Nebraska Public Power District
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NLS2007036
April 30, 2007

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

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

Dear Sir or Madam,

The purpose of this letter is to provide changes to the Cooper Nuclear Station (CNS) Technical Specification Bases implemented without prior Nuclear Regulatory Commission approval. In accordance with the requirements of CNS Technical Specification 5.5.10.d, these changes are provided on a frequency consistent with 10 CFR 50.71(e). The enclosed Bases changes are for the time period from June 1, 2005, through March 14, 2007. Also enclosed are filing instructions and an updated List of Effective Pages for the CNS Technical Specification Bases.

If you have any questions regarding this submittal, please contact me at (402) 825-2774.

Sincerely,

Paul V. Fleming
Licensing Manager

/lb

Enclosure

cc: Regional Administrator, w/enclosure
USNRC - Region IV

Cooper Project Manager, w/enclosure
USNRC - NRR Project Directorate IV-1

Senior Resident Inspector, w/enclosure
USNRC - CNS

NPG Distribution, w/o enclosure

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A001

0.ATTACHMENT 3 LIST OF REGULATORY COMMITMENTS©

Correspondence Number: NLS2007036

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

COMMITMENT	COMMITMENT NUMBER	COMMITTED DATE OR OUTAGE
None		

NLS2007036
ENCLOSURE

TECHNICAL SPECIFICATION
BASES CHANGES

FILING INSTRUCTIONS

TECHNICAL SPECIFICATION BASES

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B 3.0-1	B 3.0-1
B 3.0-9	B 3.0-9
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B 3.0-11	B 3.0-11
B 3.0-12	B 3.0-12
B 3.0-13	B 3.0-13
B 3.0-14	B 3.0-14
B 3.0-15	B 3.0-15
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---	B 3.0-17
B 3.1-25	B 3.1-25
B 3.1-26	B 3.1-26
B 3.1-27	B 3.1-27
B 3.2-1	B 3.2-1
B 3.3-28	B 3.3-28
B 3.3-29	B 3.3-29
B 3.3-30	B 3.3-30
B 3.3-40	B 3.3-40
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B 3.3-46	B 3.3-46
B 3.3-53	B 3.3-53
B 3.3-61	B 3.3-61
B 3.3-78	B 3.3-78
B 3.3-89	B 3.3-89
B 3.3-96	B 3.3-96
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FILING INSTRUCTIONS

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B 3.3-100	B 3.3-100
B 3.3-101	B 3.3-101
B 3.3-104	B 3.3-104
B 3.3-124	B 3.3-124
B 3.3-136	B 3.3-136
B 3.3-138	B 3.3-138
B 3.3-139	B 3.3-139
B 3.3-140	B 3.3-140
B 3.3-141	B 3.3-141
B 3.3-142	B 3.3-142
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B 3.3-149	B 3.3-149
B 3.3-156	B 3.3-156
B 3.3-157	B 3.3-157
B 3.3-165	B 3.3-165
B 3.3-167	B 3.3-167
B 3.3-168	B 3.3-168
B 3.3-170	B 3.3-170
B 3.3-171	B 3.3-171
B 3.3-172	B 3.3-172
B 3.3-176	B 3.3-176
B 3.3-183	B 3.3-183
B 3.3-185	B 3.3-185
B 3.3-187	B 3.3-187
B 3.3-188	B 3.3-188
B 3.3-189	B 3.3-189
B 3.3-190	B 3.3-190
B 3.3-191	B 3.3-191
B 3.3-192	B 3.3-192
B 3.3-193	B 3.3-193
B 3.3-204	B 3.3-204
B 3.3-210	B 3.3-210
B 3.4-46	B 3.4-46
B 3.4-50	B 3.4-50
B 3.5-14	B 3.5-14
B 3.5-15	B 3.5-15
B 3.5-29	B 3.5-29
B 3.6-4	B 3.6-4
B 3.6-28	B 3.6-28

FILING INSTRUCTIONS

TECHNICAL SPECIFICATION BASES

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B 3.6-37	B 3.6-37
B 3.6-38	B 3.6-38
B 3.6-44	B 3.6-44
B 3.6-67	B 3.6-67
B 3.6-68	B 3.6-68
B 3.6-69	B 3.6-69
B 3.6-70	B 3.6-70
B 3.6-72	B 3.6-72
B 3.6-73	B 3.6-73
B 3.6-74	B 3.6-74
B 3.6-76	B 3.6-76
B 3.6-80	B 3.6-80
B 3.6-81	B 3.6-81
B 3.6-82	B 3.6-82
B 3.6-83	B 3.6-83
B 3.7-18	B 3.7-18
B 3.7-19	B 3.7-19
B 3.7-20	B 3.7-20
B 3.7-25	B 3.7-25
B 3.7-27	B 3.7-27
B 3.7-29	B 3.7-29
B 3.8-11	B 3.8-11
B 3.8-12	B 3.8-12
B 3.8-28	B 3.8-28
B 3.8-31	B 3.8-31
B 3.8-33	B 3.8-33
B 3.8-34	B 3.8-34
B 3.8-45	B 3.8-45
B 3.8-70	B 3.8-70
B 3.9-3	B 3.9-3
B 3.9-4	B 3.9-4
B 3.9-5	B 3.9-5
B 3.9-6	B 3.9-6
B 3.9-7	B 3.9-7
B 3.9-8	B 3.9-8
B 3.9-19	B 3.9-19
B 3.9-21	B 3.9-21
B 3.10-1	B 3.10-1
B 3.10-2	B 3.10-2

FILING INSTRUCTIONS

TECHNICAL SPECIFICATION BASES

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B 3.10-4	B 3.10-4
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B 2.0-1	12/18/03	B 3.1-18	0
B 2.0-2	0	B 3.1-19	0
B 2.0-3	0	B 3.1-20	0
B 2.0-4	6/10/99	B 3.1-21	12/18/03
B 2.0-5	12/18/03	B 3.1-22	0
B 2.0-6	12/18/03	B 3.1-23	0
B 2.0-7	0	B 3.1-24	0
B 2.0-8	12/18/03	B 3.1-25	05/09/06
		B 3.1-26	02/02/06
B 3.0-1	06/30/06	B 3.1-27	05/09/06
B 3.0-2	0	B 3.1-28	12/18/03
B 3.0-3	0	B 3.1-29	0
B 3.0-4	0	B 3.1-30	0
B 3.0-5	0	B 3.1-31	0
B 3.0-6	0	B 3.1-32	0
B 3.0-7	0	B 3.1-33	01/30/03
B 3.0-8	0	B 3.1-34	0
B 3.0-9	06/30/06	B 3.1-35	1/14/05
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B 3.0-13	06/30/06	B 3.1-39	0
B 3.0-14	06/30/06	B 3.1-40	0
B 3.0-15	06/30/06	B 3.1-41	0
B 3.0-16	06/30/06	B 3.1-42	0
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		B 3.1-44	0
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B 3.1-3	6/10/99	B 3.1-47	0
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B 3.2-9	0	B 3.3-42	0
B 3.2-10	4/12/00	B 3.3-43	0
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B 3.3-1	0	B 3.3-45	02/05/07
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B 3.3-12	1	B 3.3-56	1
B 3.3-13	1	B 3.3-57	1
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B 3.3-19	6/28/01	B 3.3-63	1
B 3.3-20	6/28/01	B 3.3-64	6/10/99
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B 3.3-33	0	B 3.3-77	0
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B 3.3-35	0	B 3.3-79	1
B 3.3-36	0	B 3.3-80	0
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B 3.3-86	0	B 3.3-130	0
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B 3.3-89	02/20/07	B 3.3-133	0
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B 3.3-102	1	B 3.3-146	0
B 3.3-103	6/10/99	B 3.3-147	0
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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	8/29/02	B 3.3-152	4/12/00
B 3.3-109	0	B 3.3-153	0
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B 3.3-114	0	B 3.3-158	0
B 3.3-115	0	B 3.3-159	0
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B 3.3-118	0	B 3.3-162	0
B 3.3-119	0	B 3.3-163	6/28/01
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B 3.3-175	6/28/01	B 3.4-8	0
B 3.3-176	02/20/07	B 3.4-9	0
B 3.3-177	6/28/01	B 3.4-10	0
B 3.3-178	0	B 3.4-11	1
B 3.3-179	0	B 3.4-12	1
B 3.3-180	0	B 3.4-13	4/12/00
B 3.3-181	0	B 3.4-14	0
B 3.3-182	6/28/01	B 3.4-15	0
B 3.3-183	02/20/07	B 3.4-16	0
B 3.3-184	6/28/01	B 3.4-17	0
B 3.3-185	10/05/06	B 3.4-18	6/10/99
B 3.3-186	11/04/01	B 3.4-19	0
B 3.3-187	12/22/05	B 3.4-20	0
B 3.3-188	12/22/05	B 3.4-21	0
B 3.3-189	10/05/06	B 3.4-22	0
B 3.3-190	10/05/06	B 3.4-23	0
B 3.3-191	10/05/06	B 3.4-24	0
B 3.3-192	10/05/06	B 3.4-25	0
B 3.3-193	02/20/07	B 3.4-26	0
B 3.3-194	11/04/01	B 3.4-27	0
B 3.3-195	11/04/01	B 3.4-28	6/28/01
B 3.3-196	11/04/01	B 3.4-29	0
B 3.3-197	11/04/01	B 3.4-30	0
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B 3.3-199	11/04/01	B 3.4-32	11/04/01
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B 3.3-201	11/04/01	B 3.4-34	0
B 3.3-202	11/04/01	B 3.4-35	0
B 3.3-203	11/04/01	B 3.4-36	0
B 3.3-204	02/20/07	B 3.4-37	0
B 3.3-205	11/04/01	B 3.4-38	0
B 3.3-206	11/04/01	B 3.4-39	1
B 3.3-207	11/04/01	B 3.4-40	0
B 3.3-208	11/04/01	B 3.4-41	0
B 3.3-209	11/04/01	B 3.4-42	0
B 3.3-210	02/20/07	B 3.4-43	0
		B 3.4-44	08/11/04
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B 3.4-51	0	B 3.6-8	0
B 3.4-52	08/11/04	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
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B 3.5-1	1	B 3.6-14	3/8/00
B 3.5-2	11/24/03	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	04/26/04	B 3.6-18	0
B 3.5-6	04/26/04	B 3.6-19	11/28/01
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B 3.5-9	1	B 3.6-22	11/28/01
B 3.5-10	0	B 3.6-23	11/28/01
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	02/20/07	B 3.6-27	11/04/01
B 3.5-15	02/20/07	B 3.6-28	06/01/06
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	12/18/03	B 3.6-31	6/14/00
B 3.5-19	0	B 3.6-32	12/27/02
B 3.5-20	0	B 3.6-33	12/27/02
B 3.5-21	0	B 3.6-34	12/14/01
B 3.5-22	0	B 3.6-35	0
B 3.5-23	12/18/03	B 3.6-36	0
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B 3.5-25	1	B 3.6-38	02/20/07
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	02/20/07	B 3.6-42	0
B 3.5-30	12/18/03	B 3.6-43	0
		B 3.6-44	02/20/07
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
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B 3.6-4	11/06/06	B 3.6-48	0

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B 3.6-51	0	B 3.7-10	1
B 3.6-52	8/13/02	B 3.7-11	0
B 3.6-53	0	B 3.7-12	1
B 3.6-54	0	B 3.7-13	10/22/02
B 3.6-55	8/13/02	B 3.7-14	0
B 3.6-56	8/13/02	B 3.7-15	0
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B 3.6-62	0	B 3.7-21	11/04/01
B 3.6-63	8/13/02	B 3.7-22	0
B 3.6-64	0	B 3.7-23	0
B 3.6-65	0	B 3.7-24	0
B 3.6-66	0	B 3.7-25	10/05/06
B 3.6-67	10/05/06	B 3.7-26	0
B 3.6-68	10/05/06	B 3.7-27	10/05/06
B 3.6-69	10/05/06	B 3.7-28	0
B 3.6-70	10/05/06	B 3.7-29	01/27/06
B 3.6-71	0	B 3.7-30	1
B 3.6-72	10/05/06	B 3.7-31	1
B 3.6-73	10/05/06		
B 3.6-74	10/05/06	B 3.8-1	12/18/03
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B 3.6-80	10/05/06	B 3.8-7	11/28/01
B 3.6-81	10/05/06	B 3.8-8	3/15/01
B 3.6-82	10/05/06	B 3.8-9	1/17/05
B 3.6-83	10/05/06	B 3.8-10	07/01/04
B 3.6-84	12/18/03	B 3.8-11	12/22/05
		B 3.8-12	12/22/05
B 3.7-1	1	B 3.8-13	1
B 3.7-2	0	B 3.8-14	1
B 3.7-3	03/24/04	B 3.8-15	12/18/03
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	8/20/02	B 3.8-19	0

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B 3.8-23	0	B 3.8-67	10/14/04
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B 3.8-26	0	B 3.8-70	04/11/06
B 3.8-27	0	B 3.8-71	0
B 3.8-28	05/09/06	B 3.8-72	0
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B 3.8-30	0	B 3.8-74	0
B 3.8-31	05/09/06	B 3.8-75	0
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B 3.8-33	12/22/05	B 3.9-1	12/18/03
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B 3.8-35	10/21/04	B 3.9-3	05/09/06
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B 3.8-44	0	B 3.9-12	12/18/03
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B 3.8-47	0	B 3.9-15	12/18/03
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B 3.8-51	0	B 3.9-19	10/05/06
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B 3.10-12	0		
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B 3.10-14	0		
B 3.10-15	0		
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B 3.10-17	0		
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B 3.0 LIMITING CONDITION FOR OPERATION (LCO) APPLICABILITY

BASES

LCOs

LCO 3.0.1 through LCO 3.0.8 establish the general requirements applicable to all Specifications in Sections 3.1 through 3.10 and apply at all times, unless otherwise stated.

LCO 3.0.1

LCO 3.0.1 establishes the Applicability statement within each individual Specification as the requirement for when the LCO is required to be met (i.e., when the unit is in the MODES or other specified conditions of the Applicability statement of each Specification).

LCO 3.0.2

LCO 3.0.2 establishes that upon discovery of a failure to meet an LCO, the associated ACTIONS shall be met. The Completion Time of each Required Action for an ACTIONS Condition is applicable from the point in time that an ACTIONS Condition is entered. The Required Actions establish those remedial measures that must be taken within specified Completion Times when the requirements of an LCO are not met. This Specification establishes that:

a. Completion of the Required Actions within the specified Completion Times constitutes compliance with a Specification; and

b. Completion of the Required Actions is not required when an LCO is met within the specified Completion Time, unless otherwise specified.

There are two basic types of Required Actions. The first type of Required Action specifies a time limit in which the LCO must be met. This time limit is the Completion Time to restore an inoperable system or component to OPERABLE status or to restore variables to within specified limits. If this type of Required Action is not completed within the specified Completion Time, a shutdown may be required to place the unit in a MODE or condition in which the Specification is not applicable. (Whether stated as a Required Action or not, correction of the entered Condition is an action that may always be considered upon entering

BASES

LCO 3.0.7 (continued)

perform special maintenance activities, and to perform special evolutions. Special Operations LCOs in Section 3.10 allow specified TS requirements to be changed to permit performances of these special tests and operations, which otherwise could not be performed if required to comply with the requirements of these TS. Unless otherwise specified, all the other TS requirements remain unchanged. This will ensure all appropriate requirements of the MODE or other specified condition not directly associated with or required to be changed to perform the special test or operation will remain in effect.

The Applicability of a Special Operations LCO represents a condition not necessarily in compliance with the normal requirements of the TS. Compliance with Special Operations LCOs is optional. A special operation may be performed either under the provisions of the appropriate Special Operations LCO or under the other applicable TS requirements. If it is desired to perform the special operation under the provisions of the Special Operations LCO, the requirements of the Special Operations LCO shall be followed. When a Special Operations LCO requires another LCO to be met, only the requirements of the LCO statement are required to be met regardless of that LCO's Applicability (i.e., should the requirements of this other LCO not be met, the ACTIONS of the Special Operations LCO apply, not the ACTIONS of the other LCO). However, there are instances where the Special Operations LCO ACTIONS may direct the other LCO ACTIONS be met. The Surveillances of the other LCO are not required to be met, unless specified in the Special Operations LCO. If conditions exist such that the Applicability of any other LCO is met, all the other LCO's requirements (ACTIONS and SRs) are required to be met concurrent with the requirements of the Special Operations LCO.

LCO 3.0.8

LCO 3.0.8 establishes conditions under which systems are considered to remain capable of performing their intended safety function when associated snubbers are not capable of providing their associated support function(s). This LCO states that the supported system is not considered to be inoperable solely due to one or more snubbers not capable of performing their associated support function(s). This is appropriate because a limited length of time is allowed for maintenance, testing, or repair of one or more snubbers not capable of performing their associated support function(s) and appropriate compensatory measures are specified in the snubber requirements, which are located outside of

BASES

the Technical Specifications (TS) under licensee control. The snubber requirements do not meet the criteria in 10 CFR 50.36(c)(2)(ii), and, as such, are appropriate for control by the licensee.

If the allowed time expires and the snubber(s) are unable to perform their associated support function(s), the affected supported system's LCO(s) must be declared not met and the Conditions and Required Actions entered in accordance with LCO 3.0.2.

LCO 3.0.8.a applies when one or more snubbers are not capable of providing their associated support function(s) to a single train or subsystem of a multiple train or subsystem supported system or to a single train or subsystem supported system. LCO 3.0.8.a can be used only if one of the following two means of heat removal is available:

(1) At least one high pressure makeup path (e.g., using HPCI or RCIC or equivalent) and heat removal capability including a minimum set of supporting equipment required for success, not associated with the inoperable snubber(s),

OR

(2) At least one low pressure makeup path (e.g., LPCI) and heat removal capability including a minimum set of supporting equipment required for success, not associated with the inoperable snubber(s),

LCO 3.0.8.a allows 72 hours to restore the snubber(s) before declaring the supported system inoperable. The 72 hour Completion Time is reasonable based on the low probability of a seismic event concurrent with an event that would require operation of the supported system occurring while the snubber(s) are not capable of performing their associated support function and due to the availability of the redundant train of the supported system.

LCO 3.0.8.b applies when one or more snubbers are not capable of providing their associated support function(s) to more than one train or subsystem of a multiple train or subsystem supported system. LCO 3.0.8.b can be used only if at least one success path exists, using equipment not associated with the inoperable snubber(s), to provide makeup and core cooling needed to mitigate LOOP accident sequences.

LCO 3.0.8.b allows 12 hours to restore the snubber(s) before declaring the supported system inoperable. The 12 hour Completion Time is reasonable based on the low probability of a seismic event concurrent with an event that would require operation of the supported system

BASES

occurring while the snubber(s) are not capable of performing their associated support function.

LCO 3.0.8 requires that risk be assessed and managed. Industry and NRC guidance on the implementation of 10 CFR 50.65(a)(4) (the Maintenance Rule) does not address seismic risk. However, use of LCO 3.0.8 should be considered with respect to other plant maintenance activities, and integrated into the existing Maintenance Rule process to the extent possible so that maintenance on any unaffected train or subsystem is properly controlled, and emergent issues are properly addressed. Ths risk assessment need not be quantified, but may be a qualitative awareness of the vulnerability of systems and components when one or more snubbers are not able to perform their associated support function.

B 3.0 SURVEILLANCE REQUIREMENT (SR) APPLICABILITY

BASES

SRs

SR 3.0.1 through SR 3.0.4 establish the general requirements applicable to all Specifications in Sections 3.1 through 3.10 and apply at all times, unless otherwise stated.

SR 3.0.1

SR 3.0.1 establishes the requirement that SRs must be met during the MODES or other specified conditions in the Applicability for which the requirements of the LCO apply, unless otherwise specified in the individual SRs. This Specification is to ensure that Surveillances are performed to verify the OPERABILITY of systems and components, and that variables are within specified limits. Failure to meet a Surveillance within the specified Frequency, in accordance with SR 3.0.2, constitutes a failure to meet an LCO.

