Summary of Design Transient

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Mitsubishi Heavy Industries, Ltd.
16-5, Konan 2-chome, Minato-ku
Tokyo 108-8215 Japan

Abstract

The purpose of this technical report is to present the summary of Design transient for the RCS primary loop transient events.

This report describes outline of the regulatory requirements, selection of transients, assumed frequency and transient curves for the RCS primary loop.

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List of Acronyms

APWR	Advanced Pressurized Water Reactor
ASME	American Society of Mechanical Engineers
DCD	Design Control Document
DNBR	Departure from Nucleate Boiling Ratio
ECCS	Emergency Core Cooling System
EFW	Emergency Feedwater
LOCA	Loss of Coolant Accident
MHI	Mitsubishi Heavy Industries, LTD.
NRC	Nuclear Regulatory Commission
RCP	Reactor Coolant Pump
RCS	Reactor Coolant System
RG	Regulatory Guide
RT	Reactor Trip
SGTR	Steam Generator Tube Rupture

1.0 INTRODUCTION

This report documents the selection of transients and assumed frequency, and the calculated pressure and temperature curve as a function of time for each transient, which provides input to stress and fatigue calculations. This report describes the summary of RCS primary loop transient events, which is a part of the full set of transients for the US-APWR plant.

The transient conditions selected are based on conservative estimates of the magnitude and frequency of the temperature and pressure transients resulting from various operating conditions in the plant for a 60 year operating period. To a large extent, the specific transients considered for design are based on ANSI/ANS-51.1 (Reference 1), engineering judgment and experience.

2.0 DESIGN TRANSIENTS

2.1 Regulatory Requirements

Criterion 30 ”Quality of reactor coolant pressure boundary” of 10CFR50 App.A ”General Design Criteria for Nuclear Power Plants” requires that components which are part of the reactor coolant pressure boundary shall be designed, fabricated, erected, and tested to the highest quality standards practical. 10CFR50.55a “Codes and Standard” (c) requires that components which are part of the reactor coolant pressure boundary must meet the requirements for Class 1 components in SEC III of ASME Boiler and Pressure Vessel Code. ASME Section III requires identifying service loadings and conducting fatigue analysis for the class 1 components. In Standard Review Plan (NUREG -0800), AREAS OF REVIEW of “3.9.1 Special Topics For Mechanical Components” includes “Transients used in the design and fatigue analysis of all Code Class 1 and core support components, supports, and reactor internals”. Furthermore, Section C.I.3.9.1.1 Design Transients of RG1.206, “Combined License Applications for Nuclear Power Plants”, provide guidance that the applicant should provide a complete list of transients used in the design and fatigue analysis of all ASME Code Class 1 and core support components, component supports, and reactor internals. The list should include the number of events for each transient, as well as the number of load and stress cycles per event and for events in combination.

In order to comply with these requirements, the design transients is needed to be used in design and fatigue analysis for the ASME Code class 1 and core support components, component supports, and reactor internals.

2.2 Operating Conditions

Components are evaluated using the design transients in accordance with the requirements for Class 1 in ASME Code, Section III (Reference -2). ASME Code, Section III defines the following five conditions which apply to the design of RCS Class 1 components, auxiliary Class 1 components, RCS component supports, and reactor internals.

Level A Service Conditions – (Normal Conditions)

These conditions include any condition in the course of system startup, operation in the design power range, hot standby, and system shutdown other than Level B, Level C, or Level D service conditions or testing conditions. Tests, in which pressure is not greater than the component design pressure, are considered to be normal condition design transients.

Level B Service Conditions – (Upset Conditions, Incidents of Moderate Frequency)

These conditions include any deviations from Level A service conditions anticipated to occur often enough that the design includes the capability to withstand the conditions without operational impairment. The Level B service conditions include those transients resulting from any single operator error or control malfunction, transients caused by a fault in a system component requiring its isolation from the system, and transients due to a loss of load or power. Level B service conditions include any abnormal incidents not resulting in a forced outage and also those resulting in a forced outage for which the corrective action does not include any repair of mechanical damage. The estimated duration of Level B service condition is included in the design specifications.

