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SUBJECT: Forwards actions taken re allowable stresses for safety-related piping and piping supports.

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AEP:NRC:1100

Donald C. Cook Nuclear Plant Units 1 and 2
Docket Nos. 50-315 and 50-316
License Nos. DPR-58 and DPR-74
ALLOWABLE STRESSES FOR PIPING AND PIPING SUPPORTS

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

Attn: T. E. Murley

July 12, 1989

Dear Dr. Murley:

This letter provides information relative to operability and reportability determinations currently being made for the Donald C. Cook Nuclear Plant. Specifically, the use of an interim acceptance criteria for operability decisions and the application of 10 CFR 50.72 and 10 CFR 50.73 for reportability determinations are addressed for situations where as-found piping and piping supports are found not to meet the original FSAR design requirements. This subject has been previously discussed with NRC Region III as recently as May 18, 1989. The Region expressed no reservations with respect to the positions specified herein, although they did request that the information be docketed. This letter satisfies that request.

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

Sincerely,

M. P. Alexich
Vice President

MPA/eh
Attachment
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ATTACHMENT TO AEP:NRC:1100

ALLOWABLE STRESSES FOR PIPING AND PIPING SUPPORTS

The following information is provided to describe the actions being taken for the Cook Nuclear Plant whenever safety-related piping and/or piping supports are found that deviate from the as-designed condition.

BACKGROUND

As a result of the Inservice Inspection Program developed for the Cook Nuclear Plant to satisfy the requirements of ASME Boiler & Pressure Vessel Code Section XI 1983 Edition, a number of piping supports are examined each outage to ensure that they can perform their design functions. In addition, an examination of installed piping systems is about to commence at the Cook Nuclear Plant to further verify acceptability. This latter review was discussed with NRC-Region III on May 18, 1989 and was documented in AEP:NRC:1060N of June 2, 1989. When these or similar reviews reveal discrepancies between the as-found and the as-designed condition, an evaluation of the acceptability and reportability of the condition is conducted.

EVALUATION OF AS-FOUND INSPECTION RESULTS

Operability

A set of criteria, included herein as an appendix, has been developed for the Cook Nuclear Plant to guide the decision making process when evaluating situations where as-found piping and/or support conditions differ from the as-designed condition. The criteria, which are specifically used to support the component/system functionality determination, are identical to those previously used for the Copes-Vulcan valve issue addressed in AEP:NRC:1084B of March 13, 1989. A determination of functionality is considered equivalent to a decision of operability as defined in the plant's Technical Specifications. When satisfied, these criteria provide reasonable assurance that the as-found piping system will fulfill its intended design function. Nevertheless, as noted in the appendix, modifications required to return the piping and/or support systems within the design basis limits shall be planned and installed before or during the next refueling outage.

If the as-found condition would result in exceeding the allowable appendix criteria, the piping and/or support system is considered non-functional. Any applicable technical specification action statements would be followed as appropriate. Ultimately, the piping and/or support system would be repaired, returning it to within the FSAR allowable stress criteria.

Reportability

Reportability guidelines have been developed, consistent with the appendix findings of system/component operability. Specifically, a determination that an as-found piping and/or support system discrepancy does not adversely affect functionality will be treated as an indication that the affected component(s) is not "in a condition that [is] outside the design basis of the plant". Under such conditions, no 10 CFR 50.72 or 10 CFR 50.73 reports will be made. On the other hand, if stresses in a safety-related piping and/or support system are calculated to exceed the limits of the appendix, the affected system will be declared non-functional, applicable technical specification actions will be taken and the condition will be reported under 10 CFR 50.72(b)(ii)(B) or 10 CFR 50.73(a)(ii)(B) as appropriate.

CONCLUSION

In summary, piping and/or piping supports found to exceed the FSAR allowable stresses for the design basis conditions will be returned to within the original design requirements. Systems that are discovered in this condition will be considered operable if the calculated stress levels meet the limits of the interim acceptance criteria described in the appendix. For this case, no 10 CFR 50.72 or 10 CFR 50.73 report will be made to the NRC. When calculated stress levels exceed the limits of the appendix, the system will be considered non-functional (i.e., inoperable), applicable technical specifications will be followed, and a report pursuant to 10 CFR 50.72(b)(ii)(B) or 10 CFR 50.73(a)(ii)(B) will be made.

