Docket Files

Docket No. 50-298

SEP 16 1977

Nebraska Public Power District
ATTN: Mr. J. M. Pilant, Director
Licensing and Quality Assurance
P. O. Box 499
Columbus, Nebraska 68601

Gentlemen:

The Commission has issued the enclosed Amendment No. 38 to Facility Operating License No. DPR-46 for the Cooper Nuclear Station. This amendment consists of changes to the Technical Specifications in response to your requests for exemptions from the requirements of 10 CFR Part 50, Appendix J, dated September 10, 1975 and January 4, 1977. These exemptions are:

1. The Main Steam Isolation Valves (MSIV's) would be tested at 29 psig (P_t) instead of the required 59 psig (Pa).

2. The personnel air lock door would be tested at intervals no longer than one year at 58 psig (Pa) and at 3 psig after each opening during the one year interval between the 58 psig tests.

3. The void between the bellows located in the main steam line and feedwater line penetrations would be tested at 5 pisg instead of the required 58 pisg (Pa).

4. The feedwater check valves would be tested with water rather than air or nitrogen.

5. The test interval for Type C valve leak testing would be extended until the September 1977 refueling outage.

Exemptions 1 through 4 above were requested in your September 10, 1975 letter; exemption 5, in your January 4, 1977 letter. Our Safety Evaluation of the requested exemptions is enclosed. Based on our evaluation, we have determined that items 1, 3 and 5 above are acceptable. Therefore, pursuant to 10 CFR Section 50.12, the exemptions from the requirements of Appendix J of 10 CFR Part 50

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SEP 16 1977

for items 1, 3 and 5 are authorized by law and will not endanger life or property or the common defense and security and are otherwise in the public interest. Items 2, and 4 were the subject of our request for additional information dated February 17, 1977. By letter dated April 4, 1977, you provided additional information concerning item 2 describing the correlation which relates personnel air lock leak rates at 58 psig to leak rates at a test pressure of 3 psig. Our review of this correlation is included in the enclosed evaluation. We have determined that additional information is required before we can complete our review of item 2. Concerning item 4, you indicated that efforts were continuing to provide the additional information we requested on feedwater check valves and high pressure coolant injection and reactor core isolation cooling-to feedwater check valve testing. We are presently awaiting your response. Also, as discussed in the enclosed evaluation, you are requested to submit, within 30 days of receipt of this letter, your acceptance criterion which relates bellows leakage rates at 5 psig test pressure to the leakage rate which would be experienced at 58 psig pressure.

Copies of the Safety Evaluation and the Notice of Issuance are also enclosed.

Sincerely,

Original Signed by
V. Stello

Victor Stello, Jr., Director
Division of Operating Reactors
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 38 to DPR-46
2. Safety Evaluation
3. Notice

cc: See next page

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cc w/enclosure:
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Mr. William Siebert, Commissioner
Nemaha County Board of Commissioners
Nebraska County Courtroom
Auburn, Nebraska 68305

cc w/enclosures and copy of NPPS's
filings dtd. 9/10/75, 1/4/77 & 4/4/77:

Director, Department of Environmental Control
Executive Bldg. 2nd Floor
Lincoln, Nebraska 68509

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

NEBRASKA PUBLIC POWER DISTRICT

DOCKET NO. 50-298

COOPER NUCLEAR STATION

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 38
License No. DPR-46

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The applications for amendment by Nebraska Public Power District (the licensee) dated September 10, 1975 and January 4, 1977, as supplemented by letter dated April 4, 1977, comply with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I:

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by a change to the Technical Specifications as indicated in the attachment to this license amendment and paragraph 2.C(2) of Facility License No. DPR-46 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 38 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Don K. Davis, Acting Chief
Operating Reactors Branch #2
Division of Operating Reactors

Attachment:
Changes to the Technical
Specifications

Date of Issuance: September 16, 1977

ATTACHMENT TO LICENSE AMENDMENT NO. 38

FACILITY OPERATING LICENSE NO. DPR-46

DOCKET NO. 50-298

Replace existing pages 162 and 180 of the Appendix A Technical Specifications with the attached revised pages bearing the same numbers. Changed areas on the revised pages are identified by a marginal line.

