APS

Scott A. Bauer
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Technical Specification 5.5.14

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102-05108-SAB/TNW/RKR
May 20, 2004

ATTN: Document Control Desk
U. S. Nuclear Regulatory Commission
Mail Station P1-37
Washington, DC 20555-0001

Dear Sirs:

Subject: Palo Verde Nuclear Generating Station (PVNGS)
Units 1, 2, and 3
Docket Nos. STN 50-528/529/530
Technical Specifications Bases Revisions 28 and 29 Update

Pursuant to PVNGS Technical Specification (TS) 5.5.14, "Technical Specifications Bases Control Program," Arizona Public Service Company (APS) is submitting changes to the TS Bases incorporated into Revision 28, implemented on April 15, 2004 and Revision 29, implemented on May 20, 2004. The Revision 28 insertion instructions and replacement pages are provided in Enclosure 1. The Revision 29 insertion instructions and replacement pages are provided in Enclosure 2.

No commitments are being made to the NRC by this letter. Should you have any questions, please contact Thomas N. Weber at (623) 393-5764.

Sincerely,

T/N Weber for
SA Bauer

SAB/TNW/RKR/kg

Enclosures: 1. PVNGS Technical Specification Bases Revision 28 Insertion Instructions and Replacement Pages

2. PVNGS Technical Specification Bases Revision 29 Insertion Instructions and Replacement Pages

cc:	B. S. Mallett	NRC Region IV Regional Administrator
	M. B. Fields	NRC NRR Project Manager
	N. L. Salgado	NRC Senior Resident Inspector for PVNGS

A001

ENCLOSURE 1

PVNGS
Technical Specification Bases
Revision 28

Insertion Instructions and
Replacement Pages

PVNGS Technical Specifications Bases
Revision 28
Insertion Instructions

Remove Pages:	**Insert New Pages:**
Cover page	Cover page
List of Effective Pages, Pages 1/2 through List of Effective Pages, Page 7/8	List of Effective Pages, Pages 1/2 through List of Effective Pages, Page 7/8
B 3.1.1-1/3.1.1-2 B 3.1.1-3/3.1.1-4	B 3.1.1-1/3.1.1-2 B 3.1.1-3/3.1.1-4
B 3.1.2-1/3.1.2-2 B 3.1.2-3/3.1.2-4	B 3.1.2-1/3.1.2-2 B 3.1.2-3/3.1.2-4
B 3.1.5-1/3.1.5-2 through B 3.1.5-11/blank	B 3.1.5-1/3.1.5-2 through B 3.1.5-11/blank
B 3.1.7-3/3.1.7-4	B 3.1.7-3/3.1.7-4
B 3.1.8-1/3.1.8-2 through B 3.1.8-5/blank	B 3.1.8-1/3.1.8-2 through B 3.1.8-5/blank
B 3.1.9-5/3.1.9-6	B 3.1.9-5/3.1.9-6
B 3.1.10-1/3.1.10-2	B 3.1.10-1/3.1.10-2
B 3.1.11-1/3.1.11-2 B 3.1.11-3/3.1.11-4	B 3.1.11-1/3.1.11-2 B 3.1.11-3/3.1.11-4
B 3.2.1-1/3.2.1-2 B 3.2.1-3/3.2.1-4	B 3.2.1-1/3.2.1-2 B 3.2.1-3/3.2.1-4
B 3.2.2-1/3.2.2-2 B 3.2.2-3/3.2.2-4	B 3.2.2-1/3.2.2-2 B 3.2.2-3/3.2.2-4
B 3.2.3-1/3.2.3-2 B 3.2.3-3/3.2.3-4	B 3.2.3-1/3.2.3-2 B 3.2.3-3/3.2.3-4
B 3.2.4-1/3.2.4-2 B 3.2.4-3/3.2.4-4	B 3.2.4-1/3.2.4-2 B 3.2.4-3/3.2.4-4
B 3.2.5-1/3.2.5-2 through B 3.2.5-5/3.2.5-6	B 3.2.5-1/3.2.5-2 through B 3.2.5-5/3.2.5-6
B 3.3.1-21/3.3.1-22	B 3.3.1-21/3.3.1-22
B 3.3.1-25/3.3.1-26	B 3.3.1-25/3.3.1-26
B 3.4.1-1/3.4.1-2	B 3.4.1-1/3.4.1-2

PVNGS Technical Specifications Bases
Revision 28
Insertion Instructions

(continued)

Remove Pages:	**Insert New Pages:**
B 3.7.1-1/3.7.1-2 through B 3.7.1-7/blank	B 3.7.1-1/3.7.1-2 through B 3.7.1-5/blank
B 3.7.6-1/3.7.6-2 B 3.7.6-3/3.7.6-4	B 3.7.6-1/3.7.6-2 B 3.7.6-3/3.7.6-4

PVNGS

Palo Verde Nuclear Generating Station
Units 1, 2, and 3

Technical Specification Bases

Revision 28
April 15, 2004

PALO VERDE

TECHNICAL SPECIFICATION BASES
LIST OF EFFECTIVE PAGES

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B 2.1.1-1	0	B 3.1.4-5	0
B 2.1.1-2	0	B 3.1.5-1	0
B 2.1.1-3	21	B 3.1.5-2	28
B 2.1.1-4	21	B 3.1.5-3	28
B 2.1.1-5	23	B 3.1.5-4	28
B 2.1.2-1	0	B 3.1.5-5	28
B 2.1.2-2	0	B 3.1.5-6	28
B 2.1.2-3	0	B 3.1.5-7	1
B 2.1.2-4	23	B 3.1.5-8	28
B 2.1.2-5	0	B 3.1.5-9	28
B 3.0-1	0	B 3.1.5-10	28
B 3.0-2	0	B 3.1.5-11	28
B 3.0-3	0	B 3.1.6-1	0
B 3.0-4	0	B 3.1.6-2	0
B 3.0-5	0	B 3.1.6-3	0
B 3.0-6	1	B 3.1.6-4	0
B 3.0-7	0	B 3.1.7-1	0
B 3.0-8	0	B 3.1.7-2	0
B 3.0-9	0	B 3.1.7-3	28
B 3.0-10	14	B 3.1.7-4	0
B 3.0-11	14	B 3.1.7-5	25
B 3.0-12	14	B 3.1.7-6	0
B 3.0-13	0	B 3.1.7-7	0
B 3.0-14	0	B 3.1.7-8	0
B 3.0-15	0	B 3.1.7-9	0
B 3.0-16	17	B 3.1.8-1	28
B 3.0-17	17	B 3.1.8-2	28
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B 3.1.1-1	28	B 3.1.8-5	28
B 3.1.1-2	0	B 3.1.9-1	0
B 3.1.1-3	28	B 3.1.9-2	0
B 3.1.1-4	12	B 3.1.9-3	0
B 3.1.1-5	27	B 3.1.9-4	0
B 3.1.1-6	0	B 3.1.9-5	28
B 3.1.2-1	28	B 3.1.9-6	1
B 3.1.2-2	0	B 3.1.10-1	0
B 3.1.2-3	5	B 3.1.10-2	28
B 3.1.2-4	28	B 3.1.10-3	0
B 3.1.2-5	0	B 3.1.10-4	0
B 3.1.2-6	0	B 3.1.10-5	0
B 3.1.2-7	12	B 3.1.10-6	0
B 3.1.2-8	0	B 3.1.11-1	0
B 3.1.2-9	0	B 3.1.11-2	28
B 3.1.3-1	0	B 3.1.11-3	0
B 3.1.3-2	0	B 3.1.11-4	28
B 3.1.3-3	0	B 3.1.11-5	0
B 3.1.3-4	0	B 3.2.1-1	28
B 3.1.3-5	0	B 3.2.1-2	10
B 3.1.3-6	0	B 3.2.1-3	28
B 3.1.4-1	0	B 3.2.1-4	0
B 3.1.4-2	0	B 3.2.1-5	0
B 3.1.4-3	0	B 3.2.1-6	0
B 3.1.4-4	0	B 3.2.1-7	0

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B 3.2.2-3	0	B 3.3.1-23	25
B 3.2.2-4	28	B 3.3.1-24	25
B 3.2.2-5	1	B 3.3.1-25	25
B 3.2.2-6	0	B 3.3.1-26	25
B 3.2.2-7	0	B 3.3.1-27	25
B 3.2.3-1	28	B 3.3.1-28	25
B 3.2.3-2	10	B 3.3.1-29	25
B 3.2.3-3	0	B 3.3.1-30	27
B 3.2.3-4	28	B 3.3.1-31	27
B 3.2.3-5	0	B 3.3.1-32	27
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B 3.2.3-7	0	B 3.3.1-34	27
B 3.2.3-8	0	B 3.3.1-35	25
B 3.2.3-9	0	B 3.3.1-36	27
B 3.2.3-10	0	B 3.3.1-37	25
B 3.2.4-1	28	B 3.3.1-38	25
B 3.2.4-2	10	B 3.3.1-39	27
B 3.2.4-3	0	B 3.3.1-40	25
B 3.2.4-4	28	B 3.3.1-41	25
B 3.2.4-5	25	B 3.3.1-42	25
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B 3.3.1-5	25	B 3.3.2-2	0
B 3.3.1-6	27	B 3.3.2-3	1
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B 3.3.1-8	25	B 3.3.2-5	0
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B 3.3.1-10	27	B 3.3.2-7	15
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B 3.3.3-5	25	B 3.3.5-23	0
B 3.3.3-6	25	B 3.3.5-24	0
B 3.3.3-7	27	B 3.3.5-25	0
B 3.3.3-8	27	B 3.3.5-26	0
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B 3.3.4-2	0	B 3.3.6-12	0
B 3.3.4-3	0	B 3.3.6-13	0
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B 3.3.4-7	0	B 3.3.6-17	27
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B 3.3.4-9	0	B 3.3.6-19	0
B 3.3.4-10	0	B 3.3.6-20	0
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B 3.3.4-13	0	B 3.3.7-1	2
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B 3.3.4-15	0	B 3.3.7-3	0
B 3.3.5-1	0	B 3.3.7-4	0
B 3.3.5-2	0	B 3.3.7-5	0
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B 3.3.5-13	0	B 3.3.8-7	0
B 3.3.5-14	0	B 3.3.8-8	0
B 3.3.5-15	0	B 3.3.9-1	0
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B 3.3.9-7	0	B 3.4.5-1	0
B 3.3.10-1	0	B 3.4.5-2	6
B 3.3.10-2	0	B 3.4.5-3	6
B 3.3.10-3	0	B 3.4.5-4	0
B 3.3.10-4	0	B 3.4.5-5	6
B 3.3.10-5	18	B 3.4.6-1	0
B 3.3.10-6	0	B 3.4.6-2	6
B 3.3.10-7	0	B 3.4.6-3	6
B 3.3.10-8	14	B 3.4.6-4	6
B 3.3.10-9	14	B 3.4.6-5	6
B 3.3.10-10	14	B 3.4.7-1	0
B 3.3.10-11	14	B 3.4.7-2	6
B 3.3.10-12	14	B 3.4.7-3	6
B 3.3.10-13	14	B 3.4.7-4	2
B 3.3.10-14	14	B 3.4.7-5	0
B 3.3.10-15	14	B 3.4.7-6	0
B 3.3.10-16	14	B 3.4.7-7	27
B 3.3.10-17	14	B 3.4.8-1	0
B 3.3.10-18	14	B 3.4.8-2	6
B 3.3.10-19	14	B 3.4.8-3	6
B 3.3.10-20	14	B 3.4.9-1	0
B 3.3.10-21	14	B 3.4.9-2	0
B 3.3.11-1	0	B 3.4.9-3	1
B 3.3.11-2	2	B 3.4.9-4	0
B 3.3.11-3	2	B 3.4.9-5	0
B 3.3.11-4	2	B 3.4.9-6	0
B 3.3.11-5	19	B 3.4.10-1	0
B 3.3.11-6	2	B 3.4.10-2	7
B 3.3.11-7	2	B 3.4.10-3	0
B 3.3.12-1	15	B 3.4.10-4	0
B 3.3.12-2	15	B 3.4.11-1	0
B 3.3.12-3	5	B 3.4.11-2	7
B 3.3.12-4	5	B 3.4.11-3	0
B 3.3.12-5	6	B 3.4.11-4	0
B 3.3.12-6	6	B 3.4.11-5	0
B 3.4.1-1	10	B 3.4.11-6	0
B 3.4.1-2	28	B 3.4.12-1	1
B 3.4.1-3	0	B 3.4.12-2	1
B 3.4.1-4	0	B 3.4.12-3	0
B 3.4.1-5	0	B 3.4.12-4	0
B 3.4.2-1	7	B 3.4.12-5	0
B 3.4.2-2	1	B 3.4.13-1	0
B 3.4.3-1	0	B 3.4.13-2	0
B 3.4.3-2	0	B 3.4.13-3	1
B 3.4.3-3	0	B 3.4.13-4	0
B 3.4.3-4	2	B 3.4.13-5	0
B 3.4.3-5	2	B 3.4.13-6	0
B 3.4.3-6	0	B 3.4.13-7	2
B 3.4.3-7	0	B 3.4.13-8	2
B 3.4.3-8	2	B 3.4.13-9	0
B 3.4.4-1	0	B 3.4.13-10	2

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B 3.4.14-1	0	B 3.5.3-9	0
B 3.4.14-2	23	B 3.5.3-10	2
B 3.4.14-3	23	B 3.5.4-1	15
B 3.4.14-4	7	B 3.5.4-2	0
B 3.4.14-5	2	B 3.5.4-3	0
B 3.4.14-6	2	B 3.5.5-1	0
B 3.4.14-7	2	B 3.5.5-2	7
B 3.4.15-1	0	B 3.5.5-3	4
B 3.4.15-2	0	B 3.5.5-4	4
B 3.4.15-3	0	B 3.5.5-5	0
B 3.4.15-4	0	B 3.5.5-6	0
B 3.4.15-5	0	B 3.5.5-7	0
B 3.4.15-6	0	B 3.5.6-1	0
B 3.4.15-7	0	B 3.5.6-2	1
B 3.4.16-1	2	B 3.5.6-3	0
B 3.4.16-2	10	B 3.5.6-4	24
B 3.4.16-3	0	B 3.5.6-5	27
B 3.4.16-4	0	B 3.6.1-1	0
B 3.4.16-5	0	B 3.6.1-2	25
B 3.4.16-6	0	B 3.6.1-3	0
B 3.4.17-1	0	B 3.6.1-4	0
B 3.4.17-2	27	B 3.6.1-5	0
B 3.4.17-3	0	B 3.6.2-1	0
B 3.4.17-4	0	B 3.6.2-2	25
B 3.4.17-5	0	B 3.6.2-3	0
B 3.4.17-6	0	B 3.6.2-4	0
B 3.5.1-1	0	B 3.6.2-5	0
B 3.5.1-2	0	B 3.6.2-6	0
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B 3.5.1-5	0	B 3.6.3-1	27
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B 3.5.1-7	1	B 3.6.3-3	27
B 3.5.1-8	1	B 3.6.3-4	27
B 3.5.1-9	0	B 3.6.3-5	27
B 3.5.1-10	1	B 3.6.3-6	27
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B 3.5.2-2	0	B 3.6.3-8	27
B 3.5.2-3	0	B 3.6.3-9	27
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B 3.5.2-5	0	B 3.6.3-11	27
B 3.5.2-6	0	B 3.6.3-12	27
B 3.5.2-7	1	B 3.6.3-13	27
B 3.5.2-8	22	B 3.6.3-14	27
B 3.5.2-9	1	B 3.6.3-15	27
B 3.5.2-10	1	B 3.6.3-16	27
B 3.5.3-1	0	B 3.6.3-17	27
B 3.5.3-2	0	B.3.6.3-18	27
B 3.5.3-3	0	B.3.6.3-19	27
B 3.5.3-4	0	B 3.6.4-1	25
B 3.5.3-5	0	B 3.6.4-2	1
B 3.5.3-6	2	B 3.6.4-3	1
B 3.5.3-7	2	B 3.6.5-1	0
B 3.5.3-8	1	B 3.6.5-2	1

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B 3.6.5-4	0	B 3.7.7-5	1
B 3.6.6-1	0	B 3.7.8-1	1
B 3.6.6-2	0	B 3.7.8-2	1
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B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.1 SHUTDOWN MARGIN (SDM) - Reactor Trip Breakers Open

BASES

BACKGROUND

The reactivity control systems must be redundant and capable of holding the reactor core subcritical when shutdown under cold conditions, in accordance with GDC 26 (Ref. 1). Maintenance of the SDM ensures that postulated reactivity events will not damage the fuel. SDM requirements provide sufficient reactivity margin to ensure that acceptable fuel design limits will not be exceeded for normal shutdown and anticipated operational occurrences (AOOs). As such, the SDM defines the degree of subcriticality that would be obtained immediately following the insertion of all full strength control element assemblies (CEAs), assuming the single CEA of highest reactivity worth is fully withdrawn with Reactor Trip Breakers open. This reactivity worth is credited in establishing the required SDM.

