Nebraska Public Power District
Nebraska's Energy Leader

NLS2000073

August 31, 2000

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

Gentlemen:

Subject: Technical Specification Bases Changes
Cooper Nuclear Station
NRC Docket No. 50-298, DPR-46

In accordance with the requirements of Cooper Nuclear Station (CNS) Technical Specification 5.5.10.d and 10 CFR 50.71(e), enclosed are changes to the Technical Specification Bases implemented without prior Nuclear Regulatory Commission (NRC) approval for the current reporting period.

If you have any questions regarding this submittal, please contact me.

J. H. Swailes
Vice President of Nuclear Energy

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

COMMITMENT	COMMITTED DATE OR OUTAGE
None	

Page No.	Revision No./Date	Page No.	Revision No./Date
i	0	B 3.1-15	6/10/99
ii	0	B 3.1-16	6/10/99
iii	0	B 3.1-17	6/10/99
		B 3.1-18	0
B 2.0-1	0	B 3.1-19	0
B 2.0-2	0	B 3.1-20	0
B 2.0-3	0	B 3.1-21	6/14/00
B 2.0-4	6/10/99	B 3.1-22	0
B 2.0-5	0	B 3.1-23	0
B 2.0-6	0	B 3.1-24	0
B 2.0-7	0	B 3.1-25	0
B 2.0-8	0	B 3.1-26	0
		B 3.1-27	0
B 3.0-1	0	B 3.1-28	0
B 3.0-2	0	B 3.1-29	0
B 3.0-3	0	B 3.1-30	0
B 3.0-4	0	B 3.1-31	0
B 3.0-5	0	B 3.1-32	0
B 3.0-6	0	B 3.1-33	0
B 3.0-7	0	B 3.1-34	0
B 3.0-8	0	B 3.1-35	0
B 3.0-9	0	B 3.1-36	0
B 3.0-10	0	B 3.1-37	0
B 3.0-11	0	B 3.1-38	0
B 3.0-12	3/8/00	B 3.1-39	0
B 3.0-13	0	B 3.1-40	0
B 3.0-14	0	B 3.1-41	0
B 3.0-15	0	B 3.1-42	0
		B 3.1-43	0
B 3.1-1	6/10/99	B 3.1-44	0
B 3.1-2	6/10/99	B 3.1-45	0
B 3.1-3	6/10/99	B 3.1-46	0
B 3.1-4	6/10/99	B 3.1-47	0
B 3.1-5	6/10/99	B 3.1-48	0
B 3.1-6	6/10/99	B 3.1-49	0
B 3.1-7	6/10/99	B 3.1-50	6/10/99
B 3.1-8	6/10/99	B 3.1-51	6/10/99
B 3.1-9	6/10/99		
B 3.1-10	6/10/99	B 3.2-1	0
B 3.1-11	6/10/99	B 3.2-2	6/10/99
B 3.1-12	6/10/99	B 3.2-3	6/10/99
B 3.1-13	0	B 3.2-4	4/12/00
B 3.1-14	6/10/99	B 3.2-5	6/10/99

Page No.	Revision No./Date	Page No.	Revision No./Date
B 3.2-6	0	B 3.3-39	1
B 3.2-7	0	B 3.3-40	0
B 3.2-8	0	B 3.3-41	0
B 3.2-9	0	B 3.3-42	0
B 3.2-10	4//12/00	B 3.3-43	0
		B 3.3-44	0
B 3.3-1	0	B 3.3-45	0
B 3.3-2	0	B 3.3-46	0
B 3.3-3	0	B 3.3-47	0
B 3.3-4	1	B 3.3-48	0
B 3.3-5	1	B 3.3-49	0
B 3.3-6	1	B 3.3-50	0
B 3.3-7	1	B 3.3-51	0
B 3.3-8	1	B 3.3-52	0
B 3.3-9	1	B 3.3-53	0
B 3.3-10	1	B 3.3-54	0
B 3.3-11	1	B 3.3-55	0
B 3.3-12	1	B 3.3-56	1
B 3.3-13	1	B 3.3-57	1
B 3.3-14	1	B 3.3-58	1
B 3.3-15	1	B 3.3-59	0
B 3.3-16	0	B 3.3-60	0
B 3.3-17	0	B 3.3-61	0
B 3.3-18	0	B 3.3-62	0
B 3.3-19	0	B 3.3-63	1
B 3.3-20	0	B 3.3-64	6/10/99
B 3.3-21	0	B 3.3-65	0
B 3.3-22	0	B 3.3-66	1
B 3.3-23	0	B 3.3-67	0
B 3.3-24	0	B 3.3-68	0
B 3.3-25	0	B 3.3-69	0
B 3.3-26	0	B 3.3-70	0
B 3.3-27	0	B 3.3-71	0
B 3.3-28	12/16/98	B 3.3-72	1
B 3.3-29	1	B 3.3-73	1
B 3.3-30	0	B 3.3-74	0
B 3.3-31	1	B 3.3-75	1
B 3.3-32	4/12/00	B 3.3-76	0
B 3.3-33	0	B 3.3-77	0
B 3.3-34	0	B 3.3-78	6/10/99
B 3.3-35	0	B 3.3-79	1
B 3.3-36	0	B 3.3-80	0
B 3.3-37	0	B 3.3-81	0
B 3.3-38	0	B 3.3-82	1

Page No.	Revision No./Date	Page No.	Revision No./Date
B 3.3-83	0	B 3.3-127	0
B 3.3-84	0	B 3.3-128	6/10/99
B 3.3-85	0	B 3.3-129	1
B 3.3-86	0	B 3.3-130	0
B 3.3-87	0	B 3.3-131	0
B 3.3-88	1	B 3.3-132	0
B 3.3-89	0	B 3.3-133	0
B 3.3-90	0	B 3.3-134	0
B 3.3-91	0	B 3.3-135	0
B 3.3-92	0	B 3.3-136	1
B 3.3-93	1	B 3.3-137	0
B 3.3-94	1	B 3.3-138	0
B 3.3-95	0	B 3.3-139	0
B 3.3-96	0	B 3.3-140	0
B 3.3-97	0	B 3.3-141	0
B 3.3-98	0	B 3.3-142	0
B 3.3-99	0	B 3.3-143	0
B 3.3-100	0	B 3.3-144	0
B 3.3-101	0	B 3.3-145	0
B 3.3-102	1	B 3.3-146	0
B 3.3-103	6/10/99	B 3.3-147	0
B 3.3-104	6/10/99	B 3.3-148	1
B 3.3-105	1	B 3.3-149	0
B 3.3-106	0	B 3.3-150	4/12/00
B 3.3-107	0	B 3.3-151	4/12/00
B 3.3-108	0	B 3.3-152	4/12/00
B 3.3-109	0	B 3.3-153	0
B 3.3-110	0	B 3.3-154	0
B 3.3-111	0	B 3.3-155	0
B 3.3-112	0	B 3.3-156	1
B 3.3-113	1	B 3.3-157	6/10/99
B 3.3-114	0	B 3.3-158	0
B 3.3-115	0	B 3.3-159	0
B 3.3-116	0	B 3.3-160	0
B 3.3-117	0	B 3.3-161	0
B 3.3-118	0	B 3.3-162	0
B 3.3-119	0	B 3.3-163	0
B 3.3-120	0	B 3.3-164	1
B 3.3-121	0	B 3.3-165	6/14/00
B 3.3-122	0	B 3.3-166	1
B 3.3-123	0	B 3.3-167	0
B 3.3-124	0	B 3.3-168	0
B 3.3-125	0	B 3.3-169	0
B 3.3-126	0	B 3.3-170	0

Page No.	Revision No./Date	Page No.	Revision No./Date
B 3.3-171	4/19/00	B 3.4-6	0
B 3.3-172	0	B 3.4-7	4/12/00
B 3.3-173	0	B 3.4-8	0
B 3.3-174	0	B 3.4-9	0
B 3.3-175	0	B 3.4-10	0
B 3.3-176	0	B 3.4-11	1
B 3.3-177	0	B 3.4-12	1
B 3.3-178	0	B 3.4-13	4/12/00
B 3.3-179	0	B 3.4-14	0
B 3.3-180	0	B 3.4-15	0
B 3.3-181	0	B 3.4-16	0
B 3.3-182	0	B 3.4-17	0
B 3.3-183	0	B 3.4-18	6/10/99
B 3.3-184	0	B 3.4-19	0
B 3.3-185	0	B 3.4-20	0
B 3.3-186	0	B 3.4-21	0
B 3.3-187	0	B 3.4-22	0
B 3.3-188	0	B 3.4-23	0
B 3.3-189	0	B 3.4-24	0
B 3.3-190	0	B 3.4-25	0
B 3.3-191	0	B 3.4-26	0
B 3.3-192	0	B 3.4-27	0
B 3.3-193	0	B 3.4-28	0
B 3.3-194	0	B 3.4-29	0
B 3.3-195	0	B 3.4-30	0
B 3.3-196	0	B 3.4-31	0
B 3.3-197	0	B 3.4-32	6/14/00
B 3.3-198	0	B 3.4-33	1
B 3.3-199	0	B 3.4-34	0
B 3.3-200	0	B 3.4-35	0
B 3.3-201	0	B 3.4-36	0
B 3.3-202	0	B 3.4-37	0
B 3.3-203	0	B 3.4-38	0
B 3.3-204	0	B 3.4-39	1
B 3.3-205	0	B 3.4-40	0
B 3.3-206	0	B 3.4-41	0
B 3.3-207	0	B 3.4-42	0
B 3.3-208	0	B 3.4-43	0
		B 3.4-44	0
B 3.4-1	0	B 3.4-45	0
B 3.4-2	0	B 3.4-46	0
B 3.4-3	0	B 3.4-47	0
B 3.4-4	0	B 3.4-48	0
B 3.4-5	0	B 3.4-49	0

