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Exelon Nuclear

RS-01-029

February 28, 2001

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555-0001

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

LaSalle County Station, Units 1 and 2
Facility Operating License Nos. NPF-11 and NPF-18
NRC Docket Nos. 50-373 and 50-374

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

Subject: Revision E to Request for Technical Specifications Changes and Proposed License Conditions Supporting the Implementation of Improved Standard Technical Specifications for Dresden Nuclear Power Station, Units 2 and 3, LaSalle County Station, Units 1 and 2, and Quad Cities Nuclear Power Station, Units 1 and 2

Reference:

1) Letter from R. M. Krich (Commonwealth Edison Company) to U.S. NRC, "Request for Technical Specifications Changes for Dresden Nuclear Power Station, Units 2 and 3, LaSalle County Station, Units 1 and 2, and Quad Cities Nuclear Power Station, Units 1 and 2, to Implement Improved Standard Technical Specifications," dated March 3, 2000

2) Letter from R. M. Krich (Commonwealth Edison Company) to U.S. NRC, "Revision A to Request for Technical Specifications Changes for Dresden Nuclear Power Station, Units 2 and 3, LaSalle County Station, Units 1 and 2, and Quad Cities Nuclear Power Station, Units 1 and 2, to Implement Improved Standard Technical Specifications," dated June 5, 2000

3) Letter from R. M. Krich (Commonwealth Edison Company) to U.S. NRC, "Revision B to Request for Technical Specifications Changes for Dresden Nuclear Power Station, Units 2 and 3, LaSalle County Station, Units 1 and 2, and Quad Cities Nuclear Power Station, Units 1 and 2, to Implement Improved Standard Technical Specifications," dated September 1, 2000

A001

4) Letter from R. M. Krich (Commonwealth Edison Company) to U.S. NRC, "Revision C to Request for Technical Specifications Changes for Dresden Nuclear Power Station, Units 2 and 3, LaSalle County Station, Units 1 and 2, and Quad Cities Nuclear Power Station, Units 1 and 2, to Implement Improved Standard Technical Specifications," dated December 18, 2000

5) Letter from R. M. Krich (Commonwealth Edison Company) to U.S. NRC, "Revision D to Request for Technical Specifications Changes for Dresden Nuclear Power Station, Units 2 and 3, LaSalle County Station, Units 1 and 2, and Quad Cities Nuclear Power Station, Units 1 and 2, to Implement Improved Standard Technical Specifications," dated February 15, 2001

In Reference 1, in accordance with 10 CFR 50.90, "Application for amendment of license or construction permit," we proposed to amend Appendix A, Technical Specifications (TS) of Facility Operating License Nos. DPR-19, DPR-25, NPF-11, NPF-18, DPR-29 and DPR-30 for Dresden Nuclear Power Station, Units 2 and 3, LaSalle County Station, Units 1 and 2, and Quad Cities Nuclear Power Station, Units 1 and 2, respectively. The proposed changes revise the Dresden Nuclear Power Station, Units 2 and 3, LaSalle County Station, Units 1 and 2, and Quad Cities Nuclear Power Station, Units 1 and 2, current Technical Specifications (CTS) to a format and content consistent with NUREG-1433, Revision 1, "Standard Technical Specifications for General Electric Plants, BWR 4," and NUREG-1434, Revision 1, "Standard Technical Specifications for General Electric Plants, BWR 6," as applicable. References 2, 3, 4 and 5 subsequently supplemented the proposed amendment.

This letter submits a modification to the proposed changes. This modification and its source are described in the "Summary of Changes" section of Attachment 1. This modification was identified while planning for the implementation of setpoint changes at the Dresden Nuclear Power Station. This letter also describes license conditions that are either required to implement the Improved Technical Specifications (ITS) or no longer necessary after the implementation of ITS.

In the conversion to Improved Standard TS, Exelon Generation Company (EGC), LLC, formerly Commonwealth Edison (ComEd) Company, committed to relocate certain requirements from the CTS to licensee controlled documents as described in References 1, 2, 3, 4 and 5. To make this commitment enforceable in accordance with NRC requests, the following license condition is requested to be issued.