The system design requires that two independent reactivity control systems be provided, and that one of these systems be capable of maintaining the core subcritical under cold conditions. These requirements are provided by the use of movable CEAs and soluble boric acid in the Reactor Coolant System (RCS). The CEA System provides the SDM during power operation and is capable of making the core subcritical rapidly enough to prevent exceeding acceptable fuel design limits, assuming that the CEA of highest reactivity worth remains fully withdrawn.

The soluble boron system can compensate for fuel depletion during operation and all xenon burnout reactivity changes, and maintain the reactor subcritical under cold conditions.

During power operation, SDM control is ensured by operating with the shutdown CEAs fully withdrawn and the regulating CEAs within the limits of LCO 3.1.7, "Regulating Control Element Assembly (CEA) Insertion Limits." When the unit is in the shutdown and refueling modes, the SDM requirements are met by means of adjustments to the RCS boron concentration.

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES

The minimum required SDM is assumed as an initial condition in safety analysis. The safety analysis (Ref. 2) establishes a SDM that ensures specified acceptable fuel design limits are not exceeded for normal operation and AOOs, with the assumption of the highest worth CEA stuck out following a reactor trip. Specifically, for MODE 5, the primary safety analysis that relies on the SDM limits is the boron dilution analysis.

The acceptance criteria for SDM are that specified acceptable fuel design limits are maintained. This is done by ensuring that:

a. The reactor can be made subcritical from all operating conditions, transients, and Design Basis Events;

b. The reactivity transients associated with postulated accident conditions are controllable within acceptable limits (departure from nucleate boiling ratio (DNBR), fuel centerline temperature limit AOOs, and ≤ 280 cal/gm energy deposition for the CEA ejection accident); and

c. The reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition.

The most limiting accident for the SDM requirements is based on a main steam line break (MSLB), as described in the accident analysis (Ref. 2). The increased steam flow resulting from a pipe break in the main steam system causes an increased energy removal from the affected steam generator (SG), and consequently the RCS. This results in a reduction of the reactor coolant temperature. The resultant coolant shrinkage causes a reduction in pressure. In the presence of a negative moderator temperature coefficient, this cooldown causes an increase in core reactivity. As initial RCS temperature decreases, the severity of an MSLB decreases. The most limiting MSLB, with respect to potential fuel damage before a reactor trip occurs, is a guillotine break of a main steam line inside containment initiated at the end of core life. The positive reactivity addition from the moderator temperature decrease will terminate when the affected SG boils dry, thus terminating RCS heat removal and cooldown. Following the MSLB, a post trip return to power may occur; however, no fuel damage

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

occurs as a result of the post trip return to power, and THERMAL POWER does not violate the Safety Limit (SL) requirement of SL 2.1.1.

In addition to the limiting MSLB transient, the SDM requirement for MODES 3, 4, and 5 must also protect against:

a. Inadvertent boron dilution;

b. Startup of an inactive reactor coolant pump (RCP); and

c. CEA ejection.

Each of these is discussed below.

In the inadvertent boron dilution analysis, the amount of reactivity by which the reactor is subcritical is determined by the reactivity difference between an initial subcritical boron concentration and the corresponding critical boron concentration. The initial subcritical boron concentration assumed in the analysis corresponds to the minimum SDM requirements. These two values (initial and critical boron concentrations), in conjunction with the configuration of the Reactor Coolant System (RCS) and the assumed dilution flow rate, directly affect the results of the analysis. For this reason the event is most limiting at the beginning of core life when critical boron concentrations are highest.

The startup of an inactive RCP will not result in a "cold water" criticality, even if the maximum difference in temperature exists between the SG and the core. Although this event was considered in establishing the requirements for SDM, it is not the limiting event with respect to the specification limits.

In the analysis of the CEA ejection event, maintaining SDM ensures the reactor remains subcritical following a CEA ejection and, therefore, satisfies the radially averaged enthalpy acceptance criterion considering power redistribution effects.

SHUTDOWN MARGIN is the amount by which the core is subcritical, or would be subcritical immediately following a reactor trip, considering a single malfunction resulting in the highest worth CEA failing to insert. With any full strength CEAs not capable of being fully inserted, the withdrawn reactivity worth of these CEAs must be accounted for in the determination of SDM.

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

The SDM satisfies Criterion 2 of 10 CFR 50.36 (c)(2)(ii).

LCO

The MSLB (Ref. 2) and the boron dilution (Ref. 3) accidents are the most limiting analyses that establish the SDM value of the LCO. For MSLB accidents, if the LCO is violated, there is a potential to exceed the DNBR limit and to exceed 10 CFR 100, "Reactor Site Criterion," limits (Ref. 4). For the boron dilution accident, if the LCO is violated, then the minimum required time assumed for operator action to terminate dilution may no longer be applicable.

SDM is a core physics design condition that can be ensured through CEA positioning (regulating and shutdown CEAs) and through the soluble boron concentration.

APPLICABILITY

In MODES 3, 4 and 5 with the Reactor Trip Breakers Open or the CEA drive system not capable of CEA withdrawal, the SDM requirements are applicable to provide sufficient negative reactivity to meet the assumptions of the safety analyses discussed above. In MODES 1 and 2, SDM is ensured by complying with LCO 3.1.6, "Shutdown Control Element Assembly (CEA) Insertion Limits," and LCO 3.1.7. In MODES 3, 4 and 5 with the Reactor Trip Breakers Closed, SDM is addressed by LCO 3.1.2, "SHUTDOWN MARGIN (SDM) - Reactor Trip Breakers Closed." In MODE 6, the shutdown reactivity requirements are given in LCO 3.9.1, "Boron Concentration."

ACTIONS

A.1

If the SDM requirements are not met, boration must be initiated promptly. A Completion Time of 15 minutes is adequate for an operator to correctly align and start the required systems and components. It is assumed that boration will be continued until the SDM requirements are met.

In the determination of the required combination of boration flow rate and boron concentration, there is no unique requirement that must be satisfied. Since it is imperative to raise the boron concentration of the RCS as soon as

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B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.2 SHUTDOWN MARGIN (SDM) - Reactor Trip Breakers Closed

BASES

BACKGROUND

The reactivity control systems must be redundant and capable of holding the reactor core subcritical when shut down under cold conditions, in accordance with GDC 26 (Ref. 1). Maintenance of the SDM ensures that postulated reactivity events will not damage the fuel. SDM requirements provide sufficient reactivity margin to ensure that acceptable fuel design limits will not be exceeded for normal shutdown and anticipated operational occurrences (AOOs). As such, SDM defines the degree of subcriticality that would be obtained immediately following the insertion of all full strength control element assemblies (CEAs), assuming the single CEA of highest reactivity worth is fully withdrawn.

The system design requires that two independent reactivity control systems be provided, and that one of these systems be capable of maintaining the core subcritical under cold conditions. These requirements are provided by the use of movable CEAs and soluble boric acid in the Reactor Coolant System (RCS). The CEA System provides the SDM during power operation and is capable of making the core subcritical rapidly enough to prevent exceeding the acceptable fuel design limits, assuming that the CEA of highest reactivity worth remains fully withdrawn.

The soluble boron system can compensate for fuel depletion during operation and all xenon burnout reactivity changes, and maintain the reactor subcritical under cold conditions.

During power operation, SDM control is ensured by operating with the shutdown CEAs fully withdrawn and the regulating CEAs within the limits of LCO 3.1.7, "Regulating Control Element Assembly (CEA) Insertion Limits." When the unit is in the shutdown and refueling modes, the SDM requirements are met by means of adjustments to the RCS boron concentration.

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BASES (continued)

APPLICABLE SAFETY ANALYSES

The minimum required SDM is assumed as an initial condition in safety analysis. The safety analysis (Ref. 2) establishes a SDM that ensures specified acceptable fuel design limits are not exceeded for normal operation and AOOs with the assumption of the highest worth CEA stuck out following a reactor trip. Specifically, for MODE 5, the primary safety analysis that relies on the SDM limits is the boron dilution analysis.

The acceptance criteria for SDM requirements are that the specified acceptable fuel design limits are maintained. This is done by ensuring that:

a. The reactor can be made subcritical from all operating conditions, transients, and Design Basis Events;

b. The reactivity transients associated with postulated accident conditions are controllable within acceptable limits (departure from nucleate boiling ratio, fuel centerline temperature limits for AOOs, and ≤ 280 cal/gm energy deposition for the CEA ejection accident); and

c. The reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition.

The most limiting accident for the SDM requirements is based on a main steam line break (MSLB), as described in the accident analysis (Ref. 2). The increased steam flow resulting from a pipe break in the main steam system causes an increased energy removal from the affected steam generator (SG), and consequently the RCS. This results in a reduction of the reactor coolant temperature. The resultant coolant shrinkage causes a reduction in pressure. In the presence of a negative moderator temperature coefficient, this cooldown causes an increase in core reactivity. As initial RCS temperature decreases, the severity of an MSLB decreases. The most limiting MSLB, with respect to potential fuel damage before a reactor trip occurs, is a guillotine break of a main steam line inside containment initiated at the end of core life. The positive reactivity addition from the moderator temperature decrease will terminate when the affected SG boils dry, thus terminating RCS heat removal and cooldown. Following the

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

MSLB, a post trip return to power may occur; however, no fuel damage occurs as a result of the post trip return to power, and THERMAL POWER does not violate the Safety Limit (SL) requirement of SL 2.1.1.

In addition to the limiting MSLB transient, the SDM requirement for MODES 3, 4, and 5 must also protect against:

a. Inadvertent boron dilution;

b. An uncontrolled CEA withdrawal from a subcritical condition;

c. Startup of an inactive reactor coolant pump (RCP); and

d. CEA ejection.

Each of these is discussed below.

In the inadvertent boron dilution analysis, the amount of reactivity by which the reactor is subcritical is determined by the reactivity difference between an initial subcritical boron concentration and the corresponding critical boron concentration. The initial subcritical boron concentration assumed in the analysis corresponds to the minimum SDM requirements. These two values (initial and critical boron concentrations), in conjunction with the configuration of the Reactor Coolant System (RCS) and the assumed dilution flow rate, directly affect the results of the analysis. For this reason the event is most limiting at the beginning of core life when critical boron concentrations are highest.

The withdrawal of CEAs from subcritical conditions adds reactivity to the reactor core, causing both the core power level and heat flux to increase with corresponding increases in reactor coolant temperatures and pressure. The withdrawal of CEAs also produces a time dependent redistribution of core power.

Depending on the system initial conditions and reactivity insertion rate, the uncontrolled CEA withdrawal transient is terminated by either a high power level trip or a high pressurizer pressure trip. In all cases, power level, RCS pressure, linear heat rate, and the DNBR do not exceed allowable limits.

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

The startup of an inactive RCP will not result in a "cold water" criticality, even if the maximum difference in temperature exists between the SG and the core. Although this event was considered in establishing the requirements for SDM, it is not the limiting event with respect to the specification limits.

In the analysis of the CEA ejection event, SDM alone cannot prevent reactor criticality following a CEA ejection. At temperatures less than 500 F, the K_{N-1} requirement ensures the reactor remains subcritical and, therefore, satisfies the radially averaged enthalpy acceptance criterion considering power redistribution effects. Above 500 F, Doppler reactivity feedback is sufficient to preclude the need for a specific K_{N-1} requirement.

The function of SHUTDOWN MARGIN is to ensure that the reactor remains subcritical following a design basis accident or anticipated operational occurrence. During operation in MODES 1 and 2, with k_{eff} greater than or equal to 1.0, the transient insertion limits of Specification 3.1.3.6 ensure that sufficient SHUTDOWN MARGIN is available.

SHUTDOWN MARGIN is the amount by which the core is subcritical, or would be subcritical immediately following a reactor trip, considering a single malfunction resulting in the highest worth CEA failing to insert. With any full strength CEAs not capable of being fully inserted, the withdrawn reactivity worth of the CEAs must be accounted for in the determination of SDM.

SHUTDOWN MARGIN requirements vary throughout the core life as a function of fuel depletion and reactor coolant system (RCS) cold leg temperature (T_{cold}). The most restrictive condition occurs at EOL, with T_{cold} at no load operating temperature, and is associated with a postulated steam line break accident and the resulting uncontrolled RCS cooldown. In the analysis of this accident, the specified SHUTDOWN MARGIN is required to control the reactivity transient and ensure that the fuel performance and offsite dose criteria are satisfied.

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B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.5 Control Element Assembly (CEA) Alignment

BASES

BACKGROUND

The OPERABILITY (e.g., trippability) of the shutdown and regulating CEAs is an initial assumption in all safety analyses that assume CEA insertion upon reactor trip. Maximum CEA misalignment is an initial assumption in the safety analyses that directly affects core power distributions and assumptions of available SDM.

The applicable criteria for these reactivity and power distribution design requirements are 10 CFR 50, Appendix A, GDC 10 and GDC 26 (Ref. 1) and 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Cooled Nuclear Power Plants" (Ref. 2).

Mechanical or electrical failures may cause a CEA to become inoperable or to become misaligned from its group. CEA inoperability or misalignment may cause increased power peaking, due to the asymmetric reactivity distribution and a reduction in the total available CEA worth for reactor shutdown. Therefore, CEA alignment and operability are related to core operation in design power peaking limits and the core design requirement of a minimum SDM. If a CEA(s) is discovered to be immovable but remains trippable and aligned, the CEA is considered to be OPERABLE. At anytime, if a CEA(s) is immovable, a determination of the trippability (OPERABILITY) of that CEA(s) must be made, and appropriate action taken.

Limits on CEA alignment and operability have been established, and all CEA positions are monitored and controlled during power operation to ensure that the power distribution and reactivity limits defined by the design power peaking and SDM limits are preserved.

CEAs are moved by their control element drive mechanisms (CEDMs). Each CEDM moves its CEA one step (approximately ¾ inch) at a time, but at varying rates (steps per minute) depending on the signal output from the Control Element Drive Mechanism Control System (CEDMCS).

(continued)

BASES

BACKGROUND (continued)

The CEAs are arranged into groups that are radially symmetric. Therefore, movement of the CEAs does not introduce radial asymmetries in the core power distribution. The shutdown and regulating CEAs provide the required reactivity worth for immediate reactor shutdown upon a reactor trip. The regulating CEAs also provide reactivity (power level) control during normal operation and transients. Their movement may be automatically controlled by the Reactor Regulating System. Part length or part strength CEAs are not credited in the safety analyses for shutting down the reactor, as are the regulating and shutdown groups. The part length or part strength CEAs are used solely for ASI control.

The axial position of shutdown and regulating CEAs is indicated by two separate and independent systems, which are the Pulse Counting CEA Position Indication System (described in Ref. 4) and the Reed Switch CEA Position Indication System (described in Ref. 5).

The Pulse Counting CEA Position Indicating System indicates CEA position to the actual step, if each CEA moves one step for each command signal. However, if each CEA does not follow the commands, the system will incorrectly reflect the position of the affected CEA(s). This condition may affect the operability of COLSS (refer to Section 3.2, Power Distribution Limits for the applicable actions) and should be detected by the Reed Switch Position Indication System through surveillance or alarm. Although the Reed Switch Position Indication System is less precise that the Pulse Counting CEA Position Indicating System, it is not subject to the same error mechanisms.

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES

CEA misalignment accidents are analyzed in the safety analysis (Ref. 3). The accident analysis defines CEA misoperation as any event, with the exception of sequential group withdrawals, which could result from a single malfunction in the reactivity control systems. For example, CEA misalignment may be caused by a malfunction of the CEDM, CEDMCS, or by operator error. A stuck CEA may be caused by mechanical jamming of the CEA fingers or of the gripper.

Inadvertent withdrawal of a single CEA may be caused by an electrical failure in the CEA coil power programmers. A dropped CEA could be caused by an opening of the electrical circuit of the CEDM holding coil for a full strength, part length or part strength CEA.

