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RS-07-044
TMI 5928-07-20153
OC 2130-07-20500

10 CFR 50.90

August 8, 2007

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

Braidwood Station, Units 1 and 2
Facility Operating License Nos. NPF-72 and NPF-77
NRC Docket Nos. STN 50-456 and STN 50-457

Byron Station, Units 1 and 2
Facility Operating License Nos. NPF-37 and NPF-66
NRC Docket Nos. STN 50-454 and STN 50-455

Dresden Nuclear Power Station, Units 2 and 3
Renewed Facility Operating License Nos. DPR-19 and DPR-25
NRC Docket Nos. 50-237 and 50-249

Limerick Generating Station, Units 1 and 2
Facility Operating License Nos. NPF-39 and NPF-85
NRC Docket Nos. 50-352 and 50-353

Oyster Creek Generating Station
Facility Operating License No. DPR-16
NRC Docket No. 50-219

Peach Bottom Atomic Power Station, Units 2 and 3
Renewed Facility Operating License Nos. DPR-44 and DPR-56
NRC Docket Nos. 50-277 and 50-278

Quad Cities Nuclear Power Station, Units 1 and 2
Renewed Facility Operating License Nos. DPR-29 and DPR-30
NRC Docket Nos. 50-254 and 50-265

Three Mile Island Nuclear Station, Unit 1
Facility Operating License No. DPR-50
NRC Docket No. 50-289

Subject: Request for Amendment to Technical Specifications Section for the Inservice Testing Program

References: 1. TSTF-479-A, Revision 0, "Changes to Reflect Revision of 10 CFR 50.55a," dated December 19, 2005

2. TSTF-497, Revision 0, "Limit Inservice Testing Program SR 3.0.2 Application to Frequencies of 2 Years or Less," dated July 12, 2006

3. Letter from T. H. Boyce (U. S. NRC) to members of the Technical Specification Task Force, dated December 6, 2005

4. Letter from T. J. Kobetz (U. S. NRC) to members of the Technical Specification Task Force, dated October 4, 2006

In accordance with 10 CFR 50.90, "Application for amendment of license or construction permit," Exelon Generation Company, LLC, (EGC) and AmerGen Energy Company, LLC (AmerGen) request the following amendment to Appendix A, Technical Specifications (TS), to the Facility Operating Licenses listed above. Specifically, the proposed changes will replace references to Section XI of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code with references to the ASME Code for Operation and Maintenance of Nuclear Power Plants (OM Code) in the applicable TS Section for the Inservice Testing Program for the EGC plants that have implemented industry Improved Technical Specifications (ITS).

These proposed changes are based on Technical Specification Task Force (TSTF) 479-A, Revision 0, "Changes to Reflect Revision of 10 CFR 50.55a," (Reference 1) as modified by TSTF-497, Revision 0, "Limit Inservice Testing Program SR 3.0.2 Application to Frequencies of 2 Years or Less" (Reference 2) and approved by the NRC in References 3 and 4. These proposed changes will correct or revise the TS Sections to align with the requirements of 10 CFR 50.55a, "Codes and standards," paragraph (f), "Inservice testing requirements." In addition to the replacement or deletion of the references, EGC is also adding a provision in the applicable TS Section to only apply the extension allowance of Surveillance Requirement (SR) 3.0.2 to the frequency table listed in the TS as part of the Inservice Testing Program and to normal and accelerated inservice testing frequencies of two years or less, as applicable.

For AmerGen plants with custom Technical Specifications (i.e., Three Mile Island, Unit 1 (TMI 1) and Oyster Creek Generating Station (OCGS)) references to the ASME Boiler and Pressure Vessel Code, Section XI as it relates to the Inservice Testing Program have been deleted. This change is consistent with the intent of TSTF-479-A discussed above.

Limerick Generating Station (LGS), Units 1 and 2, TS are based on industry Standard Technical Specifications (STS). For LGS, references to the ASME Boiler and Pressure Vessel Code, Section XI as it relates to the Inservice Testing Program have been deleted. The changes proposed for LGS are consistent with the intent of TSTF-479-A and TSTF-497 as discussed above.

EGC and AmerGen request approval of the proposed changes by July 29, 2008 with the amendment being implemented within 30 days of issuance for all plants with the exception of Braidwood Station. In the case of Braidwood Station, implementation of the amendment will coincide with implementation of the Third 10-Year IST Interval.

The attached amendment request is subdivided as shown below.

Attachment 1 provides an evaluation of the proposed changes.

Attachments 2A – 2H include the marked-up TS pages for each facility with the proposed changes indicated.

Attachments 3A – 3H include the revised TS pages with the proposed changes incorporated.

Attachments 4A – 4F include the marked-up TS Bases pages for each facility with the proposed changes indicated. Note that the TS Bases pages are provided for information only and do not require prior NRC approval.

The proposed changes have been reviewed by the Plant Operations Review Committees at each of the stations and approved by the Chairman of the Nuclear Safety Review Board in accordance with the requirements of the EGC and AmerGen Quality Assurance Program.

In accordance with 10 CFR 50.91, "Notice for public comment; State consultation," EGC and AmerGen are notifying the States of Illinois, Iowa, New Jersey (Bureau of Nuclear Engineering), and the Commonwealth of Pennsylvania of this application for changes to the TS by transmitting a copy of this letter and its attachments to the designated State Official.

There are no regulatory commitments contained in this letter. Should you have any questions concerning this letter, please contact Ms. Alison Mackellar at (630) 657-2817.

I declare under penalty of perjury that the foregoing is true and correct. Executed on the 8th day of August 2007.

Respectfully,

Darin M Benyak

Darin M. Benyak
Director – Licensing and Regulatory Affairs
Exelon Generation Company, LLC
AmerGen Energy Company, LLC

Attachment 1: Evaluation of the Proposed Changes

Attachments 2A – 2H: Mark-up of Proposed Technical Specifications Page Changes

Attachments 3A – 3H: Typed Pages for Technical Specifications Changes

Attachments 4A – 4F: Mark-up of Proposed Technical Specifications Bases Page Changes.

cc: Regional Administrator - NRC Region I
Regional Administrator - NRC Region III
NRC Project Manager, NRR - Braidwood Station
NRC Project Manager, NRR - Byron Station
NRC Project Manager, NRR - Dresden Nuclear Power Station
NRC Project Manager, NRR - Limerick Generating Station
NRC Project Manager, NRR - Oyster Creek Generating Station
NRC Project Manager, NRR - Peach Bottom Atomic Power Station
NRC Project Manager, NRR - Quad Cities Nuclear Power Station
NRC Project Manager, NRR - Three Mile Island Nuclear Station Unit 1
NRC Senior Resident Inspector - Braidwood Station
NRC Senior Resident Inspector - Byron Station
NRC Senior Resident Inspector - Dresden Nuclear Power Station
NRC Senior Resident Inspector - Limerick Generating Station
NRC Senior Resident Inspector - Oyster Creek Generating Station
NRC Senior Resident Inspector - Peach Bottom Atomic Power Station
NRC Senior Resident Inspector - Three Mile Island Nuclear Station Unit 1
NRC Senior Resident Inspector - Quad Cities Nuclear Power Station
Illinois Emergency Management Agency - Division of Nuclear Safety
Iowa Emergency Management Division
Director, Bureau of Radiation Protection - Pennsylvania Department of Environmental Resources
Director, Bureau of Nuclear Engineering, New Jersey Department of Environmental Protection
Chairman, Board of County Commissioners of Dauphin County, PA
Chairman, Board of Supervisors of Londonderry Township, PA
Mayor of Lacey Township, Forked River, NJ
R. I. McLean, State of Maryland
R. R. Janati, Commonwealth of Pennsylvania

**ATTACHMENT 1**
**EVALUATION OF PROPOSED CHANGES**

**INDEX**

## 1.0 DESCRIPTION

In accordance with 10 CFR 50.90, "Application for amendment of license or construction permit," Exelon Generation Company, LLC, (EGC) and AmerGen Energy Company, LLC (AmerGen) request the following amendment to Appendix A, Technical Specifications (TS), for the following operating licenses.

| Plant | Facility Operating License Nos. |
|---|---|
| Braidwood Station, Units 1 and 2 | NPF-72 and NPF-77 |
| Byron Station, Units 1 and 2 | NPF-37 and NPF-66 |
| Dresden Nuclear Power Station, Units 2 and 3 | DPR-19 and DPR-25 |
| Limerick Generating Station, Units 1 and 2 | NPF-39 and NPF-85 |
| Oyster Creek Generating Station | DPR-16 |
| Peach Bottom Atomic Power Station, Units 2 and 3 | DPR-44 and DPR-56 |
| Quad Cities Nuclear Power Station, Units 1 and 2 | DPR-29 and DPR-30 |
| Three Mile Island Nuclear Station, Unit 1 | DPR-50 |

Specifically, the proposed changes will replace references to Section XI of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code with references to the ASME Code for Operation and Maintenance of Nuclear Power Plants (OM Code) in the applicable TS Section for the Inservice Testing Program for the EGC plants that have implemented industry Improved Technical Specifications (ITS).

These proposed changes are based on Technical Specification Task Force (TSTF) 479-A, Revision 0, "Changes to Reflect Revision of 10 CFR 50.55a," (Reference 1) as modified by TSTF-497, Revision 0, "Limit Inservice Testing Program SR 3.0.2 Application to Frequencies of 2 Years or Less" (Reference 2) and approved by the NRC in References 3 and 4. These proposed changes will correct or revise the TS Sections to align with the requirements of 10 CFR 50.55a, "Codes and standards," paragraph (f), "Inservice testing requirements." In addition to the replacement or deletion of the references, EGC is also adding a provision in the applicable TS Section to only apply the extension allowance of Surveillance Requirement (SR) 3.0.2 to the frequency table listed in the TS as part of the Inservice Testing Program and to normal and accelerated inservice testing frequencies of two years or less, as applicable.

For AmerGen plants with custom Technical Specifications (i.e., Three Mile Island, Unit 1 (TMI 1) and Oyster Creek Generating Station (OCGS)) references to the ASME Boiler and Pressure Vessel Code, Section XI as it relates to the Inservice Testing Program have been deleted. This change is consistent with the intent of TSTF-479-A discussed above and also corrects the TS to align with the requirements of 10 CFR 50.55a, paragraph (f), "Inservice testing requirements."

Limerick Generating Station (LGS), Units 1 and 2, TS are based on industry Standard Technical Specifications (STS). For LGS, references to the ASME Boiler and Pressure Vessel Code, Section XI as it relates to the Inservice Testing Program have been deleted. The changes proposed for LGS are consistent with the intent of TSTF-479-A and TSTF-497 as discussed above and also correct the TS to align with the requirements of 10 CFR 50.55a, paragraph (f), "Inservice testing requirements."

## 2.0 PROPOSED CHANGES

EGC and AmerGen propose to revise the existing wording of the applicable Section of TS to replace references to the ASME Boiler and Pressure Vessel Code Section XI with references to the ASME OM Code for the EGC plants that have implemented industry ITS. For plants with STS or custom TS, references to the ASME Boiler and Pressure Vessel Code, Section XI as it relates to the Inservice Testing Program have been deleted. The applicable TS sections for each facility are listed below. The revised TS with changes indicated are presented in Attachments 2A through 2H.

OCGS and TMI 1 are in a custom format that does not have a distinct separate TS Bases Section, therefore the TS and TS Bases changes are indicated in Attachments 2G and 2H respectively.

**Braidwood Station**

TS 5.5.8 "Inservice Testing Program"

**Byron Station**

TS 5.5.8 "Inservice Testing Program"

**Dresden Nuclear Power Station**

TS 5.5.6 "Inservice Testing Program"

**Limerick Generating Station**

TS 3/4 "Surveillance Requirements"

**Oyster Creek Nuclear Generating Station**

TS 4.3 "Reactor Coolant"
TS 4.4 "Emergency Cooling"
TS 4.5 "Containment System"
TS 4.8 "Isolation Condenser"

**Peach Bottom Atomic Power Station**

TS 5.5.6 "Inservice Testing Program"

**Quad Cities Nuclear Power Station**

TS 5.5.6 "Inservice Testing Program"

**Three Mile Island Unit 1**

TS 4.2 "Reactor Coolant System Inservice Inspection and Testing"
TS 4.9 "Decay Heat Removal (DHR) Capability – Periodic Testing"

TS Bases sections will also be revised to replace or delete the references to Section XI of the ASME Boiler and Pressure Vessel Code as applicable. The affected TS Bases pages affected are contained in Attachments 3A through 3F. The TS Bases are provided for information only, and do not require NRC approval.

## 3.0 BACKGROUND

In 1990, the ASME published the initial edition of the ASME OM Code that provided rules for inservice testing of pumps and valves. The ASME OM Code replaced Section XI of the Boiler and Pressure Vessel Code for inservice testing of pumps and valves. The 1995 edition with the 1996 Addenda of the ASME OM Code (Reference 6) was incorporated by reference into 10 CFR 50.55a paragraph (b) on September 22, 1999 (Reference 5). 10 CFR 50.55a paragraph (f), "Inservice testing requirements," section (4)(ii) requires that inservice testing during successive 120-month intervals comply with the requirements of the latest edition and addenda of the Code incorporated by reference into 10 CFR 50.55a(b), 12 months before the start of the 120-month interval.

The ASME OM Code is the Code of record for the current 10-Year IST Interval for Byron Station, Units 1 and 2, Dresden Nuclear Power Station, Units 2 and 3, Limerick Generating Station, Units 1 and 2, Oyster Creek Generating Station, Peach Bottom Atomic Power Station, Units 2 and 3, Quad Cities Nuclear Power Station, Units 1 and 2, and Three Mile Island Unit 1. The ASME OM Code will become the Code of record for Braidwood Station, Units 1 and 2, upon initiation of their next 10-year IST interval.

Therefore the applicable TS Section must be revised to reflect the current ASME Code of record for all plants to correct this administrative oversight with the exception of Braidwood Station. In the case of Braidwood Station, the applicable TS Section must be revised to reflect the upcoming 10-Year Interval.

The following details the ASME OM Code for the current or upcoming interval for each facility.

**<u>Braidwood Station, Units 1 and 2</u>**

The 2001 edition with the 2003 Addenda of the ASME OM Code will be the ASME Code of record for the upcoming Third 10-Year IST Interval for Braidwood Station, Units 1 and 2. The Third 10-Year Intervals for Units 1 and 2 are scheduled to start on July 29, 2008.

**<u>Byron Station, Units 1 and 2</u>**

The 2001 edition with the 2003 Addenda of the ASME OM Code is the ASME Code of record for the current 10-Year IST Interval for Byron Station Units 1 and 2.

**<u>Dresden Nuclear Power Station, Units 2 and 3</u>**

The 1998 edition with the 2000 Addenda of the ASME OM Code is the ASME

Code of record for the current 10-Year IST Interval for Dresden Nuclear Power Station Units 2 and 3.

**Limerick Generating Station, Units 1 and 2**

The 1990 edition of the ASME OM Code is the ASME Code of record for the current 10-Year IST Interval for Limerick Generating Station Units 1 and 2.

**Oyster Creek Generating Station**

The 1995 edition with the 1996 Addenda of the ASME OM Code is the ASME Code of record for the current 10-Year IST Interval for Oyster Creek Generating Station.

**Peach Bottom Atomic Power Station, Units 2 and 3**

The 1990 edition of the ASME OM Code is the ASME Code of record for the current 10-Year IST Interval for Peach Bottom Atomic Power Station Units 2 and 3.

**Quad Cities Nuclear Power Station, Units 1 and 2**

The 1998 edition with the 2000 Addenda of the ASME OM Code is the ASME Code of record for the current 10-Year IST Interval for Quad Cities Nuclear Power Station Units 1 and 2.

**Three Mile Island Nuclear Station, Unit 1**

The 1998 edition with the 2000 Addenda of the ASME OM Code is the ASME Code of record for the current 10-Year IST Interval for TMI Unit 1.

For the ITS plants these proposed changes are consistent with TSTF 479-A, Revision 0, (Reference 1) that was approved by the NRC in Reference 3 and incorporated into NUREG-1431, "Standard Technical Specifications Westinghouse Plants," Revision 3.1 and NUREG-1433, "Standard Technical Specifications General Electric Plants – BWR/4," Revision 3.1.

On February 23, 2006 at a meeting between the TSTF and the NRC, the NRC stated that they did not agree with the portion of TSTF-479 referring to the application of a 25% IST interval extension for SR 3.0.2 to test frequencies and would not approve plant-specific amendments incorporating that portion of TSTF-479. Specifically, the NRC expressed a concern that frequency extensions may be applied to frequencies greater than two years and requested that the TSTF be revised to apply the provisions of SR 3.0.2 to the table listed in the TS as part of the Inservice Testing Program and to normal and accelerated inservice testing frequencies of two years or less. The NRC stated that they would accept applying SR 3.0.2 to IST Frequencies not listed in the Inservice Testing Program table provided that those Frequencies are specified in the Inservice Testing Program as 2 years or less.

On July 12, 2006, TSTF-497, Revision 0, (Reference 2) was submitted to reflect the revised NRC position. These proposed changes to the applicable TS Sections are based on TSTF 479-A, Revision 0, as modified by TSTF-497, Revision 0, which was approved by the NRC on October 4, 2006 (Reference 4).

## 4.0 TECHNICAL ANALYSIS

On September 22, 1999 the NRC amended 10 CFR 50.55a, "Codes and Standards," by Final Rule (64 FR 51370) (Reference 5) to incorporate by reference more recent editions and addenda of the ASME Boiler and Pressure Vessel Code and the ASME OM Code for construction, inservice inspection, and inservice testing. These provisions provide updated rules for the construction of components of light-water-cooled nuclear power plants, and for the inservice inspection and inservice testing of those components.

The 2001 edition and the 2002 and 2003 Addenda of the ASME OM Code was approved for use by the NRC and was incorporated by reference into 10 CFR 50.55a paragraph (b) on October 1, 2004 (Reference 7).

The ASME OM Code is the Code of record for the current 10-Year IST Interval for Byron Station, Units 1 and 2, Dresden Nuclear Power Station, Units 2 and 3, Limerick Generating Station, Units 1 and 2, Oyster Creek Generating Station, Peach Bottom Atomic Power Station, Units 2 and 3, Quad Cities Nuclear Power Station, Units 1 and 2, and Three Mile Island Unit 1. The ASME OM Code will become the Code of record for Braidwood Station, Units 1 and 2, upon initiation of their next 10-year IST interval.

The applicable TS Section currently reference the ASME Boiler and Pressure Vessel Code, Section XI, as the standard for testing frequencies and inservice testing of ASME Code Class 1, 2, and 3 pumps and valves. The proposed changes to the applicable TS Section will replace references to Section XI of the ASME Boiler and Pressure Vessel Code with references to the ASME OM Code as applicable to meet the requirements of 10 CFR 50.55a(f)(4), as amended in Reference 7.

OCGS and TMI 1 TS are in a custom format that does not coincide with the industry Improved Technical Specifications (ITS). For these stations the changes made are modeled after TSTF-479-A, Revision 0. Based on the administrative nature of this change, the change to the custom TS is considered acceptable as they meet the intent of TSTF-479-A, Revision 0.

LGS Units 1 and 2 STS are based on NUREG-0123, "Standard Technical Specifications for General Electric Boiling Water Reactors (BWR/5)," and are in a format that does not coincide with the industry ITS. For this station the changes made are modeled after TSTF-479-A, Revision 0 and TSTF-497 as applicable. Based on the administrative nature of this change, the change to the custom TS is considered acceptable as they meet the intent of TSTF-479-A, Revision 0 and TSTF-497, Revision 0.

## 5.0 REGULATORY ANALYSIS

### 5.1 No Significant Hazards Consideration

**Overview**

In accordance with 10 CFR 50.90, "Application for amendment of license or construction permit," Exelon Generation Company, LLC, (EGC) and AmerGen Energy Company, LLC (AmerGen) request the following amendment to Appendix A, Technical Specifications (TS), to the Facility Operating Licenses listed above. Specifically, the proposed changes will replace references to Section XI of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code with references to the ASME Code for Operation and Maintenance of Nuclear Power Plants (OM Code) in the applicable TS Section for the Inservice Testing Program for the EGC plants that have implemented industry Improved Technical Specifications (ITS).

These proposed changes are based on Technical Specification Task Force (TSTF) 479-A, Revision 0, "Changes to Reflect Revision of 10 CFR 50.55a," (Reference 1) as modified by TSTF-497, Revision 0, "Limit Inservice Testing Program SR 3.0.2 Application to Frequencies of 2 Years or Less" (Reference 2) and approved by the NRC in References 3 and 4. These proposed changes will correct or revise the TS Sections to align with the requirements of 10 CFR 50.55a, "Codes and standards," paragraph (f), "Inservice testing requirements." In addition to the replacement or deletion of the references, EGC is also adding a provision in the applicable TS Section to only apply the extension allowance of Surveillance Requirement (SR) 3.0.2 to the frequency table listed in the TS as part of the Inservice Testing Program and to normal and accelerated inservice testing frequencies of two years or less, as applicable.