Systems and components are assumed to be OPERABLE when the associated SRs have been met. Nothing in this Specification, however, is to be construed as implying that systems or components are OPERABLE when:

a. The systems or components are known to be inoperable, although still meeting the SRs; or

b. The requirements of the Surveillance(s) are known to be not met between required Surveillance performances.

Surveillances do not have to be performed when the unit is in a MODE or other specified condition for which the requirements of the associated LCO are not applicable, unless otherwise specified. The SRs associated with a Special Operations LCO are only applicable when the Special Operations LCO is used as an allowable exception to the requirements of a Specification.

Surveillances, including Surveillances invoked by Required Actions, do not have to be performed on inoperable equipment because the ACTIONS define the remedial measures that apply. Surveillances have to be met and performed in accordance with SR 3.0.2, prior to returning equipment to OPERABLE status.

BASES

SR 3.0.1 (continued)

Upon completion of maintenance, appropriate post maintenance testing is required to declare equipment OPERABLE. This includes ensuring applicable Surveillances are not failed and their most recent performance is in accordance with SR 3.0.2. Post maintenance testing may not be possible in the current MODE or other specified conditions in the Applicability due to the necessary unit parameters not having been established. In these situations, the equipment may be considered OPERABLE provided testing has been satisfactorily completed to the extent possible and the equipment is not otherwise believed to be incapable of performing its function. This will allow operation to proceed to a MODE or other specified condition where other necessary post maintenance tests can be completed. Some examples of this process are:

a. Control Rod Drive maintenance during refueling that requires scram testing at > 800 psi. However, if other appropriate testing is satisfactorily completed and the scram time testing of SR 3.1.4.3 is satisfied, the control rod can be considered OPERABLE. This allows startup to proceed to reach 800 psi to perform other necessary testing.

b. High pressure coolant injection (HPCI) maintenance during shutdown that requires system functional tests at a specified pressure. Provided other appropriate testing is satisfactorily completed, startup can proceed with HPCI considered OPERABLE. This allows operation to reach the specified pressure to complete the necessary post maintenance testing.

SR 3.0.2

SR 3.0.2 establishes the requirements for meeting the specified Frequency for Surveillances and any Required Action with a Completion Time that requires the periodic performance of the Required Action on a "once per..." interval.

SR 3.0.2 permits a 25% extension of the interval specified in the Frequency. This extension facilitates Surveillance scheduling and considers plant operating conditions that may not be suitable for conducting the Surveillance (e.g., transient conditions or other ongoing Surveillance or maintenance activities).

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 greater, applies from the point in time that it is discovered that the Surveillance has not been

BASES

SR 3.0.3 (continued)

performed in accordance with SR 3.0.2, and not at the time that the specified Frequency was not met. This delay period provides adequate time to complete Surveillances that have been missed. This delay period permits the completion of a Surveillance before complying with Required Actions or other remedial measures that might preclude completion of the Surveillance.

The basis for this delay period includes consideration of unit conditions, adequate planning, availability of personnel, the time required to perform the Surveillance, the safety significance of the delay in completing the required Surveillance, and the recognition that the most probable result of any particular Surveillance being performed is the verification of conformance with the requirements.

When a Surveillance with a Frequency based not on time intervals, but upon specified unit conditions, operating situations, or requirements of regulations (e.g., prior to entering MODE 1 after each fuel loading, or in accordance with 10 CFR 50, Appendix J, as modified by approved exemptions, etc.) is discovered to not to have been performed when specified, SR 3.0.3 allows the full delay period of up to the specified Frequency to perform the Surveillance. However, since there is not a time interval specified, the missed Surveillance should be performed at the first reasonable opportunity.

SR 3.0.3 provides a time limit for, and allowances for the performance of, Surveillances that become applicable as a consequence of MODE changes imposed by Required Actions.

Failure to comply with specified Frequencies for SRs is expected to be an infrequent occurrence. Use of the delay period established by SR 3.0.3 is a flexibility which is not intended to be used as an operational convenience to extend Surveillance intervals. While up to 24 hours or the limit of the specified Frequency is provided to perform the missed Surveillance, it is expected that the missed Surveillance will be performed at the first reasonable opportunity. The determination of the first reasonable opportunity should include consideration of the impact on plant risk (from delaying the Surveillance as well as any plant configuration changes required or shutting the plant down to perform the Surveillance) and impact on any analysis assumptions, in addition to unit conditions, planning, availability of personnel, and the time required to perform the Surveillance. This risk impact should be managed through the program in place to implement 10 CFR 50.65(a)(4) and its implementation guidance, NRC Regulatory Guide 1.182, 'Assessing and Managing Risk Before Maintenance Activities at Nuclear Power Plants.'

BASES

SR 3.0.3 (continued)

This Regulatory Guide addresses consideration of temporary and aggregate risk impacts, determination of risk management action thresholds, and risk management action up to and including plant shutdown. The missed Surveillance should be treated as an emergent condition as discussed in the Regulatory Guide. The risk evaluation may use quantitative, qualitative, or blended methods. The degree of depth and rigor of the evaluation should be commensurate with the importance of the component. Missed Surveillances for important components should be analyzed quantitatively. If the results of the risk evaluation determine the risk increase is significant, this evaluation should be used to determine the safest course of action. All missed Surveillances will be placed in the licensee's Corrective Action Program.

If a Surveillance is not completed within the allowed delay period, then the equipment is considered inoperable or the variable is considered outside the specified limits and the Completion Times of the Required Actions for the applicable LCO Conditions begin immediately upon expiration of the delay period. If a Surveillance is failed within the delay period, then the equipment is inoperable, or the variable is outside the specified limits and the Completion Times of the Required Actions for the applicable LCO Conditions begin immediately upon the failure of the Surveillance.

Completion of the Surveillance within the delay period allowed by this Specification, or within the Completion Time of the ACTIONS, restores compliance with SR 3.0.1.

SR 3.0.4

SR 3.0.4 establishes the requirement that all applicable SRs must be met before entry into a MODE or other specified condition in the Applicability.

This Specification ensures that system and component OPERABILITY requirements and variable limits are met before entry into MODES or other specified conditions in the Applicability for which these systems and components ensure safe operation of the unit.

The provisions of this Specification should not be interpreted as endorsing the failure to exercise the good practice of restoring systems or components to OPERABLE status before entering an associated MODE or other specified condition in the Applicability.

However, in certain circumstances, failing to meet an SR will not result in SR 3.0.4 restricting a MODE change or other specified condition change.

BASES

SR 3.0.4 (continued)

When a system, subsystem, division, component, device, or variable is inoperable or outside its specified limits, the associated SR(s) are not required to be performed per SR 3.0.1, which states that Surveillances do not have to be performed on inoperable equipment. When equipment is inoperable, SR 3.0.4 does not apply to the associated SR(s) since the requirement for the SR(s) to be performed is removed. Therefore, failing to perform the Surveillance(s) within the specified Frequency, on equipment that is inoperable, does not result in an SR 3.0.4 restriction to changing MODES or other specified conditions of the Applicability. However, since the LCO is not met in this instance, LCO 3.0.4 will govern any restrictions that may (or may not) apply to MODE or other specified condition changes.

The provisions of SR 3.0.4 shall not prevent changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS. In addition, the provisions of SR 3.0.4 shall not prevent changes in MODES or other specified conditions in the Applicability that result from any unit shutdown.

The precise requirements for performance of SRs are specified such that exceptions to SR 3.0.4 are not necessary. The specific time frames and conditions necessary for meeting the SRs are specified in the Frequency, in the Surveillance, or both. This allows performance of Surveillances when the prerequisite condition(s) specified in a Surveillance procedure require entry into the MODE or other specified condition in the Applicability of the associated LCO prior to the performance or completion of a Surveillance. A Surveillance that could not be performed until after entering the LCO Applicability would have its Frequency specified such that it is not "due" until the specific conditions needed are met. Alternately, the Surveillance may be stated in the form of a Note as not required (to be met or performed) until a particular event, condition, or time has been reached. Further discussion of the specific formats of SRs' annotation is found in Section 1.4, Frequency.

SR 3.0.4 is only applicable when entering MODE 3 from MODE 4, MODE 2 from MODE 3 or 4, or MODE 1 from MODE 2. Furthermore, SR 3.0.4 is applicable when entering any other specified condition in the Applicability only while operating in MODE 1, 2, or 3. The requirements of SR 3.0.4 do not apply in MODES 4 and 5, or in other specified conditions of the Applicability (unless in MODE 1, 2, or 3) because the ACTIONS of individual Specifications sufficiently define the remedial measures to be taken.

BASES

SURVEILLANCE REQUIREMENTS

The four SRs of this LCO are modified by a Note stating that during a single control rod scram time surveillance, the CRD pumps shall be isolated from the associated scram accumulator. With the CRD pump isolated, (i.e., charging valve closed) the influence of the CRD pump head does not affect the single control rod scram times. During a full core scram, the CRD pump head would be seen by all control rods and would have a negligible effect on the scram insertion times.

SR 3.1.4.1

The scram reactivity used in DBA and transient analyses is based on an assumed control rod scram time. Measurement of the scram times with reactor steam dome pressure ≥ 800 psig demonstrates acceptable scram times for the transients analyzed in References 3 and 4.

Maximum scram insertion times occur at a reactor steam dome pressure of approximately 800 psig because of the competing effects of reactor steam dome pressure and stored accumulator energy. Therefore, demonstration of adequate scram times at reactor steam dome pressure ≥ 800 psig ensures that the measured scram times will be within the specified limits at higher pressures. Limits are specified as a function of reactor pressure to account for the sensitivity of the scram insertion times with pressure and to allow a range of pressures over which scram time testing can be performed. To ensure that scram time testing is performed within a reasonable time following a shutdown duration of ≥ 120 days, control rods are required to be tested before exceeding 40% RTP following the shutdown. This Frequency is acceptable considering the additional surveillances performed for control rod OPERABILITY, the frequent verification of adequate accumulator pressure, and the required testing of control rods affected by fuel movement within the associated core cell and by work on control rods or the CRD System.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.1.4.2

Additional testing of a sample of control rods is required to verify the continued performance of the scram function during the cycle. A representative sample contains at least 10% of the control rods. The sample remains representative if no more than 7.5% of the control rods in the sample tested are determined to be "slow." With more than 7.5% of the sample declared to be "slow" per the criteria in Table 3.1.4-1, additional control rods are tested until this 7.5% criterion (i.e., 7.5% of the entire sample size) is satisfied, or until the total number of "slow" control rods (throughout the core, from all surveillances) exceeds the LCO limit. For planned testing, the control rods selected for the sample should be different for each test. Data from inadvertent scrams should be used whenever possible to avoid unnecessary testing at power, even if the control rods with data may have been previously tested in a sample. The 200 day Frequency is based on operating experience that has shown control rod scram times do not significantly change over an operating cycle. This Frequency is also reasonable based on the additional Surveillances done on the CRDs at more frequent intervals in accordance with LCO 3.1.3 and LCO 3.1.5, "Control Rod Scram Accumulators."

SR 3.1.4.3

When work that could affect the scram insertion time is performed on a control rod or the CRD System, testing must be done to demonstrate that each affected control rod retains adequate scram performance over the range of applicable reactor pressures from zero to the maximum permissible pressure. The scram testing must be performed once before declaring the control rod OPERABLE. The required scram time testing must demonstrate the affected control rod is still within acceptable limits. The limits for reactor pressures $<$ 800 psig are found in the Technical Requirements Manual (Ref. 8) and are established based on a high probability of meeting the acceptance criteria at reactor pressures $\geq$ 800 psig. Limits for $\geq$ 800 psig are found in Table 3.1.4-1. If testing demonstrates the affected control rod does not meet these limits, but is within the 7-second limit of Table 3.1.4-1, Note 2, the control rod can be declared OPERABLE and "slow."

BASES

SURVEILLANCE REQUIREMENTS (continued)

Specific examples of work that could affect the scram times are (but are not limited to) the following: removal of any CRD for maintenance or modification; replacement of a control rod; and maintenance or modification of a scram solenoid pilot valve, scram valve, accumulator, isolation valve or check valve in the piping required for scram.

The Frequency of once prior to declaring the affected control rod OPERABLE is acceptable because of the capability to test the control rod over a range of operating conditions and the more frequent surveillances on other aspects of control rod OPERABILITY.

SR 3.1.4.4

When work that could affect the scram insertion time is performed on a control rod or CRD System, or when fuel movement within the reactor pressure vessel occurs, testing must be done to demonstrate each affected control rod is still within the limits of Table 3.1.4-1 with the reactor steam dome pressure $\geq$ 800 psig. Where work has been performed at high reactor pressure, the requirements of SR 3.1.4.3 and SR 3.1.4.4 can be satisfied with one test. For a control rod affected by work performed while shut down, however, a zero pressure and high pressure test may be required. This testing ensures that, prior to withdrawing the control rod for continued operation, the control rod scram performance is acceptable for operating reactor pressure conditions. Alternatively, a control rod scram test during hydrostatic pressure testing could also satisfy both criteria. When fuel movement within the reactor pressure vessel occurs, only those control rods associated with the core cells affected by the fuel movement are required to be scram time tested. During a routine refueling outage, it is expected that all control rods will be affected.

The Frequency of once prior to exceeding 40% RTP is acceptable because of the capability to test the control rod over a range of operating conditions and the more frequent surveillances on other aspects of control rod OPERABILITY.

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.1 AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)

BASES

BACKGROUND

The APLHGR is a measure of the average LHGR of all the fuel rods in a fuel assembly at any axial location. Limits on the APLHGR are specified to ensure that the fuel design limits identified in Reference 1 are not exceeded during abnormal operational transients and that the peak cladding temperature (PCT) during the postulated design basis loss of coolant accident (LOCA) does not exceed the limits specified in 10 CFR 50.46.

APPLICABLE SAFETY ANALYSES

The analytical methods and assumptions used in evaluating the fuel design limits are presented in References 1 and 2. The analytical methods and assumptions used in evaluating Design Basis Accidents (DBAs), abnormal operational transients, and normal operation that determine the APLHGR limits are presented in References 1, 2, 3, 4, 5, and 6.

APLHGR limits are equivalent to the LHGR limit for each fuel rod divided by the local peaking factor of the fuel assembly. APLHGR limits are developed as a function of exposure and the various operating core flow and power states to ensure adherence to fuel design limits during the limiting abnormal operational transients (Refs. 5 and 6). Flow dependent APLHGR limits are determined using the three dimensional BWR simulator code (Ref. 7) to analyze slow flow runout transients. The flow dependent multiplier, $MAPFAC_f$, is dependent on the maximum core flow runout capability. The maximum runout flow is dependent on the existing setting of the core flow limiter in the Recirculation Flow Control System.

Based on analyses of limiting plant transients (other than core flow increases) over a range of power and flow conditions, power dependent multipliers, $MAPFAC_p$, are also

BASES

SURVEILLANCE REQUIREMENTS (continued)

which the most recent IRM calibration was performed will be mechanically blocked. Overlap between SRMs and IRMs similarly exists when, prior to withdrawing the SRMs from the fully inserted position, all operable IRM channels shall be on scale.

As noted, SR 3.3.1.1.6 is only required to be met during entry into MODE 2 from MODE 1. That is, after the overlap requirement has been met and indication has transitioned to the IRMs, maintaining overlap is not required (APRMs may be reading downscale once in MODE 2).

If overlap for a group of channels is not demonstrated (e.g., IRM/APRM overlap), the reason for the failure of the Surveillance should be determined and the appropriate channel(s) declared inoperable. Only those appropriate channels that are required in the current MODE or condition should be declared inoperable.

A Frequency of 7 days is reasonable based on engineering judgment and the reliability of the IRMs and APRMs.

SR 3.3.1.1.7

This SR ensures that the total loop drive flow signals from the flow units used to vary the setpoint is appropriately compared to a valid core flow signal to verify the flow signal trip setpoint and, therefore, the APRM Function accurately reflects the required setpoint as a function of flow. If the flow unit signal is not within the appropriate flow limit, the affected APRMs that receive an input from the inoperable flow unit must be declared inoperable.

The Frequency of 31 days is based on engineering judgment, operating experience, and the reliability of this instrumentation.

SR 3.3.1.1.8

LPRM gain settings are determined from the local flux profiles measured by the Traversing Incore Probe (TIP) System. When the measured local flux profile is unavailable, the predicted LPRM reading may be used. This establishes the relative local flux profile for appropriate representative input to the APRM System. The 1000 MWD/T Frequency is based on operating experience with LPRM sensitivity changes.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.1.1.9 and SR 3.3.1.1.11

A CHANNEL FUNCTIONAL TEST is performed on each required channel to ensure that the channel will perform the intended function. 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 non-Technical Specifications tests at least once per refueling interval with applicable extensions. Any setpoint adjustment shall be consistent with the assumptions of the current plant specific setpoint methodology. The 92 day Frequency of SR 3.3.1.1.9 is based on the reliability analysis of Reference 10.

The 18 month Frequency of SR 3.3.1.1.11 is based on the need to perform some of the surveillance procedures which satisfy this SR under the conditions that apply during a plant outage and the potential for an unplanned transient if those particular procedures were performed with the reactor at power. Testing of Function 10 requires placing the mode switch in "Shutdown". Operating experience has shown that these components usually pass the Surveillance when performed at the 18 month Frequency.

SR 3.3.1.1.10 and SR 3.3.1.1.12

A CHANNEL CALIBRATION is a complete check of the instrument loop and the sensor. This test verifies that the channel responds to the measured parameter within the necessary range and accuracy. CHANNEL CALIBRATION leaves the channel adjusted to account for instrument drifts between successive calibrations consistent with the plant specific setpoint methodology. Physical inspection of the position switches is performed in conjunction with SR 3.3.1.1.12 for Functions 5, 7.b, and 8 to ensure that the switches are not corroded or otherwise degraded.

Note 1 of SR 3.3.1.1.10 and SR 3.3.1.1.12 states that neutron detectors are excluded from CHANNEL CALIBRATION because they are passive devices, with minimal drift, and because of the difficulty of simulating a meaningful signal. Changes in neutron detector sensitivity are compensated for by performing the 7 day calorimetric calibration (SR 3.3.1.1.2) and the 1000 MWD/T LPRM calibration against the TIPs

BASES

SURVEILLANCE REQUIREMENTS (continued)

(SR 3.3.1.1.8). Note 1 of SR 3.3.1.1.10 states that recirculation loop flow transmitters are excluded from CHANNEL CALIBRATION. This exclusion is based on calculation results and site-specific instrument setpoint drift data, which alternately supports an 18-month calibration interval for the recirculation loop flow transmitters. As such, the flow transmitters are calibrated on an 18-month frequency as required by SR 3.3.1.1.12 for Function 2b.

A second Note to SR 3.3.1.1.12 is provided that requires the APRM and IRM SRs to be performed within 12 hours of entering MODE 2 from MODE 1. Testing of the MODE 2 APRM and IRM Functions cannot be performed in MODE 1 without utilizing jumpers, lifted leads, or movable links. This Note allows entry into MODE 2 from MODE 1 if the associated Frequency is not met per SR 3.0.2. Twelve hours is based on operating experience and in consideration of providing a reasonable time in which to complete the SR.

The Frequency of SR 3.3.1.1.10 is based upon the assumption of a 184 day calibration interval in the determination of the magnitude of equipment drift in the setpoint analysis. The Frequency of SR 3.3.1.1.12 is based upon the assumption of an 18 month calibration interval in the determination of the magnitude of equipment drift in the setpoint analysis.

SR 3.3.1.1.13

The LOGIC SYSTEM FUNCTIONAL TEST demonstrates the OPERABILITY of the required trip logic for a specific channel. The functional testing of control rods (LCO 3.1.3), and SDV vent and drain valves (LCO 3.1.8), overlaps this Surveillance to provide complete testing of the assumed safety function.