The acceptance criteria for addressing CEA inoperability or misalignment are that:

a. There shall be no violations of:

   1. specified acceptable fuel design limits, or

   2. Reactor Coolant System (RCS) pressure boundary integrity; and

b. The core must remain subcritical after accident transients.

Three types of misalignment are distinguished. During movement of a group, one CEA may stop moving while the other CEAs in the group continue. This condition may cause excessive power peaking. The second type of misalignment occurs if one CEA fails to insert upon a reactor trip and remains stuck fully withdrawn. This condition requires an evaluation to determine that sufficient reactivity worth is held in the remaining CEAs to meet the SDM requirement with the maximum worth CEA stuck fully withdrawn. If a CEA is stuck in the fully withdrawn position, its worth is added to the SDM requirement, since the safety analysis does not take two stuck CEAs into account. The third type of misalignment occurs when one CEA drops partially or fully into the core. This event causes an initial power reduction followed by a return towards the original power due to positive reactivity feedback from the negative moderator temperature coefficient. Increased peaking during the power increase may result in erosion of DNB margin.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

Analysis considers the case of a single CEA withdrawn approximately 10 inches from a bank inserted to its insertion limit. Satisfying limits on departure from nucleate boiling ratio (DNBR) bounds the situation when a CEA is misaligned from its group by 6.6 inches.

The effect of any misoperated CEA on the core power distribution will be assessed by the CEA calculators, and an appropriately augmented power distribution penalty factor will be supplied as input to the core protection calculators (CPCs). As the reactor core responds to the reactivity changes caused by the misoperated CEA and the ensuing reactor coolant and Doppler feedback effects, the CPCs will initiate a low DNBR or high local power density trip signal if specified acceptable fuel design limits (SAFDLs) are approached.

The part strength CEA drop incidents result in the most rapid approach to SAFDLs caused by a CEA misoperation. The accident analysis analyzed a single full strength CEA drop, a single part length CEA drop, and a part length CEA subgroup drop. The most rapid approach to the DNBR SAFDL or the fuel centerline melt SAFDL is caused by a single full strength CEA drop.

In the case of the full strength CEA drop, a prompt decrease in core average power and a distortion in radial power are initially produced, which when conservatively coupled result in local power and heat flux increases, and a decrease in DNBR. A part strength CEA drop would cause a similar reactivity response although with less of a magnitude due to the full strength CEAs having a more significant reactivity worth. As the dropped CEA is detected, core power and an appropriately augmented power distribution penalty factor are supplied to the CPCs. For plant operation within the DNBR and local power density (LPD) LCOs, DNBR and LPD trips can normally be avoided on a dropped 4-finger CEA.

For a part length or part strength CEA subgroup drop, a distortion in power distribution, and a decrease in core power are produced. As the position of the dropped part length or part strength CEA subgroup is detected, an appropriate power distribution penalty factor is supplied to

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSIS (continued)	the CPCs, and a reactor trip signal on low DNBR is generated. For the part length CEA drop, both core average power and three dimensional peak to average power density increase promptly. As the dropped part length CEA is detected, core power and an appropriately augmented power distribution penalty factor are supplied to the CPCs. CEA alignment satisfies Criteria 2 and 3 of 10 CFR 50.3 (c)(2)(ii).
LCO	The limits on part length or part strength, shutdown, and regulating CEA alignments ensure that the assumptions in the safety analysis will remain valid. The requirements on OPERABILITY ensure that upon reactor trip, the CEAs will be available and will be inserted to provide enough negative reactivity to shut down the reactor. The OPERABILITY requirements also ensure that the CEA banks maintain the correct power distribution and CEA alignment. The requirement is to maintain the CEA alignment to within 6.6 inches between any CEA and all other CEAs in its group. Failure to meet the requirements of this LCO may produce unacceptable power peaking factors, DNBR, and LHRs, or unacceptable SDMs, all of which may constitute initial conditions inconsistent with the safety analysis.
APPLICABILITY	The requirements on CEA OPERABILITY and alignment are applicable in MODES 1 and 2 because these are the only MODES in which neutron (or fission) power is generated, and the OPERABILITY (e.g., trippability) and alignment of CEAs have the potential to affect the safety of the plant. In MODES 3, 4, 5, and 6, the alignment limits do not apply because the reactor is shut down and not producing fission power. In the shutdown modes, the OPERABILITY of the shutdown and regulating CEAs has the potential to affect the required SDM, but this effect can be compensated for by an increase in the boron concentration of the RCS. See LCO 3.1.2, "SHUTDOWN MARGIN (SDM) - Reactor Trip Breakers Closed," for SDM in MODES 3, 4, and 5, and LCO 3.9.1, "Boron Concentration," for boron concentration requirements during refueling.

(continued)

BASES (continued)

ACTIONS

A.1 and A.2

A CEA may become misaligned, yet remain trippable. In this condition, the CEA can still perform its required function of adding negative reactivity should a reactor trip be necessary.

If one or more CEAs (regulating, shutdown, part length, or part strength) are misaligned by 6.6 inches and ≤ 9.9 inches but trippable, or one CEA misaligned by > 9.9 inches but trippable, continued operation in MODES 1 and 2 may continue, provided, within 1 hour, the power is reduced in accordance with the limits in the COLR, and within 2 hours CEA alignment is restored. Regulating and part length or part strength CEA alignment can be restored by either aligning the misaligned CEA(s) to within 6.6 inches of its group or aligning the misaligned CEA's group to within 6.6 inches of the misaligned CEA(s). Shutdown CEA alignment can be restored by aligning the misaligned CEA(s) to within 6.6 inches of its group.

Xenon redistribution in the core starts to occur as soon as a CEA becomes misaligned. Reducing THERMAL POWER in accordance with the limits in the COLR ensures acceptable power distributions are maintained (Ref. 3). For small misalignments (< 9.9 inches) of the CEAs, there is:

a. A small effect on the time dependent long term power distributions relative to those used in generating LCOs and limiting safety system settings (LSSS) setpoints;

b. A negligible effect on the available SDM; and

c. A small effect on the ejected CEA worth used in the accident analysis.

With a large CEA misalignment (≥ 9.9 inches), however, this misalignment would cause distortion of the core power distribution. This distortion may, in turn, have a significant effect on the time dependent, long term power distributions relative to those used in generating LCOs and LSSS setpoints. The effect on the available SDM and the ejected CEA worth used in the accident analysis remain small.

Therefore, this condition is limited to the single CEA misalignment, while still allowing 2 hours for recovery.

(continued)

BASES

ACTIONS

A.1 and A.2 (continued)

In both cases, a 2 hour time period is sufficient to:

a. Identify cause of a misaligned CEA;

b. Take appropriate corrective action to realign the CEAs; and

c. Minimize the effects of xenon redistribution.

The CEA must be returned to OPERABLE status within 2 hours. If a CEA misalignment results in the COLSS programs being declared INOPERABLE, refer to Section 3.2 Power Distribution Limits for applicable actions.

B.1 and B.2

At least two of the following three CEA position indicator channels shall be OPERABLE for each CEA:

a. CEA Reed Switch Position Transmitter (RSPT 1) with the capability of determining the absolute CEA positions within 5.2 inches,

b. CEA Reed Switch Position Transmitter (RSPT 2) with the capability of determining the absolute CEA positions within 5.2 inches, and

c. The CEA pulse counting position indicator channel.

If only one CEA position indicator channel is OPERABLE, continued operation in MODES 1 and 2 may continue, provided, within 6 hours, at least two position indicator channels are returned to OPERABLE status, or within 6 hours and once per 12 hours, verify that the CEA group with the inoperable position indicators are either fully withdrawn or fully inserted while maintaining the insertion limits of LCO 3.1.6, LCO 3.1.7 and LCO 3.1.8. CEAs are fully withdrawn (Full Out) when withdrawn to at least 144.75 inches.

C.1

If a Required Action or associated Completion Time of Condition A or Condition B is not met, or if one or more regulating or shutdown CEAs are untrippable (immovable as a result of excessive friction or mechanical interference or

(continued)

BASES

ACTIONS

C.1 (continued)

known to be untrippable), the unit is required to be brought to MODE 3. By being brought to MODE 3, the unit is brought outside its MODE of applicability.

When a Required Action cannot be completed within the required Completion Time, a controlled shutdown should be commenced. The allowed Completion Time of 6 hours is reasonable, based on operating experience, for reaching MODE 3 from full power conditions in an orderly manner and without challenging plant systems.

If a full strength CEA is untrippable, it is not available for reactivity insertion during a reactor trip. With an untrippable CEA, meeting the insertion limits of LCO 3.1.6, "Shutdown Control Element Assembly (CEA) Insertion Limits," and LCO 3.1.7, "Regulating Control Element Assembly (CEA) Insertion Limits," does not ensure that adequate SDM exists. Therefore, the plant must be shut down in order to evaluate the SDM required boron concentration and power level for critical operation. Continued operation is allowed with untrippable part length or part strength CEAs if the alignment and insertion limits are met.

Continued operation is not allowed with one or more full length CEAs untrippable. This is because these cases are indicative of a loss of SDM and power distribution, and a loss of safety function, respectively.

D.1

Continued operation is not allowed in the case of more than one CEA misaligned from any other CEA in its group by > 9.9 inches. For example, two CEAs in a group misaligned from any other CEA in that group by > 9.9 inches, or more than one CEA group that has a least one CEA misaligned from any other CEA in that group by > 9.9 inches. This is indicative of a loss of power distribution and a loss of safety function, respectively. Multiple CEA misalignments should result in automatic protective action. Therefore, with two or more CEAs misaligned more than 9.9 inches, this could result in a situation outside the design basis and immediate action would be required to prevent any potential fuel damage. Immediately opening the reactor trip breakers minimizes these effects.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.5.1

Verification that individual CEA positions are within 6.6 inches (indicated reed switch positions) of all other CEAs in the group at a 12 hour Frequency allows the operator to detect a CEA that is beginning to deviate from its expected position. The specified Frequency takes into account other CEA position information that is continuously available to the operator in the control room, so that during actual CEA motion, deviations can immediately be detected.

SR 3.1.5.2

OPERABILITY of at least two CEA position indicator channels is required to determine CEA positions, and thereby ensure compliance with the CEA alignment and insertion limits. The CEA full in and full out limits provide an additional independent means for determining the CEA positions when the CEAs are at either their fully inserted or fully withdrawn positions.

SR 3.1.5.3

Verifying each full strength CEA is trippable would require that each CEA be tripped. In MODES 1 and 2 tripping each full strength CEA would result in radial or axial power tilts, or oscillations. Therefore individual full strength CEAs are exercised every 92 days to provide increased confidence that all full strength CEAs continue to be trippable, even if they are not regularly tripped. A movement of 5 inches is adequate to demonstrate motion without exceeding the alignment limit when only one full strength CEA is being moved. The 92 day Frequency takes into consideration other information available to the operator in the control room and other surveillances being performed more frequently, which add to the determination of OPERABILITY of the CEAs (Ref. 3). Between required performances of SR 3.1.5.3, if a CEA(s) is discovered to be immovable but remains trippable and aligned, the CEA is considered to be OPERABLE. At anytime, if a CEA(s) is immovable, a determination of the trippability (OPERABILITY) of that CEA(s) must be made, and appropriate action taken.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.1.5.4

Performance of a CHANNEL FUNCTIONAL TEST of each reed switch position transmitter channel ensures the channel is OPERABLE and capable of indicating CEA position. Since this test must be performed when the reactor is shut down, an 18 month Frequency to be coincident with refueling outage was selected. Operating experience has shown that these components usually pass this Surveillance when performed at a Frequency of once every 18 months. Furthermore, the Frequency takes into account other factors, which determine the OPERABILITY of the CEA Reed Switch Indication System. These factors include:

a. Other, more frequently performed surveillances that help to verify OPERABILITY;

b. On-line diagnostics performed automatically by the CPCs, CEACs, and the Plant Computer which include CEA position comparisons and sensor validation; and

c. The CHANNEL CALIBRATIONs for the CPCs (SR 3.3.1.9) and CEACs (SR 3.3.3.4) input channels that are performed at 18 month intervals and is an overlapping test.

SR 3.1.5.5

Verification of full strength CEA drop times determines that the maximum CEA drop time permitted is consistent with the assumed drop time used in the safety analysis (Ref. 3). Measuring drop times prior to reactor criticality, after reactor vessel head removal, ensures the reactor internals and CEDM will not interfere with CEA motion or drop time, and that no degradation in these systems has occurred that would adversely affect CEA motion or drop time. Individual CEAs whose drop times are greater than safety analysis assumptions are not OPERABLE. This SR is performed prior to criticality due to the plant conditions needed to perform the SR and the potential for an unplanned plant transient if the Surveillance were performed with the reactor at power.

The 4 second CEA drop time is the maximum time it takes for a fully withdrawn individual full strength CEA to reach its 90% insertion position when electrical power is interrupted to the CEA drive mechanism with RCS T_{cold} greater than or equal to 550°F and all reactor coolant pumps operating.

(continued)

BASES

The CEA drop time of full strength CEAs shall also be demonstrated through measurement prior to reactor criticality for specifically affected individual CEAs following any maintenance on or modification to the CEA drive system which could affect the drop time of those specific CEAs.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 10 and GDC 26.
2. 10 CFR 50.46.
3. UFSAR, Section 15.4.
4. UFSAR, Section 7.7.1.3.2.3.
5. UFSAR, Section 7.5.1.1.4.

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BASES

BACKGROUND (continued)

event of a CEA ejection accident, and the shutdown and regulating bank insertion limits ensure the required SDM is maintained.

Operation within the subject LCO limits will prevent fuel cladding failures that would breach the primary fission product barrier and release fission products to the reactor coolant in the event of a LOCA, loss of flow, ejected CEA, or other accident requiring termination by a Reactor Protection System trip function.

APPLICABLE SAFETY ANALYSES

The fuel cladding must not sustain damage as a result of normal operation (Condition I) and anticipated operational occurrences (Condition II). The acceptance criteria for the regulating CEA insertion, part length or part strength CEA insertion, ASI, and T_q LCOs preclude core power distributions from occurring that would violate the following fuel design criteria:

a. During a large break LOCA, the peak cladding temperature must not exceed a limit of 2200°F, 10 CFR 50.46 (Ref. 2);

b. During CEA misoperation events, there must be at least a 95% probability at a 95% confidence level (the 95/95 DNB criterion) that the hot fuel rod in the core does not experience a DNB condition;

c. During an ejected CEA accident, the fission energy input to the fuel must not exceed 280 cal/gm (Ref. 3); and

d. The CEAs must be capable of shutting down the reactor with a minimum required SDM, with the highest worth CEA stuck fully withdrawn, GDC 26 (Ref. 1).

Regulating CEA position, ASI, and T_q are process variables that together characterize and control the three dimensional power distribution of the reactor core.

Fuel cladding damage does not occur when the core is operated outside these LCOs during normal operation. However, fuel cladding damage could result, should an

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

accident occur with simultaneous violation of one or more of these LCOs. Changes in the power distribution can cause increased power peaking and corresponding increased local LHRs.

The SDM requirement is ensured by limiting the regulating and shutdown CEA insertion limits, so that the allowable inserted worth of the CEAs is such that sufficient reactivity is available in the CEAs to shut down the reactor to hot zero power with a reactivity margin that assumes the maximum worth CEA remains fully withdrawn upon trip (Ref. 4).

The most limiting SDM requirements for MODE 1 and 2 conditions at BOC are determined by the requirements of several transients, e.g., Loss of Flow, Seized Rotor, etc. However, the most limiting SDM requirements for MODES 1 and 2 at EOC come from just one transient, Steam Line Break (SLB). The requirements of the SLB event at EOC for both the full power and no load conditions are significantly larger than those of any other event at that time in cycle and, also, considerably larger than the most limiting requirements at BOC.

Although the most limiting SDM requirements at EOC are much larger that those at BOC, the available SDMs obtained via the scramming of the CEAs are also substantially larger due to the much lower boron concentration at EOC. To verify that adequate SDMs are available throughout the cycle to satisfy the changing requirements, calculations are performed at both BOC and EOC. It has been determined that calculations at these two times in cycle are sufficient since the differences between available SDMs and the limiting SDM requirements are the smallest at these times in the cycle. The measurement of CEA bank worth performed as part of the Startup Testing Program demonstrates that the core has expected shutdown capability. Consequently, adherence to LCOs 3.1.6 and 3.1.7 provides assurance that the available SDMs at any time in cycle will exceed the limiting SDM requirements at that time in the cycle.

