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SOUTHERN COMPANY
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September 19, 2001

Docket Nos. 50-321
50-366

HL-6105

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

Edwin I. Hatch Nuclear Plant
Request to Revise Technical Specifications:
Excess Flow Check Valve Relaxation

Ladies and Gentlemen:

In accordance with the provisions set forth in 10 CFR 50.90, as required by 10 CFR 50.59(c)(1), Southern Nuclear Operating Company (SNC) is proposing changes to the Plant Hatch, Unit 1 and Unit 2 Technical Specifications, Appendix A to operating licenses DPR-57 and NPF-5, respectively. The proposed changes revise surveillance requirement (SR) 3.6.1.3.8. This SR requires verification of the actuation capability of each reactor instrumentation excess flow check valve every 18 months.

This amendment is consistent with GE NEDO-32977-A, "Excess Flow Check Valve relaxation. This report was commissioned by the BWR Owner's Group with the final report being issued in June of 2000. Plant Hatch was not an original member of the Owner's Group EFCV committee and, as a result, the report's data concerning industry EFCV failure rates did not include Hatch data. However, this data has been compiled for Hatch for the purposes of this submittal and, as described in the enclosure, the design data and failure rates are comparable to the other utilities as detailed in the report.

Per the report, we are not proposing to eliminate the EFCV testing, but merely to allow a representative sample of the EFCVs to be tested each cycle.

Enclosure 1 provides a description and justification of the proposed changes. Enclosure 2 describes SNC's determination that the proposed changes do not involve a significant hazards consideration as well as the environmental evaluation. Enclosure 3 includes the page change instructions, and the marked up and published Tech Spec changes. A mark-up of the changed Bases pages are also included.

In accordance with the requirements of 10CFR50.91, a copy of this letter and all applicable enclosures will be sent to the designated State official of the Environmental Protection Division of the Georgia Department of Natural Resources.

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Mr. H. L. Sumner, Jr. states he is Vice President of Southern Nuclear Operating Company and is authorized to execute this oath on behalf of Southern Nuclear Operating Company, and to the best of his knowledge and belief, the facts set forth in this letter are true.

Respectfully submitted,

Lewis Sumner

H. L. Sumner, Jr.

Sworn to and subscribed before me this 19th *day of* September *2001.*

Elaine E. Bolton
Notary Public

Commission Expiration Date: 5-25-2003

OCV/eb

Enclosures:
1. Basis for Change Request
2. 10 CFR 50.92 Evaluation
3. Page Change Instructions
4. Bases Changes

cc: Southern Nuclear Operating Company
Mr. P. H. Wells, Nuclear Plant General Manager
SNC Document Management (R-Type A02.001)

U.S. Nuclear Regulatory Commission, Washington, D.C.
Mr. L. N. Olshan, Project Manager - Hatch

U.S. Nuclear Regulatory Commission, Region II
Mr. L. A. Reyes, Regional Administrator
Mr. J. T. Munday, Senior Resident Inspector - Hatch

State of Georgia
Mr. L. C. Barrett, Commissioner - Department of Natural Resources

Enclosure 1

Edwin I. Hatch Nuclear Plant
Request to Revise Technical Specifications:
Excess Flow Check Valve Relaxation

Basis for Change Request

Description of Changes

This amendment changes surveillance requirement 3.6.1.3.8 of the Hatch Unit 1 and 2 Technical Specifications which presently requires that each reactor instrumentation line excess flow check valve (EFCV) be verified to properly actuate every 18 months. The change is consistent with TSTF –334 which was in turn developed from NEDO-32977-A, "Excess Flow Check Valve relaxation" a report commissioned by the BWR Owners' Group.

Testing of the EFCVs is not being eliminated. Instead, a representative sample of the EFCVs will be tested each cycle (18 months) such that each EFCV will be tested at least once every 10 years. This essentially means that approximately 15 % of the EFCVs will be tested every cycle.

