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GA 1485 (REV. 10/82)

GA Technologies Inc.

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ISSUE SUMMARY

TITLE		APPROVAL LEVEL
MODULAR HIGH TEMPERATURE GAS-COOLED REACTOR PLANT DESIGN DUTY CYCLE	☐ R & D ☐ DV & S ☒ DESIGN	5

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DISCLAIMER

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GA Technologies Inc.

ISSUE SUMMARY CONTINUATION SHEET

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MODULAR HIGH TEMPERATURE GAS-COOLED REACTOR PLANT DESIGN DUTY CYCLE	DOE-HTGR-86029	3

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LIST OF ABBREVIATIONS

AOO	anticipated operational occurrence
ASME	American Society of Mechanical Engineers
ATWS	anticipated transient without scram
B&PV	boiler and pressure vessel
CFR	code of federal regulations
DBE	design basis event
DH	decay heat
DOE	Department of Energy
EAB	exclusion area boundary
ECA	energy conversion area
EPBE	emergency planning basis event
F	frequency
FMEA	failure mode and effect analysis
FW	feedwater
HTGR	high temperature gas-cooled reactor
HTS	heat transport system
IPS	investment protection system
ISI	in-service inspection
LBE	licensing basis event
LOSP	loss of offsite power
MHTGR	modular high temperature gas-cooled reactor

NDTT	nil ductile transition temperature
NI	nuclear island
NSSS	nuclear steam supply system
OPDS	overall plant design specification
PAGs	protective action guidelines
PCDIS	plant control, data, and instrumentation system
PDDC	Plant Design Duty Cycle
P/F	power to flow ratio
RCCS	reactor cavity cooling system
RPS	reactor protection system
RSCE	reserve shutdown control equipment
RSCM	reserve shutdown control material
SCS	shutdown cooling system
SCHE	shutdown cooling heat exchanger
SG	steam generator
SGIT	steam generator inlet temperature (helium)
SRDC	safety-related design condition
SSCs	structures, systems, components
T-G	turbine-generator
TLRC	top-level regulatory criteria

DEFINITIONS

Affected Module: The reactor module affected by the transient event in concern.

Backflow: Reverse flow of helium through the main (or shutdown cooling) loop when the shutdown cooling (or main) loop is in operation.

Beginning-of-Cycle: At the beginning of a reactor fuel cycle.

Best Estimate: Transient computation conducted with nominal expected design and physical data without conservative margins.

Bimetallic Weld: Dissimilar weld of 2-1/4 Cr - 1 Mo and Incoloy 800H tube material in the steam generator bundle.

Blowdown: Rapid loss of coolant mass and energy from the primary or secondary cooling system.

Boundary Condition: The physical condition(s) specified at the end boundary(s) of a calculational model.

Building or Structure: A collection of structural and associated mechanical and/or electrical components treated as a unit for technical, administrative, or contractual purposes. The primary function of a building or structure, as opposed to a system, is to support a mechanical system against gravity effects, and to protect systems and their operating and maintenance personnel against certain environmental effects.

DEFINITIONS (Continued)

Bypass: Small quantity(s) of diverted flow(s) bypassing the main flow of a major component.

Category: A fundamental class or division.

Characterizer: A mathematical function used in control system signal synthesis.

Chemical Attack: The chemical reaction of a foreign fluid with reactor system materials following a steam generator leak or primary coolant leak.

Cold End: The cold coolant temperature end of the primary coolant loop between the steam generator outlet and the core inlet.

Condition: The thermodynamic or mechanical state of a process fluid or device which is identified through observable macroscopic properties.

Conduction Cooldown: The passive mode of reactor heat removal following failure of all normal active heat transfer systems. Reactor heat removal is accomplished by thermal conduction, radiation, and natural convection to the ambient.

Consequence: End result of a transient event which is quantified by radiological release and plant conditions.

Control Rod Group: A control rod group consists of three control rods which are inserted or withdrawn in unison.

Depressurized: The primary coolant inventory is reduced to near atmospheric condition.

DEFINITIONS (Continued)

Design Basis: Information which identifies the specific functions to be performed by a structure, system, or component of the plant, and the specific values or ranges of values chosen for controlling parameters as reference bounds for design. The values may be:

- Constraints derived from generally accepted state-of-the-art practices for achieving function.

- Requirements derived from analysis (based on theory and/or experiments) of postulated conditions for which a structure, system, or component must meet its functions.

Design Criterion: A basis for judging the acceptability of a particular quantity, action, procedure, or design solution.

Design Value: The nominal value of a plant or system operating parameter plus an appropriate margin.

Dump: Blowdown of the steam generator's secondary coolant inventory to the dump tank.

Encompassing: Bounding or enveloping in terms of consequence.

End-of-Cycle: At the end of a reactor fuel cycle.

End-of-Life: At the end of the MHTGR plant life (40 years).

Energy Conversion Area: That portion of the plant not included within the Nuclear Island.

Equilibrium: A thermodynamic or mechanical condition wherein the properties are not changing with time.

Equivalent Availability: Equivalent availability is mathematically defined as:

$$Ae = \int_{o}^{T_p} \frac{Pa(t)}{T_p} dt \quad ,$$

where: $Pa(t)$ = available net electrical power expressed as a percentage of design power as a function of time (t),

t = time,

T_p = time period (operational lifetime).

Event: A shift from an equilibrium condition due to a scheduled operation or an off-normal condition followed by a transition to an equilibrium condition.

Failure: Occurrence which results in a loss of capability of a component to perform its functions.

Floodout: A phase of steam generator cooldown wherein the tube bundle becomes completely filled with water.

Forced Cooling/Circulation: Active heat removal from the reactor via the main or shutdown cooling loop.

Forced Outage: Unplanned loss of power generation generally due to failure of a plant system or component.

Function: A statement of what is to be achieved. Also, a mathematical relationship between two or more thermodynamic or mechanical properties.

DEFINITIONS (Continued)

Functional Analysis: A system engineering technique or method used to develop (1) functions required to meet established goals, (2) the relationship between functions and related requirements, and (3) the basis and justification for design selections through which the prescribed goals are met. (Functional analysis is one part of the Integrated Approach.)

Functional Requirement: A bounding quantification that is derived from a functional need versus being imposed by an institutional standard.

Goal: An endpoint or accomplishment towards which an effort is directed. The Integrated Approach has identified the following specific goals supporting the overall objective:

1. Safe, economical power (top level - Goal 0).
2. Maintain plant operation (Goal 1).
3. Maintain plant protection (Goal 2).
4. Maintain control of radionuclide release (Goal 3).
5. Maintain emergency preparedness (Goal 4).

Hot End: The hot coolant temperature end of the primary coolant loop between the reactor core outlet and the steam generator inlet.

Institution: A significant practice, relationship, or organization in a society. Institutions include the utility/user, the federal government, and the Department of Energy, and state and local governments.

Institutional Requirements: A bounding quantification that is imposed by an institution as opposed to a quantification derived from a functional need.

Integrated Approach: A systems engineering technique for establishing and defending a well-developed nuclear plant design.

Isolation: A physical separation or removal of a component from the normal process flow path.

Leak: An off-normal condition due to failure of a component resulting in loss of coolant from its intended environment.

Limit: A maximum/minimum value for a parameter that is established by design considerations to ensure that plant goals are achieved.

Load: The level of steam production at rated conditions by one or more modules, usually expressed as a percentage of full power for a single module.

Load-Following: A mode of normal operation wherein the plant is in automatic control and responds to an external load demand signal.

Load Index: The load demand signal which establishes the intended operating level of a module.

Long Outage: Forced outage of greater than six months duration.

Main Loop: The heat transport system and associated components which provide normal reactor heat removal.

Main Loop Overcooling: An off-normal condition which results in decrease of steam temperature due to mismatch of primary coolant and secondary coolant conditions.

Malfunction: See Failure.

Middle-of-Life: At the middle of the MHTGR plant life.

Minimum/Maximum Control Value: Limits designed into control system hardware to restrict the variation of controlled parameters so that a protective action (and forced outage) is not invoked during normal control system action.

Model: A computer simulation of the physical system or component using mathematical expressions.

Module: The systems and components which provide steam production from one reactor.

Nil Ductile Transition Temperature: Vessel pressure versus temperature functions that define the permissible operating region to prevent embrittlement failure.

Nominal Value: Expected median value of a parameter at middle-of-cycle and middle-of-life operation for an average plant.