> EGC shall relocate certain Technical Specifications requirements to EGC-controlled documents. Implementation of the Improved Technical Specifications shall include relocating these certain Technical Specifications requirements to the appropriate documents, as described in Table LA – Removal of Details Matrix and Specification Requirements and Table R – Relocated Specifications, that are attached to the NRC's Safety Evaluation enclosed with this amendment.

As part of the overall conversion to the Improved Standard Technical Specifications, EGC is adopting numerous new and revised Surveillance Requirements (SRs). In order to facilitate implementation of these SRs, it is requested that a license condition be issued that defines the schedule for performance of the new and revised SRs during or after the implementation of the ITS. The License Condition should read as follows.

> The schedule for performing new and revised Surveillance Requirements (SRs) shall be as follows.
>
> 1. For SRs that are new in this amendment, the first performance is due at the end of the first surveillance interval that begins on the date of implementation of this amendment.
>
> 2. For SRs that existed prior to this amendment whose intervals of performance are being reduced, the first reduced surveillance interval begins upon completion of the first surveillance performed after implementation of this amendment.
>
> 3. For SRs that existed prior to this amendment that have modified acceptance criteria, the first performance is due at the end of the first surveillance interval that began on the date the surveillance was last performed prior to the implementation of this amendment.
>
> 4. For SRs that existed prior to this amendment whose intervals of performance are being extended, the first extended surveillance interval begins upon completion of the last surveillance performed prior to implementation of this amendment.

Additionally, because of inaccessibility of the Main Steam Isolation Valves (MSIVs) during plant operation, it will not be possible to set certain MSIV limit switches within the allowable values required by ITS Table 3.3.1.1-1 Function 5. These devices are reliable mechanical switches that have demonstrated good operating performance at Dresden and Quad Cities. These switches are mechanical devices and are not subject to significant drift between calibration cycles. The MSIV limit switches are operable under the CTS setpoint allowances. The current field setpoints are within the analytical limit for these instruments; however, due to the new setpoint methodology utilized under ITS, a minor adjustment to the setpoint is required to meet the resultant ITS allowable value. For this reason, the following license condition is requested for the Dresden Nuclear Power Station (DNPS) and the Quad Cities Nuclear Power Stations (QCNPS). The MSIVs located in DNPS Unit 2 and QCNPS Unit 2 currently are set below the analytical limit, but not at or below the ITS allowable value, creating the need for this license condition.

> For ITS Table 3.3.1.1-1 Function 5, Main Steam Isolation Valve – Closure, the former Technical Specification setpoint (i.e., $\leq$ 10% closed) is required to be met until the unit startup after the first outage of sufficient duration following the ITS implementation date.

In addition, with the incorporation of the changes to Specification 3.10.5, "Multiple Control Rod Withdrawal - Refueling" and Surveillance Requirement 3.10.5.3 described in Reference 5 at the LaSalle County Station, two existing license conditions are no longer required and may be deleted. The affected license conditions are Condition 37 for LaSalle County Station Unit 1 and Condition 21 for LaSalle County Station Unit 2.

The modification and the proposed license conditions have been reviewed and approved by the respective Plant Operations Review Committees and the Nuclear Safety Review Board in accordance with the Quality Assurance Program. We have reviewed the information contained in this letter and its attachment and determined that it does not affect the information supporting a finding of no significant hazards consideration provided in References 1, 2, 3, 4, and 5.

EGC is notifying the State of Illinois of these modifications to previously submitted license amendment requests by transmitting a copy of this letter, including attachments and enclosures, to the designated state official.

Should you have any questions concerning this submittal, please contact Mr. J. V. Sipek at (630) 663-3741.

