Nebraska Public Power District
Nebraska's Energy Leader

50.55a(f)(5)(iii)
50.90

NLS2001022
April 12, 2001

U.S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, D.C. 20555-0001

Gentlemen:

Subject: Proposed License Amendment To Revise The Testing Frequency of SR 3.6.1.3.8 for Actuation of Excess Flow Check Valves (EFCVs)
Proposed Inservice Testing (IST) Relief Request Related to EFCV Testing Frequency
Cooper Nuclear Station, NRC Docket 50-298, DPR-46

Reference: 1. Boiling Water Reactor (BWR) Owners' Group Report, NEDO-32977-A, "Excess Flow Check Valve Testing Relaxation," dated June 2000.

In accordance with provisions of 10 CFR 50.4, 50.55a(f)(5)(iii), and 50.90 the Nebraska Public Power District (District) hereby submits a request for an amendment to License DPR-46 to change the Cooper Nuclear Station (CNS) Technical Specifications (TS) and requests the approval of Inservice Testing (IST) relief request number RV-10. The proposed changes will modify TS Surveillance Requirement (SR) 3.6.1.3.8 to relax the SR frequency by allowing a representative sample of Excess Flow Check Valves (EFCVs) to be tested every 18 months, such that each EFCV will be tested once every ten years. The IST relief request is being submitted to modify the IST Program to be consistent with the proposed TS change.

This amendment request and associated IST relief request are consistent with an approved generic change to the Standard Technical Specifications, TSTF-334, Revision 2, and the Nuclear Regulatory Commission (NRC) Safety Evaluation of the Boiling Water Reactor (BWR) Owners' Group Topical Report, B21-00658-01, which are both contained in NEDO-32977-A, "Excess Flow Check Valve Testing Relaxation," dated June 2000 (Reference 1).

Attachment 1 contains the IST relief request revision (RV-10, revision 1) for the third 10-year interval for NRC approval. Attachment 2 contains the description of the TS change, basis for the change, attendant 10 CFR 50.92 no significant hazards consideration evaluation, and 10 CFR 51.22 environmental impact evaluation. Attachment 3 identifies the specific changes to the current CNS TS and Bases (provided for information) on marked up pages. Attachment 4 contains the final, clean versions of the affected TS and Bases pages.

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These proposed changes have been reviewed by the necessary safety review committees (Station Operations Review Committee and Safety Review and Audit Board) and incorporate all amendments to the CNS Facility Operating License through Amendment 185 issued March 13, 2001. By copy of this letter and attachments the appropriate State of Nebraska official is being notified in accordance with 10 CFR 50.91(b)(1). Copies to the Region IV Office and the CNS Resident Inspector are also being sent in accordance with 10 CFR 50.4(b)(1).

The District requests NRC approval of the proposed TS change and approval of the relief request by September 30, 2001, with a 30-day implementation time to support the upcoming refueling outage (RO20) scheduled to start on November 2, 2001.

Should you have any questions concerning this matter, please contact Mr. Michael Boyce at (402) 825-5100.

Sincerely,

John H. Swailes
Vice President of Nuclear Energy

/dw
Attachments

cc: Regional Administrator w/attachments
USNRC - Region IV

Senior Project Manager w/attachments
USNRC - NRR Project Directorate IV-1

Senior Resident Inspector w/attachments
USNRC

Nebraska Health and Human Services w/attachments
Department of Regulation and Licensure

NPG Distribution w/o attachments

Records w/ attachments

STATE OF NEBRASKA)
)
NEMAHA COUNTY)

John H. Swailes, being first duly sworn, deposes and says that he is an authorized representative of the Nebraska Public Power District, a public corporation and political subdivision of the State of Nebraska; that he is duly authorized to submit this correspondence on behalf of Nebraska Public Power District; and that the statements contained herein are true to the best of his knowledge and belief.

John H. Swailes

Subscribed in my presence and sworn to before me this 11 day of April, 2001.