Page No.	Revision No./Date	Page No.	Revision No./Date
B 3.4-50	0	B 3.6-7	0
B 3.4-51	0	B 3.6-8	0
B 3.4-52	0	B 3.6-9	0
B 3.4-53	0	B 3.6-10	0
B 3.4-54	0	B 3.6-11	0
B 3.4-55	0	B 3.6-12	3/8/00
		B 3.6-13	4/12/00
B 3.5-1	1	B 3.6-14	3/8/00
B 3.5-2	1	B 3.6-15	0
B 3.5-3	0	B 3.6-16	1
B 3.5-4	0	B 3.6-17	0
B 3.5-5	1	B 3.6-18	0
B 3.5-6	0	B 3.6-19	0
B 3.5-7	0	B 3.6-20	3/8/00
B 3.5-8	0	B 3.6-21	3/8/00
B 3.5-9	1	B 3.6-22	3/8/00
B 3.5-10	0	B 3.6-23	0
B 3.5-11	0	B 3.6-24	1
B 3.5-12	0	B 3.6-25	1
B 3.5-13	4/19/00	B 3.6-26	3/8/00
B 3.5-14	4/19/00	B 3.6-27	0
B 3.5-15	4/19/00	B 3.6-28	3/8/00
B 3.5-16	0	B 3.6-29	4/12/00
B 3.5-17	11/23/99	B 3.6-30	0
B 3.5-18	0	B 3.6-31	6/14/00
B 3.5-19	0	B 3.6-32	4/12/00
B 3.5-20	0	B 3.6-33	0
B 3.5-21	0	B 3.6-34	4/12/00
B 3.5-22	0	B 3.6-35	0
B 3.5-23	0	B 3.6-36	0
B 3.5-24	0	B 3.6-37	0
B 3.5-25	1	B 3.6-38	0
B 3.5-26	0	B 3.6-39	0
B 3.5-27	0	B 3.6-40	0
B 3.5-28	4/19/00	B 3.6-41	0
B 3.5-29	4/19/00	B 3.6-42	0
B 3.5-30	4/19/00	B 3.6-43	0
		B 3.6-44	0
B 3.6-1	3/8/00	B 3.6-45	0
B 3.6-2	3/8/00	B 3.6-46	6/10/99
B 3.6-3	3/8/00	B 3.6-47	0
B 3.6-4	3/8/00	B 3.6-48	0
B 3.6-5	3/8/00	B 3.6-49	0
B 3.6-6	0	B 3.6-50	6/10/99

Page No.	Revision No./Date	Page No.	Revision No./Date
B 3.6-51	0	B 3.7-9	0
B 3.6-52	0	B 3.7-10	1
B 3.6-53	0	B 3.7-11	0
B 3.6-54	0	B 3.7-12	1
B 3.6-55	0	B 3.7-13	0
B 3.6-56	0	B 3.7-14	0
B 3.6-57	0	B 3.7-15	0
B 3.6-58	0	B 3.7-16	10/13/99
B 3.6-59	0	B 3.7-17	0
B 3.6-60	0	B 3.7-18	0
B 3.6-61	0	B 3.7-19	0
B 3.6-62	0	B 3.7-20	0
B 3.6-63	0	B 3.7-21	0
B 3.6-64	0	B 3.7-22	0
B 3.6-65	0	B 3.7-23	0
B 3.6-66	0	B 3.7-24	0
B 3.6-67	0	B 3.7-25	0
B 3.6-68	0	B 3.7-26	0
B 3.6-69	0	B 3.7-27	0
B 3.6-70	1	B 3.7-28	0
B 3.6-71	0	B 3.7-29	0
B 3.6-72	0	B 3.7-30	1
B 3.6-73	3/8/00	B 3.7-31	1
B 3.6-74	0		
B 3.6-75	3/8/00	B 3.8-1	0
B 3.6-76	3/8/00	B 3.8-2	0
B 3.6-77	3/8/00	B 3.8-3	1
B 3.6-78	0	B 3.8-4	0
B 3.6-79	0	B 3.8-5	1
B 3.6-80	0	B 3.8-6	1
B 3.6-81	0	B 3.8-7	1
B 3.6-82	0	B 3.8-8	1
B 3.6-83	0	B 3.8-9	1
B 3.6-84	0	B 3.8-10	1
B 3.6-85	0	B 3.8-11	1
		B 3.8-12	1
B 3.7-1	1	B 3.8-13	1
B 3.7-2	0	B 3.8-14	1
B 3.7-3	0	B 3.8-15	1
B 3.7-4	0	B 3.8-16	1
B 3.7-5	0	B 3.8-17	0
B 3.7-6	0	B 3.8-18	0
B 3.7-7	0	B 3.8-19	0
B 3.7-8	1	B 3.8-20	0

Page No.	Revision No./Date	Page No.	Revision No./Date
B 3.8-21	0	B 3.8-65	0
B 3.8-22	0	B 3.8-66	0
B 3.8-23	0	B 3.8-67	0
B 3.8-24	0	B 3.8-68	0
B 3.8-25	0	B 3.8-69	0
B 3.8-26	0	B 3.8-70	0
B 3.8-27	0	B 3.8-71	0
B 3.8-28	0	B 3.8-72	0
B 3.8-29	0	B 3.8-73	0
B 3.8-30	0	B 3.8-74	0
B 3.8-31	0	B 3.8-75	0
B 3.8-32	0		
B 3.8-33	0	B 3.9-1	0
B 3.8-34	0	B 3.9-2	0
B 3.8-35	0	B 3.9-3	0
B 3.8-36	0	B 3.9-4	0
B 3.8-37	1	B 3.9-5	0
B 3.8-38	0	B 3.9-6	0
B 3.8-39	0	B 3.9-7	0
B 3.8-40	1	B 3.9-8	0
B 3.8-41	0	B 3.9-9	0
B 3.8-42	0	B 3.9-10	0
B 3.8-43	0	B 3.9-11	0
B 3.8-44	0	B 3.9-12	0
B 3.8-45	0	B 3.9-13	0
B 3.8-46	0	B 3.9-14	0
B 3.8-47	0	B 3.9-15	0
B 3.8-48	0	B 3.9-16	0
B 3.8-49	0	B 3.9-17	0
B 3.8-50	0	B 3.9-18	0
B 3.8-51	0	B 3.9-19	0
B 3.8-52	0	B 3.9-20	0
B 3.8-53	0	B 3.9-21	0
B 3.8-54	0	B 3.9-22	0
B 3.8-55	0	B 3.9-23	0
B 3.8-56	0	B 3.9-24	0
B 3.8-57	0	B 3.9-25	0
B 3.8-58	0	B 3.9-26	0
B 3.8-59	0	B 3.9-27	0
B 3.8-60	0	B 3.9-28	0
B 3.8-61	0	B 3.9-29	0
B 3.8-62	0	B 3.9-30	0
B 3.8-63	0		
B 3.8-64	0	B 3.10-1	0

Page No.	Revision No./Date	Page No.	Revision No./Date
B 3.10-2	0		
B 3.10-3	0		
B 3.10-4	0		
B 3.10-5	0		
B 3.10-6	0		
B 3.10-7	0		
B 3.10-8	0		
B 3.10-9	0		
B 3.10-10	0		
B 3.10-11	0		
B 3.10-12	0		
B 3.10-13	0		
B 3.10-14	0		
B 3.10-15	0		
B 3.10-16	0		
B 3.10-17	0		
B 3.10-18	0		
B 3.10-19	0		
B 3.10-20	0		
B 3.10-21	0		
B 3.10-22	0		
B 3.10-23	0		
B 3.10-24	0		
B 3.10-25	0		
B 3.10-26	6/10/99		
B 3.10-27	6/10/99		
B 3.10-28	0		
B 3.10-29	6/10/99		
B 3.10-30	0		
B 3.10-31	0		
B 3.10-32	0		
B 3.10-33	0		
B 3.10-34	0		
B 3.10-35	0		
B 3.10-36	0		
B 3.10-37	0		
B 3.10-38	0		
B 3.10-39	0		

BASES

SR 3.0.2 (continued)

The 25% extension does not significantly degrade the reliability that results from performing the Surveillance at its specified Frequency. This is based on the recognition that the most probable result of any particular Surveillance being performed is the verification of conformance with the SRs. The exceptions to SR 3.0.2 are those Surveillances for which the 25% extension of the interval specified in the Frequency does not apply. These exceptions are stated in the individual Specifications. The requirements of regulations take precedence over the TS. An example of where SR 3.0.2 does not apply is in the Containment Leakage Rate Testing Program. This program establishes testing requirements and Frequencies in accordance with the requirements of the regulations. The TS cannot in and of themselves extend a test interval specified in the regulations.