For AmerGen plants with custom Technical Specifications (i.e., Three Mile Island, Unit 1 (TMI 1) and Oyster Creek Generating Station (OCGS)) references to the ASME Boiler and Pressure Vessel Code, Section XI as it relates to the Inservice Testing Program have been deleted. This change is consistent with the intent of TSTF-479-A discussed above.

Limerick Generating Station (LGS), Units 1 and 2, TS are based on industry Standard Technical Specifications (STS). For LGS, references to the ASME Boiler and Pressure Vessel Code, Section XI as it relates to the Inservice Testing Program have been deleted. The changes proposed for LGS are consistent with the intent of TSTF-479-A and TSTF-497 as discussed above.

According to 10 CFR 50.92, "Issuance of amendment," paragraph (c), a proposed amendment to an operating license involves no significant hazards consideration if operation of the facility in accordance with the proposed amendment would not:

(1) Involve a significant increase in the probability or consequences of an accident previously evaluated; or

(2) Create the possibility of a new or different kind of accident from any accident previously evaluated; or

(3) Involve a significant reduction in a margin of safety.

In support of this determination, an evaluation of each of the three criteria set forth in

10 CFR 50.92 is provided below regarding the proposed license amendment.

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

   Response: No

   The proposed changes revise the applicable TS Section to conform to the requirements of 10 CFR 50.55a, "Codes and standards," paragraph (f) regarding the inservice testing of pumps and valves. The current TS reference the ASME Boiler and Pressure Vessel Code, Section XI, requirements for the inservice testing of ASME Code Class 1, 2, and 3 pumps and valves. The proposed changes would reference the ASME OM Code as applicable, which is consistent with 10 CFR 50.55a, paragraph (f), "Inservice testing requirements." In addition, the proposed changes clarify that the extension allowance of SR 3.0.2 only applies to the frequency table listed in the TS, if applicable, as part of the Inservice Testing Program and to normal and accelerated inservice testing frequencies of two years or less. The definitions of the frequencies are not changed by this license amendment request.

   The proposed changes are administrative in nature, do not affect any accident initiators, do not affect the ability to successfully respond to previously evaluated accidents and do not affect radiological assumptions used in the evaluations. Thus, the probability or radiological consequences of any accident previously evaluated are not increased.

   Therefore, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

   Response: No

   The proposed changes revise the applicable TS Section to conform to the requirements of 10 CFR 50.55a(f) regarding the inservice testing of pumps and valves. The current TS Section references the ASME Boiler and Pressure Vessel Code, Section XI, requirements for the inservice testing of ASME Code Class 1, 2, and 3 pumps and valves. The proposed changes would reference the ASME OM Code as applicable, which is consistent with 10 CFR 50.55a(f). In addition, the proposed changes clarify that the extension allowance of SR 3.0.2 only applies to the frequency table listed in the TS, if applicable, as part of the Inservice Testing Program and to normal and accelerated inservice testing frequencies of two years or less. The definitions of the frequencies are not changed by this license amendment request.

   The proposed changes to the applicable TS Section do not affect the performance of any structure, system, or component credited with mitigating any accident previously evaluated and do not introduce any new modes of system operation or failure mechanisms.

Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any previously evaluated.

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

   Response: No

   The proposed changes revise the applicable TS Section for Braidwood Station Units 1 and 2, Byron Station Units 1 and 2, Dresden Nuclear Power Station Units 2 and 3, Limerick Generating Station Units 1 and 2, Oyster Creek Generating Station, Peach Bottom Atomic Power Station Units 2 and 3, Quad Cities Nuclear Power Station Units 1 and 2, and Three Mile Island Unit 1 to conform to the requirements of 10 CFR 50.55a(f) regarding the inservice testing of pumps and valves.

   The current TS Section references the ASME Boiler and Pressure Vessel Code, Section XI, requirements for the inservice testing of ASME Code Class 1, 2, and 3 pumps and valves. The proposed changes would reference the ASME OM Code as applicable, which is consistent with the 10 CFR 50.55a(f). In addition, the proposed changes clarify that the extension allowance of SR 3.0.2 only applies to the frequency table listed in the TS, if applicable, as part of the Inservice Testing Program and to normal and accelerated inservice testing frequencies of two years or less. The definitions of the frequencies are not changed by this license amendment request.

   The proposed changes do not modify the safety limits or setpoints at which protective actions are initiated and do not change the requirements governing operation or availability of safety equipment assumed to operate to preserve the margin of safety.

   Therefore, the proposed changes do not involve a significant reduction in a margin of safety.

Based on the above information, EGC and AmerGen conclude that the proposed amendment presents no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

## 5.2 Applicable Regulatory Requirements / Criteria

10 CFR 50.55a defines the requirements for applying industry Codes to a licensed boiling or pressurized water-cooled nuclear power facility. 10 CFR 50.55a(f)(4) requires that throughout the service life of a boiling or pressurized water-cooled nuclear power facility, pumps and valves which are classified as ASME Code Class 1, Class 2 and Class 3, must meet the inservice test requirements that are incorporated by reference in 10 CFR 50.55a(b) to the extent practical within the limitations of design, geometry and materials of construction of the components.

10 CFR 50.55a(f)(4)(ii) further states that inservice tests to verify operational readiness of pumps and valves, whose function is required for safety, conducted during successive 120-month intervals must comply with the latest edition and addenda of the Code, incorporated by reference in 10 CFR 50.55a(b), 12 months before the start of the 120-month interval.

10 CFR 50.55a(f)(5)(ii) also states that if a revised inservice test program for a facility conflicts with the TS for the facility, the licensee shall apply to the NRC for amendment of the TS to conform the TS to the revised program. This application shall be submitted at least six months before the start of the period during which the provisions become applicable.

EGC and AmerGen have identified that the implementation of the Byron Station, Units 1 and 2, Dresden Nuclear Power Station, Units 2 and 3, Limerick Generating Station, Units 1 and 2, Oyster Creek Generating Station, Peach Bottom Atomic Power Station, Units 2 and 3, Quad Cities Nuclear Power Station, Units 1 and 2, and Three Mile Island Unit 1 current 10-Year IST Interval Program does not reflect the requirements specified in the applicable Section of TS. Therefore, in accordance with the requirements of 10 CFR 50.55a(f)(5)(ii), EGC and AmerGen are submitting this License Amendment Request to correct this administrative oversight.

TS 5.5.8 for Braidwood Station, Units 1 and 2 currently reference the ASME Boiler and Pressure Vessel Code, Section XI, as the standard for testing frequencies and inservice testing of ASME Code Class 1, 2, and 3 pumps and valves. The ASME Code of record for the upcoming Third 10-Year IST Interval Program for Braidwood Station Units 1 and 2 will be the ASME OM Code 2001 Edition through the 2003 Addenda.
The proposed changes to TS 5.5.8 will replace references to Section XI of the ASME Boiler and Pressure Vessel Code with references to the ASME OM Code to conform the TS to the updated Inservice Testing Program to meet the requirements of the Code currently endorsed by reference in 10 CFR 50.55a(f)(4), as amended in Reference 8, for the upcoming Braidwood Units 1 and 2 Third 10-Year IST intervals.

## 6.0 ENVIRONMENTAL EVALUATION

A review has determined that the proposed amendment would not change a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, "Standards for Protection Against Radiation," and would not change an inspection or surveillance requirement. The proposed amendment does not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluent that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22, "Criterion for categorical exclusion; identification of licensing and regulatory actions eligible for categorical exclusion or otherwise not requiring environmental review," paragraph (c)(9).

## 7.0 REFERENCES

1. TSTF-479-A, Revision 0, "Changes to Reflect Revision of 10 CFR 50.55a," dated December 19, 2005

2. TSTF-497, Revision 0, "Limit Inservice Testing Program SR 3.0.2 Application to Frequencies of 2 Years or Less," dated July 12, 2006

3. Letter from T. H. Boyce (U. S. NRC) to members of the Technical Specification Task Force, dated December 6, 2005

4. Letter from T. J. Kobetz (U. S. NRC) to members of the Technical Specification Task Force, dated October 4, 2006

5. Federal Register, Volume 64, Number 183, "10 CFR Part 50 - Industry Codes and Standards; Amended Requirements," dated September 22, 1999

6. American Society of Mechanical Engineers (ASME), "Operation and Maintenance of Nuclear Power Plants (OM Code)," 1995 Edition through the 1996 Addenda

7. Federal Register, Volume 69, Number 190, "10 CFR Part 50 - Industry Codes and Standards; Amended Requirements," dated October 1, 2004

8. American Society of Mechanical Engineers (ASME), "Operation and Maintenance of Nuclear Power Plants (OM Code)," 2001 Edition through the 2003 Addenda

9. American Society of Mechanical Engineers (ASME), "Operation and Maintenance of Nuclear Power Plants (OM Code)," 1998 Edition through the 2000 Addenda

10. American Society of Mechanical Engineers (ASME), "Operation and Maintenance of Nuclear Power Plants (OM Code)," 1990 Edition

**ATTACHMENT 2A**

BRAIDWOOD STATION
UNITS 1 and 2

Docket Nos. STN 50-456 and STN 50-457

License Nos. NPF-72 and NPF-77

**Markup of Proposed Technical Specifications Page Changes**

REVISED TS PAGES

5.5-6

## 5.5 Programs and Manuals

### 5.5.8 Inservice Testing Program

This program provides controls for inservice testing of ASME Code Class 1, 2, and 3 components. The program shall include the following:

a. Testing frequencies ~~specified in Section XI of~~ applicable to the ASME ~~Boiler and Pressure Vessel Code~~ Code for Operation and Maintenance of Nuclear Power Plants (ASME OM Code) and applicable Addenda as follows:

| ASME ~~Boiler and Pressure Vessel~~ OM Code and applicable Addenda terminology for inservice testing activities | Required Frequencies for performing inservice testing activities |
|---|---|
| Weekly | At least once per 7 days |
| Monthly | At least once per 31 days |
| Quarterly or every 3 months | At least once per 92 days |
| Semiannually or every 6 months | At least once per 184 days |
| Every 9 months | At least once per 276 days |
| Yearly or annually | At least once per 366 days |
| Biennially or every 2 years | At least once per 731 days; |

b. The provisions of SR 3.0.2 are applicable to the above required Frequencies and to other normal and accelerated Frequencies specified as 2 years or less in the Inservice Testing Program for performing inservice testing activities;

c. The provisions of SR 3.0.3 are applicable to inservice testing activities; and

d. Nothing in the ASME ~~Boiler and Pressure Vessel~~ OM Code shall be construed to supersede the requirements of any Technical Specification.

INSERT

**ATTACHMENT 2B**

BYRON STATION
UNITS 1 and 2

Docket Nos. 50-454 and 50-455

License Nos. NPF-37 and NPF-66

**Markup of Proposed Technical Specifications Page Changes**

<u>REVISED TS PAGES</u>

5.5-6

5.5 Programs and Manuals

5.5.8 Inservice Testing Program

This program provides controls for inservice testing of ASME Code Class 1, 2, and 3 components. The program shall include the following:

applicable to

a. Testing frequencies ~~specified in Section XI~~ of the ASME ~~Boiler and Pressure Vessel Code~~ and applicable Addenda as follows:

Code for Operation and Maintenance of Nuclear Power Plants (ASME OM Code)

OM

| ASME ~~Boiler and Pressure Vessel~~ Code and applicable Addenda terminology for inservice testing activities | Required Frequencies for performing inservice testing activities |
|---|---|
| Weekly | At least once per 7 days |
| Monthly | At least once per 31 days |
| Quarterly or every 3 months | At least once per 92 days |
| Semiannually or every 6 months | At least once per 184 days |
| Every 9 months | At least once per 276 days |
| Yearly or annually | At least once per 366 days |
| Biennially or every 2 years | At least once per 731 days; |

b. The provisions of SR 3.0.2 are applicable to the above required Frequencies for performing inservice testing activities;

c. The provisions of SR 3.0.3 are applicable to inservice testing activities; and

OM Code

d. Nothing in the ASME ~~Boiler and Pressure Vessel Code~~ shall be construed to supersede the requirements of any Technical Specification.

INSERT

and to other normal and accelerated Frequencies specified as 2 years or less in the Inservice Testing Program

**ATTACHMENT 2C**

DRESDEN NUCLEAR POWER STATION
UNITS 2 and 3

Docket Nos. 50-237 and 50-249

License Nos. DPR-19 and DPR-25

**Markup of Proposed Technical Specifications Page Changes**

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5.5-4
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5.5 Programs and Manuals

5.5.4 Radioactive Effluent Controls Program (continued)

1. For noble gases: a dose rate ≤ 500 mrems/yr to the whole body and a dose rate ≤ 3000 mrems/yr to the skin, and

2. For iodine-131, iodine-133, tritium, and all radionuclides in particulate form with half-lives greater than 8 days: a dose rate ≤ 1500 mrems/yr to any organ;

h. Limitations on the annual and quarterly air doses resulting from noble gases released in gaseous effluents from each unit to areas beyond the site boundary, conforming to 10 CFR 50, Appendix I;

i. Limitations on the annual and quarterly doses to a member of the public from iodine-131, iodine-133, tritium, and all radionuclides in particulate form with half lives > 8 days in gaseous effluents released from each unit to areas beyond the site boundary, conforming to 10 CFR 50, Appendix I; and

j. Limitations on the annual dose or dose commitment to any member of the public, beyond the site boundary, due to releases of radioactivity and to radiation from uranium fuel cycle sources, conforming to 40 CFR 190.

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Radioactive Effluents Control Program Surveillance Frequencies.

5.5.5 Component Cyclic or Transient Limit

This program provides controls to track the UFSAR Section 3.9, cyclic and transient occurrences to ensure that components are maintained within the design limits.

5.5.6 Inservice Testing Program

This program provides controls for inservice testing of ASME Code Class 1, 2, and 3 pumps and valves.

a. Testing Frequencies applicable to ~~specified in Section XI of~~ the ASME Code for Operation and Maintenance of Nuclear Power Plants (ASME OM Code) ~~Boiler and Pressure Vessel Code~~ and applicable Addenda are as follows:

(continued)

5.5 Programs and Manuals

5.5.6 Inservice Testing Program (continued)

| ASME OM ~~Boiler and Pressure Vessel~~ Code and applicable Addenda terminology for inservice testing activities | Required Frequencies for performing inservice testing activities |
|---|---|
| Weekly | At least once per 7 days |
| Monthly | At least once per 31 days |
| Quarterly or every 3 months | At least once per 92 days |
| Semiannually or every 6 months | At least once per 184 days |
| Every 9 months | At least once per 276 days |
| Yearly or annually | At least once per 366 days |
| Biennially or every 2 years | At least once per 731 days |
| Every 48 months | At least once per 1461 days |

And to other normal and accelerated Frequencies specified as 2 years or less in the Inservice Testing Program

INSERT

b. The provisions of SR 3.0.2 are applicable to the above required Frequencies for performing inservice testing activities;

c. The provisions of SR 3.0.3 are applicable to inservice testing activities; and

d. Nothing in the ASME OM ~~Boiler and Pressure Vessel~~ Code shall be construed to supersede the requirements of any TS.

5.5.7 Ventilation Filter Testing Program (VFTP)

The VFTP shall establish the required testing of Engineered Safety Feature (ESF) filter ventilation systems. Tests described in Specification 5.5.7.a and 5.5.7.b shall be performed once per 24 months; after each complete or partial replacement of the HEPA filter bank or charcoal adsorber bank; after any structural maintenance on the HEPA filter bank or charcoal adsorber bank housing; and, following painting, fire, or chemical release in any ventilation zone communicating with the subsystem while it is in operation that could adversely affect the filter bank or charcoal adsorber capability.

(continued)

**ATTACHMENT 2D**

LIMERICK GENERATING STATION
UNITS 1 and 2

Docket Nos. 50-352 and 50-353

License Nos. NPF-39 and NPF-85

**Markup of Proposed Technical Specifications Page Changes**

<u>REVISED TS PAGES</u>

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<u>UNIT 2</u>
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APPLICABILITY

SURVEILLANCE REQUIREMENTS

4.0.1 Surveillance Requirements shall be met during the OPERATIONAL CONDITIONS or other specified conditions in the Applicability for individual Limiting Conditions for Operation, unless otherwise stated in the Surveillance Requirement. Failure to meet a Surveillance, whether such failure is experienced during the performance of the Surveillance or between performances of the Surveillance, shall be failure to meet the Limiting Condition for Operation. Failure to perform a Surveillance within the specified Surveillance time interval and allowed extension per Specification 4.0.2, shall be failure to meet the Limiting Condition for Operation except as provided in Specification 4.0.3. Surveillances do not have to be performed on inoperable equipment or variables outside specified limits.

4.0.2 Each Surveillance Requirement shall be performed within the specified surveillance time interval with a maximum allowable extension not to exceed 25% of the surveillance interval.

4.0.3 If it is discovered that a Surveillance was not performed within its specified Surveillance time interval and allowed extension per Specification 4.0.2, then compliance with the requirement to declare the Limiting Condition for Operation not met may be delayed, from the time of discovery, up to 24 hours or up to the limit of the specified Surveillance time interval, whichever is greater. This delay period is permitted to allow performance of the Surveillance. A risk evaluation shall be performed for any Surveillance delayed greater than 24 hours and the risk impact shall be managed.

If the Surveillance is not performed within the delay period, the Limiting Condition for Operation must immediately be declared not met, and the applicable ACTION requirements must be entered.

When the Surveillance is performed within the delay period and the Surveillance is not met, the Limiting Condition for Operation must immediately be declared not met, and the applicable ACTION requirements must be entered.

4.0.4 Entry into an OPERATIONAL CONDITION or other specified condition in the Applicability of a Limiting Condition for Operation shall only be made when the Limiting Condition for Operation's Surveillance Requirements have been met within their Surveillance time interval, except as provided in Specification 4.0.3. When a Limiting Condition for Operation is not met due to its Surveillance Requirements not having been met, entry into an OPERATIONAL CONDITION or other specified condition in the Applicability shall only be made in accordance with Specification 3.0.4.

This provision shall not prevent entry into OPERATIONAL CONDITIONS or other specified conditions in the Applicability that are required to comply with ACTION requirements or that are part of a shutdown of the unit.

4.0.5 Surveillance Requirements for inservice inspection and testing of ASME Code Class 1, 2, & 3 components shall be applicable as follows:

INSERT A

a. Inservice inspection of ASME Code Class 1, 2, and 3 components and inservice testing of ASME Code Class 1, 2, and 3 pumps and valves shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda ~~as required by 10 CFR Part 50, Section 50.55a.~~

INSERT B

b. Surveillance intervals specified in Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda for the inservice inspection ~~and testing~~ activities ~~required by the ASME Boiler and Pressure Vessel Code and applicable Addenda shall be applicable as follows in these Technical Specifications:~~

APPLICABILITY

SURVEILLANCE REQUIREMENTS (Continued)

| ASME Boiler and Pressure Vessel Code and applicable Addenda terminology for inservice inspection and testing activities | Required frequencies for performing inservice inspection and testing activities |
|---|---|
| Weekly | At least once per 7 days |
| Monthly | At least once per 31 days |
| Quarterly or every 3 months | At least once per 92 days |
| Semiannually or every 6 months | At least once per 184 days |
| Every 9 months | At least once per 276 days |
| Yearly or annually | At least once per 366 days |
| Biennially or every 2 years | At least once per 731 days |

c. The provisions of Specification 4.0.2 are applicable to the above required frequencies for performing inservice inspection and testing activities.

d. Performance of the above inservice inspection and testing activities shall be in addition to other specified Surveillance Requirements.

e. Nothing in the ASME Boiler and Pressure Vessel Code shall be construed to supersede the requirements of any Technical Specification.

f. The Inservice Inspection (ISI) Program for piping identified in NRC Generic Letter 88-01 shall be performed in accordance with the staff positions on schedule, methods and personnel, and sample expansion included in the Generic Letter, or in accordance with alternate measures approved by the NRC staff. Details for implementation of these requirements are included as augmented inspection requirements in the ISI Program.

In addition, the provisions of Specification 4.0.2 are applicable to other normal and accelerated frequencies specified as 2 years or less in the Inservice Testing Program for performing inservice testing activities.

APPLICABILITY

SURVEILLANCE REQUIREMENTS

4.0.1 Surveillance Requirements shall be met during the OPERATIONAL CONDITIONS or other specified conditions in the Applicability for individual Limiting Conditions for Operation, unless otherwise stated in the Surveillance Requirement. Failure to meet a Surveillance, whether such failure is experienced during the performance of the Surveillance or between performances of the Surveillance, shall be failure to meet the Limiting Condition for Operation. Failure to perform a Surveillance within the specified Surveillance time interval and allowed extension per Specification 4.0.2, shall be failure to meet the Limiting Condition for Operation except as provided in Specification 4.0.3. Surveillances do not have to be performed on inoperable equipment or variables outside specified limits.

4.0.2 Each Surveillance Requirement shall be performed within the specified surveillance time interval with a maximum allowable extension not to exceed 25% of the surveillance interval.

4.0.3 If it is discovered that a Surveillance was not performed within its specified Surveillance time interval and allowed extension per Specification 4.0.2, then compliance with the requirement to declare the Limiting Condition for Operation not met may be delayed, from the time of discovery, up to 24 hours or up to the limit of the specified Surveillance time interval, whichever is greater. This delay period is permitted to allow performance of the Surveillance. A risk evaluation shall be performed for any Surveillance delayed greater than 24 hours and the risk impact shall be managed.