Normal Event: Planned event including scheduled outage, load-following, startup/shutdown, and base load operation.

Nuclear Island: That portion of the plant that has within its boundary the following:

1. The standard reactor modules and "safety-related" buildings, structures, systems, portions of systems, and components dedicated to assuring reactor shutdown, decay heat removal, fission product retention, and security of vital areas including new (unirradiated) fuel.

2. At the designer's discretion, buildings, structures, systems, portions of systems, and components that support reactor operation or investment protection, and are not "safety-related."

Nuclear Island Control: The portion of the plant control, data, and instrumentation system that deals with control of NI components and processes.

Off-Normal Event: Random, unplanned event which causes a departure from normal operation.

Offset Rupture: A complete circumferential break with full separation of the two severed pipe or tube ends.

Operating Life: The calendar time from receipt of the operating license to completion of plant power production.

Operating Limit: An allowed deviation of a parameter from nominal value to accommodate the normal operating envelopes for the various components.

Overshoot: The initial transient excursion of a controlled variable above the specified change followed by recovery to the specified value.

Parameter: A specific measurable and quantifiable aspect of a physical item.

Plant: All buildings, structures, systems, and components that together accomplish the process of energy production and conversion and support the human activities of administration, operation, and maintenance.

Plant Level: A level of detail in the plant analysis which encompasses major processes that affect energy production and power generation.

Plant Life: The calendar time from construction permit issuance to completion of plant power production.

Plant State: The condition of the physical plant (or module) at a particular time and place, as described by a set of appropriate variables associated with a plant goal.

Postcooling: Restoration of forced cooling decay heat removal following an interval of conduction cooldown.

Power-to-Flow Ratio: A dimensionless parameter derived by the ratio of normalized reactor neutron power to normalized reactor coolant flow rate. At design conditions, the power-to-flow ratio is unity.

Precooling: A period of forced cooling decay heat removal preceding a conduction cooldown.

Pressurized: The primary coolant mass inventory is at rated-power value.

Primary Circuit: The main flow path of the primary coolant through the core, reactor internals, and either the main or shutdown cooling loop.

Primary Coolant: The total helium inventory with its impurities contained within the reactor and steam generator vessels' pressure boundary is called primary coolant. Some of the primary coolant continuously flows to the helium purification system for purification and to the moisture monitor and analytical instrumentation systems.

Radionuclide Control Function: Any function performed to limit radionuclide release, or to limit the dose from direct radiation.

Ramp: A linear change of state of a specified parameter from an initial value to a final value.

Rated Value: The nominal value of a parameter at 100% plant energy conditions.

Reactor Cold Plenum: The plenum above the reactor core nominally containing essentially the coldest helium in the primary circuit.

Reactor Internals: The mechanical components (metallic and graphite) in the reactor system exclusive of the reactor core.

Recovery: Actions which may include cleanup and repair following an off-normal event to permit plant/module return to normal operation.

Relief Valve: A valve, or valves, venting primary or secondary fluid to the atmosphere to relieve overpressure conditions.

Requirement: The bounding quantification (limits) of a function.

Reshim: The insertion/withdrawal of a control rod group to force the control rod group being used for reactor control to be within a specified insertion range.

Runback: A controlled reduction in operating level in response to an upset or limit to a level of operation supportable under the condition existing.

Rupture: A sudden break in a vessel (pipe, tank, or vessel) containing secondary coolant.

Safety: A modifier, which when used in conjunction with nouns such as function and requirement, limits the scope of the nouns to radiological considerations only.

"Safety-Related": Identifier on equipment relied on to perform the functions required to limit releases under accident conditions to those allowed by 10CFR100.

Saturation: The condition at which a mixture of steam and water may coexist at a given pressure.

Scheduled Outage: Planned loss of power generation due to refueling, maintenance, and in-service inspection.

Secondary Coolant: Steam and/or water which is passed through the steam generator.

Service Limit: American Society of Mechanical Engineers definition of design requirements.

Setpoint: The demanded value of a controlled variable sent to the controller.

Shutdown: The transition from an operating state to a shutdown or nonoperating state.

Simulation: Mathematical model of a physical process, mechanism, or overall behavior of a part of the power plant.

<u>Spurious</u>: A nonspecific origin of a particular occurrence (such as a spurious reactor trip which might result from any electrical/electronic failure).

<u>Standard Reactor Module</u>: That portion of the Nuclear Island which is duplicated with the addition of each reactor. This would include, in part, the reactor, steam generator, main circulator, reactor building, reactor cavity cooling system, etc.

<u>Startup</u>: The transition from a nonoperating condition to an operating condition. For a module, going from a decay heat removal condition to a steaming condition, generally to the start of delivery of steam to the turbine.

<u>Standby</u>: The module standby condition is at or near rated steam temperature and pressure with reduced flow and power.

<u>Steady-State</u>: A condition where the main parameters of the system are essentially unchanging, having been at their current values for a significant period of time.

<u>Step</u>: An essentially instantaneous change of a variable from one value to another, generally only physically possible for electronic "demand" signals.

<u>Structure</u>: See definition of "building."

<u>Superheat</u>: The temperature increase of steam above that temperature at which water and steam may coexist (see saturation) at a given pressure.

System: A grouping of electrical, mechanical, and/or structural components and associated software treated as a unit for technical, administrative, or contractual purposes.

Transient: The changing of major plant parameters (i.e., steady-state) from an existing set of values to a new set of values. This is generally due to a demanded change in operating level or to the failure of a plant component.

Trip: Trip is the action to initiate the shutdown of a component, system, module, or plant.

Undershoot: The amount a controlled variable initially rebounds to below the specified change after having gone beyond the specified change (see overshoot).

1. INTRODUCTION/SUMMARY

The plant design duty cycle events and their design number of occurrences specified in Table 4-1 provide requirements for the design of plant structures, systems, and components to ensure that the plant goals are met. The duty cycle events are grouped into four major categories which encompass a range from normal operation to off-normal/accident conditions. These categories are derived by identifying the frequency (F) ranges and allocating top-level requirements of the Overall Plant Design Specification which govern the design duty cycle event selection. The four categories are:

1. Normal operation (includes scheduled outage).
2. Off-normal/forced outage ($F > 2.5 \times 10^{-2}$ per plant year).
3. Off-normal/long outage ($2.5 \times 10^{-2} > F > 10^{-3}$).
4. Off-normal/safety design ($10^{-3} > F > 10^{-4}$).

Specific design duty cycle events are derived from plant-level requirements and plant assessments which relate to plant performance, scheduled outage, forced outage, investment risk, and safety risk. Event selection and allocation into the duty cycle event categories is performed systematically by considering the following criteria:

1. Plant-level requirements.
2. Event frequency.
3. Plant response/consequence family.
4. Component design limits.

An encompassing set of duty cycle events is developed which covers all plant states and the functions required during these states. This ensures completeness of the event selection without encumbering the design duty cycle with superfluous events.

The design duty cycle presented herein is representative of events expected for the reference 2 (2 x 1) (two units with two reactor modules per turbine) configuration. The basis is the 4 x 2 (single unit with four reactor modules and two turbines) configuration which includes the capability of 2 x 1 operation. Pending update of the overall plant design specification (OPDS) and plant assessments to reflect the 2 (2 x 1) configuration, the duty cycle events/frequencies which are sensitive to turbine configuration will be updated and incorporated into the duty cycle.

Category 1 events are determined by plant performance/economics and scheduled outage requirements. Normal events include base load and load-following operation; refueling and other scheduled shutdowns for maintenance/inspection; and startups following scheduled and unscheduled outages. Load-following events include step load changes (±10% of rated load) and ramp load changes (up to 5% of rated load per minute).

Category 2 events result in an unscheduled reduction or complete loss of power production due to component failures or protective trips. These events are anticipated to occur within the plant lifetime and are included in the design duty cycle to ensure that the plant availability goal is achieved. Examples of forced outage events are main loop trip, reactor trip, turbine trip, primary/secondary coolant leaks, small earthquake, accidental control rod withdrawal, and control system failures. There is a total of 20 events which are driven primarily by the forced outage requirement of this category.

Category 3 events are included in the design duty cycle to ensure that the financial risk of severe transients complies with the property damage cost and long outage requirements. These events include moderate earthquake, pressurized conduction cooldown, depressurized conduction cooldown, moderate primary coolant leak, steam generator leak with moisture monitor failure, main loop trip with helium shutoff valve failure, and feedwater/main steam pipe rupture.

