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SUBJECT: Application for amends to licenses DPR-58 & DPR-74,request change re reactor coolant system pressure isolation valves.

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AEP:NRC:1180A

Donald C. Cook Nuclear Plant Units 1 and 2
Docket Nos. 50-315 and 50-316
License Nos. DPR-58 and DPR-74
ADDITIONAL INFORMATION FOR TECHNICAL SPECIFICATION CHANGE REQUEST
REACTOR COOLANT SYSTEM PRESSURE ISOLATION VALVES

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

Attn: W. T. Russell

October 7, 1994

Dear Mr. Russell:

Reference: AEP:NRC:1180, Technical Specification Change Request, November 15, 1993

In accord with requests from your staff, this letter and its attachments supply additional information to support and amend our previous application, contained in the referenced letter, for a technical specification (T/S) change for Donald C. Cook Nuclear Plant Units 1 and 2. Specifically, in the referenced letter we proposed and supported the deletion of the overly restrictive leakage limitation requirements for Reactor Coolant System Pressure Isolation Valves listed in T/S Table 3.4-0, while maintaining the more reasonable testing requirements for the subject valves in accordance with ASME Section XI. In conversations held on September 26, 1994, your staff indicated that there were no technical concerns with the subject request; however, they suggested a slightly different approach to achieving the same change. Instead of deleting the subject testing requirements from the T/Ss, the staff suggested and we agreed to replace the current T/S testing requirements with those required by ASME Section XI.

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Attachment 1 to this letter contains the additional information requested by your staff to support these modifications to our T/Ss. For your convenience, all of the previously supplied information in the referenced letter has been included in this attachment, with changes indicated in the right hand margin to address the NRC's requested approach as outlined above. The affected T/S pages, marked to show proposed revisions, are included in Attachment 2. Attachment 3 contains the affected T/S pages, with proposed changes typed in place.

We believe that the proposed changes will not result in (1) a significant change in the types of effluents or a significant increase in the amounts of any effluent that may be released offsite, or (2) a significant increase in individual or cumulative occupational radiation exposure.

The T/S changes originally proposed in the referenced letter were reviewed by the Plant Nuclear Safety Review Committee and the Nuclear Safety and Design Review Committee. The changes to that submittal included herein have been reviewed by the Plant Nuclear Safety Review Committee and will be reviewed by the Nuclear Safety and Design Review Committee at its next scheduled meeting.

In compliance with the requirements of 10 CFR 50.91(b)(1), copies of this letter and its attachments have been transmitted to the Michigan Public Service Commission and to the Michigan Department of Public Health.

This letter is submitted pursuant to 10 CFR 50.30(b) and, as such an oath statement is attached.

Sincerely,

E E Fitzpatrick

E. E. Fitzpatrick
Vice President

cad

Attachments

cc: A. A. Blind
G. Charnoff
J. B. Martin - Region III
NFEM Section Chief
NRC Resident Inspector - Bridgman
J. R. Padgett

STATE OF OHIO)
COUNTY OF FRANKLIN)

E. E. Fitzpatrick, being duly sworn, deposes and says that he is the Vice President of licensee Indiana Michigan Power Company, that he has read the forgoing Technical Specification Change Request Reactor Coolant System Pressure Isolation Valves and knows the contents thereof; and that said contents are true to the best of his knowledge and belief.

E E Fitzpatrick

Subscribed and sworn to before me this 7th

day of October, 19 94.

Rita D Hill
NOTARY PUBLIC

RITA D. HILL
NOTARY PUBLIC, STATE OF OHIO
MY COMMISSION EXPIRES 6-28-99

ATTACHMENT 1 TO AEP:NRC:1180A

DESCRIPTION OF PROPOSED CHANGES AND

10CFR50.92 SIGNIFICANT HAZARDS CONSIDERATION ANALYSIS

BACKGROUND

The Reactor Safety Study WASH-1400/NUREG 75/014 (Reference 1) analyzed a so-called Event V Sequence resulting in an intersystem loss-of-coolant accident (ISLOCA). It was concluded that the ISLOCA was a significant contributor to core damage frequency since the containment is bypassed and reactor coolant is released directly to the auxiliary building.