If the Surveillance is not performed within the delay period, the Limiting Condition for Operation must immediately be declared not met, and the applicable ACTION requirements must be entered.

When the Surveillance is performed within the delay period and the Surveillance is not met, the Limiting Condition for Operation must immediately be declared not met, and the applicable ACTION requirements must be entered.

4.0.4 Entry into an OPERATIONAL CONDITION or other specified condition in the Applicability of a Limiting Condition for Operation shall only be made when the Limiting Condition for Operation's Surveillance Requirements have been met within their Surveillance time interval, except as provided in Specification 4.0.3. When a Limiting Condition for Operation is not met due to its Surveillance Requirements not having been met, entry into an OPERATIONAL CONDITION or other specified condition in the Applicability shall only be made in accordance with Specification 3.0.4.

This provision shall not prevent entry into OPERATIONAL CONDITIONS or other specified conditions in the Applicability that are required to comply with ACTION requirements or that are part of a shutdown of the unit.

4.0.5 Surveillance Requirements for inservice inspection and testing of ASME Code Class 1, 2, & 3 components shall be applicable as follows:

INSERT A

a. Inservice inspection of ASME Code Class 1, 2, and 3 components and inservice testing of ASME Code Class 1, 2, and 3 pumps and valves shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable ~~Addenda as~~ required by 10 CFR Part 50, Section 50.55a.

INSERT B

b. Surveillance intervals specified in Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda for the inservice inspection and testing activities required by the ASME Boiler and Pressure Vessel ~~Code and~~ applicable Addenda shall be applicable as ~~follows in these Technical~~ Specifications:

SURVEILLANCE REQUIREMENTS (Continued)

| ASME Boiler and Pressure Vessel Code and applicable Addenda terminology for inservice inspection and testing activities | Required frequencies for performing inservice inspection and testing activities |
|---|---|
| Weekly | At least once per 7 days |
| Monthly | At least once per 31 days |
| Quarterly or every 3 months | At least once per 92 days |
| Semiannually or every 6 months | At least once per 184 days |
| Every 9 months | At least once per 276 days |
| Yearly or annually | At least once per 366 days |
| Biennially or every 2 years | At least once per 731 days |

c. The provisions of Specification 4.0.2 are applicable to the above required frequencies for performing inservice inspection and testing activities.

d. Performance of the above inservice inspection and testing activities shall be in addition to other specified Surveillance Requirements.

e. Nothing in the ASME Boiler and Pressure Vessel Code shall be construed to supersede the requirements of any Technical Specification.

f. The Inservice Inspection (ISI) Program for piping identified in NRC Generic Letter 88-01 shall be performed in accordance with the staff positions on schedule, methods and personnel, and sample expansion included in the Generic Letter, or in accordance with alternate measures approved by the NRC staff. Details for implementation of these requirements are included as augmented inspection requirements in the ISI Program.

In addition, the provisions of Specification 4.0.2 are applicable to other normal and accelerated frequencies specified as 2 years or less in the Inservice Testing Program for performing inservice testing activities.

INSERT A

Inservice inspection of ASME Code Class 1, 2, and 3 components shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR Part 50, Section 50.55a. Inservice testing of ASME Code Class 1, 2, and 3 pumps and valves shall be performed in accordance with the ASME Code for Operation and Maintenance of Nuclear Power Plants (ASME OM Code) and applicable Addenda as required by 10 CFR Part 50, Section 50.55a.

INSERT B

, and the ASME Code for Operation and Maintenance of Nuclear Power Plants (ASME OM Code) and applicable Addenda for inservice testing activities,

**ATTACHMENT 2E**

PEACH BOTTOM ATOMIC POWER STATION
UNITS 2 and 3

Docket Nos. 50-277 and 50-278

License Nos. DPR-44 and DPR-56

**Markup of Proposed Technical Specifications Page Changes**

<u>REVISED TS PAGES</u>

<u>UNIT 2</u>
5.0-11

<u>UNIT 3</u>
5.0-11

applicable to the ASME Code for Operation and Maintenance of Nuclear Power Plants (ASME OM Code)

5.5 Programs and Manuals (continued)

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5.5.6 Inservice Testing Program

This program provides controls for inservice testing of ASME Code Class 1, 2, and 3 components. The program shall include the following:

a. Testing frequencies specified in Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda are as follows:

OM

| ASME Boiler and Pressure Vessel Code and applicable Addenda terminology for inservice testing activities | Required Frequencies for performing inservice testing activities |
|---|---|
| Weekly | At least once per 7 days |
| Monthly | At least once per 31 days |
| Quarterly or every 3 months | At least once per 92 days |
| Semiannually or every 6 months | At least once per 184 days |
| Every 9 months | At least once per 276 days |
| Yearly or annually | At least once per 366 days |
| Biennially or every 2 years | At least once per 732 days |

b. The provisions of SR 3.0.2 are applicable to the Frequencies for performing inservice testing activities;

the above required Frequencies and to other normal and accelerated Frequencies specified as 2 years or less in the Inservice Testing Program

c. The provisions of SR 3.0.3 are applicable to inservice testing activities; and

OM

d. Nothing in the ASME Boiler and Pressure Vessel Code shall be construed to supersede the requirements of any TS.

5.5.7 Ventilation Filter Testing Program (VFTP)

The VFTP shall establish the required testing of Engineered Safety Feature (ESF) filter ventilation systems.

Tests described in Specifications 5.5.7.a, 5.5.7.b, and 5.5.7.c shall be performed:

(continued)

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applicable to the ASME Code for Operation and Maintenance of Nuclear Power Plants (ASME OM Code)

5.5 Programs and Manuals (continued)

5.5.6 Inservice Testing Program

This program provides controls for inservice testing of ASME Code Class 1, 2, and 3 components. The program shall include the following:

a. Testing frequencies specified in Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda are as follows:

OM

| ASME Boiler and Pressure Vessel Code and applicable Addenda terminology for inservice testing activities | Required Frequencies for performing inservice testing activities |
|---|---|
| Weekly | At least once per 7 days |
| Monthly | At least once per 31 days |
| Quarterly or every 3 months | At least once per 92 days |
| Semiannually or every 6 months | At least once per 184 days |
| Every 9 months | At least once per 276 days |
| Yearly or annually | At least once per 366 days |
| Biennially or every 2 years | At least once per 732 days |

the above required Frequencies and to other normal and accelerated Frequencies specified as 2 years or less in the Inservice Testing Program

b. The provisions of SR 3.0.2 are applicable to the Frequencies for performing inservice testing activities;

c. The provisions of SR 3.0.3 are applicable to inservice testing activities; and

OM

d. Nothing in the ASME Boiler and Pressure Vessel Code shall be construed to supersede the requirements of any TS.

5.5.7 Ventilation Filter Testing Program (VFTP)

The VFTP shall establish the required testing of Engineered Safety Feature (ESF) filter ventilation systems.

Tests described in Specifications 5.5.7.a, 5.5.7.b, and 5.5.7.c shall be performed:

(continued)

**ATTACHMENT 2F**

QUAD CITIES NUCLEAR POWER STATION
UNITS 1 and 2

Docket Nos. 50-254 and 50-265

License Nos. DPR-29 and DPR-30

**Markup of Proposed Technical Specifications Page Changes**

<u>REVISED TS PAGES</u>

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5.5 Programs and Manuals

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5.5.4 Radioactive Effluent Controls Program (continued)

1. For noble gases: a dose rate ≤ 500 mrems/yr to the whole body and a dose rate ≤ 3000 mrems/yr to the skin, and

2. For iodine-131, iodine-133, tritium, and all radionuclides in particulate form with half-lives greater than 8 days: a dose rate ≤ 1500 mrems/yr to any organ;

h. Limitations on the annual and quarterly air doses resulting from noble gases released in gaseous effluents from each unit to areas beyond the site boundary, conforming to 10 CFR 50, Appendix I;

i. Limitations on the annual and quarterly doses to a member of the public from iodine-131, iodine-133, tritium, and all radionuclides in particulate form with half lives > 8 days in gaseous effluents released from each unit to areas beyond the site boundary, conforming to 10 CFR 50, Appendix I; and

j. Limitations on the annual dose or dose commitment to any member of the public, beyond the site boundary, due to releases of radioactivity and to radiation from uranium fuel cycle sources, conforming to 40 CFR 190.

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Radioactive Effluents Control Program Surveillance Frequencies.

5.5.5 Component Cyclic or Transient Limit

This program provides controls to track the UFSAR Section 3.9, cyclic and transient occurrences to ensure that components are maintained within the design limits.

5.5.6 Inservice Testing Program

This program provides controls for inservice testing of ASME Code Class 1, 2, and 3 pumps and valves.

applicable to

Code for Operation and Maintenance of Nuclear Power Plants (ASME OM Code)

a. Testing Frequencies ~~specified in Section XI of~~ the ASME ~~Boiler and Pressure Vessel Code~~ and applicable Addenda are as follows:

(continued)

## 5.5 Programs and Manuals

5.5.6 Inservice Testing Program (continued)

| ASME ~~Boiler and Pressure Vessel~~ OM Code and applicable Addenda terminology for inservice testing activities | Required Frequencies for performing inservice testing activities |
|---|---|
| Weekly | At least once per 7 days |
| Monthly | At least once per 31 days |
| Quarterly or every 3 months | At least once per 92 days |
| Semiannually or every 6 months | At least once per 184 days |
| Every 9 months | At least once per 276 days |
| Yearly or annually | At least once per 366 days |
| Biennially or every 2 years | At least once per 731 days |
| Every 48 months | At least once per 1461 days |

INSERT
and to other normal and accelerated Frequencies specified as 2 years or less in the Inservice Testing Program

b. The provisions of SR 3.0.2 are applicable to the above required Frequencies for performing inservice testing activities;

c. The provisions of SR 3.0.3 are applicable to inservice testing activities; and

d. Nothing in the ASME ~~Boiler and Pressure Vessel~~ OM Code shall be construed to supersede the requirements of any TS.

5.5.7 Ventilation Filter Testing Program (VFTP)

The VFTP shall establish the required testing of Engineered Safety Feature (ESF) filter ventilation systems. Tests described in Specification 5.5.7.a and 5.5.7.b shall be performed once per 24 months; after each complete or partial replacement of the HEPA filter bank or charcoal adsorber bank; after any structural maintenance on the HEPA filter bank or charcoal adsorber bank housing; and, following painting, fire, or chemical release in any ventilation zone communicating with the subsystem while it is in operation that could adversely affect the filter bank or charcoal adsorber capability.

(continued)

**ATTACHMENT 2G**

OYSTER CREEK GENERATING STATION
UNITS 1 and 2

Docket No. 50-219

License No. DPR-16

**Markup of Proposed Technical Specifications and Bases Page Changes**

<u>REVISED TS and TS BASES PAGES</u>

4.3-1
4.3-3
4.4-2
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4.8 -1

## 4.3 REACTOR COOLANT

Applicability: Applies to the surveillance requirements for the reactor coolant system.

Objective: To determine the condition of the reactor coolant system and the operation of the safety devices related to it.

Specification:

A. Materials surveillance specimens and neutron flux monitors shall be installed in the reactor vessel adjacent to the wall at the midplane of the active core. Specimens and monitors shall be periodically removed, tested, and evaluated to determine the effects of neutron fluence on the fracture toughness of the vessel shell materials. The results of these evaluations shall be used to assess the adequacy of the P-T curves A, B, and C in Figures 3.3.1, 3.3.2 and 3.3.3. New curves shall be generated as required.

B. Inservice inspection of ASME Code Class 1, Class 2 and Class 3 systems and components shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR, Section 50.55a(g), except where specific written relief has been granted by the NRC pursuant to 10 CFR, Section 50.55a(g)(6)(i).

C. Inservice testing of ASME Code Class 1, Class ~~2 and~~ Class 3 pumps and valves shall be performed in accordance with ~~Section XI of the ASME Boiler and Pressure Vessel Code~~ and applicable Addenda as required by 10 CFR, Section 50.55a(f), except where specific written relief has been granted by the NRC pursuant to 10 CFR, Section 50.55a(f)(6)(i).

D. A visual examination for leaks shall be made with the reactor coolant system at pressure during each scheduled refueling outage or after major repairs have been made to the reactor coolant system in accordance with Article 5000, Section XI. The requirements of specification 3.3.A shall be met during the test.

E. Each replacement safety valve or valve that has been repaired shall be tested in accordance with Specification C above. Setpoints shall be as follows:

| Number of Valves | Set Points (psig) |
|---|---|
| 4 | 1212 ± 36 |
| 5 | 1221 ± 36 |

F. A sample of reactor coolant shall be analyzed at least every 72 hours for the purpose of determining the content of chloride ion and to check the conductivity.

the ASME Code for Operation and Maintenance of Nuclear Power Plants (ASME OM Code) the existence of ASME Section XI. For this reason, the degree of access required by ASME Section XI is not generally available and will be addressed as "requests for relief" in accordance with 10 CFR 50.55a(g).

Experience in safety valve operation shows ~~testing~~ in accordance with Section XI of the ASME Boiler and Pressure Vessel Code is adequate to detect failures or deterioration. ~~The as-found setpoint tolerance value is specified in Section XI of the ASME Code at ± 3% of design pressure.~~ An analysis has been performed which shows that with all safety valves set 36 psig higher, the safety limit of 1375 psig is not exceeded.

Conductivity instruments continuously monitor the reactor coolant. Experience indicates that a check of the conductivity instrumentation at least every 72 hours is adequate to ensure accurate readings. The reactor water sample will also be used to determine the chloride ion content to assure that the limits of 3.3.E are not exceeded. The chloride ion content will not change rapidly over a period of several days; therefore, the sampling frequency is adequate.

The as-found setpoint tolerence is specified in the ASME OM code as the owner-defined tolerence or ±3% of valve nameplate set pressure

| Item | Frequency |
|---|---|
| C. Containment Cooling System | |
| 2. Motor-operated valve operability | Every 3 months |
| 3. Pump compartment water-tight doors closed | Once/week and after each entry |
| D. Emergency Service Water System | |
| 1. Pump Operability | Once/3 months. Also after major maintenance and prior to startup following a refueling outage. |
| E. Control Rod Drive Hydraulic System | |
| 1. Pump Operability | Once/month. Also after major maintenance and prior to startup following a refueling outage. |
| F. Fire Protection System | |
| 1. Pump Operability | Once/month. Also after major maintenance and prior to startup following a refueling outage. |
| 2. Isolation valve operability | Once/3 months. Also after major maintenance and prior to startup following a refueling outage. |

Bases:

It is during major maintenance or repair that a system's design intent may be violated accidentally. Therefore, a functional test is required after every major maintenance operation. During an extended outage, such as a refueling outage, major repair and maintenance may be performed on many systems. To be sure that these repairs on other systems do not encroach unintentionally on critical standby cooling systems, they should be given a functional test prior to startup.

Motor operated pumps, valves and other active devices that are normally on standby should be exercised periodically to make sure that they are free to operate. Motors on pumps should operate long enough to approach equilibrium temperature to ensure there is no overheat problem. the Whenever practical, valves should be stroked full length to ensure that nothing impedes their motion. Testing of components per OC Inservice Testing Program in accordance with ~~ASME~~ Code Section XI once every 3 months provides assurances of the availability of the system. The Control Rod Hydraulic pumps and Fire Protection System pumps are not part of the Inservice Test Program per ASME Section XI and will continue to be tested for operability once per month. the Engineering judgment based on experience and availability analyses of the type presented in Appendix L of the FDSAR indicates that testing these components more often than once a month over a long period of time does not significantly improve the system reliability. Also, at this frequency of testing wearout should not be a problem through the life of the plant.

Code

A Primary Containment Leakage Rate Testing Program has been established to implement the requirements of 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. Guidance for implementation of Option B is contained in NRC Regulatory Guide 1.163, "Performance Based Containment Leak Test Program", Revision 0, dated September 1995. Additional guidance for NRC Regulatory Guide 1.163 is contained in Nuclear Energy Institute (NEI) 94-01, "Industry Guideline for Implementing Performance Based Option of 10 CFR 50, Appendix J," Revision 0, dated July 26, 1995, and ANSI/ANS 56.8-1994, "Containment System Leakage Testing Requirements." The Primary Containment Leakage Rate Testing Program conforms with this guidance as modified by approved exemptions.

The maximum allowable leakage rate for the primary containment ($L_a$) is 1.0 percent by weight of the containment air per 24 hours at the design basis LOCA maximum peak containment pressure ($P_a$). As discussed below, $P_a$ for the purpose of containment leak rate testing is 35 psig.

The penetration and air purge piping leakage test frequency, along with the containment leak rate tests, is adequate to allow detection of leakage trends. Whenever a double gasketed penetration (primary containment head equipment hatches and the absorption chamber access hatch) is broken and remade, the space between the gaskets is pressurized to determine that the seals are performing properly. The test pressure of 35 psig is consistent with the accident analyses and the maximum preoperational leak rate test pressure.

Monitoring the nitrogen makeup requirements of the inerting system provides a method of observing leak rate trends and would detect gross leaks in a very short time. This equipment must be periodically removed from service for test and maintenance, but this out-of-service time be kept to a practical minimum.

Automatic primary containment isolation valves are provided to maintain PRIMARY CONTAINMENT INTEGRITY following the design basis loss-of-coolant accident. Closure times for the automatic primary containment isolation valves are not critical because it is on the order of minutes before significant fission product release to the containment atmosphere for the design basis loss of coolant accident. These valves are highly reliable, see infrequent service and most of them are normally in the closed position. Therefore, a test during each REFUELING OUTAGE is sufficient.

Large lines connecting to the reactor coolant system, whose failure could result in uncovering the reactor core, are supplied with automatic isolation valves (except containment cooling). Closure times restrict coolant loss from the circumferential rupture of any of these lines outside primary containment to less than that for a main steam line break (the design basis accident for outside containment line breaks). The minimum time for main steam isolation valve (MSIV) closure of 3 seconds is based on the transient analysis that shows the pressure peak 76 psig below the lowest safety valve setting. The maximum time for MSIV closure of 10 seconds is based on the value assumed for the main steam line break dose calculations and restricts coolant loss to prevent uncovering the reactor core. Per ASME Boiler and Pressure Vessel Code, Section XI the full closure test of the MSIVs during COLD SHUTDOWNs will ensure OPERABILITY and provide assurance that the valves maintain the required closing time. The provision for a minimum of 92 days between the tests ensures that full closure testing is not too frequent. The MSIVs are partially stroked quarterly as part of reactor protection system instrument surveillance testing.

the

## 4.8 ISOLATION CONDENSER

Applicability: Applies to periodic testing requirements for the isolation condenser system.

Objective: To verify the operability of the isolation condenser system.

Specification: A. Surveillance of each isolation condenser loop shall be as follows:

| Item | Frequency |
|---|---|
| 1. Operability of motor-operated isolation valves and condensate makeup valves. | Once/3 months |
| 2. Automatic actuation and functional test. | Each refueling outage (interval not to exceed 20 months) or following major repair. |
| 3. Shell side water volume check | Once/day |
| 4. Isolation valve (steam side) | |
| a. Visual inspection | Each refueling outage |
| b. External leakage check | Each primary system Leak test |
| c. Area temperature check | Once/shift |

Basis: Motor-operated valves on the isolation condenser steam and condensate lines and on the condensate makeup line that are normally on standby should be exercised periodically to make sure that they are free to operate. The valves will be stroked full length every time they are tested to verify proper functional performance. This frequency of OM testing is consistent with instrumentation tests discussed in Specification 4.1. Testing of these components per ASME ~~section XI~~ Code once every 3 months provides assurance of availability of the system. Also, at this frequency of testing, wearout should not be a problem throughout the life of the plant.

The automatic actuation and functional test will demonstrate the automatic opening of the condensate return line valves and the automatic closing of the isolation valves on the vent lines to the main steam lines. Automatic closure of the isolation condenser steam and condensate lines on actuation of the condenser pipe break detectors will also be verified by the test. It is during a major maintenance or repair that a system's design intent may be violated accidentally. This makes the functional test necessary after every major repair operation.

By virtue of normal plant operation the operators daily observe the water level in the isolation condensers. In addition, isolation condenser shell side water level sensors provide control room annunciation of condenser high or low water level.

**ATTACHMENT 2H**

THREE MILE ISLAND
UNITS 1

Docket No. 50-289

License No. DPR-50

**Markup of Proposed Technical Specifications and Bases Page Changes**

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Bases

The limitation on power operation with one idle RC pump in each loop has been imposed since the ECCS cooling performance has not been calculated in accordance with the Final Acceptance Criteria requirements specifically for this mode of reactor operation. A time period of 24 hours is allowed for operation with one idle RC pump in each loop to effect repairs of the idle pump(s) and to return the reactor to an acceptable combination of operating RC pumps. The 24 hours for this mode of operation is acceptable since this mode is expected to have considerable margin for the peak cladding temperature limit and since the likelihood of a LOCA within the 24-hour period is considered very remote.

A reactor coolant pump or decay heat removal pump is required to be in operation before the boron concentration is reduced by dilution with makeup water. Either pump will provide mixing which will prevent sudden positive reactivity changes caused by dilute coolant reaching the reactor. One decay heat removal pump will circulate the equivalent of the reactor coolant system volume in one-half hour or less.