Category 4 events are included in the design duty cycle to ensure that the plant design meets the safety requirements. Category 4 events are derived from licensing basis events to demonstrate that the safety-related structures, systems and components are, by themselves, sufficient to control the release of radionuclides to acceptable levels. Events for this category include pressurized conduction cooldown with large earthquake, depressurized conduction cooldown with moderate moisture ingress, pressurized conduction cooldown with failure of control rod trip, and pressurized conduction cooldown with control rod withdrawal. Plant/module recovery following a Category 4 event is not required.

Design guidance, including minimum level of service limit for each event category is provided to assist the designer in the selection of appropriate design standards for evaluating component response to duty cycle events. Unless precluded by event initiator or the specific event sequence, plant components must withstand the attendant service loadings for each event to ensure compliance with top-level goals.

Design number of occurrences over a 40-year plant lifetime are specified for each event based on top-level requirements and 95th percentile frequency estimate. This provides assurance that the expected number of occurrences of all events will be accommodated in the plant design.

1.1. OBJECTIVE

This document defines the Plant Design Duty Cycle (PDDC) for the Modular High Temperature Gas-cooled Reactor (MHTGR). The duty cycle is a set of events and their design number of occurrences over the life of the plant for which the MHTGR plant shall be designed to ensure that the plant meets all the top-level requirements. The duty cycle is representative of the types of events to be expected in multiple reactor module-turbine plant configurations of the MHTGR.

A synopsis of each PDDC event is presented to provide an overview of the plant response and consequence.

1.2. BASIS

The PDDC was developed from the plant-level requirements specified in Ref. 1 and the plant assessments which relate to scheduled outage, forced outage/investment risk, and safety requirements. Plant requirements provide the basis for defining PDDC event categories and their mean frequency ranges. Specific duty cycle events were screened from the plant assessments and allocated into the duty cycle event categories using the methodology described below.

The PDDC was developed based on a plant operating life of 40 calendar years from the start of power operation and assumes an equivalent availability factor of 80% as specified in Ref. 1. The reference plant configuration is four standard reactor modules with two turbine-generators which operate in a two unit uncoupled 2 (2 x 1) configuration. Each unit is composed of two reactor modules headered together supplying steam to a single turbine-generator. Both base load and load-following operation are accommodated (Ref. 1), as well as occurrences which result from unplanned component failures or protective trips.

The design number of occurrences for duty cycle events was based on:

1. Plant performance requirements as stated in the OPDS (Ref. 1) and the plant scheduled outage report (Ref. 4) for normal events.

2. The 95th percentile frequency estimate for off-normal (unscheduled) events to assure confidence that the expected number of events are designed for.

1.3. METHODOLOGY

The Integrated Approach, a "top down," goal-oriented procedure, was employed to allocate PDDC requirements for the design of structures, systems, components (SSC) to ensure compliance with the plant goals. These top-level goals specify the nominal plant performance including economics, limit financial risk to the user due to unplanned events, and limit public risk attributable to unplanned radionuclide release.

Duty cycle events and their design number of occurrences were developed by comparing the plant assessments to the plant-level requirements.

The methodology of deriving the PDDC is summarized in the following steps:

1. Identify the frequency ranges/event categories to which the plant-level requirements apply.

2. Identify appropriate design standards for each frequency range/event category to ensure goal compliance.

3. Select events from plant assessments and plant requirements which lie within the frequency ranges/event categories.

4. Determine appropriate design number of occurrences for each event.

2. PLANT-LEVEL REQUIREMENTS

Plant-level requirements are top-level requirements imposed on the plant design by the utility/user, the government regulatory institutions, and the Department of Energy (DOE). These requirements were obtained from Ref. 1 and were used to govern the PDDC development. A screening procedure described in Section 3 was used to allocate requirements to form the basic event categories for the PDDC. Programmatic-related requirements and design selection/plant configuration requirements were not included because they do not directly control the PDDC selection. Where duplication of requirements existed in Ref. 1, the requirements were intentionally not repeated herein.

Four basic categories of requirements were identified and are described below:

1. Normal operation requirements.
2. Off-normal/forced outage requirements.
3. Long outage/investment protection requirements.
4. Safety/licensing requirements.

Category 1 requirements relate to the Integrated Approach Goal 1 objective to maintain plant operation. Category 2 and 3 requirements relate to Goal 2 objective to maintain plant protection. A distinction was made between forced outage and long outage requirements in order to delineate frequent, low-consequence events and less frequent, high-consequence events under Goal 2. Category 4 requirements relate to Goal 3 objective to maintain control of radionuclide release.

Requirement traceability numbers relating to Ref. 1 are included in the following sections for reference. The Top-Level Regulatory Criteria

(TLRC) for permissible radioactivity release and public dose limits (01.0201.121) is identified in all the requirements categories. Only the controlling release/dose limits of each category are specified below.

2.1. NORMAL OPERATION REQUIREMENTS

The normal operation requirements include scheduled outage allocation, plant operation/performance capabilities, and regulatory criteria for permissible dose levels and activity levels as follows:

1. ≤10% scheduled outage (01.0201.102).
2. Base-load and daily load cycle for 40 years (01.0201.212 and 215).
3. Utility/user load-following (01.0201.213).
4. Continuous automatic load range operation between 25% and 100% nominal feedwater flow (01.0201.228).
5. 5%/min load change capability (01.0201.229).
6. Capability to operate reactor modules and turbine-generators independently and at different power levels (01.0201.230).
7. Capability to shutdown/startup a single reactor module or turbine-generator system with remaining units in operation (01.0201.302 and 412).
8. ≤10% of 10CFR20 exposure for plant personnel (01.0201.311).
9. Startup/shutdown range between 0 and 25% nominal feedwater flow (01.0302.041).

10. Meet TLRC 10CFR20/10CFR50 Appendix I radioactivity release and public dose limits (01.0201.121, Goal 1).

Requirements for plant performance are used to derive the normal PDDC events involving load-following. Load-following events and their conservative design number of occurrences are taken directly from OPDS Table 2.1-1 (Ref. 1). Startup is categorized as a normal event with the design number of occurrences derived from the sum of shutdowns due to scheduled and unscheduled outages.

Plant scheduled outages include shutdowns for refueling, maintenance, and in-service inspection. The scheduled shutdowns were derived from the plant scheduled outage report (Ref. 4) and include pressurized and depressurized shutdowns. The expected scheduled outage allocation complies with the 10% per year averaged plant unavailability requirement for scheduled outages.

2.2. OFF-NORMAL/FORCED OUTAGE REQUIREMENTS

The off-normal/forced outage requirements include unscheduled (forced) outage allocation, off-normal plant operation capabilities, and regulatory criteria for permissible dose levels as follows:

1. ≤10% forced outage (01.0201.011).

2. ≥80% equivalent availability (01.0201.101).

3. Capability to sustain continuous operation at reduced plant output upon loss of function of major plant components where more than one is employed (01.0201.214 and 231).

4. Load rejection with hold of house load; turbine trip without reactor trip; 10% step load changes (01.0201.232).

5. Reloading/recovery at 5%/min following load rejection/turbine trip (01.0201.233).

6. Meet TLRC 10CFR50 Appendix I public dose limits (01.0201.121, Goal 3).

Plant forced outage requirements provide the basis for off-normal PDDC event selection. These events are expected to occur during a plant's lifetime (≥0.025 per plant year). The plant design must accommodate the forced outage events with recovery to rated power operation within the forced outage downtime allocation. Due to high frequency of forced outage events, which may vary from a few to several hundred cycles over the plant lifetime, downtime owing to forced outage events is limited to repair of initiator of events in order to meet the 10% per year averaged plant unavailability requirement for unscheduled outages.

2.3. LONG OUTAGE/INVESTMENT PROTECTION REQUIREMENTS

Long outage/investment protection requirements include long outage allocation, investment risk criteria, and regulatory criteria for permissible dose levels as follows:

1. Outages ≥6 months shall not contribute more than 10% of total forced outage unavailability (01.0201.111).

2. Annual value for risk to plant equipment or property shall not exceed the annual property damage insurance premium (01.0201.112).

3. Mean likelihood of exceeding safety-related design conditions (SRDCs) and risk of regulatory shutdown of other MHTGR plants shall be $<10^{-5}$ per plant year (01.0201.113).