As stated in SR 3.0.2, the 25% extension also does not apply to the initial portion of a periodic Completion Time that requires performance on a "once per..." basis. The 25% extension applies to each performance after the initial performance. The initial performance of the Required Action, whether it is a particular Surveillance or some other remedial action, is considered a single action with a single Completion Time. One reason for not allowing the 25% extension to this Completion Time is that such an action usually verifies that no loss of function has occurred by checking the status of redundant or diverse components or accomplishes the function of the inoperable equipment in an alternative manner.

The provisions of SR 3.0.2 are not intended to be used repeatedly merely as an operational convenience to extend Surveillance intervals (other than those consistent with refueling intervals) or periodic Completion Time intervals beyond those specified.

SR 3.0.3

SR 3.0.3 establishes the flexibility to defer declaring affected equipment inoperable or an affected variable outside the specified limits when a Surveillance has not been completed within the specified Frequency. A delay period of up to 24 hours or up to the limit of the specified Frequency, whichever is less, applies from the point in time that it is discovered that the Surveillance has not been

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.1.3.5

Coupling verification is performed to ensure the control rod is connected to the CRDM and will perform its intended function when necessary. The Surveillance requires verifying a control rod does not go to the withdrawn overtravel position. The overtravel position feature provides a positive check on the coupling integrity since only an uncoupled CRD can reach the overtravel position. The verification is required to be performed any time a control rod is withdrawn to the "full out" position (notch position 48) or prior to declaring the control rod OPERABLE after work on the control rod or CRD System that could affect coupling. This includes control rods inserted one notch and then returned to the "full out" position during the performance of SR 3.1.3.2. This Frequency is acceptable, considering the low probability that a control rod will become uncoupled when it is not being moved and operating experience related to uncoupling events.

REFERENCES

1. USAR, Appendix F.
2. USAR, Section XIV-5.3.1
3. USAR, Section XIV-6.2.
4. USAR, Appendix G.
5. 10 CFR 50.36(c)(2)(ii).
6. NEDO-21231, "Banked Position Withdrawal Sequence," Section 7.2, January 1977.

BASES

ACTIONS

B.1 (continued)

be brought to a MODE or other specified condition in which the LCO does not apply. To achieve this status, THERMAL POWER must be reduced to < 25% RTP within 4 hours. The allowed Completion Time is reasonable, based on operating experience, to reduce THERMAL POWER to < 25% RTP in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.2.1.1

APLHGRs are required to be initially calculated within 12 hours after THERMAL POWER is ≥ 25% RTP and then every 24 hours thereafter. They are compared to the specified limits in the COLR to ensure that the reactor is operating within the assumptions of the safety analysis. The 24 hour Frequency is based on both engineering judgment and recognition of the slowness of changes in power distribution during normal operation. The 12 hour allowance after THERMAL POWER ≥ 25% RTP is achieved is acceptable given the large inherent margin to operating limits at low power levels.

REFERENCES

1. NEDE-24011-P-A, "General Electric Standard Application for Reactor Fuel," (Revision specified in the COLR).

2. USAR, Section III.

3. USAR, Section VI.

4. USAR, Section XIV.

5. NEDO-24258, "Cooper Nuclear Station Single-Loop Operation," May 1980.

6. NEDC-32914P, "Maximum Extended Load Line Limit and Increased Core Flow for Cooper Nuclear Station," Revision 0, January 2000.

7. NEDO-30130-A, "Steady State Nuclear Methods," May 1985.

(continued)

BASES

REFERENCES (continued)

6. NEDO-24258, "Cooper Nuclear Station Single-Loop Operation," May 1980.

7. NEDC-32914P, "Maximum Extended Load Line Limit and Increased Core Flow for Cooper Nuclear Station," Revision 0, January 2000.

8. NEDO-30130-A, "Steady State Nuclear Methods," May 1985.

9. NEDO-24154, "Qualification of the One-Dimensional Core Transient Model for Boiling Water Reactors," October 1978.

10. 10 CFR 50.36(c)(2)(ii).

BASES

SURVEILLANCE REQUIREMENTS (continued)

that all components are tested. The RPS RESPONSE TIME acceptance criteria are included in Reference 12.

As noted, neutron detectors are excluded from RPS RESPONSE TIME testing because the principles of detector operation virtually ensure an instantaneous response time.

The 18 month Frequency is consistent with the typical industry refueling cycle and is based upon plant operating experience, which shows that random failures of instrumentation components causing serious response time degradation, but not channel failure, are infrequent occurrences.

REFERENCES

1. USAR, Section VII-2.
2. USAR, Chapter XIV.
3. NEDO-23842, "Continuous Control Rod Withdrawal in the Startup Range," April 18, 1978.
4. USAR, Section VI-5.
5. 10 CFR 50.36(c)(2)(ii).
6. USAR, Section IV-4.9.
7. USAR, Section XIV-6.2.
8. USAR, Section XIV-5.4.3.
9. P. Check (NRC) letter to G. Lainas (NRC), "BWR Scram Discharge System Safety Evaluation," December 1, 1980.
10. NEDO-30851-P-A , "Technical Specification Improvement Analyses for BWR Reactor Protection System," March 1988.
11. MDE-94-0485, "Technical Specification Improvement Analysis for the Reactor Protection System for Cooper Nuclear Station," April 1985.
12. USAR, VII-2.3.9.10.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

2.d. Main Steam Line Radiation - High

The Main Steam Line Radiation - High isolation signal has been removed from the MSIVs (Ref. 7); however, this isolation Function has been retained for other valves (e.g., recirculation sample valves and mechanical vacuum pump inlet and outlet valves) to ensure that the assumptions utilized to determine that acceptable offsite doses resulting from a control rod drop accident are maintained.

Main Steam Line Radiation - High signals are generated from four radiation elements and associated monitors, each of which is located near one of the main steam lines in the steam tunnel. Four instrumentation channels of the Main Steam Line Radiation - High Function are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function. The Allowable Value was selected to be low enough that a high radiation trip results from the fission products released in the design basis control rod drop accident (which occurs at a low steam flow condition) and high enough above the background radiation level in the vicinity of the main steam lines so that spurious trips are avoided at rated power.

The allowable value is stated in terms of a multiple of "normal full power background." With the injection of hydrogen into the RCS from Optimum Water Chemistry System, the background radiation levels seen by the radiation monitors will be greater than during periods of no hydrogen injection. The allowable value is fixed at the normal full power background associated with hydrogen injection at the nominal rate. The allowable value is not adjusted during periods when hydrogen injection is not in service. This is acceptable since the setpoint will continue to provide automatic actuation protection in the event of a DBA Control Rod Drop Accident (CRDA). Even during periods of no hydrogen injection, the expected radiation levels in the event of a DBA-CRDA will exceed the higher allowable value that is based on full power with hydrogen injection.

This Function isolates the recirculation sample valves, trips the mechanical vacuum pumps, and closes the inlet and outlet valves to the mechanical vacuum pumps.

2.e. Reactor Vessel Water Level - Low Low Low (Level 1)

Low reactor water level indicates that the capability to cool the fuel may be threatened. Should the water level decrease too far, fuel damage could result. Therefore, isolation of the recirculation sample line valves

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

from the reactor vessel occurs as part of the isolation of Primary Containment to prevent offsite dose limits from being exceeded. The Reactor Vessel Water Level - Low Low Low (Level 1) Function is one of the many Functions assumed to be OPERABLE and capable of providing isolation signals. This Function, associated with Primary Containment isolation, is assumed in the analysis of the recirculation line break (Ref. 4). The isolation of the valves of the recirculation sample line on Level 1 supports actions to ensure that offsite dose limits are not exceeded for a DBA.

Reactor vessel water level signals are initiated from four level transmitters that sense the difference between the pressure due to a constant column of water (reference leg) and the pressure due to the actual water level (variable leg) in the vessel. Four channels of Reactor Vessel Water Level - Low Low Low (Level 1) Function are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function.

The Reactor Vessel Water Level - Low Low Low (Level 1) Allowable Value is chosen to be the same as the ECCS Level 1 Allowable Value (LCO 3.3.5.1) to ensure that the recirculation sample valves will isolate on a potential LOCA to prevent offsite doses from exceeding 10 CFR 100 limits.