William D. Shanks
NOTARY PUBLIC

STATE OF NEBRASKA NOTARY
WILLIAM D. SHANKS
MY COMMISSION EXPIRES
November 27, 2003

COOPER NUCLEAR STATION
THIRD INTERVAL
INSERVICE TESTING PROGRAM

RELIEF REQUEST RV-10, Revision 1

VALVES: Excess Flow Check Valves

CLASS: 1 **CATEGORY:** A/C

FUNCTION: Excess flow check valves (EFCVs) have an active safety function in the closed position to provide containment isolation and are reactor coolant system pressure boundary isolation valves. They prevent excess flow of reactor water should an instrument line break occur outside containment. EFCVs are installed in instrument lines which connect to the reactor vessel. Each line contains a 1/4 inch restriction orifice inside the containment in order to limit flow leakage in the event the instrument line breaks.

These instrument lines are in compliance with the requirements of Safety Guide 11, Supplement 1, except there is no remote indication of the EFCV. This design ensures that in the event of a postulated piping or component failure (1) leakage is reduced to the maximum extent practical, (2) the rate and extent of coolant loss is within the capability of the reactor coolant make up system, (3) the integrity and performance of the secondary containment and associated safety systems will be maintained, and (4) the potential offsite exposure will be substantially less than 10CFR100 guidelines.

REQUIRED TEST: OMa Part 10, 4.3.2.1 requires check valves to be individually exercised nominally every 3 months.

BASIS FOR RELIEF: Uninterrupted function of these valves is essential for the safe operation of the plant. Quarterly testing in accordance with Section XI would interrupt instruments required for safety-system actuation, reactor shutdown, or sensing accident conditions. In addition, these valves cannot be exercised during cold shutdown because removal of multiple instruments from service could prevent or interrupt the operation of systems required for decay heat removal. Testing this frequently could jeopardize the safety of the reactor. EFCVs are reliable devices. The major components consist of a poppet and spring. The spring holds the poppet open only under static conditions. The valve will close upon sufficient differential pressure across the poppet.

EFCVs have been proven to be highly reliable at Cooper Nuclear Station (CNS) and throughout the industry. CNS testing results of EFCVs from the ten-year period of 1991 through 2000 were evaluated and revealed zero closure failures out of 476 tests. General Electric (GE) Nuclear Energy Topical Report B21-00658-01, "Excess Flow Check Valve Testing Relaxation," dated November 1998 (SER to BWR Owners Group from NRC, dated March 14, 2000, subject: Safety Evaluation of General Electric Nuclear Energy Topical Report B21-00658-01, "Excess Flow Check Valve Testing Relaxation" [TAC NOS. MA7884 And M84809]), also provides evidence of EFCV reliability. The Topical Report evaluated EFCV testing history from 12 BWR plants and reported a low failure rate (i.e., 11 failures in 12,424.5 valve-years of service or one failure in 1129 valve-years of service).

The proposed alternate test involves testing in accordance with CNS Technical Specification (TS) Surveillance Requirement (SR) 3.6.1.3.8. A representative sample of EFCVs will be functionally tested every 18 months. The SR 3.6.1.3.8 test frequency is adequate to maintain a high degree of reliability and availability, and provides an acceptable level of quality and safety. Justification for the adequacy of this test frequency is contained in license amendment request letter NLS2001022, Attachment 2, and is based on information contained in the above referenced SER.

ALTERNATE TEST: In lieu of Section XI quarterly functional testing, a representative sample of EFCVs will be functionally tested every 18 months such that each EFCV will be tested at least once each ten year interval.

PROPOSED LICENSE AMENDMENT TO REVISE THE TESTING FREQUENCY OF SR 3.6.1.3.8 FOR ACTUATION OF EXCESS FLOW CHECK VALVEs (EFCVs)

Cooper Nuclear Station, NRC Docket 50-298, DPR-46

Revised Pages
3.6-14

1.0 Introduction

Cooper Nuclear Station (CNS) Technical Specifications (TS) Surveillance Requirement (SR) 3.6.1.3.8 currently requires verification that each reactor instrumentation line excess flow check valve (EFCV) actuates to the isolation position on an actual or simulated instrument line break every 18 months. The requested change would revise the surveillance frequency from testing each excess flow check valve once per 18 months to testing "a representative sample" of EFCVs once per 18 months.