4. Meet TLRC 10CFR100/protective action guidelines (PAGs) public dose limits (01.0201.121, Goal 3).

The long outage (1) and property damage (2) requirements are the controlling plant investment protection requirements which are used to develop investment protection PDDC events, including some that are not expected to occur in a single plant's lifetime, for which the plant design must accommodate with return to service within the forced outage downtime allocation. Some deformation or damage to components is permitted. In addition to repair of initiator of events, inspection or repair of consequential damage may be required before continued operation.

Plant unavailability due to investment protection outages must not exceed 10% of the total equivalent unavailability due to forced outages. The annual value for risk to plant equipment or property must not exceed the annual property damage insurance premium which is currently $4.5 million.

2.4. SAFETY/LICENSING REQUIREMENTS

Safety/licensing requirements include regulatory criteria for permissible dose levels and the manner in which compliance with these levels shall be achieved as follows:

1. Meet PAGs without public sheltering beyond exclusion area boundary (EAB) (01.0201.122).

2. Meet 10CFR100 without reliance on control room, control system, or operator (01.0203.002).

Safety, licensing, and regulatory requirements provide criteria for mortality risk and radiological release consequence of events. These requirements are used to develop licensing basis events (LBEs) (Ref. 3) from which safety design basis PDDC events are derived. The LBEs are screened using a hierarchy method which allocates LBEs into PDDC event categories beginning with normal operation and ending with safety design basis events. The LBEs with frequency between 10^{-3} and 10^{-4} per plant year form the basis for the PDDC safety design basis events for which the plant design must withstand to meet safety requirements. Only safety-related SSCs are employed in the mitigation of release consequence of the safety design basis events. Events with frequency below 10^{-4} per plant year are beyond PDDC basis. They are, however, evaluated in the plant-level assessments to demonstrate compliance with safety requirements.

3. METHODOLOGY

This section describes the approach used in formulating the PDDC events and their design number of occurrences. The Integrated Approach was used to define goal-oriented event categories from the plant-level requirements identified in Section 2. Design standards were specified for each event category to facilitate event selection and to assist the component designer in evaluating the duty cycle events. Event selection criteria were developed and screening of the wide spectrum of events from the plant assessments and plant performance requirements (Section 2.1) was performed to derive the encompassing events for the PDDC. Finally, the PDDC events were assigned to the respective event categories and the design number of occurrences for each PDDC event was evaluated.

3.1. GENERAL CONSIDERATIONS

The PDDC events may be characterized as normal events and off-normal events. The former are driven by the Goal 1 objective to maintain plant operation; events include load-following, scheduled outages, and startup/shutdown. The latter are driven by Goal 2 and 3 objectives to maintain plant protection and to maintain control of radionuclide release; events include the forced outage, long outage/investment protection, and safety design basis events.

The normal events are planned events which are obtained from the scheduled outage assessment (Ref. 4) and the normal operation requirements (Section 2.1). Normal events encompass startups, load-following, and scheduled shutdowns for refueling, maintenance, and in-service inspection.

Off-normal events are random, unplanned events which cause a departure from normal operation. These unplanned events are identified in the plant assessments (Refs. 2, 5, and 7) and specific event initiators such as main loop trip, turbine trip, earthquake, primary coolant leaks, etc., were delineated for the PDDC based on the event selection criteria described below.

3.2. EVENT CATEGORIES

Goal-oriented event categories were identified from the plant-level requirements as shown in Table 3-1. For each category the frequency range, a summary of requirements, and the controlling requirements are identified. Category 1 (normal operation) is a Goal 1 entity to maintain plant operation. Categories 2 through 5 are off-normal operation and, therefore, must comply with Goal 2 and Goal 3 objectives to maintain plant protection and maintain control of radionuclide release.

3.2.1. Frequency Ranges

Event frequency ranges are presented in Table 3-1 on a per plant basis. Category 1 events are specified to occur during normal operation. Frequency range is not applicable since these events are planned. Category 2 off-normal events are anticipated to occur at least once in the plant lifetime of 40 years and consistent with the LBE selection criteria (Ref. 3) for anticipated operational occurrences (AOOs) the frequency is greater than or equal to 2.5×10^{-2}.

Category 3 off-normal events may occur in the lifetime of a single plant. This category is driven by the long outage requirement and the property damage requirement. The frequency range is established by considering risk/consequence/requirements as follows:

$$\text{Risk} = \text{Frequency} \times \text{Consequence} \quad . \tag{1}$$

TABLE 3-1
PDDC EVENT CATEGORIES

Event Category Frequency/Plant Year	Top-Level Requirements			Controlling Requirements
	Goal 1	Goal 2	Goal 3	
1. Normal operation planned events	Load-following 10% scheduled outage 10CFR20 10CFR50 Appendix I	--	--	Performance/economics/scheduled outage
2. Off-normal $F \geq 2.5 \times 10^{-2}$	--	10% forced outages	10CFR50 Appendix I	Forced outages
3. Off-normal $2.5 \times 10^{-2} > F \geq 10^{-3}$	--	1% long-term outages property damage/insurance	10CFR100/PAGs	Long outages
4. Off-normal $10^{-3} > F \geq 10^{-4}$	--	Property damage/insurance	10CFR100/PAGs	10CFR100/PAGs
5. Beyond PDDC basis $10^{-4} > F > 5 \times 10^{-7}$	--	Property damage/insurance	PAGs	PAGs

For the upper frequency limit, risk is the downtime requirement (01.0201.111 per Ref. 1) for long outages (less than 10% of the overall 10% forced outage requirement) and consequence is outage exceeding 6 months.

$$(0.10)(0.10) = \text{Frequency} \times \frac{6\ \text{months}}{12\ \text{month/year}} \quad ,$$

$$\text{Frequency} = 0.02/\text{year} \quad .$$

This is close to the 0.025/year lower boundary for Category 2. For consistency with LBE criteria, the PDDC will use 0.025/year.

The lower frequency limit for Category 3 is driven by the property damage requirement (01.0201.112). Discussions with nuclear insurers indicate that about half of the annual insurance premium is generally returned to the plant owners for claims associated with frequent and relatively small losses. The other half is available for larger repairs and equipment replacement. The previous risk equation is used to derive an approximate frequency for which insurance alone is sufficient investment protection. Risk is one half of the annual insurance premium and consequence is the maximum claim of a plant write-off.

$$\frac{\$4.5 \times 10^6/\text{year}}{2} = (\text{Frequency})\ (\$2 \times 10^9) \quad ,$$

$$\text{Frequency} = 1.125 \times 10^{-3}/\text{year} \quad .$$

Roundoff to 10^{-3}/year for PDDC.

Category 4 off-normal events are not expected to occur in the lifetime of a single plant, but may occur within the lifetime of a population of plants. This category is driven by the safety requirements. Consistent with the LBE selection criteria (Ref. 3), the lower frequency limit is 10^{-4}/year for the design basis region.

Category 5 is beyond PDDC basis and is shown for comparison purposes. This category is identified in the LBE criteria as the emergency planning basis event (EPBE) region which extends down to 5 x 10^{-7}/year. EPBEs are not included in the PDDC.

3.3. SERVICE LEVELS AND DESIGN GUIDANCE

Design guidance is specified for the PDDC event categories in Table 3-2 to identify design limits which are considered in the event selection procedure (Section 3.4) and by the component designers in the assessment of duty cycle events. The American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code Section III (Ref. 8) is an applicable design standard for major pressure-retaining components such as the vessel and steam generator. Table 3-2 shows the ASME Code Section III allowable level of service limits (Service Levels) and provides qualitative guidance for damage consequence consistent with the specified Service Level. The Service Levels and design guidance indicated in Table 3-2 are minimum standards. The capability of plant components to perform their specified functions to ensure compliance with top-level requirements must be assured, even if this requires a more conservative design than that dictated by the specified standards.

The ASME Code Section III service limits are provided in Table 3-2 as guidance for designing the plant components even though most of them are not safety-related and are therefore not governed by ASME Code Section III. Qualitative design guidance for components not governed by ASME Code Section III is given in the right-hand side of Table 3-2. For these components, the designer may use the ASME Section III or other Code as appropriate to ensure the design meets the duty cycle requirements.

The definition of the Service Level A requires that all planned ("normal") events have to meet Service Level A. This is applicable to all plant SSCs.