(continued)

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.8 Part Length or Part Strength Control Element Assembly (CEA) Insertion Limits

BASES

BACKGROUND

The insertion limits of the part length or part strength CEAs are initial assumptions in the safety analyses for CEA misoperation events. The insertion limits directly affect core power distributions. The applicable criteria for these power distribution design requirements are 10 CFR 50, Appendix A, GDC 10, "Reactor Design" (Ref. 1), and 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Plants" (Ref. 2). Limits on part length or part strength CEA insertion have been established, and all CEA positions are monitored and controlled during power operation to ensure that the power distribution defined by the design power peaking limits is preserved.

The part length or part strength CEAs are used for axial power shape control of the reactor. The positions of the part length or part strength CEAs are manually controlled. They are capable of changing reactivity very quickly (compared to borating or diluting).

The power density at any point in the core must be limited to maintain specified acceptable fuel design limits, including limits that preserve the criteria specified in 10 CFR 50.46 (Ref. 2). Together, LCO 3.1.7. "Regulating Control Element Assembly (CEA) Insertion Limits"; LCO 3.1.8; LCO 3.2.4, "Departure From Nucleate Boiling Ratio (DNBR)"; and LCO 3.2.5, "AXIAL SHAPE INDEX (ASI)," provide limits on control component operation and on monitored process variables to ensure the core operates within the linear heat rate (LHR) (LCO 3.2.1, "Linear Heat Rate (LHR)"); planar peaking factor (F_{xy}) (LCO 3.2.2, "Planar Radial Peaking Factors (F_{xy})"); and LCO 3.2.4 limits in the COLR.

Operation within the limits given in the COLR prevents power peaks that would exceed the loss of coolant accident (LOCA) limits derived by the Emergency Core Cooling Systems analysis. Operation within the F_{xy} and departure from nucleate boiling (DNB) limits given in the COLR prevents DNB during a loss of forced reactor coolant flow accident.

(continued)

BASES

BACKGROUND (continued)

The establishment of limiting safety system settings and LCOs requires that the expected long and short term behavior of the radial peaking factors be determined. The long term behavior relates to the variation of the steady state radial peaking factors with core burnup; it is affected by the amount of CEA insertion assumed, the portion of a burnup cycle over which such insertion is assumed, and the expected power level variation throughout the cycle. The short term behavior relates to transient perturbations to the steady state radial peaks due to radial xenon redistribution. The magnitudes of such perturbations depend upon the expected use of the CEAs during anticipated power reductions and load maneuvering. Analyses are performed, based on the expected mode of operation of the Nuclear Steam Supply System (base loaded, maneuvering, etc.). From these analyses, CEA insertions are determined, and a consistent set of radial peaking factors are defined. The long term (steady state) and short term insertion limits are determined, based upon the assumed mode of operation used in the analyses; they provide a means of preserving the assumptions on CEA insertions used. The long and short term insertion limits of LCO 3.1.8 are specified for the plant, which has been designed primarily for base loaded operation, but has the ability to accommodate a limited amount of load maneuvering.

APPLICABLE SAFETY ANALYSES

The fuel cladding must not sustain damage as a result of normal operation (Condition I) and anticipated operational occurrences (Condition II). The regulating CEA insertion, part length or part strength CEA insertion, ASI, and T_q LCOs preclude core power distributions from occurring that would violate the following fuel design criteria:

a. During a large break LOCA, the peak cladding temperature must not exceed 2200°F (Ref. 2);

b. During a loss of forced reactor coolant flow accident, there must be at least a 95% probability at a 95% confidence level (the 95/95 DNB criterion) that the hot fuel rod in the core does not experience a DNB condition;

c. During an ejected CEA accident, the fission energy input to the fuel must not exceed 280 cal/gm (Ref. 3); and

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

d. The CEAs must be capable of shutting down the reactor with a minimum required SDM, with the highest worth CEA stuck fully withdrawn, GDC 26 (Ref. 1).

Regulating CEA position, part length or part strength CEA position, ASI, and T_q are process variables that together characterize and control the three dimensional power distribution of the reactor core.

Fuel cladding damage does not occur when the core is operated outside these LCOs during normal operation. However, fuel cladding damage could result, should an accident occur with simultaneous violation of one or more of these LCOs. Changes in the power distribution can cause increased power peaking and corresponding increased local LHRs.

The part length or part strength CEA insertion limits satisfy Criterion 2 of 10 CFR 50.36 (c)(2)(ii). The part length or part strength CEAs are required due to the potential peaking factor violations that could occur if part length or part strength CEAs exceed insertion limits.

LCO

The limits on part length or part strength CEA insertion, as defined in the COLR, must be maintained because they serve the function of preserving power distribution.

APPLICABILITY

The part length or part strength insertion limits shall be maintained with the reactor in MODES 1 and 2. These limits must be maintained, since they preserve the assumed power distribution. Applicability in lower MODES is not required, since the power distribution assumptions would not be exceeded in these MODES.

(continued)

BASES (continued)

ACTIONS

A.1, A.2 and B.1

If the part length or part strength CEA groups are inserted beyond the following limits, flux patterns begin to develop that are outside the range assumed for long term fuel burnup;

1) Transient insertion limits;

2) Between the long term (steady-state) insertion limit and the transient insertion limit for;

    a) 7 or more effective full power days (EFPD) out of any 30 EFPD period;

    b) 14 EFPD or more out of any 365 EFPD period.

If allowed to continue beyond this limit, the peaking factors assumed as initial conditions in the accident analysis may be invalidated (Ref. 4). Restoring the CEAs to within limits or reducing THERMAL POWER to that fraction of RTP that is allowed by CEA group position, using the limits specified in the COLR, ensures that acceptable peaking factors are maintained.

Since these effects are cumulative, actions are provided to limit the total time the part length or part strength CEAs can be out of limits in any 30 EFPD or 365 EFPD period. Since the cumulative out of limit times are in days, an additional Completion Time of 2 hours is reasonable for restoring the part length or part strength CEAs to within the allowed limits.

C.1

When a Required Action cannot be completed within the required Completion Time, a controlled shutdown should commence. A Completion Time of 6 hours is reasonable, based on operating experience, for reaching Mode 3 from full power conditions in an orderly manner and without challenging plant systems.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.1.8.1

Verification of each part length or part strength CEA group position every 12 hours is sufficient to detect CEA positions that may approach the limits, and provide the operator with time to undertake the Required Action(s), should insertion limits be found to be exceeded. The 12 hour frequency also takes into account the indication provided by the power dependent insertion limit alarm circuit and other information about CEA group positions available to the operator in the control room.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 10 and GDC 26.
2. 10 CFR 50.46.
3. Regulatory Guide 1.77, Rev. 0, May 1974.
4. UFSAR, Section 15.4.

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BASES (continued)

ACTIONS

A.1

With any CEA not fully inserted and less than the minimum required reactivity equivalent available for insertion, or with all CEAs inserted and the reactor subcritical by less than the reactivity equivalent of the highest worth withdrawn CEA, restoration of the minimum shutdown reactivity requirements must be accomplished by increasing the RCS boron concentration. The required Completion Time of 15 minutes for initiating boration allows the operator sufficient time to align the valves and start the boric acid pumps and is consistent with the Completion Time of LCO 3.1.2.

In the determination of the required combination of boration flow rate and boron concentration, there is no unique requirement that must be satisfied. Since it is imperative to raise the boron concentration of the RCS as soon as possible, the boron concentration should be a highly concentrated solution, such as that normally found in the refueling water tank. The operator should borate with the best source available for the plant conditions.

In determining the boration flow rate the time in core life must be considered. For instance, the most difficult time in core life to increase the RCS boron concentration is at the beginning of cycle, when boron concentration may approach or exceed 2000 ppm. Assuming that a value of 1% $\Delta k/k$ must be recovered and a boration flow rate of 26 gpm, it is possible to increase the boron concentration of the RCS by 100 ppm in approximately 35 minutes with a 4000 ppm source. If a boron worth of 10 pcm/ppm is assumed, this combination of parameters will increase the SDM by 1% $\Delta k/k$. These boration parameters of 26 gpm and 4000 ppm represent typical values and are provided for the purpose of offering a specific example.

SURVEILLANCE REQUIREMENTS

SR 3.1.9.1

Verification of the position of each partially or fully withdrawn full strength, part length, or part strength CEA is necessary to ensure that the minimum negative reactivity requirements for insertion on a trip are preserved. A 2 hour Frequency is sufficient for the operator to verify that each CEA position is within the acceptance criteria.

(continued)

BASES (continued)

SR 3.1.9.2

Prior demonstration that each CEA to be withdrawn from the core during PHYSICS TESTS is capable of full insertion, when tripped from at least a 50% withdrawn position, ensures that the CEA will insert on a trip signal. The 7 day Frequency ensures that the CEAs are OPERABLE prior to reducing SDM requirements to less than the limits of LCO 3.1.2.

SR 3.1.9.3

During MODE 3, verification that the reactor is subcritical by at least the reactivity equivalent of the highest estimated CEA worth ensures that the minimum negative reactivity requirements are preserved. The negative reactivity requirements are verified by performing a reactivity balance calculation, considering the listed reactivity effects:

a. RCS boron concentration;

b. CEA positions;

c. RCS average temperature;

d. Fuel burnup based on gross thermal energy generation;

e. Xenon concentration; and

f. Samarium concentration.

The Frequency of 2 hours is based on the generally slow change in required boron concentration, and it allows sufficient time for the operator to collect the required data.

REFERENCES

1. 10 CFR 50, Appendix B, Section XI.
2. 10 CFR 50.59.
3. Regulatory Guide 1.68, Revision 2, August 1978.
4. ANSI/ANS-19.6.1-1985, December 13, 1985.
5. UFSAR, Chapter 14.
6. 10 CFR 50.46.
7. UFSAR, Chapter 15.

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.10 Special Test Exceptions (STE) - MODES 1 and 2

BASES

BACKGROUND

The primary purpose of these MODES 1 and 2 STEs is to permit relaxation of existing LCOs to allow the performance of certain PHYSICS TESTS. These tests are conducted to determine specific reactor core characteristics.

Section XI of 10 CFR 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Processing Plants" (Ref. 1), requires that a test program be established to ensure that structures, systems, and components will perform satisfactorily in service. All functions necessary to ensure that specified design conditions are not exceeded during normal operation and anticipated operational occurrences must be tested. Testing is required as an integral part of the design, fabrication, construction, and operation of the power plant. Requirements for notification of the NRC, for the purpose of conducting tests and experiments, are specified in 10 CFR 50.59, "Changes, Tests, and Experiments" (Ref. 2).

The key objectives of a test program are to (Ref. 3):

a. Ensure that the facility has been adequately designed;

b. Validate the analytical models used in design and analysis;

c. Verify assumptions used for predicting plant response;

d. Ensure that installation of equipment in the facility has been accomplished in accordance with design; and

e. Verify that operating and emergency procedures are adequate.

To accomplish these objectives, testing is required prior to initial criticality, after each refueling shutdown, and during startup, low power operation, power ascension, and at power operation. The PHYSICS TESTS requirements for reload fuel cycles ensure that the operating characteristics of the core are consistent with the design predictions and that the core can be operated as designed (Ref. 4).

(continued)

BASES

BACKGROUND (continued)

PHYSICS TESTS procedures are written and approved in accordance with established formats. The procedures include all information necessary to permit a detailed execution of testing required to ensure that design intent is met. PHYSICS TESTS are performed in accordance with these procedures and test results are approved prior to continued power escalation and long term power operation.

Examples of PHYSICS TESTS include determination of critical boron concentration, CEA group worth, reactivity coefficients, flux symmetry, and core power distribution.

APPLICABLE SAFETY ANALYSES

It is acceptable to suspend certain LCOs for PHYSICS TESTS because fuel damage criteria are not exceeded. Even if an accident occurs during PHYSICS TESTS with one or more LCOs suspended, fuel damage criteria are preserved because the limits on power distribution and shutdown capability are maintained during PHYSICS TESTS.

Reference 5 defines requirements for initial testing of the facility, including PHYSICS TESTS. Requirements for reload fuel cycle PHYSICS TESTS are defined in ANSI/ANS-19.6.1-1985 (Ref. 4). Although these PHYSICS TESTS are generally accomplished within the limits of all LCOs, conditions may occur when one or more LCOs must be suspended to make completion of PHYSICS TESTS possible or practical. This is acceptable as long as the fuel design criteria are not violated. As long as the linear heat rate (LHR) remains within its limit, fuel design criteria are preserved.

In this test, the following LCOs are suspended:

LCO 3.1.4, "Moderator Temperature Coefficient (MTC)";
LCO 3.1.5, "Control Element Assembly (CEA) Alignment";
LCO 3.1.6, "Shutdown Control Element Assembly (CEA) Insertion Limits";
LCO 3.1.7, "Regulating Control Element Assembly (CEA) Insertion Limits (F^T_{xy})";
LCO 3.1.8, "Part Length or Part Strength Control Element Assembly (CEA) Insertion Limits";
LCO 3.2.2, "Planar Radial Peaking Factors";
LCO 3.2.3, "AZIMUTHAL POWER TILT (T_q)";
LCO 3.2.5, "AXIAL SHAPE INDEX (ASI)"; and
LCO 3.3.3, "Control Element Assembly Calculators (CEACs)".

(continued)

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.11 Special Test Exceptions (STE) – Reactivity Coefficient Testing

BASES

BACKGROUND

The primary purpose of Reactivity Coefficient Testing is to permit relaxation of existing LCOs to allow the performance of certain PHYSICS TESTS. These tests are conducted to determine isothermal temperature coefficient, moderator temperature coefficient, and power coefficient.

Section XI of 10 CFR 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Processing Plants" (Ref. 1), requires that a test program be established to ensure that structures, systems, and components will perform satisfactorily in service. All functions necessary to ensure that specified design conditions are not exceeded during normal operation and anticipated operational occurrences must be tested. Testing is required as an integral part of the design, fabrication, construction, and operation of the power plant. Requirements for notification of the NRC, for the purpose of conducting tests and experiments, are specified in 10 CFR 50.59, "Changes, Tests, and Experiments" (Ref. 2).

The key objectives of a test program are to (Ref. 3):

a. Ensure that the facility has been adequately designed;

b. Validate the analytical models used in design and analysis;

c. Verify assumptions used for predicting plant response;

d. Ensure that installation of equipment in the facility has been accomplished in accordance with design; and

e. Verify that operating and emergency procedures are adequate.

To accomplish these objectives, testing is required prior to initial criticality, after each refueling shutdown, and during startup, low power operation, power ascension, and at power operation.

(continued)

BASES

BACKGROUND (continued)	The PHYSICS TESTS requirements for reload fuel cycles ensure that the operating characteristics of the core are consistent with the design predictions and that the core can be operated as designed (Ref. 4). PHYSICS TESTS procedures are written and approved in accordance with established formats. The procedures include all information necessary to permit a detailed execution of testing required to ensure that design intent is met. PHYSICS TESTS are performed in accordance with these procedures and test results are approved prior to continued power escalation and long term power operation. Examples of PHYSICS TESTS include determination of critical boron concentration, CEA group worth, reactivity coefficients, flux symmetry, and core power distribution.
APPLICABLE SAFETY ANALYSES	It is acceptable to suspend certain LCOs for PHYSICS TESTS because fuel damage criteria are not exceeded. Even if an accident occurs during PHYSICS TESTS with one or more LCOs suspended, fuel damage criteria are preserved because the limits on power distribution and shutdown capability are maintained during PHYSICS TESTS. Reference 5 defines requirements for initial testing of the facility, including PHYSICS TESTS. Requirements for reload fuel cycle PHYSICS TESTS are defined in ANSI/ANS-19.6.1-1985 (Ref. 4). Although these PHYSICS TESTS are generally accomplished within the limits of all LCOs, conditions may occur when one or more LCOs must be suspended to make completion of PHYSICS TESTS possible or practical. This is acceptable as long as the fuel design criteria are not violated. As long as the linear heat rate (LHR) and DNBR remain within its limits, fuel design criteria are preserved. In this test, the following LCOs are suspended: LCO 3.1.7, "Regulating Control Element Assembly (CEA) Insertion Limits"; LCO 3.1.8, "Part Length or Part Strength Control Element Assembly (CEA) Insertion Limits"; and LCO 3.4.1, "RCS Pressure, Temperature, and Flow Limits" (LCO 3.4.1.b, RCS Cold Leg Temperature only).

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

The safety analysis (Ref. 6) requires that the LHR and the departure from nucleate boiling (DNB) parameter be maintained within limits. The associated trip setpoints are required to ensure these limits are maintained.