3.7.A (cont'd)

4.7.A (cont'd)

repeated provided locally measured leakage reductions, achieved by repairs, reduced the containment's overall measured leakage rate sufficiently to meet the acceptance criteria.

f. With the exception of main steam isolation valves and main steam line and feedwater line bellows, (see below) local leak rate tests (LLRT's) shall be performed on the primary containment testable penetrations and isolation valves at a pressure of 58 psig during each reactor shutdown for refueling but in no case at intervals greater than two years. Bolted double-gasket seals shall be tested after each opening and during each reactor shutdown for refueling but in no case at intervals greater than two years.

* The main steam isolation valves (MSIV's) shall be tested a pressure of 29 psig. If a total leakage rate of 11.5 scf/hr for any one MSIV is exceeded, repair and retest shall be performed to correct the condition.

* Main steam line and feedwater line expansion bellows shall be tested at a pressure of 5 psig.

g. Continuous Leak Rate Monitor

When the primary containment is inerted, the containment shall be continuously monitored for gross leakage by review of the inerting system makeup requirements. This monitoring system may be taken out of service for maintenance but shall be returned to service as soon as practicable.

h. Drywell Surfaces

The interior surfaces of the drywell and torus shall be visually inspected each operating cycle for evidence of

* Exemption to Appendix J of 10 CFR 50.

trends. Whenever a bolted double-gasketed penetration is broken and remade, the space between the gaskets is pressurized to determine that the seals are performing properly. It is expected that the majority of the leakage from valves,penetrations and seals would be into the reactor building. However, it is possible that leakage into other parts of the facility could occur. Such leakage paths that may affect significantly the consequences of accidents are to be minimized.

Table 3.7.4 identifies certain isolation valves that are tested by pressurizing the volume between the inboard and outboard isolation valves. This results in conservative test results since the inboard valve, if a globe valve, will be tested such that the test pressure is tending to lift the globe off its seat. Additionally, the measured leak rate for such a test is conservatively assigned to both of the valves equally and not divided between the two.

The main steam and feedwater testable penetrations consist of a double layered metal bellows. The inboard high pressure side of the bellows is subjected to drywell pressure. Therefore the bellows is tested in its entirety when the drywell is tested. The bellows layers are tested for the integrity of both layers by pressurizing the void between the layers to 5 psig. Any higher pressure could cause permanent deformation, damage and possible reptures of the bellows.

The primary containment pre-operational test pressures are based upon the calculated primary containment pressure response in the event of a loss-of-coolant accident. The peak drywell pressure would be about 58 psig which would rapidly reduce to 29 psig following the pipe break. Following the pipe break, the suppression chamber pressure rises to 27 psig, equalizes with drywell pressure and therefore rapidly decays with the drywell pressure decay. The design pressure of the drywell and suppression chamber is 56 psig. Based on the calculated containment pressure response discussed above, the primary containment pre-operational test pressure was chosen. Also, based on the primary containment pressure response and the fact that the drywell and suppression chamber function as a unit, the primary containment will be tested as a unit rather than the individual components separately.

The design basis loss-of-coolant accident was evaluated at the primary containment maximum allowable accident leak rate of 0.635%/day at 58 psig. Calculations made by the NRC staff with leak rate and a standby gas treatment system filter efficiency of 90% for halogens and assuming the fission product release fractions stated in NRC Regulatory Guide 1.3, show that the maximum total whole body passing cloud dose is about 1.0 REM and the maximum total thyroid dose is about 12 REM at 1100 meters from the stack over an exposure duration of two hours. The resultant doses reported are the maximum that would be expected in the unlikely event of a design basis loss-of-coolant accident. These doses are also based on the assumption of no holdup in the secondary containment resulting in a direct release of fission products from the primary containment through the filters and stack to the environs. Therefore, the specified primary containment leak rate and filter efficiency are conservative and provide margin between expected off-site doses and 10 CFR 100 guidelines.

The water in the suppression chamber is used only for cooling in the event of an accident; i.e., it is not used for normal operation; therefore, a daily

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 38 TO FACILITY OPERATING LICENSE NO. DPR-46

AND

EXEMPTIONS TO 10 CFR PART 50, APPENDIX J

NEBRASKA PUBLIC POWER DISTRICT

COOPER NUCLEAR STATION

DOCKET NO. 50-298

INTRODUCTION

By letters dated September 10, 1975 and January 4, 1977, Nebraska Public Power District (NPPD, the license) requested certain exemptions from the requirements of Appendix J to 10 CFR Part 50 for the Cooper Nuclear Station (CNS). The requested exemptions are:

1. The Main Steam Isolation Valves (MSIV's) would be tested at 29 psig (P_t) instead of the required 58 psig (Pa).

2. The personnel air lock door would be tested at intervals no longer than one year at 58 psig (Pa) and at 3 psig after each opening during the one year interval between the 58 psig tests.