All of level B service conditions result in a reactor trip except for “Cold over-pressure”. After these conditions, the reactor coolant system is assumed to return to no load condition with applying temperature and pressure changes within the design limits defined in the normal condition.

Level C Service Conditions – (Emergency Conditions, Infrequent Incidents)

These conditions include those deviations from Level A service conditions that require shutdown for correction of the conditions or repair of damage. These conditions have a low probability of occurrence but are included to establish that no gross loss of structural integrity will result as a concurrent effect of any damage developed in a system. The postulated occurrences for such events which result in more than 25 strong stress cycles are evaluated for cyclic fatigue using Level B service limits. Strong stress cycles are those having an alternating stress intensity value greater than that for 10^6 cycles obtained from the applicable

fatigue design curves. All of level C service conditions reach cold shutdown conditions basically.

Level D Service Conditions – (Faulted Conditions, Limiting Faults)

These conditions include those combinations of conditions associated with extremely low-probability postulated events whose consequences are such that the integrity and operability of the nuclear energy system may be impaired to the extent that considerations of public health and safety are involved. Such considerations require compliance with safety criteria as may be specified by regulatory authorities.

All of level D service conditions reach cold shutdown conditions basically.

Testing Conditions

Testing conditions are those pressure overload tests that include primary and secondary hydrostatic tests specified. Other types of tests are classified under one of the other service condition categories.

The design transients and the number of occurrences for fatigue analysis of components are shown in Table 2.2-1.

Table 2.2-1 RCS Design Transients (1/2)

Event	Cycles	Table	Figure
Level A			
Reactor Coolant Pump startup	3,000	-	2.4.1-1
Reactor Coolant Pump shutdown	3,000	-	2.4.1-2
Plant heat-up	120	-	2.4.1-3
Plant cooldown	120	-	2.4.1-4
Ramp load increase between 0% and 15% of full power	600	-	2.4.1-5
Ramp load decrease between 0% and 15% of full power	600	-	2.4.1-6
Ramp load increase between 15% and 100% of full power (5% of full power per minute)	600	2.4.1-1	2.4.1-7
Ramp load increase between 50% and 100% of full power (5% of full power per minute)	19,200	2.4.1-2	2.4.1-8
Ramp load decrease between 15% and 100% of full power (5% of full power per minute)	600	2.4.1-3	2.4.1-9
Ramp load decrease between 50% and 100% of full power (5% of full power per minute)	19,200	2.4.1-4	2.4.1-10
Step load increase of 10% of full power	600	2.4.1-5	2.4.1-11
Step load decrease of 10% of full power	600	2.4.1-6	2.4.1-12
Large step load decrease with turbine bypass	60	2.4.1-7	2.4.1-13
Steady-state fluctuation and load regulation			
Steady-state fluctuation	1×10^{6}	-	-
Load regulation	8×10^{5}	-	-
Boron concentration equalization	39,600	-	-
Main feedwater cycling	2,100	2.4.1-8	2.4.1-14
Core lifetime extension	60	-	2.4.1-15
Refueling	60	-	2.4.1-16
Turbine roll test	10	-	2.4.1-17
Primary leakage test	120	-	2.4.1-18
Secondary leakage test	120	-	-

Table 2.2-2 RCS Design Transients (2/2)