The individual LCOs governing CEA group height, insertion and alignment, ASI, total planar radial peaking factor, total integrated radial peaking factor, and T_q, preserve the LHR limits. Additionally, the LCOs governing Reactor Coolant System (RCS) flow, reactor inlet temperature (T_c), and pressurizer pressure contribute to maintaining DNB parameter limits. The initial condition criteria for accidents sensitive to core power distribution are preserved by the LHR and DNB parameter limits. The criteria for the loss of coolant accident (LOCA) are specified in 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors" (Ref. 7). The criteria for the loss of forced reactor coolant flow accident are specified in Reference 7. Operation within the LHR limit preserves the LOCA criteria: operation within the DNB parameter limits preserves the loss of flow criteria.

During PHYSICS TESTS, one or more of the LCOs that normally preserve the LHR and DNB parameter limits may be suspended. The results of the accident analysis are not adversely impacted, however, if LHR and DNB parameters are verified to be within their limits while the LCOs are suspended. Therefore, SRs are placed as necessary to ensure that LHR and DNB parameters remain within limits during PHYSICS TESTS. Performance of these Surveillances allows PHYSICS TESTS to be conducted without decreasing the margin of safety.

PHYSICS TESTS include measurement of core parameters or exercise of control components that affect process variables. Among the process variables involved are total planar radial peaking factor, total integrated radial peaking factor, T_q, and ASI, which represent initial condition input (power peaking) to the accident analysis. Also involved are the shutdown and regulating CEAs, which affect power peaking and are required for shutdown of the reactor. The limits for these variables are specified for each fuel cycle in the COLR.

(continued)

BASES

APPLICABLE SAFETY ANALYSIS (continued)	PHYSICS TESTS meet the criteria for inclusion in the Technical Specifications, since the component and process variable LCOs suspended during PHYSICS TESTS meet Criteria 1, 2, and 3 of 10 CFR 50.36 (c)(2)(ii).
LCO	This LCO permits Part Length or Part Strength CEAs and Regulating CEAs to be positioned outside of their normal group heights and insertion limits, and RCS cold leg temperature to be outside its limits during the performance of PHYSICS TESTS. These PHYSICS TESTS are required to determine the isothermal temperature coefficient (ITC), MTC, and power coefficient. The requirements of LCO 3.1.7, LCO 3.1.8, and LCO 3.4.1, (for RCS cold leg temperature only) may be suspended during the performance of PHYSICS TESTS provided COLSS is in service.
APPLICABILITY	This LCO is applicable in MODE 1 with THERMAL POWER > 20% RTP because the reactor must be critical at THERMAL POWER levels > 20% RTP to perform the PHYSICS TESTS described in the LCO section.
ACTIONS	A.1 With the LHR or DNBR outside the limits specified in the COLR, adequate safety margin is not assured and power must be reduced to restore LHR and DNBR to within limits. The required Completion Time of 15 minutes ensure prompt action is taken to restore LHR or DNBR to within limits.

(continued)

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.1 Linear Heat Rate (LHR)

BASES

BACKGROUND

The purpose of this LCO is to limit the core power distribution to the initial values assumed in the accident analyses. Operation within the limits imposed by this LCO limits or prevents potential fuel cladding failures that could breach the primary fission product barrier and release fission products to the reactor coolant in the event of a Loss Of Coolant Accident (LOCA), ejected Control Element Assembly (CEA) accident, or other postulated accident requiring termination by a Reactor Protection System (RPS) trip function. This LCO limits the damage to the fuel cladding during an accident by ensuring that the plant is operating within acceptable bounding conditions at the onset of a transient.

Methods of controlling the power distribution include:

a. Using full strength, part length, or part strength CEAs to alter the axial power distribution;

b. Decreasing CEA insertion by boration, thereby improving the radial power distribution; and

c. Correcting off optimum conditions (e.g., a CEA drop or misoperation of the unit) that cause margin degradations.

The core power distribution is controlled so that, in conjunction with other core operating parameters (e.g., CEA insertion and alignment limits), the power distribution does not result in violation of this LCO. The limiting safety system settings and this LCO are based on the accident analyses (Refs. 1 and 2), so that specified acceptable fuel design limits are not exceeded as a result of Anticipated Operational Occurrences (AOOs), and the limits of acceptable consequences are not exceeded for other postulated accidents.

Limiting power distribution skewing over time also minimizes xenon distribution skewing, which is a significant factor in controlling the axial power distribution.

(continued)

BASES

BACKGROUND (continued)

Power distribution is a product of multiple parameters, various combinations of which may produce acceptable power distributions. Operation within the design limits of power distribution is accomplished by generating operating limits on the LHR and Departure from Nucleate Boiling (DNB).

Proximity to the DNB condition is expressed by the Departure from Nucleate Boiling Ratio (DNBR), defined as the ratio of the cladding surface heat flux required to cause DNB to the actual cladding surface heat flux. The minimum DNBR value during both normal operation and AOOs is the DNBR Safety Limit as calculated by the CE-1 Correlation (Ref. 3) and corrected for such factors as rod bow and grid spacers. It is accepted as an appropriate margin to DNB for all operating conditions.

There are two systems that monitor core power distribution online: the Core Operating Limit Supervisory System (COLSS) and the Core Protection Calculators (CPCs). The COLSS and CPCs that monitor the core power distribution are capable of verifying that the LHR and the DNBR do not exceed their limits. The COLSS performs this function by continuously monitoring the core power distribution and calculating core power operating limits corresponding to the allowable peak LHR and DNBR. The CPCs perform this function by continuously calculating an actual value of DNBR and Local Power Density (LPD) for comparison with the respective trip setpoints.

The COLSS indicates continuously to the operator how far the core is from the operating limits and provides an audible alarm if an operating limit is exceeded. Such a condition signifies a reduction in the capability of the plant to withstand an anticipated transient, but does not necessarily imply an immediate violation of fuel design limits. If the margin to fuel design limits continues to decrease, the RPS ensures that the specified acceptable fuel design limits are not exceeded by initiating a reactor trip.

The COLSS continually generates an assessment of the calculated margin for specified LHR and DNBR limits. The data required for these assessments include measured incore neutron flux, CEA positions, and Reactor Coolant System (RCS) inlet temperature, pressure, and flow.

(continued)

BASES

BACKGROUND (continued)

In addition to the monitoring performed by the COLSS, the RPS (via the CPCs) continually infers the core power distribution and thermal margins by processing reactor coolant data, signals from excore neutron flux detectors, and input from redundant reed switch assemblies that indicate CEA positions. In this case, the CPCs assume a minimum core power of 20% RTP because the power range excore neutron flux detecting system is inaccurate below this power level. If power distribution or other parameters are perturbed as a result of an AOO, the high LPD or low DNBR trips in the RPS initiate a reactor trip prior to exceeding fuel design limits.

The LHR and DNBR algorithms are valid within the limits on ASI, F_{xy} and T_q. These limits are obtained directly from initial core or reload analysis.

APPLICABLE SAFETY ANALYSES

The fuel cladding must not sustain damage as a result of normal operation or AOOs (Ref. 4).

The power distribution and CEA insertion and alignment LCOs prevent core power distributions from reaching levels that violate the following fuel design criteria:

a. During a LOCA, peak cladding temperature must not exceed 2200°F (Ref. 5);

b. During a loss of flow accident, there must be at least 95% probability at the 95% confidence level (the 95/95 DNB criterion) that the hot fuel rod in the core does not experience a DNB condition (Ref. 4);

c. During an ejected CEA accident, the fission energy input to the fuel must not exceed 280 al/gm (Ref. 6); and

d. The control rods (excluding part length or part strength rods) must be capable of shutting down the reactor with a minimum required SDM with the highest worth control rod stuck fully withdrawn (Ref. 7).

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

The power density at any point in the core must be limited to maintain the fuel design criteria (Refs. 4 and 5). This is accomplished by maintaining the power distribution and reactor coolant conditions so that the peak LHR and DNB parameters are within operating limits supported by the accident analyses (Ref. 1) with due regard for the correlations between measured quantities, the power distribution, and uncertainties in determining the power distribution.

Fuel cladding failure during a LOCA is limited by restricting the maximum Linear Heat Generation Rate (LHGR) so that the peak cladding temperature does not exceed 2200°F (Ref. 5). Peak cladding temperatures exceeding 2200°F cause severe cladding failure by oxidation due to a Zircaloy water reaction.

The LCOs governing the LHR, ASI, CEAs, and RCS ensure that these criteria are met as long as the core is operated within the ASI and F_{xy} limits specified in the COLR, and within the T_q limits. The latter are process variables that characterize the three dimensional power distribution of the reactor core. Operation within the limits for these variables ensures that their actual values are within the ranges used in the accident analyses (Ref. 1).

Fuel cladding damage does not occur from conditions outside the limits of these LCOs during normal operation. However, fuel cladding damage could result if an accident occurs from initial conditions outside the limits of these LCOs. This potential for fuel cladding damage exists because changes in the power distribution can cause increased power peaking and can correspondingly increase local LHR.

The LHR satisfies Criterion 2 of 10 CFR 50 36 (c)(2)(ii).

LCO

The power distribution LCO limits are based on correlations between power peaking and certain measured variables used as inputs to the LHR and DNBR operating limits. The power distribution LCO limits are provided in the COLR. The limitation on LHR ensures that in the event of a LOCA the peak temperature of the fuel cladding does not exceed 2200°F.

(continued)

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.2 Planar Radial Peaking Factors (F_{xy})

BASES

BACKGROUND

The purpose of this LCO is to limit the core power distribution to the initial values assumed in the accident analyses. Operation within the limits imposed by this LCO either limits or prevents potential fuel cladding failures that could breach the primary fission product barrier and release fission products to the reactor coolant in the event of a Loss Of Coolant Accident (LOCA), loss of flow accident, ejected Control Element Assembly (CEA) accident, or other postulated accident requiring termination by a Reactor Protection System (RPS) trip function. This LCO limits damage to the fuel cladding during an accident by ensuring that the plant is operating within acceptable conditions at the onset of a transient.

Methods of controlling the power distribution include:

a. Using full strength, part length, or part strength CEAs to alter the axial power distribution;

b. Decreasing CEA insertion by boration, thereby improving the radial power distribution; and

c. Correcting off optimum conditions (e.g., a CEA drop or misoperation of the unit) that cause margin degradations.

The core power distribution is controlled so that, in conjunction with other core operating parameters (CEA insertion and alignment limits), the power distribution does not result in violation of this LCO. Limiting safety system settings and this LCO are based on the accident analyses (Refs. 1 and 2), so that specified acceptable fuel design limits are not exceeded as a result of Anticipated Operational Occurrences (AOOs), and the limits of acceptable consequences are not exceeded for other postulated accidents.

Limiting power distribution skewing over time also minimizes xenon distribution skewing, which is a significant factor in controlling axial power distribution. Power distribution is a product of multiple parameters, various combinations of

(continued)

BACKGROUND (continued)

which may produce acceptable power distributions. Operation within the design limits of power distribution is accomplished by generating operating limits on Linear Heat Rate (LHR) and Departure from Nucleate Boiling (DNB).

Proximity to the DNB condition is expressed by the Departure from Nucleate Boiling Ratio (DNBR), defined as the ratio of the cladding surface heat flux required to cause DNB to the actual cladding surface heat flux. The minimum DNBR value during both normal operation and AOOs is the DNBR Safety Limit as calculated by the CE-1 Correlation (Ref. 3) and corrected for such factors as rod bow and grid spacers, and it is accepted as an appropriate margin to DNB for all operating conditions.

There are two systems that monitor core power distribution online: the Core Operating Limit Supervisory System (COLSS) and the Core Protection Calculators (CPCs). The COLSS and CPCs that monitor the core power distribution are capable of verifying that the LHR and the DNBR do not exceed their limits. The COLSS performs this function by continuously monitoring the core power distribution and calculating core power operating limits corresponding to the allowable peak LHR and DNBR values. The CPCs perform this function by continuously calculating actual values of DNBR and Local Power Density (LPD) for comparison with the respective trip setpoints.

DNBR penalty factors are included in both the COLSS and CPC DNBR calculations to accommodate the effects of rod bow. The amount of rod bow in each assembly is dependent upon the average burnup experienced by that assembly. Fuel assemblies that incur higher than average burnup experience greater rod bow. Conversely, fuel assemblies that receive lower than average burnup experience less rod bow. In design calculations for a reload core, each batch of fuel is assigned a penalty applied to the maximum integrated planar radial power peak of the batch. This penalty is correlated with the amount of rod bow determined from the maximum average assembly burnup of the batch. A single net penalty for the COLSS and CPCs is then determined from the penalties associated with each batch that comprises a core reload, accounting for the offsetting margins due to the lower radial power peaks in the higher burnup batches.

The COLSS indicates continuously to the operator how far the core is to the operating limits and provides an audible

(continued)

BACKGROUND (continued)

alarm if an operating limit is exceeded. Such a condition signifies a reduction in the capability of the plant to withstand an anticipated transient, but does not necessarily imply an immediate violation of fuel design limits. If the margin to fuel design limits continues to decrease, the RPS ensures that the specified acceptable fuel design limits are not exceeded for AOOs by initiating a reactor trip.

The COLSS continually generates an assessment of the calculated margin for LHR and DNBR specified limits. The data required for these assessments include measured incore neutron flux, CEA positions, and Reactor Coolant System (RCS) inlet temperature, pressure, and flow.

In addition to monitoring performed by the COLSS, the RPS (via the CPCs) continually infers the core power distribution and thermal margins by processing reactor coolant data, signals from excore neutron flux detectors, and input from redundant reed switch assemblies that indicates CEA position. In this case, the CPCs assume a minimum core power of 20% RTP. This threshold is set at 20% RTP because the power range excore neutron flux detecting system is inaccurate below this power level. If power distribution or other parameters are perturbed as a result of an AOO, the high LPD or low DNBR trips in the RPS initiate a reactor trip prior to exceeding fuel design limits.

The limits on ASI, F_{xy}, and T_q represent limits within which the LHR and DNBR algorithms are valid. These limits are obtained directly from the initial core or reload analysis.

APPLICABLE SAFETY ANALYSES

The fuel cladding must not sustain damage as a result of normal operation or AOOs (Ref. 4). The power distribution and CEA insertion and alignment LCOs prevent core power distributions from reaching levels that violate the following fuel design criteria:

a. During a LOCA, peak cladding temperature must not exceed 2200°F (Ref. 5);

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

b. During a loss of flow accident, there must be at least 95% probability at the 95% confidence level (the 95/95 DNB criterion) that the hot fuel rod in the core does not experience a DNB condition (Ref. 4);

c. During an ejected CEA accident, the fission energy input to the fuel must not exceed 280 cal/gm (Ref. 6); and

d. The control rods (excluding part length or part strength rods) must be capable of shutting down the reactor with a minimum required SDM with the highest worth control rod stuck fully withdrawn (Ref. 7).

The power density at any point in the core must be limited to maintain the fuel design criteria (Refs. 4 and 5). This result is accomplished by maintaining the power distribution and reactor coolant conditions so that the peak LHR and DNB parameters are within operating limits supported by the accident analyses (Ref. 1) with due regard for the correlations between measured quantities, the power distribution, and the uncertainties in the determination of power distribution.

Fuel cladding failure during a LOCA is limited by restricting the maximum Linear Heat Generation Rate (LHGR) so that the peak cladding temperature does not exceed 2200°F (Ref. 5). Peak cladding temperatures exceeding 2200°F cause severe cladding failure by oxidation due to a Zircaloy water reaction.

The LCOs governing LHR, ASI, CEAs, and RCS ensure that these criteria are met as long as the core is operated within the ASI and F_{xy} limits specified in the COLR, and within the T_q limits. The latter are process variables that characterize the three dimensional power distribution of the reactor core. Operation within the limits for these variables ensures that their actual values are within the ranges used in the accident analyses (Ref. 1).

Fuel cladding damage does not occur because of conditions outside the limits of these LCOs for ASI, F_{xy}, and T_q during normal operation. However, fuel cladding damage results if an accident occurs from initial conditions outside the limits of these LCOs. This potential for fuel cladding damage exists because changes in the power distribution can

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B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.3 AZIMUTHAL POWER TILT (T_q)

BASES

BACKGROUND

The purpose of this LCO is to limit the core power distribution to the initial values assumed in the accident analyses. Operation within the limits imposed by this LCO either limits or prevents potential fuel cladding failures that could breach the primary fission product barrier and release fission products to the reactor coolant in the event of a Loss Of Coolant Accident (LOCA), loss of flow accident, ejected Control Element Assembly (CEA) accident, or other postulated accident requiring termination by a Reactor Protection System (RPS) trip function. This LCO limits the amount of damage to the fuel cladding during an accident by ensuring that the plant is operating within acceptable conditions at the onset of a transient.