3. The void between the bellows located in the main steam line and feedwater line penetrations would be tested at 5 psig instead of the required 58 psig (Pa).

4. The feedwater check valves would be tested with water rather than air or nitrogen.

5. The test interval for Type C valve leak testing would be extended until the September 1977 refueling outage.

Additional information concerning personnel air lock door testing was provided in the licensee's April 4, 1977 letter. NPPD is currently evaluating our request for additional information on feedwater check valve testing.

BACKGROUND

Appendix J to 10 CFR 50 was published on February 14, 1973. Since many operating nuclear plants had either received an operating license or were in advanced stages of design or construction at that time, some plants may not now be in full compliance with the requirements of this regulation. Therefore, beginning in August 1975, requests to establish the degree of compliance with the requirements of Appendix J were made of each licensee. Following the initial responses to these requests, we developed positions which would provide assurance that the objective of the testing program were satisfied. These NRC staff positions have since been applied in our review of reports filed by NPPD and the results are reflected in the following evaluation.

EVALUATION

Main Steam Isolation Valves

Paragraph III.C.2 of Appendix J requires that the containment isolation valves be locally leak tested (Type C test) at the peak calculated containment pressure, Pa.

Each main steam line at CNS contains two containment isolation valves, called main steam isolation valves (MSIV) in series, one inboard valve and one outboard valve with respect to the reactor containment. These valves are designed to provide a leak tight seal in the main steam lines when the valves are closed and pressure is applied to the reactor vessel side of the valve. Therefore, if the MSIV's are shut in conjunction with containment isolation, reactor vessel pressure or containment pressure, in the event of a loss of coolant accident, on the valve disc will help achieve a tight seal between the valve disc and seat.

The current procedure for leak testing MSIV's at CNS requires pressurizing the main steam pipe volume between the inboard and outboard valves. The procedure pressurizes the outboard valve in the direction for which it was designed to seal. However, the procedure pressurizes the inboard valve in the reverse direction and, therefore, tends to open the valve by lifting the disc off of its seat. This results in greater leakage through the inboard valve than would be experienced if the valve were pressurized in the proper direction. The effect of reverse loading on the inboard valve was considered when the original test pressure of 29 psig was established and incorporated into the CNS technical specifications.

We have determined that since the test procedure used at CNS results in reverse loading of the inboard MSIV and therefore in a greater measured leak rate, testing MSIV's at 29 psig results in a conservative determination of leak rate through the valves and is acceptable.

Personnel Airlocks

Paragraphs II.D.2 and III.B.2 of Appendix J require that reactor containment airlocks be leak tested at the peak calculated accident pressure (Pa) at six month intervals. Further, should the airlocks be opened during such intervals, the airlocks will be tested after each opening.

The objective of the airlock leak testing requirements are: (1) that the six month test will provide an integrated leakage rate for the entire assembly, including electrical and mechanical penetrations, the airlock cylinder, hinge assemblies, welded connections, and other potential leakage paths; and (2) that the "after each opening" test would provide a means of assuring that the door seals had not been damaged or seated improperly during airlock use.

The airlock design for the Cooper Station includes an inner and an outer door, both of which seat with containment pressure. Pressurizing the airlock to Pa lifts the inner airlock door off its seat which results in excessive leakage into the containment. This condition does not reflect the post-accident condition of the airlock. To leak test the airlock at Pa, a strongback must be installed, inside the containment, on the inner airlock door. The strongback prevents the inner door from lifting off its seat. To conduct the test, the airlock doors must be opened both before and after the test to install and remove the strongback. Consequently, NPPD has requested an exemption to allow testing of the airlock at a reduced pressure (3 psig) which would not require the use of the strongback.

We agree with NPPD's proposed approach for the "after opening" airlock test. Conducting the tests at Pa would necessitate breaking the door seals to remove the strongback, thus defeating the purpose of the tests. Also, the 3 psig test provided an acceptable test of the integrity of the air lock door seals.

