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SUBJECT:
RESPONSE TO NRC"S LTR OF 03/03/78... FURNISHING ADDL INFO CONCERNING THE STEAM GENERATOR SUBCOMPARTMENT PRESSURE RESPONSE ANALYSIS AND THE STAFF REVIEW OF RESULTS OF AUDIT OF ENVIRON QUALIFICATION RECORDS, FOR UNIT 1... W/ATT INFO CONCERNING CABLE TESTING.

PLANT NAME: COOK - UNIT 2
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INDIANA & MICHIGAN POWER COMPANY
P. O. BOX 18
BOWLING GREEN STATION
NEW YORK, N. Y. 10004

March 7, 1978

Donald C. Cook Nuclear Plant Unit 2
Docket No. 50-316
DPR No. 74

Mr. Edson G. Case, Acting Director
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Mr. Case:

This letter is in response to Mr. Karl Kniel's letters of March 3, 1978 wherein the NRC Staff requested additional information concerning the Steam Generator Subcompartment Pressure Response Analysis and the Staff Review of Results of Audit of Environmental Qualification Records, for Donald C. Cook Nuclear Plant Unit 2. The staff requires resolution of this issue prior to initial criticality of D.C. Cook Unit 2. Attachment 1 to this letter presents an item by item response to your staff's request for additional information on the Steam Generator Enclosure Analysis. Attachment 2 to this letter presents information on cable testing of Cerro and Continental instrument cable.

Very truly yours,

John Tillinghast

John Tillinghast
Vice President

JT:em

Sworn and subscribed to before me
on this 7th day of March , 1978 in
New York County, New York

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, 1979

cc: (attached)

3001
S
1/1

cc: G. Charnoff
R. C. Callen
P. W. Steketee
R. J. Vollen
R. Walsh
R. W. Jurgensen
D. V. Shaller-Bridgman

Attachment - 1

Question 1 - Mechanical Engineering Branch

For the steam generator vertical support columns:

a) Provide "Faulted Condition" design criteria utilized to provide assurance against column buckling under the loading resulting from the worst case postulated steam line break, i.e., at the steam generator outlet nozzle.

b) Provide calculated stresses for the columns in a) resulting from compressive/buckling load effect of this postulated break.

Response

a) The AISC-69 formulae used to establish allowable buckling load for the faulted condition are the same as those used for the normal, upset and emergency conditions. These equations are used directly without adjustments for normal and upset conditions, whereas factors 1.5 and 1.67 are applied to obtain allowables for the emergency and faulted conditions, respectively.

b) For the steam generator vertical support columns, the calculated stress in the column is 41.3 KSI. The allowable stress in the column is given by AISC Equation 1.5-1 times the faulted condition factor of 1.67 and was found to be 43.5 KSI. Thus, the actual stress is 95% of allowable.

Question 2 - Mechanical Engineering Branch

Describe or demonstrate how material properties at operating temperatures of the steam generator supports were taken into account in designing the supports to AISC-69 standard.

Response

AISC-69 does not address properties of materials at elevated temperatures. Hence, these effects were not discussed in our previous submittal of February, 1978. However, studies were performed to investigate elevated temperature effects on material properties based on tests(1) published by U.S. Steel which demonstrate a loss of tensile strength of approximately 7% for temperatures of 500°F to 600°F.

(1) R.L. Brockenbrough, B. G. Johnstone, "USS Steel Design Manual, United States Steel Corporation, Pittsburgh, Pennsylvania July, 1968

Question 2 - Mechanical Engineering Branch

Response (Cont'd)

The original design load capacity for the steam generator upper lateral support was 3000 kips. For this load a maximum extreme fiber stress in the belly band was found to be 46 KSI or 8% less than the yield stress at room temperature. For the 2858 kip load corresponding to the governing main steam line break load combination, a stress of 43.8 KSI was calculated, thus, achieving a margin of 12%, compared to the yield stress at room temperature. This margin more than offsets the decrease in strength at operating temperature.

The lower support columns have a normal operating temperature of 120°F. Hence, the properties are insignificantly changed by operating temperature effects.

The lower lateral support design is not governed by the main steam line break accident.

Question 3 -Mechanical Engineering Branch

Provide assurance that under the postulated main steam pipe breaks resulting stresses in unbroken piping attached to the steam generator do not exceed the Faulted Condition Stress limits (Service Limit D) per ASME Section III of the Boiler and Pressure Vessel Code.

Response

D.C. Cook Unit 2 was designed under the rules of ANSI B31.1 and not ASME III. However, a steam line break was considered in the design analysis. The calculated (loop) stresses for the steam line break were quite low and are well within the allowables for ASME Level D stress limits.

Question 4 - Mechanical Engineering Branch

Provide additional information or the results of analysis that demonstrates that utilizing the loads derived from the nine node cavity model to analyze for support integrity indeed envelopes, i.e. , assures that worst case loading combination has been accounted for, any similar analysis performed with loads derived from the 17 node cavity model.

Response

A. Summary of steam generator upper support loads (in Kips):

Item	Fx	Fz
9 node model (as previously transmitted.	2233	445
17 node model @t= .00991 sec	1639	253
17 node model @t= .01477 sec.	1617	362
17 node model @t= .01619 sec	1521	351

Therefore it is concluded that the previous calculations using the 9 node model were more conservative as previously stated and do represent the upper bound steam generator support loads for postulated steam line breaks.