The 18 month Frequency is based on the need to perform some of the surveillance procedures which satisfy this SR under the conditions that apply during a plant outage and the potential for an unplanned transient if those particular procedures were performed with the reactor at power. Operating experience has shown that these components usually pass the Surveillance when performed at the 18 month Frequency.

BASES

SURVEILLANCE REQUIREMENTS (continued)

CHANNEL CHECK), that ensure proper functioning between CHANNEL FUNCTIONAL TESTS.

SR 3.3.1.2.6 is required in MODE 2 with IRMs on Range 2 or below, and in MODES 3 and 4. Since core reactivity changes do not normally take place in MODES 3 and 4, and core reactivity changes are due only to control rod movement in MODE 2, the Frequency has been extended from 7 days to 31 days. The 31 day Frequency is based on operating experience and on other Surveillances (such as CHANNEL CHECK) that ensure proper functioning between CHANNEL FUNCTIONAL TESTS.

Verification of the signal to noise ratio also ensures that the detectors are inserted to an acceptable operating level. In a fully withdrawn condition, the detectors are sufficiently removed from the fueled region of the core to essentially eliminate neutrons from reaching the detector. Any count rate obtained while the detectors are fully withdrawn is assumed to be "noise" only. An alternative to fully withdrawing the detector is to configure the assembly cabling such that only the noise signal is observed.

The Note to SR 3.3.1.2.6 allows the Surveillance to be delayed until entry into the specified condition of the Applicability (THERMAL POWER decreased to IRM Range 2 or below). The SR must be performed within 12 hours after IRMs are on Range 2 or below. The allowance to enter the Applicability with the 31 day Frequency not met is reasonable, based on the limited time of 12 hours allowed after entering the Applicability and the inability to perform the Surveillance while at higher power levels. Although the Surveillance could be performed while on IRM Range 3, the plant would not be expected to maintain steady state operation at this power level. In this event, the 12 hour Frequency is reasonable, based on the SRMs being otherwise verified to be OPERABLE (i.e., satisfactorily performing the CHANNEL CHECK) and the time required to perform the Surveillances.

SR 3.3.1.2.7

Performance of a CHANNEL CALIBRATION at a Frequency of 18 months verifies the performance of the SRM detectors and associated circuitry. The Frequency considers the plant conditions required to perform the test, the ease of performing the test, and the likelihood of a change in the system or component status. The neutron detectors are excluded from the CHANNEL CALIBRATION (Note 1) because they cannot readily be adjusted. The detectors are fission chambers that are

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

1. Rod Block Monitor (continued)

Nominal trip setpoints are specified in the setpoint calculations. The setpoint calculations are performed using methodology described in NEDC-31336P-A, "General Electric Instrument Setpoint Methodology," dated September 1996. The nominal setpoints are selected to ensure that the setpoints do not exceed the Allowable Values between successive CHANNEL CALIBRATIONS. Operation with a trip setpoint less conservative than the nominal trip setpoint, but within its Allowable Value, is acceptable. Trip setpoints are those predetermined values of output at which an action should take place. The setpoints are compared to the actual process parameter (e.g., reactor power), and when the measured output value of the process parameter exceeds the setpoint, the associated device changes state. The analytic limits are derived from the limiting values of the process parameters obtained from the safety analysis. The Allowable Values are derived from the analytic limits, corrected for calibration, process, and some of the instrument errors. The trip setpoints are then determined accounting for the remaining instrument errors (e.g., drift). The trip setpoints derived in this manner provide adequate protection because instrumentation uncertainties, process effects, calibration tolerances, instrument drift, and severe environment errors (for channels that must function in harsh environments as defined by 10 CFR 50.49) are accounted for.

The RBM is assumed to mitigate the consequences of an RWE event when operating ≥ 30% RTP and a peripheral control rod is not selected. Below this power level or if a peripheral control rod is selected, the consequences of an RWE event will not exceed the MCPR SL and, therefore, the RBM is not required to be OPERABLE (Ref. 3). When operating < 90% RTP, analyses (Ref. 3) have shown that with an initial MCPR ≥ 1.70, no RWE event will result in exceeding the MCPR SL. Also, the analyses demonstrate that when operating at ≥ 90% RTP with MCPR ≥ 1.40, no RWE event will result in exceeding the MCPR SL (Ref. 3). Therefore, under these conditions, the RBM is also not required to be OPERABLE.

2. Rod Worth Minimizer

The RWM is a backup to operator control of the rod sequences. The RWM enforces the banked position withdrawal sequence (BPWS) by alerting the operator when the rod pattern is not in accordance with BPWS. Compliance with BPWS ensures that the initial conditions of the CRDA analysis are not violated.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

The analytical methods and assumptions used in evaluating the CRDA are summarized in References 5 and 6. The BPWS requires that control rods be moved in groups, with all control rods assigned to a specific group required to be within specified banked positions. Requirements that the control rod sequence is in compliance with the BPWS are specified in LCO 3.1.6, "Rod Pattern Control."

When performing a shutdown of the plant, an optional BPWS control rod sequence (Ref. 7) may be used if the coupling of each withdrawn control rod has been confirmed. The rods may be inserted without the need to stop at intermediate positions. When using the Reference 7 control rod insertion sequence for shutdown, the rod worth minimizer may be reprogrammed to enforce the requirements of the improved BPWS control rod insertion, or may be bypassed and the improved BPWS shutdown sequence implemented under the controls in Condition D.

The RWM Function satisfies Criterion 3 of Reference 4.

Since the RWM is a system designed to act as a backup to operator control of the rod sequences, only one channel of the RWM is available and required to be OPERABLE (Ref. 7). Special circumstances provided for in the Required Action of LCO 3.1.3, "Control Rod OPERABILITY," and LCO 3.1.6 may necessitate bypassing the RWM to allow continued operation with inoperable control rods, or to allow correction of a control rod pattern not in compliance with the BPWS. The RWM may be bypassed as required by these conditions, but then it must be considered inoperable and the Required Actions of this LCO followed.

Compliance with the BPWS, and therefore OPERABILITY of the RWM, is required in MODES 1 and 2 when THERMAL POWER is ≤ 10% RTP. When THERMAL POWER is > 10% RTP, there is no possible control rod configuration that results in a control rod worth that could exceed the 280 cal/gm fuel damage limit during a CRDA (Ref. 5). In MODES 3 and 4, all control rods are required to be inserted into the core; therefore, a CRDA cannot occur. In MODE 5, since only a single control rod can be withdrawn from a core cell containing fuel assemblies, adequate SDM ensures that the consequences of a CRDA are acceptable, since the reactor will be subcritical.

BASES

SURVEILLANCE REQUIREMENTS (continued)

is not met per SR 3.0.2. The 1 hour allowance is based on operating experience and in consideration of providing a reasonable time in which to complete the SRs.

The 18 month Frequency is based on the need to perform some of the surveillance procedures which satisfy this SR under the conditions that apply during a plant outage and the potential for an unplanned transient if those particular procedures were performed with the reactor at power. Operating experience has shown these components usually pass the Surveillance when performed at the 18 month Frequency.

SR 3.3.2.1.8

The RWM will only enforce the proper control rod sequence if the rod sequence is properly input into the RWM computer. This SR ensures that the proper sequence is loaded into the RWM so that it can perform its intended function. The Surveillance is performed once prior to declaring RWM OPERABLE following loading of sequence into RWM, since this is when rod sequence input errors are possible.

BASES

SURVEILLANCE REQUIREMENTS (continued)

main turbine valves is included as part of this Surveillance and overlaps the LOGIC SYSTEM FUNCTIONAL TEST to provide complete testing of the assumed safety function. Therefore, if a valve is incapable of operating, the associated instrumentation would also be inoperable. The 18 month Frequency is based on the need to perform some of the surveillance procedures which satisfy this SR under the conditions that apply during a plant outage and the potential for an unplanned transient if those particular procedures were performed with the reactor at power. Operating experience has shown that these components usually pass the Surveillance when performed at the 18 month Frequency.

REFERENCES

1. USAR, Section XIV-5.8.1.

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

3. GENE-770-06-1, "Bases for Changes to Surveillance Test Intervals and Allowed Out-Of-Service Times for Selected Instrumentation Technical Specifications," February 1991.

BASES

SURVEILLANCE REQUIREMENTS (continued)

The Frequency is based upon plant operating experience that demonstrates channel failure is rare.

SR 3.3.3.2.2

SR 3.3.3.2.2 verifies each required Alternate Shutdown System transfer switch and control circuit performs the intended function. This verification is performed from the alternate shutdown panel and locally, as appropriate. Operation of the equipment from the alternate shutdown panel is not necessary. The Surveillance can be satisfied by performance of a continuity check. This will ensure that if the control room becomes inaccessible, the plant can be placed and maintained in a safe shutdown condition from the alternate shutdown panel and the local control stations. However, this Surveillance is not required to be performed only during a plant outage. Operating experience demonstrates that Alternate Shutdown System control channels usually pass the Surveillance when performed at the 18 month Frequency.

SR 3.3.3.2.3

CHANNEL CALIBRATION is a complete check of the instrument loop and the sensor. The test verifies the channel responds to measured parameter values with the necessary range and accuracy.

The 18 month Frequency is based upon operating experience and consistency with the typical industry refueling cycle.

REFERENCES

1. USAR, Section VII-18.0.
2. USAR, Section XIV-5.9.
3. 10 CFR 50.36(c)(2)(ii).

BASES

SURVEILLANCE REQUIREMENTS (continued)

an RRMG field breaker is incapable of operating, the associated instrument channel(s) would be inoperable.

The 18 month Frequency is based on the need to perform some of the surveillance procedures which satisfy this SR under the conditions that apply during a plant outage and the potential for an unplanned transient if those particular procedures 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, Section VII-9.4.4.2.
2. 10 CFR 50.36(c)(2)(ii).
3. GENE-770-06-1, "Bases for Changes To Surveillance Test Intervals and Allowed Out-of-Service Times For Selected Instrumentation Technical Specifications," February 1991.

BASES

BACKGROUND (continued)

in approximately 14 seconds, and will run in standby conditions (rated voltage and speed, with the DG output breaker open). The DGs will only energize their respective Engineered Safety Feature buses if a loss of offsite power occurs. (Refer to Bases for LCO 3.3.8.1.)

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

The actions of the ECCS are explicitly assumed in the safety analyses of References 5, 6, and 7. The ECCS is initiated to preserve the integrity of the fuel cladding by limiting the post LOCA peak cladding temperature to less than the 10 CFR 50.46 limits.

ECCS instrumentation satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii) (Ref. 4). Certain instrumentation Functions are retained for other reasons and are described below in the individual Functions discussion.

The OPERABILITY of the ECCS instrumentation is dependent upon the OPERABILITY of the individual instrumentation channel Functions specified in Table 3.3.5.1-1. Each Function must have a required number of OPERABLE channels, with their setpoints within the specified Allowable Values, where appropriate. The actual setpoint is calibrated consistent with applicable setpoint methodology assumptions. Table 3.3.5.1-1 contains several footnotes. Footnote (a) clarifies that the associated functions are required to be OPERABLE in MODES 4 and 5 only when their supported ECCS are required to be OPERABLE per LCO 3.5.2, ECCS - Shutdown. Footnote (b), is added to show that certain ECCS instrumentation Functions also perform DG initiation.

Allowable Values are specified for each ECCS Function specified in the table. Nominal trip setpoints are specified in the setpoint calculations. The setpoint calculations are performed using methodology described in NEDC-31336P-A, "General Electric Instrument Setpoint Methodology," dated September 1996. The nominal setpoints are selected to ensure that the setpoints do not exceed the Allowable Value between CHANNEL CALIBRATIONS. Operation with a trip setpoint less conservative than the nominal trip setpoint, but within its Allowable Value, is acceptable. A channel is inoperable if its actual trip setpoint is not within its required Allowable Value. Trip setpoints are those predetermined values of output at which an action should take place. The setpoints are compared to the actual

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

Function 1.a signals. The Reactor Vessel Water Level — Low Low Low (Level 1) is one of the Functions assumed to be OPERABLE and capable of initiating the ECCS during the transients analyzed in References 5 and 7. In addition, the Reactor Vessel Water Level — Low Low Low (Level 1) Function is directly assumed in the analysis of the recirculation line break (Ref. 6). The core cooling function of the ECCS, along with the scram action of the Reactor Protection System (RPS), ensures that the fuel peak cladding temperature remains below the limits of 10 CFR 50.46.

Reactor Vessel Water Level — Low Low Low (Level 1) signals are initiated from four level switches 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.

The Reactor Vessel Water Level — Low Low Low (Level 1) Allowable Value is chosen to allow time for the low pressure core flooding systems to activate and provide adequate cooling.

Four channels of Reactor Vessel Water Level — Low Low Low (Level 1) Function are only required to be OPERABLE when the ECCS are required to be OPERABLE to ensure that no single instrument failure can preclude ECCS initiation. Per Footnote (a) to Table 3.3.5.1-1, this ECCS function is only required to be OPERABLE in MODES 4 and 5 whenever the associated ECCS is required to be OPERABLE per LCO 3.5.2. Refer to LCO 3.5.1 and LCO 3.5.2, "ECCS — Shutdown," for Applicability Bases for the low pressure ECCS subsystems; LCO 3.8.1, "AC Sources — Operating"; and LCO 3.8.2, "AC Sources — Shutdown," for Applicability Bases for the DGs.

<u>1.b, 2.b. Drywell Pressure-High</u>

High pressure in the drywell could indicate a break in the reactor coolant pressure boundary (RCPB). The low pressure ECCS and associated DGs are initiated upon receipt of the Drywell Pressure — High Function in order to minimize the possibility of fuel damage. The DGs are initiated from Function 1.b signals. The Drywell Pressure — High Function, along with the Reactor Water Level — Low Low Low (Level 1) Function, is directly assumed in the analysis of the

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

The Allowable Value is low enough to prevent overpressuring the equipment in the low pressure ECCS, but high enough to ensure that the ECCS injection prevents the fuel peak cladding temperature from exceeding the limits of 10 CFR 50.46.

Four channels of Reactor Pressure — Low Function are only required to be OPERABLE when the ECCS is required to be OPERABLE to ensure that no single instrument failure can preclude ECCS initiation. Per Footnote (a) to Table 3.3.5.1-1, this ECCS function is only required to be OPERABLE in MODES 4 and 5 whenever the associated ECCS is required to be OPERABLE per LCO 3.5.2. Refer to LCO 3.5.1 and LCO 3.5.2 for Applicability Bases for the low pressure ECCS subsystems.

<u>1.d, 2.g. Core Spray and Low Pressure Coolant Injection Pump Discharge Flow-Low (Bypass)</u>

The minimum flow instruments are provided to protect the associated low pressure ECCS pump from overheating when the pump is operating and the associated injection valve is not fully open. The minimum flow line valve is opened when low flow is sensed, and the valve is automatically closed when the flow rate is adequate to protect the pump. The LPCI and CS Pump Discharge Flow — Low Functions are assumed to be OPERABLE. The minimum flow valves for CS and LPCI are not required to close to ensure that the low pressure ECCS flows assumed during the transients and accidents analyzed in References 5, 6, and 7 are met. The core cooling function of the ECCS, along with the scram action of the RPS, ensures that the fuel peak cladding temperature remains below the limits of 10 CFR 50.46.

One flow transmitter per CS pump and one differential pressure switch per LPCI subsystem are used to detect the associated subsystems' flow rates. The logic is arranged such that each switch or transmitter causes its associated minimum flow valve to open. The logic will close the minimum flow valve once the closure setpoint is exceeded. The LPCI minimum flow valves are time delayed such that the valves will not open for approximately 3.5 seconds after the switches detect low flow. The time delay is provided to limit reactor vessel inventory loss during the startup of the RHR shutdown cooling mode. The Pump Discharge Flow — Low Allowable Values are high enough to ensure that the pump

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APLICABILITY (continued)

flow rate is sufficient to protect the pump.

Each channel of Pump Discharge Flow — Low Function (two CS channels and four LPCI channels) is only required to be OPERABLE when the associated ECCS is required to be OPERABLE to ensure that no single instrument failure can preclude the ECCS function. Per Footnote (a) to Table 3.3.5.1-1, this ECCS function is only required to be OPERABLE in MODES 4 and 5 whenever the associated ECCS is required to be OPERABLE per LCO 3.5.2. Refer to LCO 3.5.1 and LCO 3.5.2 for Applicability Bases for the low pressure ECCS subsystems.

1.e. Core Spray Pump Start-Time Delay Relay

The purpose of this time delay is to delay the start of the CS pumps to enable sequential loading of the appropriate AC source. This Function is necessary when power is being supplied from the offsite sources or the standby power sources (DG). The CS Pump Start-Time Delay Relays are assumed to be OPERABLE in the accident analyses requiring ECCS initiation. That is, the analyses assume that the pumps will initiate when required and excess loading will not cause failure of the power sources.

There are two Core Spray Pump Start-Time Delay Relays, one for each CS pump. Each time delay relay is dedicated to a single pump start logic, such that a single failure of a Core Spray Pump Start-Time Delay Relay will not result in the failure of more than one CS pump. In this condition, one of the two CS pumps will remain OPERABLE; thus, the single failure criterion is met (i.e., loss of one instrument does not preclude ECCS initiation). The Allowable Value for the Core Spray Pump Start-Time Delay Relays is chosen to be long enough so that the power source will not be overloaded and short enough so that ECCS operation is not degraded.

Each channel of Core Spray Pump Start-Time Delay Relay Function is required to be OPERABLE only when the associated CS subsystem is required to be OPERABLE. Refer to LCO 3.5.1 and LCO 3.5.2 for Applicability Bases for the CS subsystems.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

There are four LPCI Pump Start — Time Delay Relays, one in each of the RHR pump start logic circuits. While each time delay relay is dedicated to a single pump start logic, a single failure of a LPCI Pump Start — Time Delay Relay could result in the failure of the two low pressure ECCS pumps, powered for the same ESF bus, to perform their intended function (e.g., as in the case where both ECCS pumps on one ESF bus start simultaneously due to an inoperable time delay relay). This still leaves four of the six low pressure ECCS pumps OPERABLE; thus, the single failure criterion is met (i.e., loss of one instrument does not preclude ECCS initiation). The Allowable Value for the LPCI Pump Start — Time Delay Relays is chosen to be long enough so that most of the starting transient of the first pump is complete before starting the second pump on the same 4.16 kV emergency bus and short enough so that ECCS operation is not degraded.

Each LPCI Pump Start — Time Delay Relay Function is required to be OPERABLE only when the associated LPCI subsystem is required to be OPERABLE. Per Footnote (a) to Table 3.3.5.1-1, this ECCS function is only required to be OPERABLE in MODES 4 and 5 whenever the associated ECCS is required to be OPERABLE per LCO 3.5.2. Refer to LCO 3.5.1 and LCO 3.5.2 for Applicability Bases for the LPCI subsystems.

<u>High Pressure Coolant Injection (HPCI) System</u>

<u>3.a. Reactor Vessel Water Level-Low Low (Level 2)</u>

Low RPV water level indicates that the capability to cool the fuel may be threatened. Should RPV water level decrease too far, fuel damage could result. Therefore, the HPCI System is initiated at Level 2 to maintain level above fuel zone zero. The Reactor Vessel Water Level — Low Low (Level 2) is one of the Functions assumed to be OPERABLE and capable of initiating HPCI during the transients analyzed in References 5 and 7. Additionally, the Reactor Vessel Water Level — Low Low (Level 2) Function associated with HPCI is directly assumed in the analysis of the recirculation line break (Ref. 6). The core cooling function of the ECCS, along with the scram action of the RPS, ensures that the fuel peak cladding temperature remains below the limits of 10 CFR 50.46.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.5.1.3 and SR 3.3.5.1.4

A CHANNEL CALIBRATION is a complete check of the instrument loop and the sensor. This test verifies the channel responds to the measured parameter within the necessary range and accuracy. CHANNEL CALIBRATION leaves the channel adjusted to account for instrument drifts between successive calibrations consistent with the plant specific setpoint methodology.