Event	Cycles	Table	Figure
Level B			
Loss of load	60	2.4.2-1	2.4.2-1
Loss of offsite power	60	2.4.2-2	2.4.2-2
RT from full power			
With no inadvertent cooldown	60	2.4.2-3	2.4.2-3
With cooldown and no safety injection	30	2.4.2-4	2.4.2-4
With cooldown and safety injection	10	2.4.2-5	2.4.2-5
Control rod drop	30	2.4.2-6	2.4.2-6
Cold over-pressure	30	2.4.2-7	2.4.2-7
Inadvertent safeguards actuation	30	2.4.2-8	2.4.2-8
Partial loss of reactor coolant flow	30	2.4.2-9	2.4.2-9
Inadvertent RCS depressurization			
Umbrella case	30	2.4.2-10	2.4.2-10
Inadvertent auxiliary spray	15	2.4.2-11	2.4.2-11
Emergency feedwater cycling	700	2.4.2-12	2.4.2-12
Partial loss of emergency feedwater	30	-	-
Level C			
Reactor coolant pipe break (Small LOCA)	5	2.4.3-1	2.4.3-1
Small steam line break	5	2.4.3-2	2.4.3-2
Complete loss of flow	5	2.4.3-3	2.4.3-3
Small feedwater line break	5	2.4.3-4	2.4.3-4
SG tube rupture	5	2.4.3-5	2.4.3-5
Level D			
Reactor coolant pipe break (Large LOCA)	1	2.4.4-1	2.4.4-1
Large steam line break	1	2.4.4-2	2.4.4-2
Large feedwater line break	1	2.4.4-3	2.4.4-3
RCP locked rotor	1	2.4.4-4	2.4.4-4
Control rod ejection	1	2.4.4-5	2.4.4-5
Test Condition			
Primary-side hydrostatic test	10	-	-
Secondary-side hydrostatic test	10	-	-

2.3 Selection of Transients

ANSI/ANS51.1(Reference-1) provides the guidance for plant conditions (normal, upset, emergency, and faulted) in Table 3-3, and relation between plant conditions and ASME Section III service levels in Table 3-6. The above Table 3-3 includes the transients affecting RCS temperature and pressure, and also the transients not affecting them. Transient items of US-APWR are selected and categorized by engineering judgment referring to the transients affecting RCS temperature and pressure listed in Table 3-3, but some items are categorized in different service levels from Table 3-6. (For example: complete loss of flow and steam generator tube rupture (SGTR) are categorized as service level C in US-APWR while these transients are categorized as service level B in ANSI/ANS51.1).

2.4 Description of Transients

Outlines of these transients are described in DCD 3.9.1(Reference -4). Computer Codes used for transient analysis are shown below. Details of these codes are described in DCD 15.0.2.2(Reference -4).

- MARVEL-M
- M-RELAP5
- WCOBRA/TRAC
- TWINKLE-M
- VIPRE-01M

Regarding to the transients obtained from computer codes, analysis conditions for each transient are shown in Table 2.4.1-1 to 2.4.4-5. These conditions have been chosen to maximize temperature and pressure difference in order to maximize stress on components.

Transient curves are shown in Figure 2.4.1-1 to 2.4.4-5.

Table 2.4.1-1 Ramp load increase between 15% and 100% of full power (5% of full power per minute)

Item		Analytical Conditions
Disturbance		Ramp load increase from 15% to 100% (5% of full power per minutes)
Initial reactor power		15% power
Control system	Rod control system	Actuated
	Pressurizer pressure control system	Actuated
	Turbine bypass control system	Not actuated
	Steam generator water level control system	Actuated
Moderator density reactivity coefficient		Minimum feedback
Doppler reactivity coefficient		Maximum feedback
Decay heat		With decay heat
Emergency feedwater		Not actuated
ECCS		Not actuated

Table 2.4.1-2 Ramp load increase between 50% and 100% of full power (5% of full power per minute)

Item		Analytical Conditions
Disturbance		Ramp load increase from 50% to 100% (5% of full power per minutes)
Initial reactor power		50% power
Control system	Rod control system	Actuated
	Pressurizer pressure control system	Actuated
	Turbine bypass control system	Not actuated
	Steam generator water level control system	Actuated
Moderator density reactivity coefficient		Minimum feedback
Doppler reactivity coefficient		Maximum feedback
Decay heat		With decay heat
Emergency feedwater		Not actuated
ECCS		Not actuated

Table 2.4.1-3 Ramp load decrease between 15% and 100% of full power (5% of full power per minute)