As a result of the above finding, the USNRC issued a letter, "LWR Primary Coolant System Pressure Isolation Valves," dated February 25, 1980, (Reference 2), requesting LWR licensees to provide the following information:

1. Describe the valve configuration at your plant and indicate if an Event V isolation valve configuration exists within the Class I boundary of the high pressure piping connection PCS piping to low pressure system piping; e.g., (1) two check valves in series, or (2) two check valves in series with a MOV;

2. If either of the above Event V configurations exists at your facility, indicate whether continuous surveillance or periodic tests are being accomplished on such valves to ensure integrity. Also indicate whether valves have been known, or found, to lack integrity; and

3. If either of the above Event V configurations exist at your facility, indicate whether plant procedures should be revised or if plant modifications should be made to increase reliability.

AEPSC responded to the above USNRC letter with letter AEP:NRC:0371, "Reactor Coolant System Pressure Isolation Valves," dated March 24, 1980, (Reference 3). In this letter, the USNRC was informed that the following valve configurations are used at Cook Nuclear Plant:

1) A minimum of three check valves in series,

2) Two check valves with a minimum of a closed motor operated valve in series,

3) Two check valves with a closed hand operated valve in series, and/or

4) A check valve with two closed air operated valves in series.

AEP:NRC:0371 concluded that: "Therefore no Event V configuration exists at the Cook Nuclear Plant. Consequently, the requests in Items 2 and 3 of Mr. Eisenhut's letter are not applicable to the Cook Nuclear Plant."

On April 20, 1981, the USNRC issued an "Order for Modification of Licenses Concerning Primary Coolant System Pressure Isolation Valves" for Cook Nuclear Plant (Reference 4). This order stated that "We have concluded that a WASH-1400 Event V valve configuration exists at your facility and that the corrective action as defined in the attached order is necessary." Attached to the order were the Technical Evaluation Report (TER) supporting the order and the new technical specifications (T/Ss) which, according to the order, ". . .will ensure public health and safety over the operating life of your facility." These new T/Ss were incorporated at that time into the operating licenses for Cook Nuclear Plant.

In 1987, the USNRC issued generic letter (GL) 87-06, "Periodic Verification of Leak Tight Integrity of Pressure Isolation Valves" (Reference 5). In our response to that GL, AEP:NRC:1041 (Reference 6), we stated that 12-SI-170L2, 12-SI-170L3, 12-RH-133, and 12-RH-134 are leak tested as per the surveillance requirements in the T/Ss. The proposed changes in this submittal, therefore, modify our response to GL 87-06.

Finally, in 1992, the USNRC issued information notice 92-36, "Intersystem LOCA Outside Containment" (Reference 7). This information notice listed eleven "Observed Plant Vulnerabilities to ISLOCA Precursors." A review of this information notice concluded that the concerns raised are adequately addressed by controls currently in place or planned at the Cook Nuclear Plant. The proposed changes to the T/Ss will not alter this conclusion because our response to the information notice was not based on existing T/Ss.

COOK NUCLEAR PLANT IPE

On May 1, 1992, AEPSC responded to generic letter 88-20 (Reference 8), in AEP:NRC:1082E, Individual Plant Examination (IPE) submittal (Reference 9), and provided further information on the analysis of the ISLOCA in Reference 10. (See the Enclosure to this Attachment: Donald C. Cook Nuclear Plant, Paths Considered as Possible Event V-Sequence LOCA, taken from Reference 10). In the determination of the ISLOCA initiating event frequency, leak testing of 12-SI-170L2, 12-SI-170L3, 12-RH-133, and 12-RH-134, as required by the surveillance requirements in the T/Ss, was addressed. The IPE for Cook Nuclear Plant concluded that the ISLOCA was the accident that contributed the least to the overall core damage frequency. The calculated ISLOCA initiating event frequency was approximately 6.70E-07, and the calculated probability of ISLOCA core damage was approximately 5.4E-08. The contribution to overall core damage frequency from an ISLOCA is less than 0.1% of the 6.26E-05 IPE calculated core damage frequency per reactor year. It is seen, therefore, that the contribution from an ISLOCA event at Cook Nuclear Plant is negligible.