The decay heat removal system suction piping is designed for 300°F and 370 psig; thus, the system can remove decay heat when the reactor coolant system is below this temperature (References 1, 2, and 3).

Both steam generators must be operable before heatup of the Reactor Coolant System to insure system integrity against leakage under normal and transient conditions. Only one steam generator is required for decay heat removal purposes.

One pressurizer code safety valve is capable of preventing overpressurization when the reactor is not critical since its relieving capacity is greater than that required by the sum of the available heat sources which are pump energy, pressurizer heaters, and reactor decay heat. Both pressurizer code safety valves are required to be in service prior to criticality to conform to the system design relief capabilities. The code safety valves prevent overpressure for a rod withdrawal or feedwater line break accidents (Reference 4). The pressurizer code safety valve lift set point shall be set at 2500 psig $\pm 1\%$ allowance for error. **Surveillance requirements are specified in the Inservice Testing Program. Pressurizer code safety valve setpoint drift of up to 3% is acceptable in accordance with ASME Section XI (Reference 5) and the assumptions of TMI-1 safety analysis.**

the (Reference 5)

References


(1) UFSAR, Tables 9.5-1 and 9.5-2

(2) UFSAR, Sections 4.2.5.1 and 9.5 – "Decay Heat Removal"

(3) UFSAR, Section 4.2.5.4 – "Secondary System"

(4) UFSAR, Section 4.3.10.4 – "System Minimum Operational Components"

(5) UFSAR, Section 4.3.7 – "Overpressure Protection"

## 4.2 REACTOR COOLANT SYSTEM INSERVICE INSPECTION AND TESTING

### Applicability

This technical specification applies to the inservice inspection (ISI) and inservice testing (IST) of the reactor coolant system pressure boundary and portions of other safety oriented system pressure boundaries.

### Objective

The objective of the ISI and IST programs is to provide assurance of the continuing integrity of the reactor coolant system while at the same time minimizing radiation exposure to personnel in the performance of inservice inspections and tests.

### Specification

4.2.1 ISI of ASME Code Class 1, Class 2, and Class 3 components shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR 50, Section 50.55a(g), except where specific written relief has been granted by the NRC.

4.2.2 IST of ASME Code Class 1, Class 2 and Class 3 pumps and valves shall be performed in accordance with ~~Section XI of the ASME Boiler and Pressure Vessel Code~~ and applicable Addenda as ~~required by 10 CFR~~ 50, Section 50.55a(f), except where specific written relief has been granted by the NRC.

4.2.3 (Deleted)

4.2.4 The accessible portions of one reactor coolant pump motor flywheel assembly will be ultrasonically inspected within the first ISI period, two reactor coolant pump motor flywheel assemblies within the first two ISI periods and all four by the end of the 10 year inspection interval. However, the U.T. procedure is developmental and will be used only to the extent that it is shown to be meaningful. The extent of coverage will be limited to those areas of the flywheel which are accessible without motor disassembly, i.e., can be reached through the access ports. Also, if radiation levels at the lower access ports are prohibitive, only the upper access ports will be used.

the ASME Code for Operation and Maintenance of Nuclear Power Plants (ASME OM Code)



Amendment No. ~~13~~, ~~29~~, ~~54~~, ~~69~~, ~~71~~, ~~118~~, 172

Bases

Specifications 4.2.1 and 2 ensure that inservice inspection of ASME Code Class 1, 2 and 3 components and inservice testing of ASME Code Class 1, 2 and 3 pumps and valves will be performed in accordance with a periodically updated version of Section XI of the ASME Boiler and Pressure Vessel Code and Addenda as required by 10 CFR 50, paragraphs 55a(f) and 55a(g). Relief from any of the above requirements has been provided in writing by the NRC and is not a part of these technical specifications.

50.55a

4.3 DELETED

## 4.9 DECAY HEAT REMOVAL (DHR) CAPABILITY - PERIODIC TESTING

Applicability

Applies to the periodic testing of systems or components which function to remove decay heat.

Objective

To verify that systems/components required for DHR are capable of performing their design function.

Specification

4.9.1 Reactor Coolant System (RCS) Temperature greater than 250 degrees F.

4.9.1.1 Verify each Emergency Feedwater (EFW) Pump is tested in accordance with the requirements and acceptance criteria of the ASME Section XI Inservice Test Program.

Note: This surveillance is not required to be performed for the turbine-driven EFW Pump (EF-P-1) until 24 hours after exceeding 750 psig.

4.9.1.2 DELETED

4.9.1.3 At least once per 31 days, each EFW System flowpath valve from both Condensate Storage Tanks (CSTs) to the OTSGs via the motor-driven pumps and the turbine-driven pump shall be verified to be in the required status.

4.9.1.4 On a refueling interval basis:

a) Verify that each EFW Pump starts automatically upon receipt of an EFW test signal.

b) Verify that each EFW control valve responds upon receipt of an EFW test signal.

c) Verify that each EFW control valve responds in manual control from the control room and remote shutdown panel.

4.9.1.5 Prior to STARTUP, following a REFUELING SHUTDOWN or a COLD SHUTDOWN greater than 30 days, conduct a test to demonstrate that the motor driven EFW Pumps can pump water from the CSTs to the Steam Generators.

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4.9 DECAY HEAT REMOVAL (DHR) CAPABILITY-PERIODIC TESTING (Continued)

4.9.1.6 Acceptance Criteria

These tests shall be considered satisfactory if control board indication and visual observation of the equipment demonstrates that all components have operated properly except for the tests required by Specification 4.9.1.1.

4.9.2 RCS Temperature less than or equal to 250 degrees F.*

4.9.2.1 On a daily basis, verify operability of the means for DHR required by Specification 3.4.2 by observation of console status indication.

* These requirements supplement the requirements of Specifications 4.5.2.2 and 4.5.4.

Bases

The Code

~~1.~~ ASME Section XI specifies requirements and acceptance standards for the testing of nuclear safety ~~related~~ pumps. The quarterly EFW Pump test frequency specified by the ASME Section XI Code will be sufficient to verify that the turbine-driven and both motor-driven EFW Pumps are operable. Compliance with the normal acceptance criteria assures that the EFW Pumps are operating as expected. The surveillance requirements ensure that the overall EFW System functional capability is maintained.

Deferral of the requirement to perform IST on the turbine-driven EFW Pump is necessary to assure sufficient OTSG pressure to perform the test using Main Steam.

Daily verification of the operability of the required means for DHR ensures that sufficient DHR capability will be maintained.

**ATTACHMENT 3A**

BRAIDWOOD STATION
UNITS 1 and 2

Docket Nos. STN 50-456 and STN 50-457

License Nos. NPF-72 and NPF-77

**Typed Pages**

**for**

**Technical Specifications Changes**

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5.5 Programs and Manuals

5.5.8 Inservice Testing Program

This program provides controls for inservice testing of ASME Code Class 1, 2, and 3 components. The program shall include the following:

a. Testing frequencies applicable to the ASME Code for Operation and Maintenance of Nuclear Power Plants (ASME OM Code) and applicable Addenda as follows:

| ASME OM Code and applicable Addenda terminology for inservice testing activities | Required Frequencies for performing inservice testing activities |
|---|---|
| Weekly | At least once per 7 days |
| Monthly | At least once per 31 days |
| Quarterly or every 3 months | At least once per 92 days |
| Semiannually or every 6 months | At least once per 184 days |
| Every 9 months | At least once per 276 days |
| Yearly or annually | At least once per 366 days |
| Biennially or every 2 years | At least once per 731 days; |

b. The provisions of SR 3.0.2 are applicable to the above required Frequencies and to other normal and accelerated Frequencies specified as 2 years or less in the Inservice Testing Program for performing inservice testing activities;

c. The provisions of SR 3.0.3 are applicable to inservice testing activities; and

d. Nothing in the ASME OM Code shall be construed to supersede the requirements of any Technical Specification.

**ATTACHMENT 3B**

BYRON STATION
UNITS 1 and 2

Docket Nos. 50-454 and 50-455

License Nos. NPF-37 and NPF-66

**Typed Pages**

**for**

**Technical Specifications Changes**

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5.5 Programs and Manuals

5.5.8 Inservice Testing Program

This program provides controls for inservice testing of ASME Code Class 1, 2, and 3 components. The program shall include the following:

a. Testing frequencies applicable to the ASME Code for Operation and Maintenance of Nuclear Power Plants (ASME OM Code) and applicable Addenda as follows:

| ASME OM Code and applicable Addenda terminology for inservice testing activities | Required Frequencies for performing inservice testing activities |
|---|---|
| Weekly | At least once per 7 days |
| Monthly | At least once per 31 days |
| Quarterly or every 3 months | At least once per 92 days |
| Semiannually or every 6 months | At least once per 184 days |
| Every 9 months | At least once per 276 days |
| Yearly or annually | At least once per 366 days |
| Biennially or every 2 years | At least once per 731 days; |

b. The provisions of SR 3.0.2 are applicable to the above required Frequencies and to other normal and accelerated Frequencies specified as 2 years or less in the Inservice Testing Program for performing inservice testing activities;

c. The provisions of SR 3.0.3 are applicable to inservice testing activities; and

d. Nothing in the ASME OM Code shall be construed to supersede the requirements of any Technical Specification.

**ATTACHMENT 3C**

DRESDEN NUCLEAR POWER STATION
UNITS 2 and 3

Docket Nos. 50-237 and 50-249

License Nos. DPR-19 and DPR-25

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**for**

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5.5 Programs and Manuals

5.5.4 Radioactive Effluent Controls Program (continued)

1. For noble gases: a dose rate ≤ 500 mrems/yr to the whole body and a dose rate ≤ 3000 mrems/yr to the skin, and

2. For iodine-131, iodine-133, tritium, and all radionuclides in particulate form with half-lives greater than 8 days: a dose rate ≤ 1500 mrems/yr to any organ;

h. Limitations on the annual and quarterly air doses resulting from noble gases released in gaseous effluents from each unit to areas beyond the site boundary, conforming to 10 CFR 50, Appendix I;

i. Limitations on the annual and quarterly doses to a member of the public from iodine-131, iodine-133, tritium, and all radionuclides in particulate form with half lives > 8 days in gaseous effluents released from each unit to areas beyond the site boundary, conforming to 10 CFR 50, Appendix I; and

j. Limitations on the annual dose or dose commitment to any member of the public, beyond the site boundary, due to releases of radioactivity and to radiation from uranium fuel cycle sources, conforming to 40 CFR 190.

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Radioactive Effluents Control Program Surveillance Frequencies.

5.5.5 Component Cyclic or Transient Limit

This program provides controls to track the UFSAR Section 3.9, cyclic and transient occurrences to ensure that components are maintained within the design limits.

5.5.6 Inservice Testing Program

This program provides controls for inservice testing of ASME Code Class 1, 2, and 3 pumps and valves.

a. Testing Frequencies applicable to the ASME Code for Operation and Maintenance of Nuclear Power Plants (ASME OM Code) and applicable Addenda are as follows:

(continued)

5.5 Programs and Manuals

5.5.6 Inservice Testing Program (continued)

| ASME OM Code and applicable Addenda terminology for inservice testing activities | Required Frequencies for performing inservice testing activities |
|---|---|
| Weekly | At least once per 7 days |
| Monthly | At least once per 31 days |
| Quarterly or every 3 months | At least once per 92 days |
| Semiannually or every 6 months | At least once per 184 days |
| Every 9 months | At least once per 276 days |
| Yearly or annually | At least once per 366 days |
| Biennially or every 2 years | At least once per 731 days |
| Every 48 months | At least once per 1461 days |

b. The provisions of SR 3.0.2 are applicable to the above required Frequencies and to other normal and accelerated Frequencies specified as 2 years or less in the Inservice Testing Program for performing inservice testing activities;

c. The provisions of SR 3.0.3 are applicable to inservice testing activities; and

d. Nothing in the ASME OM Code shall be construed to supersede the requirements of any TS.

5.5.7 Ventilation Filter Testing Program (VFTP)

The VFTP shall establish the required testing of Engineered Safety Feature (ESF) filter ventilation systems. Tests described in Specification 5.5.7.a and 5.5.7.b shall be performed once per 24 months; after each complete or partial replacement of the HEPA filter bank or charcoal adsorber bank; after any structural maintenance on the HEPA filter bank or charcoal adsorber bank housing; and, following painting, fire, or chemical release in any ventilation zone communicating with the subsystem while it is in operation that could adversely affect the filter bank or charcoal adsorber capability.

(continued)

**ATTACHMENT 3D**

LIMERICK GENERATING STATION
UNITS 1 and 2

Docket Nos. 50-352 and 50-353

License Nos. NPF-39 and NPF-85

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APPLICABILITY

SURVEILLANCE REQUIREMENTS

4.0.1 Surveillance Requirements shall be met during the OPERATIONAL CONDITIONS or other specified conditions in the Applicability for individual Limiting Conditions for Operation, unless otherwise stated in the Surveillance Requirement. Failure to meet a Surveillance, whether such failure is experienced during the performance of the Surveillance or between performances of the Surveillance, shall be failure to meet the Limiting Condition for Operation. Failure to perform a Surveillance within the specified Surveillance time interval and allowed extension per Specification 4.0.2, shall be failure to meet the Limiting Condition for Operation except as provided in Specification 4.0.3. Surveillances do not have to be performed on inoperable equipment or variables outside specified limits.

4.0.2 Each Surveillance Requirement shall be performed within the specified surveillance time interval with a maximum allowable extension not to exceed 25% of the surveillance interval.

4.0.3 If it is discovered that a Surveillance was not performed within its specified Surveillance time interval and allowed extension per Specification 4.0.2, then compliance with the requirement to declare the Limiting Condition for Operation not met may be delayed, from the time of discovery, up to 24 hours or up to the limit of the specified Surveillance time interval, whichever is greater. This delay period is permitted to allow performance of the Surveillance. A risk evaluation shall be performed for any Surveillance delayed greater than 24 hours and the risk impact shall be managed.

If the Surveillance is not performed within the delay period, the Limiting Condition for Operation must immediately be declared not met, and the applicable ACTION requirements must be entered.

When the Surveillance is performed within the delay period and the Surveillance is not met, the Limiting Condition for Operation must immediately be declared not met, and the applicable ACTION requirements must be entered.

4.0.4 Entry into an OPERATIONAL CONDITION or other specified condition in the Applicability of a Limiting Condition for Operation shall only be made when the Limiting Condition for Operation's Surveillance Requirements have been met within their Surveillance time interval, except as provided in Specification 4.0.3. When a Limiting Condition for Operation is not met due to its Surveillance Requirements not having been met, entry into an OPERATIONAL CONDITION or other specified condition in the Applicability shall only be made in accordance with Specification 3.0.4.

This provision shall not prevent entry into OPERATIONAL CONDITIONS or other specified conditions in the Applicability that are required to comply with ACTION requirements or that are part of a shutdown of the unit.

4.0.5 Surveillance Requirements for inservice inspection and testing of ASME Code Class 1, 2, & 3 components shall be applicable as follows:

a. Inservice inspection of ASME Code Class 1, 2, and 3 components shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR Part 50, Section 50.55a. Inservice testing of ASME Code Class 1, 2, and 3 pumps and valves shall be performed in accordance with the ASME Code for Operation and Maintenance of Nuclear Power Plants (ASME OM Code) and applicable Addenda as required by 10 CFR Part 50, Section 50.55a.

APPLICABILITY

SURVEILLANCE REQUIREMENTS (Continued)

b. Surveillance intervals specified in Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda for the inservice inspection activities, and the ASME Code for Operation and Maintenance of Nuclear Power Plants (ASME OM Code) and applicable Addenda for inservice testing activities, shall be applicable as follows in these Technical Specifications:

| ASME Code and applicable Addenda terminology for inservice inspection and testing activities | Required frequencies for performing inservice inspection and testing activities |
|---|---|
| Weekly | At least once per 7 days |
| Monthly | At least once per 31 days |
| Quarterly or every 3 months | At least once per 92 days |
| Semiannually or every 6 months | At least once per 184 days |
| Every 9 months | At least once per 276 days |
| Yearly or annually | At least once per 366 days |
| Biennially or every 2 years | At least once per 731 days |

c. The provisions of Specification 4.0.2 are applicable to the above required frequencies for performing inservice inspection and testing activities. In addition, the provision of Specification 4.0.2 are applicable to other normal and accelerated frequencies specified as 2 years or less in the Inservice Testing Program for performing inservice testing activities.

d. Performance of the above inservice inspection and testing activities shall be in addition to other specified Surveillance Requirements.

e. Nothing in the ASME Code shall be construed to supersede the requirements of any Technical Specification.

f. The Inservice Inspection (ISI) Program for piping identified in NRC Generic Letter 88-01 shall be performed in accordance with the staff positions on schedule, methods and personnel, and sample expansion included in the Generic Letter, or in accordance with alternate measures approved by the NRC staff. Details for implementation of these requirements are included as augmented inspection requirements in the ISI Program.

APPLICABILITY

SURVEILLANCE REQUIREMENTS

4.0.1 Surveillance Requirements shall be met during the OPERATIONAL CONDITIONS or other specified conditions in the Applicability for individual Limiting Conditions for Operation, unless otherwise stated in the Surveillance Requirement. Failure to meet a Surveillance, whether such failure is experienced during the performance of the Surveillance or between performances of the Surveillance, shall be failure to meet the Limiting Condition for Operation. Failure to perform a Surveillance within the specified Surveillance time interval and allowed extension per Specification 4.0.2, shall be failure to meet the Limiting Condition for Operation except as provided in Specification 4.0.3. Surveillances do not have to be performed on inoperable equipment or variables outside specified limits.

4.0.2 Each Surveillance Requirement shall be performed within the specified surveillance time interval with a maximum allowable extension not to exceed 25% of the surveillance interval.

4.0.3 If it is discovered that a Surveillance was not performed within its specified Surveillance time interval and allowed extension per Specification 4.0.2, then compliance with the requirement to declare the Limiting Condition for Operation not met may be delayed, from the time of discovery, up to 24 hours or up to the limit of the specified Surveillance time interval, whichever is greater. This delay period is permitted to allow performance of the Surveillance. A risk evaluation shall be performed for any Surveillance delayed greater than 24 hours and the risk impact shall be managed.

If the Surveillance is not performed within the delay period, the Limiting Condition for Operation must immediately be declared not met, and the applicable ACTION requirements must be entered.

When the Surveillance is performed within the delay period and the Surveillance is not met, the Limiting Condition for Operation must immediately be declared not met, and the applicable ACTION requirements must be entered.

4.0.4 Entry into an OPERATIONAL CONDITION or other specified condition in the Applicability of a Limiting Condition for Operation shall only be made when the Limiting Condition for Operation's Surveillance Requirements have been met within their Surveillance time interval, except as provided in Specification 4.0.3. When a Limiting Condition for Operation is not met due to its Surveillance Requirements not having been met, entry into an OPERATIONAL CONDITION or other specified condition in the Applicability shall only be made in accordance with Specification 3.0.4.

This provision shall not prevent entry into OPERATIONAL CONDITIONS or other specified conditions in the Applicability that are required to comply with ACTION requirements or that are part of a shutdown of the unit.

4.0.5 Surveillance Requirements for inservice inspection and testing of ASME Code Class 1, 2, & 3 components shall be applicable as follows:

a. Inservice inspection of ASME Code Class 1, 2, and 3 components shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR Part 50, Section 50.55a. Inservice testing of ASME Code Class 1, 2, and 3 pumps and valves shall be performed in accordance with the ASME Code for Operation and Maintenance of Nuclear Power Plants (ASME OM Code) and applicable Addenda as required by 10 CFR Part 50, Section 50.55a.

SURVEILLANCE REQUIREMENTS (Continued)

b. Surveillance intervals specified in Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda for the inservice inspection activities, and the ASME Code for Operation and Maintenance of Nuclear Power Plants (ASME OM Code) and applicable Addenda for inservice testing activities, shall be applicable as follows in these Technical Specifications:

| ASME Code and applicable Addenda terminology for inservice inspection and testing activities | Required frequencies for performing inservice inspection and testing activities |
|---|---|
| Weekly | At least once per 7 days |
| Monthly | At least once per 31 days |
| Quarterly or every 3 months | At least once per 92 days |
| Semiannually or every 6 months | At least once per 184 days |
| Every 9 months | At least once per 276 days |
| Yearly or annually | At least once per 366 days |
| Biennially or every 2 years | At least once per 731 days |

c. The provisions of Specification 4.0.2 are applicable to the above required frequencies for performing inservice inspection and testing activities. In addition, the provision of Specification 4.0.2 are applicable to other normal and accelerated frequencies specified as 2 years or less in the Inservice Testing Program for performing inservice testing activities.

d. Performance of the above inservice inspection and testing activities shall be in addition to other specified Surveillance Requirements.

e. Nothing in the ASME Code shall be construed to supersede the requirements of any Technical Specification.

f. The Inservice Inspection (ISI) Program for piping identified in NRC Generic Letter 88-01 shall be performed in accordance with the staff positions on schedule, methods and personnel, and sample expansion included in the Generic Letter, or in accordance with alternate measures approved by the NRC staff. Details for implementation of these requirements are included as augmented inspection requirements in the ISI Program.