Item		Analytical Conditions
Disturbance		Ramp load decrease from 100% to 15% (5% of full power per minutes)
Initial reactor power		100% power
Control system	Rod control system	Actuated
	Pressurizer pressure control system	Actuated
	Turbine bypass control system	Not actuated
	Steam generator water level control system	Actuated
Moderator density reactivity coefficient		Minimum feedback
Doppler reactivity coefficient		Maximum feedback
Decay heat		With decay heat
Emergency feedwater		Not actuated
ECCS		Not actuated

Table 2.4.1-4 Ramp load decrease between 50% and 100% of full power (5% of full power per minute)

Item		Analytical Conditions
Disturbance		Ramp load decrease from 100% to 50% (5% of full power per minutes)
Initial reactor power		100% power
Control system	Rod control system	Actuated
	Pressurizer pressure control system	Actuated
	Turbine bypass control system	Not actuated
	Steam generator water level control system	Actuated
Moderator density reactivity coefficient		Minimum feedback
Doppler reactivity coefficient		Maximum feedback
Decay heat		With decay heat
Emergency feedwater		Not actuated
ECCS		Not actuated

Table 2.4.1-5 Step load increase of 10% of full power

Item		Analytical Conditions
Disturbance		Step load increase from 90% to 100%
Initial reactor power		90% power
Control system	Rod control system	Actuated
	Pressurizer pressure control system	Actuated
	Turbine bypass control system	Not actuated
	Steam generator water level control system	Actuated
Moderator density reactivity coefficient		Minimum feedback
Doppler reactivity coefficient		Maximum feedback
Decay heat		With decay heat
Emergency feedwater		Not actuated
ECCS		Not actuated

Table 2.4.1-6 Step load decrease of 10% of full power

Item		Analytical Conditions
Disturbance		Step load decrease from 100% to 90%
Initial reactor power		100% power
Control system	Rod control system	Actuated
	Pressurizer pressure control system	Actuated
	Turbine bypass control system	Not actuated
	Steam generator water level control system	Actuated
Moderator density reactivity coefficient		Minimum feedback
Doppler reactivity coefficient		Maximum feedback
Decay heat		With decay heat
Emergency feedwater		Not actuated
ECCS		Not actuated

Table 2.4.1-7 Large step load decrease with turbine bypass

Item		Analytical Conditions
Disturbance		Step load decrease in the turbine load from 100% to 0%
Initial reactor power		100% power
Control system	Rod control system	Actuated
	Pressurizer pressure control system	Actuated
	Turbine bypass control system	Actuated
	Steam generator water level control system	Actuated
Moderator density reactivity coefficient		Minimum feedback
Doppler reactivity coefficient		Maximum feedback
Decay heat		With decay heat
Emergency feedwater		Not actuated
ECCS		Not actuated

Table 2.4.1-8 Main feedwater cycling

Item		Analytical Conditions
Disturbance		Main feedwater flow step increase and decrease
Initial reactor power		0% power
Control system	Rod control system	Not actuated
	Pressurizer pressure control system	Actuated
	Turbine bypass control system	Actuated
	Steam generator water level control system	Not actuated
Moderator density reactivity coefficient		-
Doppler reactivity coefficient		-
Decay heat		No decay heat
Emergency feedwater		Not actuated
ECCS		Not actuated

Table 2.4.2-1 Loss of load

Item		Analytical Conditions
Disturbance		Instantaneous step load decrease from 100% to 0%
Initial reactor power		102% power
Control system	Rod control system	Not actuated
	Pressurizer pressure control system	Not actuated
	Turbine bypass control system	Not actuated
	Steam generator water level control system	Not actuated
Moderator density reactivity coefficient		Minimum feedback
Doppler reactivity coefficient		Minimum feedback
Decay heat		No decay heat
Emergency feedwater		Not actuated
ECCS		Not actuated

