AREVA NP Inc.,

an AREVA and Siemens company

Engineering Information Record

Document No: 51 - 9105791 - 001

NGNP CONCEPTUAL DESIGN - PLANT DESIGN DUTY CYCLE

BEA Contract No. 000 75310

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Safety Related? ☐ YES ☒ NO

Does this document contain assumptions requiring verification? ☐ YES ☒ NO

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R/LR designates Reviewer (R), Lead Reviewer (LR)
A/A-CRF designates Approver (A), Approver of Customer Requested Format A-CRF

Record of Revision

Revision No.	Date	Pages/Sections/ Paragraphs Changed	Brief Description / Change Authorization
001	04/27/2009	Throughout	Miscellaneous editorial changes
		Page 6	Inserted a list of acronyms
		Tables 3.1 and 4.1 Event # 47	Replaced "operating" with "moderate" for consistency
		Table 4.1	Replaced HTS with MHTS for consistency with other AREVA NGNP documents
		Section 6.0 Reference 2	Updated revision number
000	03/19/2009	All	Initial Issue

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List of Acronyms

ASME	American Society of Mechanical Engineers
B&PV	Boiler and Pressure Vessel
BEA	Battelle Energy Alliance
DOE	Department of Energy
FC	Final Condition
FOAK	First-of-a-Kind
HTR	High Temperature Reactor
IC	Initial Condition
INL	Idaho National Laboratory
LR	Load Rejection
MHTGR	Modular High Temperature Gas-Cooled Reactor
MHTS	Main Heat Transport System
NGNP	Next Generation Nuclear Plant
NLR	Normal Load Ramp
NOAK	N^{th}-of-a-Kind
NP-MHTGR	New Production MHTGR
PDDC	Plant Design Duty Cycle
PH	Process Heat
PRA	Probabilistic Risk Assessment
RCCS	Reactor Cavity Cooling System
RLR	Rapid Load Ramp
RSCM	Reserve Shutdown Control Material
SCS	Shutdown Cooling System
SLC	Step Load Changes
SSCs	Structure, Systems and Components

1.0 INTRODUCTION

The Next Generation Nuclear Plant (NGNP) project will demonstrate the applicability of high temperature reactor (HTR) technology to high efficiency electricity production and process heat (PH) applications. The ultimate objective is to enable commercial deployment of HTR technology which will reduce the current reliance of process heat applications on fossil fuels. The Idaho National Laboratory (INL) is facilitating this project for the U.S. Department of Energy (DOE).

In the context of the NGNP project, several areas were identified as high priorities during the initial conceptual design phase. This includes the initial Plant Design Duty Cycle (PDDC). The PDDC establishes design events and their associated design number of occurrence over the plant lifetime. It also associates each event to a service level as defined in the ASME Boiler and Pressure Vessel Code. The plant systems, subsystems, and components must be designed to accommodate the full set of events specified in the PDDC as appropriate for each event's level of service.

This report contains an initial list of events for the AREVA NGNP Plant Design Duty Cycle. It will be updated periodically during the design process as requirement definition continues, the plant design matures, and as NGNP Probabilistic Risk Assessment (PRA) results become available.

1.1 Purpose and Scope

The Plant Design Duty Cycle provides supporting requirements for the design of the plant structures, systems and components (SSCs). The events are grouped into three categories with different operating conditions: 1) normal, 2) off-normal (forced/long outages), and 3) safety design.

Each category is generally defined by plant-level requirements and associated with an event frequency of occurrence range. Table 1-1 presents the three categories and their associated characteristics.

Table 1-1: Event Categories

EVENT CATEGORY	CHARACTERISTICS
1 – Normal Events	This category includes planned events determined by plant operational requirements and scheduled outages.
2 – Off-Normal Events	These events result from unplanned initiators such as normal failures or upsets. They are expected to happen during a typical plant lifetime and are included in the PDDC to ensure that plant availability and investment protection requirements are met.
3 – Safety Events	These events are included to ensure that the plant is properly designed to tolerate credible failure conditions and accidents. This allows the plant to meet safety requirements.

The development of the NGNP initial PDDC requires defining the events and event number of occurrences based on anticipated user requirements (defined in Section 1.2) and operating strategies (Section 2.2). Each event is also assigned a service level as defined in Section III of the ASME Boiler and Pressure Vessel (B&PV) Code [4] and summarized in Table 1-2 below.