**ATTACHMENT 3E**

PEACH BOTTOM ATOMIC POWER STATION
UNITS 2 and 3

Docket Nos. 50-277 and 50-278

License Nos. DPR-44 and DPR-56

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**for**

**Technical Specifications Changes**

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5.5 Programs and Manuals (continued)

5.5.6 Inservice Testing Program

This program provides controls for inservice testing of ASME Code Class 1, 2, and 3 components. The program shall include the following:

a. Testing frequencies applicable to the ASME Code for Operation and Maintenance of Nuclear Power Plants (ASME OM Code) and applicable Addenda are as follows:

| ASME OM Code and applicable Addenda terminology for inservice testing activities | Required Frequencies for performing inservice testing activities |
|---|---|
| Weekly | At least once per 7 days |
| Monthly | At least once per 31 days |
| Quarterly or every 3 months | At least once per 92 days |
| Semiannually or every 6 months | At least once per 184 days |
| Every 9 months | At least once per 276 days |
| Yearly or annually | At least once per 366 days |
| Biennially or every 2 years | At least once per 732 days |

b. The provisions of SR 3.0.2 are applicable to the above required Frequencies and to other normal and accelerated Frequencies specified as 2 years or less in the Inservice Testing Program for performing inservice testing activities;

c. The provisions of SR 3.0.3 are applicable to inservice testing activities; and

d. Nothing in the ASME OM Code shall be construed to supersede the requirements of any TS.

5.5.7 Ventilation Filter Testing Program (VFTP)

The VFTP shall establish the required testing of Engineered Safety Feature (ESF) filter ventilation systems.

Tests described in Specifications 5.5.7.a, 5.5.7.b, and 5.5.7.c shall be performed:

(continued)

5.5 Programs and Manuals (continued)

5.5.6 Inservice Testing Program

This program provides controls for inservice testing of ASME Code Class 1, 2, and 3 components. The program shall include the following:

a. Testing frequencies applicable to the ASME Code for Operation and Maintenance of Nuclear Power Plants (ASME OM Code) and applicable Addenda are as follows:

| ASME OM Code and applicable Addenda terminology for inservice testing activities | Required Frequencies for performing inservice testing activities |
|---|---|
| Weekly | At least once per 7 days |
| Monthly | At least once per 31 days |
| Quarterly or every 3 months | At least once per 92 days |
| Semiannually or every 6 months | At least once per 184 days |
| Every 9 months | At least once per 276 days |
| Yearly or annually | At least once per 366 days |
| Biennially or every 2 years | At least once per 732 days |

b. The provisions of SR 3.0.2 are applicable to the above required Frequencies and to other normal and accelerated Frequencies specified as 2 years or less in the Inservice Testing Program for performing inservice testing activities;

c. The provisions of SR 3.0.3 are applicable to inservice testing activities; and

d. Nothing in the ASME OM Code shall be construed to supersede the requirements of any TS.

5.5.7 Ventilation Filter Testing Program (VFTP)

The VFTP shall establish the required testing of Engineered Safety Feature (ESF) filter ventilation systems.

Tests described in Specifications 5.5.7.a, 5.5.7.b, and 5.5.7.c shall be performed:

(continued)

**ATTACHMENT 3F**

QUAD CITIES NUCLEAR POWER STATION
UNITS 1 and 2

Docket Nos. 50-254 and 50-265

License Nos. DPR-29 and DPR-30

**Typed Pages**

**for**

**Technical Specifications Changes**

REVISED TS PAGES

5.5-4
5.5-5

5.5 Programs and Manuals

5.5.4 Radioactive Effluent Controls Program (continued)

1. For noble gases: a dose rate ≤ 500 mrems/yr to the whole body and a dose rate ≤ 3000 mrems/yr to the skin, and

2. For iodine-131, iodine-133, tritium, and all radionuclides in particulate form with half-lives greater than 8 days: a dose rate ≤ 1500 mrems/yr to any organ;

h. Limitations on the annual and quarterly air doses resulting from noble gases released in gaseous effluents from each unit to areas beyond the site boundary, conforming to 10 CFR 50, Appendix I;

i. Limitations on the annual and quarterly doses to a member of the public from iodine-131, iodine-133, tritium, and all radionuclides in particulate form with half lives > 8 days in gaseous effluents released from each unit to areas beyond the site boundary, conforming to 10 CFR 50, Appendix I; and

j. Limitations on the annual dose or dose commitment to any member of the public, beyond the site boundary, due to releases of radioactivity and to radiation from uranium fuel cycle sources, conforming to 40 CFR 190.

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Radioactive Effluents Control Program Surveillance Frequencies.

5.5.5 Component Cyclic or Transient Limit

This program provides controls to track the UFSAR Section 3.9, cyclic and transient occurrences to ensure that components are maintained within the design limits.

5.5.6 Inservice Testing Program

This program provides controls for inservice testing of ASME Code Class 1, 2, and 3 pumps and valves.

a. Testing Frequencies applicable to the ASME Code for Operation and Maintenance of Nuclear Power Plants (ASME OM Code) and applicable Addenda are as follows:

(continued)

5.5 Programs and Manuals

5.5.6 Inservice Testing Program (continued)

| ASME OM Code and applicable Addenda terminology for inservice testing activities | Required Frequencies for performing inservice testing activities |
|---|---|
| Weekly | At least once per 7 days |
| Monthly | At least once per 31 days |
| Quarterly or every 3 months | At least once per 92 days |
| Semiannually or every 6 months | At least once per 184 days |
| Every 9 months | At least once per 276 days |
| Yearly or annually | At least once per 366 days |
| Biennially or every 2 years | At least once per 731 days |
| Every 48 months | At least once per 1461 days |

b. The provisions of SR 3.0.2 are applicable to the above required Frequencies and to other normal and accelerated Frequencies specified as 2 years or less in the Inservice Testing Program for performing inservice testing activities;

c. The provisions of SR 3.0.3 are applicable to inservice testing activities; and

d. Nothing in the ASME OM Code shall be construed to supersede the requirements of any TS.

5.5.7 Ventilation Filter Testing Program (VFTP)

The VFTP shall establish the required testing of Engineered Safety Feature (ESF) filter ventilation systems. Tests described in Specification 5.5.7.a and 5.5.7.b shall be performed once per 24 months; after each complete or partial replacement of the HEPA filter bank or charcoal adsorber bank; after any structural maintenance on the HEPA filter bank or charcoal adsorber bank housing; and, following painting, fire, or chemical release in any ventilation zone communicating with the subsystem while it is in operation that could adversely affect the filter bank or charcoal adsorber capability.

(continued)

**ATTACHMENT 3G**

OYSTER CREEK GENERATING STATION
UNITS 1 and 2

Docket No. 50-219

License No. DPR-16

**Typed Pages**

**for**

**Technical Specifications and Bases Changes**

<u>REVISED TS and TS BASES PAGES</u>

4.3-1
4.3-3
4.4-2
4.5-11
4.8 -1

## 4.3 REACTOR COOLANT

Applicability: Applies to the surveillance requirements for the reactor coolant system.

Objective: To determine the condition of the reactor coolant system and the operation of the safety devices related to it.

Specification:

A. Materials surveillance specimens and neutron flux monitors shall be installed in the reactor vessel adjacent to the wall at the midplane of the active core. Specimens and monitors shall be periodically removed, tested, and evaluated to determine the effects of neutron fluence on the fracture toughness of the vessel shell materials. The results of these evaluations shall be used to assess the adequacy of the P-T curves A, B, and C in Figures 3.3.1, 3.3.2 and 3.3.3. New curves shall be generated as required.

B. Inservice inspection of ASME Code Class 1, Class 2 and Class 3 systems and components shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR, Section 50.55a, except where specific written relief has been granted by the NRC pursuant to 10 CFR, Section 50.55a.

C. Inservice testing of ASME Code Class 1, Class 2 and Class 3 pumps and valves shall be performed in accordance with the ASME Code for Operation and Maintenance of Nuclear Power Plants (ASME OM Code) and applicable Addenda as required by 10 CFR, Section 50.55a, except where specific written relief has been granted by the NRC pursuant to 10 CFR, Section 50.55a.

D. A visual examination for leaks shall be made with the reactor coolant system at pressure during each scheduled refueling outage or after major repairs have been made to the reactor coolant system in accordance with Article 5000, Section XI. The requirements of specification 3.3.A shall be met during the test.

E. Each replacement safety valve or valve that has been repaired shall be tested in accordance with Specification C above. Setpoints shall be as follows:

| Number of Valves | Set Points (psig) |
|---|---|
| 4 | 1212 ± 36 |
| 5 | 1221 ± 36 |

F. A sample of reactor coolant shall be analyzed at least every 72 hours for the purpose of determining the content of chloride ion and to check the conductivity.

the existence of ASME Section XI. For this reason, the degree of access required by ASME Section XI is not generally available and will be addressed as "requests for relief" in accordance with 10 CFR 50.55a(g).

Experience in safety valve operation shows testing in accordance with the ASME Code is adequate to detect failures or deterioration. The as-found setpoint tolerance is specified in the ASME OM Code as the owner-defined tolerance or ± 3% of valve nameplate set pressure. An analysis has been performed which shows that with all safety valves set 36 psig higher, the safety limit of 1375 psig is not exceeded.

Conductivity instruments continuously monitor the reactor coolant. Experience indicates that a check of the conductivity instrumentation at least every 72 hours is adequate to ensure accurate readings. The reactor water sample will also be used to determine the chloride ion content to assure that the limits of 3.3.E are not exceeded. The chloride ion content will not change rapidly over a period of several days; therefore, the sampling frequency is adequate.

| Item | Frequency |
|---|---|
| C. Containment Cooling System | |
| 2. Motor-operated valve operability | Every 3 months |
| 3. Pump compartment water-tight doors closed | Once/week and after each entry |
| D. Emergency Service Water System | |
| 1. Pump Operability | Once/3 months. Also after major maintenance and prior to startup following a refueling outage. |
| E. Control Rod Drive Hydraulic System | |
| 1. Pump Operability | Once/month. Also after major maintenance and prior to startup following a refueling outage. |
| F. Fire Protection System | |
| 1. Pump Operability | Once/month. Also after major maintenance and prior to startup following a refueling outage. |
| 2. Isolation valve operability | Once/3 months. Also after major maintenance and prior to startup following a refueling outage. |

Bases:

It is during major maintenance or repair that a system's design intent may be violated accidentally. Therefore, a functional test is required after every major maintenance operation. During an extended outage, such as a refueling outage, major repair and maintenance may be performed on many systems. To be sure that these repairs on other systems do not encroach unintentionally on critical standby cooling systems, they should be given a functional test prior to startup.

Motor operated pumps, valves and other active devices that are normally on standby should be exercised periodically to make sure that they are free to operate. Motors on pumps should operate long enough to approach equilibrium temperature to ensure there is no overheat problem. Whenever practical, valves should be stroked full length to ensure that nothing impedes their motion. Testing of components per OC Inservice Testing Program in accordance with the ASME Code once every 3 months provides assurances of the availability of the system. The Control Rod Hydraulic pumps and Fire Protection System pumps are not part of the Inservice Test Program per the ASME Code and will continue to be tested for operability once per month. Engineering judgment based on experience and availability analyses of the type presented in Appendix L of the FDSAR indicates that testing these components more often than once a month over a long period of time does not significantly improve the system reliability. Also, at this frequency of testing wearout should not be a problem through the life of the plant.

A Primary Containment Leakage Rate Testing Program has been established to implement the requirements of 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. Guidance for implementation of Option B is contained in NRC Regulatory Guide 1.163, "Performance Based Containment Leak Test Program", Revision 0, dated September 1995. Additional guidance for NRC Regulatory Guide 1.163 is contained in Nuclear Energy Institute (NEI) 94-01, "Industry Guideline for Implementing Performance Based Option of 10 CFR 50, Appendix J," Revision 0, dated July 26, 1995, and ANSI/ANS 56.8-1994, "Containment System Leakage Testing Requirements." The Primary Containment Leakage Rate Testing Program conforms with this guidance as modified by approved exemptions.

The maximum allowable leakage rate for the primary containment ($L_a$) is 1.0 percent by weight of the containment air per 24 hours at the design basis LOCA maximum peak containment pressure ($P_a$). As discussed below, $P_a$ for the purpose of containment leak rate testing is 35 psig.

The penetration and air purge piping leakage test frequency, along with the containment leak rate tests, is adequate to allow detection of leakage trends. Whenever a double gasketed penetration (primary containment head equipment hatches and the absorption chamber access hatch) is broken and remade, the space between the gaskets is pressurized to determine that the seals are performing properly. The test pressure of 35 psig is consistent with the accident analyses and the maximum preoperational leak rate test pressure.

Monitoring the nitrogen makeup requirements of the inerting system provides a method of observing leak rate trends and would detect gross leaks in a very short time. This equipment must be periodically removed from service for test and maintenance, but this out-of-service time be kept to a practical minimum.

Automatic primary containment isolation valves are provided to maintain PRIMARY CONTAINMENT INTEGRITY following the design basis loss-of-coolant accident. Closure times for the automatic primary containment isolation valves are not critical because it is on the order of minutes before significant fission product release to the containment atmosphere for the design basis loss of coolant accident. These valves are highly reliable, see infrequent service and most of them are normally in the closed position. Therefore, a test during each REFUELING OUTAGE is sufficient.

Large lines connecting to the reactor coolant system, whose failure could result in uncovering the reactor core, are supplied with automatic isolation valves (except containment cooling). Closure times restrict coolant loss from the circumferential rupture of any of these lines outside primary containment to less than that for a main steam line break (the design basis accident for outside containment line breaks). The minimum time for main steam isolation valve (MSIV) closure of 3 seconds is based on the transient analysis that shows the pressure peak 76 psig below the lowest safety valve setting. The maximum time for MSIV closure of 10 seconds is based on the value assumed for the main steam line break dose calculations and restricts coolant loss to prevent uncovering the reactor core. Per the ASME Code, the full closure test of the MSIVs during COLD SHUTDOWNs will ensure OPERABILITY and provide assurance that the valves maintain the required closing time. The provision for a minimum of 92 days between the tests ensures that full closure testing is not too frequent. The MSIVs are partially stroked quarterly as part of reactor protection system instrument surveillance testing.

## 4.8 ISOLATION CONDENSER

Applicability: Applies to periodic testing requirements for the isolation condenser system.

Objective: To verify the operability of the isolation condenser system.

Specification: A. Surveillance of each isolation condenser loop shall be as follows:

| Item | Frequency |
|---|---|
| 1. Operability of motor-operated isolation valves and condensate makeup valves. | Once/3 months |
| 2. Automatic actuation and functional test. | Each refueling outage (interval not to exceed 20 months) or following major repair. |
| 3. Shell side water volume check | Once/day |
| 4. Isolation valve (steam side) | |
| a. Visual inspection | Each refueling outage |
| b. External leakage check | Each primary system Leak test |
| c. Area temperature check | Once/shift |

Basis: Motor-operated valves on the isolation condenser steam and condensate lines and on the condensate makeup line that are normally on standby should be exercised periodically to make sure that they are free to operate. The valves will be stroked full length every time they are tested to verify proper functional performance. This frequency of testing is consistent with instrumentation tests discussed in Specification 4.1. Testing of these components per the ASME Code once every 3 months provides assurance of availability of the system. Also, at this frequency of testing, wearout should not be a problem throughout the life of the plant.

The automatic actuation and functional test will demonstrate the automatic opening of the condensate return line valves and the automatic closing of the isolation valves on the vent lines to the main steam lines. Automatic closure of the isolation condenser steam and condensate lines on actuation of the condenser pipe break detectors will also be verified by the test. It is during a major maintenance or repair that a system's design intent may be violated accidentally. This makes the functional test necessary after every major repair operation.

By virtue of normal plant operation the operators daily observe the water level in the isolation condensers. In addition, isolation condenser shell side water level sensors provide control room annunciation of condenser high or low water level.

**ATTACHMENT 3H**

THREE MILE ISLAND
UNITS 1

Docket No. 50-289

License No. DPR-50

**Typed Pages**

**for**

**Technical Specifications and Bases Changes**

REVISED TS and TS BASES PAGES

3-2
4-11
4-13
4-52
4-52a

Bases

The limitation on power operation with one idle RC pump in each loop has been imposed since the ECCS cooling performance has not been calculated in accordance with the Final Acceptance Criteria requirements specifically for this mode of reactor operation. A time period of 24 hours is allowed for operation with one idle RC pump in each loop to effect repairs of the idle pump(s) and to return the reactor to an acceptable combination of operating RC pumps. The 24 hours for this mode of operation is acceptable since this mode is expected to have considerable margin for the peak cladding temperature limit and since the likelihood of a LOCA within the 24-hour period is considered very remote.

A reactor coolant pump or decay heat removal pump is required to be in operation before the boron concentration is reduced by dilution with makeup water. Either pump will provide mixing which will prevent sudden positive reactivity changes caused by dilute coolant reaching the reactor. One decay heat removal pump will circulate the equivalent of the reactor coolant system volume in one-half hour or less.

The decay heat removal system suction piping is designed for 300°F and 370 psig; thus, the system can remove decay heat when the reactor coolant system is below this temperature (References 1, 2, and 3).

Both steam generators must be operable before heatup of the Reactor Coolant System to insure system integrity against leakage under normal and transient conditions. Only one steam generator is required for decay heat removal purposes.

One pressurizer code safety valve is capable of preventing overpressurization when the reactor is not critical since its relieving capacity is greater than that required by the sum of the available heat sources which are pump energy, pressurizer heaters, and reactor decay heat. Both pressurizer code safety valves are required to be in service prior to criticality to conform to the system design relief capabilities. The code safety valves prevent overpressure for a rod withdrawal or feedwater line break accidents (Reference 4). The pressurizer code safety valve lift set point shall be set at 2500 psig ±1% allowance for error. Surveillance requirements are specified in the Inservice Testing Program. Pressurizer code safety valve setpoint drift of up to 3% is acceptable in accordance with the assumptions of the TMI-1 safety analysis (Reference 5).

References

(1) UFSAR, Tables 9.5-1 and 9.5-2

(2) UFSAR, Sections 4.2.5.1 and 9.5 - "Decay Heat Removal"

(3) UFSAR, Section 4.2.5.4 - "Secondary System"

(4) UFSAR, Section 4.3.10.4 - "System Minimum Operational Components"

(5) UFSAR, Section 4.3.7 - "Overpressure Protection"

## 4.2 REACTOR COOLANT SYSTEM INSERVICE AND TESTING

Applicability

This technical specification applies to the inservice inspection (ISI) and inservice testing (IST) of the reactor coolant system pressure boundary and portions of other safety oriented system pressure boundaries.

Objective

The objective of the ISI and IST programs is to provide assurance of the continuing integrity of the reactor coolant system while at the same time minimizing radiation exposure to personnel in the performance of inservice inspections and tests.

Specification

4.2.1 ISI of ASME Code Class 1, Class 2, and Class 3 components shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR 50, Section 50.55a, except where specific written relief has been granted by the NRC.

4.2.2 IST of ASME Code Class 1, Class 2 and Class 3 pumps and valves shall be performed in accordance with the ASME Code for Operation and Maintenance of Nuclear Power Plants (ASME OM Code) and applicable Addenda as required by 10 CFR 50, Section 50.55a, except where specific written relief has been granted by the NRC.

4.2.3 (Deleted)

4.2.4 The accessible portions of one reactor coolant pump motor flywheel assembly will be ultrasonically inspected within the first ISI period, two reactor coolant pump motor flywheel assemblies within the first two ISI periods and all four by the end of the 10 year inspection interval. However, the U.T. procedure is developmental and will be used only to the extend that it is shown to be meaningful. The extent of coverage will be limited to those areas of the flywheel which are accessible without motor disassembly, i.e., can be reached through the access ports. Also, if radiation levels at the lower access ports are prohibitive, only the upper access ports will be used.

Bases

Specifications 4.2.1 and 2 ensure that inservice inspection of ASME Code Class 1, 2 and 3 components and inservice testing of ASME Code Class 1, 2 and 3 pumps and valves will be performed in accordance with a periodically updated version of the ASME Code and Addenda as required by 10 CFR 50.55a. Relief from any of the above requirements has been provided in writing by the NRC and is not a part of these technical specifications.

4.3 DELETED

## 4.9 DECAY HEAT REMOVAL (DHR) CAPABILITY - PERIODIC TESTING

Applicability

Applies to the periodic testing of systems or components which function to remove decay heat.

Objective

To verify that systems/components required for DHR are capable of performing their design function.

Specification

4.9.1 Reactor Coolant System (RCS) Temperature greater than 250 degrees F.

4.9.1.1 Verify each Emergency Feedwater (EFW) Pump is tested in accordance with the requirements and acceptance criteria of the Inservice Test Program.

Note: This surveillance is not required to be performed for the turbine-driven EFW Pump (EF-P-1) until 24 hours after exceeding 750 psig.

4.9.1.2 DELETED

4.9.1.3 At least once per 31 days, each EFW System flowpath valve from both Condensate Storage Tanks (CSTs) to the OTSGs via the motor-driven pumps and the turbine-driven pump shall be verified to be in the required status.