Table 2.4.2-2 Loss of offsite power

Item		Analytical Conditions
Disturbance		Loss of main feedwater Loss of AC power followed by a reactor/turbine trip on low steam generator water level
Initial reactor power		102% power
Control system	Rod control system	Not actuated
	Pressurizer pressure control system	Not actuated
	Turbine bypass control system	Not actuated
	Steam generator water level control system	Not actuated
Moderator density reactivity coefficient		Minimum feedback
Doppler reactivity coefficient		Maximum feedback
Decay heat		With decay heat
Emergency feedwater		Actuated (3 of 4 emergency feedwater pumps actuated)
ECCS		Not actuated

Table 2.4.2-3 RT from full power with no inadvertent cooldown

Item		Analytical Conditions
Disturbance		Reactor trip
Initial reactor power		102% power
Control system	Rod control system	Not actuated
	Pressurizer pressure control system	Not actuated
	Turbine bypass control system	Not actuated
	Steam generator water level control system	Not actuated
Moderator density reactivity coefficient		Minimum feedback
Doppler reactivity coefficient		Minimum feedback
Decay heat		No decay heat
Emergency feedwater		Not actuated
ECCS		Not actuated

Table 2.4.2-4 RT from full power with cooldown and no safety injection

Item		Analytical Conditions
Disturbance		Reactor trip with one main feedwater regulation valve stuck opened
Initial reactor power		102% power
Control system	Rod control system	Not actuated
	Pressurizer pressure control system	Not actuated
	Turbine bypass control system	Actuated
	Steam generator water level control system	Not actuated
Moderator density reactivity coefficient		Minimum feedback
Doppler reactivity coefficient		Minimum feedback
Decay heat		No decay heat
Emergency feedwater		Actuated
ECCS		Not actuated

Table 2.4.2-5 RT from full power with cooldown and safety injection

Item		Analytical Conditions
Disturbance		Reactor trip with one turbine bypass valve stuck open
Initial reactor power		102% power
Control system	Rod control system	Not actuated
	Pressurizer pressure control system	Not actuated
	Turbine bypass control system	Actuated
	Steam generator water level control system	Not actuated
Moderator density reactivity coefficient		Minimum feedback
Doppler reactivity coefficient		Minimum feedback
Decay heat		No decay heat
Emergency feedwater		Actuated
ECCS		Actuated

Table 2.4.2-6 Control rod drop

Item		Analytical Conditions
Disturbance		One control rod cluster bank drop
Initial reactor power		102% power
Control system	Rod control system	Not actuated
	Pressurizer pressure control system	Not actuated
	Turbine bypass control system	Actuated
	Steam generator water level control system	Not actuated
Moderator density reactivity coefficient		Minimum feedback
Doppler reactivity coefficient		Minimum feedback
Decay heat		No decay heat
Emergency feedwater		Not actuated
ECCS		Not actuated

Table 2.4.2-7 Cold over-pressure

Item		Analytical Conditions
Disturbance		Mass Input : Inadvertent startup of two Safety injection pumps Heat Input : Inadvertent startup of one reactor coolant pump
Initial reactor power		0% power
Control system	Rod control system	Not actuated
	Pressurizer pressure control system	Not actuated
	Turbine bypass control system	Not actuated
	Steam generator water level control system	Not actuated
Moderator density reactivity coefficient		-
Doppler reactivity coefficient		-
Decay heat		No decay heat
Emergency feedwater		Not actuated
ECCS		Actuated(Mass input case)

Table 2.4.2-8 Inadvertent safeguards actuation

Item		Analytical Conditions
Disturbance		Inadvertent actuation of ECCS actuation signal
Initial reactor power		102% power
Control system	Rod control system	Not actuated
	Pressurizer pressure control system	Not actuated
	Turbine bypass control system	Actuated
	Steam generator water level control system	Not actuated
Moderator density reactivity coefficient		Minimum feedback
Doppler reactivity coefficient		Minimum feedback
Decay heat		No decay heat
Emergency feedwater		Actuated
ECCS		Actuated