Table 1-2: Service Levels and Associated Descriptions

SERVICE LEVEL	ASME B&PV CODE, SECTION III DESCRIPTIONS
A	Loadings arising from system startup, operation in the design power range, hot standby, and system shutdown.
B	Events that are anticipated to occur often enough that design should include a capability to withstand them without operational impairment. These events include any abnormal incidents not resulting in a forced outage and also forced outages for which the corrective action does not include any repair of mechanical damage.
C	Events that require shutdown for correction of the loadings or repair of damage in the system. These conditions have a low probability of occurrence, but are included to provide assurance that no gross loss of structural integrity will result as a concomitant effect of any damage developed in the system.
D	Combinations of loadings 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 only considerations of public health and safety are involved.

Category 1 (normal) events stem from customer operational requirements. In the absence of such requirements, the normal events were based on the Modular High Temperature Gas-Cooled Reactor (MHTGR) PDDC [1], NP-MHTGR PDDC [3], as well as on the anticipated process heat/cogeneration user requirements (Section 1.2). Category 2 (off-normal) and 3 (safety) events would usually be based on PRA. However, due to the lack of PRA specific to the NGNP, category 2 and 3 events and their associated design number of occurrences were based primarily on the MHTGR PDDC [1] and NP-MHTGR PDDC [3], which in turn are based on the MHTGR PRA [5].

1.2 Anticipated User Requirements

In the absence of customer operational requirements, a list of anticipated user requirements was developed and can be found in Table 1-3. This is a non-inclusive list of requirements which directly affect the PDDC.

Table 1-3: Overview of the Anticipated User Operational Requirements Affecting the PDDC

DESCRIPTION	ANTICIPATED REQUIREMENT
Applications	1 – Cogeneration 2 – Electricity production only 3 – PH production only
Modes of operation	Base load and load following
Plant design lifetime	60 calendar years
Refueling interval capability	18 month minimum
Number of modules	FOAK = 1 module NOAK = multiple modules
Load rejection in cogeneration mode	Accept: - External electric load rejection without disrupting PH supply - Full electric load rejection without disrupting PH supply - Full PH load rejection without disrupting electricity generation
Load rejection in full PH mode	Accept [75%] load rejection without trip
Load rejection in full electricity mode	Accept full external load rejection to house load without trip

1.3 Organization of Report

Section 2 of this report first includes a detailed description of the process followed to develop the initial PDDC for the NGNP. It then describes the anticipated operating strategy and the current AREVA NGNP reference design.

Section 3 contains the actual PDDC. Events and their associated number of occurrences and service levels are listed.

Section 4 provides an event description table with a concise explanation of each event listed in Section 3.

Section 5 provides conclusions of this study.

2.0 PROCESS

2.1 Approach

Conventionally, normal operational events are derived from customer operational requirements and basic plant maneuvering necessities, while off-normal and safety events come from PRA based on equipment failure data.

However, in the absence of both customer operational requirements and PRA, normal operational events and their associated design number of occurrences were derived from the normal events lists found in the MHTGR and NP-MHTGR PDDCs [1, 3] and expanded to reflect the anticipated user requirements found in Section 1.2. The MHTGR PDDC [1] and NP-MHTGR PDDC [3] (which are based on the MHTGR PRA [5]) form the basis for the off-normal and safety events tables with appropriate adjustments due to design variations between the MHTGR and the AREVA NGNP (i.e.: different number of loops and different secondary interfaces). For all three categories, levels of service were assigned in accordance with Table 1-2. All events and their associated numbers of occurrences and service levels were reviewed by an internal expert panel.

2.2 Anticipated Operating Strategy

The NGNP is a cogeneration concept expected to produce both electricity and PH as high pressure or low pressure steam. Although designed to operate primarily in cogeneration mode, the plant is capable of producing strictly electricity or strictly PH. It is estimated that the full PH mode of operation will occur approximately 25% of the time and is necessary in the event of, for instance, a turbine trip. While operating, the plant can run at base load, full load, go through transients, undergo load rejection testing, operate at part load, etc.