The Frequency of SR 3.3.5.1.3 is based upon 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 3.3.5.1.4 is based upon the assumption of an 18 month calibration interval in the determination of the magnitude of equipment drift in the setpoint analysis.

SR 3.3.5.1.5

The LOGIC SYSTEM FUNCTIONAL TEST demonstrates the OPERABILITY of the required initiation logic and simulated automatic actuation for a specific channel. The system functional testing performed in LCO 3.5.1, LCO 3.5.2, LCO 3.8.1, and LCO 3.8.2 overlaps this Surveillance to complete testing of the assumed safety function.

The 18 month Frequency is based on the need to perform some of the surveillance procedures which satisfy this SR under the conditions that apply during a plant outage and the potential for an unplanned transient if those particular procedures were performed with the reactor at power. Operating experience has shown that these components usually pass the Surveillance when performed at the 18 month Frequency.

BASES

SURVEILLANCE REQUIREMENTS (continued)

The Frequency of SR 3.3.5.2.3 is based upon 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 3.3.5.2.4 is based upon the assumption of an 18 month calibration interval in the determination of the magnitude of equipment drift in the setpoint analysis.

SR 3.3.5.2.5

The LOGIC SYSTEM FUNCTIONAL TEST demonstrates the OPERABILITY of the required initiation logic for a specific channel. The system functional testing performed in LCO 3.5.3 overlaps this Surveillance to provide complete testing of the safety function. Simulated automatic actuation is performed each operating cycle.

The 18 month Frequency is based on the need to perform some of the surveillance procedures which satisfy this SR under the conditions that apply during a plant outage and the potential for an unplanned transient if those particular procedures were performed with the reactor at power. Operating experience has shown that these components usually pass the Surveillance when performed at the 18 month Frequency.

REFERENCES

1. 10 CFR 50.36(c)(2)(ii).

2. GENE-770-06-2, "Addendum to Bases for Changes to Surveillance Test Intervals and Allowed Out-of-Service Times for Selected Instrumentation Technical Specifications," February 1991.

BACKGROUND (continued)

1. Main Steam Line Isolation

Most MSL Isolation Functions receive inputs from four channels. The outputs from these channels are combined in a one-out-of-two taken twice logic to initiate isolation of the Group I isolation valves (MSIVs and MSL drains). To initiate a Group I isolation valve closure, both trip system logics must be tripped.

The exceptions to this arrangement are the Main Steam Line Flow — High Function and Main Steam Tunnel Temperature-High Functions. The Main Steam Line Flow — High Function uses 16 flow channels, four for each steam line. One channel from each steam line inputs to one of the four trip strings. Two trip strings make up each trip system and both trip systems must trip to cause an MSL isolation. Each trip string has four inputs (one per MSL), any one of which will trip the trip string. The trip strings are arranged in a one-out-of-two taken twice logic. This is effectively a one-out-of-eight taken twice logic arrangement to initiate a Group I isolation.

The Main Steam Tunnel Temperature-High Function receives input from 16 temperature switches located in the steam tunnel. These switches are physically located along and in the vicinity of the steam lines in groups of eight (8). There are two locations in the steam tunnel (upper/east and lower/west). For each location, four of the eight switches input into trip system A, the other four into trip system B. The four switches per location are electrically connected in series with switches in other locations and with normally energized trip relays. Any one switch tripping in its trip system plus any one switch tripping in the other trip system will result in isolation of the MSIVs and MSL drains. For purposes of this specification, each temperature switch is considered a "channel".

2. Primary Containment Isolation

Most Primary Containment Isolation Functions receive inputs from four channels. The outputs from these channels are

BACKGROUND (continued)

arranged into two one-out-of-two taken twice trip system logics. One trip system logic initiates isolation of all inboard primary containment isolation valves, while the other trip system logic initiates isolation of all outboard primary containment isolation valves. Each logic closes one of the two valves on each penetration, so that operation of either logic isolates the penetration.

The exception to this arrangement is the Main Steam Line Radiation — High Function. This Function has four channels, whose outputs are arranged in two, two-out-of-two trip system logics for the recirculation sample valves, and in one, one-out-of-two taken twice trip system logic for the mechanical vacuum pump and associated isolation valves. Each of the recirculation sample valve logics isolates one of the two valves. The single mechanical vacuum pump logic must actuate to trip both mechanical vacuum pumps and isolate the associated valves.

The valves isolated by each of the Primary Containment Isolation Functions are listed in Reference 1.

3. 4. High Pressure Coolant Injection System Isolation and Reactor Core Isolation Cooling System Isolation

The Steam Line Flow-High Functions that isolate HPCI and RCIC receive input from two channels, with each channel comprising one trip system using a one-out-of-one logic. Each of the two trip system logics in each isolation group (HPCI and RCIC) is connected to one of the two valves on each associated penetration. Each HPCI and RCIC steam Line Flow-High Channel has a time delay relay to prevent isolation due to flow transients during startup.

The HPCI and RCIC Isolation Functions for Steam Supply Pressure-Low receive inputs from four channels. The outputs from these channels are combined in two trip system logics, each with two-out-of-two logic to initiate isolation of the associated valves. One trip system logic isolates the inboard valve and the other trip system logic isolates the outboard valve.

BACKGROUND (continued)

The HPCI Steam Line Space Temperature-High Function receives input from 32 bimetallic temperature switches physically located along and in the vicinity of the HPCI steam line. Additionally, 8 temperature switches located along and in the vicinity of the RHR steam condensing mode steam lines input into this Function. These 40 switches are located in groups of eight (8). The 32 HPCI steam line switches cover four locations; RHR injection valve room, torus area west, SW quadrant, and the HPCI pump room. The 8 RHR steam condensing line switches are located in torus area NW. For each location, four switches input into trip system A, the other four switches input to trip system B. Each set of four switches is arranged in a one-of-two taken twice trip system logic. One trip system logic isolates the HPCI steam line inboard isolation valves and the other trip system logic isolates the HPCI steam line outboard valves. For purposes of this specification, each temperature switch is considered a "channel".

The RCIC Steam Line Space Temperature-High Function receives input from 16 bimetallic temperature switches located along and in the vicinity of the RCIC steam line; 8 switches are located in the torus area NE, the remaining 8 are located in the NE quadrant RCIC pump room. For each location, four switches input to trip system A, the other four switches input to trip system B. Each set of four switches is arranged in a one-out-of-two taken twice trip system logic. One trip system logic isolates the RCIC Steam Line Inboard Isolation Valve, and the other trip system logic isolates the RCIC Steam Line Outboard Isolation Valve. For purposes of this specification, each temperature switch is considered a "channel".

The HPCI and RCIC Steam Line Flow-High Functions, Steam Supply Pressure-Low Functions, and Steam Line Space Temperature-High Functions isolate the associated steam supply. The Functions associated with HPCI close the HPCI pump suction valve from the suppression pool (if the ECST suction valve is open), close the HPCI turbine exhaust line

BACKGROUND (continued)

drain pot drain valves, and cause a HPCI turbine trip which closes the HPCI minimum flow valve. The Functions associated with RCIC cause a RCIC turbine trip which closes the RCIC minimum flow valve.

5. Reactor Water Cleanup System Isolation

The Reactor Vessel Water Level — Low Low (Level 2) Isolation Function receives input from four reactor vessel water level channels. The outputs from the reactor vessel water level channels are connected into two one-out-of-two taken twice trip system logics. The RWCU Flow — High Function receives input from two channels, each channel outputs to one trip system logic using a one-out-of-one logic, with one logic tripping the inboard RWCU isolation valve and one logic tripping the outboard RWCU isolation valve. The RWCU System Space Temperature-High Function receives input from 48 bimetallic temperature switches. These switches are physically located along and in the vicinity of the RWCU system high temperature piping in groups of eight (8). Thus, there are six (6) locations; RWCU HX room NW (RWCU supply line), RWCU pump rooms (2 locations), RWCU HX room (pump discharge line to Regenerative HX), torus area south, and torus area east. For each location, four switches input into trip system A, the other four switches input into trip system B. Each set of four switches is arranged in a one-out-of-two taken twice logic in series with a normally deenergized trip relay. Actuation of the correct combination of two switches will initiate the corresponding trip system logic. Trip system logic A isolates the RWCU supply line inboard isolation valve, and trip system logic B isolates the RWCU supply line outboard isolation valve. For purposes of this specification, each temperature switch is considered a "channel".

The SLC System Isolation Function receives input from two channels (one channel in each trip system), arranged in a one-out-of-one logic. A channel consists of one of the two control room SLC pump start switches which inputs directly into one of the two RWCU isolation trip system logics. Placing the SLC Pump A control switch to "Start" will isolate the RWCU inboard isolation valve. Placing the

BACKGROUND (continued)

SLC Pump B control switch to "Start" will isolate the RWCU outboard isolation valve.

6. Shutdown Cooling System Isolation

The Reactor Vessel Water Level — Low (Level 3) Function receives input from four reactor vessel water level channels. The outputs from the reactor vessel water level channels are connected to two one-out-of-two taken twice trip system logics. Each of the two trip system logics is connected to one of the two valves on the RHR shutdown cooling pump suction penetration and one of the two inboard LPCI injection valves if in the shutdown cooling mode. The Reactor Vessel Pressure — High Function receives input from two channels, with each channel in one trip system using a one-out-of-one logic. Each of the two trip system logics is connected to one of the two valves on the RHR shutdown cooling pump suction penetration.

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

The isolation signals generated by the primary containment isolation instrumentation are implicitly assumed in the safety analyses of Reference 2 to initiate closure of valves to limit offsite doses. Refer to LCO 3.6.1.3, "Primary Containment Isolation Valves (PCIVs)," Applicable Safety Analyses Bases for more detail of the safety analyses.

Primary containment isolation instrumentation satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii) (Ref. 3). Certain instrumentation Functions are retained for other reasons and are described below in the individual Functions discussion.

The OPERABILITY of the primary containment instrumentation is dependent on the OPERABILITY of the individual instrumentation channel Functions specified in Table 3.3.6.1-1. Each Function must have a required number

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

Primary Containment Isolation

2.a. Reactor Vessel Water Level - Low (Level 3)

Low RPV water level indicates that the capability to cool the fuel may be threatened. The valves whose penetrations communicate with the primary containment are isolated to limit the release of fission products. The isolation of the primary containment on Level 3 supports actions to ensure that offsite dose limits of 10 CFR 100 are not exceeded. The Reactor Vessel Water Level — Low (Level 3) Function associated with isolation is implicitly assumed in the USAR analysis as these leakage paths are assumed to be isolated post LOCA.

Reactor Vessel Water Level — Low (Level 3) signals are initiated from four vessel level instrument switches 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 (Level 3) 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 (Level 3) Allowable Value was chosen to be the same as the RPS Level 3 scram Allowable Value (LCO 3.3.1.1), since isolation of these valves is not critical to orderly plant shutdown.

This Function isolates the Group 2, 3, and 6 valves listed in Reference 1.

2.b. Drywell Pressure - High

High drywell pressure can indicate a break in the RCPB inside the primary containment. The isolation of some of the primary containment isolation valves on high drywell pressure supports actions to ensure that offsite dose limits of 10 CFR 100 are not exceeded. The Drywell Pressure — High Function, associated with isolation of the primary containment, is implicitly assumed in the USAR accident analysis as these leakage paths are assumed to be isolated post LOCA.

APPLICABLE SAFETY ANALYSES, LCO, AND APPLICABILITY (continued)

High drywell pressure signals are initiated from four pressure switches that sense the pressure in the drywell. Four channels of Drywell Pressure — High 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 the same as the ECCS Drywell Pressure — High Allowable Value (LCO 3.3.5.1), since this may be indicative of a LOCA inside primary containment.

This Function isolates the Group 2 and 6 valves listed in Reference 1.

2.c. 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. When Reactor Building Exhaust Plenum Radiation — High is detected, primary containment vent and purge valves are isolated to limit the release of fission products.

The Reactor Building Exhaust Plenum Radiation — High signals are initiated from radiation detectors that are located such that they can monitor the flow of gas through the reactor building plenum. The signal from each detector is input to an individual monitor whose trip outputs are assigned to an isolation channel. Four channels of Reactor Building Exhaust Plenum 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 Values are chosen to promptly detect gross failure of the fuel cladding.

These Functions isolate the Group 6 valves listed in Reference 1.

APPLICABLE SAFETY ANALYSES, LCO, AND APPLICABILTY (continued)

peak cladding temperature remains below the limits of 10 CFR 50.46. The Reactor Vessel Water Level — Low Low (Level 2) Function associated with RWCU isolation is not directly assumed in the USAR safety analyses because the RWCU System line break is bounded by breaks of larger systems (recirculation and MSL breaks are more limiting).

Reactor Vessel Water Level — Low Low (Level 2) signals are initiated from four level switches 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 (Level 2) 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 (Level 2) Allowable Value was chosen to be the same as the High Pressure Coolant Injection/Reactor Core Isolation Cooling (HPCI/RCIC) Reactor Vessel Water Level — Low Low (Level 2) Allowable Value (LCO 3.3.5.1and LCO 3.3.5.2), since this could indicate that the capability to cool the fuel may be threatened.

This Function isolates the Group 3 valves, as listed in Reference 1.

Shutdown Cooling System Isolation

6.a. Reactor Pressure - High

The Reactor Pressure — High Function is provided to isolate the shutdown cooling portion of the Residual Heat Removal (RHR) System. This Function is provided only for equipment protection to prevent an intersystem LOCA scenario, and credit for the interlock is not assumed in the accident or transient analysis in the USAR.

The Reactor Pressure — High signals are initiated from two pressure switches that are connected to different taps on a recirculation pump suction line. Two channels of Reactor Pressure — High Function are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function. The Function is only required to be OPERABLE in MODES 1, 2, and 3, since these

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

are the only MODES in which the reactor can be pressurized; thus, equipment protection is needed. The Allowable Value was chosen to be low enough to protect the system equipment from overpressurization.

This Function isolates both RHR shutdown cooling pump suction valves.

6.b. Reactor Vessel Water Level - Low (Level 3)

Low RPV water level indicates that the capability to cool the fuel may be threatened. Should RPV water level decrease too far, fuel damage could result. Therefore, isolation of some reactor vessel interfaces occurs to begin isolating the potential sources of a break. The Reactor Vessel Water Level — Low (Level 3) Function associated with RHR Shutdown Cooling System isolation is not directly assumed in safety analyses because a break of the RHR Shutdown Cooling System is bounded by breaks of the recirculation and MSL. The RHR Shutdown Cooling System isolation on Level 3 supports actions to ensure that the RPV water level does not drop below fuel zone zero during a vessel draindown event caused by a leak (e.g., pipe break or inadvertent valve opening) in the RHR Shutdown Cooling System.

Reactor Vessel Water Level — Low (Level 3) signals are initiated from four level switches 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 (two channels per trip system) of the Reactor Vessel Water Level — Low (Level 3) Function are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function. As noted (footnote (b) to Table 3.3.6.1-1), only one trip system of the Reactor Vessel Water Level — Low (Level 3) Function is required to be OPERABLE in MODES 4 and 5, provided the RHR Shutdown Cooling System integrity is maintained. System integrity is maintained provided the piping is intact and no maintenance is being performed that has the potential for draining the reactor vessel through the system.

The Reactor Vessel Water Level — Low (Level 3) Allowable Value was chosen to be the same as the RPS Reactor Vessel

SURVEILLANCE REQUIREMENTS (continued)

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 some of the surveillance procedures which satisfy this SR under the conditions that apply during a plant outage and the potential for an unplanned transient if those particular procedures were performed with the reactor at power. Operating experience has shown these components usually pass the Surveillance when performed at the 18 month Frequency.

B 3.3 INSTRUMENTATION

B 3.3.6.2 Secondary Containment Isolation Instrumentation

BASES

BACKGROUND

The secondary containment isolation instrumentation automatically initiates closure of appropriate secondary containment isolation valves (SCIVs) and starts the Standby Gas Treatment (SGT) System. The function of these systems, in combination with other accident mitigation systems, is to limit fission product release during and following postulated Design Basis Accidents (DBAs) (Ref. 1). Secondary containment isolation and establishment of vacuum with the SGT System within the required time limits ensures that fission products that leak from primary containment following a DBA, or are released outside primary containment, or are released during certain operations when primary containment is not required to be OPERABLE are maintained within applicable limits.

The isolation instrumentation includes the sensors, relays, and switches that are necessary to cause initiation of secondary containment isolation. Most channels include electrical equipment and/or electronic equipment (e.g., switches or trip units) that compares measured input signals with pre-established setpoints. When the setpoint is exceeded, the channel output relay actuates, which then outputs a secondary containment isolation signal to the isolation logic. Functional diversity is provided by monitoring a wide range of independent parameters. The input parameters to the isolation logic are (1) reactor vessel water level, (2) drywell pressure, and (3) reactor building ventilation exhaust plenum radiation. Redundant sensor input signals from each parameter are provided for initiation of isolation.

The outputs of the channels in a trip system are arranged into two one-out-of-two taken twice trip system logics (each sensor sends a signal to both trip system logics). One trip system logic initiates isolation of one isolation valve (damper) and starts one SGT subsystem while the other trip system logic initiates isolation of the other isolation valve in the penetration and starts the other SGT subsystem. Each logic closes one of the two valves on each penetration, starts one SGT subsystem, and initiates the other logic. Operation of either logic

BASES

BACKGROUND (continued)

isolates the secondary containment and provides for the necessary filtration of fission products.

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

The isolation signals generated by the secondary containment isolation instrumentation are implicitly assumed in the safety analyses of References 1 and 2 to initiate closure of valves and start the SGT System to limit offsite doses.

Refer to LCO 3.6.4.2, "Secondary Containment Isolation Valves (SCIVs)," and LCO 3.6.4.3, "Standby Gas Treatment (SGT) System," Applicable Safety Analyses Bases for more detail of the safety analyses.

The secondary containment isolation instrumentation satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii) (Ref. 3). Certain instrumentation Functions are retained for other reasons and are described below in the individual Functions discussion.

The OPERABILITY of the secondary containment isolation instrumentation is dependent on the OPERABILITY of the individual instrumentation channel Functions. Each Function must have the required number of OPERABLE channels with their setpoints set within the specified Allowable Values, as shown in Table 3.3.6.2-1. The actual setpoint is calibrated consistent with applicable setpoint methodology assumptions. A channel is inoperable if its actual trip setpoint is not within its required Allowable Value.

Allowable Values are specified for each Function specified in the Table. Nominal trip setpoints are specified in the setpoint calculations. The setpoint calculations are performed using methodology described in NEDC-31336P-A, "General Electric Instrument Setpoint Methodology," dated September 1996. The nominal setpoints are selected to ensure that the setpoints do not exceed the Allowable Value between CHANNEL CALIBRATIONS. Operation with a trip setpoint less conservative than the nominal setpoint, but within its Allowable Value, is acceptable.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

the actual water level (variable leg) in the vessel. Four channels of Reactor Vessel Water Level — Low Low (Level 2) Function are available and are required to be OPERABLE in MODES 1, 2, and 3 to ensure that no single instrument failure can preclude the isolation function.

The Reactor Vessel Water Level — Low Low (Level 2) Allowable Value was chosen to be the same as the High Pressure Coolant Injection/ Reactor Core Isolation Cooling (HPCI/RCIC) Reactor Vessel Water Level Low Low (Level 2) Allowable Value (LCO 3.3.5.1 and LCO 3.3.5.2) since this could indicate that the capability to cool the fuel is being threatened).

The Reactor Vessel Water Level — Low Low (Level 2) Function is required to be OPERABLE in MODES 1, 2, and 3 where considerable energy exists in the Reactor Coolant System (RCS); thus, there is a probability of pipe breaks resulting in significant releases of radioactive steam and gas. In MODES 4 and 5, the probability and consequences of these events are low due to the RCS pressure and temperature limitations of these MODES; thus, this Function is not required. In addition, the Function is also required to be OPERABLE during operations with a potential for draining the reactor vessel (OPDRVs) because the capability of isolating potential sources of leakage must be provided to ensure that offsite dose limits are not exceeded if core damage occurs.