4.9.1.4 On a refueling interval basis:

a) Verify that each EFW Pump starts automatically upon receipt of an EFW test signal.

b) Verify that each EFW control valve responds upon receipt of an EFW test signal.

c) Verify that each EFW control valve responds in manual control from the control room and remote shutdown panel.

4.9.1.5 Prior to STARTUP, following a REFUELING SHUTDOWN or a COLD SHUTDOWN greater than 30 days, conduct a test to demonstrate that the motor driven EFW Pumps can pump water from the CSTs to the Steam Generators.

4.9.1.6 Acceptance Criteria

These tests shall be considered satisfactory if control board indication and visual observation of the equipment demonstrates that all components have operated properly except for the tests required by Specification 4.9.1.1.

4.9.2 RCS Temperature less than or equal to 250 degrees F.*

4.9.2.1 On a daily basis, verify operability of the means for DHR required by Specification 3.4.2 by observation of console status indication.

* These requirements supplement the requirements of Specifications 4.5.2.2 and 4.5.4.

Bases

The ASME Code specifies requirements and acceptance standards for the testing of nuclear safety related pumps. The quarterly EFW Pump test frequency specified by the ASME Code will be sufficient to verify that the turbine-driven and both motor-driven EFW Pumps are operable. Compliance with the normal acceptance criteria assures that the EFW Pumps are operating as expected. The surveillance requirements ensure that the overall EFW System functional capability is maintained.

Deferral of the requirement to perform IST on the turbine-driven EFW Pump is necessary to assure sufficient OTSG pressure to perform the test using Main Steam.

Daily verification of the operability of the required means for DHR ensures that sufficient DHR capability will be maintained.

**ATTACHMENT 4A**

BRAIDWOOD STATION
UNITS 1 and 2

Docket Nos. STN 50-456 and STN 50-457

License Nos. NPF-72 and NPF-77

**Markup of Proposed Technical Specifications Bases Page Changes**

<u>REVISED TS BASES PAGES</u>

B 3.4.10-4
B 3.4.10-5
B 3.4.11-8
B 3.4.11-9
B 3.4.12-15
B 3.4.12-17
B 3.4.14-6
B 3.4.14-8
B 3.5.2-11
B 3.6.6-10
B 3.6.6-12
B 3.7.1-6
B 3.7.1-7
B 3.7.2-6
B 3.7.2-7
B 3.7.5-7
B 3.7.5-10

BASES

ACTIONS

A.1

With one pressurizer safety valve inoperable, restoration must take place within 15 minutes. The Completion Time of 15 minutes reflects the importance of maintaining the RCS Overpressure Protection System. An inoperable safety valve coincident with an RCS overpressure event could challenge the integrity of the pressure boundary.

B.1 and B.2

If Required Action A.1 and its associated Completion Time are not met or if two or more pressurizer safety valves are inoperable, the unit must be brought to a MODE in which the requirement does not apply. To achieve this status, the unit must be brought to at least MODE 3 within 6 hours and to MODE 4 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging plant systems. In MODE 4, overpressure protection is provided by the LTOP System. The change from MODE 1, 2, or 3 to MODE 4 reduces the RCS energy (core power and pressure), lowers the potential for large pressurizer insurges, and thereby removes the need for overpressure protection by three pressurizer safety valves.

SURVEILLANCE REQUIREMENTS

SR 3.4.10.1

SRs are specified in the Inservice Testing Program. Pressurizer safety valves are to be tested in accordance with the requirements ~~of Section XI~~ of the ASME Code (Ref. 4), which provides the activities and Frequencies necessary to satisfy the SRs. No additional requirements are specified.

The pressurizer safety valve setpoint is ± 2% of a nominal 2460 psig for OPERABILITY; however, the valves are reset to ± 1% during the Surveillance to allow for drift.

BASES

REFERENCES

1. ASME, Boiler and Pressure Vessel Code, Section III.

2. UFSAR, Chapter 15.

3. WCAP-7769, Rev. 1, June 1972.

4. ~~ASME, Boiler and Pressure Vessel Code, Section XI~~.

INSERT 1

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.11.1

Block valve cycling verifies that the valve(s) can be opened and closed if needed. The basis for the Frequency of 92 days is the ASME Code~~, Section XI~~ (Ref. 3).

The Note modifies this SR by stating that it is not required to be met with the block valve closed in accordance with the Required Actions of this LCO. If the block valve is closed to isolate an inoperable PORV that is incapable of being manually cycled, the maximum Completion Time to restore the PORV and open the block valve is 72 hours, which is well within the allowable limits (25%) to extend the block valve Frequency of 92 days. Furthermore, these test requirements would be completed by the reopening of a recently closed block valve upon restoration of the PORV to OPERABLE status (i.e., completion of the Required Actions fulfills the SR).

SR 3.4.11.2

SR 3.4.11.2 requires a complete cycle of each PORV. Operating a PORV through one complete cycle ensures that the PORV can be manually actuated for mitigation of an SGTR. The Frequency of 18 months is based on a typical refueling cycle and industry accepted practice.

The Note modifies the SR to allow entry into and operation in MODE 3 prior to performing the SR. This allows the test to be performed in MODE 3 under operating temperature and pressure conditions prior to entering MODE 1 or 2. In accordance with Reference 4, this test should be performed in MODE 3 or 4 to adequately simulate operating temperature and pressure effects on PORV operation.

SR 3.4.11.3

Operating the solenoid air control valves and check valves on the air accumulators ensures the PORV control system actuates properly when called upon. The Frequency of 18 months is based on a typical refueling cycle and the Frequency of the other Surveillances used to demonstrate PORV OPERABILITY.

BASES

REFERENCES

1. Regulatory Guide 1.32, February 1977.
2. UFSAR, Section 15.2.
3. ~~ASME, Boiler and Pressure Vessel Code, Section XI~~.

INSERT 1

4. Generic Letter 90-06, "Resolution of Generic Issue 70, "Power Operated Relief Valve and Block Valve Reliability," and Generic Issue 94, "Additional Low Temperature Overpressure Protection for Light Water Reactors," pursuant to 10 CFR 50.54(f), June 25, 1990.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.4.12.4

The RCS vent of ≥ 2.0 square inches is proven OPERABLE by verifying its open condition either:

a. Once every 12 hours for a valve that cannot be locked.

b. Once every 31 days for a valve that is locked, sealed, or secured in position. A removed pressurizer safety valve fits this category.

The passive vent arrangement must only be open to be OPERABLE. This Surveillance is required to be performed if the vent is being used to satisfy the pressure relief requirements of LCO 3.4.12.d.4.

SR 3.4.12.5

Each required RHR suction relief valve shall be demonstrated OPERABLE by verifying its RHR suction isolation valves are open. This Surveillance is only required to be performed if the RHR suction relief valve is being used to satisfy this LCO.

The RHR suction isolation valves, RH8701A and RH8701B for relief valve RH8708A, and RH8702A and RH8702B for relief valve RH8708B, are verified to be opened every 72 hours. The Frequency is considered adequate in view of other administrative controls such as valve status indications available to the operator in the control room that verify the RHR suction valves remain open.

The ASME Code~~,~~ ~~Section XI~~ (Ref. 7)~~,~~ test per Inservice Testing Program verifies OPERABILITY by proving proper relief valve mechanical motion and by measuring and, if required, adjusting the lift setpoint.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.4.12.8

Performance of a CHANNEL CALIBRATION on each required PORV actuation channel is required every 18 months to adjust the whole channel so that it responds and the valve opens within the required range and accuracy to known input.

REFERENCES

1. 10 CFR 50, Appendix G.
2. Generic Letter 88-11.
3. ASME, Boiler and Pressure Vessel Code, Section III.
4. UFSAR, Chapter 15.
5. Generic Letter 90-06.
6. Safety Evaluation Report, dated August 31, 1990.
7. ~~ASME, Boiler and Pressure Vessel Code, Section XI.~~

$\boxed{\text{INSERT 1}}$

BASES

ACTIONS (continued)

C.1 and C.2

If the Required Actions and associated Completion Times of Conditions A and B are not met, the unit must be brought to a MODE in which the requirement does not apply. To achieve this status, the unit must be brought to MODE 3 within 6 hours and MODE 5 within 36 hours. This Action may reduce the leakage and also reduces the potential for a LOCA outside the containment. The allowed Completion Times are reasonable based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.4.14.1

Performance of leakage testing on each RCS PIV or isolation valve used to satisfy Required Action A.1 and Required Action A.2 is required to verify that leakage is below the specified limit and to identify each leaking valve. The leakage limit of 0.5 gpm per inch of nominal valve diameter up to 5 gpm maximum applies to each valve. Leakage testing requires a stable pressure condition.

For two PIVs in series, the leakage requirement applies to each valve individually and not to the combined leakage across both valves. If the PIVs are not individually leakage tested, one valve may have failed completely and not be detected if the other valve in series meets the leakage requirement. In this situation, the protection provided by redundant valves would be lost.

Testing is to be performed every 18 months, a typical refueling cycle, if the plant does not go into MODE 5 for at least 7 days. The 18 month Frequency is consistent with 10 CFR 50.55a(g) (Ref. 8) as contained in the Inservice Testing Program, is within the frequency allowed by the American Society of Mechanical Engineers (ASME) Code~~,~~ ~~Section XI~~ (Ref. 7).

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.4.14.2

The interlock setpoint that prevents the RHR System suction isolation valves from being opened is set so the actual RCS pressure must be < 360 psig to open the valves. This setpoint ensures the RHR design pressure will not be exceeded and the RHR relief valves will not lift. The 18 month Frequency is based on the need to perform the Surveillance under conditions that apply during a unit outage. The 18 month Frequency is also acceptable based on consideration of the design reliability (and confirming operating experience) of the equipment.

REFERENCES

1. 10 CFR 50.2.
2. 10 CFR 50.55a(c).
3. 10 CFR 50, Appendix A, Section V, GDC 55.
4. WASH-1400 (NUREG-75/014), Appendix V, October 1975.
5. NUREG-0677, May 1980.
6. EG&G Report, EGG-NTAP-6175.
7. ~~ASME, Boiler and Pressure Vessel Code, Section XI.~~

INSERT 1

8. 10 CFR 50.55a(g).

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.5.2.4

Periodic surveillance testing of ECCS pumps to detect gross degradation caused by impeller structural damage or other hydraulic component problems is required by ~~SECTION XI of~~ the ASME Code. This type of testing may be accomplished by measuring the pump developed head at only one point of the pump characteristic curve. This verifies both that the measured performance is within an acceptable tolerance of the original pump baseline performance and that the performance at the test flow is greater than or equal to the performance assumed in the plant safety analysis. SRs are specified in the Inservice Testing Program, ~~which encompasses SECTION XI~~ of the ASME Code. ~~SECTION XI of the~~ The ASME Code provides the activities and Frequencies necessary to satisfy the requirements.

SR 3.5.2.5

This Surveillance demonstrates that each automatic ECCS valve actuates to the required position on an actual or simulated SI signal (a coincident RWST Level Low-Low signal is required to open the containment sump isolation valves). This Surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls. This Surveillance may be performed on-line at an 18 month Frequency during RH system maintenance work windows when the RH pump suction piping is drained; thus, reducing overall RH system unavailability. If there is not an on-line RH system maintenance work window that requires the RH pump suction piping to be drained, this Surveillance must be conducted during refueling outages (Ref. 11). The 18 month Frequency is acceptable based on consideration of the design reliability (and confirming operating experience) of the equipment. The actuation logic is tested as part of ESF Actuation System testing, and equipment performance is monitored as part of the Inservice Testing Program.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.6.2

Operating each containment cooling train fan unit (in slow speed) for ≥ 15 minutes ensures that all trains are OPERABLE and that all associated controls are functioning properly. It also ensures that blockage, fan or motor failure, or excessive vibration can be detected for corrective action. The 31 day Frequency was developed considering the known reliability of the fan units and controls, the two train redundancy available, and the low probability of significant degradation of the containment cooling train occurring between surveillances. It has also been shown to be acceptable through operating experience.

SR 3.6.6.3

This SR requires verifying that an SX flow rate greater than or equal to the design flow rate assumed in the safety analyses (i.e., 2660 gpm) to each containment cooling unit (RCFC) will be achieved with the primary containment refrigeration units in their specified safety configuration described in UFSAR Section 9.4.8 (Ref.2). The Frequency was developed considering the known reliability of the SX System, the two train redundancy available, and the low probability of a significant degradation of flow occurring between surveillances.

SR 3.6.6.4

Verifying each containment spray pump's developed head at the flow test point is greater than or equal to the required developed head (resulting in 265 psig discharge pressure) ensures that spray pump performance has not degraded during the cycle. Flow and differential pressure are normal tests of centrifugal pump performance required by ~~SECTION XI of~~ the ASME Code (Ref. 8). Since the containment spray pumps cannot be tested with flow through the spray headers, they are tested on recirculation flow. This test confirms one point on the pump design curve and is indicative of overall performance. Such inservice tests confirm component OPERABILITY, trend performance, and detect incipient failures by abnormal performance. The Frequency of the SR is in accordance with the Inservice Testing Program.

BASES

REFERENCES

1. 10 CFR 50, Appendix A, GDC 38, GDC 39, GDC 40, GDC 41, GDC 42, and GDC 43.
2. UFSAR, Section 9.4.8.
3. UFSAR, Section 6.5.2.
4. 10 CFR 50, Appendix K.
5. UFSAR, Section 6.2.1.1.3.
6. UFSAR, Section 6.2.2.
7. UFSAR, Section 6.2.
8. ~~ASME, Boiler and Pressure Vessel Code, SECTION XI.~~

INSERT 1

BASES

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ACTIONS (continued)

calculated value. The MSSV setpoint tolerance assumption used in the plant specific analyses is bounded by the setpoint tolerance specified in Table 3.7.1-2.

Required Action A.2 is modified by a Note, indicating that the Power Range Neutron Flux-High reactor trip setpoint reduction is only required in Mode 1. In Modes 2 and 3 the rector protection system trips specified in LCO 3.3.1, "Reactor Trip System Instrumentation," provide sufficient protection.

The allowed Completion Times are reasonable based on operating experience to accomplish the Required Actions in an orderly manner without challenging plant systems.

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B.1 and B.2

If the MSSVs cannot be restored to OPERABLE status or the Required Actions cannot be completed within the associated Completion Time, or if one or more steam generators have = 4 inoperable MSSVs, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 4 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

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SURVEILLANCE REQUIREMENTS

SR 3.7.1.1

This SR verifies the OPERABILITY of the MSSVs by the verification of each MSSV lift setpoint in accordance with the Inservice Testing Program. The ASME Code, ~~Section XI~~ (Ref. 5), requires that safety and relief valve tests be performed in accordance with ANSI/ASME OM-1-1987 (Ref. 6). According to Reference 6, the following tests are required.

a. Visual examination;

b. Seat tightness determination;

c. Setpoint pressure determination (lift setting);

d. Compliance with owner's seat tightness criteria; and

e. Verification of the balancing device integrity on balanced valves.

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BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.1.1 (continued)

The ANSI/ASME Standard requires that all valves be tested every 5 years, and a minimum of 20% of the valves be tested every 24 months. The ASME Code specifies the activities and frequencies necessary to satisfy the requirements. Table 3.7.1-2 allows a ± 3% setpoint tolerance for OPERABILITY; however, the valves are reset to ± 1% during the Surveillance to allow for drift. The lift settings, according to Table 3.7.1-2, correspond to ambient conditions of the valve at nominal operating temperature and pressure.

This SR is modified by a Note that allows entry into and operation in MODE 3 prior to performing the SR. The MSSVs may be either bench tested or tested in situ at hot conditions using an assist device to simulate lift pressure. If the MSSVs are not tested at hot conditions, the lift setting pressure shall be corrected to ambient conditions of the valve at operating temperature and pressure.

REFERENCES

1. UFSAR, Section 10.3.1.
2. ASME, Boiler and Pressure Vessel Code, Section III, Article NC-7000, Class 2 Components.
3. UFSAR, Section 15.2.
4. NRC Information Notice 94-60, "Potential Overpressurization of the Main Steam System," August 22, 1994.
5. ~~ASME, Boiler and Pressure Vessel Code, Section XI.~~ INSERT 1
6. ANSI/ASME OM-1-1987 and applicable Addenda.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.2.1

This SR verifies that MSIV closure time is ≤ 5 seconds. The MSIV closure time is assumed in the accident and containment analyses. This Surveillance is normally performed upon returning the unit to operation following a refueling outage. Based on ASME Code ~~Section XI~~ (Ref. 5), the MSIVs are not closure time tested at power.

The Frequency is in accordance with the Inservice Testing Program. This test is conducted in MODE 3 with the unit at operating temperature and pressure. This SR is modified by two Notes. Note 1 allows entry into and operation in MODE 3 prior to performing the SR. This allows a delay of testing until MODE 3, to establish conditions consistent with those under which the acceptance criterion was generated. Note 2 allows the SR to not be met until the first startup after September 27, 2001.

SR 3.7.2.2

This SR verifies that each MSIV can close on an actual or simulated actuation signal. This Surveillance is normally performed upon returning the unit to operation following a refueling outage. The frequency of MSIV testing is every 18 months. The 18 month Frequency for testing is based on the refueling cycle. Operating experience has shown that these components usually pass the Surveillance when performed at the 18 month Frequency. Therefore, this Frequency is acceptable from a reliability standpoint.

This SR is modified by two Notes. Note 1 allows entry into and operation in MODE 3 prior to performing the SR. This allows a delay of testing until MODE 3, to establish conditions consistent with those under which the acceptance criterion was generated. Note 2 allows the SR to not be met until the first startup after September 27, 2001.

BASES

REFERENCES

1. UFSAR, Section 10.3.
2. UFSAR, Section 15.1.5.
3. UFSAR, Section 6.2.
4. 10 CFR 50.67.
5. ~~ASME, Boiler and Pressure Vessel Code, Section XI.~~

INSERT 1

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.5.4

Verifying that each AF pump's developed head at the flow test point is greater than or equal to the required developed head ensures that AF pump performance has not degraded during the cycle. Flow and differential head are normal tests of centrifugal pump performance required by ~~Section XI of~~ the ASME Code (Ref. 4). Because it is undesirable to introduce cold AF into the steam generators while they are operating, this testing is performed on recirculation flow. This test confirms one point on the pump design curve and is indicative of overall performance. Such inservice tests confirm component OPERABILITY, trend performance, and detect incipient failures by indicating abnormal performance. Performance of inservice testing discussed in the ASME Code~~, Section XI~~ (Ref. 4) (only required at 3 month intervals) satisfies this requirement.

SR 3.7.5.5

This SR verifies that AF can be delivered to the steam generators in the event of any accident or transient that generates an ESFAS, by demonstrating that each automatic valve in the flow path actuates to its correct position on an actual or simulated actuation signal. This Surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls. The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. The 18 month Frequency is acceptable based on operating experience and the design reliability of the equipment.

BASES

REFERENCES

1. UFSAR, Section 10.4.9.

2. Westinghouse Nuclear Safety Evaluation Checklist, SECL 90-469, Revision 1, "Byron/Braidwood Units 1 and 2, Relaxation of MSSV Setpoint Tolerance to +/- 3% - Revised SECL."

3. UFSAR, Section 15.2.

4. ~~ASME, Boiler and Pressure Vessel Code, Section XI.~~

INSERT 1

5. ASTM Standards, D5452-98.

INSERT 1

ASME Code for Operation and Maintenance of Nuclear Power Plants.

**ATTACHMENT 4B**

BYRON STATION
UNITS 1 and 2

Docket Nos. 50-454 and 50-455

License Nos. NPF-37 and NPF-66

**Markup of Proposed Technical Specifications Bases Page Changes**

REVISED TS BASES PAGES

B 3.4.10-4
B 3.4.10-5
B 3.4.11-8
B 3.4.11-9
B 3.4.12-15
B 3.4.12-17
B 3.4.14-6
B 3.4.14-8
B 3.5.2-11
B 3.7.1-6
B 3.7.1-7
B 3.7.2-6
B 3.7.2-7
B 3.7.5-7
B 3.7.5-10

BASES

ACTIONS

A.1

With one pressurizer safety valve inoperable, restoration must take place within 15 minutes. The Completion Time of 15 minutes reflects the importance of maintaining the RCS Overpressure Protection System. An inoperable safety valve coincident with an RCS overpressure event could challenge the integrity of the pressure boundary.

B.1 and B.2

If Required Action A.1 and its associated Completion Time are not met or if two or more pressurizer safety valves are inoperable, the unit must be brought to a MODE in which the requirement does not apply. To achieve this status, the unit must be brought to at least MODE 3 within 6 hours and to MODE 4 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging plant systems. In MODE 4, overpressure protection is provided by the LTOP System. The change from MODE 1, 2, or 3 to MODE 4 reduces the RCS energy (core power and pressure), lowers the potential for large pressurizer insurges, and thereby removes the need for overpressure protection by three pressurizer safety valves.

SURVEILLANCE REQUIREMENTS

SR 3.4.10.1

SRs are specified in the Inservice Testing Program. Pressurizer safety valves are to be tested in accordance with the requirements ~~of Section XI~~ of the ASME Code (Ref. 4), which provides the activities and Frequencies necessary to satisfy the SRs. No additional requirements are specified.