Basis for Proposed Change

The justification for the proposed changes is found in the previously mentioned NEDO report. The report was put together by GE for the excess flow check valve committee of the BWR Owner's Group. Plant Hatch was not a member of this committee during the development of the report and thus, the industry data contained in the report does not include Hatch specific data. (Since then, we have joined the committee to take advantage of EFCV surveillance relaxation). Though Hatch data was not included in the NEDO, we have determined that the report bounds the Plant Hatch EFCV system, as discussed below:

The manufacturers of the Unit 1 and 2 Plant Hatch EFCVs are *Dragon Valves* and *Marrotta* respectively. Both vendors are used by many other BWR utilities and are well represented in the NEDO report. Attachment 1 to Enclosure 1 contains a table of Hatch specific EFCV data, similar to the table in the Topical Report. This data shows that the Hatch EFCVs are similar in design, and use, to the valves used by the other member utilities. Furthermore, a failure rate analysis of the Hatch EFCVs was done over three operating cycles. These results show that the failure rate study done for the Topical Report bounds the Hatch failure rate. Attachment 2 to Enclosure 1 details the results of the study. This information supports the contention that the generic radiological consequences evaluation performed by GE in Attachment B to the NEDO bounds Plant Hatch. It is thus reasonable to conclude, as the NEDO states, that similar results would be expected at Hatch.

The NEDO also states that most plants have performed radiological evaluations of reactor coolant pressure boundary instrument line breaks. That is also the case at Plant Hatch. Our evaluation is documented in section 15.4.13 of the Hatch Unit 2 FSAR. The analysis assumes a circumferential rupture of an instrument line connected to the primary coolant outside of the

primary containment but inside the excess flow check valves, thus the excess flow check valves are not taken credit for in the Hatch analysis of record. A failure of an EFCV is therefore bounded by the existing FSAR evaluation. The Hatch instrument lines do have a ¼" orifice inside the primary containment, and our specific analysis takes credit for these orifices. The evaluation results are documented in the above referenced FSAR section.

With respect to a feedback mechanism, Hatch commits to the following: any excess flow check valve that fails to check flow in its surveillance test will be documented in the Hatch corrective action program as a surveillance test failure. The failure will be evaluated and corrected and, if the valve is repaired and not replaced, it will be added to the next cycle's surveillance.

Summarizing, the Hatch EFCVs are similar in design and application to those of the utilities which participated in the BWROG committee. Additionally, the Hatch failure rate study indicates that the performance of the Hatch EFCVs is comparable to the performance of those EFCVs from the utilities listed in the report. Accordingly, it is reasonable to state that the conclusions of the topical report are bounding with respect to Plant Hatch and that we are justified in seeking the surveillance relaxation.

Attachment 1 to Enclosure 1

Edwin I. Hatch Nuclear Plant
Request to Revise Technical Specifications:
Excess Flow Check Valve Relaxation

Table of Hatch Specific Information

Attachment 1 to Enclosure 1

Table of Hatch Specific Information

	Unit 1	**Unit 2**
Design Pressure (psig)	1660	1660
Hydrostatic Pressure (psig)	3250	3250
Make and Model	Dragon	Marrotta
Nominal Line Size (inlet/outlet, inches)	1.33/.38	1.33/.38
Main Valve body orifice (inches)	.25	.25
Minimum flow for closure (range in gpm)	1.7 – 2.09	1.7 – 2.09
Test Method (online/offline)	(reactor pressure/ nitrogen bottles)	(reactor pressure/ nitrogen bottles)
Testing performed	During refueling outage: some are done offline, and some are done online during the shutdown.	
Testing on critical path	No	No
Man Rem exposure during testing (Mr)	100	100
Man-hours for testing during a cycle	1000	1000
Types of failures	Typically only limit switches and bulbs	
PM program	No, in maintenance rule and in surveillance program.	
Failed EFCVs replaced or repaired?	Depends on failure. If valve fails, it's usually replaced.	

Attachment 2 to Enclosure 1

Edwin I. Hatch Nuclear Plant
Request to Revise Technical Specifications:
Excess Flow Check Valve Relaxation

Hatch Excess Flow Check Valve Failure Rate Study Results

Attachment 2 to Enclosure 1

Hatch Excess Flow Check Valve Failure Rate Study Results

Failure Rate Analysis

This failure rate study of the Hatch 2 Excess Flow Check Valves was conducted by reviewing the data for the September, 1995, March, 1997 and September 1998 refueling outages. A total of 87 EFCVs were tested each cycle with only 3 valve failures during this time. A review of the data packages show a total of 28 failures, however, most of these failures were for double indications or bad bulbs. Only three of these failures were attributed to the valve actually failing to check flow. The operating failure rates are tabulated below.