Table 2.4.2-9 Partial loss of reactor coolant flow

Item		Analytical Conditions
Disturbance		2 of 4 reactor coolant pumps trip
Initial reactor power		102% power
Control system	Rod control system	Not actuated
	Pressurizer pressure control system	Not actuated
	Turbine bypass control system	Actuated
	Steam generator water level control system	Not actuated
Moderator density reactivity coefficient		Minimum feedback
Doppler reactivity coefficient		Minimum feedback
Decay heat		No decay heat
Emergency feedwater		Not actuated
ECCS		Not actuated

Table 2.4.2-10 Inadvertent RCS depressurization (Umbrella case)

Item		Analytical Conditions
Disturbance		Inadvertent opening of safety depressurization valve
Initial reactor power		102% power
Control system	Rod control system	Not actuated
	Pressurizer pressure control system	Not actuated
	Turbine bypass control system	Actuated
	Steam generator water level control system	Not actuated
Moderator density reactivity coefficient		Minimum feedback
Doppler reactivity coefficient		Minimum feedback
Decay heat		No decay heat
Emergency feedwater		Actuated
ECCS		Actuated

Table 2.4.2-11 Inadvertent RCS depressurization (Inadvertent auxiliary spray)

Item		Analytical Conditions
Disturbance		Inadvertent opening of auxiliary spray valve
Initial reactor power		102% power
Control system	Rod control system	Not actuated
	Pressurizer pressure control system	Not actuated
	Turbine bypass control system	Actuated
	Steam generator water level control system	Not actuated
Moderator density reactivity coefficient		Minimum feedback
Doppler reactivity coefficient		Minimum feedback
Decay heat		No decay heat
Emergency feedwater		Not actuated
ECCS		Actuated

Table 2.4.2-12 Emergency feedwater cycling

Item		Analytical Conditions
Disturbance		Emergency feedwater flow step increase and decrease
Initial reactor power		0% power
Control system	Rod control system	Not actuated
	Pressurizer pressure control system	Actuated
	Turbine bypass control system	Actuated
	Steam generator water level control system	Not actuated
Moderator density reactivity coefficient		-
Doppler reactivity coefficient		-
Decay heat		No decay heat
Emergency feedwater		Actuated
ECCS		Not actuated

Table 2.4.3-1 Reactor coolant pipe break (Small LOCA)

Item		Analytical Conditions
Disturbance		1-inch pipe break in the cold leg
Initial reactor power		102% power
Control system	Rod control system	Actuated
	Pressurizer pressure control system	Not actuated
	Turbine bypass control system	Not actuated
	Steam generator water level control system	Not actuated
Moderator density reactivity coefficient		Same as DCD Chapter 15.6.5 evaluation
Doppler reactivity coefficient		Maximum feedback
Decay heat		With decay heat
Emergency feedwater		Actuated
ECCS		Actuated

Table 2.4.3-2 Small steam line break

Item		Analytical Conditions
Disturbance		10-inch pipe break
Initial reactor power		0% power
Control system	Rod control system	Not actuated
	Pressurizer pressure control system	Not actuated
	Turbine bypass control system	Not actuated
	Steam generator water level control system	Not actuated
Moderator density reactivity coefficient		-
Doppler reactivity coefficient		-
Decay heat		No decay heat
Emergency feedwater		Actuated
ECCS		Actuated

Table 2.4.3-3 Complete loss of flow

Item		Analytical Conditions
Disturbance		Loss of power to all reactor coolant pumps
Initial reactor power		102% power
Control system	Rod control system	Not actuated
	Pressurizer pressure control system	Not actuated
	Turbine bypass control system	Not actuated
	Steam generator water level control system	Not actuated
Moderator density reactivity coefficient		Minimum feedback
Doppler reactivity coefficient		Maximum feedback
Decay heat		With decay heat
Emergency feedwater		Actuated (3 of 4 emergency feedwater pumps actuated)
ECCS		Not actuated