However, before we can conclude our review of this exemption request, we require an acceptance criterion for the reduced pressure tests which correlates the leakage rate at 3 psig to the leakage rate which would be experienced at 58 psig. By letter dated April 4, 1977, NPPD provided a correlation which equates the 3 psig leak rate to the product of the 58 psig leak rate times the square root of the quantity three divided by fifty-eight. However, NPPD provided no technical basis for this equation. When NPPD provides such a technical basis, we will continue our evaluation of the 3 psig airlock test procedure.

NPPD has also requested an exemption to allow conducting the airlock integrated leak test at one year intervals rather than at the six month intervals required by Appendix J. Insufficient justification was provided by NPPD in support of a year test interval. Accordingly, based on the lack of justification, we find this proposed exemption unacceptable.

Main Steam Line and Feedwater Line Bellows

Paragraph III.B.2 of Appendix J requires that local leak test on containment penetrations (Type B) be performed at the peak calculated containment pressure. NPPD has requested an exemption from the Type B test pressure for the expansion bellows in the main steam lines and feedwater lines. The main steam and feedwater testable penetrations consist of double layered metal bellows which are currently locally leak tested by pressurizing the annulus between the double layers to 5 psig rather than Pa. The design of the bellows does not permit local testing at a higher pressure. The bellows are exposed to the drywell atmosphere and are, therefore, tested as part of the containment integrated leakage rate test. In addition, the bellows are a static system: there are no moving parts or active components.

Based on these considerations, we conclude that the proposed exemption for the bellows test pressure is acceptable. However, the NRC staff will require NPPD to provide an acceptance criterion similar to that described above for airlock testing to relate bellows leakage rates at 5 psig to the leakage rate which would be experienced at 58 psig.

Feedwater Check Valves

Paragraph III.C.2 of Appendix J requires that isolation valves be locally leak tested with air or nitrogen.

For the Cooper Nuclear Station the feedwater system was not originally designed to permit local leak testing with air or nitrogen. NPPD therefore, requested an exemption for the feedwater check valves which are currently leak tested with water instead of air or nitrogen as required by Appendix J.

This paragraph of Appendix J required a simulation of the condition of the system following a postulated loss-of-coolant accident (LOCA) where the leakage barriers (e.g., valves, gaskets, and seals) may be exposed to the containment atmosphere. There are a number of liquid-filled systems, however, that are designed to remain intact following a LOCA. These liquid filled systems include the emergency core cooling system and the containment heat removal systems. For those systems that are designed to engineered safety feature criteria and for which there is assurance that they will remain filled with liquid following a LOCA, the liquid leakage rates should be distinguished from containment atmosphere leakage rates. These systems can be hydrostatically tested to demonstrate that the fluid inventory is sufficient to maintain a water seal during and following the postulated accident. A liquid leakage limit can be assigned for these systems. This criterion is similar in concept to a valve seal-water system criterion and will provide equivalent isolation protection. For this type of testing, radiological analyses should be performed to demonstrate that the liquid leakage limits do not result in significant doses so that the total accident dose would not be greater than the 10 CFR Part 100 guidelines.

In its September 10, 1975 submittal, NPPD requested an exemption from the requirements of Appendix J to permit local leak rate testing of the feedwater check valves using water as a test medium rather than air or nitrogen.

We have determined that the proposed hydrostatic testing would be acceptable if it can be shown that the valves will indeed be filled with water during and after a loss-of-coolant (LOCA) accident and that the liquid leakage will not result in additional radiological doses such that the total accident dose would be greater than the 10 CFR Part 100 guidelines.

By letter dated February 17, 1977, we requested that NPPD demonstrate that the feedwater, and HPCI and RCIC-to-feedwater, check valves would remain full of water following a postulated LOCA and that the fission products entrained in liquid leakage will not result in total radiological doses exceeding 10 CFR Part 100 guidelines. Alternatively, we asked NPPD to either develop a correlation to an equivalent air leakage or modify the systems to permit leak testing with air or nitrogen.

When the licensee provides the requested information, we will continue our evaluation of this exemption request.

Type C Test Interval

Paragraph III.D.3 of Appendix J requires that Type C tests be performed during each reactor shutdown for refueling but in no case at intervals greater than two years.