2. Drywell Pressure-High

High drywell pressure can indicate a break in the reactor coolant pressure boundary (RCPB). An isolation of the secondary containment and actuation of the SGT System are initiated in order to minimize the potential of an offsite dose release. The isolation on high drywell pressure supports actions to ensure that any offsite releases are within the limits calculated in the safety analysis. The Drywell Pressure — High Function associated with isolation is not assumed in any USAR accident or transient analyses, but will provide an isolation and initiation signal. It is retained for the overall redundancy and diversity of the secondary containment isolation instrumentation as required by the NRC approved licensing basis.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

High drywell pressure signals are initiated from pressure switches that sense the pressure in the drywell. Four 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

APPLICABLE SAFETY ANALYSES, LCO, AND APPLICABILITY (continued)

instrument failure can preclude the isolation function.

The Allowable Values are chosen to promptly detect gross failure of the fuel cladding.

The Reactor Building Ventilation Exhaust Plenum Radiation — High Functions are required to be OPERABLE in MODES 1, 2, and 3 where considerable energy exists; thus, there is a probability of pipe breaks resulting in significant releases of radioactive steam and gas. In MODES 4 and 5, the probability and consequences of these events are low due to the RCS pressure and temperature limitations of these MODES; thus, these Functions are not required. In addition, the Functions are also required to be OPERABLE during OPDRVs, and movement of recently irradiated fuel assemblies in the secondary containment, because the capability of detecting radiation releases due to fuel failures (due to fuel uncovery or dropped fuel assemblies) must be provided to ensure that offsite dose limits are not exceeded. Due to radioactive decay, this Function is only required to isolate secondary containment during fuel handling accidents involving handling recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous 24 hours).

ACTIONS

A Note has been provided to modify the ACTIONS related to secondary containment isolation instrumentation channels. Section 1.3, Completion Times, specifies that once a Condition has been entered, subsequent divisions, subsystems, components, or variables expressed in the Condition, discovered to be inoperable or not within limits, will not result in separate entry into the Condition. Section 1.3 also specifies that Required Actions of the Condition continue to apply for each additional failure, with Completion Times based on initial entry into the Condition. However, the Required Actions for inoperable secondary containment isolation instrumentation channels provide appropriate compensatory measures for separate inoperable channels. As such, a Note has been provided that allows separate Condition entry for each inoperable secondary containment isolation instrumentation channel.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.6.2.3

A CHANNEL CALIBRATION is a complete check of the instrument loop and the sensor. This test verifies the channel responds to the measured parameter within the necessary range and accuracy. CHANNEL CALIBRATION leaves the channel adjusted to account for instrument drifts between successive calibrations consistent with the plant specific setpoint methodology.

The Frequency of SR 3.3.6.2.3 is based on the assumption of an 18 month calibration interval, respectively, in the determination of the magnitude of equipment drift in the setpoint analysis.

SR 3.3.6.2.4

The LOGIC SYSTEM FUNCTIONAL TEST demonstrates the OPERABILITY of the required isolation logic for a specific channel. The system functional testing performed on SCIVs and the SGT System in LCO 3.6.4.2 and LCO 3.6.4.3, respectively, overlaps this Surveillance to provide complete testing of the assumed safety function.

The 18 month Frequency is based on the need to perform some of the surveillance procedures which satisfy this SR under the conditions that apply during a plant outage and the potential for an unplanned transient if those particular procedures were performed with the reactor at power. Operating experience has shown that these components usually pass the Surveillance when performed at the 18 month Frequency.

BASES

SURVEILLANCE REQUIREMENTS (continued)

The 92 day Frequency is based on the reliability analysis of Reference 3.

A portion of the SRV discharge line pressure switch instrument channels are located inside the primary containment. The Note for SR 3.3.6.3.2, "Only required to be performed prior to entering MODE 2 during each scheduled outage > 72 hours when entry is made into primary containment," is based on the location of these instruments and ALARA considerations.

SR 3.3.6.3.4

CHANNEL CALIBRATION is a complete check of the instrument loop and sensor. This test verifies the channel responds to the measured parameter within the necessary range and accuracy. CHANNEL CALIBRATION leaves the channel adjusted to account for instrument drifts between successive calibrations consistent with the plant specific setpoint methodology.

The Frequency of once every 18 months for SR 3.3.6.3.4 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 3.3.6.3.5

The LOGIC SYSTEM FUNCTIONAL TEST demonstrates the OPERABILITY of the required actuation logic for a specified channel. The system functional testing performed in LCO 3.4.3, "Safety/Relief Valves (SRVs) and Safety Valves (SVs)" and LCO 3.6.1.6, "Low-Low Set (LLS) Safety/Relief Valves (SRVs)," for SRVs overlaps this test to provide complete testing of the assumed safety function.

The Frequency of once every 18 months for SR 3.3.6.3.5 is based on the need to perform some of the surveillance procedures which satisfy this SR under the conditions that apply during a plant outage and the potential for an unplanned transient if those particular procedures were performed with the reactor at power. Operating experience has shown these components usually pass the Surveillance when performed at the 18 month Frequency.

B 3.3 INSTRUMENTATION

B 3.3.7.1 Control Room Emergency Filter (CREF) System Instrumentation

BASES

BACKGROUND

The CREF System is designed to provide a radiologically controlled environment to ensure the habitability of the control room for the safety of control room operators under all plant conditions. The instrumentation and controls for the CREF System automatically isolate the normal ventilation intake and initiate action to pressurize the main control room and filter incoming air to minimize the infiltration of radioactive material into the control room environment.

In the event of a loss of coolant accident (LOCA) signal (Reactor Vessel Water Level — Low Low, Level 2 or Drywell Pressure — High) or Reactor Building Ventilation Exhaust Plenum Radiation — High signal, the normal control room inlet supply damper closes and the CREF System is automatically started in the emergency bypass mode. The air drawn in from the outside passes through a high efficiency filter and a charcoal filter in sufficient volume to maintain the control room slightly pressurized with respect to the adjacent areas.

The CREF System instrumentation has two trip systems. Each trip system includes the sensors, relays, and switches necessary to cause initiation of the CREF System. Each trip system receives input from each of the Functions listed above (each sensor sends a signal to both trip systems). The Reactor Vessel Water Level — Low Low, Level 2, Drywell Pressure — High, and Reactor Building Ventilation Exhaust Plenum Radiation — High are each arranged in a one-out-of-two taken twice logic for each trip system. The channels include electronic and electrical equipment (e.g., switches and trip relays) that compares measured input signals with pre-established setpoints. When the setpoint is exceeded, the channel output relay actuates, which then outputs a CREF System initiation signal to the initiation logic.

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

The ability of the CREF System to maintain the habitability of the control room is explicitly assumed for certain accidents as discussed in the USAR safety analyses (Refs. 1, 2, and 3). CREF System operation ensures that the radiation exposure of control room personnel, through the duration of any one of the postulated accidents that assume CREF System operation, does not exceed the limits set by GDC 19 of 10 CFR 50, Appendix A or 10 CFR 50.67 (Fuel Handling Accident only).

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

1. Reactor Vessel Water Level — Low Low (Level 2)

Low reactor pressure vessel (RPV) water level indicates that the capability of cooling the fuel may be threatened. A low reactor vessel water level could indicate a LOCA and will automatically initiate the CREF System, since this could be a precursor to a potential radiation release and subsequent radiation exposure to control room personnel.

Reactor Vessel Water Level — Low Low (Level 2) signals are initiated from level switches 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 (Level 2) Function are available and are required to be OPERABLE in MODES 1, 2, and 3 to ensure that no single instrument failure can preclude CREF System initiation.

The Reactor Vessel Water Level — Low Low (Level 2) Allowable Value was chosen to be the same as the Secondary Containment Isolation Allowable Value (LCO 3.3.6.2) to enable initiation of the CREF System at the earliest indication of a breach in the nuclear system process barrier, yet far enough below normal operational levels to avoid spurious initiation.

The Reactor Vessel Water Level — Low Low (Level 2) Function is required to be OPERABLE in MODES 1, 2, and 3, and during operations with a potential for draining the reactor vessel (OPDRVs) to ensure that the Control Room personnel are protected during a LOCA. In MODES 4 and 5 at times other than OPDRVs, the probability of a vessel draindown event resulting in the release of radioactive material to the environment is minimal. Therefore, this Function is not required in other MODES and specified conditions.

2. Drywell Pressure — High

High drywell pressure can indicate a break in the reactor coolant pressure boundary. A high drywell pressure signal could indicate a LOCA and will automatically initiate the CREF System, since this could be a precursor to a potential radiation release and subsequent radiation exposure to control room personnel.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

Drywell Pressure — High signals are initiated from pressure switches that sense drywell pressure. Four channels of Drywell Pressure — High Function are available and are required to be OPERABLE to ensure that no single instrument failure can preclude performance of the initiation function. The Drywell Pressure — High Allowable Value was chosen to be the same as the ECCS Drywell Pressure — High Function Allowable Value (LCO 3.3.5.1).

The Drywell Pressure — High Function is required to be OPERABLE in MODES 1, 2, and 3 to ensure that control room personnel are protected in the event of a LOCA. In MODES 4 and 5, the Drywell Pressure — High Function is not required since there is insufficient energy in the reactor to pressurize the drywell to the Drywell Pressure — High setpoint.

3. Reactor Building Ventilation Exhaust Plenum Radiation — High

High radiation in the refueling floor area could be the result of a fuel handling accident. A refueling floor high radiation signal will automatically initiate the CREF System, since this radiation release could result in radiation exposure to control room personnel.

The Reactor Building Exhaust Plenum Radiation — High signals are initiated from 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 instrument failure can preclude the CREF System initiation. The Allowable Value was chosen to promptly detect gross failure of the fuel cladding.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

The Reactor Building Ventilation Exhaust Plenum Radiation — High Function is required to be OPERABLE in MODES 1, 2, and 3 and during movement of lately irradiated fuel assemblies in the secondary containment and operations with a potential for draining the reactor vessel (OPDRVs), to ensure control room personnel are protected during a pipe break resulting in significant releases of radioactive steam and gas, fuel handling event, or vessel draindown event. During MODES 4 and 5, when these specified conditions are not in progress (e.g., OPDRVs), the probability of a pipe break resulting in significant releases of radioactive steam and gas or fuel damage is low; thus, the Function is not required. Due to radioactive decay, this Function is only required to initiate the CREF System during fuel handling accidents involving handling lately irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous 7 days). During the movement of lately irradiated fuel, Reactor Building ventilation exhaust flow (provided by either a Reactor Building ventilation exhaust fan or SGT fan) is a required support function.

ACTIONS

A Note has been provided to modify the ACTIONS related to CREF System instrumentation channels. Section 1.3, Completion Times, specifies that once a Condition has been entered, subsequent divisions, subsystems, components, or variables expressed in the Condition, discovered to be inoperable or not within limits, will not result in separate entry into the Condition. Section 1.3 also specifies that Required Actions of the Condition continue to apply for each additional failure, with Completion Times based on initial entry into the Condition. However, the Required Actions for inoperable CREF System instrumentation channels provide appropriate compensatory measures for separate inoperable channels. As such, a Note has been provided that allows separate Condition entry for each inoperable CREF System instrumentation channel.

A.1

Because of the diversity of sensors available to provide isolation signals and the common interface with the Secondary Containment isolation Instrumentation, allowable out of service time of 12 hours for Functions 1

BASES

ACTIONS (continued)

and 2, and 24 hours for Function 3, has been shown to be acceptable (Refs. 5, 6, and 7) to permit restoration of any inoperable channel to OPERABLE status. However, this out of service time is only acceptable provided the associated Function is still maintaining CREF System initiation capability. If the inoperable channel cannot be restored to OPERABLE status within the Completion Time, the channel must be placed in the tripped condition per Required Action A.1. Placing the inoperable channel in trip would conservatively compensate for the inoperability, restore capability to accommodate a single failure in the trip system, and allow operation to continue. Alternately, if it is not desired to place the channel in trip, Condition C must be entered and its Required Actions taken.

B.1

Required Action B.1 is intended to ensure that appropriate actions are taken if multiple, inoperable, untripped channels within the same Function result in a complete loss of CREF System initiation capability. A Function is considered to be maintaining CREF System initiation capability when sufficient channels are OPERABLE or in trip, such that at least one trip system will generate a trip signal from the given Function on a valid signal.

The 1 hour Completion Time is acceptable because it minimizes risk while allowing time for restoration or tripping of channels.

If the CREF System initiation capability cannot be restored within the Completion Time, Condition C must be entered and its Required Actions taken.

C.1

With any Required Action and associated Completion Time of Condition A or B not met, the CREF System must be placed in operation per Required Action C.1 to ensure that control room personnel will be protected in the event of a Design Basis Accident which assumes a CREF System initiation. The method used to place the CREF System in operation must provide for automatically re-initiating the system upon restoration of power following a loss of power to the CREF System.

BASES

ACTIONS (continued)

Alternatively, if it is not desired to start the CREF System, the CREF System must be declared inoperable within 1 hour.

The 1 hour Completion Time is intended to allow the operator time to place the CREF System in operation. The 1 hour completion time is acceptable because it minimizes risk while allowing time for restoration or tripping of channels, for placing the CREF System in operation, or for entering the applicable Conditions and Required Actions for the inoperable CREF System.

SURVEILLANCE REQUIREMENTS

As noted at the beginning of the SRs, the SRs for each CREF System instrumentation Function are located in the SRs column of Table 3.3.7.1-1.

The Surveillances are modified by a Note to indicate that when a channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 6 hours, provided the associated Function maintains CREF System initiation capability. Upon completion of the Surveillance, or expiration of the 6 hour allowance, the channel must be returned to OPERABLE status or the applicable Condition entered and Required Actions taken. This Note is based on the reliability analysis (Refs. 5, 6, and 7) assumption of the average time required to perform channel surveillance. That analysis demonstrated that the 6 hour testing allowance does not significantly reduce the probability that the CREF System will initiate when necessary.

SR 3.3.7.1.1

Performance of the CHANNEL CHECK once every 12 hours ensures that a gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between the instrument channels could be an indication of excessive instrument drift in one of the channels or something even more serious. A CHANNEL CHECK will detect gross channel failure; thus, it is key to verifying the

BASES

SURVEILLANCE REQUIREMENTS (continued)

instrumentation continues to operate properly between each CHANNEL CALIBRATION.

Agreement criteria are determined by the plant staff, based on a combination of the channel instrument uncertainties, including indication and readability. If a channel is outside the criteria, it may be an indication that the instrument has drifted outside its limit.

The Frequency is based upon operating experience that demonstrates channel failure is rare. The CHANNEL CHECK supplements less formal, but more frequent, checks of channel status during normal operational use of the displays associated with channels required by the LCO.

SR 3.3.7.1.2

A CHANNEL FUNCTIONAL TEST is performed on each required channel to ensure that the channel will perform the intended function. 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 non-Technical Specifications tests at least once per refueling interval with applicable extensions.

Any setpoint adjustment shall be consistent with the assumptions of the current plant specific setpoint methodology.

The Frequency of 92 days is based on the reliability analyses of References 5, 6, and 7.

SR 3.3.7.1.3

A CHANNEL CALIBRATION is a complete check of the instrument loop and the sensor. This test verifies the channel responds to the measured parameter within the necessary range and accuracy. CHANNEL CALIBRATION leaves the channel adjusted to account for instrument drifts between successive calibrations consistent with the plant specific setpoint methodology.

BASES

SURVEILLANCE REQUIREMENTS (continued)

The Frequency is based upon the assumption of an 18 month calibration interval in the determination of the magnitude of equipment drift in the setpoint analysis.

SR 3.3.7.1.4

The LOGIC SYSTEM FUNCTIONAL TEST demonstrates the OPERABILITY of the required initiation logic for a specific channel. The system functional testing performed in LCO 3.7.4, "Control Room Emergency Filter (CREF) System," overlaps this Surveillance to provide complete testing of the assumed safety function.

The 18 month Frequency is based on the need to perform some of the surveillance procedures which satisfy this SR under the conditions that apply during a plant outage and the potential for an unplanned transient if those particular procedures 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, Section X-10.4.
2. USAR, Section XIV-6.3.
3. USAR, Section XIV-6.4.
4. 10 CFR 50.36(c)(2)(ii).
5. GENE-770-06-1, "Bases for Changes to Surveillance Test Intervals and Allowed Out-of-Service Times for Selected Instrumentation Technical Specifications," February 1991.
6. NEDC-31677P-A, "Technical Specification Improvement Analysis for BWR Isolation Actuation Instrumentation," July 1990.
7. NEDC-30851P-A Supplement 2, "Technical Specifications Improvement Analysis for BWR Isolation Instrumentation Common to RPS and ECCS Instrumentation," March 1989.

BASES

SURVEILLANCE REQUIREMENTS (continued)

The 18 month Frequency is based on the need to perform some of the surveillance procedures which satisfy this SR under the conditions that apply during a plant outage and the potential for an unplanned transient if those particular procedures 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, Section VIII-4.6.
2. USAR, Chapter XIV.
3. 10 CFR 50.36(c)(2)(ii)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.8.2.2

Performance of a system functional test demonstrates that, with a required system actuation (simulated or actual) signal, the logic of the system will automatically trip open the associated power monitoring assembly. The system functional test shall include actuation of the protective relays, tripping logic, and output circuit breakers. Only one signal per power monitoring assembly is required to be tested. This Surveillance overlaps with the CHANNEL CALIBRATION to provide complete testing of the safety function. The system functional test of the Class 1E circuit breakers is included as part of this test to provide complete testing of the safety function. If the breakers are incapable of operating, the associated electric power monitoring assembly would be inoperable.

The 18 month Frequency is based on the need to perform some of the surveillance procedures which satisfy this SR under the conditions that apply during a plant outage and the potential for an unplanned transient if those particular procedures were performed with the reactor at power. Operating experience has shown that these components usually pass the Surveillance when performed at the 18 month Frequency.

REFERENCES

1. USAR, Section VII-2.3.
2. 10 CFR 50.36(c)(2)(ii).

BASES

APPLICABLE SAFETY ANALYSES (continued)

P/T limits are not derived from any DBA, there are no acceptance limits related to the P/T limits. Rather, the P/T limits are acceptance limits themselves since they preclude operation in an unanalyzed condition.

RCS P/T limits satisfy Criterion 2 of 10 CFR 50.36(c)(2)(ii) (Ref. 8).

LCO

The elements of this LCO are:

a. RCS pressure and temperature (Beltline, Bottom Head, and Upper Vessel) are within the applicable limits of Figure 3.4.9-1 and Figure 3.4.9-2, and heatup or cooldown rates are ≤ 100°F when averaged over a one hour period during RCS heatup, cooldown, and inservice leak and hydrostatic testing (The Adjusted Reference Temperature (ART) beltline region must be determined from Figure 3.4.9-2. During RCS heatup and cooldown operation (i.e., not critical and not performing inservice leak or hydrostatic testing) verify RCS pressure and temperature are within the applicable limits specified in Figure 3.4.9-1. During RCS inservice leak and hydrostatic testing verify RCS pressure and temperature are within the applicable limits specified in Figure 3.4.9-2;

b. The temperature difference between the reactor vessel bottom head coolant and the reactor pressure vessel (RPV) coolant is ≤ 145°F during recirculation pump startup;

c. The temperature difference between the reactor coolant in the respective recirculation loop and in the reactor vessel is ≤ 50°F during recirculation pump startup;

d. RCS pressure and temperature are within the criticality limits specified in Figure 3.4.9-3, prior to achieving criticality; and

e. The reactor vessel flange and the head flange temperatures are > 80°F when tensioning the reactor vessel head bolting studs.