A COMPARISON OF WASH-1400 VS. COOK NUCLEAR PLANT WITH RESPECT TO ISLOCA

The ISLOCA accident is described in Section 5.3.2.5 of WASH-1400, and the quantification of its core damage frequency, is found in Appendix V, Section 4.4 of WASH-1400.

The core damage frequency calculated in WASH-1400 is 4.00E-06/reactor year. WASH-1400 evaluated three pathways of two check valves in series, and assumed that the 600 psi Low Pressure Injection System (LPIS), once exposed to RCS pressure, would fail and create approximately a 6" effective diameter LOCA. No other accident initiation features or mitigating actions were modeled.

In the Cook Nuclear Plant IPE, nine different pathways for an ISLOCA event were analyzed. These pathways (see Enclosure to this Attachment), consisted of either three check valves in series, two check valves and a closed motor operated valve (MOV) in series, or two normally closed MOVs in series. Although WASH-1400 looked at check valve leak testing as a sensitivity analysis, the Cook Nuclear Plant IPE accounted for the in-place leak testing when calculating the probability of an ISLOCA occurring. In addition, the Cook Nuclear Plant IPE modeled mitigating actions for the ISLOCA event in the event tree. The

low pressure injection system piping was reviewed, and found to be capable of withstanding the full pressure of the reactor coolant system. Only hoop stresses were found to be significant, since after initiation of an ISLOCA the heatup rate of the piping would be relatively slow, minimizing thermal stresses. Thus, isolation of the LOCA pathway and potentially successful mitigating activities are possible. The IPE ISLOCA initiating event frequency was approximately an order of magnitude lower than the WASH-1400 core damage frequency (4.00E-06), and the IPE ISLOCA core damage frequency value was approximately two orders of magnitude lower than the WASH-1400 values.

Finally, to address the impact of removing the T/S requirements and only testing 12-SI-170L2, 12-SI-170L3, 12-RH-133, and 12-RH-134 at a refueling outage frequency, a sensitivity run was made in support of this submittal. Since the inservice inspection program will discover a ruptured valve, the additional T/S requirement was only credited with the ability to identify a stuck open valve. This credit was removed for this sensitivity run. It was found that the ISLOCA initiating event frequency, and ISLOCA probability of core damage would also increase by 5.4% to the mid 5.00E-08 range, and the overall core damage frequency would remain unchanged. Thus, the proposed T/S will essentially provide protection equivalent to the existing T/S from an ISLOCA.

IST PROGRAM

The subject check valves of this submittal are currently being tested in Mode 5, Cold Shutdown, as required by the T/Ss surveillance requirements prior to going to Mode 4, Hot Shutdown. The allowable leak rate cannot exceed 1 gpm. If the T/Ss are modified as requested, these valves would continue to be tested under the IST Program (ASME Boiler and Pressure Vessel Code, Section XI), on a refueling outage frequency, like their sister valves in the other loops in the Residual Heat Removal System. The acceptance criteria for the leak testing of these valves under the IST Program has been defined here as 0.5 gpm per nominal inch of valve size, consistent with NUREG-1431 (Reference 12).

DESCRIPTION OF PROPOSED TECHNICAL SPECIFICATION CHANGES

The proposed changes are listed below. They are identical for both units.

1. T/S 3.4.6.2 f. Replace in its entirety.

2. T/S 3.4.6.2 APPLICABILITY: Delete "**" and replace with a "*."

3. T/S 3.4.6.2 ACTION c. Delete "except when ... by 50% or more."

4. T/S 3.4.6.2 ACTION c, footnote *. Delete the footnote.

5. T/S 3.4.6.2 APPLICABILITY, footnote **. Change the "**" to "*."

6. T/S 4.4.6.2.2. Replace it with the following text:

   "Each reactor coolant system pressure isolation valve specified in Table 3.4-0 shall be demonstrated OPERABLE pursuant to Specification 4.0.5."