Table 2.4.3-4 Small feedwater line break

Item		Analytical Conditions
Disturbance		6-inch pipe break
Initial reactor power		102% power
Control system	Rod control system	Not actuated
	Pressurizer pressure control system	Not actuated
	Turbine bypass control system	Not actuated
	Steam generator water level control system	Not actuated
Moderator density reactivity coefficient		Minimum feedback
Doppler reactivity coefficient		Maximum feedback
Decay heat		With decay heat
Emergency feedwater		Actuated (2 of 4 emergency feedwater pumps actuated)
ECCS		Not actuated

Table 2.4.3-5 SG tube rupture

Item		Analytical Conditions
Disturbance		Double-ended rupture of a single SG tube
Initial reactor power		102% power
Control system	Rod control system	Not actuated
	Pressurizer pressure control system	Not actuated
	Turbine bypass control system	Not actuated
	Steam generator water level control system	Actuated
Moderator density reactivity coefficient		Minimum feedback
Doppler reactivity coefficient		Maximum feedback
Decay heat		With decay heat
Emergency feedwater		Actuated (3 of 4 emergency feedwater pumps actuated)
ECCS		Actuated

Table 2.4.4-1 Reactor coolant pipe break (Large LOCA)

Item		Analytical Conditions
Disturbance		Double-ended hot leg guillotine
Initial reactor power		100% power
Control system	Rod control system	Not actuated
	Pressurizer pressure control system	Not actuated
	Turbine bypass control system	Not actuated
	Steam generator water level control system	Not actuated
Moderator density reactivity coefficient		Same as DCD Chapter 15.6.5 evaluation
Doppler reactivity coefficient		Maximum feedback
Decay heat		With decay heat
Emergency feedwater		Actuated
ECCS		Actuated

Table 2.4.4-2 Large steam line break

Item		Analytical Conditions
Disturbance		Double-ended rupture of a main steam line
Initial reactor power		0% power
Control system	Rod control system	Not actuated
	Pressurizer pressure control system	Not actuated
	Turbine bypass control system	Not actuated
	Steam generator water level control system	Not actuated
Moderator density reactivity coefficient		-
Doppler reactivity coefficient		-
Decay heat		No decay heat
Emergency feedwater		Actuated
ECCS		Actuated

Table 2.4.4-3 Large feedwater line break

Item		Analytical Conditions
Disturbance		Double-ended rupture of a main feedwater line
Initial reactor power		102% power
Control system	Rod control system	Not actuated
	Pressurizer pressure control system	Not actuated
	Turbine bypass control system	Not actuated
	Steam generator water level control system	Not actuated
Moderator density reactivity coefficient		Minimum feedback
Doppler reactivity coefficient		Maximum feedback
Decay heat		With decay heat
Emergency feedwater		Actuated (2 of 4 emergency feedwater pumps actuated)
ECCS		Not actuated

Table 2.4.4-4 RCP locked rotor

Item		Analytical Conditions
Disturbance		Seizure of the rotating assembly of a RCP rotor
Initial reactor power		102% power
Control system	Rod control system	Not actuated
	Pressurizer pressure control system	Not actuated
	Turbine bypass control system	Not actuated
	Steam generator water level control system	Not actuated
Moderator density reactivity coefficient		Minimum feedback
Doppler reactivity coefficient		Maximum feedback
Decay heat		With decay heat
Emergency feedwater		Not actuated
ECCS		Not actuated

Table 2.4.4-5 Control rod ejection

Item		Analytical Conditions
Disturbance		Single most reactive control rod is instantaneously ejected from the core
Initial reactor power		102% power
Control system	Rod control system	Not actuated
	Pressurizer pressure control system	Not actuated
	Turbine bypass control system	Not actuated
	Steam generator water level control system	Not actuated
Moderator density reactivity coefficient		- *1
Doppler reactivity coefficient		Temperature coefficient -20% from design
Decay heat		With decay heat
Emergency feedwater		Not actuated
ECCS		Not actuated

*1 Moderator temperature reactivity coefficient is assumed -20% from design.