Question 1 - Structural Engineering Branch

The calculation for the section factor of safety is based purely on moment and neglects the tensile forces in the section. Your factor of safety calculation should be computed on the basis of the stresses in the reinforcing steel.

Response

We have recalculated the section factor of safety based upon the stresses in the reinforcing steel. In no case does the section factor of safety fall below 1.5.

Question 2 - Structural Engineering Branch

The allowable shear capacity should consider the effects of membrane tension per EQ 11-8 of ACI 318-71 (Section 11-4.4). The applicant has used the equation as stated in Section 11.4.3 of ACI 318-71 which is for members subjected to axial compression. Provide justification for your approach; show that such a deviation is not significant to the functional requirement of the wall.

Response

The Donald C. Cook Nuclear Plant was designed using the design criteria of ACI 318-63 code. The steam generator enclosure was reanalyzed for the new loading conditions using the design criteria of this code. At the request of the NRC we have investigated the sections using the shear criteria of the ACI 318-71 code. Under this new criteria one section has a factor of safety in shear below 1.5. The location and stress mode is as follows:

Location	Type of Stress	F.S.
3-2 (Perimeter Wall)	Hoop Shear	1.48

However, the factors of safety are computed based on a concrete 28 day design strength of 3500 psi. The actual minimum 28 day concrete strength for the steam generator enclosure area is 4400 psi and the actual minimum 90 day strength is 5800 psi.

Response

Using the actual concrete 28 day and 90 day strengths, yields safety factors as follows:

Location	28 Days	90 Days
3-2	1.66	1.90

Attachment - 2

REQUEST FOR ADDITIONAL INFORMATION

The following provides additional information regarding environmental qualification of instrumentation cable which has been requested by the Staff by a letter dated March 3, 1978 from Mr. Karl Kniel to Mr. John Tillinghast.

Cerro Wire & Cable - Instrumentation Cable

The documentation for this cable identifies tests performed on single conductor No. 12 AWG copper wire with 30 mils of crosslinked polyethelene insulation. The test profile exceeds the worst anticipated accident conditions for both radiation and temperature. The test conductor demonstrated more than adequate physical and electrical integrity following the test.

The cable supplied to the D. C. Cook Plant is an instrumentation cable consisting of 4 conductors. Each conductor is a No. 16 AWG copper wire with 30 mils of crosslinked polyethelene insulation. The insulation material and thickness is identical to the samples tested. The 4 wires are grouped together and wrapped with a shield consisting of 2 mil copper tape backed by mylar. A single No. 18 AWG (minimum) wire is added in contact with the copper shield to maintain shield continuity. The shielded conductor bundle is then covered with a 45 mil jacket of hypalon (chlorosulphonated polyethelene) insulation for mechanical protection and electrical isolation of the shield.

The testing done on the single insulated conductor demonstrated the adequacy of the cable used in the containment. The individual wires of the cable can withstand the containment environment without the additional protection provided by the shield and hypalon jacket. Hypalon insulation and jacketing materials have demonstrated excellent resistance to containment environment and offer excellent protection from the containment environment to the individual wires which are capable of withstanding the containment environment without this additional protection.

Continental Wire & Cable Co - Instrumentation Cable

The documentation for the instrumentation cable supplied by Continental Wire & Cable covered tests performed on samples of the insulating material subjected to a test profile which exceeds the worst anticipated accident conditions inside the containment. These tests were not conducted on cable samples. However, the only material subject to modification of physical and electrical properties is the insulating materials. The copper wire remains essentially unchanged except for minor physical property changes through the temperature excursions and other environmental exposures which occur in the worst anticipated accident condition.

The tests on the insulating material samples resulted in minor changes in the physical and electrical properties of the test samples but also demonstrates their suitability for continued use following the environmental test.

Testing of the finished cable was done during the instrumentation splice qualification tests performed in November 1977 at Conax Corp in Buffalo, New York. Samples of Continental Wire & Cable twisted shielded quad (4 insulated wires plus shield) on hand at the D. C. Cook Plant and identical to the cable used in the containment were used to make the splices to the electrical penetration feedthroughs and were subjected to both test profiles at that time. One set of samples was subjected to 250°F steam for 6 hours followed by immersion in sodium borated water at a minimum temperature of 190°F for 194 hours. The same test samples were then subjected to a test profile of 340°F steam for 1 hour, 250°F steam for 5 hours followed by an additional immersion in sodium borated water at a minimum temperature of 190°F for 18 hours. The second set of samples was subjected to a test profile of 340°F steam for 1 hour, 250°F steam for 5 hours followed by immersion in sodium borated water at 190°F or above for 18 hours.

In all of the above tests no evidence of failure or serious degradation of the physical or electrical properties of the cables was detected. Following both tests the cable splices and electrical penetration feedthrough passed high potential tests equal to that required for new cable. The above tests are identified as IPS-316, IPS-317 and IPS-319. Documentation reports of these tests identified as IPS-326, IPS-327 and IPS-329, respectively are in the possession of D. W. Hayes of the Region III Office in Chicago.