2.0 Discussion

The EFCVs, which are the subject of the requested revised testing frequency, are associated with instrumentation piping connecting to the Reactor Coolant Pressure Boundary, which leaves Primary Containment and is dead-ended at instruments located in the Reactor Building. These instrument lines which connect directly to the Reactor Pressure Vessel or to systems containing fluids connecting to the Reactor Pressure Vessel are provided with a manual shutoff valve and an automatic EFCV outside Primary Containment. In addition, these lines are provided with 1/4 inch opening restriction orifices inside Primary Containment in order to restrict flow in the event of an instrument line break. This isolation system design is in compliance with the requirements of Safety Guide 11, Supplement 1 (except that there is no remote indication of excess flow check valve position).

3.0 Description

The proposed change relaxes the surveillance requirement frequency by allowing a representative sample of EFCVs to be tested every 18 months. This representative sample consists of an approximately equal number of EFCVs that will be tested each 18 months (nominal), such that each EFCV will be tested at least once every ten years (nominal).

The Boiling Water Reactor (BWR) Owners' Group has issued a topical report (Reference 1) that provides a generic technical basis for this request. This NRC approved topical report provides justification for the relaxation in the SR frequency as described above. The report demonstrates the high degree of EFCV reliability and the low consequences of

an EFCV failure. CNS has evaluated the plant-specific application of this topical report, and provided that evaluation in the Justification section to follow. Similar Technical Specification amendments have been submitted and approved for several BWRs. Furthermore, the format and content of these proposed Technical Specification and Bases changes, is consistent with the NRC approved generic change to the Improved Standard Technical Specifications, NUREG-1433, reflected in TSTF-334, Revision 2 (Reference 2).

4.0 Justification

The proposed change is being requested to minimize personnel radiation exposure during refueling outages, cut down on outage critical path time, and increase the availability of instrumentation during outages without significantly impacting the risk to the general public.

The BWR Owners' Group Topical Report provided detailed information about EFCV surveillance testing at 12 BWR plants. This testing history indicated that there is generally a low failure rate in EFCV testing industry-wide (11 failures reported in 12,424.5 valve-years of service, or one failure in 1129 valve-years of service). For CNS, specific EFCV testing reliability was evaluated based on the test results from the ten year period of 1991 through 2000. There have been no failures to close associated with EFCV isolation testing at CNS in 476 tests (approximately 680 valve-years of service) since 1991. Thus, the EFCVs at CNS have been very reliable performers.

The acceptance criterion used from 1974 to 1990 was found to be excessively restrictive (>0.2 gpm and <0.7 gpm), which was the cause of most failures prior to 1991. In 1991, the acceptance criterion was revised to reflect limits appropriate to ASME Inservice Testing limits (>0.01 and <1.50 gpm) to ensure appropriate corrective actions are taken for failures of the magnitude to be a safety concern.

The NRC safety evaluation report associated with the BWR Owners' Group Topical Report requires that each plant's corrective action program evaluate equipment failures and establish appropriate corrective actions. In order to assure there is no significant degradation in EFCV performance due to aging effects, the CNS Maintenance Rule Program will be used to monitor EFCV performance. For any future EFCV failures identified at CNS, as part of the implementation of this TS amendment, the 10CFR 50.65 Maintenance Rule Program at CNS will be revised to include a specific EFCV performance criterion of ≤ 2 failures per rolling 36-month period. When this performance criterion is exceeded, a 10CFR 50.65(a)(1) determination will be performed in accordance with CNS procedures.