TABLE 3-2
PDDC DESIGN GUIDANCE FOR ALL SSC

Frequency (Per Plant Year)	Minimum Service Level	Design Guidance Consistent with ASME B&PV Code Section III
Not applicable for planned events (normal operation)	A	Perform specified service. No downtime except for scheduled outages.
$\geq 2.5 \times 10^{-2}$	B	Component must withstand loadings without repair. Downtime for repair/cleanup of event must not exceed forced outage allocation.
$2.5 \times 10^{-2} > F \geq 10^{-3}$	C	Permit large deformation or damage. May require long outage (>6 months) for inspection or repair before continued operation.
$10^{-3} > F \geq 10^{-4}$	D	Component must be designed to perform functions to meet safety requirement. Permit gross general deformation or damage requiring extensive repair or removal from service. Return to service not required.
$10^{-4} > F > 5 \times 10^{-7}$	--	Beyond PDDC basis. Component must have capability to perform function to meet safety requirements.

The following pertinent investment protection requirements, as specified in the OPDS (Ref. 1), are driving the selection of the Service Level B and C:

1. It shall be a design goal that the calculated equivalent unavailability averaged over the lifetime of the plant owing to forced outages shall not exceed 10% (average of 876 h/year).

The above specified forced outage time is being met, as reported in the forced outage allocation report (Ref. 5) allowing time only for repair of initiator of events and not for repair of consequential damages. Therefore, for events that can be expected to occur during a plant life ("forced outage" events) Service Level B is assigned.

2. Outages of six months or greater shall not contribute more than 10% of the total equivalent unavailability from forced outages, including those not expected to occur in an individual plant's lifetime.

This requirement forces the lower probability events (frequencies between 2.5×10^{-2} and 1×10^{-3} per plant year) to result in no more than limited damage, such that the allocated repair time is not exceeded. These "long outage" events are, therefore, assigned Service Level C.

The safety requirements determine the Service Level D events and are only applicable to safety-related systems, structures, and components. These "safety design basis" events are not expected to occur in the lifetime of a population of plants ($10^{-3} > F > 10^{-4}$ per plant year).

As indicated in Table 3-2, events with frequency between 1×10^{-4} and 5×10^{-7} per plant year are beyond PDDC basis. These events are evaluated in the plant assessments to demonstrate that components have capability to perform their functions to meet safety requirements.

The ASME Code Section III (Ref. 8) define the following service level limits:

1. Level A Service Limits

   Level A Service Limits are those sets of limits which must be satisfied for all Level A Service loadings identified in the Design Specifications to which the component or structure may be subjected in the performance of its specified service function.

2. Level B Service Limits

   Level B Service Limits are those sets of limits which must be satisfied for all Level B Service loadings identified in the Design Specifications for which these Service Limits are designated. The component or structure must withstand these loadings without damage requiring repair.

3. Level C Service Limits

   Level C Service Limits are those sets of limits which must be satisfied for all Level C Service loadings identified in the Design Specification for which these Service Limits are designated. These sets of limits permit large deformations in areas of structural discontinuity which may necessitate the removal of the component from service for inspection or repair of damage to the component or structure. Therefore, the selection of these limits shall be reviewed by the Owner for compatibility with established system safety criteria.

4. Level D Service Limits

Level D Service Limits are those sets of limits which must be satisfied for all Level D Service loadings identified in the Design Specifications for which these Service Limits are designated. These sets of limits permit gross general deformations with some consequential loss of dimensional stability and damage requiring repair, which may require removal of the component from service. Therefore, the selection of this limit shall be reviewed by the Owner for compatibility with established system safety criteria.

3.4. EVENT SELECTION CRITERIA

The selection of the plant design duty cycle events is developed systematically based upon functional analyses. The functional analysis takes into consideration the plant-level requirements, event frequency, plant response/consequence family, and component design constraint criteria. The functional analysis covers all plant states and the functions required during these states and, therefore, provides a firm foundation for ensuring the completeness of the design duty cycle events without burdening the PDDC with superfluous events.

3.4.1. Plant-Level Requirements

The basic ground rules for event selection are derived from the plant-level requirements (Ref. 1). The requirements were screened into four groups as defined in Sections 2.1 through 2.4. These requirement groups were used to formulate the four event categories defined in Section 3.2. Event selection/screening is performed by comparing the candidate event frequency and consequence with the frequency ranges and requirements defined for the event categories.

3.4.2. Event Frequency

Events were screened according to the frequency criteria in Table 3-1. Those events with frequency below 10^{-4} per plant year were eliminated from the PDDC basis. The safety design basis events take exception to this rule. The basis for these events is the LBE (Ref. 3) selection which does use a 10^{-4} per plant year lower cutoff frequency for design basis events (DBEs). The safety design basis events were derived from the DBEs, using only safety-related SSCs to mitigate the event consequence. The actual frequencies of the safety design basis events may be considerably lower than 10^{-4} due to the nonmechanistic assumption of ignoring the mitigating responses of nonsafety-related SSCs.

3.4.3. Plant Response/Consequence

Events with similar plant response and safety consequence were combined into a single event family which encompasses a continuum of events, e.g., steam generator tube leaks between 0.1 to 12.5 lbm/s were characterized as moderate leaks with 12.5 lbm/s leak rate. A similar approach is used to encompass primary coolant leak events into small and moderate leak families. The frequency of occurrence for each event family is the integrated value for all events within that family.

3.4.4. Component Design Constraints

Consolidation of the continuum of events into discrete encompassing event families may impose hardships on component design. This may take the form of an excessive number of cycles of a worst case event or classification of the event into a more stringent Service Level category per Table 3-2. If necessary to avoid an overly conservative and costly design, the PDDC event selection is, or will be, reformulated to allocate events/frequencies into a more realistic number of event families.

Reformulation of event families involves an iterative procedure between the systems engineering and component design groups. The initial duty cycle event selection is evaluated by the component designer to determine if the component will meet the duty cycle requirements. If all requirements are met, then no change is required. If changes are required, the following options are available:

1. Change the duty cycle event allocation.
2. Change the affected component.
3. Change another component/system which will benefit the affected component.

Option 1 is available for a limited number of event categories. Service Level A and D events are essential to meeting top-level requirements for plant performance and safety. Therefore, option 1 is not available. For Service Level B and C events the event tree is reexamined to identify appropriate branches where the event family may be subdivided to produce two or more discrete events which have a significantly different consequence on the impacted component. Although the plant or system consequence may be very similar for the constituent events, this reformulation is reevaluated by the component designer to determine compliance with the duty cycle requirement.

Options 2 and 3 are involved if option 1 is unavailable or is ineffective. The changes are evaluated at the plant level to determine impact on plant assessments which may affect duty cycle event selections and at the component level to determine compliance with duty cycle requirements. Trade studies may need to be performed to determine the most cost-effective changes.

3.5. CATEGORIZATION OF EVENT FAMILIES

PDDC normal events are allocated to the normal operation (Service Level A) event category by virtue of Goal 1 requirements.

Each PDDC off-normal event family is assigned to an event category (Table 3-2) using a frequency of occurrence criterion. Consistent with the LBE selection procedure (Ref. 3), a factor of two over the mean frequency is used to provide margin and to dispel concern over events falling just below the frequency boundaries of a category and therefore, being designed to less rigorous criteria than those of a higher-frequency category.

3.6. DESIGN NUMBER OF OCCURRENCES

The design number of occurrences for normal events are obtained from either the user/utility requirement as stated in the OPDS Section 2 or the plant scheduled outage report (Ref. 4). Some of the number of occurrences are a summarization of other occurrences; e.g., the total number of startups is calculated from the number of scheduled shutdowns and the number of unscheduled or forced shutdowns. The number of occurrences for shutdown is the sum of the number of scheduled (typical refueling and scheduled maintenance) and unscheduled shutdowns. For transients which are not scheduled, but are predicted to occur, over a reasonable range of frequencies, the design number of cycles are derived using probabilistic assessment methods (Ref. 2) as follows:

The Failure Mode and Effect Analyses and Fault Tree/Event Tree analyses are used to identify the various event sequences which can result in forced outage. The analyses also allow identifying these various transients as either:

1. Single module events.
2. Multimodule events.

Events that are designated as single module events actually have some secondary effect on the other modules. The number of cycles due to the single module events on other modules with second order effect are currently not included. Events with similar plant response/consequence are grouped into a single event family.

The quantification of the Fault Tree/Event Tree Analysis, using established failure rate database, provides the expected frequency of occurrence, including uncertainties, for these transients. For the design duty cycles, the numbers of occurrences of the unplanned events are based on a 95% confidence level in order to provide design margin and to assure that the expected number of occurrences will be designed for.