Respectfully,

R. M. Krich
Director-Licensing
Mid-West Regional Operating Group

Attachments: Affidavit
Attachment 1 - Revision E to Dresden Improved Technical Specifications Document

cc: Regional Administrator - NRC Region III
NRC Senior Resident Inspector - Dresden Nuclear Power Station
NRC Senior Resident Inspector - LaSalle County Station
w/o Attachment
NRC Senior Resident Inspector - Quad Cities Nuclear Power Station
w/o Attachment
Office of Nuclear Facility Safety - Illinois Department of Nuclear Safety

STATE OF ILLINOIS	)	
COUNTY OF DUPAGE	)	
IN THE MATTER OF	)	
COMMONWEALTH EDISON (COMED) COMPANY	)	**Docket Nos.**
DRESDEN NUCLEAR POWER STATION - UNITS 2 and 3	)	**50- 237and 50-249**
LASALLE COUNTY STATION - UNITS 1 and 2	)	**50- 373 and 50-374**
QUAD CITIES NUCLEAR POWER STATION - UNITS 1 and 2	)	**50- 254 and 50-265**

SUBJECT: **Revision E to Request for Technical Specifications Changes and Proposed License Conditions Supporting the Implementation of Improved Standard Technical Specifications for Dresden Nuclear Power Station, Units 2 and 3, LaSalle County Station, Units 1 and 2, and Quad Cities Nuclear Power Station, Units 1 and 2,**

AFFIDAVIT

I affirm that the content of this transmittal is true and correct to the best of my knowledge, information and belief.

R. M. Krich
Director, Licensing
Mid-West Regional Operating Group

Subscribed and sworn to before me, a Notary Public in and for the State above named, this 28th day of February, 2001.

Notary Public

* OFFICIAL SEAL *
Timothy A. Byam
Notary Public, State of Illinois
My Commission Expires 11/24/2001

ATTACHMENT 1

Revision E to Dresden Nuclear Power Station, Units 2 and 3
Proposed Improved Technical Specifications Submittal
dated March 3, 2000

Revision E to Dresden Nuclear Power Station
Improved Technical Specifications Summary of Changes

This attachment provides a brief summary of the changes in Revision E of the proposed Improved Technical Specifications (ITS) submittal for Dresden Nuclear Power Station, Units 2 and 3. The original Technical Specifications amendment request (i.e., Revision 0) was submitted to the NRC by letter dated March 3, 2000, as revised in Revisions A, B, C, and D submitted to the NRC by letters dated June 5, 2000, September 1, 2000, December 18, 2000, and February 15, 2001, respectively.

The changes included in this revision are editorial corrections.

Section 3.3

1. The inequality sign for the Allowable Value of ITS 3.3.5.1 Function 2.g has been corrected to be consistent with the current setpoint calculations. The change affects ITS 3.3.5.1 page 3.3.5.1-10 and the ISTS markup insert page 3.3-44.

2. The Allowable Values for ITS 3.3.6.1 Function 2.b and ITS 3.3.6.2 Function 2 (Drywell Pressure - High Functions) have been changed to be consistent with the Allowable Value for the same Function in ITS 3.3.1.1 (Function 6). The new Allowable Values are consistent with the current setpoint calculations. The change affects ITS 3.3.6.1 page 3.3.6.1-5, ITS 3.3.6.2 page 3.3.6.2-4, and the ISTS markup pages 3.3-58 and 3.3-66.

Section 3.4

1. The Bases of ITS 3.4.5 have been modified based on discussions with the NRC. This change affects ITS 3.4.5 pages B 3.4.5-2, B 3.4.5-3, and B 3.4.5-5 and the ISTS Bases markup pages B 3.4-28, B 3.4-29, and B 3.4-32.

Dresden ITS Rev. E Submittal
DISCARD AND INSERT INSTRUCTIONS

VOLUME 3	
SECTION 3.3	
DISCARD	**INSERT**
ITS page 3.3.5.1-10	ITS page 3.3.5.1-10
ITS page 3.3.6.1-5	ITS page 3.3.6.1-5
ITS page 3.3.6.2-4	ITS page 3.3.6.2-4

Dresden ITS Rev. E Submittal
DISCARD AND INSERT INSTRUCTIONS

VOLUME 4	
SECTION 3.3	
DISCARD	**INSERT**
ISTS markup insert page 3.3-44	ISTS markup insert page 3.3-44
ISTS markup page 3.3-58	ISTS markup page 3.3-58
ISTS markup page 3.3-66	ISTS markup page 3.3-66