When producing only electricity, the NGNP acts similarly to the MHTGR which is used as a reference for this document. The electric load is raised or lowered with reactor power following the same trend. In cogeneration mode, the plant is producing both electricity and PH. In that case, various operational scenarios exist and include:

- raise/lower electric load, keep PH constant (reactor power increase/decrease follows)
- raise/lower electric load, raise/lower PH (reactor power increase/decrease follows)
- raise/lower PH, keep electric load constant (reactor power increase/decrease follows)
- keep the reactor power constant and shift load back and forth between PH and electricity production

Under normal operation, maximum utilization of fuel and facility will be achieved by running at rated reactor power and trading between electric and PH loads instead of changing loads independently. Additionally, constant rated reactor power will incur less stress on the reactor and primary systems. In that case, the load will shift back and forth between PH and electricity production depending on need.

The first-of-a-kind (FOAK) plant will be designed as a single module while the n^{th}-of-a-kind (NOAK) will operate with several modules in parallel.

2.3 NGNP Reference Design

The NGNP reference design description can be found in the AREVA NGNP Design Baseline Document [2]. The design concept is based on a 750°C steam cycle with two main heat transport loops per module to supply high temperature steam for both electricity generation and PH applications. The primary coolant (helium) carries reactor heat to a steam generator to produce steam in the secondary loop. The steam then transfers heat to steam reboilers where process steam is generated in the tertiary loop for various industrial processes. The secondary steam can also drive steam turbines for dedicated electricity production or cogeneration.

3.0 RESULTS

Table 3-1 provides the results of the Plant Design Duty Cycle. Each event is associated with a unique event number, 60-year design number of occurrences per module and level of service.

Table 3-1: Plant Design Duty Cycle Events with Associated Numbers of Occurrences and Service Levels

EVENT No.	EVENT	60-YEAR DESIGN NUMBER OF OCCURRENCES PER MODULE	SERVICE LEVEL
	Normal		
1	Cogeneration mode startup from depressurized conditions	[200]	A
2	Cogeneration mode startup with full helium inventory	[600]	A
3	Full electricity mode transition from cogeneration mode	[300]	A
4	Full PH mode transition from cogeneration mode	[300]	A
5	Scheduled shutdown from cogeneration mode to depressurized conditions	[150]	A
6	Scheduled shutdown from cogeneration mode with full helium inventory	[100]	A
7	Cogeneration mode transition from full electricity mode	[300]	A
8	Cogeneration mode transition from full PH mode	[300]	A
9	Rapid load ramp (5% per minute) (RLR)		
9a	PH RLR in full PH mode (increase/decrease)	[375 / 375]	A
9b	PH RLR in cogeneration mode (increase/decrease)	[1,500 / 1,500]	A
9c	Electric RLR in full electricity mode (increase/decrease)	[375 / 375]	A

EVENT No.	EVENT	60-YEAR DESIGN NUMBER OF OCCURRENCES PER MODULE	SERVICE LEVEL
9d	Electric RLR in cogeneration mode (increase/decrease)	[1,500 / 1,500]	A
10	Normal load ramp (NLR) (.5% per minute)		
10a	PH NLR in full PH mode (increase/decrease)	[1,250 / 1,250]	A
10b	PH NLR in cogeneration mode (increase/decrease)	[5,000 / 5,000]	A
10c	Electric NLR in full electricity mode (increase/decrease)	[1,250 / 1,250]	A
10d	Electric NLR in cogeneration mode (increase/decrease)	[5,000 / 5,000]	A
11	Step load changes (SLC) (±10%)		
11a	PH SLC in full PH mode (increase/decrease)	[375 / 375]	A
11b	PH SLC in cogeneration mode (increase/decrease)	[1,500 / 1,500]	A
11c	Electric SLC in full electricity mode (increase/decrease)	[375 / 375]	A
11d	Electric SLC in cogeneration mode (increase/decrease)	[1,500 / 1,500]	A
12	Grid demand variations	[1,500,000]	A
13	PH/electric load transfer (5% per minute) in cogeneration mode (transfer PH load/electric load)	[20,000 / 20,000]	A
	Off-Normal		
14	Main loop trip (1 loop)	[200]	B
15	Both loops trip	[100]	B
16	Reactor trip from 100% load	[345]	B
17	Reactor trip from 25% load	[125]	B