3.6.1. Normal Events

The design number of occurrences of normal events were obtained directly from the utility/user requirements for load-following events (Ref. 1) and from the scheduled outage report (Ref. 4) for the scheduled shutdown events. The total number of startups was derived from the sum of scheduled shutdowns and unscheduled shutdowns due to forced outage/ long outage events, i.e., return to service is anticipated for shutdowns attributed to normal and off-normal events exclusive of the safety design basis events.

3.6.2. Off-Normal Events

Roundoff and integerization of design numbers of occurrences was performed to eliminate fractional occurrences. This ensures that the plant will be designed to withstand at least one cycle of any PDDC event. For off-normal events, the number of plant/module restarts following an unscheduled outage is calculated from the 95% confidence. The design number of occurrences for each off-normal event is determined by the sum of number of restarts for that event plus one to ensure at least one design occurrence. The following equation was used to calculate design number of occurrences for multimodule events:

$$N_D = \text{INTEGER}\ (F_{95} \times 40 + 0.5) + 1 \quad , \qquad (2)$$

where N_D = design number of occurrences per plant or module,

F_{95} = 95% confidence frequency per plant year,

INTEGER = mathematical function for eliminating the fractional component of a real number.

For single module events, the design number of occurrences per module is calculated by using the appropriate module frequency instead of plant frequency in the above equation. When the calculated per plant occurrence is less than or equal to two, the per module occurrence is set equal to the per plant occurrence. This allocation ensures that any given module in the 4-module plant will be designed to accommodate a sufficient number of occurrences of the lower frequency single module events without a prior knowledge of which module(s) will actually experience these events.

4. DESIGN DUTY CYCLE

The events which make up the design duty cycle for the MHTGR plant are listed in Table 4-1. The daily load following cycle shown in Fig. 4-1 is used when the plant is to be operated in a load-following mode. The resultant number of normal load ramps are a part of the total normal load increases and decreases.

Included in the list of events found in Table 4-1 are normal operating events such as load changes and startups and shutdowns, as well as events which result from protective trips or from the failure of plant components or systems. The individual events may have several initiating causes. The total number of design occurrences for each event is the sum of occurrences due to all initiating causes. The number of occurrences is based on the 95th percentile frequency estimate to assure that the expected number of occurrences will be designed for.

The number of occurrences for each event has been specified on a per plant and per reactor module basis. When performing an analysis of balance of plant equipment such as the turbine, the design number of occurrences per module must be multiplied by the number of reactor modules associated with that component for event sequences in which the initiating event is at the module level. Thus, if two reactor modules supply steam to a single turbine, then that turbine would be exposed to twice as many reactor trip transients as the number shown in Table 4-1. For event sequences in which the initiating event is external to the plant [e.g., earthquake, loss of offsite power (LOSP)], the design number of occurrences is determined by the per plant occurrences.

TABLE 4-1
PLANT DESIGN DUTY CYCLE EVENTS

		Frequency Per Plant Year		Design No. of Occurrences Per		
		Mean	95% Confidence	Module	Plant	Basis[a]
1.	Normal Events (Service Level A)[b]					
1.1	Module startup from depressurized conditions	--	8.5	85	[340]	1
1.2	Module startup with full helium inventory	--	59.1	591	[2,364]	2
1.3	Scheduled module shutdown to depressurized conditions	--	3.4	34	136	3
1.4	Scheduled module shutdown with full helium inventory	--	8.6	86	344	3
1.5	Rapid load increase (5% per minute, 25% to 100%)	--	25.0	1,000	1,000	4
1.6	Normal load increase (0.5% per minute, 25% to 100%)	--	520.0	20,800	20,800	4
1.7	Rapid load decrease (5% per minute, 100% to 25%)	--	25.0	1,000	1,000	4
1.8	Normal load decrease (0.5% per minute, 100% to 25%)	--	437.5	17,500	17,500	4
1.9a	Step load increase (+10%, 90% to 100%)	--	25.0	1,000	1,000	4
1.9b	Step load increase (+10%, 25% to 35%)					
1.10a	Step load decrease (-10%, 100% to 90%)	--	25.0	1,000	1,000	4
1.10b	Step load decrease (-10%, 35% to 25%)					

TABLE 4-1 (Continued)

		Frequency Per Plant Year		Design No. of Occurrences Per		
		Mean	95% Confidence	Module	Plant	Basis[a]
1.11	Grid frequency variations (±1% load fluctuations about steady-state)	--	2.5×10^4	1×10^6	1×10^6	4
2.	<u>Off-Normal Events (Service Level B)</u>[b]					
2.1	Main loop trip	3.2	9.5	85	[340]	5, 7
2.2	Reactor trip from 100% load	7.42	22.3	223[c]	892	5, 7
2.3	Reactor trip from 25% load	2.74	8.3	83[c]	332	5
2.4	Turbine trip with recovery	1.0	2.25	90	90	4, 5, 7
2.5	Turbine trip with plant shutdown	2.4×10^{-1}	7.25×10^{-1}	30	30	4, 5
2.6	Loss of condenser vacuum	1.58×10^{-1}	4.03×10^{-1}	17	17	5
2.7	Loss of offsite power with turbine trip	2.5×10^{-2}	7.5×10^{-2}	4	4	5, 7
2.8	Grid load reject with hold of house load and recovery	7.0×10^{-2}	9.8×10^{-1}	40	40	5
2.9a	Feedwater flow decrease	2.6×10^{-1}	7.7×10^{-1}	8	32	5
2.9b	Circulator overspeed	2.8×10^{-1}	8.4×10^{-1}	9	35	5
2.9c	Excess feedwater heating	7.0×10^{-2}	2.1×10^{-1}	9	9	5
2.10a	Feedwater flow increase	2.6×10^{-1}	7.7×10^{-1}	8	32	5
2.10b	Circulator underspeed	2.8×10^{-1}	8.4×10^{-1}	9	35	5
2.10c	Loss of feedwater heating	7.0×10^{-2}	2.1×10^{-1}	9	9	5

TABLE 4-1 (Continued)

		Frequency Per Plant Year		Design No. of Occurrences Per		
		Mean	95% Confidence	Module	Plant	Basis[a]
2.11	Accidental control rod group or RSCM insertion	1.4×10^{-1}	4.6×10^{-1}	6	19	5, 6
2.12	Small primary coolant leak (up to 1 in.^{2})	2.02×10^{-1}	6.06×10^{-1}	7	25	5, 7
2.13	Small steam generator tube leak (up to 0.1 lbm/s)	2.64×10^{-1}	1.18	12	48	5, 7
2.14	Small earthquake (≤0.06 g)	2.5×10^{-2}	1×10^{-1}		5	5
2.15	Shutdown cooling heat exchanger tube leak	7.01×10^{-2}	2.10×10^{-1}	3	9	5
2.16	Loss of shutdown cooling heat exchanger cooling water during standby	1.75×10^{-1}	8.30×10^{-1}	9	34	5
2.17	Pressurized conduction cooldown with ~6 h precooling on SCS	3.07×10^{-2}	6.18×10^{-2}	2	3	6
2.18a	Accidental control rod group continuous withdrawal	8.2×10^{-2}	1.8×10^{-1}	3	8	5, 7
2.18b	Accidental control rod group intermittent withdrawal	2.5×10^{-2}	7.5×10^{-2}	1	4	5
2.19	Moderate steam generator tube leak (up to 12.5 lbm/s)	1.84×10^{-2}	1.20×10^{-1}	2	6	5, 7
2.20	Small feedwater or main steam pipe break	2.5×10^{-2}	7.5×10^{-2}	2	4	5

TABLE 4-1 (Continued)