The calculation of the upper limit failure rate was performed as follows:

For Unit 2, the operating time used was 3 operating cycles, since failure rate history was obtained over three 18 month operating cycles. The number of failures (for Unit 2) was 3. The Chi-squared value for 8 degrees of freedom and a confidence level of .95 is 15.5. From these values, and using the formula listed in the Topical Report Section 4.2, the upper limit failure value is 2.26×10^{-6} hr^{-1}.

A similar analysis was performed for Unit 1, spanning three refueling outages and three operating cycles. Unit 1 had total of 1 failure to check flow. The upper limit failure value for Unit 1 is 1.38×10^{-6} hr^{-1}.

A summary table is provided below.

Unit	**Total Valve Operating Time (hr)**	**Failures**	**Best Estimate Failure Rate (hr^{-1})**	**Chi-Squared Value (χ^2) 95% Confidence**	**Upper Limit of Expected Failures (hr^{-1})**
1	3.43E+06	1	2.92×10^{-7}	9.49	1.38×10^{-6}
2	3.43E+06	3	8.75×10^{-7}	15.5	2.26×10^{-6}

Enclosure 2

Edwin I. Hatch Nuclear Plant
Request to Revise Technical Specifications:
Excess Flow Check Valve Relaxation

10 CFR 50.92 Evaluation

10 CFR 50.92 no significant hazards evaluation and environmental assessment

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

   The Excess Flow Check Valves are designed to limit the flow from an instrument line break downstream of the check valve itself. Thus the previously analyzed event is the instrument line break, documented in the Unit 2 FSAR, section 15.4.13, for both units. This proposed revision does not alter the operation or maintenance of any instrument line; the revision is made to reduce the surveillance requirements for the EFCVs. This revision does nothing which jeopardizes the integrity of the instrument lines and thus increase the probability of a line break.

   The line break analysis does not take credit for operation of the excess flow check valves, therefore, the radiological consequences of this event are not affected by this proposed TS revision.

   This amendment request does not affect any other previously evaluated line or pipe break analysis.

   For the above reasons, the probability of occurrence, or the consequences of a previously evaluated event are not increased by this proposed change.

2. *Do the proposed changes create the possibility of a new type event different from any previously evaluated?*

   No changes are being made to the way in which the EFCVs are operated, or maintained; they will continue to be operated within the conditions for which they were designed. Since no new operational modes are proposed, no new failure modes are introduced.

   Furthermore, no changes to any systems designed for the prevention of transients or accidents are being made as a result of this proposed Technical Specifications change.

   For the above reasons, this proposed change does not introduce the possibility of a different type event from any previously evaluated.

3. *Do the proposed changes involve a significant reduction in the margin of safety?*

The reactor coolant pressure boundary line break analysis documented in Unit 2 FSAR section 15.4.13 does not assume credit for the EFCVs. Additionally, the failure rate of the Unit 1 and 2 EFCVs has been small, as verified by the failure rate analysis done for this proposed revision. Accordingly, reducing the frequency of the surveillance is justified and will not significantly reduce the margin of safety with respect to EFCV failure.

Additionally, General Electric has performed a generic radiological evaluation of an instrument line break, with EFCV failure, which concluded that the dose consequences would not exceed 10 CFR 100 guidelines. This analysis is documented in NEDO-32977-A, "Excess Flow Check Valve relaxation", a report commissioned by the Boiling Water Reactors Owners' Group (BWROG). Because the Hatch EFCV design is similar to the EFCV designs assumed in the NEDO, it is reasonable to conclude that the results of this generic analysis are bounding for Plant Hatch.

Environmental Assessment

10 CFR 51.22(c)(9) provides criterion for 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 requires no environmental assessment if operation of the facility in accordance with the proposed license amendment would not:

1. Involve a significant hazards consideration,

2. Result in a significant change in the types or significant increase in the amounts of any effluents that may be released off-site, or,

3. Result in a significant increase in individual or cumulative occupational radiation exposure.

Southern Nuclear Operating Company (SNC) has determined that the proposed Technical Specifications changes, described in Enclosure 1, meet the eligibility criteria for categorical exclusion set forth in 10 CFR 50.22(c)(9). Accordingly, pursuant to 10 CFR 50.22, no environmental impact statement needs to be prepared in connection with the issuance of the license amendment for the proposed changes. The basis for this determination using the above criteria follows:

1. As demonstrated in this enclosure, the proposed changes do not involve a significant hazards consideration.

2. The proposed changes do not result in a significant change to the types of effluents or in the amounts of effluents released offsite. This proposed change involves reducing the surveillance frequency for the Plant Hatch excess flow check valves. This change does not involve any radioactive waste processing or monitoring system; accordingly, it does not result in any changes to off-site effluents.