This Function isolates the recirculation sample valves. It may be bypassed using a key-locked switch during accident conditions to obtain a sample for Post Accident Sampling System (PASS).

<u>High Pressure Coolant Injection and Reactor Core Isolation Cooling Systems Isolation</u>

<u>3.a., 3.b., 4.a., 4.b. HPCI and RCIC Steam Line Flow - High and Time Delay Relays</u>

Steam Line Flow — High Functions are provided to detect a break of the RCIC or HPCI steam lines and initiate closure of the steam line isolation valves of the appropriate system. If the steam is allowed to continue flowing out of the break, the reactor will depressurize and the core can uncover. Therefore, the isolations are initiated on high flow to prevent or minimize core damage. The isolation action, along with the scram function of the RPS, ensures that the fuel peak cladding temperature remains below the limits of 10 CFR 50.46. Specific credit for these Functions is not assumed in any USAR accident analyses since the bounding analysis is performed for large breaks such as recirculation and

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

MSL breaks. However, these instruments prevent the RCIC or HPCI steam line breaks from becoming bounding.

The HPCI and RCIC Steam Line Flow — High signals are initiated from differential pressure switches (two for HPCI and two for RCIC) that are connected to the system steam lines. A time delay is provided to prevent HPCI or RCIC isolation due to high flow transients during HPCI or RCIC startup with one Time Delay Relay channel associated with each Steam Line Flow-High channel. Two channels of both HPCI and RCIC Steam Line Flow — High Functions and the associated Time Delay Relays are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function.

The Allowable Values for the Steam Line Flow-High Function and associated Time Delay Relay Function are chosen to be low enough to ensure that the trip occurs to prevent fuel damage and maintains the MSLB event as the bounding event.

These Functions isolate the Group 4 and 5 valves, as appropriate, as listed in Reference 1.

3.c., 4.c. HPCI and RCIC Steam Supply Line Pressure - Low

Low MSL pressure indicates that the pressure of the steam in the HPCI or RCIC turbine may be too low to continue operation of the associated system's turbine. These isolations are for equipment protection and are not assumed in any transient or accident analysis in the USAR. However, they also provide a diverse signal to indicate a possible system break. These instruments are included in Technical Specifications (TS) because of the potential for risk due to possible failure of the instruments preventing HPCI and RCIC initiations (Ref. 8).

The HPCI and RCIC Steam Supply Line Pressure — Low signals are initiated from pressure switches (four for HPCI and four for RCIC) that are connected to the system steam line. Four channels of both HPCI and RCIC Steam Supply Line Pressure — Low Functions are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function.

BASES

SURVEILLANCE REQUIREMENTS (continued)

of the channel (SR 3.3.6.1.4) are performed using a standard current source.

The Frequency of SR 3.3.6.1.3 is based on the assumption of a 92 day calibration interval in the determination of the magnitude of equipment drift in the setpoint analysis. The Frequency of SR 3.3.6.1.4 and SR 3.3.6.1.5 is based on the assumption of an 18 month calibration interval in the determination of the magnitude of equipment drift in the setpoint analysis.

SR 3.3.6.1.6

The LOGIC SYSTEM FUNCTIONAL TEST demonstrates the OPERABILITY of the required isolation logic for a specific channel. The system functional testing performed on PCIVs in LCO 3.6.1.3 overlaps this Surveillance to provide complete testing of the assumed safety function. Simulated automatic actuation is performed each operating cycle. The 18 month Frequency is based on the need to perform the Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown these components usually pass the Surveillance when performed at the 18 month Frequency.

REFERENCES

1. USAR, Table V-2-2.
2. USAR, Chapter XIV.
3. 10 CFR 50.36(c)(2)(ii).
4. USAR, Section XIV-6.3.
5. USAR, Section XIV-5.4.1.
6. USAR, Section XIV-6.5.
7. USAR, Section XIV-6.7.1.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

High drywell pressure signals are initiated from pressure switches that sense the pressure in the drywell. Eight channels of Drywell Pressure — High Functions are available and are required to be OPERABLE to ensure that no single instrument failure can preclude performance of the isolation function.

The Allowable Value was chosen to be the same as the ECCS Drywell Pressure — High Function Allowable Value (LCO 3.3.5.1) since this is indicative of a loss of coolant accident (LOCA).

The Drywell Pressure — High Function is required to be OPERABLE in MODES 1, 2, and 3 where considerable energy exists in the RCS; thus, there is a probability of pipe breaks resulting in significant releases of radioactive steam and gas. This Function is not required in MODES 4 and 5 because the probability and consequences of these events are low due to the RCS pressure and temperature limitations of these MODES.

3. Reactor Building Ventilation Exhaust Plenum Radiation - High

High secondary containment exhaust radiation is an indication of possible gross failure of the fuel cladding. The release may have originated from the primary containment due to a break in the RCPB or the refueling floor due to a fuel handling accident during refueling. When Reactor Building Exhaust Plenum Radiation — High is detected, secondary containment isolation and actuation of the SGT System are initiated to limit the release of fission products as assumed in the USAR safety analyses (Ref. 4).

The Reactor Building Exhaust Plenum Radiation — High signals are initiated from four radiation detectors that are located such that they can monitor the radioactivity of gas flowing through the reactor building exhaust plenum. The signal from each detector is input to an individual monitor whose trip outputs are assigned to an isolation channel in each trip system. Four channels of Reactor Building Ventilation Exhaust Plenum Radiation — High Function are available and are required to be OPERABLE to ensure that no single

BASES

SURVEILLANCE REQUIREMENTS (continued)

core flow. The recirculation loop flow, as used in this Surveillance, can be either the summation of the flows from all of the jet pumps associated with a single recirculation loop, or the drive flow associated with each recirculation loop.

The mismatch is measured in terms of percent of rated core flow. If the flow mismatch exceeds the specified limits, the loop with the lower flow is considered inoperable. The SR is not required when both loops are not in operation since the mismatch limits are meaningless during single loop or natural circulation operation. The Surveillance must be performed within 24 hours after both loops are in operation. The 24 hour Frequency is consistent with the Surveillance Frequency for jet pump OPERABILITY verification and has been shown by operating experience to be adequate to detect off normal loop flows in a timely manner.

SR 3.4.1.2

This SR ensures the core flow, as a function of core THERMAL POWER, is within the appropriate limits to prevent uncontrolled thermal hydraulic oscillations. At low flows and high power levels the reactor exhibits increased susceptibility to thermal hydraulic instability. The power/flow map is based on the guidance provided in Reference 5. The 24 hour Frequency is based on operating experience and the operator's inherent knowledge of reactor status, including significant changes in core THERMAL POWER and flow.

REFERENCES

1. NEDE-24011-P-A (Revision specified in the COLR).

2. USAR, Section IV-3.6.

3. NEDE-24258, "Cooper Nuclear Station Single-Loop Operation," May, 1980.

4. 10 CFR 50.36(c)(2)(ii).

BASES

REFERENCES

1. USAR, Section IV-3.6.

2. 10 CFR 50.36(c)(2)(ii).

3. GE Service Information Letter No. 330, including Supplement 1, "Jet Pump Beam Cracks," June 9, 1980.

4. NUREG/CR-3052, "Closeout of IE Bulletin 80-07: BWR Jet Pump Assembly Failure," November 1984.

BASES

ACTIONS

B.1, B.2.1, B.2.2.1, and B.2.2.2 (continued)

The Completion Time of once every 4 hours is the time needed to take and analyze a sample. The 12 hour Completion Time is reasonable, based on operating experience, to isolate the main steam lines in an orderly manner and without challenging plant systems. Also, the allowed Completion Times for Required Actions B.2.2.1 and B.2.2.2 for placing the unit in MODES 3 and 4 are reasonable, based on operating experience, to achieve the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.4.6.1

This Surveillance is performed to ensure iodine remains within limit during normal operation. The 7 day Frequency is adequate to trend changes in the iodine activity level.

This SR is modified by a Note that requires this Surveillance to be performed only in MODE 1 because the level of fission products generated in other MODES is much less.

REFERENCES

1. 10 CFR 100.11, 1973.
2. USAR, Section XIV-8.1.
3. NEDC 94-157, Rev. 1, Control Room Dose due to Main Steam Line Break, January 1, 1995.
4. 10 CFR 50, Appendix A, GDC 19.
5. 10 CFR 50.36(c)(2)(ii).

BASES

SURVEILLANCE REQUIREMENTS (continued)

should overcome the RPV pressure and associated discharge line losses. Adequate reactor 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. Adequate reactor steam pressure must be $\geq$ 920 psig to perform SR 3.5.1.7 and $\geq$ 145 psig to perform SR 3.5.1.8. Adequate steam flow is represented by turbine bypass valves at least 30% open, or total steam flow $\geq 10^6$ lb/hr. Reactor startup is allowed prior to performing the low pressure Surveillance test because the reactor pressure is low and the time allowed to satisfactorily perform the Surveillance test is short. The reactor pressure is allowed to be increased to normal operating pressure since it is assumed that the low pressure test has been satisfactorily completed and there is no indication or reason to believe that HPCI is inoperable.