7. T/S 3/4.6.2, Table 3.4-0. Replace the Table in its entirety.

10CFR50.92 SIGNIFICANT HAZARDS CONSIDERATION ANALYSIS

Per 10 CFR 50.92, a proposed amendment to an operating license will not involve a significant hazards consideration if the proposed amendment satisfies the following three criteria:

1) Does not involve a significant increase in the probability or consequences of an accident previously analyzed,

2) does not create the possibility of a new or different kind of accident from an accident previously analyzed or evaluated, or

3) does not involve a significant reduction in a margin of safety.

Criterion 1

The ISLOCA is not one of the accidents previously analyzed in Chapter 14, Safety Analysis, of the Cook Nuclear Plant Updated Final Safety Analysis Report. Chapter 14 analyzes the large break LOCA in Section 14.3.1, and "loss of reactor coolant from small ruptured pipes or from cracks in large pipes which actuates the ECCS," or small break LOCA in Section 14.3.2. Therefore, the proposed T/Ss modifications to the Reactor Coolant System pressure isolation valves in Table 3.4-0, will not increase the probability or the consequences of the large break or the small break LOCAs previously analyzed for the Cook Nuclear Plant.

Criterion 2

The Reactor Coolant System pressure isolation valves in Table 3.4-0 of the T/Ss were added because WASH-1400 identified the ISLOCA as a significant contributor to core damage frequency. The proposed modifications of the subject valves Technical Specification surveillance requirements to rely on the testing requirements mandated by the In-Service Testing Program of ASME XI does not create the possibility of a new or different kind of accident from the large break or the small break LOCAs previously analyzed for the Cook Nuclear Plant.

Criterion 3

The proposed modifications to the Reactor Coolant System pressure isolation valves testing requirements in Table 3.4-0 of the T/Ss, will result in these valves only being tested on a refueling outage frequency as part of the ASME B&PV Code Section XI IST Program. This somewhat reduced testing frequency will result in a slight increase in the ISLOCA initiating event frequency, and ISLOCA contribution to core damage frequency by 5.4%, from lower 5.00E-08/reactor year to mid 5.00E-08/reactor year. This insignificant increase will not affect the overall core damage frequency of 6.26E-05/reactor year. Therefore, it is concluded that the proposed changes to the T/S requirements for the Reactor Coolant System pressure isolation valves in Table 3.4-0, will not result in a significant reduction in the margin of safety that exists at Cook Nuclear Plant to prevent an ISLOCA or to mitigate the consequences of an ISLOCA.

REFERENCES

1. WASH-1400/NUREG 75/014, Reactor Safety Study: An Assessment of Accident Risks in U.S. Commercial Nuclear Plants, USNRC, October 1975.

2. USNRC Letter to all LWR Licensees: LWR Primary Coolant System Pressure Isolation Valves, February 25, 1980.

3. AEP:NRC:0371, Reactor Coolant System Pressure Isolation Valves, March 24, 1980.

4. Letter, S. A. Varga, USNRC to J. Dolan, Indiana and Michigan Electric Company, Order for Modification of Licenses Concerning Primary Coolant System Pressure Isolation Valves, April 20, 1981.

5. USNRC Generic Letter (GL) 87-06, " Periodic Verification of Leak Tight Integrity of Pressure Isolation Valves," March 13, 1987.

6. AEP:NRC:1041, Generic Letter 87-06, Periodic Verification of Leak Tight Integrity of Pressure Isolation Valves, November 12, 1987.

7. USNRC Information Notice 92-36, "Intersystem LOCA Outside Containment," May, 7, 1992.

8. Initiation of the Individual Plant Examination for Severe Accident Vulnerabilities - 10CFR50.54(f), Generic Letter No. 88-20, November 23, 1988.

9. AEP:NRC:1082E, Individual Plant Examination Submittal, Response to Generic Letter 88-20, May 1, 1992.

10. AEP:NRC:1082F, Individual Plant Examination Response to NRC Questions, February 24, 1993.

11. AEP:NRC:1180, Technical Specification Change Request, Reactor Coolant System Pressure Isolation Valves, November 15, 1993.

12. NUREG-1431, Standard Technical Specifications for Westinghouse Plants, September 1992.

ATTACHMENT 2 TO AEP:NRC:1180A

MARKED-UP TECHNICAL SPECIFICATION PAGES