EVENT No.	EVENT	60-YEAR DESIGN NUMBER OF OCCURRENCES PER MODULE	SERVICE LEVEL
18	Load rejection (LR)		
18a	PH LR in full PH mode	[30]	B
18b	PH LR in cogeneration mode	[120]	B
18c	Electric LR in full electricity mode	[34]	B
18d	Electric LR in cogeneration mode	[135]	B
19	Turbine trip with recovery	[135]	B
20	Turbine trip with plant shutdown	[45]	B
21	Loss of condenser vacuum	[26]	B
22	Loss of offsite power with turbine trip	[6]	B
23	Loss of onsite and offsite power	[6]	B
24	Grid load reject with hold of house load and recovery	[60]	B
25	Feedwater flow decrease	[12]	B
26	Circulator overspeed	[14]	B
27	Excess feedwater heating	[14]	B
28	Feedwater flow increase	[12]	B
29	Circulator underspeed	[14]	B
30	Loss of feedwater heating	[14]	B
31	Accidental control rod group or RSCM insertion	[9]	B
32	Small primary coolant leak (up to [1 in^2])	[11]	B

EVENT No.	EVENT	60-YEAR DESIGN NUMBER OF OCCURRENCES PER MODULE	SERVICE LEVEL
33	Small steam generator tube leak (up to [0.1 lbm/s])	[24]	B
34	Small earthquake	[8]	B
35	Shutdown cooling heat exchanger tube leak	[5]	B
36	Loss of shutdown cooling heat exchanger cooling water during standby	[14]	B
37	Pressurized conduction cooldown	[15]	B
38	Pressurized conduction cooldown with restart of SCS at peak fuel temperature condition	[3]	C
39	Pressurized conduction cooldown with about 6h precooling on SCS	[3]	B
40	Pressurized conduction cooldown with reinstatement of main loop cooling within 1h	[5]	B
41	Pressurized conduction cooldown with small steam generator leak	[2]	C
42	Accidental control rod group continuous withdrawal	[5]	B
43	Accidental control rod group intermittent withdrawal	[3]	B
44	Moderate steam generator tube leak (up to [12.5 lbm/s])	[3]	B
45	Small feedwater or main steam pipe break	[3]	B
46	Moderate earthquake with main loop trip	[3]	C
47	Moderate earthquake with loss of onsite and offsite power	[3]	C
48	Small steam generator tube leak with moisture monitor failure	[3]	C
49	Main loop trip with failure of helium shutoff valve to close	[3]	C
50	Plant feedwater or main steam pipe break	[3]	C
51	Moderate primary coolant leak (up to [13 in^2])	[3]	C

EVENT No.	EVENT	60-YEAR DESIGN NUMBER OF OCCURRENCES PER MODULE	SERVICE LEVEL
52	Depressurized conduction cooldown	[3]	C
53	Depressurized conduction cooldown with small primary coolant leak (up to [1 in^2])	[3]	C
54	Depressurized conduction cooldown with moderate primary coolant leak (up to [13 in^2])	[3]	C
55	Depressurized conduction cooldown with moderate moisture ingress	[2]	C
	Safety Events		
56	Pressurized conduction cooldown with safety earthquake	[1]	D
57	Pressurized conduction cooldown with control rod withdrawal	[1]	D
58	Pressurized conduction cooldown with failure of control rod trip	[1]	D
59	Pressurized conduction cooldown with failure of helium shutoff valve to close	[1]	D

[ ] denotes preliminary results subject to change

4.0 EVENT DESCRIPTIONS

Table 4-1 provides a list of all the PDDC events with their unique event number and a concise description of the event.

Table 4-1: Plant Design Duty Cycle Events and Associated Descriptions

EVENT No.	EVENT	DESCRIPTION
	Normal Events	
1	Cogeneration mode startup from depressurized conditions	Initial Condition (IC): Reactor in cold, shutdown condition, primary coolant ~ ambient pressure. Final Condition (FC): Reactor producing rated steam condition, 25% rated load, automatic control, turbine and reboilers online, valves to PH plant open. Note: this event also encompasses startup from depressurized conditions to full PH or electricity mode.
2	Cogeneration mode startup with full helium inventory	IC: Reactor in cold, shutdown condition, full primary coolant inventory. FC: Reactor producing rated steam condition, 25% of rated load, automatic control, turbine and reboilers online, valves to PH plant open. Note: this event also encompasses startup with full helium inventory to full PH or electricity mode.
3	Full electricity mode transition from cogeneration mode	IC: Reactor producing rated steam condition, 100% rated load, automatic control, turbine and reboilers online, valves to PH plant open. FC: Reactor producing rated steam condition, 100% electric load, automatic control, turbine online, reboilers offline, valves to PH plant closed.
4	Full PH mode transition from cogeneration mode	IC: Reactor producing rated steam condition, 100% rated load, automatic control, turbine and reboilers online, valves to PH plant open. FC: Reactor producing rated steam condition, 100% PH load, automatic control, turbine offline, reboilers online, valves to PH plant open.