Dresden ITS Rev. E Submittal
DISCARD AND INSERT INSTRUCTIONS

VOLUME 5	
SECTIONS 3.4 and 3.5	
DISCARD	**INSERT**
ITS pages B 3.4.5-2 through B 3.4.5-5	ITS pages B 3.4.5-2 through B 3.4.5-5
ISTS Bases markup page B 3.4-28	ISTS Bases markup page B 3.4-28
ISTS Bases markup page B 3.4-29	ISTS Bases markup page B 3.4-29
ISTS Bases markup page B 3.4-32	ISTS Bases markup page B 3.4-32

Table 3.3.5.1-1 (page 2 of 5)
Emergency Core Cooling System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER FUNCTION	CONDITIONS REFERENCED FROM REQUIRED ACTION A.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	
2. LPCI System (continued)						
d. Reactor Steam Dome Pressure – Low (Break Detection)	1,2,3	4	B	SR 3.3.5.1.2 SR 3.3.5.1.5 SR 3.3.5.1.6	≥ 802 psig and ≤ 895 psig	(A)
e. Low Pressure Coolant Injection Pump Start – Time Delay Relay Pumps B and C	1,2,3, 4(a), 5(a)	1 per pump	C	SR 3.3.5.1.5 SR 3.3.5.1.6	≤ 8.8 seconds	(B)
f. Low Pressure Coolant Injection Pump Discharge Flow – Low (Bypass)	1,2,3, 4(a), 5(a)	1 per loop	E	SR 3.3.5.1.2 SR 3.3.5.1.3 SR 3.3.5.1.5 SR 3.3.5.1.6	≥ 1107 gpm	(A)
g. Recirculation Pump Differential Pressure–High (Break Detection)	1, 2, 3	4 per pump	C	SR 3.3.5.1.2 SR 3.3.5.1.5 SR 3.3.5.1.6	≤ 5.9 psid	(A) (E)
h. Recirculation Riser Differential Pressure–High (Break Detection)	1, 2, 3	4	C	SR 3.3.5.1.2 SR 3.3.5.1.5 SR 3.3.5.1.6	≤ 2.0 psid	(A)
i. Recirculation Pump Differential Pressure Time Delay - Relay (Break Detection)	1, 2, 3	2	C	SR 3.3.5.1.5 SR 3.3.5.1.6	≤ 0.53 seconds	(B)
j. Reactor Steam Dome Pressure Time Delay - Relay (Break Detection)	1, 2, 3	2	B	SR 3.3.5.1.5 SR 3.3.5.1.6	≤ 2.12 seconds	(B)
k. Recirculation Riser Differential Pressure Time Delay - Relay (Break Detection)	1, 2, 3	2	C	SR 3.3.5.1.5 SR 3.3.5.1.6	≤ 0.53 seconds	(B)

(continued)

(a) When associated ECCS subsystem(s) are required to be OPERABLE per LCO 3.5.2.

Table 3.3.6.1-1 (page 1 of 3)
Primary Containment Isolation Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION C.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Main Steam Line Isolation					
a. Reactor Vessel Water Level – Low Low	1,2,3	2	D	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.6 SR 3.3.6.1.7	≥ -56.77 inches
b. Main Steam Line Pressure – Low	1	2	E	SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.7	≥ 831 psig
c. Main Steam Line Pressure – Timer	1	2	E	SR 3.3.6.1.2 SR 3.3.6.1.6 SR 3.3.6.1.7	≤ 0.280 seconds (Unit 2) ≤ 0.236 seconds (Unit 3)
d. Main Steam Line Flow – High	1,2,3	2 per MSL	D	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.7	≤ 160.5 psid (Unit 2) ≤ 117.1 psid (Unit 3)
e. Main Steam Line Tunnel Temperature – High	1,2,3	2 per trip string	D	SR 3.3.6.1.5 SR 3.3.6.1.6 SR 3.3.6.1.7	≤ 200°F
2. Primary Containment Isolation					
a. Reactor Vessel Water Level – Low	1,2,3	2	G	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.6 SR 3.3.6.1.7	≥ 10.24 inches
b. Drywell Pressure – High	1,2,3	2	G	SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.7	≤ 1.94 psig
c. Drywell Radiation – High	1,2,3	1	F	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.6 SR 3.3.6.1.7	≤ 77 R/hr