3. This proposed change does not result in a significant increase in occupational radiation exposure. The change impacts the surveillance frequency for excess flow check valves. In fact, the reduction in frequency should result in a decrease in the man-rem exposure per cycle, since fewer EFCVs will be tested.

Enclosure 3

Edwin I. Hatch Nuclear Plant
Request to Revise Technical Specifications:
Excess Flow Check Valve Relaxation

Page Change Instructions

Unit 1

Page	Instructions
3.6-14	Replace

Unit 2

Page	Instructions
3.6-14	Replace

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.6.1.3.6	Verify the isolation time of each MSIV is ≥ 3 seconds and ≤ 5 seconds.	In accordance with the Inservice Testing Program
SR 3.6.1.3.7	Verify each automatic PCIV, excluding EFCVs, actuates to the isolation position on an actual or simulated isolation signal.	18 months
SR 3.6.1.3.8	Verify each reactor instrumentation line EFCV (of a representative sample) actuates to restrict flow to within limits.	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 ≤ 11.5 scfh when tested at ≥ 28.0 psig.	In accordance with the Primary Containment Leakage Rate Testing Program

(continued)

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, excluding EFCVs, actuates to the isolation position on an actual or simulated isolation signal.	18 months
SR 3.6.1.3.8	Verify each reactor instrumentation line EFCV (of a representative sample) actuates to restrict flow to within limits.	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 the combined leakage rate for all secondary containment bypass leakage paths is $\leq 0.009\ L_a$ when pressurized to $\geq P_a$.	In accordance with the Primary Containment Leakage Rate Testing Program

(continued)

Enclosure 4

Edwin I. Hatch Nuclear Plant
Request to Revise Technical Specifications:
Excess Flow Check Valve Relaxation

Bases Changes

Unit 1

Page	Instructions
B 3.6-27	Replace
B 3.6-28	Replace
B 3.6-28a	Add
B 3.6-28b	Add

Unit 2

Page	Instructions
B 3.6-27	Replace
B 3.6-28	Replace
B 3.6-29	Replace

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.1.3.8

This SR requires a demonstration that each reactor instrumentation line excess flow check valve (EFCV) (of a representative sample) is OPERABLE by verifying that the valve reduces flow to within limits on an actual or simulated instrument line break condition. (The representative sample consists of an approximately equal number of EFCVs, such that each EFCV is tested at least once every 10 years [nominal]. In addition, the EFCVs in the sample are representative of the various plant configurations, models, sizes, and operating environments. This ensures that any potentially common problem with a specific type of application of EFCV is detected at the earliest possible time.) This SR provides assurance that the instrumentation line EFCVs will perform as designed. The 18 month Frequency is based on the need to perform this 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. (The nominal 10 year interval is based on performance testing as discussed in NEDO-32977-A, "Excess Flow Check Valve Testing Relaxation." 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.) Any excess flow check valve that fails to check flow during its surveillance test will be documented in the Hatch corrective action program as a surveillance test failure. The failure will be evaluated and corrected and, if the valve is repaired and not replaced, it will be added to the next cycle's surveillance.

SR 3.6.1.3.9

The TIP shear isolation valves are actuated by explosive charges. An in place functional test is not possible with this design. The explosive squib is removed and tested to

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.1.3.9 (continued)

provide assurance that the valves will actuate when required. The replacement charge for the explosive squib shall be from the same manufactured batch as the one fired or from another batch that has been certified by having one of the batch successfully fired. The Frequency of 18 months on a STAGGERED TEST BASIS is considered adequate given the administrative controls on replacement charges and the frequent checks of circuit continuity (SR 3.6.1.3.4).

SR 3.6.1.3.10

The analyses in References 1 and 3 are based on leakage that is less than the specified leakage rate. Leakage through each MSIV must be ≤ 11.5 scfh when tested at ≥ 28.0 psig.

The Frequency is required by the Primary Containment Leakage Rate Testing Program (Ref. 6).

SR 3.6.1.3.11

The valve seats of each 18 inch purge valve (supply and exhaust) having resilient material seats must be replaced every 18 months. This will allow the opportunity for repair before gross leakage failure develops. The 18 month Frequency is based on engineering judgment and operational experience which shows that gross leakage normally does not occur when the valve seats are replaced on an 18 month Frequency.