Therefore, SR 3.5.1.7 and SR 3.5.1.8 are modified by Notes that state the Surveillances are not required to be performed until 12 hours after the reactor steam pressure and flow are adequate to perform the test. The 12 hours allowed for the flow tests after required pressure and flow are reached are sufficient to achieve stable conditions for testing and provides a reasonable time to complete the SRs. For SR 3.5.1.8, while adequate pressure can be reached prior to the required Applicability for HPCI, the 12 hour allowance of the Note would not apply until entering the Applicability (>150 psig) with adequate steam flow.

The Frequency for SR 3.5.1.6 and SR 3.5.1.7 is in accordance with the Inservice Testing Program requirements. The 18 month Frequency for SR 3.5.1.8 is based on the need to perform the Surveillance under the conditions that apply just prior to or during a startup from a plant outage. Operating experience has shown that these components usually pass the SR when performed at the 18 month Frequency, which is based on the refueling cycle. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.5.1.9

The ECCS subsystems are required to actuate automatically to perform their design functions. This Surveillance verifies that, with a required system initiation signal (actual or simulated), the automatic initiation logic of HPCI, CS, and LPCI will cause the systems or subsystems to operate as designed, including actuation of the system throughout its emergency operating sequence, automatic pump startup and actuation of all automatic valves to their required positions. This SR also ensures that the HPCI System will automatically restart on an RPV low water level (Level 2) signal received subsequent to an RPV high water level (Level 8) trip and that the suction is automatically transferred from the ECSTs to the suppression pool. The LOGIC SYSTEM FUNCTIONAL TEST performed in LCO 3.3.5.1 overlaps this Surveillance to provide complete testing of the assumed safety function.

The 18 month Frequency is based on the need to perform the Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power.

Operating experience has shown that these components usually pass the SR when performed at the 18 month Frequency, which is based on the refueling cycle. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

This SR is modified by Note 1 that says for HPCI only the Surveillance is not required to be performed until 12 hours after the reactor steam pressure and flow are adequate to perform the test. The time allowed for this test after required pressure and flow are reached is sufficient to achieve stable conditions for testing and provides a reasonable time to complete the SR. Adequate reactor pressure must be available to perform this test. 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. Thus, sufficient time is allowed after adequate pressure and flow are achieved to perform this test. Adequate reactor steam pressure is > 145 psig. Adequate steam flow is represented by turbine bypass valves at least

BASES

SURVEILLANCE REQUIREMENTS (continued)

30% open, or a total steam flow of 10^6 lb/hr. Reactor startup is allowed prior to performing this test because the reactor pressure is low and the time allowed to satisfactorily perform the test is short. For SR 3.5.1.9, while adequate pressure can be reached prior to the required Applicability for HPCI, the 12 hour allowance of the Note would not apply until entering the Applicability (>150 psig) with adequate steam flow.

This SR is modified by Note 2 that excludes vessel injection/spray during the Surveillance. Since all active components are testable and full flow can be demonstrated by recirculation through the test line, coolant injection into the RPV is not required during the Surveillance.

SR 3.5.1.10

The ADS designated SRVs are required to actuate automatically upon receipt of specific initiation signals. A system functional test is performed to demonstrate that the mechanical portions of the ADS function (i.e., solenoids) operate as designed when initiated either by an actual or simulated initiation signal, causing proper actuation of all the required components. SR 3.5.1.11 and the LOGIC SYSTEM FUNCTIONAL TEST performed in LCO 3.3.5.1 overlap this Surveillance to provide complete testing of the assumed safety function.

The 18 month Frequency is based on the need to perform the Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass the SR when performed at the 18 month Frequency, which is based on the refueling cycle. 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 3.5.1.11. This also prevents an RPV pressure blowdown.

BASES

REFERENCES (continued)

3. USAR, Section VI-4.1.

4. USAR, Section VI-4.2.

5. USAR, Section XIV-6.

6. USAR, Appendix G.

7. 10 CFR 50, Appendix K.

8. 10 CFR 50.46.

9. NEDC-32687P, Revision 1, "Cooper Nuclear Station SAFER/GESTR-LOCA Loss-of-Coolant Accident Analysis," March 1997.

10. 10 CFR 50.36(c)(2)(ii).

11. Memorandum from R.L. Baer (NRC) to V. Stello, Jr. (NRC), "Recommended Interim Revisions to LCOs for ECCS Components," December 1, 1975.

12. NEDE-24956, "BWR ADS Pneumatic System Comparison to NUREG-0737 Requirement II.K.3.2.8."

BASES

SURVEILLANCE REQUIREMENTS (continued)

RCIC System, this SR also includes the steam flow path for the turbine and the flow controller position.

The 31 day Frequency of this SR was derived from the Inservice Testing Program requirements for performing valve testing at least once every 92 days. The Frequency of 31 days is further justified because the valves are operated under procedural control and because improper valve position would affect only the RCIC System. This Frequency has been shown to be acceptable through operating experience.

SR 3.5.3.3 and SR 3.5.3.4

The RCIC pump flow rates ensure that the system can maintain reactor coolant inventory during pressurized conditions with the RPV isolated. The flow tests for the RCIC 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 both at 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 RCIC System diverts steam flow. Therefore, sufficient time is allowed after adequate pressure and flow are achieved to perform these SRs. Adequate reactor steam pressure to perform SR 3.5.3.3 is 920 psig and 145 psig to perform SR 3.5.3.4. Adequate steam flow is represented by turbine bypass valves at least 30% open, or total steam flow $\geq 10^6$ lb/hr. Reactor startup is allowed prior to performing the low pressure Surveillance because the reactor pressure is low and the time allowed to satisfactorily perform the Surveillance is short. The reactor pressure is allowed to be increased to normal operating pressure since it is assumed that the low pressure Surveillance has been satisfactorily completed and there is no indication or reason to believe that RCIC is inoperable. Therefore, these SRs are modified by Notes that state the Surveillances are not required to be performed until 12 hours after the reactor steam pressure and flow are

BASES

SURVEILLANCE REQUIREMENTS (continued)

adequate to perform the test. The 12 hours allowed for the flow tests after the required pressure and flow are reached are sufficient to achieve stable conditions for testing and provides a reasonable time to complete the SRs. For SR 3.5.3.4, while adequate pressure can be reached prior to the required Applicability for RCIC, the 12 hour allowance of the Note would not apply until entering the Applicability (>150 psig) with adequate steam flow.

A 92 day Frequency for SR 3.5.3.3 is consistent with the Inservice Testing Program requirements. The 18 month Frequency for SR 3.5.3.4 is based on the need to perform the Surveillance under conditions that apply just prior to or during a startup from a plant outage. Operating experience has shown that these components usually pass the SR when performed at the 18 month Frequency, which is based on the refueling cycle. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

SR 3.5.3.5

The RCIC System is required to actuate automatically in order to verify its design function satisfactorily. This Surveillance verifies that, with a required system initiation signal (actual or simulated), the automatic initiation logic of the RCIC System will cause the system to operate as designed, including actuation of the system throughout its emergency operating sequence; that is, automatic pump startup and actuation of all automatic valves to their required positions. This test also ensures the RCIC System will automatically restart on an RPV low water level (Level 2) signal received subsequent to an RPV high water level (Level 8) trip and that the suction is automatically transferred from the ECST to the suppression pool. The LOGIC SYSTEM FUNCTIONAL TEST performed in LCO 3.3.5.2 overlaps this Surveillance to provide complete testing of the assumed design function.

The 18 month Frequency is based on the need to perform the Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass the SR when performed at the 18 month Frequency, which is based on the refueling cycle. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

BASES

SURVEILLANCE REQUIREMENTS (continued)

This SR is modified by Note 1 that says the Surveillance is not required to be performed until 12 hours after the reactor steam pressure and flow are adequate to perform the test. The time allowed for this test after required pressure and flow are reached is sufficient to achieve stable conditions for testing and provides a reasonable time to complete the SR. Adequate reactor pressure must be available to perform this test. Additionally, adequate steam flow must be passing through the main turbine or turbine bypass valves to continue to control reactor pressure when the RCIC System diverts steam flow. Thus, sufficient time is allowed after adequate pressure and flow are achieved to perform this test. Adequate reactor steam pressure is 145 psig. Adequate steam flow is represented by turbine bypass valves at least 30% open, or a total steam flow of 10^6 lb/hr. Reactor startup is allowed prior to performing this test because the reactor pressure is low and the time allowed to satisfactorily perform the test is short. For SR 3.5.3.3, while adequate pressure can be reached prior to the required Applicability for RCIC, the 12 hour allowance of the Note would not apply until entering the Applicability (>150 psig) with adequate steam flow.

This SR is modified by Note 2 that excludes vessel injection during the Surveillance. Since all active components are testable and full flow can be demonstrated by recirculation through the test line, coolant injection into the RPV is not required during the Surveillance.