APPENDIX FOR THE ATTACHMENT TO AEP:NRC:1100

INTERIM ACCEPTANCE CRITERIA FOR SAFETY RELATED PIPING SYSTEMS

1.0 INTRODUCTION:

Donald C. Cook FSAR (hereinafter called FSAR, Ref. 4) defines design bases for various seismic category piping system. These bases provide sufficient safety margin for continued plant operation such that the plant safety is not compromised. However, during an evaluation of a specific plant condition, identified via a problem report, if the limits of the design bases for piping and its support systems are exceeded, operability of the piping system during a DBE will be assured by meeting the limits of these interim criteria (Note: Expeditious processing and reportability requirements are defined in AEPSC Procedure GP 15.1). These criteria will provide justifications for continued plant operation. Modifications required to return the piping and support system within the design bases limits shall be made by the next refueling outage or sooner.

2.0 SCOPE:

These criteria are applicable to all safety related piping and associated support systems of the Donald C. Cook Nuclear Plant.

3.0 CRITERIA:

3.1 Piping System Acceptance Criteria

An analysis of the affected piping system shall be performed in accordance with ASME Section III NC-3600 Service Level D limits (Equation 9) for loading condition associated with Design Basis Earthquake (DBE). Increased damping values as permitted by Code Case N-411 shall be used for DBE analysis.

$S_p + S_w + S_D < 2.0\, S_y$ (Ref. 1)

Where S_p = Longitudinal Pressure Stress
S_w = Dead wt. stress plus stresses due to other Mechanical loads
S_D = Design Basis Earthquake Stress
S_y = Material Yield Stress at Design Temp.

3.2 Pipe Support Acceptance Criteria

3.2.1 In addition to the support loads developed in 3.1, thermal loads and other applicable displacement induced loads (e.g. seismic anchor movements) shall be included to define design loads for each component support associated with

the piping system. These supports will then be evaluated in accordance with the limits prescribed in Appendix F (Ref. 1) subsubarticle F-1370.

3.2.2 For anchor bolts, use a factor of safety of 2 against ultimate shear and tension values, instead of the values given in Structural Design Standard SDS-88 which has large factor of safety.

3.2.3 For catalog items, prorate load capacities based on a factor of safety of 2.

3.3 Basis for the Usage of Code Case N-411

The following address the five conditions specified in ASME Section III, Division 1, Regulatory Guide 1.84 Revision 25, "Design and Fabrication Code Class Acceptability," to provide a reasonable basis for the use of the Code Case N-411-1 in the Interim Acceptance Criteria evaluation.

(1) The code case damping values for piping systems are used consistently in the response spectra analysis.

(2) Cook Nuclear Plant floor response spectra were developed by averaging the results obtained from four scaled earthquake records as noted in our response to FSAR Question Q.5.73-2, Amendment 19 dated January 1972. The Cook Nuclear Plant Ground Response Spectra are conservative with respect to seismic history at the plant site.

(3) The evaluations are being performed for interim acceptance only and not for reconciliation work or support optimization.

(4) The code case damping values are not used in evaluations where supports that are designed to dissipate energy by yielding are used.

(5) The code case damping values are not used in systems where significant stress corrosion cracking has occurred. Stress corrosion cracking has not been identified in the systems recently evaluated.

4.0 CONCLUSION:
These criteria provide allowable values greater than the FSAR, however, these values are based on the current codes, standards and practices applicable to the design of Nuclear Power Plants.

5.0 REFERENCES:

1. American Society of Mechanical Engineers (ASME), Boiler and Pressure Vessel Codes, Section III, 1983 edition.

2. ASME code case N-411 (Approval date February 20, 1986)

3. "Steel Construction Manual", American Institute of Steel Construction

4. Donald C. Cook Nuclear Plant, updated Final Safety Analysis Report (FSAR)