(continued)

Table 3.3.6.2-1 (page 1 of 1)
Secondary Containment Isolation Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	
1. Reactor Vessel Water Level - Low	1,2,3, (a)	2	SR 3.3.6.2.1 SR 3.3.6.2.2 SR 3.3.6.2.3 SR 3.3.6.2.5 SR 3.3.6.2.6	$\geq$ 10.24 inches	1(A)
2. Drywell Pressure - High	1,2,3	2	SR 3.3.6.2.2 SR 3.3.6.2.4 SR 3.3.6.2.6	$\leq$ 1.94 psig	1(A) 1(E)
3. Reactor Building Exhaust Radiation - High	1,2,3, (a),(b)	2	SR 3.3.6.2.1 SR 3.3.6.2.2 SR 3.3.6.2.4 SR 3.3.6.2.6	$\leq$ 14.9 mR/hr	1(A)
4. Refueling Floor Radiation - High	1,2,3, (a),(b)	2	SR 3.3.6.2.1 SR 3.3.6.2.2 SR 3.3.6.2.4 SR 3.3.6.2.6	$\leq$ 100 mR/hr	1(A)

(a) During operations with a potential for draining the reactor vessel.

(b) During CORE ALTERATIONS and during movement of irradiated fuel assemblies in secondary containment.

(CTS) [1]

(DOC M.1)

INSERT Functions 2.g, 2.h, 2.i, 2.j, and 2.k

g.	Recirculation Pump Differential Pressure–High (Break Detection)	1, 2, 3	4 per pump	C	SR 3.3.5.1.2 SR 3.3.5.1.5 SR 3.3.5.1.6	≤ 5.9 psid	[A] [E]
h.	Recirculation Riser Differential Pressure–High (Break Detection)	1, 2, 3	4	C	SR 3.3.5.1.2 SR 3.3.5.1.5 SR 3.3.5.1.6	≤ 2.0 psid	[A]
i.	Recirculation Pump Differential Pressure Time Delay - Relay (Break Detection)	1, 2, 3	2	C	SR 3.3.5.1.5 SR 3.3.5.1.6	≤ 0.53 seconds	[A]
j.	Reactor Steam Dome Pressure Time Delay - Relay (Break Detection)	1, 2, 3	2	B	SR 3.3.5.1.5 SR 3.3.5.1.6	≤ 2.12 seconds	[A]
k.	Recirculation Riser Differential Pressure Time Delay - Relay (Break Detection)	1, 2, 3	2	C	SR 3.3.5.1.5 SR 3.3.5.1.6	≤ 0.53 seconds	[A]

〈CTS〉

〈T3.2.A-1〉
〈Doc M.3〉
〈T4.2.A-1〉
〈T3.2.A-1 Footnote(h)〉

Table 3.3.6.1-1 (page 2 of 6)
Primary Containment Isolation Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION C.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
2. Primary Containment Isolation					
a. Reactor Vessel Water Level – Low, Level 3	1,2,3	2	M	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.6 SR 3.3.6.1.7 SR 3.3.6.1.8	≥ 10.24 inches
b. Drywell Pressure – High	1,2,3	2	M	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.6 SR 3.3.6.1.7 SR 3.3.6.1.8	≤ 1.92 psig
c. Drywell Radiation – High	1,2,3	1	F	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.6 SR 3.3.6.1.7	≤ 1.36 R/hr
d. Reactor Building Exhaust Radiation – High	1,2,3	[2]	H	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.6 SR 3.3.6.1.7 SR 3.3.6.1.8	≤ [60] mR/hr
e. Refueling Floor Exhaust Radiation – High	1,2,3	[2]	H	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.6 SR 3.3.6.1.7 SR 3.3.6.1.8	≤ [20] mR/hr
f. Manual Initiation	1,2,3	[1 per group]	G	SR 3.3.6.1.7	NA
3. High Pressure Coolant Injection (HPCI) System Isolation					
a. HPCI Steam Line Flow – High	1,2,3	1	F	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.6 SR 3.3.6.1.7 SR 3.3.6.1.8	≤ 290.16% of rated steam flow (Unit 2) ≤ 288.23% of rated steam flow (Unit 3)