The pressurizer safety valve setpoint is ± 2% of a nominal 2460 psig for OPERABILITY; however, the valves are reset to ± 1% during the Surveillance to allow for drift.

BASES

REFERENCES

1. ASME, Boiler and Pressure Vessel Code, Section III.
2. UFSAR, Chapter 15.
3. WCAP-7769, Rev. 1, June 1972.
4. ~~ASME, Boiler and Pressure Vessel Code, Section XI.~~

INSERT 1

SURVEILLANCE REQUIREMENTS

SR 3.4.11.1

Block valve cycling verifies that the valve(s) can be opened and closed if needed. The basis for the Frequency of 92 days is the ASME Code~~, Section XI~~ (Ref. 3).

The Note modifies this SR by stating that it is not required to be met with the block valve closed in accordance with the Required Actions of this LCO. If the block valve is closed to isolate an inoperable PORV that is incapable of being manually cycled, the maximum Completion Time to restore the PORV and open the block valve is 72 hours, which is well within the allowable limits (25%) to extend the block valve Frequency of 92 days. Furthermore, these test requirements would be completed by the reopening of a recently closed block valve upon restoration of the PORV to OPERABLE status (i.e., completion of the Required Actions fulfills the SR).

SR 3.4.11.2

SR 3.4.11.2 requires a complete cycle of each PORV. Operating a PORV through one complete cycle ensures that the PORV can be manually actuated for mitigation of an SGTR. The Frequency of 18 months is based on a typical refueling cycle and industry accepted practice.

The Note modifies the SR to allow entry into and operation in MODE 3 prior to performing the SR. This allows the test to be performed in MODE 3 under operating temperature and pressure conditions prior to entering MODE 1 or 2. In accordance with Reference 4, this test should be performed in MODE 3 or 4 to adequately simulate operating temperature and pressure effects on PORV operation.

SR 3.4.11.3

Operating the solenoid air control valves and check valves on the air accumulators ensures the PORV control system actuates properly when called upon. The Frequency of 18 months is based on a typical refueling cycle and the Frequency of the other Surveillances used to demonstrate PORV OPERABILITY.

BASES

REFERENCES

1. Regulatory Guide 1.32, February 1977.

2. UFSAR, Section 15.2.

3. ~~ASME, Boiler and Pressure Vessel Code, Section XI.~~ INSERT 1

4. Generic Letter 90-06, "Resolution of Generic Issue 70, "Power Operated Relief Valve and Block Valve Reliability," and Generic Issue 94, "Additional Low Temperature Overpressure Protection for Light Water Reactors," pursuant to 10 CFR 50.54(f), June 25, 1990.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.4.12.4

The RCS vent of ≥ 2.0 square inches is proven OPERABLE by verifying its open condition either:

a. Once every 12 hours for a valve that cannot be locked.

b. Once every 31 days for a valve that is locked, sealed, or secured in position. A removed pressurizer safety valve fits this category.

The passive vent arrangement must only be open to be OPERABLE. This Surveillance is required to be performed if the vent is being used to satisfy the pressure relief requirements of LCO 3.4.12.d.4.

SR 3.4.12.5

Each required RHR suction relief valve shall be demonstrated OPERABLE by verifying its RHR suction isolation valves are open. This Surveillance is only required to be performed if the RHR suction relief valve is being used to satisfy this LCO.

The RHR suction isolation valves, RH8701A and RH8701B for relief valve RH8708A, and RH8702A and RH8702B for relief valve RH8708B, are verified to be opened every 72 hours. The Frequency is considered adequate in view of other administrative controls such as valve status indications available to the operator in the control room that verify the RHR suction valves remain open.

The ASME Code, ~~Section XI~~ (Ref. 7), test per Inservice Testing Program verifies OPERABILITY by proving proper relief valve mechanical motion and by measuring and, if required, adjusting the lift setpoint.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.4.12.8

Performance of a CHANNEL CALIBRATION on each required PORV actuation channel is required every 18 months to adjust the whole channel so that it responds and the valve opens within the required range and accuracy to known input.

REFERENCES

1. 10 CFR 50, Appendix G.
2. Generic Letter 88-11.
3. ASME, Boiler and Pressure Vessel Code, Section III.
4. UFSAR, Chapter 15.
5. Generic Letter 90-06.
6. Safety Evaluation Report, dated August 31, 1990.
7. ~~ASME, Boiler and Pressure Vessel Code, Section XI.~~

INSERT 1

BASES

ACTIONS (continued)

C.1 and C.2

If the Required Actions and associated Completion Times of Conditions A and B are not met, the unit must be brought to a MODE in which the requirement does not apply. To achieve this status, the unit must be brought to MODE 3 within 6 hours and MODE 5 within 36 hours. This Action may reduce the leakage and also reduces the potential for a LOCA outside the containment. The allowed Completion Times are reasonable based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.4.14.1

Performance of leakage testing on each RCS PIV or isolation valve used to satisfy Required Action A.1 and Required Action A.2 is required to verify that leakage is below the specified limit and to identify each leaking valve. The leakage limit of 0.5 gpm per inch of nominal valve diameter up to 5 gpm maximum applies to each valve. Leakage testing requires a stable pressure condition.

For two PIVs in series, the leakage requirement applies to each valve individually and not to the combined leakage across both valves. If the PIVs are not individually leakage tested, one valve may have failed completely and not be detected if the other valve in series meets the leakage requirement. In this situation, the protection provided by redundant valves would be lost.

Testing is to be performed every 18 months, a typical refueling cycle, if the plant does not go into MODE 5 for at least 7 days. The 18 month Frequency is consistent with 10 CFR 50.55a(g) (Ref. 8) as contained in the Inservice Testing Program, is within the frequency allowed by the American Society of Mechanical Engineers (ASME) Code~~,~~ ~~Section XI~~ (Ref. 7).

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.4.14.2

The interlock setpoint that prevents the RHR System suction isolation valves from being opened is set so the actual RCS pressure must be < 360 psig to open the valves. This setpoint ensures the RHR design pressure will not be exceeded and the RHR relief valves will not lift. The 18 month Frequency is based on the need to perform the Surveillance under conditions that apply during a unit outage. The 18 month Frequency is also acceptable based on consideration of the design reliability (and confirming operating experience) of the equipment.

REFERENCES

1. 10 CFR 50.2.
2. 10 CFR 50.55a(c).
3. 10 CFR 50, Appendix A, Section V, GDC 55.
4. WASH-1400 (NUREG-75/014), Appendix V, October 1975.
5. NUREG-0677, May 1980.
6. EG&G Report, EGG-NTAP-6175.
7. ~~ASME, Boiler and Pressure Vessel Code, Section XI.~~

INSERT 1

8. 10 CFR 50.55a(g).

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.5.2.4

Periodic surveillance testing of ECCS pumps to detect gross degradation caused by impeller structural damage or other hydraulic component problems is required by ~~SECTION XI of~~ the ASME Code. This type of testing may be accomplished by measuring the pump developed head at only one point of the pump characteristic curve. This verifies both that the measured performance is within an acceptable tolerance of the original pump baseline performance and that the performance at the test flow is greater than or equal to the performance assumed in the plant safety analysis. SRs are specified in the Inservice Testing Program~~, which encompasses SECTION XI~~ of the ASME Code. ~~SECTION XI of the~~ The ASME Code provides the activities and Frequencies necessary to satisfy the requirements.

SR 3.5.2.5

This Surveillance demonstrates that each automatic ECCS valve actuates to the required position on an actual or simulated SI signal (a coincident RWST Level Low-Low signal is required to open the containment sump isolation valves). This Surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls. This Surveillance may be performed on-line at an 18 month Frequency during RH system maintenance work windows when the RH pump suction piping is drained; thus, reducing overall RH system unavailability. If there is not an on-line RH system maintenance work window that requires the RH pump suction piping to be drained, this Surveillance must be conducted during refueling outages (Ref. 11). The 18 month Frequency is acceptable based on consideration of the design reliability (and confirming operating experience) of the equipment. The actuation logic is tested as part of ESF Actuation System testing, and equipment performance is monitored as part of the Inservice Testing Program.

BASES

ACTIONS (continued)

Instrumentation System trip channel uncertainty assumption used in the plant specific analyses is bounded by the calculated value. The MSSV setpoint tolerance assumption used in the plant specific analyses is bounded by the setpoint tolerance specified in Table 3.7.1-2.

Required Action A.2 is modified by a Note, indicating that the Power Range Neutron Flux-High reactor trip setpoint reduction is only required in Mode 1. In Modes 2 and 3 the reactor protection system trips specified in LCO 3.3.1, "Reactor Trip System Instrumentation," provide sufficient protection.

The allowed Completion Times are reasonable based on operating experience to accomplish the Required Actions in an orderly manner without challenging plant systems.

B.1 and B.2

If the MSSVs cannot be restored to OPERABLE status or the Required Actions cannot be completed within the associated Completion Time, or if one or more steam generators have ≥ 4 inoperable MSSVs, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 4 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.1.1

This SR verifies the OPERABILITY of the MSSVs by the verification of each MSSV lift setpoint in accordance with the Inservice Testing Program. The ASME Code, ~~Section XI~~ (Ref. 5), requires that safety and relief valve tests be performed in accordance with ANSI/ASME OM-1-1987 (Ref. 6). According to Reference 6, the following tests are required:

a. Visual examination;

b. Seat tightness determination;

c. Setpoint pressure determination (lift setting);

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.1.1 (continued)

d. Compliance with owner's seat tightness criteria; and

e. Verification of the balancing device integrity on balanced valves.

The ANSI/ASME Standard requires that all valves be tested every 5 years, and a minimum of 20% of the valves be tested every 24 months. The ASME Code specifies the activities and frequencies necessary to satisfy the requirements. Table 3.7.1-2 allows a ± 3% setpoint tolerance for OPERABILITY; however, the valves are reset to ± 1% during the Surveillance to allow for drift. The lift settings, according to Table 3.7.1-2, in the accompanying LCO, correspond to ambient conditions of the valve at nominal operating temperature and pressure.

This SR is modified by a Note that allows entry into and operation in MODE 3 prior to performing the SR. The MSSVs may be either bench tested or tested in situ at hot conditions using an assist device to simulate lift pressure. If the MSSVs are not tested at hot conditions, the lift setting pressure shall be corrected to ambient conditions of the valve at operating temperature and pressure.

REFERENCES

1. UFSAR, Section 10.3.1.

2. ASME, Boiler and Pressure Vessel Code, Section III, Article NC-7000, Class 2 Components.

3. UFSAR, Section 15.2.

4. NRC Information Notice 94-60, "Potential Overpressurization of the Main Steam System," August 22, 1994.

5. ~~ASME, Boiler and Pressure Vessel Code, Section XI.~~ INSERT 1

6. ANSI/ASME OM-1-1987 and applicable Addenda.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.2.1

This SR verifies that MSIV closure time is ≤ 5 seconds. The MSIV closure time is assumed in the accident and containment analyses. This Surveillance is normally performed upon returning the unit to operation following a refueling outage. Based on ASME Code ~~Section XI~~ (Ref. 5), the MSIVs are not closure time tested at power.

The Frequency is in accordance with the Inservice Testing Program. This test is conducted in MODE 3 with the unit at operating temperature and pressure. This SR is modified by two Notes. Note 1 allows entry into and operation in MODE 3 prior to performing the SR. This allows a delay of testing until MODE 3, to establish conditions consistent with those under which the acceptance criterion was generated. Note 2 allows the SR to not be met until the first startup after September 27, 2001.

SR 3.7.2.2

This SR verifies that each MSIV can close on an actual or simulated actuation signal. This Surveillance is normally performed upon returning the unit to operation following a refueling outage. The frequency of MSIV testing is every 18 months. The 18 month Frequency for testing is based on the refueling cycle. Operating experience has shown that these components usually pass the Surveillance when performed at the 18 month Frequency. Therefore, this Frequency is acceptable from a reliability standpoint.

This SR is modified by two Notes. Note 1 allows entry into and operation in MODE 3 prior to performing the SR. This allows a delay of testing until MODE 3, to establish conditions consistent with those under which the acceptance criterion was generated. Note 2 allows the SR to not be met until the first startup after September 27, 2001.

BASES

REFERENCES

1. UFSAR, Section 10.3.
2. UFSAR, Section 15.1.5.
3. UFSAR, Section 6.2.
4. 10 CFR 50.67.
5. ~~ASME, Boiler and Pressure Vessel Code, Section XI.~~

INSERT 1

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.5.4

Verifying that each AF pump's developed head at the flow test point is greater than or equal to the required developed head ensures that AF pump performance has not degraded during the cycle. Flow and differential head are normal tests of centrifugal pump performance required by ~~Section XI of~~ the ASME Code (Ref. 4). Because it is undesirable to introduce cold AF into the steam generators while they are operating, this testing is performed on recirculation flow. This test confirms one point on the pump design curve and is indicative of overall performance. Such inservice tests confirm component OPERABILITY, trend performance, and detect incipient failures by indicating abnormal performance. Performance of inservice testing discussed in the ASME Code, ~~Section XI~~ (Ref. 4) (only required at 3 month intervals) satisfies this requirement.

SR 3.7.5.5

This SR verifies that AF can be delivered to the steam generators in the event of any accident or transient that generates an ESFAS, by demonstrating that each automatic valve in the flow path actuates to its correct position on an actual or simulated actuation signal. This Surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls. The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. The 18 month Frequency is acceptable based on operating experience and the design reliability of the equipment.

BASES

REFERENCES

1. UFSAR, Section 10.4.9.

2. Westinghouse Nuclear Safety Evaluation Checklist, SECL 90-469, Revision 1, "Byron/Braidwood Units 1 and 2, Relaxation of MSSV Setpoint Tolerance to +/- 3% - Revised SECL."

3. UFSAR, Section 15.2.

4. ~~ASME, Boiler and Pressure Vessel Code, Section XI.~~

INSERT 1

5. ASTM Standards, D5452-98.

<u>INSERT 1</u>

ASME Code for Operation and Maintenance of Nuclear Power Plants.

**ATTACHMENT 4C**

DRESDEN NUCLEAR POWER STATION
UNITS 2 and 3

Docket Nos. 50-237 and 50-249

License Nos. DPR-19 and DPR-25

**Markup of Proposed Technical Specifications Bases Page Changes**

<u>REVISED TS BASES PAGES</u>

B 3.4.3-7
B 3.4.3-8
B 3.5.1-12
B 3.6.1.6-4
B 3.6.1.6-5
B 3.6.2.3-4

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.3.2 (continued)

The combination of the valve testing and the valve actuator testing provide a complete check of the capability of the valves to open and close, such that full functionality is demonstrated through overlapping tests, without cycling the valves.

The 24 month Frequency ensures that each solenoid for each relief valve is tested. The 24 month Frequency was developed based on the relief valve tests required by the ASME Boiler and Pressure Vessel Code~~, Section XI~~ (Ref. 5). Operating experience has shown that these components usually pass the Surveillance when performed at the 24 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

SR 3.4.3.3

The relief valves, including the S/RV, are required to actuate automatically upon receipt of specific initiation signals. A system functional test is performed to verify that the mechanical portions (i.e., solenoids) of the relief valve operate as designed when initiated either by an actual or simulated automatic initiation signal. The LOGIC SYSTEM FUNCTIONAL TESTs in LCO 3.3.5.1, "Emergency Core Cooling System (ECCS) Instrumentation," and LCO 3.3.6.3, "Relief Valve Instrumentation," overlap this SR to provide complete testing of the safety function.

The 24 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 these components usually pass the Surveillance when performed at the 24 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

This SR is modified by a Note that excludes valve actuation since the valves are individually tested in accordance with SR 3.4.3.2.

(continued)

BASES (continued)

REFERENCES

1. UFSAR, Section 5.2.2.
2. UFSAR, Section 15.2.3.1.
3. UFSAR, Section 15.2.2.1.
4. UFSAR, Chapter 15.
5. ~~ASME, Boiler and Pressure Vessel Code, Section XI.~~

INSERT 1

BASES

SURVEILLANCE REQUIREMENTS

SR 3.5.1.4 (continued)

The Frequency of this SR is in accordance with the Inservice Testing Program. If any recirculation pump discharge valve is inoperable and in the open position, both LPCI subsystems must be declared inoperable.

SR 3.5.1.5, SR 3.5.1.6, and SR 3.5.1.7

The performance requirements of the low pressure ECCS pumps are determined through application of the 10 CFR 50, Appendix K criteria (Ref. 7) and are bounded by the requirements of SR 3.5.1.5. This periodic Surveillance is performed (in accordance with the ASME Code, ~~Section XI~~, requirements for the ECCS pumps) to verify that the ECCS pumps will develop the flow rates required by the respective analyses. The low pressure ECCS pump flow rates ensure that adequate core cooling is provided to satisfy the acceptance criteria of Reference 9. The pump flow rates are verified against a test line pressure or system head equivalent to the RPV pressure expected during a LOCA. The total system pump outlet pressure is adequate to overcome the elevation head pressure between the pump suction and the vessel discharge, the piping friction losses, and RPV pressure present during a LOCA. These values have been established analytically.

The flow tests for the HPCI System are performed at two different pressure ranges such that system capability to provide rated flow against a system head corresponding to reactor pressure is tested at both the higher and lower operating ranges of the system. The required system head should overcome the RPV pressure and associated discharge line losses. Adequate reactor steam pressure must be available to perform these tests. Additionally, adequate steam flow must be passing through the main turbine or turbine bypass valves to continue to control reactor pressure when the HPCI System diverts steam flow. Therefore, sufficient time is allowed after adequate pressure and flow are achieved to perform these tests. Reactor steam pressure must be ≥ 920 psig to perform SR 3.5.1.6 and ≥ 150 psig to perform SR 3.5.1.7. Adequate steam flow is represented by at least 2 turbine bypass

(continued)

SURVEILLANCE REQUIREMENTS

SR 3.6.1.6.1 (continued)

by verification of set pressure, and delay. The valve will then be shipped to the plant without any disassembly or alteration of the main valve or pilot valve components.

The combination of the valve testing and the valve actuator testing provide a complete check of the capability of the valves to open and close, such that full functionality is demonstrated through overlapping tests, without cycling the valves.

The 24 month Frequency was based on the relief valve tests required by the ASME Boiler and Pressure Vessel Code, ~~Section XI~~ (Ref. 2). The Frequency of 24 months ensures that each solenoid for each low set relief valve is tested. Operating experience has shown that these components usually pass the Surveillance when performed at the 24 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

SR 3.6.1.6.2

The low set relief designated relief valves are required to actuate automatically upon receipt of specific initiation signals. A system functional test is performed to verify that the mechanical portions (i.e., solenoids) of the low set relief function operate as designed when initiated either by an actual or simulated automatic initiation signal. The LOGIC SYSTEM FUNCTIONAL TEST in LCO 3.3.6.3, "Low Set Relief Valve Instrumentation," overlaps this SR to provide complete testing of the safety function.

The 24 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 these components usually pass the Surveillance when performed at the 24 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

This SR is modified by a Note that excludes valve actuation. This prevents a reactor pressure vessel pressure blowdown.

(continued)

BASES (continued)

| | | |
|---|---|---|
| REFERENCES | 1. | UFSAR, Section 6.2.1.3.5.3. INSERT 1 |
| | 2. | ~~ASME, Boiler and Pressure Vessel Code, Section X~~I. |

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.2.3.1 (continued)

event requiring initiation of the system is low, and the system is a manually initiated system. This Frequency has been shown to be acceptable based on operating experience.

SR 3.6.2.3.2

Verifying that each required LPCI pump develops a flow rate ≥ 5000 gpm while operating in the suppression pool cooling mode with flow through the associated heat exchanger ensures that the primary containment peak pressure and temperature can be maintained below the design limits during a DBA (Ref. 1). The flow is a normal test of centrifugal pump performance required by ASME Code~~, Section XI~~ (Ref. 2). This test confirms one point on the pump design curve, and the results are indicative of overall performance. Such inservice tests confirm component OPERABILITY, and detect incipient failures by indicating abnormal performance. The Frequency of this SR is in accordance with the Inservice Testing Program.

REFERENCES

1. UFSAR, Section 6.2.
2. ~~ASME, Boiler and Pressure Vessel Code, Section XI.~~

INSERT 1

INSERT 1

ASME Code for Operation and Maintenance of Nuclear Power Plants.

**ATTACHMENT 4D**

LIMERICK GENERATING STATION
UNITS 1 and 2

Docket Nos. 50-352 and 50-353

License Nos. NPF-39 and NPF-85

**Markup of Proposed Technical Specifications Bases Page Changes**

<u>REVISED TS BASES PAGES</u>

<u>UNIT 1</u>

B 3/4 0-6

<u>UNIT 2</u>

B 3/4 0-6

3/4.0 APPLICABILITY

BASES

SR(s) are not required to be performed, per Specification 4.0.1, which states that surveillances do not have to be performed on inoperable equipment. When equipment is inoperable, Specification 4.0.4 does not apply to the associated SR(s) since the requirement for the SR(s) to be performed is removed. Therefore, failing to perform the Surveillance(s) within the specified Surveillance time interval does not result in a Specification 4.0.4 restriction to changing OPERATIONAL CONDITIONS or other specified conditions of the Applicability. However, since the Limiting Condition for Operation is not met in this instance, Specification 3.0.4 will govern any restrictions that may (or may not) apply to OPERATIONAL CONDITION or other specified condition changes. Specification 4.0.4 does not restrict changing OPERATIONAL CONDITIONS or other specified conditions of the Applicability when a Surveillance has not been performed within the specified Surveillance time interval, provided the requirement to declare the Limiting Condition for Operation not met has been delayed in accordance with Specification 4.0.3.