Methods of controlling the power distribution include:

a. Using full strength, part length, or part strength CEAs to alter the axial power distribution;

b. Decreasing CEA insertion by boration, thereby improving the radial power distribution; and

c. Correcting off optimum conditions, (e.g., a CEA drop or misoperation of the unit) that cause margin degradations.

The core power distribution is controlled so that, in conjunction with other core operating parameters (e.g., CEA insertion and alignment limits), the power distribution does not result in violation of this LCO. The limiting safety system settings and this LCO are based on the accident analyses (Refs. 1 and 2), so that specified acceptable fuel design limits are not exceeded as a result of Anticipated Operational Occurrences (AOOs) and the limits of acceptable consequences are not exceeded for other postulated accidents.

Limiting power distribution skewing over time also minimizes xenon distribution skewing, which is a significant factor in controlling axial power distribution.

(continued)

BACKGROUND (continued)

Power distribution is a product of multiple parameters, various combinations of which may produce acceptable power distributions. Operation within the design limits of power distribution is accomplished by generating operating limits on the Linear Heat Rate (LHR) and the Departure from Nucleate Boiling (DNB).

Proximity to the DNB condition is expressed by the Departure from Nucleate Boiling Ratio (DNBR), defined as the ratio of the cladding surface heat flux required to cause DNB to the actual cladding surface heat flux. The minimum DNBR value during both normal operation and AOOs is the DNBR Safety Limit as calculated by the CE-1 Correlation (Ref. 3) and corrected for such factors as rod bow and grid spacers, and it is accepted as an appropriate margin to DNB for all operating conditions.

There are two systems that monitor core power distribution online: the Core Operating Limit Supervisory System (COLSS) and the Core Protection Calculators (CPCs). The COLSS and CPCs that monitor the core power distribution are capable of verifying that the LHR and the DNBR do not exceed their limits. The COLSS performs this function by continuously monitoring the core power distribution and calculating core power operating limits corresponding to the allowable peak LHR and DNBR. The CPCs perform this function by continuously calculating actual values of DNBR and Local Power Density (LPD) for comparison with the respective trip setpoints.

A DNBR penalty factor is included in the COLSS and CPC DNBR calculation to accommodate the effects of rod bow. The amount of rod bow in each assembly is dependent upon the average burnup experienced by the assembly. Fuel assemblies that incur higher than average burnup experience greater magnitude of rod bow. Conversely, fuel assemblies that receive lower than average burnup experience less rod bow. In design calculations for a reload core, each batch of fuel is assigned a penalty applied to the maximum integrated planar radial power peak of the batch. This penalty is correlated with the amount of rod bow that is determined from the maximum average assembly burnup of the batch. A single net penalty for the COLSS and CPCs is then determined from the penalties associated with each batch that comprises a core reload, accounting for the offsetting margins caused by the lower radial power peaks in the higher burnup batches.

(continued)

BACKGROUND (continued)

The COLSS indicates continuously to the operator how far the core is from the operating limits and provides an audible alarm if an operating limit is exceeded. Such a condition signifies a reduction in the capability of the plant to withstand an anticipated transient, but does not necessarily imply an immediate violation of fuel design limits. If the margin to fuel design limits continues to decrease, the RPS ensures that the specified acceptable fuel design limits are not exceeded for AOOs by initiating a reactor trip.

The COLSS continually generates an assessment of the calculated margin for LHR and DNBR specified limits. The data required for these assessments include measured incore neutron flux data, CEA positions, and Reactor Coolant System (RCS) inlet temperature, pressure, and flow.

In addition to the monitoring performed by the COLSS, the RPS (via the CPCs) continually infers the core power distribution and thermal margins by processing reactor coolant data, signals from excore neutron flux detectors, and input from redundant reed switch assemblies that indicates CEA position. In this case, the CPCs assume a minimum core power of 20% RTP. This threshold is set at 20% RTP because the power range excore neutron flux detection system is inaccurate below this power level. If power distribution or other parameters are perturbed as a result of an AOO, the high local power density or low DNBR trips in the RPS initiate a reactor trip prior to exceeding fuel design limits.

The limits on the ASI, F_{xy}, and T_q represent limits within which the LHR and DNBR algorithms are valid. These limits are obtained directly from the initial core or reload analysis.

APPLICABLE SAFETY ANALYSES

The fuel cladding must not sustain damage as a result of operation and AOOs (Ref. 4). The power distribution and CEA insertion and alignment LCOs preclude core power distributions that violate the following fuel design criteria:

a. During a LOCA, peak cladding temperature must not exceed 2200°F (Ref. 5);

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

b. During a loss of flow accident, there must be at least 95% probability at the 95% confidence level (the 95/95 DNB criterion) that the hot fuel rod in the core does not experience a DNB condition (Ref. 4);

c. During a CEA ejection accident, the fission energy input to the fuel must not exceed 280 cal/gm (Ref. 6); and

d. The control rods (excluding part length or part strength rods) must be capable of shutting down the reactor with a minimum required SDM with the highest worth control rod stuck fully withdrawn (Ref. 7).

The power density at any point in the core must be limited to maintain the fuel design criteria (Ref. 1). This result is accomplished by maintaining the power distribution and reactor coolant conditions so that the peak LHR and DNB parameters are within operating limits supported by the accident analysis (Ref. 2) with due regard for the correlations between measured quantities, the power distribution, and uncertainties in the determination of power distribution.

Fuel cladding failure during a LOCA is limited by restricting the maximum Linear Heat Generation Rate (LHGR) so that the peak cladding temperature does not exceed 2200°F (Ref. 1). Peak cladding temperatures exceeding 2200°F cause severe cladding failure by oxidation due to a Zircaloy water reaction.

The LCOs governing LHR, ASI, CEAs, and RCS ensure that these criteria are met as long as the core is operated within the ASI and F_{xy} limits specified in the COLR, and within the T_q limits. The latter are process variables that characterize the three dimensional power distribution of the reactor core. Operation within the limits of these variables ensures that their actual values are within the range used in the accident analyses (Ref. 1).

(continued)

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.4 Departure from Nucleate Boiling Ratio (DNBR)

BASES

BACKGROUND

The purpose of this LCO is to limit the core power distribution to the initial value assumed in the accident analyses. Specifically, operation within the limits imposed by this LCO either limits or prevents potential fuel cladding failures that could breach the primary fission product barrier and release fission products to the reactor coolant in the event of a Loss Of Coolant Accident (LOCA), loss of flow accident, ejected Control Element Assembly (CEA) accident, or other postulated accident requiring termination by a Reactor Protection System (RPS) trip function. This LCO limits the amount of damage to the fuel cladding during an accident by ensuring that the plant is operating within acceptable conditions at the onset of a transient.

Methods of controlling the power distribution include:

a. Using full strength, part length, or part strength CEAs to alter the axial power distribution;

b. Decreasing CEA insertion by boration, thereby improving the radial power distribution; and

c. Correcting off optimum conditions (e.g., a CEA drop or misoperation of the unit) that cause margin degradations.

The core power distribution is controlled so that, in conjunction with other core operating parameters (e.g., CEA insertion and alignment limits), the power distribution does not result in violation of this LCO. The limiting safety system settings and this LCO are based on the accident analysis (Refs. 1 and 2), so that specified acceptable fuel design limits are not exceeded as a result of Anticipated Operational Occurrences (AOOs) and the limits of acceptable consequences are not exceeded for other postulated accidents.

Limiting power distribution skewing over time also minimizes the xenon distribution skewing, which is a significant factor in controlling axial power distribution.

(continued)

BASES

BACKGROUND (continued)

Power distribution is a product of multiple parameters, various combinations of which may produce acceptable power distributions. Operation within the design limits of power distribution is accomplished by generating operating limits on the Linear Heat Rate (LHR) and the Departure from nucleate boiling (DNB).

Proximity to the DNB condition is expressed by the DNBR, defined as the ratio of the cladding surface heat flux required to cause DNB to the actual cladding surface heat flux. The minimum DNBR value during both normal operation and AOOs is the DNBR Safety Limit as calculated by the CE-1 Correlation (Ref. 3) and corrected for such factors as rod bows and grid spacers and it is accepted as an appropriate margin to DNB for all operating conditions.

There are two systems that monitor core power distribution online: the Core Operating Limits Supervisory System (COLSS) and the Core Protection Calculators (CPCs). The COLSS and CPCs that monitor the core power distribution are capable of verifying that the LHR and DNBR do not exceed their limits. The COLSS performs this function by continuously monitoring the core power distribution and calculating core power operating limits corresponding to the allowable peak LHR and DNBR. The CPCs perform this function by continuously calculating an actual value of DNBR and LPD for comparison with the respective trip setpoints.

A DNBR penalty factor is included in both the COLSS and CPC DNBR calculation to accommodate the effects of rod bow. The amount of rod bow in each assembly is dependent upon the average burnup experienced by that assembly. Fuel assemblies that incur higher than average burnup experience a greater magnitude of rod bow. Conversely, fuel assemblies that receive lower than average burnup experience less rod bow. In design calculations for a reload core, each batch of fuel is assigned a penalty that is applied to the maximum integrated planar radial power peak of the batch. This penalty is correlated with the amount of rod bow that is determined from the maximum average assembly burnup of the batch. A single net penalty for the COLSS and CPCs is then determined from the penalties associated with each batch that comprises a core reload, accounting for the offsetting margins due to the lower radial power peaks in the higher burnup batches.

(continued)

BASES

BACKGROUND (continued)

The COLSS indicates continuously to the operator how far the core is from the operating limits and provides an audible alarm when an operating limit is exceeded. Such a condition signifies a reduction in the capability of the plant to withstand an anticipated transient, but does not necessarily imply an immediate violation of fuel design limits. If the margin to fuel design limits continues to decrease, the RPS ensures that the specified acceptable fuel design limits are not exceeded during AOOs by initiating a reactor trip.

The COLSS continually generates an assessment of the calculated margin for LHR and DNBR specified limits. The data required for these assessments include measured incore neutron flux, CEA positions, and Reactor Coolant System (RCS) inlet temperature, pressure, and flow.

In addition to the monitoring performed by the COLSS, the RPS (via the CPCs) continually infers the core power distribution and thermal margins by processing reactor coolant data, signals from excore neutron flux detectors, and input from redundant reed switch assemblies that indicates CEA position. In this case, the CPCs assume a minimum core power of 20% RTP because the power range excore neutron flux detecting system is inaccurate below this power level. If power distribution or other parameters are perturbed as a result of an AOO, the high local power density or low DNBR trips in the RPS initiate a reactor trip prior to exceeding fuel design limits.

The limits on ASI, F_{xy}, and T_q represent limits within which the LHR and DNBR algorithms are valid. These limits are obtained directly from the initial core or reload analysis.

APPLICABLE SAFETY ANALYSES

The fuel cladding must not sustain damage as a result of normal operation or AOOs (Ref. 4). The power distribution and CEA insertion and alignment LCOs prevent core power distributions from reaching levels that violate the following fuel design criteria:

a. During a LOCA, peak cladding temperature must not exceed 2200°F (Ref. 5);

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

b. During a loss of flow accident, there must be at least 95% probability at the 95% confidence level (the 95/95 DNB criterion) that the hot fuel rod in the core does not experience a DNB condition (Ref. 3);

c. During an ejected CEA accident, the fission energy input to the fuel must not exceed 280 cal/gm (Ref. 6); and

d. The control rods (excluding part length or part strength rods) must be capable of shutting down the reactor with a minimum required SDM with the highest worth control rod stuck fully withdrawn (Ref. 7).

The power density at any point in the core must be limited to maintain the fuel design criteria (Ref. 4). This is accomplished by maintaining the power distribution and reactor coolant conditions so that the peak LHR and DNB parameters are within operating limits supported by the accident analyses (Ref. 1) with due regard for the correlations between measured quantities, the power distribution, and uncertainties in the determination of power distribution.

Fuel cladding failure during a LOCA is limited by restricting the maximum Linear Heat Generation Rate (LHGR) so that the peak cladding temperature does not exceed 2200°F (Ref. 4). Peak cladding temperatures exceeding 2200°F may cause severe cladding failure by oxidation due to a Zircaloy water reaction.

The LCOs governing LHR, ASI, CEAs, and RCS ensure that these criteria are met as long as the core is operated within the ASI and F_{xy} limits specified in the COLR, and within the T_q limits. The latter are process variables that characterize the three dimensional power distribution of the reactor core. Operation within the limits for these variables ensures that their actual values are within the range used in the accident analyses (Ref. 1).

(continued)

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.5 AXIAL SHAPE INDEX (ASI)

BASES

BACKGROUND

The purpose of this LCO is to limit the core power distribution to the initial values assumed in the accident analysis. Operation within the limits imposed by this LCO either limits or prevents potential fuel cladding failures that could breach the primary fission product barrier and release fission products to the reactor coolant in the event of a Loss Of Coolant Accident (LOCA), loss of flow accident, ejected Control Element Assembly (CEA) accident, or other postulated accident requiring termination by a Reactor Protection System (RPS) trip function. This LCO limits the amount of damage to the fuel cladding during an accident by ensuring that the plant is operating within acceptable conditions at the onset of a transient.

Methods of controlling the axial power distribution include:

a. Using full strength, part length, or part strength CEAs to alter the axial power distribution;

b. Decreasing CEA insertion by boration, thereby improving the axial power distribution; and

c. Correcting off optimum conditions (e.g., a CEA drop or misoperation of the unit) that cause margin degradations.

The core power distribution is controlled so that, in conjunction with other core operating parameters (CEA insertion and alignment limits), the power distribution does not result in violation of this LCO. The limiting safety system settings are based on the accident analyses (Refs. 1 and 2), so that specified acceptable fuel design limits are not exceeded as a result of Anticipated Operational Occurrences (AOOs) and the limits of acceptable consequences are not exceeded for other postulated accidents.

Limiting power distribution skewing over time also minimizes xenon distribution skewing, which is a significant factor in controlling axial power distribution.

(continued)

BASES

BACKGROUND (continued)

Power distribution is a product of multiple parameters, various combinations of which may produce acceptable power distributions. Operation within the design limits of power distribution is accomplished by generating operating limits on the Linear Heat Rate (LHR) and the Departure from Nucleate Boiling (DNB).

Proximity to the DNB condition is expressed by the Departure from Nucleate Boiling Ratio (DNBR), defined as the ratio of the cladding surface heat flux required to cause DNB to the actual cladding surface heat flux. The minimum DNBR value during both normal operation and AOOs is the DNBR Safety Limit as calculated by the CE-1 Correlation (Ref. 3), and corrected for such factors as rod bow and grid spacers, and it is accepted as an appropriate margin to DNB for all operating conditions.

There are two systems that monitor core power distribution online: the Core Operating Limit Supervisory System (COLSS) or the Core Protection Calculators (CPCs). The COLSS and CPCs monitor the core power distribution and are capable of verifying that the LHR and DNBR do not exceed their limits. The COLSS performs this function by continuously monitoring the core power distribution and calculating core power operating limits corresponding to the allowable peak LHR and DNBR. The CPCs perform this function by continuously calculating actual values of DNBR and local power density (LPD) for comparison with the respective trip setpoints.

A DNBR penalty factor is included in both the COLSS and CPC DNBR calculations to accommodate the effects of rod bow. The amount of rod bow in each assembly is dependent upon the average burnup experienced by that assembly. Fuel assemblies that incur higher than average burnup experience greater rod bow. Conversely, fuel assemblies that receive lower than average burnup experience less rod bow. In design calculations for a reload core, each batch of fuel is assigned a penalty that is applied to the maximum integrated planar radial power peak of the batch. This penalty is correlated with the amount of rod bow that is determined from the maximum average assembly burnup of the batch. A single net penalty for the COLSS and CPC is then determined from the penalties associated with each batch that comprises a core reload, accounting for the offsetting margins due to the lower radial power peaks in the higher burnup batches.

(continued)

BASES

BACKGROUND (continued)

The COLSS indicates continuously to the operator how far the core is from the operating limits and provides an audible alarm if an operating limit is exceeded. Such a condition signifies a reduction in the capability of the plant to withstand an anticipated transient, but does not necessarily imply an immediate violation of fuel design limits. If the margin to fuel design limits continues to decrease, the RPS ensures that the specified acceptable fuel design limits are not exceeded for AOOs by initiating a reactor trip.