EVENT No.	EVENT	DESCRIPTION
5	Scheduled shutdown from cogeneration mode to depressurized conditions	IC: Reactor at 25% rated load, turbine and reboilers online, valves to PH plant open. FC: Reactor in cold shutdown condition, primary coolant depressurized, steam generator flooded out. Note: this event also encompasses shutdown from full PH or electricity mode to depressurized conditions.
6	Scheduled shutdown from cogeneration mode with full helium inventory	IC: Reactor at 25% rated load, turbine and reboilers online, valves to PH plant open. FC: Reactor in cold shutdown condition with full helium inventory, steam generator flooded out. Note: this event also encompasses shutdown from full PH or electricity mode with full helium inventory.
7	Cogeneration mode transition from full electricity mode	IC: Reactor producing rated steam condition, 100% electric load, automatic control, turbine online, reboilers offline, valves to PH plant closed. FC: Reactor producing rated steam condition, 100% rated load, automatic control, turbine and reboilers online, valves to PH plant open.
8	Cogeneration mode transition from full PH mode	IC: Reactor producing rated steam condition, 100% PH load, automatic control, turbine offline, reboilers online, valves to PH plant open. FC: Reactor producing rated steam condition, 100% rated load, automatic control, turbine and reboilers online, valves to PH plant open.
9	Rapid load ramp (5% per minute) (RLR)	All rapid ramp load increases/decrease (25-100% / 100-25%) at a rate up to 5% of rated load per minute. Causes increase/decrease in feedwater flow, helium flow, and reactor power.
9a	PH RLR in full PH mode	Reboiler load increase/decrease.
9b	PH RLR in cogeneration mode	Reboiler load increase/decrease.
9c	Electric RLR in full electricity mode	Turbine load increase/decrease.
9d	Electric RLR in cogeneration mode	Turbine load increase/decrease.
10	Normal load ramp (NLR) (.5% per minute)	All normal ramp load increases/decreases (25-100% / 100-25%) at a rate up to .5% of rated load per minute. Causes increase/decrease in feedwater flow, helium flow, and reactor power.

EVENT No.	EVENT	DESCRIPTION
10a	PH NLR in full PH mode	Reboiler load increase/decrease.
10b	PH NLR in cogeneration mode	Reboiler load increase/decrease.
10c	Electric NLR in full electricity mode	Turbine load increase/decrease.
10d	Electric NLR in cogeneration mode	Turbine load increase/decrease.
11	Step load changes (SLC) (±10%)	This event is due to grid or process plant upsets. Plant designed to accommodate step increase/decrease up to 10%. In response to step load increase/decrease, feedwater flow, helium flow, reactor power increase/decrease.
11a	PH SLC in full PH mode	Reboiler load increase/decrease.
11b	PH SLC in cogeneration mode	Reboiler load increase/decrease.
11c	Electric SLC in full electricity mode	Turbine load increase/decrease.
11d	Electric SLC in cogeneration mode	Turbine load increase/decrease.
12	Grid demand variations	Small amplitude (up to 1% of rated about steady state) load change due to load demand change on electric grid. Induced by small frequency variation of grid during steady state operation.
13	PH/electric load transfer (5% per minute) in cogeneration mode	Load transfer (up to 25%) at a rate up to 5% per minute between PH and electricity plant due to load demand or needs of the industrial process being fed.
Off-Normal Events		
14	Main loop trip (1 loop)	Loss of feedwater flow in one loop or one of the main circulators trips. If did not occur during the event, the circulator is tripped. Feedwater, isolation valves are closed. Turbine load is reduced enough for the remaining modules to support. Cooling is provided using the other loop's Main Heat Transport System (MHTS) for decay heat removal.

