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SUBJECT: Revises AEP:NRC:1100 to address comments made by NRR during 891213 telcon re allowable stresses for piping.

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AEP:NRC:1100A

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
Attn: Document Control Desk
Washington, D.C. 20555

Attn: T. E. Murley

February 16, 1990

Dear Dr. Murley:

AEP:NRC:1100 provided information relative to operability and reportability determinations 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 were addressed therein for situations where as-found piping and piping supports were found not to meet the original FSAR design requirements. The purpose of this letter is to revise AEP:NRC:1100 to address comments made by NRR during a December 13, 1989, conference call and to clarify the schedular commitments for modifying discrepant piping and/or support systems. Concurrence was obtained on changes addressing the latter issue during a conference call with J. Giitter (NRR Project Manager) and D. Danielson (Region III) on February 9, 1990.

Three changes of any consequence have been made to the Appendix for the Attachment to AEP:NRC:1100 and all changes are noted with a bar in the margin. In the first case, the last sentence in the introduction, which addresses modification schedules, has been deleted and a clarification has been added to the operability paragraph on page 1 of the attachment. This clarification specifically deals with the schedule for addressing discrepancies found during refueling outages. In the second case, paragraph 3.2.1 has been modified to indicate that the pipe support acceptance criteria for Cook Nuclear Plant is the same as that used on the Prairie Island Nuclear Plant. Finally, paragraph 3.3(2) has been clarified to indicate the specific spectra that are being used at the Cook Nuclear Plant in lieu of RG 1.60 floor response spectra. For convenience, the entire attachment to AEP:NRC:1100 (as revised) is attached hereto such that this letter supersedes AEP:NRC:1100 in its entirety.

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Other than the issues addressed above, NRR expressed no objections during the December 13, 1989, conference call to the concept of using the interim acceptance criteria for operability decision making for piping and piping supports. As such, Cook Nuclear Plant will use the revised attachment as the basis for our operability and reportability determinations.

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

ldp

Attachment

cc: D. H. Williams, Jr.
A. A. Blind - Bridgman
R. C. Callen
G. Charnoff
A. B. Davis - Region III
NRC Resident Inspector - Bridgman
NFEM Section Chief

ATTACHMENT TO AEP:NRC:1100A

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 in progress 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 similar 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, when discrepancies are discovered during unit operation which require modifications to return the piping and/or support systems within the design basis limits, the modifications shall be planned and installed before or during the next refueling outage. Reasonable efforts will be made to similarly disposition discrepancies found during a refueling outage before the end of that outage to the extent that the modifications can be planned and installed without adversely affecting the outage schedule. Region III will be informed when modifications cannot be completed prior to unit startup.

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:1100A

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 the design safety margin for continued plant operation. 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. These criteria provide sufficient safety margins to justify continued plant operation (Note: Expeditious processing and reportability requirements are defined in AEPSC Procedure GP 15.1). These criteria will provide justifications for continued plant operations.

2.0 SCOPE:

These criteria are applicable to all safety related piping and associated support systems for 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.

$$S_p + S_w + S_D \leq 2.0\ S_y \qquad \text{(Ref. 1)}$$

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 (Ref.-1 Appendices)

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 analyzed using the allowables listed below to meet operability requirements.

<u>Structural Steel</u>

Tension	F_t = 1.20 Sy but $\leq$ 0.70 Su
Bending	F_b = 1.20 Sy but $\leq$ 0.70 Su
Shear	F_v = 0.72 Sy but $\leq$ 0.42 Su
Compression	$F_a \leq F_t$ but not to exceed 2/3 Pcr
Combined Stress	For axial compression and bending <u>or</u> axial tension and bending, use AISC 1.6., (Ref. 3)
Web Cripling	$\leq$ 1.0 Sy
<u>Weld Stress</u>	$F_w \leq$ 0.42 Su (of weld material)
<u>Anchor Bolts</u>	Use Factor of Safety of 2 against ultimate tension and shear values.
<u>Snubbers</u>	
Hydraulic:	Load $\leq$ manufacturers one time load capacity or Level D limits. Movement $\leq$ total travel
<u>Springs</u>	Load within catalog range without bottoming out
<u>Struts</u>	FS = 2 and $\leq$ 2/3 Pcr
All remaining Catalog Items	Use manufacturers published faulted load rating. Where level D allowables are not given, and the factor of safety is specified in the catalog, use design allowables but with FS = 2. (Typical catalog FS=5, therefore use 2.5x catalog capacity)

Where: F_t = Allowable Tensile Stress

F_b = Allowable Bending Stress

F_v = Allowable Shear Stress

F_a = Allowable Axial Compressive Stress

F_w = Allowable Weld Stress

Pcr = Maximum Strength of Axially Loaded Compression Member

S_y = Specified Minimum Yield Strength at Temperature (See Note 1)

S_u = Specified Minimum tensile Strength at Temperature

FS = Factor of Safety

NOTE 1: Actual yield strength may be used where CMTR's are available for the material.

If a support fails using the linear elastic method, then a more refined analysis may be performed using plastic analysis techniques. The plastic analysis will follow the design rules of ASME Section III, Appendix F, (Ref. 1).

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 for the Containment Structure were developed by the application of a frequency dependent magnification factor to the Ground Response spectra for the site. Four earthquake records were used in the time history analysis in determining the magnification factor. The final magnification factor used was an envelope of the four earthquakes. The floor response spectra for the Auxiliary Building were developed by averaging the results obtained from a time history analysis of four scaled earthquake records.

The use of multiple time histories in the development of floor response spectra introduced additional conservatism into the resulting floor response spectra.

(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 <u>CONCLUSION</u>:
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 <u>REFERENCES</u>:

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)