REFERENCES

1. USAR, Appendix F.
2. USAR, Section IV-7.
3. 10 CFR 50.36(c)(2)(ii).
4. Memorandum from R.L. Baer (NRC) to V. Stello, Jr. (NRC), "Recommended Interim Revisions to LCOs for ECCS Components," December 1, 1975.

B 3.6 CONTAINMENT SYSTEMS

B 3.6.1.1 Primary Containment

BASES

BACKGROUND

The function of the primary containment is to isolate and contain fission products released from the Reactor Primary System following a design basis Loss of Coolant Accident and to confine the postulated release of radioactive material. The primary containment consists of a steel pressure vessel in the shape of an inverted light bulb with a torus-shaped suppression chamber located below and encircling the drywell, which surrounds the Reactor Primary System and provides an essentially leak tight barrier against an uncontrolled release of radioactive material to the environment.

The isolation devices for the penetrations in the primary containment boundary are a part of the containment leak tight barrier. To maintain this leak tight barrier:

a. All penetrations required to be closed during accident conditions are either:

   1. capable of being closed by an OPERABLE automatic containment isolation system, or

   2. closed by manual valves, blind flanges, or de-activated automatic valves secured in their closed positions, except as provided in LCO 3.6.1.3, "Primary Containment Isolation Valves (PCIVs)";

b. The primary containment air lock is OPERABLE, except as provided in LCO 3.6.1.2, "Primary Containment Air Lock"; and

c. All manways and equipment hatches are closed.

This Specification ensures that the performance of the primary containment, in the event of a Design Basis Accident (DBA), meets the assumptions used in the safety analyses of References 1 and 2. SR 3.6.1.1.1 leakage rate requirements are in conformance with 10 CFR 50, Appendix J, Option B (Ref. 3), as modified by approved exemptions.

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES

The safety design basis for the primary containment is that it must withstand the pressures and temperatures of the limiting DBA without exceeding the design leakage rate.

The DBA that postulates the maximum release of radioactive material within primary containment is a LOCA. In the analysis of this accident, it is assumed that primary containment is OPERABLE such that release of fission products to the environment is controlled by the rate of primary containment leakage.

Analytical methods and assumptions involving the primary containment are presented in References 1 and 2. The safety analyses assume a nonmechanistic fission product release following a DBA, which forms the basis for determination of offsite doses. The fission product release is, in turn, based on an assumed leakage rate from the primary containment. OPERABILITY of the primary containment ensures that the leakage rate assumed in the safety analyses is not exceeded.

The maximum allowable leakage rate for the primary containment (L_a) is 0.635% by weight of the containment air per 24 hours at the design basis LOCA maximum peak containment pressure (P_a) of 58.0 psig.

Primary containment satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii) (Ref. 4).

LCO

Primary containment OPERABILITY is maintained by limiting leakage to $\leq 1.0\ L_a$, except prior to the first startup after performing a required Primary Containment Leakage Rate Testing Program leakage test. At this time, the applicable leakage limits must be met. In addition, the leakage from the drywell to the suppression chamber must be limited to ensure the pressure suppression function is accomplished and the suppression chamber pressure does not exceed design limits. Compliance with this LCO will ensure a primary containment configuration, including equipment hatches, that is

(continued)

BASES

LCO (continued)

structurally sound and that will limit leakage to those leakage rates assumed in the safety analyses.

Individual leakage rates specified for the primary containment air lock are addressed in LCO 3.6.1.2.

APPLICABILITY

In MODES 1, 2, and 3, a DBA could cause a release of radioactive material to primary containment. In MODES 4 and 5, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Therefore, primary containment is not required to be OPERABLE in MODES 4 and 5 to prevent leakage of radioactive material from primary containment.

ACTIONS

A.1

In the event primary containment is inoperable, primary containment must be restored to OPERABLE status within 1 hour. The 1 hour Completion Time provides a period of time to correct the problem commensurate with the importance of maintaining primary containment OPERABILITY during MODES 1, 2, and 3. This time period also ensures that the probability of an accident (requiring primary containment OPERABILITY) occurring during periods where primary containment is inoperable is minimal.

B.1 and B.2

If primary containment cannot be restored to OPERABLE status within the required Completion Time, 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 to MODE 4 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.6.1.1.1

Maintaining the primary containment OPERABLE requires compliance with the visual examinations and leakage rate test requirements of the Primary Containment Leakage Rate Testing Program. Failure to meet the air lock leakage limit (SR 3.6.1.2.1) or the main steam isolation valve leakage limit (SR 3.6.1.3.10) does not necessarily result in a failure of this SR. The impact of the failure to meet these SRs must be evaluated against the Type A, B, and C acceptance criteria of the Primary Containment Leakage Rate Testing Program.

As left leakage prior to the first startup after performing a required Primary Containment Leakage Rate Testing Program leakage test is required to be < 0.6 L_a for combined Type B and C leakage, and ≤ 0.75 L_a for overall Type A leakage. At all other times between required leakage rate tests, the acceptance criteria is based on an overall Type A leakage limit of ≤ 1.0 L_a. At ≤ 1.0 L_a the offsite dose consequences are bounded by the assumptions of the safety analysis. The Frequency is required by the Primary Containment Leakage Rate Testing Program.

SR 3.6.1.1.2

Maintaining the pressure suppression function of primary containment requires limiting the leakage from the drywell to the suppression chamber. Thus, if an event were to occur that pressurized the drywell, the steam would be directed through the downcomers into the suppression pool. This SR is a leak test that confirms that the bypass area between the drywell and the suppression chamber is less than a one inch diameter hole. This ensures that the leakage paths that would bypass the suppression pool are within allowable limits.

Satisfactory performance of this SR can be achieved by establishing a known differential pressure between the drywell and the suppression chamber and verifying that the pressure in either the suppression chamber or the drywell

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.6.1.1.2 (continued)

does not change by more than the calculated amount per minute over a 10 minute period. The leakage test is performed every 18 months. The 18 month Frequency was developed considering it is prudent that this Surveillance be performed during a unit outage and also in view of the fact that component failures that might have affected this test are identified by other primary containment SRs. Two consecutive test failures, however, would indicate unexpected primary containment degradation; in this event, as the Note indicates, increasing the Frequency to once every 9 months is required until the situation is remedied as evidenced by passing two consecutive tests.

REFERENCES

1. USAR, Section V-2.4.
2. USAR, Section XIV-6.3.
3. 10 CFR 50, Appendix J, Option B.
4. 10 CFR 50.36(c)(2)(ii).

BASES

ACTIONS

D.1 and D.2 (continued)

required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.6.1.2.1

Maintaining the primary containment air lock OPERABLE requires compliance with the leakage rate test requirements of the Primary Containment Leakage Rate Testing Program. This SR reflects the leakage rate testing requirements with respect to air lock leakage (Type B leakage tests). The acceptance criteria were established as a small fraction of the total allowable primary containment leakage. The periodic testing requirements verify that the air lock leakage does not exceed the allowed fraction of the overall primary containment leakage rate. The Frequency is required by the Primary Containment Leakage Rate Testing Program.

The SR has been modified by two Notes. Note 1 states that an inoperable air lock door does not invalidate the previous

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

successful performance of the overall air lock leakage test. This is considered reasonable since either air lock door is capable of providing a fission product barrier in the event of a DBA. Note 2 has been added to this SR, requiring the results to be evaluated against the acceptance criteria which are applicable to SR 3.6.1.1.1. This ensures that air lock leakage is properly accounted for in determining the combined Type B and C primary containment leakage rate.

SR 3.6.1.2.2

The air lock interlock mechanism is designed to prevent simultaneous opening of both doors in the air lock. Since both the inner and outer doors of an air lock are designed to withstand the maximum expected post accident primary containment pressure, closure of either door will support primary containment OPERABILITY. Thus, the interlock feature supports primary containment OPERABILITY while the air lock is being used for personnel transit in and out of the containment. Periodic testing of this interlock demonstrates that the interlock will function as designed and that simultaneous inner and outer door opening will not inadvertently occur. Due to the purely mechanical nature of this interlock, and given that the interlock mechanism is not normally challenged when primary containment is used for entry and exit (procedures require strict adherence to single door opening), this test is only required to be performed every 24 months. 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 loss of primary containment OPERABILITY if the Surveillance were performed with the reactor at power. The 24 month Frequency is based on engineering judgement and is considered adequate given that the interlock is not challenged during use of the airlock.

REFERENCES

1. USAR, Section V-2.3.4.5.
2. USAR, Section V-2.3.2.
3. 10 CFR 50.36(c)(2)(ii).

BASES

REFERENCES (continued)

4. 10 CFR 50, Appendix J, Option B.