Figure 2.4.1-1 Reactor Coolant Pump startup

Figure 2.4.1-2 Reactor Coolant Pump shutdown

Figure 2.4.1-3 Plant heat-up

Figure 2.4.1-4 Plant cooldown

Figure 2.4.1-5 Ramp load increase between 0% and 15% of full power

Figure 2.4.1-6 Ramp load decrease between 0% and 15% of full power

Figure 2.4.1-7 Ramp load increase between 15% and 100% of full power (5% of full power per minute)

Figure 2.4.1-8 Ramp load increase between 50% and 100% of full power (5% of full power per minute)

Figure 2.4.1-9 Ramp load decrease between 15% and 100% of full power (5% of full power per minute)

Figure 2.4.1-10 Ramp load decrease between 50% and 100% of full power (5% of full power per minute)

Figure 2.4.1-11 Step load increase of 10% of full power

Figure 2.4.1-12 Step load decrease of 10% of full power

Figure 2.4.1-13 Large step load decrease with turbine bypass

Figure 2.4.1-14 Main feedwater cycling

Figure 2.4.1-15 Core lifetime extension

Figure 2.4.1-16 Refueling

Figure 2.4.1-17 Turbine roll test

Figure 2.4.1-18 Primary leakage test

Figure 2.4.2-1 Loss of load

Figure 2.4.2-2 (1) Loss of offsite power

Figure 2.4.2-2 (2) Loss of offsite power

Figure 2.4.2-3 RT from full power

Case A - RT with no inadvertent cooldown

Figure 2.4.2-4 (1) RT from full power

Case B - RT with cooldown and no safety injection

Figure 2.4.2-4 (2) RT from full power

Case B - RT with cooldown and no safety injection

Figure 2.4.2-5 RT from full power

Case C - RT with cooldown and safety injection

Figure 2.4.2-6 Control rod drop

Figure 2.4.2-7 (1) Cold over-pressure (Mass input)

Figure 2.4.2-7 (2) Cold over-pressure (Heat input)

Figure 2.4.2-7 (3) Cold over-pressure (Heat input)

Figure 2.4.2-8 Inadvertent safeguards actuation

Figure 2.4.2-9 (1) Partial loss of reactor coolant flow

Figure 2.4.2-9 (2) Partial loss of reactor coolant flow

Figure 2.4.2-10 Inadvertent RCS depressurization - Umbrella case

Figure 2.4.2-11 Inadvertent RCS depressurization
- Inadvertent auxiliary spray

Figure 2.4.2-12 Emergency feedwater cycling

Figure 2.4.3-1 (1) Reactor coolant pipe break (Small LOCA)

Figure 2.4.3-1 (2) Reactor coolant pipe break (Small LOCA)

Figure 2.4.3-1 (3) Reactor coolant pipe break (Small LOCA)

Figure 2.4.3-2 (1) Small steam line break

Figure 2.4.3-2 (2) Small steam line break

Figure 2.4.3-3 (1) Complete loss of flow

Figure 2.4.3-3 (2) Complete loss of flow

Figure 2.4.3-4 (1) Small feedwater line break

Figure 2.4.3-4 (2) Small feedwater line break

Figure 2.4.3-5 (1) SG tube rupture

Figure 2.4.3-5 (2) SG tube rupture

Figure 2.4.4-1 (1) Reactor coolant pipe break (Large LOCA)

Figure 2.4.4-1 (2) Reactor coolant pipe break (Large LOCA)

Figure 2.4.4-2 (1) Large steam line break

Figure 2.4.4-2 (2) Large steam line break

Figure 2.4.4-3 (1) Large feedwater line break

Figure 2.4.4-3 (2) Large feedwater line break

Figure 2.4.4-4 (1) RCP locked rotor

Figure 2.4.4-4 (2) RCP locked rotor

Figure 2.4.4-5 Control rod ejection

3.0 REFERENCES

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2. ASME Boiler and Pressure Vessel Code Section III
3. 10CFR50.App.G,"Fracture Toughness Requirements"
4. Design Control Document for the US-APWR Revision 1, Mitsubishi Heavy Industries, Ltd., 2008