These limits define allowable operating regions and permit a large number of operating cycles while also providing a wide margin to nonductile failure.

BASES

SURVEILLANCE REQUIREMENTS (continued)

reasonable in view of the control room indication available to monitor RCS status. Also, since temperature rate of change limits are specified in hourly increments, 30 minutes permits a reasonable time for assessment and correction of minor deviations.

Surveillance for heatup, cooldown, or inservice leakage and hydrostatic testing may be discontinued when the difference between any two readings taken over a 45 minute period is less than 50°F.

This SR has been modified with a Note that requires this Surveillance to be performed only during system heatup and cooldown operations and inservice leakage and hydrostatic testing. During RCS heatup and cooldown operation (i.e., not critical and not performing inservice leak or hydrostatic testing) verify RCS pressure and temperature are within the applicable limits specified in Figure 3.4.9-1. During RCS inservice leak and hydrostatic testing verify RCS pressure and temperature are within the applicable limits specified in Figure 3.4.9-2.

<u>SR 3.4.9.2</u>

A separate limit is used when the reactor is approaching criticality. Consequently, the RCS pressure and temperature must be verified within the appropriate limits before withdrawing control rods that will make the reactor critical.

Performing the Surveillance within 15 minutes before control rod withdrawal for the purpose of achieving criticality provides adequate assurance that the limits will not be exceeded between the time of the Surveillance and the time of the control rod withdrawal.

<u>SR 3.4.9.3 and SR 3.4.9.4</u>

Differential temperatures within the specified limits ensure that thermal stresses resulting from the startup of an idle recirculation pump will not exceed design allowances. In addition, compliance with these limits ensures that the assumptions of the analysis for the startup of an idle recirculation loop (Ref. 9) are satisfied.

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 some of the surveillance procedures which satisfy this SR under the conditions that apply during a plant outage and the potential for an unplanned transient if those particular procedures 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 some of the surveillance procedures which satisfy this SR under the conditions that apply during a plant outage and the potential for an unplanned transient if those particular procedures 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

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 some of the surveillance procedures which satisfy this SR under the conditions that apply during a plant outage and the potential for an unplanned transient if those particular procedures 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 (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) does not necessarily result in a failure of this SR. The impact of the failure to meet this SR must be evaluated against the Type A and Type 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 $\leq$ 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 $\leq$ 1.0 L_a. At $\leq$ 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)

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. The combined main steam line leakage rate must be $\leq$ 46 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.

BASES

ACTIONS (continued)

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 3.6.1.6.1

A manual actuation of each LLS valve is performed to verify that the valve and solenoids are functioning properly and no blockage exists in the valve discharge line. This can be demonstrated by the response of the turbine control or bypass valve, by a change in the measured steam flow, or by any other method that is suitable to verify steam flow. Adequate reactor steam dome pressure must be available to perform this test to avoid damaging the valve. Adequate pressure at which this test is to be performed is $\geq$ 500 psig (consistent with the recommendations of the vendor). Also, adequate steam flow must be passing through the main turbine or turbine bypass valves to continue to control reactor pressure when the LLS valves divert steam flow upon opening. Adequate steam flow is represented by turbine bypass valves at least 30% open, or total steam flow $\geq 10^6$ lb/hr. The 18 month Frequency was based on the SRV tests required by the ASME Boiler and Pressure Vessel Code, Section XI (Ref. 3). Operating experience has shown that these components usually pass the Surveillance when performed at the 18 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

Since steam pressure is required to perform the Surveillance, however, and steam may not be available during a unit outage, the Surveillance may be performed during the startup following a unit outage. Unit startup is allowed prior to performing the test because valve OPERABILITY and the setpoints for overpressure protection are verified by Reference 3 prior to valve installation. After adequate reactor steam dome pressure and flow are reached, 12 hours is allowed to prepare for and perform the test.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.1.6.2

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

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

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

REFERENCES

1. 10 CFR 50.36(c)(2)(ii).

2. NEDE-22197, Safety Relief Valve Low Low Set System and Lower MSIV Water Level Trip for Cooper Nuclear Station, Unit 1, December 1982.

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

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.1.7.2

Each vacuum breaker must be cycled to ensure that it opens properly to perform its design function and returns to its fully closed position. This ensures that the safety analysis assumptions are valid. The 92 day Frequency of this SR was developed based upon Inservice Testing Program requirements to perform valve testing at least once every 92 days.

SR 3.6.1.7.3

Demonstration of vacuum breaker opening setpoint is necessary to ensure that the safety analysis assumption regarding vacuum breaker full open differential pressure of $\leq$ 0.5 psid is valid. The 18 month Frequency is based on the need to perform some of the surveillance procedures which satisfy this SR under the conditions that apply during a plant outage and the potential for an unplanned transient if those particular procedures were performed with the reactor at power. For this unit, the 18 month Frequency has been shown to be acceptable, based on operating experience, and is further justified because of other Surveillances performed at shorter Frequencies that convey the proper functioning status of each vacuum breaker.

REFERENCES

1. Bodega Bay Preliminary Hazards Summary Report, Appendix I, Docket 50-205, December 28, 1962.
2. USAR, Section V-2.3.6.
3. 10 CFR 50.36(c)(2)(ii).

B 3.6 CONTAINMENT SYSTEMS

B 3.6.4.1 Secondary Containment

BASES

BACKGROUND

The function of the secondary containment is to contain, dilute, and hold up fission products that may leak from primary containment following a Design Basis Accident (DBA) to limit fission product release to the environment. In conjunction with operation of the Standby Gas Treatment (SGT) System and closure of certain valves whose lines penetrate the secondary containment, the secondary containment is designed to reduce the activity level of the fission products released to the environment and to limit fission products that are released during certain operations that take place inside primary containment, when primary containment is not required to be OPERABLE, or that take place outside primary containment.

The secondary containment is a structure that completely encloses the primary containment and those components that may be postulated to contain primary system fluid. This structure forms a control volume that serves to hold up and dilute the fission products. It is possible for the pressure in the control volume to rise relative to the environmental pressure (e.g., due to pump and motor heat load additions). To prevent ground level exfiltration while allowing the secondary containment to be designed as a conventional structure, the secondary containment requires support systems to maintain the control volume pressure at less than the external pressure. Requirements for these systems are specified separately in LCO 3.6.4.2, "Secondary Containment Isolation Valves (SCIVs)," and LCO 3.6.4.3, "Standby Gas Treatment (SGT) System."

APPLICABLE SAFETY ANALYSIS

There are two principal accidents for which credit is taken for secondary containment OPERABILITY. These are a loss of coolant accident (LOCA) (Ref. 1) and a fuel handling accident involving handling recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous 24 hours) inside secondary containment (Ref. 2). The secondary containment performs no active function in response to each of these limiting events; however, its leak tightness is required to ensure that fission products entrapped within the secondary containment structure following secondary containment isolation will be treated by the SGT System prior to discharge to the environment.

BASES

APPLICABLE SAFETY ANALYSES (continued)

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

LCO

An OPERABLE secondary containment provides a control volume into which fission products that leak from primary containment, or are released from the reactor coolant pressure boundary components located in secondary containment, following secondary containment isolation can be processed prior to release to the environment. For the secondary containment to be considered OPERABLE, it must have adequate leak tightness to ensure that the required vacuum can be established and maintained.

APPLICABILITY

In MODES 1, 2, and 3, a LOCA could lead to a fission product release to primary containment that leaks to secondary containment. Therefore, secondary containment OPERABILITY is required during the same operating conditions that require primary containment OPERABILITY.

In MODES 4 and 5, the probability and consequences of the LOCA are reduced due to the pressure and temperature limitations in these MODES. Therefore, maintaining secondary containment OPERABLE is not required in MODE 4 or 5 to ensure a control volume, except for other situations for which significant releases of radioactive material can be postulated, such as during operations with a potential for draining the reactor vessel (OPDRVs), or during movement of recently irradiated fuel assemblies in the secondary containment. Due to radioactive decay, secondary containment is only required to be OPERABLE during fuel handling involving handling recently irradiated fuel (i.e., fuel that has occupied part of critical reactor core within the previous 24 hours).

ACTIONS

A.1

If secondary containment is inoperable, it must be restored to OPERABLE status within 4 hours. The 4 hour Completion Time provides a period of time to correct the problem that is commensurate with the importance of maintaining secondary

BASES

ACTIONS (continued)

containment during MODES 1, 2, and 3. This time period also ensures that the probability of an accident (requiring secondary containment OPERABILITY) occurring during periods where secondary containment is inoperable is minimal.

B.1 and B.2

If secondary 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.

C.1 and C.2

Movement of recently irradiated fuel assemblies in the secondary containment and OPDRVs can be postulated to cause significant fission product release to the secondary containment. In such cases, the secondary containment is the only barrier to release of fission products to the environment. Therefore, movement of recently irradiated fuel assemblies must be immediately suspended if the secondary containment is inoperable.

Suspension of these activities shall not preclude completing an action that involves moving a component to a safe position. Also, action must be immediately initiated to suspend OPDRVs to minimize the probability of a vessel draindown and subsequent potential for fission product release. Actions must continue until OPDRVs are suspended.

Required Action C.1 has been modified by a Note stating that LCO 3.0.3 is not applicable. If moving recently irradiated fuel assemblies while in MODE 4 or 5, LCO 3.0.3 would not specify any action. If moving recently irradiated fuel assemblies while in MODE 1, 2, or 3, the fuel movement is independent of reactor operations. Therefore, in either case, inability to suspend

BASES

ACTIONS

C.1 and C.2 (continued)

movement of recently irradiated fuel assemblies would not be a sufficient reason to require a reactor shutdown.

SURVEILLANCE REQUIREMENTS

SR 3.6.4.1.1

This SR ensures that the secondary containment boundary is sufficiently leak tight to preclude exfiltration under expected wind conditions. Momentary transients on installed instrumentation due to gusty wind conditions are considered acceptable and are not cause for failure to meet this SR. The 24 hour Frequency of this SR was developed based on operating experience related to secondary containment vacuum variations 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 secondary containment vacuum condition.

SR 3.6.4.1.2 and SR 3.6.4.1.3

Verifying that secondary containment equipment hatches and one access door in each access opening are closed ensures that the infiltration of outside air of such a magnitude as to prevent maintaining the desired negative pressure does not occur. Verifying that all such openings are closed provides adequate assurance that exfiltration from the secondary containment will not occur. SR 3.6.4.1.2 also requires equipment hatches to be sealed. In this application, the term "sealed" has no connotation of leak tightness. Maintaining secondary containment OPERABILITY requires verifying one door in the access opening is closed. However, each secondary containment access door is normally kept closed, except when the access opening is being used for normal transient entry and exit or when maintenance is being performed on an access. The 31 day Frequency for these SRs has been shown to be adequate, based on operating experience, and is considered adequate in view of the other indications of door and hatch status that are available to the operator.

B 3.6 CONTAINMENT SYSTEMS

B 3.6.4.2 Secondary Containment Isolation Valves (SCIVs)

BASES

BACKGROUND

The function of the SCIVs, in combination with other accident mitigation systems, is to limit fission product release during and following postulated Design Basis Accidents (DBAs) (Refs. 1 and 2). Secondary containment isolation within the time limits specified for those isolation valves designed to close automatically ensures that fission products that leak from primary containment following a DBA, or that are released during certain operations when primary containment is not required to be OPERABLE or take place outside primary containment, are maintained within the secondary containment boundary.

The OPERABILITY requirements for SCIVs help ensure that an adequate secondary containment boundary is maintained during and after an accident by minimizing potential paths to the environment. These isolation devices consist of either passive devices or active (automatic) devices. Manual valves, de-activated automatic valves secured in their closed position (including check valves with flow through the valve secured), and blind flanges are considered passive devices.

Automatic SCIVs close on a secondary containment isolation signal to establish a boundary for untreated radioactive material within secondary containment following a DBA or other accidents.

Other penetrations are isolated by the use of valves in the closed position or blind flanges.

APPLICABLE SAFETY ANALYSES

The SCIVs must be OPERABLE to ensure the secondary containment barrier to fission product releases is established. The principal accidents for which the secondary containment boundary is required are a loss of coolant accident (Ref. 3) and a fuel handling accident involving handling recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous 24 hours) (Ref. 4). The secondary containment performs no active function in response to either of these limiting events, but the boundary established by SCIVs is required to ensure that leakage from the primary containment is processed by the Standby Gas Treatment System (SGT) System following secondary containment isolation, before being released to the environment.

BASES

APPLICABLE SAFETY ANALYSES (continued)

Maintaining SCIVs OPERABLE with isolation times within limits ensures that fission products will remain trapped inside secondary containment following secondary containment isolation 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 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) or during movement of recently irradiated fuel assemblies in the secondary containment. Moving recently irradiated fuel assemblies in the secondary containment may also occur in MODES 1, 2, and 3. Due to radioactive decay, SCIVs are only required to be OPERABLE during fuel handling involving handling recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous 24 hours).

BASES

ACTIONS

The ACTIONS are modified by three Notes. The first Note allows penetration flow paths to be unisolated intermittently under administrative controls. These controls consist of stationing a dedicated operator, who is in continuous communication with the control room, at the controls of the isolation device. In this way, the penetration can be rapidly isolated when a need for secondary containment isolation is indicated.

The second Note provides clarification that, for the purpose of this LCO, separate Condition entry is allowed for each penetration flow path. This is acceptable, since the Required Actions for each Condition provide appropriate compensatory actions for each inoperable SCIV. Complying with the Required Actions may allow for continued operation, and subsequent inoperable SCIVs are governed by subsequent Condition entry and application of associated Required Actions.

The third Note ensures appropriate remedial actions are taken, if necessary, if the affected system(s) are rendered inoperable by an inoperable SCIV.

A.1 and A.2

In the event that there are one or more penetration flow paths with one SCIV inoperable, the affected penetration flow path(s) 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 SCIV, a closed manual valve, and a blind flange. For penetrations isolated in accordance with Required Action A.1, the device used to isolate the penetration should be the closest available device to secondary containment. The Required Action must be completed within the 8 hour Completion Time. The specified time period is reasonable considering the time required to

BASES

ACTIONS (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, the movement of recently 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.

Required Action D.1 has been modified by a Note stating that LCO 3.0.3 is not applicable. If moving recently 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.

BASES

BACKGROUND (continued)

g. A centrifugal fan.

The capacity of the SGT System is sufficient to reduce and maintain the reactor building at a subatmospheric pressure of -0.25 inches water gauge (under neutral wind conditions of greater than 2 mph but less than 5 mph) with an air infiltration rate of no more than 100% of the reactor building volume per day.

The demister is provided to remove entrained water in the air, while the electric heater reduces the relative humidity of the airstream to less than 70% (Ref. 2). The prefilter removes large particulate matter, while the HEPA filter removes fine particulate matter and protects the charcoal from fouling. The charcoal adsorber removes gaseous elemental iodine and organic iodides, and the final HEPA filter collects any carbon fines exhausted from the charcoal adsorber.

The SGT System automatically starts and operates in response to actuation signals indicative of conditions or an accident that could require operation of the system. Following initiation, both charcoal filter train fans start. Upon verification that both subsystems are operating, the redundant subsystem is normally shut down.

APPLICABLE SAFETY ANALYSES

The design basis for the SGT System is to mitigate the consequences of a loss of coolant accident and fuel handling accidents involving handling recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous 24 hours) (Ref. 2). For all events analyzed, the SGT System is shown to be automatically initiated to reduce, via filtration and adsorption, the radioactive material released to the environment.

The SGT System satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii) (Ref. 3).

LCO

Following a DBA, a minimum of one SGT subsystem is required to maintain the secondary containment at a negative pressure with respect to the environment and to process gaseous releases. Meeting the LCO requirements for two OPERABLE subsystems ensures operation of at least one SGT subsystem in the event of a single active failure. An OPERABLE SGT

BASES

LCO (continued)

subsystem consists of a demister, prefilter, an electric heater, HEPA filter, charcoal adsorber, a final HEPA filter, exhaust fan, and associated ductwork, dampers, valves and controls.

When the required decay heat removal flow through the cross tie damper is not met, only ONE SGT subsystem may be considered OPERABLE.

APPLICABILITY

In MODES 1, 2, and 3, a DBA could lead to a fission product release to primary containment that leaks to secondary containment. Therefore, SGT System OPERABILITY is required during these MODES.

In MODES 4 and 5, the probability and consequences of these events are reduced due to the pressure and temperature limitations in these MODES. Therefore, maintaining the SGT System in OPERABLE status is not required in MODE 4 or 5, except for other situations under which significant releases of radioactive material can be postulated, such as during operations with a potential for draining the reactor vessel (OPDRVs) or during movement of recently irradiated fuel assemblies in the secondary containment. Due to radioactive decay, the SGT System is only required to be OPERABLE during fuel handling involving handling recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous 24 hours).

ACTIONS

A.1

With one SGT subsystem inoperable, the inoperable subsystem must be restored to OPERABLE status in 7 days. In this Condition, the remaining OPERABLE SGT subsystem is adequate to perform the required radioactivity release control function. However, the overall system reliability is reduced because a single failure in the OPERABLE subsystem could result in the radioactivity release control function not being adequately performed. The 7 day Completion Time is based on consideration of such factors as the availability of the OPERABLE redundant SGT subsystem and the low probability of a DBA occurring during this period.

B.1 and B.2

If the SGT subsystem cannot be restored to OPERABLE status within the required Completion Time in MODE 1, 2, or 3, 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

BASES

ACTIONS (continued)

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.

C.1, C.2.1, and C.2.2

During movement of recently irradiated fuel assemblies, in the secondary containment or during OPDRVs, when Required Action A.1 cannot be completed within the required Completion Time, the OPERABLE SGT subsystem should immediately be placed in operation. This action ensures that the remaining subsystem is OPERABLE, that no failures that could prevent automatic actuation have occurred, and that any other failure would be readily detected.

An alternative to Required Action C.1 is to immediately suspend activities that represent a potential for releasing a significant amount of radioactive material to the secondary containment, thus placing the plant in a condition that minimizes risk. If applicable, movement of recently irradiated fuel assemblies must immediately be suspended. Suspension of these activities must not preclude completion of movement of a component to a safe position. Also, if applicable, actions must immediately be initiated to suspend OPDRVs in order to minimize the probability of a vessel draindown and subsequent potential for fission product release. Actions must continue until OPDRVs are suspended.

Required Actions of Condition C have been modified by a Note stating that LCO 3.0.3 is not applicable. If moving recently irradiated fuel assemblies while in MODE 4 or 5, LCO 3.0.3 would not specify any action. If moving recently irradiated fuel assemblies 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.

BASES

ACTIONS (continued)

D.1

If both SGTS subsystems are inoperable in MODE 1, 2, or 3, the SGT system may not be capable of supporting the required radioactivity release control function. Therefore, actions are required to enter LCO 3.0.3 immediately.

E.1 and E.2

When two SGT subsystems are inoperable, if applicable, movement of recently irradiated fuel assemblies in secondary containment must immediately be suspended. Suspension of these activities shall not preclude completion of movement of a component to a safe position. Also, if applicable, actions must immediately be initiated to suspend OPDRVs in order to minimize the probability of a vessel draindown and subsequent potential for fission product release. Actions must continue until OPDRVs are suspended.

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

SURVEILLANCE REQUIREMENTS

SR 3.6.4.3.1

Operating each SGT subsystem, including each filter train fan, for ≥ 10 continuous hours ensures that both subsystems are OPERABLE and that all associated controls are functioning properly. It also ensures that blockage, fan or motor failure, or excessive vibration can be detected for corrective action. Operation with the heaters on for ≥ 10 continuous hours every 31 days eliminates moisture on the adsorbers and HEPA filters. The 31 day Frequency was developed in consideration of the known reliability of

BASES

APPLICABLE SAFETY ANALYSES

The ability of the CREF System to maintain the habitability of the control room is an explicit assumption for the safety analyses presented in the USAR, Chapters X and XIV (Refs. 1 and 2, respectively). The CREF System is assumed to operate following a loss of coolant accident and a fuel handling accident involving handling lately irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous 7 days).