		Frequency Per Plant Year		Design No. of Occurrences Per		
		Mean	95% Confidence	Module	Plant	Basis[a]
3.	<u>Off-Normal Events (Service Level C)</u>[b]					
3.1	Moderate earthquake (0.18 g) with main loop trip	6×10^{-3}	2.5×10^{-2}	2	2	5, 7
3.2	Small steam generator tube leak with moisture monitor failure	6×10^{-4}	1.8×10^{-3}	2	2	6, 7
3.3a	Pressurized conduction cooldown	6.7×10^{-3}	1.5×10^{-2}	2	2	6, 7
3.3b	Pressurized conduction cooldown with post-cooling on SCS after ~36 h	8.0×10^{-3}	1.8×10^{-2}	2	2	6
3.4	Main loop trip with failure of helium shutoff valve to close	9.6×10^{-4}	7.65×10^{-3}	1	1	5, 6
3.5	Plant feedwater or main steam pipe break	7.01×10^{-4}	7.01×10^{-3}	2	2	5
3.6	Moderate primary coolant leak (up to 13 in.^{2}) with cooling on HTS or SCS	6.13×10^{-3}	1.84×10^{-2}	2	2	5, 6, 7
3.7	Depressurized conduction cooldown	1.4×10^{-3}	8.7×10^{-3}	1	1	5, 6, 7
4.	<u>Off-Normal Events (Service Level D)</u>[b]					
4.1	Pressurized conduction cooldown with large (0.35 g) earthquake	6.7×10^{-6}	$\sim 2 \times 10^{-5}$	1	1	7[d]
4.2	Depressurized conduction cooldown with moderate moisture ingress	1.7×10^{-6}	$\sim 5 \times 10^{-5}$	1	1	7[d]

TABLE 4-1 (Continued)

		Frequency Per Plant Year		Design No. of Occurrences Per		
		Mean	95% Confidence	Module	Plant	Basis[a]
4.3	Pressurized conduction cooldown with control rod withdrawal	7.5×10^{-5}	$\sim 2 \times 10^{-4}$	1	1	7[d]
4.4	Pressurized conduction cooldown with failure of control rod trip	3.1×10^{-6}	$\sim 1 \times 10^{-5}$	1	1	7[d]

[a] Basis for PDDC Events

1. Sum of scheduled depressurized shutdowns and unscheduled (off-normal) depressurized shutdowns.
2. Sum of scheduled pressurized shutdowns and unscheduled (off-normal) pressurized shutdowns.
3. DOE-HTGR-86046, Rev. 7, "Plant Scheduled Outage Report Update 4 x 350 MW(t) MHTGR Plant," August 1986.
4. DOE-HTGR-86004, Rev. 7, Overall Plant Design Specification Modular High Temperature Gas-cooled Reactor," Section 2.1 Utility/User Requirements, August 30, 1988.
5. DOE-HTGR-86011, Rev. 3, "Probabilistic Risk Assessment for the Standard MHTGR," January 1987.
6. DOE-HTGR-88235, "Status of Investment Risk Studies," November 10, 1988.
7. DOE-HTGR-86034, Rev. 1, "Licensing Basis Events for the Standard MHTGR," February 27, 1987.

[b] American Society of Mechanical Engineers, Boiler and Pressure Vessel Code, Section III, 1986, through 1986 addenda.

[c] For components where reactor trip from 25% load is worse, the breakdown is 107 trips from 100% and 199 trips from 25%.

[d] Basis derived from corresponding DBE using only safety-related SSC to mitigate consequence.

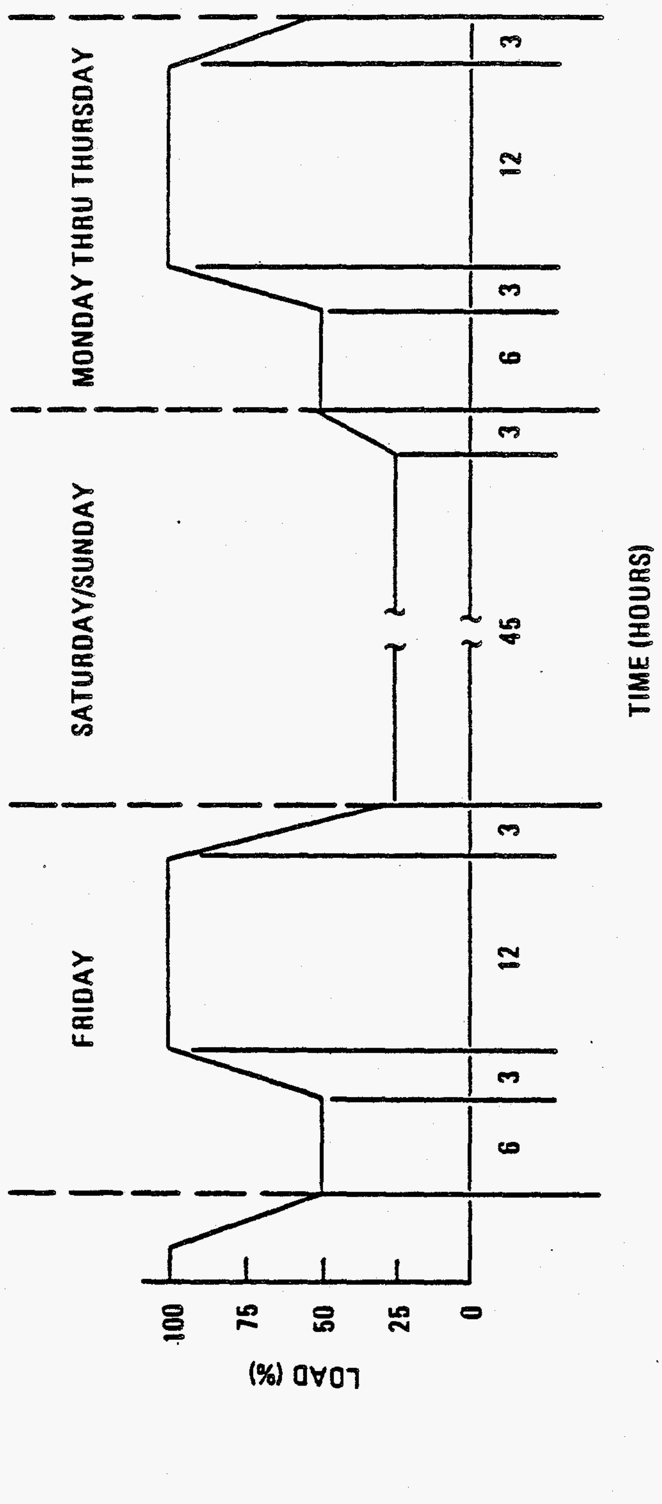

Fig. 4-1. Weekly load-following cycle

The appropriate ASME B&PV Code Level of Service Limits has been identified for each of the four event categories listed in Table 4-1. Although these Levels of Service Limits are intended primarily for passive, pressure-retaining components, they are also permitted for active components, provided that the capability of the component to perform its specified functions (as specified in the component's design requirements) can be demonstrated under the service loading combination. As indicated in Section 3.3, the capability of plant components to perform their specified functions and to meet plant requirements must be assured, even if this requires a more conservative design than that dictated by the specified service limit level alone. This ensures a component design which meets the component's design requirements for all design events.

5. DESCRIPTION OF DESIGN DUTY CYCLE EVENTS

The following event descriptions describe the transients which are representative of the duty cycle events. Table 5-1 presents a summary of plant response/consequences for the duty cycle events. The table enumerates on a per module basis the number of significant event sequence actions including reactor heat removal, reactor trip, primary coolant pumpup/pumpdown, radionuclide release, and chemical attack for each PDDC event.

5.1. NORMAL EVENTS

Normal events are planned activities which the plant must accommodate in order to fulfill the Goal 1 requirements. Normal events encompass startups, load-following, and scheduled shutdowns for refueling, maintenance, and in-service inspection.

5.1.1. Module Startup From Depressurized Conditions

This event begins with the reactor in a cold, shutdown condition with the primary coolant depressurized to near ambient pressure. The number of depressurized startups encompasses the sum of depressurized shutdowns due to schedule outages (Section 5.1.3) and unscheduled outages (off-normal events, Table 5-1) which result in primary coolant depressurization. There are a total of 85 module startups from depressurized conditions for each module, of which 34 are startups following scheduled shutdowns.