EVENT No.	EVENT	DESCRIPTION
15	Both loops trip	Feedwater flow is lost in both loops or both main circulators are tripped. The reactor and circulators are tripped. Feedwater, isolation valves are closed. Turbine is either tripped or load reduced enough for remaining modules to support. Cooling on Shutdown Cooling System (SCS) for decay heat removal.
16	Reactor trip from 100% load	IC: Reactor producing rated steam condition, 100% rated load, automatic control, turbine and reboilers online, valves to PH plant open. FC: At least one reactor module trips, rapid cooldown of reactor and MHTS components, isolation from main steam header which supplies turbine and reboilers. Turbine load reduced enough for remaining modules to support. Flow through steam generator of tripped reactor sustained to provide cooldown and decay heat removal.
17	Reactor trip from 25% load	IC: Reactor at 25% rated load, automatic control, turbine and reboilers online, valves to PH plant open. FC: At least one reactor module trips, rapid cooldown of reactor and MHTS components, isolation from main steam header which supplies turbine and reboilers. Turbine load reduced enough for remaining modules to support. Flow through steam generator of tripped reactor sustained to provide cooldown and decay heat removal.
18	Load rejection (LR)	
18a	PH LR in full PH mode	Large PH load rejection while running in full PH mode. Load rejection can be due to PH pump trip, accidental PH valve closure, etc. Greater than [75%] load rejection causes reactor and main loop trip.
18b	PH LR in cogeneration mode	Large PH load rejection while running in cogeneration mode. The modules accept full PH load rejection without disrupting the electricity generation.
18c	Electric LR in full electricity mode	Large electric load rejection while running in full electricity mode. Load rejection can be due to turbine trip, grid issues, etc. The modules accept full external load rejection to house load without trip.
18d	Electric LR in cogeneration mode	Large electric load rejection while running in cogeneration mode. The modules accept full electric load rejection without disrupting PH supply.

EVENT No.	EVENT	DESCRIPTION
19	Turbine trip with recovery	Turbine trip occurs and all modules are brought down to 25% rated load until turbine is restored.
20	Turbine trip with plant shutdown	Same as turbine trip above except that recovery cannot be achieved within 2 hours and modules are shutdown.
21	Loss of condenser vacuum	Loss of condenser vacuum causes loss of condensate flow to the deaerator tank and therefore loss of feedwater flow. All modules brought down to 25% rate load until condensate flow restored. Loss of feedwater flow causes reactor and circulators to be tripped. SCS provides cooling for decay heat removal.
22	Loss of offsite power with turbine trip	Loss of offsite power causes all modules to trip. SCS initially provides cooling for decay heat removal until offsite power is restored and the modules are restarted to conclude decay heat removal.
23	Loss of onsite and offsite power	Loss of onsite and offsite power causes all modules and the turbine to trip. The diesel generators are started and then SCS can provide cooling for decay heat removal.
24	Grid load reject with hold of house load and recovery	Grid load rejection causes large step load reduction on turbine. Turbine load reduced to house load. Turbine throttle valve controls speed (when the grid is restored, rapid load increase of the turbine load provides recovery).
25	Feedwater flow decrease	A large feedwater flow decreases causes reactor and main loop trip. SCS provides cooling for decay heat removal.
26	Circulator overspeed	A significant circulator overspeed causes reactor and main loop trip. SCS provides cooling for decay heat removal.
27	Excess feedwater heating	Excessive feedwater heating causes reactor and main loop trip. SCS provides cooling for decay heat removal.
28	Feedwater flow increase	A large feedwater flow increase causes reactor and main loop trip. SCS provides cooling for decay heat removal.
29	Circulator underspeed	A significant circulator underspeed causes reactor and main loop trip. SCS provides cooling for decay heat removal.

