosure

  FFC-210, -211
  FFI-210, -240
  FMO-211, -212, -241, -242
  MPP-210, -211, -240, -241

- West Main Steam Enclosure

  FFI-220, -230
  FMO-221, -222, -231, -232
  MCM-221, -231
  MPP-220, -221, -230, -231

- Main Steam Accessway

  FFC-220, -221, -230, -231
  XSO-293, -294, -295, -296

It is noted that some of the FMOs identified above are located in an area outside containment which may not see the superheated environment resulting from a postulated MSLB. These FMOs have, however, been conservatively included within the scope of review.

An ongoing review is being conducted to identify those safety related cables routed through the East and West Main Steam Enclosures. In addition to identifying those cables which supply power to the equipment items of concern within each enclosure, it has been determined that some cabling for FMO-211 and -241 (in the East Main Steam Enclosure) is routed through the West Main Steam Enclosure. The effect of this cable routing on the ability to cool down the plant is discussed in further detail below. The safety related cables within the two (2) enclosures are also being reviewed to determine if they supply power to other 10 CFR 50.49 equipment items outside containment, or to those alternate items inside containment providing redundant functions. No additional 10 CFR 50.49 equipment items have as yet been identified as being impacted due to a potential loss of power resulting from postulated cable damage.

Additionally, a review was conducted to ensure that the cables servicing the TDAFP trip-throttle valve and the MDAFPs are not routed through the Main Steam Accessway. Our review findings indicate that these cables are not in an area subject to postulated superheated steam environmental conditions, and therefore the auxiliary feedwater pumps may be assumed operable until such time as the operator isolates auxiliary feedwater to the faulted steam generator.

Operability Requirements and Availability of Alternate Equipment

Each of the 10 CFR 50.49 equipment items identified above was reviewed to determine its required operating time and, if its required operating time was less than the earliest time at which superheated steam could be expected to bepresent, i.e., about 1.5 minutes following a MSLB, it was determined if subsequent failure would lead to an unsafe plant condition. If the required operating time was greater than about 1.5 minutes, the potential failure modes of the equipment were considered as was the availability of alternate equipment not subject to the superheated steam environment. The results of this review are discussed below:

(1) FFC-210, -211, -220, -221, -230, and -231 are the steam generator main feedwater flow transmitters for lead Nos. 1 through 3, respectively. These transmitters are required to operate during the first twenty-five (25) seconds following a MSLB outside containment. Since their required operating time is less than the earliest time at which superheated steam may be expected to be present, i.e., about 1.5 minutes following a MSLB, failure of these transmitters under postulated superheated steam conditions should not affect plant nuclear safety.

(2) FFI-210, -220, -230, and -240 are the steam generator auxiliary feedwater flow transmitters for lead Nos. 1 through 4, respectively. Should these transmitters fail under postulated adverse environmental conditions, the operator may still monitor secondary side conditions through the use of the steam generator water level transmitters located inside containment. The steam generator water level transmitters located inside containment are not subject to the adverse environmental conditions resulting from a MSLB outside containment.

(3) FMO-211, -221, -231, and -241 are the auxiliary feedwater isolation valves for the TDAFP supply to lead Nos. 1 through 4, respectively. FMO-212, -222, -232, and -242 are the auxiliary feedwater isolation valves for the Motor Driven Auxiliary Feedwater Pumps (MDAFPs) supply to lead Nos. 1 through 4, respectively. In the event of a MSLB outside containment, the operator is called upon to isolate auxiliary feedwater to the faulted steam generator. This operator action, which is expected to take place about ten (10) minutes following the break, may be hampered by postulated failures of the FMOs under the adverse superheated steam conditions. In case of such a failure, auxiliary feedwater flow to the faulted steam generator may be isolated by tripping the TDAFP and one (1) MDAFP, or by closing the manual isolation valves at the discharge of these pumps. This action will result in the isolation of all auxiliary feedwater to the faulted steam generator and to the associated steam generator supplied by the tripped MDAFP, i.e., lead Nos. 1 and 4 or lead Nos. 2 and 3 will be completely isolated at the same time. The auxiliary feedwater flow to the other two (2) steam generators from the TDAFP will also be isolated; however, auxiliary feedwater flow to these steam generators will be supplied by the other MDAFP, ensuring plant cooldown. The ability to regulate auxiliary feed-

water flow to these two (2) steam generators from the running MDAFP is still provided by the appropriate FMOs, e.g., FMO-212 and -242 (in the East Main Steam Enclosure) will not be rendered inoperable as a result of a MSLB in lead Nos. 2 or 3 (in the West Main Steam Enclosure). It is also noted that a review of the electrical installations has indicated that cabling for FMO-211 and -241 (in the East Main Steam Enclosure) runs through the West Main Steam Enclosure area. This cable routing does not affect the conclusions above since it is assumed that the TDAFP will be tripped or isolated if the FMOs for the faulted steam generator fail open.

(4) MCM-221 and -231 are the isolation valves for the main steam supply to the TDAFPs. These valves are normally open and, during postulated accident conditions, must remain open to supply steam to the TDAFPs. These valves are motor operated and designed to fail in an "as is" condition, i.e., they should fail open if their control cables, terminations, etc., are damaged by the postulated adverse environmental conditions.

(5) MPP-210, -211, -220, -221, -230, -231, -240, and -241 are the main steam pressure transmitters located outside containment for lead Nos. 1 through 4, respectively. Should these transmitters fail under postulated adverse environmental conditions, the operator may still monitor secondary side conditions through the use of the steam generator auxiliary feedwater flow transmitters (if available) or the steam generator water level transmitters located inside containment.

(6) XSO-293, -294, -295, and -296 are the solenoid valves which control the main feedwater regulating valves for lead Nos. 2 and 3, i.e., FRV-220 and -230, respectively. These solenoids are required to complete their functions within twenty-five (25) seconds following a MSLB outside containment. Since their required operating time is less than the earliest time at which superheated steam may be expected to be present, i.e., about 1.5 minutes following a MSLB, and since subsequent failure of the solenoids will not cause a spurious opening of the FRVs, failure of the solenoids under postulated adverse environmental conditions should not affect plant nuclear safety.

Effects on Steel and Concrete

In general, the effects of superheated steam on the steel and concrete in the areas of concern were considered. With regard to the concrete, it is believed that there may be a 5% to 20% decrease in compressive strength and similar reductions in stiffness, for temperatures up to about 500°F. However, as noted by AEPSC's consultant in this matter, i.e., Stevenson & Associates, ". . .the in place concrete, even if the twenty percent reduction occurred, should be above the design strength. These values are effected by the rate of heating, amount of moisture confinement and temperature at which the load tests are performed as well as the maximum concrete temperature. . ." We expect that there may be some local spalling of concrete at the heated surface of the concrete, but that this spalling will not cause structural distress to the concrete structures.

With regard to the steel, AEPSC's consultant noted that the ". . .effect of elevated temperature on structural steel up to 450°F has been to reduce the yield strength of such steels by 5 to 10 percent. By way of comparison, the ASME Section III Code for Class 2 and 3 components does not consider such strength changes in design until temperatures exceed 600°F. . . ." Based on the above and the expected duration of a superheated environment, i.e., a maximum of eight (8) or nine (9) minutes assuming operator action to isolate auxiliary feedwater, it is believed that the structures at the Donald C. Cook Nuclear Plant would remain structurally sound for the postulated event.