The CREF System satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii) (Ref. 4).

LCO

The CREF System is required to be OPERABLE, since total system failure could result in exceeding a dose of 5 rem to the control room operators in the event of a DBA.

The CREF System is considered OPERABLE when the individual components necessary to control operator exposure are OPERABLE. The system is considered OPERABLE when its associated:

a. Fans are OPERABLE (one supply fan, the emergency booster fan and the exhaust booster fan);

b. HEPA filter and charcoal adsorber are not excessively restricting flow and are capable of performing their filtration functions; and

c. Ductwork, valves, and dampers are OPERABLE, and air circulation can be maintained.

In addition, the control room boundary must be maintained, including the integrity of the walls, floors, ceilings, ductwork, and access doors, such that the pressurization limit of SR 3.7.4.4 can be met. However, it is acceptable for access doors to be open for normal control room entry and exit, and not consider it to be a failure to meet the LCO.

BASES

APPLICABILITY

In MODES 1, 2, and 3, the CREF System must be OPERABLE to control operator exposure during and following a DBA, since the DBA could lead to a fission product release.

In MODES 4 and 5, the probability and consequences of a DBA are reduced because of the pressure and temperature limitations in these MODES. Therefore, maintaining the CREF System OPERABLE is not required in MODE 4 or 5, except for the following situations under which significant radioactive releases can be postulated:

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

b. During movement of lately irradiated fuel assemblies in the secondary containment. Due to radioactive decay, the CREF System is only required to be OPERABLE during fuel handling involving handling lately irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous 7 days).

ACTIONS

A.1

The inoperable CREF System must be restored to OPERABLE status within 7 days. With the unit in this condition, there is no other system to perform control room radiation protection. The 7 day Completion Time is based on the low probability of a DBA occurring during this time period.

B.1 and B.2

In MODE 1, 2, or 3, if the inoperable CREF System cannot be restored to OPERABLE status within the associated Completion Time, the unit must be placed in a MODE that minimizes risk. To achieve this status, the unit must be placed in at least MODE 3 within 12 hours and in MODE 4 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

BASES

ACTIONS

C.1, C.2, and C.3

The Required Actions of Condition C are modified by a Note indicating that LCO 3.0.3 does not apply. If moving lately irradiated fuel assemblies while in MODE 1, 2, or 3, the fuel movement is independent of reactor operations. Therefore, inability to suspend movement of lately irradiated fuel assemblies is not sufficient reason to require a reactor shutdown.

During movement of lately irradiated fuel assemblies in the secondary containment or during OPDRVs, if the inoperable CREF System cannot be restored to OPERABLE status within the required Completion Time, activities that present a potential for releasing radioactivity that might require isolation of the control room must be immediately suspended. This places the unit in a condition that minimizes risk.

If applicable, movement of lately irradiated fuel assemblies in the secondary containment must be suspended immediately. Suspension of these activities shall not preclude completion of movement of a component to a safe position. Also, if applicable, actions must be initiated immediately to suspend OPDRVs to minimize the probability of a vessel draindown and the subsequent potential for fission product release. Actions must continue until the OPDRVs are suspended.

SURVEILLANCE REQUIREMENTS

SR 3.7.4.1

This SR verifies that the CREF System in a standby mode starts on demand and continues to operate. The system should be checked periodically to ensure that it starts and functions properly. As the environmental and normal operating conditions of this system are not severe, testing the system once every month provides an adequate check on this system. Since the CREF System does not contain heaters, the system need only be operated for $\geq$ 15 minutes to demonstrate the function of the system. The 31 day Frequency is based on the known reliability of the equipment.

B 3.7 PLANT SYSTEMS

B 3.7.6 Spent Fuel Storage Pool Water Level

BASES

BACKGROUND	The minimum water level in the spent fuel storage pool meets the assumptions of iodine decontamination factors following a fuel handling accident. A general description of the spent fuel storage pool design is found in the USAR, Section X-3.0 (Ref. 1). The assumptions of the fuel handling accident are found in the USAR, Section XIV-6.4 (Ref. 2).
APPLICABLE SAFETY ANALYSES	The water level above the irradiated fuel assemblies is an implicit assumption of the fuel handling accident. A fuel handling accident is evaluated to ensure that the radiological consequences (calculated total effective dose equivalent at the exclusion area and low population zone boundaries) are within 10 CFR 50.67 limits (Ref. 4). A fuel handling accident could release a fraction of the fission product inventory by breaching the fuel rod cladding as discussed in the Regulatory Guide 1.183 (Ref. 5). The fuel handling accident is evaluated for the dropping of an irradiated fuel assembly onto the reactor core. The consequences of a fuel handling accident over the spent fuel storage pool are no more severe than those of the fuel handling accident over the reactor core, as discussed in the USAR, Section XIV-6.1 (Ref. 6). The water level in the spent fuel storage pool provides for absorption of water soluble fission product gases and transport delays of soluble and insoluble gases that must pass through the water before being released to the secondary containment atmosphere. This absorption and transport delay reduces the potential radioactivity of the release during a fuel handling accident. The spent fuel storage pool water level satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii) (Ref. 7).

BASES

REFERENCES

1. USAR, Section X-3.0.
2. USAR, Section XIV-6.4.
3. Not used.
4. 10 CFR 50.67.
5. Regulatory Guide 1.183, July 2000.
6. USAR, Section XIV-6.1.
7. 10 CFR 50.36(c)(2)(ii).

BASES (continued)

LCO

The Main Turbine Bypass System is required to be OPERABLE to limit peak pressure in the main steam lines and maintain reactor pressure within acceptable limits during events that cause rapid pressurization, so that the Safety Limit MCPR is not exceeded. With one Main Turbine Bypass Valve inoperable, modifications to the MCPR operating limits (LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)") may be applied to allow this LCO to be met. The MCPR operating limits for one inoperable Main Turbine Bypass Valve are specified in the COLR. An OPERABLE Main Turbine Bypass System requires all three bypass valves to open in response to increasing main steam line pressure. This response is within the assumptions of the applicable analyses (Ref. 4).

APPLICABILITY

The Main Turbine Bypass System is required to be OPERABLE at ≥ 25% RTP to ensure that the fuel cladding integrity Safety Limit and the cladding 1% plastic strain limit are not violated during the Applicable Safety Analyses transients. As discussed in the Bases for TLCO 3.2.1, "LINEAR HEAT GENERATION RATE (LHGR)," and LCO 3.2.2, sufficient margin to these limits exists at < 25% RTP. Therefore, these requirements are only necessary when operating at or above this power level.

ACTIONS

A.1

If one Main Turbine Bypass Valve is inoperable, and the MCPR operating limits for one inoperable Main Turbine Bypass Valve, as specified in the COLR, are not applied, the assumptions of the design basis transient analyses may not be met. Under such circumstances, prompt action should be taken to restore the inoperable Main Turbine Bypass Valve to OPERABLE status or adjust the MCPR operating limits accordingly. The 2 hour Completion Time is reasonable, based on the time to complete the Required Action and the low probability of an event occurring during this period requiring the Main Turbine Bypass System.

ACTIONS (continued)

B.4

In Condition B, the remaining OPERABLE DG and offsite circuits are adequate to supply electrical power to the onsite Class 1E Distribution System. The 7 day Completion Time takes into account the capacity and capability of the remaining AC sources, reasonable time for repairs, and low probability of a DBA occurring during this period.

The second Completion Time for Required Action B.4 establishes a limit on the maximum time allowed for any combination of required AC power sources to be inoperable during any single contiguous occurrence of failing to meet the LCO. If Condition B is entered while, for instance, an offsite circuit is inoperable and that circuit is subsequently restored OPERABLE, the LCO may already have been not met for up to 7 days. This situation could lead to a total of 14 days, since initial failure of the LCO, to restore the DG. At this time, an offsite circuit could again become inoperable, the DG restored OPERABLE, and an additional 7 days (for a total of 21 days) allowed prior to complete restoration of the LCO. The 14 day Completion Time provides a limit on the time allowed in a specified condition after discovery of failure to meet the LCO. This limit is considered reasonable for situations in which Conditions A and B are entered concurrently. The "AND" connector between the 7 day and 14 day Completion Times means that both Completion Times apply simultaneously, and the more restrictive must be met.

Similar to Required Action B.2, the second Completion Time of Required Action B.4 allows for an exception to the normal "time zero" for beginning the allowed outage time "clock." This exception results in establishing the "time zero" at the time that the LCO was initially not met, instead of the time that Condition B was entered.

ACTIONS (continued)

C.1 and C.2

Required Action C.1 addresses actions to be taken in the event of inoperability of redundant required features concurrent with inoperability of two offsite circuits. Required Action C.1 reduces the vulnerability to a loss of function. The Completion Time for taking these actions is reduced to 12 hours from that allowed with one division without offsite power (Required Action A.2). The rationale for the reduction to 12 hours is that Regulatory Guide 1.93 (Ref. 8) allows a Completion Time of 24 hours for two required offsite circuits inoperable, based upon the assumption that two complete safety divisions are OPERABLE. When a concurrent redundant required feature failure exists, this assumption is not the case, and a shorter Completion Time of 12 hours is appropriate. These features are designed with redundant safety related divisions, (i.e., single division systems are not included in the list). Redundant required features failures consist of any of these features that are inoperable because any inoperability is on a division redundant to a division with inoperable offsite circuits.

The Completion Time for Required Action C.1 is intended to allow the operator time to evaluate and repair any discovered inoperabilities. This Completion Time also allows for an exception to the normal "time zero" for beginning the allowed outage time "clock." In this Required Action, the Completion Time only begins on discovery that both:

a. Both offsite circuits are inoperable; and

b. A redundant required feature is inoperable.

If, at any time during the existence of this Condition (both offsite circuits inoperable), a redundant required feature subsequently becomes inoperable, this Completion Time begins to be tracked.

According to the recommendations in Regulatory Guide 1.93 (Ref. 8), operation may continue in Condition C for a period that should not exceed 24 hours. This level of degradation means that the offsite electrical power system may not have the capability to effect a safe shutdown and to mitigate the effects of an accident; however, the onsite AC sources have

BASES

LCO (continued)

ensures that a diverse power source is available for providing electrical power support assuming a loss of the offsite circuit. Together, OPERABILITY of the required offsite circuit and DG ensures the availability of sufficient AC sources to operate the plant in a safe manner and to mitigate the consequences of postulated events during shutdown (e.g., fuel handling accidents and reactor vessel draindown). Automatic initiation of the required DG during shutdown conditions is specified in LCO 3.3.5.1, ECCS Instrumentation, and LCO 3.3.8.1, LOP Instrumentation.

The qualified offsite circuit must be capable of maintaining rated frequency and voltage while connected to its respective critical bus, and of accepting required loads during an accident. Qualified offsite circuits are those that are described in the USAR and are part of the licensing basis for the unit. The offsite circuit consists of incoming breaker and disconnect to the startup or emergency station service transformer, associated startup or emergency station service transformer, and the respective circuit path including feeder breakers to all 4.16 kV critical buses required by LCO 3.8.8.

The required DG must be capable of starting, accelerating to rated speed and voltage, connecting to its respective critical bus on detection of bus undervoltage, and accepting required loads. This sequence must be accomplished within 14 seconds. Each DG must also be capable of accepting required loads within the assumed loading sequence intervals, and must continue to operate until offsite power can be restored to the critical buses. These capabilities are required to be met from a variety of initial conditions such as DG in standby with engine hot and DG in standby with engine at ambient conditions.

Proper sequencing of loads, including tripping of nonessential loads, is a required function for DG OPERABILITY. The necessary portions of the Service Water System and Ultimate Heat Sink are also required to provide appropriate cooling to the required DGs.

It is acceptable during shutdown conditions, for a single offsite power circuit to supply both 4.16 kV critical buses. No fast transfer capability is required for offsite circuits to be considered OPERABLE.

BASES

ACTIONS (continued)

to any required 4.16 kV critical bus, ACTIONS for LCO 3.8.8 must be immediately entered. This Note allows Condition A to provide requirements for the loss of the offsite circuit whether or not a division is de-energized. LCO 3.8.8 provides the appropriate restrictions for the situation involving a de-energized division.

SURVEILLANCE REQUIREMENTS

SR 3.8.2.1

SR 3.8.2.1 requires the SRs from LCO 3.8.1 that are necessary for ensuring the OPERABILITY of the AC sources in other than MODES 1, 2, and 3. SR 3.8.1.8 is not required to be met since only one offsite circuit is required to be OPERABLE. Refer to the corresponding Bases for LCO 3.8.1 for a discussion of each SR.

This SR is modified by a Note. The reason for the Note is to preclude requiring the OPERABLE DG(s) from being paralleled with the offsite power network or otherwise rendered inoperable during the performance of SRs, and to preclude deenergizing a required 4.16 kV critical bus or disconnecting a required offsite circuit during performance of SRs. With limited AC sources available, a single event could compromise both the required circuit and the DG. It is the intent that these SRs must still be capable of being met, but actual performance is not required during periods when the DG and offsite circuit is required to be OPERABLE.

Note 2 states SR 3.8.1.11 is considered to be met without the ECCS initiation signals OPERABLE when associated ECCS initiation signals are not required to be OPERABLE per Table 3.3.5.1-1. This SR demonstrates the DG response to an ECCS signal in conjunction with a loss of power signal. When ECCS system(s) are not required to be OPERABLE per LCO 3.5.2, "ECCS - Shutdown," the DG is not required to start in response to ECCS initiation signals. This is consistent with the ECCS instrumentation requirements. However, the DG is still required to meet the other attributes of SR 3.8.1.11 when associated ECCS initiation signals are not required to be OPERABLE per Table 3.3.5.1-1.

REFERENCES

1. 10 CFR 50.36(c)(2)(ii).

BASES

APPLICABLE SAFETY ANALYSES

The initial conditions of Design Basis Accident (DBA) and transient analyses in USAR, Chapter VI (Ref. 4), and Chapter XIV (Ref. 5), assume Engineered Safety Feature (ESF) systems are OPERABLE. The DGs are designed to provide sufficient capacity, capability, redundancy, and reliability to ensure the availability of necessary power to ESF systems so that fuel, Reactor Coolant System, and containment design limits are not exceeded. These limits are discussed in more detail in the Bases for Section 3.2, Power Distribution Limits; Section 3.5, Emergency Core Cooling Systems (ECCS) and Reactor Core Isolation Cooling (RCIC) System; and Section 3.6, Containment Systems.

Since diesel fuel oil, lube oil, and starting air subsystems support the operation of the standby AC power sources, they satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii) (Ref. 6).

LCO

Stored diesel fuel oil is required in sufficient supply for 7 days of operation at maximum post-LOCA load demand. It is also required to meet specific standards for quality. Additionally, sufficient lube oil supply must be available to ensure the capability to operate for 7 days at maximum post-LOCA load demand. This requirement, in conjunction with an ability to obtain replacement supplies within 7 days, supports the availability of DGs required to shut down the reactor and to maintain it in a safe condition for an abnormal operational transient or a postulated DBA with loss of offsite power. DG day tank fuel oil requirements, as well as transfer capability from the storage tank to the day tank, are addressed in LCO 3.8.1, "AC Sources — Operating," and LCO 3.8.2, "AC Sources — Shutdown."

The starting air system is required to have a minimum capacity for multiple DG start attempts in accordance with Reference 7, without recharging the air start receivers. Only one air receiver (and associated airstart header) per DG is required, since each air receiver has the required capacity.

BASES

APPLICABILITY

The AC sources (LCO 3.8.1 and LCO 3.8.2) are required to ensure the availability of the required power to shut down the reactor and maintain it in a safe shutdown condition after an abnormal operational transient or a postulated DBA. Because stored diesel fuel oil, lube oil, and starting air subsystems support LCO 3.8.1 and LCO 3.8.2, stored diesel fuel oil, lube oil, and starting air are required to be within limits when the associated DG is required to be OPERABLE.

ACTIONS

The ACTIONS Table is modified by a Note indicating that separate Condition entry is allowed for each DG except for Conditions A, C, and D. This is acceptable, since the Required Actions for each Condition provide appropriate compensatory actions for each inoperable DG subsystem. Complying with the Required Actions for one inoperable DG subsystem may allow for continued operation, and subsequent inoperable DG subsystem(s) governed by separate Condition entry and application of associated Required Actions. The Note does not apply to Conditions A, C and D since the CNS design has two fuel oil storage tanks that supply fuel oil to both DGs.

BASES

ACTIONS (continued)

inoperable battery), the remaining 125 V DC electrical power subsystem has the capacity to support a safe shutdown and to mitigate an accident condition. Since a subsequent worst case single failure could, however, result in the loss of minimum necessary 125 V DC electrical subsystems to mitigate a worst case accident, continued power operation should not exceed 2 hours. The 2 hour Completion Time is based on Regulatory Guide 1.93 (Ref. 6) and reflects a reasonable time to assess unit status as a function of the inoperable DC electrical power subsystem and, if the DC electrical power subsystem is not restored to OPERABLE status, to prepare to effect an orderly and safe unit shutdown.

B.1 and B.2

If the 125 V DC electrical power subsystem cannot be restored to OPERABLE status within the required Completion Time, the unit must be brought to a MODE in which the LCO does not apply. To achieve this status, the unit 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. The Completion Time to bring the unit to MODE 4 is consistent with the time required in Regulatory Guide 1.93 (Ref. 6).

C.1

With the Division 1 250 V DC electrical power subsystem inoperable, one LPCI subsystem is rendered inoperable. Loss of the Division 2 250 V DC electrical power subsystem renders HPCI and the other LPCI subsystem inoperable. Required Action C.1 therefore requires with one 250 V DC electrical power subsystem inoperable that the associated supported features be declared inoperable immediately. This declaration also requires entry into applicable Conditions and Required Actions for the associated supported features.

BASES

ACTIONS (continued)

D.1

With the Division 1 250 V DC electrical power subsystem inoperable, one LPCI subsystem is rendered inoperable. Loss of the Division 2 250 V DC electrical power subsystem renders HPCI and the other LPCI subsystem inoperable. Required Action D.1 therefore requires with one 250 V DC electrical power subsystem inoperable that the associated supported features be declared inoperable immediately. This declaration also requires entry into applicable Conditions and Required Actions for the associated supported features.

E.1

Condition E corresponds to a level of degradation in the electrical power distribution system that causes a required safety function to be lost. When more than one AC or DC electrical power distribution subsystem is lost, and this results in the loss of a required function, the plant is in a condition outside the accident analysis. Therefore, no additional time is justified for continued operation. LCO 3.0.3 must be entered immediately to commence a controlled shutdown.

SURVEILLANCE REQUIREMENTS

SR 3.8.7.1

This Surveillance verifies that the AC and DC electrical power distribution systems are functioning properly, with the correct circuit breaker alignment. The correct breaker alignment ensures the appropriate separation and independence of the electrical buses are maintained, and the appropriate voltage is available to each required bus. The verification of proper voltage availability on the buses ensures that the required voltage is readily available for motive as well as control functions for critical system loads connected to these buses. The 7 day Frequency takes into account the redundant capability of the AC and DC electrical power distribution subsystems, and other indications available in the control room that alert the operator to subsystem malfunctions.

BASES

LCO

To prevent criticality during refueling, the refueling interlocks associated with the reactor mode switch refuel position ensure that fuel assemblies are not loaded into the core with any control rod withdrawn.

To prevent these conditions from developing, the all-rods-in, the refueling platform position, the refueling platform fuel grapple fuel loaded, the refueling platform frame mounted hoist fuel loaded, the refueling platform monorail mounted hoist fuel loaded, the refueling platform fuel grapple not full up position, and the service platform hoist fuel loaded inputs are required to be OPERABLE. These inputs are combined in logic circuits, which provide refueling equipment or control rod blocks to prevent operations that could result in criticality during refueling operations.