Add SR 3.3.6.1.4

Insert Function 3.b

(continued)

all changes are [4] unless otherwise identified

<CTS>

<T3.2.A-1>
<T4.2.A-1>

Table 3.3.6.2-1 (page 1 of 1)
Secondary Containment Isolation Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Reactor Vessel Water Level – Low [Low, Level 2] [1]	1,2,3, [(a)]	[2]	SR 3.3.6.2.1 SR 3.3.6.2.2 [SR 3.3.6.2.3] SR 3.3.6.2.5 SR 3.3.6.2.6 [5] [SR 3.3.6.2.7]	≥ [177] inches (10.24)
2. Drywell Pressure – High	1,2,3	[2]	[SR 3.3.6.2.1] SR 3.3.6.2.2 [6] [SR 3.3.6.2.3] [SR 3.3.6.2.5] SR 3.3.6.2.6 [SR 3.3.6.2.7] [5]	≤ [1.92] psig (1.94)
3. Reactor Building Exhaust Radiation – High	1,2,3, [(a),(b)]	[2]	SR 3.3.6.2.1 SR 3.3.6.2.2 [SR 3.3.6.2.4] [6] SR 3.3.6.2.6 [SR 3.3.6.2.7] [5]	≤ [100] mR/hr (14.9)
4. Refueling Floor [Exhaust] Radiation – High	1,2,3, [(a),(b)]	[2]	SR 3.3.6.2.1 SR 3.3.6.2.2 [SR 3.3.6.2.4] SR 3.3.6.2.6 [SR 3.3.6.2.7] [5]	≤ [100] mR/hr (100)
5. Manual Initiation	1,2,3, [(a),(b)]	[1 per group]	SR 3.3.6.2.6	NA [2]

6 — SR 3.3.6.2.4

A | A | A | A | E

(a) During operations with a potential for draining the reactor vessel.

(b) During [CORE ALTERATIONS and during] movement of irradiated fuel assemblies in [secondary] containment.

BASES

BACKGROUND (continued)

Two drywell floor drain sump pumps take suction from the drywell floor drain sump and discharge to the Liquid Radioactive Waste Management Systems. The pumps alternate as lead and backup on each successive start. When a high level is reached in the floor drain sump, a level switch actuates to start the lead floor drain sump pump when the pump discharge valves are open. In the event the level continues to rise, a second level switch actuates to start the backup floor drain sump pump and initiates an alarm in the control room. When the level decreases to a low level, both floor drain sump pumps are stopped. A flow transmitter in the discharge line of the drywell floor drain sump pumps provides flow indication in the control room. In addition, a leak rate recorder is provided capable of identifying a 1 gpm change over an 8 hour period. The pumps can also be started from the control room.

The primary containment atmospheric particulate sampling system provides a means to monitor the primary containment atmosphere for airborne particulate radioactivity. An increase of radioactivity may be attributed to RCPB steam or reactor water LEAKAGE. The primary containment atmospheric particulate sampling system is not capable of quantifying LEAKAGE rates. The primary containment atmospheric particulate sampling system consists of a manifold rack that allows drywell atmospheric grab samples to be obtained for analysis and a continuous air monitor that contains particulate and charcoal filters for monitoring of the drywell atmosphere. (D)

APPLICABLE SAFETY ANALYSES

A threat of significant compromise to the RCPB exists if the barrier contains a crack that is large enough to propagate rapidly. LEAKAGE rate limits are set low enough to detect the LEAKAGE emitted from a single crack in the RCPB (Refs. 3 and 4). The drywell floor drain sump monitoring system is designed with the capability of detecting LEAKAGE less than the established LEAKAGE rate limits and providing appropriate alarm of excess LEAKAGE in the control room. (D) The primary containment atmospheric particulate sampling system provides a means to detect changes in LEAKAGE rates (Ref. 5). (D) (E)

A control room alarm provided by the drywell floor drain sump monitoring system allows the operators to evaluate the significance of the indicated LEAKAGE and, if necessary, shut down the reactor for further investigation and (D)