The proposed change to the Technical Specifications (TS) Surveillance Requirement (SR) 3.6.1.3.8 will relax the SR frequency by allowing "a representative sample" of EFCVs to be tested every 18 months. The"representative sample" is not defined in the TS itself,

however, the proposed Bases for SR 3.6.1.3.8 state that the representative sample consists of an approximately equal number of EFCVs, such that each EFCV is tested at least once every 10 years (nominal). This presentation has been found acceptable in the NRC approved TSTF-334, Revision 2. In this regard, CNS commits to make the following change to the Bases for SR 3.6.1.3.8 upon implementation of NRC issuance of the requested license amendment:

> This SR requires a demonstration that a representative sample of reactor instrumentation line excess flow check valves (EFCV) are OPERABLE by verifying that each valve actuates to the isolation position on an actual or simulated instrument line break. The representative sample consists of an approximately equal number of EFCVs, such that each EFCV is tested at least once every 10 years (nominal). This SR provides assurance that the instrumentation line EFCVs will perform so that predicted radiological consequences will not be exceeded during the postulated instrument line break event. The 18 month Frequency is based on the need to perform the Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power.
>
> The nominal 10 year interval is based on other performance-based testing programs, such as Inservice Testing (snubbers) and Option B to 10 CFR 50, Appendix J. Furthermore, any EFCV failures will be evaluated to determine if additional testing in that test interval is warranted to ensure overall reliability is maintained. Operating experience has demonstrated that these components are highly reliable and that failures to isolate are very infrequent. Therefore, testing of a representative sample was concluded to be acceptable from a reliability standpoint.

The Bases provided in TSTF-334, Revision 2, also include a statement that "the EFCVs in the sample are representative of the various plant configurations, models, sizes and operating environments." However, the EFCVs at CNS are of the same manufacturer and model and, with the exception of the EFCV associated with the reactor flange leakoff pressure instrumentation, similar operating environment. For the EFCV associated with the reactor flange leakoff pressure instrumentation, it is noted that the leakoff line is connected to the reactor into the annulus between the two metallic seal rings used to seal the reactor vessel and top head flanges. As such, this line is normally isolated from the reactor coolant system by the inner o-ring. As such, no specific grouping of the representative sample is necessary and this statement will not be included in the CNS Bases. Changes to these Bases and the associated clarifying details of "representative sample" are subject to appropriate controls, which are specified in CNS Technical Specification 5.5.10, Bases Control Program. Based on the low failure rate of EFCVs and the low safety significance of a failure of an EFCV (discussed in further detail below), the level of detail in the proposed SR itself is appropriate.

Based on the NRC staff's conclusions reached in the evaluation of the BWR Owners' Group Topical Report (contained within Reference 1) the acceptability of the methods applied to estimate the release frequency, a relatively low release frequency estimate in conjunction with extremely low likelihood that this release could impact core damage frequency (CDF) and negligible consequence of a release in the reactor building, it was concluded that the increase in risk associated with this request for relaxation of EFCV surveillance testing is sufficiently low and acceptable.

A review of the performance history of the EFCVs at CNS over 26 years of operation confirms the low failure frequency of these valves as discussed in NEDO-32977-A. Thus, the reliability of these valves at CNS is in line with general industry experience. However, even if the failure frequency were to increase to a level which would trigger action in the CNS Maintenance Rule Program, i.e., 2 failures in 36 months, the CDF would still not be significantly impacted.

As stated in the Safety Evaluation for the CNS operating license, dated February 14, 1973 (Reference 4), the radiological consequences for an instrument line break credit the 1/4 inch orifices to prevent overpressurization of the reactor building and limit offsite doses to substantially below the 10CFR Part 100 values. The radiation dose consequences for an instrument line break are not impacted by the proposed change since there is no change in the function or operation of the restricting orifice to limit the blowdown, nor any change to a source term.

However, the District has further evaluated the potential dose consequences of an instrument line break with failure of the EFCV to isolate by a CNS plant-specific comparison to NEDO-32977-A (Ref.1), Attachment B, "Instrument Line Break Radiological Analysis." The Reference 1 analysis is presented as a "typical GE radiological evaluation ... using a GE methodology which has been accepted by the NRC in GE FSAR submittals." Reference 1 concludes that the radiological consequence of EFCVs failing to function upon demand is sufficiently low to be considered insignificant, and that the EFCVs are not needed to assure a containment isolation function.