EVENT No.	EVENT	DESCRIPTION
30	Loss of feedwater heating	A severe loss of feedwater heating causes reactor and main loop trip. SCS provides cooling for decay heat removal.
31	Accidental control rod group or RSCM insertion	Accidental control rod group or Reserve Shutdown Control Material (RSCM) insertion causes a decrease in the reactor power, primary coolant and steam temperatures. Reactor module trips and main steam isolation valves are closed.
32	Small primary coolant leak (up to [1 in^2])	Breach (up to [1 in^2]) in the primary coolant pressure boundary causes loss of helium coolant and primary system pressure. As coolant flow slows, reactor power and circulator speed increase. Eventually, reactor trips due to low pressure. MHTS provides cooling for decay heat removal.
33	Small steam generator tube leak (up to [0.1 lbm/s])	Moisture ingress caused by small (up to [0.1 lbm/s]) steam generator tube leak into the primary system. Moisture ingress causes primary pressure increase and is detected on high moisture content by the moisture monitors causing reactor and main loop trip. Steam generator dump system is activated and the steam generator isolated for repair. SCS provides cooling for decay heat removal.
34	Small earthquake	A small earthquake (up to [0.06 g]) would generally cause operators to manually trip the reactor modules. Once event is over, normal startup is initiated. MHTS provides cooling for decay heat removal.
35	Shutdown cooling heat exchanger tube leak	In the event of a shutdown cooling heat exchanger tube leak, the heat exchanger is isolated. During operation when primary coolant pressure is greater than SCS pressure, the primary coolant leaks into the SCS loop. Reactor is shutdown to depressurize system to repair the leak. During depressurized conditions when primary coolant pressure is lower than SCS pressure, the SCS loop coolant leaks into the primary loop. When the leak is detected the SCS is isolated and the system is shutdown for repair.
36	Loss of shutdown cooling heat exchanger cooling water during standby	Loss of shutdown cooling heat exchanger cooling water during standby causes the SCS loop coolant temperature to increase. If online repair cannot be accomplished, the reactor is orderly shutdown for repair.

EVENT No.	EVENT	DESCRIPTION
37	Pressurized conduction cooldown	Pressurized conduction cooldown is a main loop trip with failure of SCS. The Reactor Cavity Cooling System (RCCS) provides cooling of reactor decay heat. After approximately 50 hours, the system temperature starts to decrease.
38	Pressurized conduction cooldown with restart of SCS at peak fuel temperature condition	Pressurized conduction cooldown is a main loop trip with failure of SCS. RCCS provides cooling of reactor decay heat. After 36 hours (peak fuel temperature condition), SCS is restarted and the system temperature starts to decrease.
39	Pressurized conduction cooldown with about 6h precooling on SCS	Main loop trip occurs and SCS provides cooling for the first 6 hours of the event. After 6 hours, SCS fails and RCCS provides cooling of reactor decay heat.
40	Pressurized conduction cooldown with reinstatement of main loop cooling within 1h	Pressurized conduction cooldown is a main loop trip with failure of SCS. RCCS provides cooling of reactor decay heat. After 1hour, MHTS is reinstated and the system temperature starts to decrease.
41	Pressurized conduction cooldown with small steam generator leak	Small steam generator tube leak. Reactor and main loop trip on high pressure. MHTS and SCS fail. RCCS succeeds.
42	Accidental control rod group continuous withdrawal	Accidental continuous control rod group withdrawal causes an increase in the reactor power, core outlet temperature until the reactor module trips. Cooling provided by MHTS.
43	Accidental control rod group intermittent withdrawal	Accidental intermittent control rod group withdrawal causes an increase in the reactor power and core outlet temperatures until the reactor module trips on high steam generator inlet helium temperature. Cooling provided on MHTS and SCS.
44	Moderate steam generator tube leak (up to [12.5 lbm/s])	Moisture ingress caused by moderate (up to [12.5 lbm/s]) steam generator tube leak into the primary system. Moisture ingress causes primary pressure increase and is detected on high moisture content by the moisture monitors causing reactor and main loop trip. Steam generator dump system is activated and the steam generator isolated for repair. SCS provides cooling for decay heat removal.
45	Small feedwater or main steam pipe break	A small feedwater or main steam pipe break causes a release of secondary coolant into the reactor or turbine building. Upon detection, main loop and reactor trip are initiated for the affected module. SCS provides cooling for decay heat removal.