BASES

ACTIONS

A.1 and A.2 (continued)

occur. This Required Action does not require any testing or device manipulation. Rather, it involves verification that those devices outside containment and capable of potentially being mispositioned are in the correct position. The Completion Time of "once per 31 days for isolation devices outside primary containment" is appropriate because the devices are operated under administrative controls and the probability of their misalignment is low. For the devices inside primary containment, the time period specified "prior to entering MODE 2 or 3 from MODE 4, if primary containment was de-inerted while in MODE 4, if not performed within the previous 92 days" is based on engineering judgment and is considered reasonable in view of the inaccessibility of the devices and other administrative controls ensuring that device misalignment is an unlikely possibility.

Condition A is modified by a Note indicating that this Condition is only applicable to those penetration flow paths with two PCIVs. For penetration flow paths with one PCIV, Condition C provides the appropriate Required Actions.

Required Action A.2 is modified by 2 Notes. Note 1 applies to isolation devices located in high radiation areas, and allows them to be verified by use of administrative means. Allowing verification by administrative means is considered acceptable, since access to these areas is typically restricted. Note 2 applies to isolation devices that are locked, sealed, or otherwise secured in position and allows these devices to be verified closed by use of administrative means. Allowing verification by administrative means is considered acceptable, since the function of locking, sealing, or securing components is to ensure that these devices are not inadvertently repositioned. Therefore, the probability of misalignment, once they have been verified to be in the proper position, is low.

B.1

With one or more penetration flow paths with two PCIVs inoperable, except due to MSIV leakage not within a limit, either the inoperable PCIVs must be restored to OPERABLE status or the affected penetration flow path must be isolated within 1 hour. The method of isolation must include the use of at least one isolation barrier that cannot be adversely

(continued)

BASES

affected by a single active failure. Isolation barriers that meet this criterion are a closed and de-activated automatic valve, a closed manual valve, and a blind flange. The 1 hour Completion Time is consistent with the ACTIONS of LCO 3.6.1.1.

Condition B is modified by a Note indicating this Condition is only applicable to penetration flow paths with two PCIVs. For penetration flow paths with one PCIV, Condition C provides the appropriate Required Actions.

C.1 and C.2

With one or more penetration flow paths with one PCIV inoperable, the inoperable valve must be restored to OPERABLE status or the affected penetration flow path must be isolated. The method of isolation must include the use of at least one isolation barrier that cannot be adversely affected by a single active failure. Isolation barriers that meet this criterion are a closed and de-activated automatic valve, a closed manual valve, and a blind flange. A check valve may not be used to isolate the affected penetration. Required Action C.1 must be completed within 4 hours for lines other than excess flow check valve (EFCV) lines and 12 hours for EFCV lines. The Completion Time of 4 hours is reasonable considering the relative stability of the closed system (hence, reliability) to act as a penetration isolation boundary and the relative importance of supporting primary containment OPERABILITY during MODES 1, 2, and 3. The Completion Time of 12 hours is reasonable considering the instrument and the small pipe diameter of penetration (hence, reliability) to act as a penetration isolation boundary and the small pipe diameter of the affected penetrations. In the event the affected penetration flow path is isolated in accordance with Required Action C.1, the affected penetration must be verified to be isolated on a periodic basis. This is necessary to ensure that primary containment penetrations required to be isolated following an accident are isolated. This Required Action does not require any testing or valve manipulation. Rather, it involves verification, through a system walkdown, that those valves outside containment and capable of potentially being mispositioned are in the correct position. The Completion Time of once per 31 days for verifying each affected penetration is isolated is appropriate because the devices are operated under administrative controls and the probability of their misalignment is low.

(continued)

BASES

Condition C is modified by a Note indicating that this Condition is only applicable to penetration flow paths with only one PCIV. For penetration flow paths with two PCIVs, Conditions A and B provide the appropriate Required Actions.

Required Action C.2 is modified by 2 Notes. Note 1 applies to valves and blind flanges located in high radiation areas and allows them to be verified by use of administrative means. Allowing verification by administrative means is considered acceptable, since access to these areas is typically restricted. Note 2 applies to isolation devices that are locked, sealed, or otherwise secured in position and allows these devices to be verified closed by use of administrative means. Allowing verification by administrative means is considered acceptable, since the function of locking, sealing, or securing components is to ensure that these devices are not inadvertently repositioned. Therefore, the probability of misalignment, once they have been verified to be in the proper position, is low.

D.1

With any MSIV leakage rate not within limit, the assumptions of the safety analysis may not be met. Therefore, the leakage must be restored to within limit within 8 hours. Restoration can be accomplished by isolating the penetration that caused the limit to be exceeded by use of one closed and de-activated automatic valve, closed manual valve, or blind flange. When a penetration is isolated, the leakage rate for the isolated penetration is assumed to be the actual pathway leakage through the isolation device. If two isolation devices are used to isolate the penetration, the leakage rate is assumed to be the lesser actual pathway leakage of the two devices. The 8 hour Completion Time is reasonable considering the time required to restore the leakage by isolating the penetration, the fact that MSIV closure will result in isolation of the main steam line(s) and a potential for plant shutdown, and the relative importance of MSIV leakage to the overall containment function.

E.1 and E.2

If any Required Action and associated Completion Time cannot be met in MODE 1, 2, or 3, the plant must be brought to

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.1.3.3 (continued)

controls consist of stationing a dedicated operator at the controls of the valve, who is in continuous communication with the control room. In this way, the penetration can be rapidly isolated when a need for primary containment isolation is indicated.

SR 3.6.1.3.4

The traversing incore probe (TIP) shear isolation valves are actuated by explosive charges. Surveillance of explosive charge continuity provides assurance that TIP valves will actuate when required. Other administrative controls, such as those that limit the shelf life of the explosive charges, must be followed. The 31 day Frequency is based on operating experience that has demonstrated the reliability of the explosive charge continuity.

SR 3.6.1.3.5

Verifying the isolation time of each power operated automatic PCIV is within limits is required to demonstrate OPERABILITY. MSIVs may be excluded from this SR since MSIV full closure isolation time is demonstrated by SR 3.6.1.3.6. The isolation time test ensures that the valve will isolate in a time period less than or equal to that assumed in the safety analyses. The isolation time and Frequency of this SR are in accordance with the requirements of the Inservice Testing Program.

SR 3.6.1.3.6

Verifying that the isolation time of each MSIV is within the specified limits is required to demonstrate OPERABILITY. The isolation time test ensures that the MSIV will isolate in a time period that does not exceed the times assumed in the DBA and transient analyses. This ensures that the

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.1.3.9

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

SR 3.6.1.3.10

The analyses in References 8 and 9 are based on leakage that is less than the specified leakage rate. Leakage through each MSIV must be $\leq$ 11.5 scfh when tested at $\geq P_t$ (29 psig). The Frequency is required by the Primary Containment Leakage Rate Testing Program.

SR 3.6.1.3.11

Verifying each inboard 24 inch primary containment purge and vent valve (PC-230 MV, PC-231 MV, PC-232 MV, and PC-233 MV) is blocked to restrict the maximum opening angle to 60° is required to ensure that the valves can close under DBA conditions within the times assumed in the analysis of References 7 and 8. If a LOCA occurs, the purge and vent valves must close to maintain containment leakage within the values assumed in the accident analysis. At other times, pressurization concerns are not present, thus the purge valves can be fully open. The 18 month Frequency is appropriate because the blocking devices may be removed during a refueling outage.

(continued)

BASES

REFERENCES

1. USAR, Chapter XIV.

2. Amendment 25 to the FSAR.

3. NEDC 96-006, "Estimate of Steam Tunnel's HELB," dated March 30, 1996.

4. USAR Section IV-4.9.

5. 10 CFR 50.36(c)(2)(ii).

6. USAR, Table V-2-2.

7. USAR, Burns and Roe Drawing 4259, Sheets 1 and 1A, and Burns and Roe Drawing 4260, Sheets 2A and 2B (Incorporated by Reference).

8. USAR, Section V-2.0.

9. USAR, Section XIV-6.3.

10. 10 CFR 50, Appendix J, Option B.

BASES (continued)

ACTIONS

A.1

With drywell pressure not within the limit of the LCO, drywell pressure must be restored within 1 hour. The Required Action is necessary to return operation to within the bounds of the primary containment analysis. The 1 hour Completion Time is consistent with the ACTIONS of LCO 3.6.1.1, "Primary Containment," which requires that primary containment be restored to OPERABLE status within 1 hour.

B.1 and B.2

If drywell pressure cannot be restored to within limit within the required Completion Time, 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 to MODE 4 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.6.1.4.1

Verifying that drywell pressure is within limit ensures that unit operation remains within the limit assumed in the primary containment analysis. The 12 hour Frequency of this SR was developed, based on operating experience related to trending of drywell pressure variations during the applicable MODES. Furthermore, the 12 hour Frequency is considered adequate in view of other indications available in the control room, including alarms, to alert the operator to an abnormal drywell pressure condition.

REFERENCES

1. USAR, Section V-2.

2. 10 CFR 50.36(c)(2)(ii).