For the CNS specific comparison, a sensitivity evaluation was conducted to determine the impact that CNS specific design inputs may have on the instrument line break radiological analysis contained in Attachment B to NEDO-32977-A. In particular, the input data of Table B-2 and the Technical Specification dose equivalent I-131 values of Attachment B were compared to CNS values. No other assumptions made in the Attachment B analysis were assumed to change.

Table B-1 Dose Equivalent I-131 Comparison

Item	NEDO-32977-A Value	CNS Value	NEDO-32977-A Value x DCF Rem/gm	CNS Value x DCF Rem/gm
I-131 Reactor Water Concentration	0.047 uCi/gm	0.083 uCi/gm	0.06956	0.1228
I-132 Reactor Water Concentration	0.415 uCi/gm	0.46 uCi/gm	2.220E-3	2.461E-3
I-133 Reactor Water Concentration	0.326 uCi/gm	0.49 uCi/gm	0.1304	0.1960
I-134 Reactor Water Concentration	1.207 uCi/gm	0.66 uCi/gm	3.018E-2	1.650E-2
I-135 Reactor Water Concentration	0.755 uCi/gm	0.63 uCi/gm	9.362E-2	7.812E-2
Overall Dose Equivalent I-131 Summation	N/A	N/A	Sum= 0.3260	Sum=0.4159

Based on the information presented in Table B-1 the CNS Technical Specification Dose Equivalent I-131 value results in an approximate 28% increase in Rem/gm over the values used in the NEDO-32977-A Appendix B radiological analysis for the reactor water inventory contribution.

Table B-2 Comparison

Item	NEDO-32977-A Value	CNS Value
EAB X/Q	2.6E-4 sec/m^3	5.2E-4 sec/m^3
LPZ X/Q	1.1E-5 sec/m^3	2.9E-4 sec/m^3
Number of Bundles	724	548
Mass of RPV Water	590,000 lbs	437,000 lbs

From Table B-2 it can been seen that the decrease in reactor water volume for the CNS would result in a corresponding decrease in the reactor water inventory of Dose Equivalent I-131. This effectively cancels out the overall increase in the CNS Dose Equivalent I-131 Rem/gm value determined in Table B-1 above.

The reduction in fuel bundles in the CNS case would be directly offset by the increase in corresponding bundle gap inventory per bundle if it is assumed that the total bundle gap inventory is constant regardless of core type. However, the fraction of reactor water volume which flashes to steam will be greater for CNS case because of its smaller reactor pressure vessel water mass. Thus, using the NEDO bundle gap inventory, it was conservatively assumed in the CNS case that the total bundle gap inventory released in the steam cloud would be greater by the ratio 590,000/437,000 or 1.35.

The CNS specific X/Q values are greater than those used in the NEDO-32977-A radiological analysis. Since X/Q values are a direct multiplier in the determination of the radiological dose, the ratio between the CNS X/Q values and the related NEDO X/Q values is used to adjust the NEDO-32977-A radiological analysis results.

Therefore, the differences between the NEDO-32977-A and CNS X/Q and mass of RPV water parameters are the primary factors which would affect the NEDO-32977-A results. The impact of the CNS X/Q and mass of RPV water values on the NEDO-32977-A results is summarized in the following table.

Table B-3 Sensitivity Results

Dose Receptor	Without Orifice Result (Rem)	With Orifice Result (Rem)	Without Orifice Result Fraction of 10CFR100 Limit (%)
EAB- Thyroid	43.2	2.43	14.4
EAB- Whole Body	0.135	0.0216	0.54
LPZ- Thyroid	17.8	3.20	5.93
LPZ- Whole Body	0.107	0.021	0.43

The CNS specific adjusted values from NEDO-32977-A, determined in this sensitivity evaluation, remain well within the exposure guideline values of 10CFR100.11, as required by Safety Guide 11, Supplement 1.

5.0 No Significant Hazards Consideration

10 CFR 50.91(a)(1) requires that licensee requests for operating license amendments be accompanied by an evaluation of significant hazard posed by issuance of an amendment. This evaluation is performed with respect to the criteria given in 10 CFR 50.92 (c).