EVENT No.	EVENT	DESCRIPTION
46	Moderate earthquake with main loop trip	A moderate (up to [0.18 g]) earthquake causes loss of feedwater flow or circulator trip. Main loop is tripped and SCS provides cooling for decay heat removal.
47	Moderate earthquake with loss of onsite and offsite power	Moderate earthquake (up to [0.18 g]) earthquake causes loss of onsite and offsite power. All modules and the turbine trip. Diesel generators are started and then SCS can provide cooling for decay heat removal.
48	Small steam generator tube leak with moisture monitor failure	Moisture ingress caused by small (up to [0.1 lbm/s]) steam generator tube leak into the primary system. Moisture ingress causes primary pressure increase and is detected a few hours later on high pressure due to failure of moisture monitor. This causes a much larger ingress than a normal small steam generator tube leak. The reactor pressure system causes reactor and main loop trip. Steam generator dump system is activated and the steam generator isolated for repair. SCS provides cooling for decay heat removal.
49	Main loop trip with failure of helium shutoff valve to close	Main loop trip with helium shutoff valve failing in the open position. SCS initiated. Helium backflow to steam generator causes decrease in reactor coolant flow, higher than normal reactor outlet temperature, and higher steam generator cooldown rate.
50	Plant feedwater or main steam pipe break	A plant feedwater or main steam pipe break causes a release of secondary coolant into the reactor or turbine building and affects multiple modules. Upon detection, main loop and reactor trip are initiated for the affected modules. SCS provides cooling for decay heat removal.
51	Moderate primary coolant leak (up to [13 in^2])	Breach (up to [13 in^2]) in the primary coolant pressure boundary causes loss of helium coolant and primary system pressure. As coolant flow slows, reactor power and circulator speed increase. Eventually, reactor trips due to low pressure. MHTS or SCS provides cooling for decay heat removal.
52	Depressurized conduction cooldown	Loss of forced cooling and coolant. MHTS and SCS are therefore unavailable and decay heat is removed by RCCS. Reactor trips on low pressure and the event results in higher reactor and vessel temperatures than for a pressurized conduction cooldown event.
53	Depressurized conduction cooldown with small primary coolant leak (up to [1 in^2])	Small primary coolant leak. Main loop and reactor trip on low pressure. Due to lack of significant natural circulation when primary coolant becomes depressurized, temperatures are higher than in a pressurized conduction cooldown event. MHTS and SCS fail. Cooling provided through RCCS.

EVENT No.	EVENT	DESCRIPTION
54	Depressurized conduction cooldown with moderate primary coolant leak (up to [13 in^2])	Moderate primary coolant leak. Main loop and reactor trip on low pressure. Due to lack of significant natural circulation when primary coolant becomes depressurized, temperatures are higher than in a pressurized conduction cooldown event. MHTS and SCS fail. Cooling provided through RCCS.
55	Depressurized conduction cooldown with moderate moisture ingress	Moderate steam generator tube leak. Reactor and main loop trip on high pressure but primary coolant pressure relief valve fails to close. MHTS and SCS fail. RCCS succeeds.
Safety Events		
56	Pressurized conduction cooldown with safety earthquake	An earthquake with ground acceleration greater than [0.35 g] causes a main loop and reactor trip. MHTS and SCS cooling fail. RCCS succeeds.
57	Pressurized conduction cooldown with control rod withdrawal	Spurious withdrawal of a control rod group causes reactor trip. MHTS and SCS cooling fail. RCCS succeeds.
58	Pressurized conduction cooldown with failure of control rod trip	Main loop trip with failure of most reactive control rod assembly. RSCM trip provides reactor shutdown. MHTS and SCS cooling fail. RCCS succeeds.
59	Pressurized conduction cooldown with failure of helium shutoff valve to close	Main loop trip with failure of helium shutoff valve in the open position. MHTS and SCS cooling fail. RCCS succeeds.

5.0 CONCLUSIONS

This initial AREVA NGNP Plant Design Duty Cycle provides designers with a preliminary list of events to refer to based on anticipated user requirements. This list of events, associated numbers of occurrences as well as service levels may be updated upon completion of the PRA and as requirements are finalized and design details become available.

6.0 REFERENCES

1. DOE-HTGR-86029, Rev. 3, *Modular High Temperature Gas-Cooled Reactor Plant Design Duty Cycle*, 1989.

2. 51-9103803-001, *NGNP Conceptual Design Baseline Document for Conventional Steam Cycle for Process Heat and Cogeneration*, March 2009.

3. CEGA-000002, Rev. 2, *NP-MHTGR Preliminary Plant Design Requirements Document*, October 1991.

4. 2007 ASME Boiler & Pressure Vessel Code, Section III, American Society of Mechanical Engineers, July 2008.

5. HTGR-86-011, Rev. 1, *Probabilistic Risk Assessment of the Modular HTGR Plant*, June 1986.