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

corrective action. The allowed LEAKAGE rates are well below the rates predicted for critical crack sizes (Ref. 6). Therefore, these actions provide adequate response before a significant break in the RCPB can occur. (E)

The drywell floor drain sump monitoring system satisfies Criterion 1 of 10 CFR 50.36(c)(2)(ii). The primary containment atmospheric particulate sampling system is maintained to be consistent with NUREG-1433. (D) (E)

LCO

The drywell floor drain sump monitoring system is required to quantify the unidentified LEAKAGE from the RCS. Thus, for the system to be considered OPERABLE, the flow monitoring portion of the system must be OPERABLE. The primary containment atmospheric particulate sampling system is available to the operators so closer examination can be made to determine the extent of any corrective action that may be required. Only one sampling method (either the manifold rack or the continuous air monitor) is required to meet the OPERABILITY requirements. With the leakage detection systems inoperable, monitoring for LEAKAGE in the RCPB is degraded. (D) (D)

APPLICABILITY

In MODES 1, 2, and 3, the leakage detection systems are required to be OPERABLE to support LCO 3.4.4. This Applicability is consistent with that for LCO 3.4.4.

ACTIONS

A.1

With the drywell floor drain sump monitoring system inoperable, no other form of sampling can provide the equivalent information to quantify leakage. However, other monitoring systems are normally available that will provide indication of changes in leakage. (D)

With the drywell floor drain sump monitoring system inoperable, but with RCS unidentified and total LEAKAGE being determined every 12 hours (SR 3.4.4.1), operation may continue for 24 hours. The 24 hour Completion Time of Required Action A.1 is acceptable, based on operating experience, considering the alternative form of leakage detection that is normally available and the fact that the LEAKAGE is still being determined every 12 hours. (D)

(continued)

BASES

ACTIONS (continued)

B.1

With the primary containment atmospheric particulate sampling system inoperable, operation may continue for 24 hours. The 24 hour Completion Time of Required Action B.1 is acceptable, based on operating experience, considering the alternative form of leakage detection that is normally available and the fact that the LEAKAGE is still being determined every 12 hours (SR 3.4.4.1).

D

C.1 and C.2

D

If the Required Action and associated Completion Time of Condition A or B cannot be met, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 12 hours and MODE 4 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to perform the actions in an orderly manner and without challenging plant systems.

D

SURVEILLANCE REQUIREMENTS

SR 3.4.5.1

This SR requires performance of a primary containment atmospheric particulate sample every 12 hours. This is performed by either removing and analyzing the particulate and charcoal filters from the continuous air monitor or by analyzing a grab sample.

D

SR 3.4.5.2

D

This SR is for the performance of a CHANNEL FUNCTIONAL TEST of the drywell floor drain sump monitoring system instrumentation. The test ensures that the system can perform its function in the desired manner. The test also verifies the relative accuracy of the instrument string. A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable CHANNEL FUNCTIONAL TEST of a relay. This is acceptable because all of the other required contacts of the relay are verified by other Technical Specifications and

C

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.4.5.2

non-Technical Specifications tests at least once per refueling interval with applicable extensions. The Frequency of 31 days considers instrument reliability, and operating experience has shown it proper for detecting degradation.

SR 3.4.5.3

This SR is for the performance of a CHANNEL CALIBRATION of the drywell floor drain sump monitoring system instrumentation channel (i.e., drywell floor drain sump pump discharge flow integrator). The calibration verifies the accuracy of the instrument string. The Frequency of SR 3.4.5.3 is based on the assumption of a 12 month calibration interval in the determination of the magnitude of equipment drift in the setpoint analysis.

REFERENCES

1. UFSAR, Section 3.1.2.4.1.

2. Regulatory Guide 1.45, May 1973.

3. GEAP-5620, "Failure Behavior in ASTM A106B Pipes Containing Axial Through-Wall Flaws," April 1968.

4. NUREG-75/067, "Investigation and Evaluation of Cracking in Austenitic Stainless Steel Piping of Boiling Water Reactor Plants," October 1975.