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

   The current SR frequency requires each reactor instrumentation line EFCV to be tested every 18 months. The EFCVs at CNS are designed to close automatically in the event of a line break downstream of the valve. This proposed change allows a reduced number of EFCVs to be tested every 18 months. Industry operating experience, documented in BWR Owners' Group Topical Report NEDO-32977-A (Reference 1), concludes that a change in surveillance test frequency has a minimal impact on the reliability for these valves. A failure of an EFCV to isolate cannot initiate previously evaluated accidents. Furthermore, neither the EFCV actuation test, nor the frequency of testing is considered an initiator of any analyzed event. Therefore, there is no increase in the probability of occurrence of an accident as a result of this proposed change.

   The consequences of a previously analyzed event are dependent on the initial conditions assumed for the analysis, and the availability and successful functioning of the equipment assumed to operate in response to the analyzed event, and the setpoints at which these actions are initiated. This change does not affect the performance of any credited equipment. The installed restricting orifice

on each associated instrument line provides assurance that any instrument line break will limit offsite doses to substantially below 10CFR Part 100 values. Neither the EFCV actuation test, nor the frequency of testing is an analysis assumption. Therefore, there is no increase in the previously evaluated consequences of the rupture of an instrument line and there is no potential increase in the radiological consequences of an accident previously evaluated as a result of this change.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

   This proposed change allows a reduced number of EFCVs to be tested each operating cycle. No other changes in requirements are being proposed. Industry operating experience as documented in Reference 1 provides supporting evidence that the reduced testing frequency will not affect the high reliability of these valves. The potential failure of an EFCV to isolate as a result of the proposed reduction in test frequency is bounded by the previous evaluation of an instrument line pipe break. This change will not physically alter the plant (no new or different type of equipment will be installed). This change will not alter the operation of process variables, structures, systems, or components as described in the safety analysis. Thus, a new or different kind of accident will not be created.

3. Does this change involve a significant reduction in a margin of safety?

   The margin of safety is established through equipment design, operating parameters, and the setpoints at which automatic actions are initiated. EFCV design, operation, and flow actuation criteria remain unaffected by this change. Restricting orifices for each associated instrument line remains available to mitigate an instrument line break. The proposed change, which impacts the frequency of testing EFCVs is acceptable because the tests continue to require appropriate confirmation of the assumed function of the system (and thereby assure continued operability), and has been shown to reflect an acceptable frequency for detecting failures. There is no detrimental impact on any other equipment design parameter, and the plant will still be required to operate within prescribed limits. Therefore, the change does not involve a significant reduction in the margin of safety.

6.0 Environmental Impact Evalution

10 CFR 51.22(c)(9) provides criteria for, and identification of, licensing and regulatory actions eligible for categorical exclusion from performing an environmental assessment. A proposed amendment to an operating license for a facility does not require environmental assessment if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant hazards consideration, (2) result in a

significant change in the types or significant increase in the amount of any effluents that may be released off-site, or (3) result in an increase in individual or cumulative occupational radiation exposure. The District has reviewed the proposed license amendment and concludes that it meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(c), no environmental impact statement or environmental assessment needs to be prepared in connection with issuance of the proposed license change. The basis for this determination is as follows:

1. The proposed license amendment does not involve significant hazards as described previously in the No Significant Hazards Consideration Evaluation.

2. As discussed in the No Significant Hazards Consideration Evaluation, the proposed changes to SR 3.6.1.3.8 do not introduce any new equipment, nor do they require any existing equipment or systems to perform a different type of function than they are presently designed to perform during normal operation. The District has concluded that there will not be a significant increase in the types or amounts of effluents that may be released off-site and these changes do not involve irreversible environmental consequences beyond those already associated with normal operation.

3. The proposed change involves a revision to the Technical Specification requirements for the frequency of performing EFCV flow actuation testing. This reduced testing will result in directly reducing worker radiation exposure. Thus, the proposed changes do not increase individual or cumulative occupational radiation exposure.