SR 3.6.1.3.12

The Surveillance Requirement provides assurance that the excess flow isolation dampers can close following an isolation signal. The 18 month Frequency is based on vendor recommendations and engineering judgment. Operating experience has shown that these dampers usually pass the Surveillance when performed at the 18 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

(continued)

BASES (continued)

REFERENCES

1. FSAR, Section 14.4.
2. Technical Requirements Manual
3. FSAR, Section 5.2.
4. 10 CFR 50, Appendix J, Option B.
5. NRC No. 93-102, "Final Policy Statement on Technical Specification Improvements," July 23, 1993.
6. Primary Containment Leakage Rate Testing Program.

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BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.1.3.8

This SR requires a demonstration that each reactor instrumentation line excess flow check valve (EFCV) (of a representative sample) is OPERABLE by verifying that the valve reduces flow to within limits on an actual or simulated instrument line break condition. (The representative sample consists of an approximately equal number of EFCVs, such that each EFCV is tested at least once every 10 years [nominal]. In addition, the EFCVs in the sample are representative of the various plant configurations, models, sizes, and operating environments. This ensures that any potentially common problem with a specific type of application of EFCV is detected at the earliest possible time.) This SR provides assurance that the instrumentation line EFCVs will perform as designed. The 18 month Frequency is based on the need to perform this 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. (The nominal 10 year interval is based on performance testing as discussed in NEDO-32977-A, "Excess Flow Check Valve Testing Relaxation." 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.) Any excess flow check valve that fails to check flow during its surveillance test will be documented in the Hatch corrective action program as a surveillance test failure. The failure will be evaluated and corrected and, if the valve is repaired and not replaced, it will be added to the next cycle's surveillance.

SR 3.6.1.3.9

The TIP shear isolation valves are actuated by explosive charges. An in place functional test is not possible with this design. The explosive squib is removed and tested to

(continued)

SURVEILLANCE REQUIREMENTS

SR 3.6.1.3.9 (continued)

provide assurance that the valves will actuate when required. The replacement charge for the explosive squib shall be from the same manufactured batch as the one fired or from another batch that has been certified by having one of the batch successfully fired. The Frequency of 18 months on a STAGGERED TEST BASIS is considered adequate given the administrative controls on replacement charges and the frequent checks of circuit continuity (SR 3.6.1.3.4).

SR 3.6.1.3.10

This SR ensures that the leakage rate of secondary containment bypass leakage paths is less than the specified leakage rate. This provides assurance that the assumptions in the radiological evaluations that form the basis of the FSAR (Ref. 3) are met. The secondary containment bypass leakage paths are: 1) main steam condensate drain, penetration 8; 2) reactor water cleanup, penetration 14; 3) equipment drain sump discharge, penetration 18; 4) floor drain sump discharge, penetration 19; and 5) chemical drain sump discharge, penetration 55. The leakage rate of each bypass leakage path is assumed to be the maximum pathway leakage (leakage through the worse of the two isolation valves) unless the penetration is isolated by use of one closed and de-activated automatic valve, closed manual valve, or blind flange. In this case, the leakage rate of the isolated bypass leakage path is assumed to be the actual pathway leakage through the isolation device. If both isolation valves in the penetration are closed, the actual leakage rate is the lesser leakage rate of the two valves. The Frequency is required by the Primary Containment Leakage Rate Testing Program (Ref. 7).

SR 3.6.1.3.11

The analyses in References 1 and 4 are based on leakage that is less than the specified leakage rate. Leakage through each MSIV must be ≤ 100 scfh, and a combined maximum pathway leakage ≤ 250 scfh for all four main steam lines when tested at ≥ 28.8 psig. In addition, if any MSIV exceeds the 100 scfh limit, the as left leakage shall be ≤ 11.5 scfh for that MSIV.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.1.3.11 (continued)

The Frequency is required by the Primary Containment Leakage Rate Testing Program.

SR 3.6.1.3.12

The valve seats of each 18 inch purge valve (supply and exhaust) having resilient material seats must be replaced every 18 months. This will allow the opportunity for repair before gross leakage failure develops. The 18 month Frequency is based on engineering judgment and operational experience which shows that gross leakage normally does not occur when the valve seats are replaced on an 18 month Frequency.