B 3.6 CONTAINMENT SYSTEMS

B 3.6.1.5 Drywell Air Temperature

BASES

BACKGROUND

The drywell contains the reactor vessel and piping, which add heat to the airspace. Drywell fan coil units remove heat and maintain a suitable environment. The average airspace temperature affects the calculated response to postulated Design Basis Accidents (DBAs). The limitation on the drywell average air temperature was developed as reasonable, based on operating experience. The limitation on drywell air temperature is used in the Reference 1 safety analyses.

APPLICABLE SAFETY ANALYSES

Primary containment performance is evaluated for a spectrum of break sizes for postulated loss of coolant accidents (LOCAs) (Ref. 1). Among the inputs to the design basis analysis is the initial drywell average air temperature (Ref. 2). Analyses assume an initial average drywell air temperature of 150°F. This limitation ensures that the safety analysis remains valid by maintaining the expected initial conditions and ensures that the peak LOCA drywell temperature does not exceed the maximum allowable temperature of 309°F. Exceeding this design temperature may result in the degradation of the primary containment structure under accident loads. Equipment inside primary containment required to mitigate the effects of a DBA is designed to operate and be capable of operating under environmental conditions expected for the accident.

Drywell air temperature satisfies Criterion 2 of 10 CFR 50.36 (c)(2)(ii) (Ref. 3).

LCO

In the event of a DBA, with an initial drywell average air temperature less than or equal to the LCO temperature limit, the resultant peak accident temperature is maintained below the maximum allowable drywell temperature. As a result, the ability of primary containment to perform its design function is ensured.

BASES

SURVEILLANCE REQUIREMENTS (continued)

The 24 hour Frequency of the SR was developed based on operating experience related to drywell average air temperature variations and temperature instrument drift during the applicable MODES and the low probability of a DBA occurring between surveillances. Furthermore, the 24 hour Frequency is considered adequate in view of other indications available in the control room, including alarms, to alert the operator to an abnormal drywell air temperature condition.

REFERENCES

1. USAR, Section XIV-6.3.
2. USAR, Table V-2-1.
3. 10 CFR 50.36(c)(2)(ii).

BASES

APPLICABLE SAFETY ANALYSES (continued)

that leakage from the primary containment is processed by the Standby Gas Treatment (SGT) System before being released to the environment.

Maintaining SCIVs OPERABLE with isolation times within limits ensures that fission products will remain trapped inside secondary containment so that they can be treated by the SGT System prior to discharge to the environment.

SCIVs satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii) (Ref. 5).

LCO

SCIVs form a part of the secondary containment boundary. The SCIV safety function is related to control of offsite radiation releases resulting from DBAs.

The power operated automatic isolation valves are considered OPERABLE when their isolation times are within limits and the valves actuate on an automatic isolation signal. The valves covered by this LCO, along with their associated stroke times, are listed in Reference 6.

The normally closed isolation valves or blind flanges are considered OPERABLE when manual valves are closed or open in accordance with appropriate administrative controls, automatic SCIVs are de-activated and secured in their closed position, and blind flanges are in place. These passive isolation valves or devices are listed in Reference 6.

APPLICABILITY

In MODES 1, 2, and 3, a DBA could lead to a fission product release to the primary containment that leaks to the secondary containment. Therefore, the OPERABILITY of SCIVs is required.

In MODES 4 and 5, the probability and consequences of these events are reduced due to pressure and temperature limitations in these MODES. Therefore, maintaining SCIVs OPERABLE is not required in MODE 4 or 5, except for other situations under which significant radioactive releases can be postulated, such as during operations with a potential for draining the reactor vessel (OPDRVs), during CORE

(continued)

BASES

ACTIONS

A.1 and A.2 (continued)

isolate the penetration, and the probability of a DBA, which requires the SCIVs to close, occurring during this short time is very low.

For affected penetrations that have been isolated in accordance with Required Action A.1, the affected penetration must be verified to be isolated on a periodic basis. This is necessary to ensure that [secondary] containment penetrations required to be isolated following an accident, but no longer capable of being automatically isolated, will be in the isolation position should an event occur. The Completion Time of once per 31 days is appropriate because the isolation devices are operated under administrative controls and the probability of their misalignment is low. This Required Action does not require any testing or device manipulation. Rather, it involves verification that the affected penetration remains isolated.

Required Action A.2 is modified by two Notes. Note 1 applies to devices located in high radiation areas and allows them to be verified closed by use of administrative controls. Allowing verification by administrative controls is considered acceptable, since access to these areas is typically restricted. Note 2 applies to isolation devices that are locked, sealed, or otherwise secured in position and allows these devices to be verified closed by use of administrative means. Allowing verification by administrative means is considered acceptable, since the function of locking, sealing, or securing components is to ensure that these devices are not inadvertently repositioned. Therefore, the probability of misalignment, once they have been verified to be in the proper position, is low.

B.1

With two SCIVs in one or more penetration flow paths inoperable, the affected penetration flow path must be isolated within 4 hours. The method of isolation must include the use of at least one isolation barrier that cannot be adversely affected by a single active failure. Isolation barriers that meet this criterion are a closed and de-activated automatic valve, a closed manual valve, and a blind flange. The 4 hour Completion Time is reasonable considering the time required to isolate the penetration and the probability of a DBA, which requires the SCIVs to close, occurring during this short time, is very low.

(continued)

BASES

ACTIONS B.1 (continued)

The Condition has been modified by a Note stating that Condition B is only applicable to penetration flow paths with two isolation valves. This clarifies that only Condition A is entered if one SCIV is inoperable in multiple penetrations.

C.1 and C.2

If any Required Action and associated Completion Time 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 to MODE 4 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

D.1, D.2, and D.3

If any Required Action and associated Completion Time are not met, the plant must be placed in a condition in which the LCO does not apply. If applicable, CORE ALTERATIONS and the movement of irradiated fuel assemblies in the secondary containment must be immediately suspended. Suspension of these activities shall not preclude completion of movement of a component to a safe position. Also, if applicable, actions must be immediately initiated to suspend OPDRVs in order to minimize the probability of a vessel draindown and the subsequent potential for fission product release. Actions must continue until OPDRVs are suspended.

LCO 3.0.3 is not applicable while in MODE 4 or 5. However, since irradiated fuel assembly movement can occur in MODE 1, 2, or 3, Required Action D.1 has been modified by a Note stating that LCO 3.0.3 is not applicable. If moving irradiated fuel assemblies while in MODE 4 or 5, LCO 3.0.3 would not specify any action. If moving fuel while in MODE 1, 2, or 3, the fuel movement is independent of reactor operations. Therefore, in either case, inability to suspend movement of irradiated fuel assemblies would not be a sufficient reason to require a reactor shutdown.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.6.4.2.1

This SR verifies that each secondary containment manual isolation valve and blind flange that is required to be closed during accident conditions is closed. The SR helps to ensure that post accident leakage of radioactive fluids or gases outside of the secondary containment boundary is within design limits. This SR does not require any testing or valve manipulation. Rather, it involves verification that those SCIVs in secondary containment that are capable of being mispositioned are in the correct position.

Since these SCIVs are readily accessible to personnel during normal operation and verification of their position is relatively easy, the 31 day Frequency was chosen to provide added assurance that the SCIVs are in the correct positions.

Two Notes have been added to this SR. The first Note applies to valves and blind flanges located in high radiation areas and allows them to be verified by use of administrative controls. Allowing verification by administrative controls is considered acceptable, since access to these areas is typically restricted during MODES 1, 2, and 3 for ALARA reasons. Therefore, the probability of misalignment of these isolation devices, once they have been verified to be in the proper position, is low.

A second Note has been included to clarify that SCIVs that are open under administrative controls are not required to meet the SR during the time the SCIVs are open. These controls consist of stationing a dedicated operator at the controls of the valve, who is in continuous communication with the control room. In this way, the penetration can be rapidly isolated when a need for secondary containment isolation is indicated.

SR 3.6.4.2.2

Verifying that the isolation time of each power operated automatic SCIV is within limits is required to demonstrate OPERABILITY. The isolation time test ensures that the SCIV will isolate in a time period less than or equal to that a`ssumed in the safety analyses. The isolation time and Frequency of this SR are in accordance with the Inservice Testing Program.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.3.4 (continued)

the safety related equipment during an accident event. This is demonstrated by the use of an actual or simulated initiation signal. The initiation signal is caused by low REC heat exchanger outlet pressure (which has an analytically determined limit of 55 psig decreasing). Also, a Group VI isolation signal will open the REC heat exchanger service water outlet valves and the REC critical loop supply valves to provide cooling water to essential components.

Operating experience has shown that these components usually pass the SR when performed at the 18 month Frequency. Therefore, this Frequency is concluded to be acceptable from a reliability standpoint.

REFERENCES

1. USAR, Section X-6.
2. 10 CFR 50.36(c)(2)(ii).
3. DC 93-057
4. NEDC 92-050X and NEDC 97-087