7.0 Conclusion

The requested change would revise the surveillance frequency from testing each excess flow check valve once per 18 months, to testing "a representative sample" of EFCVs once per 18 months. This representative sample consists of an approximately equal number of EFCVs that will be tested each 18 months, such that each EFCV will be tested at least once every ten years (nominal). The EFCVs have been shown, both plant specifically as well as industry generically, to be very reliable. Furthermore, given the relatively low release frequency estimate in conjunction with extremely low likelihood that this release could impact core damage frequency and negligible consequence of a release in the reactor building, it was concluded that the increase in risk associated with this request for relaxation of EFCV surveillance testing is sufficiently low and acceptable.

8.0 References

1. Boiling Water Reactor (BWR) Owners' Group Report, NEDO-32977-A, "Excess Flow Check Valve Testing Relaxation," dated June 2000.

2. Technical Specification Task Force Change TSTF-334, Revision 2.

3. BWR Owners' Group Generic Response to NRC Request for Additional Information on Lead Plant Technical Specification Change Request Regarding Excess Flow Check Valve Surveillance Requirements, dated January 6, 2000 (BWROG-00001).

4. Safety Evaluation for the CNS operating license, dated February 14, 1973.

**Mark-Up to show Specific Changes to
Existing Technical Specifications
and Associated Bases (for information)**

Cooper Nuclear Station, NRC Docket 50-298, DPR-46

<u>Revised TS Pages</u>
3.6-14

<u>Revised Bases Pages</u>
B 3.6-27

Note: Bases are provided for information. Following approval of the proposed TS change, Bases changes will be implemented in accordance with TS 5.5.10, Technical Specification (TS) Bases Control Program.

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.6.1.3.6	Verify the isolation time of each MSIV is $\geq$ 3 seconds and $\leq$ 5 seconds.	In accordance with the Inservice Testing Program
SR 3.6.1.3.7	Verify each automatic PCIV actuates to the isolation position on an actual or simulated isolation signal.	18 months
SR 3.6.1.3.8	Verify ~~each~~ a representative sample of reactor instrumentation line EFCVs actuate~~s~~ to the isolation position on an actual or simulated instrument line break.	18 months
SR 3.6.1.3.9	Remove and test the explosive squib from each shear isolation valve of the TIP System.	18 months on a STAGGERED TEST BASIS
SR 3.6.1.3.10	Verify leakage rate through each MSIV is $\leq$ 11.5 scfh when tested at $\geq$ 29 psig.	In accordance with the Primary Containment Leakage Rate Testing Program

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.1.3.6 (continued)

calculated radiological consequences of these events remain within 10 CFR 100 limits. The Frequency of this SR is in accordance with the requirements of the Inservice Testing Program.

SR 3.6.1.3.7

Automatic PCIVs close on a primary containment isolation signal to prevent leakage of radioactive material from primary containment following a DBA. This SR ensures that each automatic PCIV will actuate to its isolation position on a primary containment isolation signal. The LOGIC SYSTEM FUNCTIONAL TEST in LCO 3.3.6.1, "Primary Containment Isolation Instrumentation," overlaps this SR to provide complete testing of the safety function. The 18 month Frequency was developed considering it is prudent that this Surveillance be performed only during a unit outage since isolation of penetrations would disrupt the normal operation of many critical components. Operating experience has shown that these components usually pass this Surveillance when performed at the 18 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

SR 3.6.1.3.8

each

a representative sample of

This SR requires a demonstration that ~~each~~ reactor(s) instrumentation line excess flow check valves (EFCVs) ~~is~~ are OPERABLE by verifying that ~~the~~ valve actuates to the isolation position on an actual or simulated instrument line break. This SR provides assurance that the instrumentation line EFCVs will perform so that predicted radiological consequences will not be exceeded during the postulated instrument line break event. The 18 month Frequency is based on the need to perform the Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. ~~Operating experience has shown that these components usually pass this Surveillance when performed at the 18 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.~~

INSERT 1

INSERT 2

(continued)

INSERT 1

The representative sample consists of an approximately equal number of EFCVs, such that each EFCV is tested at least once every 10 years (nominal). ...