The Technical Specifications for Cooper Nuclear Station (CNS) specify that Type C tests shall be performed each operating cycle but in no case at intervals greater than two years.

As stated in NPPD's January 4, 1977 letter, during the refueling of CNS which occurred between September 18 and November 10, 1976, Type C testing was not performed because it had been performed during a maintenance outage in October 1975 and was scheduled to be performed during the next refueling outage in September 1977 which is within the 24 month period required by both 10 CFR 50 Appendix J and the CNS Technical Specifications. In discussions subsequent to the startup of CNS following the October 1976 refueling outage, we indicated that NPPD's interpretation of the Type C test frequency was incorrect; and the Type C testing should have been performed during that refueling outage. Therefore, in a letter dated January 4, 1977, NPPD requested an exemption to permit the performance of Type C tests during the refueling outage scheduled for September 1977.

The frequency for Type C tests specified in Appendix J was selected to coincide with the refueling outage which is normally not more than two years after the first refueling. This is because a shutdown and cooldown solely for these tests would result in an unnecessary plant thermal cycle. Such thermal cycles are limited by design to minimize the effects of thermal and mechanical stresses on plant systems. Therefore, it is desirable to conduct these tests during some other scheduled shutdown and cooldown event, such as refueling, but in no case at intervals greater than two years.

Because approval of this exemption would not result in an unnecessary thermal cycle or exceeding the maximum specified test interval and because approval would result in placing CNS back on the inspection schedule specified in Appendix J, we conclude that the proposed one-time exemption is acceptable. However, to prevent future misinterpretation, technical specification 4.7.A.f will be changed to bring it into verbatim agreement with paragraph III.D.3 of Appendix J.

Environmental Consideration

We have determined that the amendment does not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. Having made this determination, we have further concluded that the amendment involves an action which is insignificant from the standpoint of environmental impact and pursuant to 10 CFR §51.5(d)(4) that an environmental impact statement, or negative declaration and environemntal impact appraisal need not be prepared in connection with the issuance of this amendment.

Conclusion

Based on the foregoing, we have determined that, pursuant to 10 CFR Section 50.12, specific exemptions for MSIV testing, steam and feedwater line bellows testing, and Type C test interval, as discussed above, can be granted without endangering life or property, or the common defense and security, and are otherwise in the public interest.

We have also concluded, based on the considerations discussed above, that: (1) because the amendment does not involve a significant increase in the probability or consequences of accidents previously considered and does not involve a significant decrease in a safety margin, the amendment does not involve a significant hazards consideration, (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (3) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Date: September 16, 1977

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-298

NEBRASKA PUBLIC POWER DISTRICT

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 38 to Operating License No. DPR-46, issued to the Nebraska Public Power District (the licensee), which revised Technical Specifications for operation of the Cooper Nuclear Station (the facility) located in Nemaha County, Nebraska. The amendment is effective as of its date of issuance.

The amendment consists of Technical Specification changes to incorporate approved exemptions from certain requirements of 10 CFR Part 50 Appendix J regarding main steam isolation valve leak rate testing, main steam line and feedwater line bellows leak rate testing, and extension of the test interval for Type C leak rate testing for the Cooper Nuclear Station.

The applications for amendment comply with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendment. Prior public notice of this amendment was not required since the amendment does not involve a significant hazards consideration.

The Commission has determined that the issuance of this amendment will not result in any significant environmental impact and that pursuant to 10 CFR §51.5(d)(4) an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with issuance of this amendment.

For further details with respect to this action, see (1) the requests for exemption dated September 10, 1975 and January 4, 1977, as supplemented by letter dated April 4, 1977, (2) Amendment No. 38 to License No. DPR-46, and (3) the Commission's concurrently issued Safety Evaluation. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N. W., Washington, D. C. and at the Auburn Public Library, 118 15th Street, Auburn, Nebraska 68305. A single copy of items (2) and (3) may be obtained upon request addressed to the United States Nuclear Regulatory Commission, Washington, D. C. 20555, Attention: Director, Division of Operating Reactors.

Dated at Bethesda, Maryland, this 16th day of September, 1977.

FOR THE NUCLEAR REGULATORY COMMISSION

Don K. Davis, Acting Chief
Operating Reactors Branch #2
Division of Operating Reactors