The provisions of Specification 4.0.4 shall not prevent entry into OPERATIONAL CONDITIONS or other specified conditions in the Applicability that are required to comply with ACTION requirements. In addition, the provisions of Specification 4.0.4 shall not prevent changes in OPERATIONAL CONDITIONS or other specified conditions in the Applicability that result from any unit shutdown. In this context, a unit shutdown is defined as a change in OPERATIONAL CONDITION or other specified condition in the Applicability associated with transitioning from OPERATIONAL CONDITION 1 to OPERATIONAL CONDITION 2, OPERATIONAL CONDITION 2 to OPERATIONAL CONDITION 3, and OPERATIONAL CONDITION 3 to OPERATIONAL CONDITION 4.

Specification 4.0.5 establishes the requirement that inservice inspection of ASME Code Class 1, 2 and 3 components and inservice testing of ASME Code Class 1, 2 and 3 pumps and valves shall be performed in accordance with a periodically updated version of Section XI of the ASME Boiler and Pressure Vessel Code and Addenda as required by 10 CFR 50.55a. Additionally, the Inservice Inspection Program conforms to the NRC staff positions identified in NRC Generic Letter 88-01, "NRC Position on IGSCC in BWR Austinetic Stainless Steel Piping," as approved in NRC Safety Evaluations dated March 6, 1990 and October 22, 1990, or in accordance with alternate measures approved by the NRC staff. INSERT C

INSERT C

This specification includes a clarification of the frequencies for performing the inservice inspection and testing activities required by Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda. This clarification is provided to ensure consistency in surveillance intervals throughout the Technical Specifications and to remove any ambiguities relative to the frequencies for performing the required inservice inspection and testing activities.

Under the terms of this specification, the more restrictive requirements of the Technical Specifications take precedence over the ASME ~~Boiler and Pressure Vessel~~ Code and applicable Addenda. The requirements of ~~Specification 4.0.4 to~~ perform surveillance activities before entry into an OPERATIONAL CONDITION or other specified condition takes precedence over the ASME ~~Boiler and Pressure Vessel~~ Code provision that allows pumps and valves to be tested up ~~to one week after~~ return to normal operation. The Technical Specification definition of OPERABLE does not allow a grace period before a component, which is not capable of performing its specified function, is declared inoperable and takes precedence over the ASME ~~Boiler and Pressure Vessel~~ Code provision that allows a valve to be incapable of performing its specified function for up to 24 hours before being declared inoperable.

condition change. When a system, subsystem, division, component, device, or variable is inoperable or outside its specified limits, the associated SR(s) are not required to be performed, per Specification 4.0.1, which states that surveillances do not have to be performed on inoperable equipment. When equipment is inoperable, Specification 4.0.4 does not apply to the associated SR(s) since the requirement for the SR(s) to be performed is removed. Therefore, failing to perform the Surveillance(s) within the specified Surveillance time interval does not result in a Specification 4.0.4 restriction to changing OPERATIONAL CONDITIONS or other specified conditions of the Applicability. However, since the Limiting Condition for Operation is not met in this instance, Specification 3.0.4 will govern any restrictions that may (or may not) apply to OPERATIONAL CONDITION or other specified condition changes. Specification 4.0.4 does not restrict changing OPERATIONAL CONDITIONS or other specified conditions of the Applicability when a Surveillance has not been performed within the specified Surveillance time interval, provided the requirement to declare the Limiting Condition for Operation not met has been delayed in accordance with Specification 4.0.3.

The provisions of Specification 4.0.4 shall not prevent entry into OPERATIONAL CONDITIONS or other specified conditions in the Applicability that are required to comply with ACTION requirements. In addition, the provisions of Specification 4.0.4 shall not prevent changes in OPERATIONAL CONDITIONS or other specified conditions in the Applicability that result from any unit shutdown. In this context, a unit shutdown is defined as a change in OPERATIONAL CONDITION or other specified condition in the Applicability associated with transitioning from OPERATIONAL CONDITION 1 to OPERATIONAL CONDITION 2, OPERATIONAL CONDITION 2 to OPERATIONAL CONDITION 3, and OPERATIONAL CONDITION 3 to OPERATIONAL CONDITION 4.

Specification 4.0.5 establishes the requirement that inservice inspection of ASME Code Class 1, 2 and 3 components and inservice testing of ASME Code Class 1, 2 and 3 pumps and valves shall be performed in accordance with a periodically updated version of Section XI of the ASME Boiler and Pressure Vessel Code and Addenda as required by 10 CFR 50.55a. Additionally, the Inservice Inspection Program conforms to the NRC staff positions identified in NRC Generic Letter 88-01, "NRC Position on IGSCC in BWR Austinetic Stainless Steel Piping," as approved in NRC Safety Evaluations dated March 6, 1990 and October 22, 1990, or in accordance with alternate measures approved by the NRC staff. INSERT C

INSERT C

This specification includes a clarification of the frequencies for performing the inservice inspection and testing activities required by Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda. This clarification is provided to ensure consistency in surveillance intervals throughout the Technical Specifications and to remove any ambiguities relative to the frequencies for performing the required inservice inspection and testing activities.

Under the terms of this specification, the more restrictive requirements of the Technical Specifications take precedence over the ASME Boiler and pressure Vessel Code and applicable Addenda. The requirements of ~~Specification 4.0.4 to~~ perform surveillance activities before entry into an OPERATIONAL CONDITION or other specified condition takes precedence over the ASME Boiler and Pressure Vessel Code provision that allows pumps and valves ~~to be tested~~ up to one week after return to normal operation. The Technical Specification definition of OPERABLE does not allow a grace period before a component, which is not capable of performing its specified function, is declared inoperable and takes precedence over the ASME ~~Boiler and Pressure Vessel~~ Code provision that allows a valve to be incapable ~~of performing its specified~~ function for up to 24 hours before being declared inoperable.

INSERT C

, and the ASME Code for Operation and Maintenance of Nuclear Power Plants (ASME OM Code) and applicable Addenda

**ATTACHMENT 4E**

PEACH BOTTOM ATOMIC POWER STATION
UNITS 2 and 3

Docket Nos. 50-277 and 50-278

License Nos. DPR-44 and DPR-56

**Markup of Proposed Technical Specifications Bases Page Changes**

<u>REVISED TS BASES PAGES</u>

<u>UNIT 2</u>
B 3.4-15
B 3.6-59

<u>UNIT 3</u>
B 3.4-15
B 3.6-59

## B 3.4 REACTOR COOLANT SYSTEM (RCS)

## B 3.4.3 Safety Relief Valves (SRVs) and Safety Valves (SVs)

BASES

BACKGROUND

The ASME Boiler and Pressure Vessel Code requires the reactor pressure vessel be protected from overpressure during upset conditions by self-actuated safety valves. As part of the nuclear pressure relief system, the size and number of SRVs and SVs are selected such that peak pressure in the nuclear system will not exceed the ASME Code limits for the reactor coolant pressure boundary (RCPB).

The SRVs and SVs are located on the main steam lines between the reactor vessel and the first isolation valve within the drywell. The SRVs can actuate by either of two modes: the safety mode or the depressurization mode. In the safety mode, the pilot disc opens when steam pressure at the valve inlet expands the bellows to the extent that the hydraulic seating force on the pilot disc is reduced to zero. Opening of the pilot stage allows a pressure differential to develop across the second stage disc which opens the second stage disc, thus venting the chamber over the main valve piston. This causes a pressure differential across the main valve piston which opens the main valve. The SVs are spring loaded valves that actuate when steam pressure at the inlet overcomes the spring force holding the valve disc closed. This satisfies the Code requirement.

Each of the 11 SRVs discharge steam through a discharge line to a point below the minimum water level in the suppression pool. The two SVs discharge steam directly to the drywell. In the depressurization mode, the SRV is opened by a pneumatic actuator which opens the second stage disc. The main valve then opens as described above for the safety mode. The depressurization mode provides controlled depressurization of the reactor coolant pressure boundary. All 11 of the SRVs function in the safety mode and have the capability to operate in the depressurization mode via manual actuation from the control room. Five of the SRVs are allocated to the Automatic Depressurization System (ADS). The ADS requirements are specified in LCO 3.5.1, "ECCS—Operating."

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.2.3.1 (continued)

the time assumed in the accident analysis. This is acceptable since the RHR suppression pool cooling mode is manually initiated. This SR does not require any testing or valve manipulation; rather, it involves verification that those valves capable of being mispositioned are in the correct position. This SR does not apply to valves that cannot be inadvertently misaligned, such as check valves.

The Frequency of 31 days is justified because the valves are operated under procedural control, improper valve position would affect only a single subsystem, the probability of an event requiring initiation of the system is low, and the subsystem is a manually initiated system. This Frequency has been shown to be acceptable based on operating experience.

SR 3.6.2.3.2

Verifying that each required RHR pump develops a flow rate ≥ 10,000 gpm while operating in the suppression pool cooling mode with flow through the associated heat exchanger ensures that pump performance has not degraded during the cycle. Flow is a normal test ~~of centrifugal~~ pump performance required by ASME Code Section XI (Ref. 2). This test confirms one point on ~~the pump~~ design curve, and the results are indicative of overall performance. Such inservice inspections confirm component OPERABILITY, trend performance, and detect incipient failures by indicating abnormal performance. The Frequency of this SR is in accordance with the Inservice Testing Program.

REFERENCES

1. UFSAR, Section 14.6.3.
2. ASME, Boiler and Pressure Vessel Code, Section XI.

ASME Code for Operation and Maintenance of Nuclear Power Plants.

# B 3.4 REACTOR COOLANT SYSTEM (RCS)

## B 3.4.3 Safety Relief Valves (SRVs) and Safety Valves (SVs)

BASES

**BACKGROUND**

The ASME Boiler and Pressure Vessel Code requires the reactor pressure ~~vessel be protect~~ed from overpressure during upset conditions by self-actuated safety valves. As part of the nuclear pressure relief system, the size and number of SRVs and SVs are selected such that peak pressure in the nuclear system will not exceed the ASME Code limits for the reactor coolant pressure boundary (RCPB).

The SRVs and SVs are located on the main steam lines between the reactor vessel and the first isolation valve within the drywell. The SRVs can actuate by either of two modes: the safety mode or the depressurization mode. In the safety mode, the pilot disc opens when steam pressure at the valve inlet expands the bellows to the extent that the hydraulic seating force on the pilot disc is reduced to zero. Opening of the pilot stage allows a pressure differential to develop across the second stage disc which opens the second stage disc, thus venting the chamber over the main valve piston. This causes a pressure differential across the main valve piston which opens the main valve. The SVs are spring loaded valves that actuate when steam pressure at the inlet overcomes the spring force holding the valve disc closed. This satisfies the Code requirement.

Each of the 11 SRVs discharge steam through a discharge line to a point below the minimum water level in the suppression pool. The two SVs discharge steam directly to the drywell. In the depressurization mode, the SRV is opened by a pneumatic actuator which opens the second stage disc. The main valve then opens as described above for the safety mode. The depressurization mode provides controlled depressurization of the reactor coolant pressure boundary. All 11 of the SRVs function in the safety mode and have the capability to operate in the depressurization mode via manual actuation from the control room. Five of the SRVs are allocated to the Automatic Depressurization System (ADS). The ADS requirements are specified in LCO 3.5.1, "ECCS—Operating."

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.2.3.1 (continued)

the time assumed in the accident analysis. This is acceptable since the RHR suppression pool cooling mode is manually initiated. This SR does not require any testing or valve manipulation; rather, it involves verification that those valves capable of being mispositioned are in the correct position. This SR does not apply to valves that cannot be inadvertently misaligned, such as check valves.

The Frequency of 31 days is justified because the valves are operated under procedural control, improper valve position would affect only a single subsystem, the probability of an event requiring initiation of the system is low, and the subsystem is a manually initiated system. This Frequency has been shown to be acceptable based on operating experience.

SR 3.6.2.3.2

Verifying that each required RHR pump develops a flow rate ≥ 10,000 gpm while operating in the suppression pool cooling mode with flow through the associated heat exchanger ensures that pump performance has not degraded during the cycle. Flow is a normal test of ~~centrifugal~~ pump performance required by ASME Code, Section XI (Ref. 2). This test confirms one point on the pump design curve, and the results are indicative of overall performance. Such inservice inspections confirm component OPERABILITY, trend performance, and detect incipient failures by indicating abnormal performance. The Frequency of this SR is in accordance with the Inservice Testing Program.

REFERENCES

1. UFSAR, Section 14.6.3.

2. ASME, Boiler and Pressure Vessel Code, Section XI.

ASME Code for Operation and Maintenance of Nuclear Power Plants.

**ATTACHMENT 4F**

QUAD CITIES NUCLEAR POWER STATION
UNITS 1 and 2

Docket Nos. 50-254 and 50-265

License Nos. DPR-29 and DPR-30

**Markup of Proposed Technical Specifications Bases Page Changes**

<u>REVISED TS BASES PAGES</u>

B 3.4.3-6
B 3.4.3-7
B 3.5.1-12
B 3.5.1-18
B 3.6.1.6-4
B 3.6.1.6-5
B 3.6.2.3-4

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.3.2 (continued)

For the ERVs, the actuator test is performed with the pilot valve actuator mounted in its normal position. This will allow testing of the manual actuation electrical circuitry, solenoid actuator, pilot operating lever, and pilot plunger. This test will verify pilot valve movement. However, since this test is performed prior to establishing the reactor pressure needed to overcome main valve closure spring force, the main valve will not stroke during the test.

This SR, together with the valve testing performed as required by the ASME Code for pressure relieving devices (ASME OM Code-1998 through 2000 Addenda), verify the capability of each relief valve to perform its function.

Valve testing will be performed at a steam test facility, where the valve (i.e., main valve and pilot valve) and an actuator representative of the actuator used at the plant will be installed on a steam header in the same orientation as the plant installation. The test conditions in the test facility will be similar to those in the plant installation, including ambient temperature, valve insulation, and steam conditions. The valve will then be leak tested, functionally tested to ensure the valve is capable of opening and closing (including stroke time), and leak tested a final time. Valve seat tightness will be verified by a cold bar test, and if not free of fog, leakage will be measured and verified to be below design limits. In addition, for the safety mode of S/RVs, an as-found setpoint verification and as-found leak check are performed, followed by verification of set pressure, and delay. The valve will then be shipped to the plant without any disassembly or alteration of the main valve or pilot valve components.

The combination of the valve testing and the valve actuator testing provide a complete check of the capability of the valves to open and close, such that full functionality is demonstrated through overlapping tests, without cycling the valves.

The 24 month Frequency ensures that each solenoid for each relief valve is tested. The 24 month Frequency was developed based on the relief valve tests required by the ASME ~~Boiler and Pressure Vessel~~ Code~~, Section XI~~ (Ref. 5). Operating experience has shown that these components usually

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.3.2 (continued)

pass the Surveillance when performed at the 24 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

SR 3.4.3.3

The relief valves, including the S/RV, are required to actuate automatically upon receipt of specific initiation signals. A system functional test is performed to verify that the mechanical portions (i.e., solenoids) of the relief valve operate as designed when initiated either by an actual or simulated automatic initiation signal. The LOGIC SYSTEM FUNCTIONAL TESTs in LCO 3.3.5.1, "Emergency Core Cooling System (ECCS) Instrumentation," and LCO 3.3.6.3, "Relief Valve Instrumentation," overlap this SR to provide complete testing of the safety function.

The 24 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 these components usually pass the Surveillance when performed at the 24 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

This SR is modified by a Note that excludes valve actuation since the valves are individually tested in accordance with SR 3.4.3.2.

REFERENCES

1. UFSAR, Section 5.2.2.1.
2. UFSAR, Section 15.2.3.1.
3. UFSAR, Section 15.2.2.1.
4. UFSAR, Chapter 15.
5. ~~ASME, Boiler and Pressure Vessel Code, Section XI.~~ INSERT 1

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.5.1.3

Verification every 31 days of the correct breaker alignment to the LPCI swing bus demonstrates that the AC electrical power is available to ensure proper operation of the associated LPCI injection valves and the recirculation pump discharge valves. The 31 day Frequency has been found acceptable based on engineering judgment and operating experience.

SR 3.5.1.4

Cycling the recirculation pump discharge valves through one complete cycle of full travel demonstrates that the valves are mechanically OPERABLE and will close when required. Upon initiation of an automatic LPCI subsystem injection signal, these valves are required to be closed to ensure full LPCI subsystem flow injection in the reactor via the recirculation jet pumps. De-energizing the valve in the closed position will also ensure the proper flow path for the LPCI subsystem. Acceptable methods of de-energizing the valve include de-energizing breaker control power, racking out the breaker or removing the breaker.

The Frequency of this SR is in accordance with the Inservice Testing Program. If any recirculation pump discharge valve is inoperable and in the open position, both LPCI subsystems must be declared inoperable.

SR 3.5.1.5, SR 3.5.1.6, and SR 3.5.1.7

The performance requirements of the low pressure ECCS pumps are determined through application of the 10 CFR 50, Appendix K criteria (Ref. 7). This periodic Surveillance is performed (in accordance with the ASME Code~~, Section XI,~~ (Ref. 11) requirements for the ECCS pumps) to verify that the ECCS pumps will develop the flow rates required by the respective analyses. The low pressure ECCS pump flow rates ensure that adequate core cooling is provided to satisfy the acceptance criteria of Reference 9. The pump flow rates are verified against a test line pressure or system head equivalent to

(continued)

BASES (continued)

REFERENCES

1. UFSAR, Section 6.3.2.1.
2. UFSAR, Section 6.3.2.2.
3. UFSAR, Section 6.3.2.3.
4. UFSAR, Section 6.3.2.4.
5. UFSAR, Section 15.6.4.
6. UFSAR, Section 15.6.5.
7. 10 CFR 50, Appendix K.
8. UFSAR, Section 6.3.3.
9. 10 CFR 50.46.
10. Memorandum from R.L. Baer (NRC) to V. Stello, Jr. (NRC), "Recommended Interim Revisions to LCOs for ECCS Components," December 1, 1975.
11.

INSERT 1

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.1.6.1 (continued)

verification and as-found leak check are performed, followed by verification of set pressure, and delay. The valve will then be shipped to the plant without any disassembly or alteration of the main valve or pilot valve components.

The combination of the valve testing and the valve actuator testing provide a complete check of the capability of the valves to open and close, such that full functionality is demonstrated through overlapping tests, without cycling the valves.

The 24 month Frequency was based on the relief valve tests required by the ASME ~~Boiler and Pressure Vessel~~ Code, ~~Section XI~~ (Ref. 2). The Frequency of 24 months ensures that each solenoid for each low set relief valve is tested. Operating experience has shown that these components usually pass the Surveillance when performed at the 24 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

SR 3.6.1.6.2

The low set relief designated relief valves are required to actuate automatically upon receipt of specific initiation signals. A system functional test is performed to verify that the mechanical portions (i.e., solenoids) of the low set relief function operate as designed when initiated either by an actual or simulated automatic initiation signal. The LOGIC SYSTEM FUNCTIONAL TEST in LCO 3.3.6.3, "Low Set Relief Valve Instrumentation," overlaps this SR to provide complete testing of the safety function.

The 24 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 these components usually pass the Surveillance when performed at the 24 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

This SR is modified by a Note that excludes valve actuation. This prevents a reactor pressure vessel pressure blowdown.

(continued)

BASES (continued)

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REFERENCES

1. UFSAR, Section 6.2.1.3.4.2.
2. ~~ASME, Boiler and Pressure Vessel Code, Section XI.~~ $\boxed{\text{INSERT 1}}$

BASES

| | |
|---|---|
| SURVEILLANCE REQUIREMENTS | SR 3.6.2.3.1 (continued)<br><br>The Frequency of 31 days is justified because the valves are operated under procedural control, improper valve position would affect only a single subsystem, the probability of an event requiring initiation of the system is low, and the system is a manually initiated system. This Frequency has been shown to be acceptable based on operating experience.<br><br>SR 3.6.2.3.2<br><br>Verifying that each required RHR pump develops a flow rate ≥ 5000 gpm while operating in the suppression pool cooling mode with flow through the associated heat exchanger ensures that the primary containment peak pressure and temperature can be maintained below the design limits during a DBA (Ref. 1). The flow is a normal test of centrifugal pump performance required by ASME Code~~, Section XI~~ (Ref. 2). This test confirms one point on the pump design curve, and the results are indicative of overall performance. Such inservice tests confirm component OPERABILITY, and detect incipient failures by indicating abnormal performance. The Frequency of this SR is in accordance with the Inservice Testing Program. |
| REFERENCES | 1. UFSAR, Section 6.2.<br><br>INSERT 1<br><br>2. ~~ASME, Boiler and Pressure Vessel Code, Section XI.~~ |

INSERT 1

ASME Code for Operation and Maintenance of Nuclear Power Plants.