INSERT 2

... The nominal 10 year interval is based on other performance-based testing programs, such as Inservice Testing (snubbers) and Option B to 10 CFR 50, Appendix J. Furthermore, any EFCV failures will be evaluated to determine if additional testing in that test interval is warranted to ensure overall reliability is maintained. Operating experience has demonstrated that these components are highly reliable and that failures to isolate are very infrequent. Therefore, testing of a representative sample was concluded to be acceptable from a reliability standpoint.

Clean Copy of Proposed Changes to
Technical Specifications
and Associated Bases (for information)

Cooper Nuclear Station, NRC Docket 50-298, DPR-46

Revised TS Pages
3.6-14

Revised Bases Pages
B 3.6-27

Note: Bases are provided for information. Following approval of the proposed TS change, Bases changes will be implemented in accordance with TS 5.5.10, Technical Specification (TS) Bases Control Program.

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.6.1.3.6	Verify the isolation time of each MSIV is $\geq$ 3 seconds and $\leq$ 5 seconds.	In accordance with the Inservice Testing Program
SR 3.6.1.3.7	Verify each automatic PCIV actuates to the isolation position on an actual or simulated isolation signal.	18 months
SR 3.6.1.3.8	Verify a representative sample of reactor instrumentation line EFCVs actuate to the isolation position on an actual or simulated instrument line break.	18 months
SR 3.6.1.3.9	Remove and test the explosive squib from each shear isolation valve of the TIP System.	18 months on a STAGGERED TEST BASIS
SR 3.6.1.3.10	Verify leakage rate through each MSIV is $\leq$ 11.5 scfh when tested at $\geq$ 29 psig.	In accordance with the Primary Containment Leakage Rate Testing Program

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.1.3.6 (continued)

calculated radiological consequences of these events remain within 10 CFR 100 limits. The Frequency of this SR is in accordance with the requirements of the Inservice Testing Program.

SR 3.6.1.3.7

Automatic PCIVs close on a primary containment isolation signal to prevent leakage of radioactive material from primary containment following a DBA. This SR ensures that each automatic PCIV will actuate to its isolation position on a primary containment isolation signal. The LOGIC SYSTEM FUNCTIONAL TEST in LCO 3.3.6.1, "Primary Containment Isolation Instrumentation," overlaps this SR to provide complete testing of the safety function. The 18 month Frequency was developed considering it is prudent that this Surveillance be performed only during a unit outage since isolation of penetrations would disrupt the normal operation of many critical components. Operating experience has shown that these components usually pass this Surveillance when performed at the 18 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

SR 3.6.1.3.8

This SR requires a demonstration that a representative sample of reactor instrumentation line excess flow check valves (EFCVs) are OPERABLE by verifying that each valve actuates to the isolation position on an actual or simulated instrument line break. The representative sample consists of an approximately equal number of EFCVs, such that each EFCV is tested at least once every 10 years (nominal). This SR provides assurance that the instrumentation line EFCVs will perform so that predicted radiological consequences will not be exceeded during the postulated instrument line break event. The 18 month Frequency is based on the need to perform the Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power.

The nominal 10 year interval is based on other performance-based testing programs, such as Inservice Testing (snubbers) and Option B to 10 CFR 50, Appendix J. Furthermore, any EFCV failures will be evaluated to determine if additional testing in that test interval is warranted to ensure overall reliability is maintained. Operating experience has demonstrated that these components are highly reliable and that failures to isolate are very infrequent. Therefore, testing of a representative sample was concluded to be acceptable from a reliability standpoint.

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The following table identifies those actions committed to by the District in this document. Any other actions discussed in the submittal represent intended or planned actions by the District. They are described for information only and are not regulatory commitments. Please notify the NL&S Manager at Cooper Nuclear Station of any questions regarding this document or any associated regulatory commitments.

COMMITMENT	COMMITTED DATE OR OUTAGE
None	