5. UFSAR, Section 5.2.5.2.

6. UFSAR, Section 5.2.5.6.4.

BASES

BACKGROUND (continued)

If the sump fills to the high level setpoint before the timer ends, an alarm sounds in the control room, indicating a LEAKAGE rate into the sump in excess of a preset limit.

Insert BKGD 1

A flow indicator in the discharge line of the drywell floor drain sump pumps provides flow indication in the control room. The pumps can also be started from the control room.

Insert BKGD 2

The primary containment air monitoring system continuously monitor the primary containment atmosphere for airborne particulate and gaseous radioactivity. A sudden increase of radioactivity, which may be attributed to RCPB steam or reactor water LEAKAGE, is annunciated in the control room. The primary containment atmospheric particulate and gaseous radioactivity monitoring systems are not capable of quantifying LEAKAGE rates, but are sensitive enough to indicate increased LEAKAGE rates of 1 gpm within 1 hour. Larger changes in LEAKAGE rates are detected in proportionally shorter times (Ref. 3).

Insert BKD 3

Condensate from four of the six primary containment coolers is routed to the primary containment floor drain sump and is monitored by a flow transmitter that provides indication and alarms in the control room. This primary containment air cooler condensate flow rate monitoring system serves as an added indicator, but not quantifier, of RCS unidentified LEAKAGE.

APPLICABLE SAFETY ANALYSES

A threat of significant compromise to the RCPB exists if the barrier contains a crack that is large enough to propagate rapidly. LEAKAGE rate limits are set low enough to detect the LEAKAGE emitted from a single crack in the RCPB (Refs. 4 and 5). Each of the leakage detection systems inside the drywell is designed with the capability of detecting LEAKAGE less than the established LEAKAGE rate limits and providing appropriate alarm of excess LEAKAGE in the control room.

A control room alarm allows the operators to evaluate the significance of the indicated LEAKAGE and, if necessary, shut down the reactor for further investigation and corrective action. The allowed LEAKAGE rates are well below the rates predicted for critical crack sizes (Ref. 6). Therefore, these actions provide adequate response before a significant break in the RCPB can occur.

The primary containment atmospheric particulate sampling system provides a means to detect changes in LEAKAGE rates (Ref. 5).

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

LCO

APPLICABILITY

ACTIONS

A.1

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.4.6.1

This SR is for the performance of a CHANNEL CHECK of the required primary containment atmospheric monitoring system. The check gives reasonable confidence that the channel is operating properly. The Frequency of 12 hours is based on instrument reliability and is reasonable for detecting off normal conditions.

This is performed either by removing and analyzing the particulate and charcoal filters from the continuous air monitor or by analyzing a grab sample.

SR 3.4.6.2

This SR is for the performance of a CHANNEL FUNCTIONAL TEST of the required RCS leakage detection instrumentation. The test ensures that the monitors can perform their function in the desired manner. The test also verifies the alarm setpoint and relative accuracy of the instrument string. The Frequency of 31 days considers instrument reliability, and operating experience has shown it proper for detecting degradation.

INSERT SR 3.4.5.1 (TSTF-205)

SR 3.4.6.3

This SR is for the performance of a CHANNEL CALIBRATION of required leakage detection instrumentation channels. The calibration verifies the accuracy of the instrument string, including the instruments located inside containment. The Frequency of [18] months is a typical refueling cycle and considers channel reliability. Operating experience has proven this frequency is acceptable.

SR 3.4.5.3 is based on the assumption of a 12 month calibration interval in the determination of the magnitude of equipment drift in the setpoint analysis.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 30. (UFSAR, Section 3.1.2.4.1)
2. Regulatory Guide 1.45, May 1973.
3. FSAR, Section [5.2.7.2.1]. ("Failure Behavior in ASTM A106B Pipes Containing Axial Through-Wall Flaws,")
4. GEAP-5620, April 1968.
5. NUREG-75/067, October 1975. ("Investigation and Evaluation of Cracking in Austenitic Stainless Steel Piping of Boiling Water Reactor Plants.")
6. FSAR, Section [5.2.1.5.2]. (5.2.5.6.4)

5. UFSAR, Section 5.2.5.2.