The COLSS continually generates an assessment of the calculated margin for LHR and DNBR specified limits. The data required for these assessments include measured incore neutron flux, CEA positions, and Reactor Coolant System (RCS) inlet temperature, pressure, and flow.

In addition to the monitoring performed by the COLSS, the RPS (via the CPCs) continually infers the core power distribution and thermal margins by processing reactor coolant data, signals from excore neutron flux detectors, and input from redundant reed switch assemblies that indicates CEA position. In this case, the CPCs assume a minimum core power of 20% RTP because the power range excore neutron flux detecting system is inaccurate below this power level. If power distribution or other parameters are perturbed as a result of an AOO, the high local power density or low DNBR trips in the RPS initiate a reactor trip prior to exceeding fuel design limits.

The limits on ASI, F_{xy}, and T_q represent limits within which the LHR and DNBR algorithms are valid. These limits are obtained directly from the initial core or reload analysis.

APPLICABLE SAFETY ANALYSES

The fuel cladding must not sustain damage as a result of operation or AOOs (Ref. 4). The power distribution and CEA insertion and alignment LCOs prevent core power distributions from reaching levels that violate the following fuel design criteria:

a. During a LOCA, peak cladding temperature must not exceed 2200°F (Ref. 5);

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

b. During a loss of flow accident, there must be at least 95% probability at the 95% confidence level (the 95/95 DNB criterion) that the hot fuel rod in the core does not experience a DNB condition (Ref. 4);

c. During an ejected CEA accident, the fission energy input to the fuel must not exceed 280 cal/gm (Ref. 6);

d. The control rods (excluding part length or part strength rods) must be capable of shutting down the reactor with a minimum required SDM with the highest worth control rod stuck fully withdrawn (Ref. 7).

The power density at any point in the core must be limited to maintain the fuel design criteria (Refs. 4 and 5). This is accomplished by maintaining the power distribution and reactor coolant conditions so that the peak LHR and DNB parameters are within operating limits supported by the accident analyses (Ref. 1) with due regard for the correlations among measured quantities, the power distribution, and uncertainties in the determination of power distribution.

Fuel cladding failure during a LOCA is limited by restricting the maximum Linear Heat Generation Rate (LHGR) so that the peak cladding temperature does not exceed 2200°F (Ref. 5). Peak cladding temperatures exceeding 2200°F may cause severe cladding failure by oxidation due to a Zircaloy water reaction.

The LCOs governing LHR, ASI, and RCS ensure that these criteria are met as long as the core is operated within the ASI and F_{xy} limits specified in the COLR, and within the T_q limits. The latter are process variables that characterize the three dimensional power distribution of the reactor core. Operation within the limits for these variables ensures that their actual values are within the range used in the accident analysis (Ref. 1).

Fuel cladding damage does not occur from conditions outside these LCOs during normal operation. However, fuel cladding damage results when an accident occurs due to initial conditions outside the limits of these LCOs. This potential for fuel cladding damage exists because changes in the power distribution can cause increased power peaking and correspondingly increased local LHRs.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

The ASI satisfies Criterion 2 of 10 CFR 50.36 (c)(2)(ii).

LCO

The power distribution LCO limits are based on correlations between power peaking and certain measured variables used as inputs to LHR and DNBR operating limits. The power distribution LCO limits are provided in the COLR. The COLR provides separate limits that are based on different combinations of COLSS and CEACs being in and out of service.

The limitation on ASI ensures that the actual ASI value is maintained within the range of values used in the accident analyses. The ASI limits ensure that with T_q at its maximum upper limit, the DNBR does not drop below the DNBR Safety Limit for AOOs.

APPLICABILITY

Power distribution is a concern any time the reactor is critical. The power distribution LCOs, however, are only applicable in MODE 1 above 20% RTP. The reasons these LCOs are not applicable below 20% RTP are:

a. The incore neutron detectors that provide input to the COLSS, which then calculates the operating limits, are inaccurate due to the poor signal to noise ratio that they experience at relatively low core power levels.

b. As a result of this inaccuracy, the CPCs assume a minimum core power of 20% RTP when generating the LPD and DNBR trip signals. When the core power is below this level, the core is operating well below the thermal limits and the resultant CPC calculated LPD and DNBR trips are strongly conservative.

(continued)

BASES

ACTIONS

A.1

The ASI limits specified in the COLR ensure that the LOCA and loss of flow accident criteria assumed in the accident analyses remain valid. If the ASI exceeds its limit, a Completion Time of 2 hours is allowed to restore the ASI to within its specified limit. This duration gives the operator sufficient time to reposition the regulating or part length or part strength CEAs to reduce the axial power imbalance. The magnitude of any potential xenon oscillation is significantly reduced if the condition is not allowed to persist for more than 2 hours.

B.1

If the ASI is not restored to within its specified limits within the required Completion Time, the reactor continues to operate with an axial power distribution mismatch. Continued operation in this configuration induces an axial xenon oscillation, and results in increased LHGRs when the xenon redistributes. Reducing thermal power to ≤ 20% RTP reduces the maximum LHR to a value that does not exceed the fuel design limits if a design basis event occurs. The allowed Completion Time of 4 hours is reasonable, based on operating experience, to reduce power in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.2.5.1

The ASI can be monitored by both the incore (COLSS) and excore (CPC) neutron detector systems. The COLSS provides the operator with an alarm if an ASI limit is approached.

Verification of the ASI every 12 hours ensures that the operator is aware of changes in the ASI as they develop. A 12 hour Frequency for this Surveillance is acceptable because the mechanisms that affect the ASI, such as xenon redistribution or CEA drive mechanism malfunctions, cause slow changes in the ASI, which can be discovered before the limits are exceeded.

(continued)

BASES

APPLICABLE SAFETY ANALYSES

Design Basis Definition (continued)

12, 13. Reactor Coolant Flow – Low

The Reactor Coolant Flow Steam Generator #1-Low and Reactor Coolant Flow Steam Generator #2-Low trips provide protection against an RCP Sheared Shaft Event. A trip is initiated when the pressure differential across the primary side of either steam generator decreases below a variable setpoint. This variable setpoint stays below the pressure differential by a preset value called the step function, unless limited by a preset maximum decreasing rate determined by the Ramp Function, or a set minimum value determined by the Floor Function. The setpoints ensure that a reactor trip occurs to limit fuel failure and ensure offsite doses are within 10 CFR 100 guidelines.

14. Local Power Density – High

The CPCs perform the calculations required to derive the DNBR and LPD parameters and their associated RPS trips. The DNBR – Low and LPD – High trips provide plant protection during the following AOOs and assist the ESF systems in the mitigation of the following accidents.

The LPD – High trip provides protection against fuel centerline melting due to the occurrence of excessive local power density peaks during the following AOOs:

- Decrease in Feedwater Temperature;
- Increase in Feedwater Flow;
- Increased Main Steam Flow (not due to the steam line rupture) Without Turbine Trip;
- Uncontrolled CEA Withdrawal From Low Power;
- Uncontrolled CEA Withdrawal at Power; and
- CEA Misoperation; Single Part Length CEA Drop (for Units that have Part Length CEAs).

For the events listed above (except CEA Misoperation; Single Part Length CEA Drop), DNBR – Low will trip the reactor first, since DNB would occur before fuel centerline melting would occur.

(continued)

BASES

APPLICABLE SAFETY ANALYSES

Design Basis Definition (continued)

15. Departure from Nucleate Boiling Ratio (DNBR) – Low

The CPCs perform the calculations required to derive the DNBR and LPD parameters and their associated RPS trips. The DNBR – Low and LPD – High trips provide plant protection during the following AOOs and assist the ESF systems in the mitigation of the following accidents.

The DNBR – Low trip provides protection against core damage due to the occurrence of locally saturated conditions in the limiting (hot) channel during the following events and is the primary reactor trip (trips the reactor first) for these events:

- Decrease in Feedwater Temperature;
- Increase in Feedwater Flow;
- Increased Main Steam Flow (not due to steam line rupture) Without Turbine Trip;
- Increased Main Steam Flow (not due to steam line rupture) With a Concurrent Single Failure of an Active Component;
- Steam Line Break With Concurrent Loss of Offsite AC Power;
- Loss of Normal AC Power;
- Partial Loss of Forced Reactor Coolant Flow;
- Total Loss of Forced Reactor Coolant Flow;
- Single Reactor Coolant Pump (RCP) Shaft Seizure;
- Uncontrolled CEA Withdrawal From Low Power;
- Uncontrolled CEA Withdrawal at Power;
- CEA Misoperation; CEA Drop;
- CEA Misoperation; Part Length or Part Strength CEA Subgroup Drop;
- Primary Sample or Instrument Line Break; and
- Steam Generator Tube Rupture.

In the above list, only the steam generator tube rupture, the RCP shaft seizure, and the sample or instrument line break are accidents. The rest are AOOs.

(continued)

BASES

LCO

The LCO requires all instrumentation performing an RPS Function to be OPERABLE. Failure of any required portion of the instrument channel renders the affected channel(s) inoperable and reduces the reliability of the affected Functions.

Actions allow maintenance (trip channel) bypass of individual channels, but the bypass activates interlocks that prevent operation with a second channel in the same Function bypassed. With one channel in each Function trip channel bypassed, this effectively places the plant in a two-out-of-three logic configuration in those Functions.

Only the Allowable Values are specified for each RPS trip Function in the LCO. Nominal trip setpoints are specified in the plant specific setpoint calculations. The nominal setpoints are selected to ensure the setpoints measured by CHANNEL FUNCTIONAL TESTS do not exceed the Allowable Value if the bistable is performing as required. Operation with a trip setpoint less conservative than the nominal trip setpoint, but within its Allowable Value, is acceptable, provided that operation and testing are consistent with the assumptions of the plant specific setpoint calculations. A channel is inoperable if its actual trip setpoint is not within its required Allowable Value. Each Allowable Value specified is more conservative than the analytical limit assumed in the safety analysis in order to account for instrument uncertainties appropriate to the trip Function. These uncertainties are defined in the "Plant Protection System Selection of Trip Setpoint Values" (Ref. 7).

The Bases for the individual Function requirements are as follows:

1. <u>Variable Over Power-High (RPS)</u>

   This LCO requires all four channels of Variable Over Power High (RPS) to be OPERABLE in MODES 1 and 2.

   The Allowable Value is high enough to provide an operating envelope that prevents unnecessary Variable Over Power High (RPS) reactor trips during normal plant operations. When the RPS VOPT trip function is credited in the safety analyses, the Allowable Value is based on the analyses and is low enough for the system to maintain a margin to unacceptable fuel or fuel cladding damage should a positive reactivity excursion event occur.

(continued)

BASES

LCO (continued)

2. Logarithmic Power Level – High

This LCO requires all four channels of Logarithmic Power Level – High to be OPERABLE in MODE 2.

In MODES 3, 4, or 5 when the RTCBs are shut and the CEA Drive System is capable of CEA withdrawal conditions are addressed in LCO 3.3.2.

The Allowable Value is high enough to provide an operating envelope that prevents unnecessary Logarithmic Power Level – High reactor trips during normal plant operations. The Allowable Value is low enough for the system to maintain a margin to unacceptable fuel cladding damage should a CEA withdrawal event occur.

The Logarithmic Power Level – High trip may be bypassed when logarithmic power is above 1E-4% NRTP to allow the reactor to be brought to power during a reactor startup. This operating bypass is automatically removed when logarithmic power decreases below 1E-4% NRTP. Above 1E-4% NRTP, the Variable Over Power – High and Pressurizer Pressure – High trips provide protection for reactivity transients.

The automatic bypass removal channel is INOPERABLE when the associated Log power channel has failed. The bypass function is manually controlled via station operating procedures and the bypass removal circuitry itself is fully capable of responding to a change in the associated input bistable. Footnotes (a) and (b) in Table 3.3.1-1 and (d) in Table 3.3.2-1 clearly require an "automatic" removal of trip bypasses. A failed Log channel may prevent, depending on the failure mode, the associated input bistable from changing state as power transitions through the automatic bypass removal setpoint. Specifically, when the indicated Log power channel is failed high (above 1E-4%), the automatic Hi-Log power trip bypass removal feature in that channel cannot function. Similarly, when the indicated Log power channel is failed low (below 1E-4%), the automatic DNBR-LPD trip bypass removal feature in that channel cannot function. Although one bypass removal feature is applicable above 1E-4% NRTP and the other is applicable below 1E-4% NRTP, both are affected by a failed Log power channel and should therefore be considered INOPERABLE.

(continued)

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.1 RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits

BASES

BACKGROUND

These Bases address requirements for maintaining RCS pressure, temperature, and flow rate within limits assumed in the safety analyses. The safety analyses (Ref. 1) of normal operating conditions and anticipated operational occurrences assume initial conditions within the normal steady state envelope. The limits placed on DNB related parameters ensure that these parameters will not be less conservative than were assumed in the analyses and thereby provide assurance that the minimum Departure from Nucleate Boiling Ratio (DNBR) will meet the required criteria for each of the transients analyzed.

The LCO limits for minimum and maximum RCS pressures as measured at the pressurizer are consistent with operation within the nominal operating envelope and are bounded by those used as the initial pressures in the analyses.

The LCO limit for minimum and maximum RCS cold leg temperatures are in accordance with the area of acceptable operation shown in Figure 3.4.1-1, are consistent with operation at the indicated power level, and are bounded by those used as the initial temperatures in the analyses.

The LCO limit for minimum RCS flow rate is bounded by those used as the initial flow rates in the analyses. The RCS flow rate is not expected to vary during plant operation with all pumps running.

APPLICABLE SAFETY ANALYSES

The requirements of LCO 3.4.1 represent the initial conditions for DNB limited transients analyzed in the safety analyses (Ref. 1). The safety analyses have shown that transients initiated from the limits of this LCO will meet the DNBR criterion of greater than or equal to the DNBR Safety Limit. This is the acceptance limit for the RCS DNB parameters. Changes to the facility that could impact these parameters must be assessed for their impact on the DNBR criterion.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

The transients analyzed for include loss of coolant flow events and dropped or stuck Control Element Assembly (CEA) events. A key assumption for the analysis of these events is that the core power distribution is within the limits of LCO 3.1.7, "Regulating CEA Insertion Limits"; LCO 3.1.8, "Part Length or Part Strength CEA Insertion Limits"; LCO 3.2.3, "AZIMUTHAL POWER TILT (T_q)"; and LCO 3.2.5, "AXIAL SHAPE INDEX (ASI)."

The RCS DNB limits satisfy Criterion 2 of 10 CFR 50.56(c)(2)(ii).

LCO

This LCO specifies limits on the monitored process variables – RCS pressurizer pressure, RCS cold leg temperature, and RCS total flow rate – to ensure that the core operates within the limits assumed for the plant safety analyses. Operating within these limits will result in meeting the DNBR criterion in the event of a DNB limited transient.

The LCO numerical value for minimum flow rate is given for the measurement location but has not been adjusted for instrument error. Plant specific limits of instrument error are established by the plant staff to meet the operational requirements of minimum flow rate.

APPLICABILITY

In MODE 1 for RCS flow rate, MODES 1 and 2 for RCS pressurizer pressure, Mode 1 for RCS cold leg temperature, and MODE 2 with $K_{eff} \geq 1$ for RCS cold leg temperature, the limits must be maintained during steady state operation in order to ensure that DNBR criteria will be met in the event of an unplanned loss of forced coolant flow or other DNB limited transient. In all other MODES, the power level is low enough so that DNBR is not a concern.

(continued)

B 3.7 PLANT SYSTEMS

B 3.7.1 Main Steam Safety Valves (MSSVs)

BASES

BACKGROUND

The primary purpose of the MSSVs is to provide overpressure protection for the secondary system. The MSSVs also provide protection against overpressurizing the Reactor Coolant Pressure Boundary (RCPB) by providing a heat sink for the removal of energy from the Reactor Coolant System (RCS) if the preferred heat sink, provided by the Condenser and Circulating Water System, is not available.

Five MSSVs are located on each of the four main steam lines, outside containment, upstream of the main steam isolation valves, as described in the UFSAR, Section 5.2 (Ref. 1). The MSSV rated capacity passes the full steam flow at 102% RTP (100% + 2% for instrument error) with the valves full open. This meets the requirements of the ASME Code, Section III (Ref. 2). The MSSV design includes staggered setpoints, according to Table 3.7.1-2, in the accompanying LCO, so that only the number of valves needed will actuate. Staggered setpoints reduce the potential for valve chattering if there is insufficient steam pressure to fully open all valves.