SR 3.6.1.3.13

The Surveillance Requirement provides assurance that the excess flow isolation dampers can close following an isolation signal. The 18 month Frequency is based on vendor recommendations and engineering judgment. Operating experience has shown that these dampers usually pass the Surveillance when performed at the 18 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

REFERENCES

1. FSAR, Chapter 15.
2. Technical Requirements Manual.
3. FSAR, Section 15.1.39.
4. FSAR, Section 6.2.
5. 10 CFR 50, Appendix J, Option B.
6. NRC No. 93-102, "Final Policy Statement on Technical Specification Improvements," July 23, 1993.
7. Primary Containment Leakge Rate Testing 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, excluding EFCVs, actuates to the isolation position on an actual or simulated isolation signal.	18 months
SR 3.6.1.3.8	Verify each reactor instrumentation line EFCV [of a representative sample] actuates to restrict flow to within limits.	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 $\geq$ 28.0 psig.	In accordance with the Primary Containment Leakage Rate Testing Program

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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, excluding EFCVs, actuates to the isolation position on an actual or simulated isolation signal.	18 months
SR 3.6.1.3.8	Verify each reactor instrumentation line EFCV (of a representative sample) actuates to restrict flow to within limits.	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 the combined leakage rate for all secondary containment bypass leakage paths is $\leq 0.009\ L_a$ when pressurized to $\geq P_a$.	In accordance with the Primary Containment Leakage Rate Testing Program

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.1.3.8

(of a representative sample)

This SR requires a demonstration that each reactor instrumentation line excess flow check valve (EFCV) is OPERABLE by verifying that the valve reduces flow to within limits on an actual or simulated instrument line break condition. This SR provides assurance that the instrumentation line EFCVs will perform as designed. The 18 month Frequency is based on the need to perform this 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

Insert 3

SR 3.6.1.3.9

The TIP shear isolation valves are actuated by explosive charges. An in place functional test is not possible with this design. The explosive squib is removed and tested to provide assurance that the valves will actuate when required. The replacement charge for the explosive squib shall be from the same manufactured batch as the one fired or from another batch that has been certified by having one of the batch successfully fired. The Frequency of 18 months on a STAGGERED TEST BASIS is considered adequate given the administrative controls on replacement charges and the frequent checks of circuit continuity (SR 3.6.1.3.4).

SR 3.6.1.3.10

The analyses in References 1 and 3 are based on leakage that is less than the specified leakage rate. Leakage through each MSIV must be ≤ 11.5 scfh when tested at ≥ 28.0 psig.

(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). In addition, the EFCVs in the sample are representative of the various plant configurations, models, sizes, and operating environments. This ensures that any potentially common problem with a specific type of application of EFCV is detected at the earliest possible time.)

Insert 2

(The nominal 10 year interval is based on performance testing as discussed in NEDO-32977-A, "Excess Flow Check Valve Testing Relaxation." 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.)

Insert 3

Any excess flow check valve that fails to check flow during its surveillance test will be documented in the Hatch corrective action program as a surveillance test failure. The failure will be evaluated and corrected and, if the valve is repaired and not replaced, it will be added to the next cycle's surveillance.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.1.3.8

[of a representative sample]

This SR requires a demonstration that each reactor instrumentation line excess flow check valve (EFCV) is OPERABLE by verifying that the valve reduces flow to within limits on an actual or simulated instrument line break condition. This SR provides assurance that the instrumentation line EFCVs will perform as designed. The 18 month Frequency is based on the need to perform this 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.

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SR 3.6.1.3.9

The TIP shear isolation valves are actuated by explosive charges. An in place functional test is not possible with this design. The explosive squib is removed and tested to provide assurance that the valves will actuate when required. The replacement charge for the explosive squib shall be from the same manufactured batch as the one fired or from another batch that has been certified by having one of the batch successfully fired. The Frequency of 18 months on a STAGGERED TEST BASIS is considered adequate given the administrative controls on replacement charges and the frequent checks of circuit continuity (SR 3.6.1.3.4).

SR 3.6.1.3.10

This SR ensures that the leakage rate of secondary containment bypass leakage paths is less than the specified leakage rate. This provides assurance that the assumptions in the radiological evaluations that form the basis of the FSAR (Ref. 3) are met. The secondary containment bypass leakage paths are: 1) main steam condensate drain, penetration 8; 2) reactor water cleanup, penetration 14; 3) equipment drain sump discharge, penetration 18; 4) floor drain sump discharge, penetration 19; and 5) chemical drain sump discharge, penetration 55. The leakage rate of each bypass leakage path is assumed to be the maximum pathway

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