APPLICABILITY

In MODE 5, a prompt reactivity excursion could cause fuel damage and subsequent release of radioactive material to the environment. The refueling equipment interlocks protect against prompt reactivity excursions during MODE 5. The interlocks are required to be OPERABLE during in-vessel fuel movement with refueling equipment associated with the interlocks when the reactor mode switch is in the refuel position. The interlocks are not required when the reactor mode switch is in the shutdown position because a control rod block (LCO 3.3.2.1, "Control Rod Block Instrumentation") ensures control rod withdrawal cannot occur simultaneously with in-vessel fuel movements.

In MODES 1, 2, 3, and 4, the reactor pressure vessel head is on, and fuel loading activities are not possible. Therefore, the refueling interlocks are not required to be OPERABLE in these MODES.

ACTIONS

A.1, A.2.1, and A.2.2

With one or more of the required refueling equipment interlocks inoperable, the unit must be placed in a condition in which the LCO does not apply (Required Action A.1) or the interlocks are not needed (Required Action A.2). Therefore, Required Action A.1 requires that in-vessel fuel movement with the affected refueling equipment must be immediately suspended. This action ensures that operations are not performed with equipment that would potentially not be blocked from unacceptable operations (e.g., loading fuel into a cell with a control rod withdrawn). Suspension of in-vessel fuel movement shall not preclude completion of movement of a component to a safe position.

BASES

ACTIONS (continued)

Alternatively, Required Actions A.2.1 and A.2.2 will permit continued fuel movement with the interlocks inoperable if a control rod withdrawal block is inserted, and all control rods are subsequently verified to be fully inserted. Required Action A.2.1 (rod block) ensures no control rods can be withdrawn. The withdrawal block utilized must ensure that if rod withdrawal is requested, the rod will not respond (i.e., it will remain inserted). Required Action A.2.2 is performed after placing the rod withdrawal block in effect, and provides a verification that all control rods are fully inserted. This verification that all control rods are fully inserted is in addition to the periodic verifications required by SR 3.9.3.1.

Like Required Action A.1, Required Actions A.2.1 and A.2.2 ensure unacceptable operations are blocked (e.g., loading fuel into a cell with the control rod withdrawn).

One use for the A.2 Required Actions is to permit performance of SR 3.9.1.1 once, prior to fuel movement, without the need for subsequent performance if the fuel movement extends longer than the 7 day Frequency of the SR. This permits continued fuel movement under the protection of the continuous rod block inserted by the Required Actions.

SURVEILLANCE REQUIREMENTS

SR 3.9.1.1

Performance of a CHANNEL FUNCTIONAL TEST demonstrates each required refueling equipment interlock will function properly when a simulated or actual signal indicative of a required condition is injected into the logic. 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 non-Technical Specifications tests at least once per refueling interval with applicable extensions. The CHANNEL FUNCTIONAL TEST may be performed by any series of sequential, overlapping, or total channel steps so that the entire channel is tested.

The 7 day Frequency is based on engineering judgment and is considered adequate in view of other indications of refueling interlocks and their associated input status that are available to unit operations personnel.

REFERENCES

1. USAR, Appendix F, Section F-2.5.

2. USAR, Section VII-6.

BASES

3. USAR, Section XIV-5.3.3.

4. USAR, Section XIV-5.3.4.

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

B 3.9 REFUELING OPERATIONS

B 3.9.2 Refuel Position One-Rod-Out Interlock

BASES

BACKGROUND

The refuel position one-rod-out interlock restricts the movement of control rods to reinforce unit procedures that prevent the reactor from becoming critical during refueling operations. During refueling operations, no more than one control rod is permitted to be withdrawn.

The USAR, Appendix F, specifies that at least one of the two required independent reactivity control systems provided be capable of making and holding the core subcritical under any conditions with appropriate margins for contingencies (Ref. 1). The control rods serve as the system capable of maintaining the reactor subcritical in cold conditions.

The refuel position one-rod-out interlock prevents the selection of a second control rod for movement when any other control rod is not fully inserted (Ref. 2). It is a logic circuit that has redundant channels. It uses the all-rods-in signal (from the control rod full-in position indicators discussed in LCO 3.9.4, "Control Rod Position Indication") and a rod selection signal (from the Reactor Manual Control System).

This Specification ensures that the performance of the refuel position one-rod-out interlock in the event of a Design Basis Accident meets the assumptions used in the safety analysis of Reference 3.

APPLICABLE SAFETY ANALYSES

The refueling position one-rod-out interlock is explicitly assumed in the USAR analysis for the control rod removal error during refueling (Ref. 3). This analysis evaluates the consequences of control rod withdrawal during refueling. A prompt reactivity excursion during refueling could potentially result in fuel failure with subsequent release of radioactive material to the environment.

The refuel position one-rod-out interlock and adequate SDM(LCO 3.1.1, "SHUTDOWN MARGIN (SDM)" prevent criticality by preventing withdrawal of more than one control rod. With one control rod withdrawn, the core will remain subcritical, thereby preventing any prompt critical excursion.

The refuel position one-rod-out interlock satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii) (Ref. 4).

BASES

LCO

To prevent criticality during MODE 5, the refuel position one-rod-out interlock ensures no more than one control rod may be withdrawn. Both channels of the refuel position one-rod-out interlock are required to be OPERABLE and the reactor mode switch must be locked in the refuel position to support the OPERABILITY of these channels.

APPLICABILITY

In MODE 5, with the reactor mode switch in the refuel position, the OPERABLE refuel position one-rod-out interlock provides protection against prompt reactivity excursions.

In MODES 1, 2, 3, and 4, the refuel position one-rod-out interlock is not required to be OPERABLE and is bypassed. In MODES 1 and 2, the Reactor Protection System (LCO 3.3.1.1, "Reactor Protection System (RPS) Instrumentation") and the control rods (LCO 3.1.3, "Control Rod OPERABILITY") provide mitigation of potential reactivity excursions. In MODES 3 and 4, with the reactor mode switch in the shutdown position, a control rod block (LCO 3.3.2.1, "Control Rod Block Instrumentation") ensures all control rods are inserted, thereby preventing criticality during shutdown conditions.

ACTIONS

A.1 and A.2

With one or both channels of the refueling position one-rod-out interlock inoperable, the refueling interlocks may not be capable of preventing more than one control rod from being withdrawn. This condition may lead to criticality.

Control rod withdrawal must be immediately suspended, and action must be immediately initiated to fully insert all insertable control rods in core cells containing one or more fuel assemblies. Action must continue until all such control rods are fully inserted. Control rods in core cells containing no fuel assemblies do not affect the reactivity of the core and, therefore, do not have to be inserted.

SURVEILLANCE REQUIREMENTS

SR 3.9.2.1

Proper functioning of the refueling position one-rod-out interlock requires the reactor mode switch to be in Refuel. During control rod withdrawal in MODE 5, improper positioning of the reactor mode switch could, in some instances, allow improper bypassing of required interlocks.

BASES

SURVEILLANCE REQUIREMENTS (continued)

Therefore, this Surveillance imposes an additional level of assurance that the refueling position one-rod-out interlock will be OPERABLE when required. By "locking" the reactor mode switch in the proper position (i.e., removing the reactor mode switch key from the console while the reactor mode switch is positioned in refuel), an additional administrative control is in place to preclude operator errors from resulting in unanalyzed operation.

The Frequency of 12 hours is sufficient in view of other administrative controls utilized during refueling operations to ensure safe operation.

SR 3.9.2.2

Performance of a CHANNEL FUNCTIONAL TEST on each channel demonstrates the associated refuel position one-rod-out interlock will function properly when a simulated or actual signal indicative of a required condition is injected into the logic. 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 non-Technical Specifications tests at least once per refueling interval with applicable extensions. The CHANNEL FUNCTIONAL TEST may be performed by any series of sequential, overlapping, or total channel steps so that the entire channel is tested. The 7 day Frequency is considered adequate because of demonstrated circuit reliability, procedural controls on control rod withdrawals, and visual and audible indications available in the control room to alert the operator to control rods not fully inserted. To perform the required testing, the applicable condition must be entered (i.e., a control rod must be withdrawn from its full-in position). Therefore, SR 3.9.2.2 has been modified by a Note that states the CHANNEL FUNCTIONAL TEST is not required to be performed until 1 hour after any control rod is withdrawn.

REFERENCES

1. USAR, Appendix F, Section F-2.5.
2. USAR, Section VII-6.
3. USAR, Section XIV-5.3.3.
4. 10 CFR 50.36(c)(2)(ii).

B 3.9 REFUELING OPERATIONS

B 3.9.6 Reactor Pressure Vessel (RPV) Water Level

BASES

BACKGROUND

The movement of fuel assemblies or handling of control rods within the RPV requires a minimum water level of 21 ft above the top of the RPV flange. During refueling, this maintains a sufficient water level in the reactor vessel cavity. Sufficient water is necessary to retain iodine fission product activity in the water in the event of a refueling accident (Refs. 1 and 2). Sufficient iodine activity would be retained to limit offsite doses from the accident to 10 CFR 50.67 limits (Ref. 3).

APPLICABLE SAFETY ANALYSES

During movement of fuel assemblies or handling of control rods, the water level in the RPV is an initial condition design parameter in the analysis of a refueling accident in containment postulated by Reference 1. A minimum water level of 21 ft allows a decontamination factor of 200 to be used in the accident analysis for halogens (Ref. 1). This relates to the assumption that 99.5% of the total halogens released from the pellet to cladding gap of all the damaged fuel assembly rods is retained by the water. The fuel pellet to cladding gap activity assumes RG 1.183, Table 3 non-loss-of-coolant-accident gap fractions (Ref. 5).

Analysis of the refueling accident inside containment is described in Reference 1. With a minimum water level of 21 ft and a minimum decay time of 24 hours prior to fuel handling, the analysis and test programs demonstrate that the iodine release due to a postulated refueling accident is adequately captured by the water and that offsite doses are maintained within allowable limits (Ref. 3). The worst case assumptions include the dropping of the irradiated fuel assembly being handled onto the reactor core loaded with irradiated fuel assemblies. The possibility exists of the dropped assembly striking the RPV flange and releasing fission products. Therefore, the minimum depth for water coverage to ensure

BASES

SURVEILLANCE REQUIREMENTS

SR 3.9.6.1

Verification of a minimum water level of 21 ft above the top of the RPV flange ensures that the design basis for the postulated refueling accident analysis during refueling operations is met. Water at the required level limits the consequences of damaged fuel rods, which are postulated to result from a refueling accident in containment (Ref. 1).

The Frequency of 24 hours is based on engineering judgment and is considered adequate in view of the large volume of water and the normal procedural controls on valve positions, which make significant unplanned level changes unlikely.

REFERENCES

1. USAR, Section XIV-6.4
2. USAR, Section X-3.0.
3. 10 CFR 50.67.
4. 10 CFR 50.36(c)(2)(ii).
5. Regulatory Guide 1.183, July 2000.

B 3.10 SPECIAL OPERATIONS

B 3.10.1 Inservice Leak and Hydrostatic Testing Operation

BASES

BACKGROUND

The purpose of this Special Operations LCO is to allow certain reactor coolant pressure tests to be performed in MODE 4 when the metallurgical characteristics of the reactor pressure vessel (RPV) require the pressure testing at temperatures > 212°F (normally corresponding to MODE 3) or to allow completing these reactor coolant pressure tests when the initial conditions do not require temperatures > 212°F. Furthermore, the purpose is to allow continued performance of control rod scram time testing required by SR 3.1.4.1 or SR 3.1.4.4 if reactor coolant temperatures exceed 212°F when the control rod scram time testing is initiated in conjunction with an inservice leak or hydrostatic test. These control rod scram time tests would be performed in accordance with LCO 3.10.4, "Single Control Rod Withdrawal - Cold Shutdown," during MODE 4 operation.

Inservice hydrostatic testing and system leakage pressure tests required by Section XI of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (Ref. 1) are performed prior to the reactor going critical after a refueling outage. Recirculation pump operation, decay heat and a water solid RPV (except for an air bubble for pressure control) are used to achieve the necessary temperatures and pressures required for these tests. The minimum temperatures (at the required pressures) allowed for these tests are determined from the RPV pressure and temperature (P/T) limits required by LCO 3.4.9, "Reactor Coolant System (RCS) Pressure and Temperature (P/T) Limits." These limits are conservatively based on the fracture toughness of the reactor vessel, taking into account anticipated vessel neutron fluence.

With increased reactor vessel fluence over time, the minimum allowable vessel temperature increases at a given pressure. Periodic updates to the RPV P/T limit curves are performed as necessary, based upon the results of analyses of irradiated surveillance specimens removed from the vessel. Hydrostatic and leak testing may eventually be required with minimum reactor coolant temperature > 212°F. However, even with required minimum reactor coolant temperatures < 212°F, maintaining RCS temperatures within a small band during the test can be impractical. Removal of heat addition from recirculation pump operation and reactor core decay heat is coarsely controlled by control rod drive hydraulic system flow and reactor water cleanup system non-regenerative heat exchanger operation. Test conditions are focused on maintaining a steady state pressure, and tightly limited temperature control poses an unnecessary burden on the operator and may not be achievable in certain instances.

BASES

BACKGROUND (continued)

Scram time testing required by SR 3.1.4.1 and SR 3.1.4.4 requires reactor steam dome pressure $\geq$ 800 psig. The hydrostatic and/or RCS leakage tests require pressure of approximately 1,000 psig.

Other testing (Excess Flow Check Valve testing for example) may be performed in conjunction with the allowances for inservice leak or hydrostatic tests and control rod scram time tests.

APPLICABLE SAFETY ANALYSES

Allowing the reactor to be considered in MODE 4 when the reactor coolant temperature is > 212°F during or as a consequence of, hydrostatic or leak testing, or as a consequence of control rod scram time testing initiated in conjunction with an inservice leak or hydrostatic test, effectively provides an exception to MODE 3 requirements, including OPERABILITY of primary containment and the full complement of redundant Emergency Core Cooling Systems. Since the tests are performed nearly water solid, at low decay heat values, and near MODE 4 conditions, the stored energy in the reactor core will be very low. Under these conditions, the potential for failed fuel and a subsequent increase in coolant activity above the LCO 3.4.6, "RCS Specific Activity," limits are minimized. In addition, the secondary containment will be OPERABLE, in accordance with this Special Operations LCO, and will be capable of handling any airborne radioactivity or steam leaks that could occur during the performance of hydrostatic or leak testing. The required pressure testing conditions provide adequate assurance that the consequences of a steam leak will be conservatively bounded by the consequences of the postulated main steam line break outside of primary containment described in Reference 2. Therefore, these requirements will conservatively limit radiation releases to the environment.

In the event of a large primary system leak, the reactor vessel would rapidly depressurize, allowing the low pressure core cooling systems to operate. The capability of the low pressure coolant injection and core spray subsystems, as required in MODE 4 by LCO 3.5.2, "ECCS — Shutdown," would be more than adequate to keep the core flooded under this low decay heat load condition. Small system leaks would be detected by leakage inspections before significant inventory loss occurred.

For the purposes of this test, the protection provided by normally required MODE 4 applicable LCOs, in addition to the secondary containment requirements required to be met by this Special Operations LCO, will ensure acceptable consequences during normal hydrostatic test conditions and during postulated accident conditions.

BASES

APPLICABLE SAFETY ANALYSIS (continued)

As described in LCO 3.0.7, compliance with Special Operations LCOs is optional, and therefore, no criteria of 10 CFR 50.36 (c)(2)(ii) (Ref. 3) apply. Special Operations LCOs provide flexibility to perform certain operations by appropriately modifying requirements of other LCOs. A discussion of the criteria satisfied for the other LCOs is provided in their respective Bases.

LCO

As described in LCO 3.0.7, compliance with this Special Operations LCO is optional. Operation at reactor coolant temperatures > 212°F can be in accordance with Table 1.1-1 for MODE 3 operation without meeting this Special Operations LCO or its ACTIONS. This option may be required due to P/T limits, however, which require testing at temperatures > 212°F, while performance of inservice leak and hydrostatic testing results in the inoperability of subsystems required when > 212°F (i.e., MODE 3). Additionally, even with required minimum reactor coolant temperatures < 212°F, RCS temperatures may drift above 212°F during the performance of inservice leak and hydrostatic testing or during subsequent control rod scram time testing, which is typically performed in conjunction with inservice leak and hydrostatic testing. While this Special Operations LCO is provided for inservice leak and hydrostatic testing, and for scram time testing initiated in conjunction with an inservice leak or hydrostatic test, parallel performance of other tests and inspections is not precluded.

If it is desired to perform these tests while complying with this Special Operations LCO, then the MODE 4 applicable LCOs and specified MODE 3 LCOs must be met. This Special Operations LCO allows changing Table 1.1-1 temperature limits for MODE 4 to "NA" and suspending the requirements of LCO 3.4.8, "Residual Heat Removal (RHR) Shutdown Cooling System — Cold Shutdown." The additional requirements for secondary containment LCOs to be met will provide sufficient protection for operations at reactor coolant temperatures > 212°F for the purpose of performing an inservice leak or hydrostatic test, and for control rod scram time testing initiated in conjunction with an inservice leak or hydrostatic test.

This LCO allows primary containment to be open for frequent unobstructed access to perform inspections, and for outage activities on various systems to continue consistent with the MODE 4 applicable requirements.

BASES

APPLICABILITY

The MODE 4 requirements may only be modified for the performance of, or as a consequence of, inservice leak or hydrostatic tests, or as a consequence of control rod scram time testing initiated in conjunction with an inservice leak or hydrostatic test, so that these operations can be considered as in MODE 4, even though the reactor coolant temperature is > 212°F. The additional requirement for secondary containment OPERABILITY according to the imposed MODE 3 requirements provides conservatism in the response of the unit to any event that may occur. Operations in all other MODES are unaffected by this LCO.

ACTIONS

A Note has been provided to modify the ACTIONS related to inservice leak and hydrostatic testing operation. Section 1.3, Completion Times, specifies that once a Condition has been entered, subsequent divisions, subsystems, components, or variables expressed in the Condition discovered to be inoperable or not within limits, will not result in separate entry into the Condition.

Section 1.3 also specifies that Required Actions of the Condition continue to apply for each additional failure, with Completion Times based on initial entry into the Condition. However, the Required Actions for each requirement of the LCO not met provide appropriate compensatory measures for separate requirements that are not met. As such, a Note has been provided that allows separate Condition entry for each requirement of the LCO.

A.1

If an LCO specified in LCO 3.10.1 is not met, the ACTIONS applicable to the stated requirements are entered immediately and complied with. Required Action A.1 has been modified by a Note that clarifies the intent of another LCO's Required Action to be in MODE 4 includes reducing the average reactor coolant temperature to ≤ 212°F.

A.2.1 and A.2.2

Required Action A.2.1 and Required Action A.2.2 are alternate Required Actions that can be taken instead of Required Action A.1 to restore compliance with the normal MODE 4 requirements, and thereby exit this Special Operation LCO's Applicability. Activities that could further increase reactor coolant temperature or pressure are suspended immediately, in accordance with Required Action A.2.1, and the reactor coolant temperature is reduced to establish normal MODE 4 requirements. The allowed Completion Time of 24 hours for Required Action A.2.2 is based on engineering judgment and provides sufficient

BASES

ACTIONS (continued)

time to reduce the average reactor coolant temperature from the highest expected value to ≤ 212°F with normal cooldown procedures. The Completion Time is also consistent with the time provided in LCO 3.0.3 to reach MODE 4 from MODE 3.

SURVEILLANCE REQUIREMENTS

SR 3.10.1.1

The LCOs made applicable are required to have their Surveillances met to establish that this LCO is being met. A discussion of the applicable SRs is provided in their respective Bases.

REFERENCES

1. American Society of Mechanical Engineers, Boiler and Pressure Vessel Code, Section XI.

2. USAR, Section XIV-6.5.

3. 10 CFR 50.36(c)(2)(ii).