APPLICABLE SAFETY ANALYSES

The design basis for the MSSVs comes from Reference 2; its purpose is to limit secondary system pressure to ≤ 110% of design pressure when passing 100% of design steam flow. This design basis is sufficient to cope with any Anticipated Operational Occurrence (AOO) or accident considered in the Design Basis Accident (DBA) and transient analysis.

The events that challenge the MSSV relieving capacity, and thus RCS pressure, are those characterized as decreased heat removal events, and are presented in the FSAR, Section 15.2 (Ref. 3). Of these, the full power Loss Of Condenser Vacuum (LOCV) event is the limiting AOO. An LOCV isolates the turbine and condenser, and terminates normal feedwater flow to the steam generators. Peak Main Steam System and Reactor Coolant System (RCS) pressure occur before delivery of auxiliary feedwater to the steam generators. The peak pressures become high enough to actuate both the Main Steam Safety Valves (MSSVs) and Pressurizer Safety Valves, but remain less than 110% of the design (1397 and 2750 psia for main steam system and RCS, respectively). The LOCV Secondary Peak Pressure event is the limiting decrease in heat removal transient for determining the maximum allowed thermal power with inoperable MSSVs.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

The limiting accident for peak RCS pressure is the full power feedwater line break (FWLB), inside containment, with the failure of the backflow check valve in the feedwater line from the affected steam generator. Water from the affected steam generator is assumed to be lost through the break with minimal additional heat transfer from the RCS. With heat removal limited to the unaffected steam generator, the reduced heat transfer causes an increase in RCS temperature, and the resulting RCS fluid expansion causes an increase in pressure. The increase in Main Steam and Reactor Coolant System pressure is mitigated by the relief capacity of the Main Steam Safety Valves (MSSVs) and pressurizer safety valves. The peak pressures do not exceed 120% of the design pressure (1524 psia and 3000 psia for main steam and RCS, respectively). These results were found acceptable by the NRC based on the low probability of the event.

The MSSVs satisfy Criterion 3 of 10CFR 50.36 (c)(2)(ii).

LCO

This LCO requires all MSSVs to be OPERABLE in compliance with Reference 2, even though this is not a requirement of the DBA analysis. This is because operation with less than the full number of MSSVs requires limitations on allowable THERMAL POWER (to meet Reference 2 requirements), and adjustment to the Reactor Protection System trip setpoints. These limitations are according to those shown in Table 3.7.1-1 and Required Action A.2 in the accompanying LCO. An MSSV is considered inoperable if it fails to open upon demand.

The OPERABILITY of the MSSVs is defined as the ability to open within the setpoint tolerances, relieve steam generator overpressure, and reseat when pressure has been reduced. The OPERABILITY of the MSSVs is determined by periodic surveillance testing in accordance with the Inservice Testing Program.

The lift settings, according to Table 3.7.1-2 in the accompanying LCO, correspond to ambient conditions of the valve at nominal operating temperature and pressure.

This LCO provides assurance that the MSSVs will perform their designed safety function to mitigate the consequences of accidents that could result in a challenge to the RCPB.

(continued)

BASES

APPLICABILITY

In MODES 1, 2 and 3, a minimum of six MSSVs per steam generator are required to be OPERABLE, according to Table 3.7.1-1 in the accompanying LCO, which is limiting and bounds all lower MODES.

In MODES 4 and 5, there are no credible transients requiring the MSSVs.

The steam generators are not normally used for heat removal in MODES 5 and 6, and thus cannot be overpressurized; there is no requirement for the MSSVs to be OPERABLE in these MODES.

ACTIONS

The ACTIONS table is modified by a Note indicating that separate Condition entry is allowed for each MSSV.

A.1 and A.2

When 10 MSSVs are OPERABLE per steam generator, THERMAL POWER is limited to 100% RTP per the Operating Licenses, and the VOPT allowable trip setpoint is limited to 111.0% RTP per TS Table 3.3.1-1.

An alternative to restoring inoperable MSSV(s) to OPERABLE status is to reduce power in accordance with Table 3.7.1-1. These reduced power levels, derived from the transient analysis, compensate for degraded relieving capacity and ensure that the results of the transient analysis are acceptable.

The operator should limit the maximum steady state power level to the value determined from Table 3.7.1-1 to avoid an inadvertent overpower trip.

The Completion Time of 12 hours for Required Action A.2 is based on operating experience in resetting all channels of a protective function and on the low probability of the occurrence of a transient that could result in steam generator overpressure during this period.

(continued)

BASES

ACTIONS (continued)

B.1 and B.2

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

SURVEILLANCE REQUIREMENTS

SR 3.7.1.1

This SR verifies the OPERABILITY of the MSSVs by the verification of each MSSV lift setpoints in accordance with the Inservice Testing Program. The ASME Code, Section XI (Ref. 4), requires that safety and relief valve tests be performed in accordance with ANSI/ASME OM-1-1987 (Ref. 5). According to Reference 5, the following tests are required for MSSVs:

a. Visual examination;

b. Seat tightness determination;

c. Setpoint pressure determination (lift setting);

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

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

The ASME Standard requires that all valves be tested every 5 years, and a minimum of 20% of the valves tested every 24 months. The ASME Code specifies the activities and frequencies necessary to satisfy the requirements. Table 3.7.1-2 allows a ± 3% setpoint tolerance for OPERABILITY; however, the valves are reset to ± 1% during the Surveillance to allow for drift.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.1.1 (continued)

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

REFERENCES

1. UFSAR, Section 5.2.
2. ASME, Boiler and Pressure Vessel Code, Section III, Article NC-7000, Class 2 Components.
3. UFSAR, Section 15.2.
4. ASME, Boiler and Pressure Vessel Code, Section XI, Subsection IWV.
5. ANSI/ASME OM-1-1987.

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B 3.7 PLANT SYSTEMS

B 3.7.6 Condensate Storage Tank (CST)

BASES

BACKGROUND	The CST provides a safety grade source of water to the steam generators for removing decay and sensible heat from the Reactor Coolant System (RCS). The CST is the primary source of water for the Auxiliary Feedwater (AFW) System (LCO 3.7.5, "Auxiliary Feedwater (AFW) System"). The steam produced is released to the atmosphere by the Main Steam Safety Valves (MSSVs) or the atmospheric dump valves. When the main steam isolation valves are open, the preferred means of heat removal is to discharge steam to the condenser by the nonsafety grade path of the steam bypass control valves. The condensed steam is returned to the CST by the condensate pump draw-off. This has the advantage of conserving condensate while minimizing releases to the environment. Because the CST is a principal component in removing residual heat from the RCS, it is designed to withstand earthquakes and other natural phenomena. The CST is designed to Seismic Category I requirements to ensure availability of the feedwater supply. Feedwater is also available from the Reactor Makeup Water Tank (RMWT). A description of the CST is found in the UFSAR, Section 9.2.6 (Ref. 1).
APPLICABLE SAFETY ANALYSES	The CST provides cooling water to remove decay heat and to cool down the unit following all events in the accident analysis, discussed in the UFSAR, Chapters 6 and 15 (Refs. 2 and 3, respectively). For anticipated operational occurrences and accidents which do not affect the OPERABILITY of the steam generators, the CST has sufficient volume to maintain the plant for 8 hours at MODE 3, followed by a cooldown to shutdown cooling (SDC) entry conditions at the design cooldown rate.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

The design basis events for the Auxiliary Feedwater System which are presented in UFSAR Section 5.1.4 consider a loss of offsite power as the single failure, coincident with the initiating event. The limiting Chapter 15 safety analysis event for condensate volume is the large feedwater line break event followed by a loss of offsite power, as a result of turbine trip, plus a single failure. This bounds the design basis assumptions in Chapter 5.1.4. Single failures that affect this event include the following:

a. The failure of the diesel generator powering the motor driven AFW pump (requiring additional steam to drive the remaining AFW pump turbine); and

b. The failure of the steam driven AFW pump (requiring a longer time for cooldown using only one motor driven AFW pump).

The limiting Single failure for FWLB with LOP is the failure of the steam driven AFW pump.

A nonlimiting event considered in CST inventory determinations is a break either in the main feedwater, or essential AFW line near where the two join. This break has the potential for dumping condensate until terminated by operator action, as the Auxiliary Feedwater Actuation System would not detect a difference in pressure between the steam generators for this break location. This loss of condensate inventory is partially compensated by the retaining of steam generator inventory. A break in the non-essential AFW line could have similar consequences on CST level but is not controlled by AFAS. Actuation required operator action.

The CST satisfies Criterion 3 of 10 CFR 50.36 (c)(2)(ii).

LCO

To satisfy accident analysis assumptions, the CST must contain sufficient cooling water to remove decay heat for 4 hours following a reactor trip from 102% RTP, and then cool down the RCS to SDC entry conditions, assuming a coincident loss of offsite power and the most adverse single failure. In doing this it must retain sufficient water to ensure adequate net positive suction head for the AFW pumps during the cooldown, as well as to account for any losses from the steam driven AFW pump turbine, or before isolating AFW to a broken line.

(continued)

BASES

LCO (continued)

The CST level required is a usable volume of ≥ 300,000 gallons, which is based on holding the unit in MODE 3 for 8 hours, followed by a cooldown to SDC entry conditions at 75°F per hour. This basis exceeds the level required by the NRC Standard Review Plan Branch Technical Position, Reactor Systems Branch 5-1 (Ref. 4).

OPERABILITY of the CST is determined by maintaining the tank level at or above the minimum required level.

APPLICABILITY

In MODES 1, 2, and 3, and in MODE 4, when steam generator is being relied upon for heat removal, the CST is required to be OPERABLE.

In MODES 5 and 6, the CST is not required because the AFW System is not required.

ACTIONS

A.1 and A.2

If the CST level is not within the limit, the OPERABILITY of the backup water supply (RMWT) must be verified within 4 hours.

OPERABILITY of the RMWT must include initial alignment and verification of the OPERABILITY of flow paths from the RMWT to the AFW pumps, and availability of 26 ft. (300,000 gal.) of water in the RMWT. The CST level must be returned to OPERABLE status within 7 days, as the RMWT may be performing this function in addition to its normal functions. The 4 hour Completion Time is reasonable, based on operating experience, to verify the OPERABILITY of the RMWT. The 7 day Completion Time is reasonable, based on an OPERABLE RMWT being available, and the low probability of an event requiring the use of the water from the CST occurring during this period.

(continued)

BASES

ACTIONS (continued)

B.1 and B.2

If the CST cannot be restored to OPERABLE status within the associated Completion Time, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 4, without reliance on steam generator for heat removal, within 24 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.6.1

This SR verifies that the CST contains the required volume of cooling water. (This level $\geq$ 29.5 ft (300,000 gallons)). The 12 hour Frequency is based on operating experience, and the need for operator awareness of unit evolutions that may affect the CST inventory between checks. The 12 hour Frequency is considered adequate in view of other indications in the control room, including alarms, to alert the operator to abnormal CST level deviations.

REFERENCES

1. UFSAR, Section 9.2.6.
2. UFSAR, Chapter 6.
3. UFSAR, Chapter 15.
4. NRC Standard Review Plan Branch Technical Position (BTP) RSB5-1.

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PVNGS

Palo Verde Nuclear Generating Station
Units 1, 2, and 3

Technical Specification Bases

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PALO VERDE

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B 3.8.1-28	20	B 3.8.7-3	0
B 3.8.1-29	20	B 3.8.7-4	0
B 3.8.1-30	20	B 3.8.8-1	1
B 3.8.1-31	20	B 3.8.8-2	1
B 3.8.1-32	20	B 3.8.8-3	21
B 3.8.1-33	20	B 3.8.8-4	21
B 3.8.1-34	20	B 3.8.8-5	1
B 3.8.1-35	20	B 3.8.9-1	0
B 3.8.1-36	20	B 3.8.9-2	0
B 3.8.1-37	23	B 3.8.9-3	0
B 3.8.1-38	27	B 3.8.9-4	0
B 3.8.1-39	20	B 3.8.9-5	0
B 3.8.1-40	20	B 3.8.9-6	0
B 3.8.2-1	0	B 3.8.9-7	0
B 3.8.2-2	0	B 3.8.9-8	0
B 3.8.2-3	0	B 3.8.9-9	0
B 3.8.2-4	21	B 3.8.9-10	0
B 3.8.2-5	21	B 3.8.9-11	0
B 3.8.2-6	0	B 3.8.10-1	0
B 3.8.3-1	0	B 3.8.10-2	21
B 3.8.3-2	0	B 3.8.10-3	0
B 3.8.3-3	0	B 3.8.10-4	0
B 3.8.3-4	0	B 3.9.1-1	0
B 3.8.3-5	27	B 3.9.1-2	0
B 3.8.3-6	0	B 3.9.1-3	0
B 3.8.3-7	0	B 3.9.1-4	0
B 3.8.3-8	0	B 3.9.2-1	15
B 3.8.3-9	0	B 3.9.2-2	15
B 3.8.4-1	0	B 3.9.2-3	15
B 3.8.4-2	0	B 3.9.2-4	15
B 3.8.4-3	0	B 3.9.3-1	18
B 3.8.4-4	2	B 3.9.3-2	19
B 3.8.4-5	2	B 3.9.3-3	27
B 3.8.4-6	2	B 3.9.3-4	19
B 3.8.4-7	2	B 3.9.3-5	19
B 3.8.4-8	2	B.3.9.3-6	19
B 3.8.4-9	2	B 3.9.4-1	0
B 3.8.4-10	2	B 3.9.4-2	1

BASES (continued)

LCO (continued)

Type A leakage rate testing measures the overall leakage rate of the containment. Type B leakage rate testing measures the local leakage rate of blind flanges, air locks and other devices which employ resilient seals. Type C leakage rate testing measures the local leakage rate of valves. Refer to reference 1 for a more detailed definition.

Compliance with this LCO will ensure a containment configuration, including equipment hatches, that is structurally sound and that will limit leakage to those leakage rates assumed in the safety analysis.

Individual leakage rates specified for the containment air lock (LCO 3.6.2) and purge valves with resilient seals (LCO 3.6.3) are not specifically part of the acceptance criteria of 10 CFR 50, Appendix J, Option B. Therefore, leakage rates exceeding these individual limits only result in the containment being inoperable when the leakage results in exceeding the overall acceptance criteria of 1.0 L_a.

APPLICABILITY

In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material into containment. In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Therefore, containment is not required to be OPERABLE in MODE 5 to prevent leakage of radioactive material from containment. The requirements for containment during MODE 6 are addressed in LCO 3.9.3, "Containment Penetrations."

ACTIONS

A.1

In the event containment is inoperable, containment must be restored to OPERABLE status within 1 hour. The 1 hour Completion Time provides a period of time to correct the problem commensurate with the importance of maintaining containment during MODES 1, 2, 3, and 4. This time period also ensures that the probability of an accident (requiring containment OPERABILITY) occurring during periods when containment is inoperable is minimal.

(continued)

BASES (continued)

ACTIONS (continued)

B.1 and B.2

If 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 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.6.1.1

Maintaining the containment OPERABLE requires compliance with the visual examinations and leakage rate test requirements of the Containment Leakage Rate Testing Program. The containment concrete visual examinations may be performed during either power operation, e.g., performed concurrently with other containment inspection-related activities such as tendon testing, or during a maintenance/refueling outage. The visual examinations of the steel liner plate inside containment are performed during maintenance or refueling outages since this is the only time the liner plate is fully accessible.

Failure to meet air lock and purge valve with resilient seal leakage limits specified in LCO 3.6.2 and LCO 3.6.3 does not invalidate the acceptability of these overall leakage determinations unless their contribution to overall Type A, B, and C leakage causes that to exceed limits. As left leakage prior to the first startup after performing a required 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.

SR Frequencies are as required by the Containment Leakage Rate Testing Program. These periodic testing requirements verify that the containment leakage rate does not exceed the leakage rate assumed in the safety analysis.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.1.2

For ungrouted, post tensioned tendons, this SR ensures that the structural integrity of the containment will be maintained in accordance with the provisions of the Containment Tendon Surveillance Program. Testing and Frequency are in accordance with ASME Code Section XI, Subsection IWL (Ref. 4) and applicable addenda as required by 10 CFR 50.55a, except where an exemption or relief has been authorized by the NRC.

REFERENCES

1. 10 CFR 50, Appendix J, Option B.
2. UFSAR, Section 3.8.
3. UFSAR, Section 6.2.
4. ASME Code Section XI, Subsection IWL.

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