The extreme of this initial condition is the startup of all modules with the turbine initially in a cold, offline state. The startup is

TABLE 5-1
PLANT DESIGN DUTY CYCLE EVENT CONSEQUENCE SUMMARY

		Design Number of Occurrences Per Module										
		Reactor Heat Removal				Reactor Trip		Primary Coolant Radionuclide			Chemical Attack	
		Power Level (%)	HTS	SCS	RCCS	Control Rods	RSCM	Release	Pumpdown	Pumpup	Moisture	Air
1.	Normal Events											
1.1	Module startup from depressurized conditions	25	85							85		
1.2	Module startup with full helium inventory	25	591									
1.3	Scheduled module shutdown to depressurized conditions	DH	34	34		34			34			
1.4	Scheduled module shutdown with full helium inventory	DH	86	86		86						
1.5	Rapid load increase (5% per minute, 25% to 100%)	25 to 100	1,000									
1.6	Normal load increase (0.5% per minute, 25% to 100%)	25 to 100	20,800									
1.7	Rapid load decrease (5% per minute, 100% to 25%)	100 to 25	1,000									
1.8	Normal load decrease (0.5% per minute, 100% to 25%)	100 to 25	17,500									
1.9a	Step load increase (+10%, 90% to 100%)	90 to 100	1,000									
1.9b	Step load increase (+10%, 25% to 35%)	25 to 35										
1.10a	Step load decrease (-10%, 100% to 90%)	100 to 90	1,000									
1.10b	Step load decrease (-10%, 35% to 25%)	35 to 25										
1.11	Grid frequency variations (±1% load fluctuations about steady-state)	25 to 100	10^6									

TABLE 5-1 (Continued)

		Design Number of Occurrences Per Module										
		Reactor Heat Removal				Reactor Trip		Primary Coolant Radionuclide			Chemical Attack	
		Power Level (%)	HTS	SCS	RCCS	Control Rods	RSCM	Release	Pumpdown	Pumpup	Moisture	Air
2.	Off-Normal/Forced Outage Events											
2.1	Main loop trip	DH		85		85			15			
2.2	Reactor trip from 100% load	DH	201	22		223						
2.3	Reactor trip from 25% load	DH	75	8		83						
2.4	Turbine trip with recovery	25	90									
2.5	Turbine trip with plant shutdown	DH	30			30						
2.6	Loss of condenser vacuum	DH		17		17						
2.7	Loss of offsite power with turbine trip	DH		4		4						
2.8	Grid load reject with hold of house load and recovery	25	40									
2.9a	Feedwater flow decrease	DH		8		8						
2.9b	Circulator overspeed	DH		9		9						
2.9c	Excess feedwater heating	DH	9			9						
2.10a	Feedwater flow increase	DH		8		8						
2.10b	Circulator underspeed	DH		9		9						
2.10c	Loss of feedwater heating	DH	9			9						
2.11	Accidental control rod group or RSCM insertion	DH	6			6	3		6			
2.12	Small primary coolant leak (up to 1 in.^{2})	DH	7			7		7				7
2.13	Small steam generator tube leak (up to 0.1 lbm/s)	DH		12		12			12		12	
2.14	Small earthquake (≤0.06 g)	25 to 100, DH	4 1			 1						

TABLE 5-1 (Continued)

		Design Number of Occurrences Per Module										
		Reactor Heat Removal				Reactor Trip		Primary Coolant Radionuclide			Chemical Attack	
		Power Level (%)	HTS	SCS	RCCS	Control Rods	RSCM	Release	Pumpdown	Pumpup	Moisture	Air
2.15	Shutdown cooling heat exchanger tube leak	DH	3			3			3			
2.16	Loss of shutdown cooling heat exchanger cooling water during standby	DH	9			9						
2.17	Pressurized conduction cool-down with ~6 h precooling on SCS	DH		2	2	2						
2.18a	Accidental control rod group continuous withdrawal	DH	3			3						
2.18b	Accidental control rod group intermittent withdrawal	DH	1			1						
2.19	Moderate steam generator tube leak (up to 12.5 lbm/s)	DH		2		2			2		2	
2.20	Small feedwater or main steam pipe break	DH		2		2						
3.	<u>Off-Normal/Long Outage Events</u>											
3.1	Moderate earthquake (0.18 g) with main loop trip	DH		2		2						
3.2	Small steam generator tube leak with moisture monitor failure	DH		2		2	2		2		2	
3.3a	Pressurized conduction cooldown	DH			2	2						
3.3b	Pressurized conduction cool-down with postcooling on SCS after ~36 h	DH		2	2	2						

TABLE 5-1 (Continued)

		Design Number of Occurrences Per Module										
		Reactor Heat Removal				Reactor Trip		Primary Coolant Radionuclide			Chemical Attack	
		Power Level (%)	HTS	SCS	RCCS	Control Rods	RSCM	Release	Pumpdown	Pumpup	Moisture	Air
3.4	Main loop trip with failure of helium shutoff valve to close	DH		1		1			1			
3.5	Plant feedwater or main steam pipe break	DH		2		2						
3.6	Moderate primary coolant leak (up to 13 in.^{2}) with cooling on HTS or SCS	DH	1	1		2		2				2
3.7	Depressurized conduction cooldown	DH			1	1		1				1
4.	<u>Off-Normal/Safety Design Basis Events</u>											
4.1	Pressurized conduction cool-down with large (0.35 g) earthquake	DH			1	1						
4.2	Depressurized conduction cooldown with moderate moisture ingress	DH			1	1	1	1			1	1
4.3	Pressurized conduction cool-down with control rod withdrawal	DH			1	1						
4.4	Pressurized conduction cool-down with failure of control rod trip	DH			1		1					
	Totals		475[a]	318	11	679[b]	7	11	75	85	17	11

[a]Decay heat (DH) removal occurrences only.

[b]Number of module shutdowns excluding safety design events is 676 to reconcile with the number of startups.

completed when the module is producing rated steam condition, at 25% of rated load and is on automatic control, with the turbine online.

Depending on the initial condition, some or all of the following major activities may be required during this event:

1. Clean up/polish feedwater to achieve required chemical purity.

2. Begin primary coolant pressurization and vessel system warmup using preheated feedwater.

3. Achieve reactor criticality and begin rise to 25% of rated power.

4. Warm up primary/secondary coolant and vessel systems and complete primary coolant pressurization.

5. Warm up turbine on turning gear.

6. Establish boiling in steam generator.

7. Establish primary/secondary coolant conditions consistent for producing rated steam condition of 25% of rated load.

8. Synchronize turbine-generator and establish 25% of rated load on automatic control.

5.1.2. Module Startup With Full Helium Inventory

This event begins with the reactor in a cold, shutdown condition with full primary coolant inventory. The number of pressurized startups encompasses the sum of pressurized shutdowns due to scheduled outages (Section 5.1.4) and unscheduled outages (off-normal events, Table 5-1) which result in shutdown with full helium inventory. There are a total of 591 startups with full helium inventory for each module, of which 86 are startups following scheduled shutdowns.

The extreme of this initial condition is the startup of all modules with the turbine initially in a cold, offline state. The startup is completed when the module is producing rated steam condition, at 25% of rated load and is on automatic control, with the turbine online.

Depending on the initial condition, some or all of the following major activities may be required during this event:

1. Clean up/polish feedwater to achieve required chemical purity.

2. Achieve reactor criticality and begin rise to 25% of rated power.

3. Warm up primary/secondary coolant and vessel systems.

4. Warm up turbine on turning gear.

5. Establish boiling in steam generator.

6. Establish primary/secondary coolant conditions consistent for producing rated steam condition of 25% of rated load.

7. Synchronize turbine-generator and establish 25% of rated load on automatic control.

5.1.3. Scheduled Module Shutdown to Depressurized Conditions

This event begins with the reactor at 25% of rated load condition and is completed when the reactor reaches the cold shutdown condition with the primary coolant depressurized. The reference shutdown event is the shutdown of all modules to neutron flux of less than 0.001% of rated power, the steam generators flooded out, and the primary coolant depressurized to slightly subatmospheric pressure. There are a total of 34 scheduled shutdowns which involve pumpdown of the primary coolant to storage.

The following major functions must be accomplished during this event:

1. Shutdown and secure turbine-generators (if all modules are shut down).
2. Reduce power level and shutdown reactor.
3. Depressurize primary coolant.
4. Control cooldown and flood out of steam generator.
5. Maintain decay heat removal.

5.1.4. Scheduled Module Shutdown With Full Helium Inventory

This event begins with the reactor at 25% of rated load condition and is completed when the reactor reaches the cold shutdown condition with full helium inventory. The reference event is the shutdown of all modules to neutron flux of less than 0.001% of rated power with the steam generators flooded out.

The following major functions must be accomplished during this event:

1. Shutdown and secure turbine-generators (if all modules are shut down).
2. Reduce power level and shutdown reactor.
3. Control cooldown and flood out of steam generator.
4. Maintain decay heat removal.

5.1.5. Rapid Load Increase (5% Per Minute) (25% to 100%)

This event encompasses all rapid ramp load increases from any load between 25% and 100% of rated load at a rate between 0.5% and 5% of rated load per minute. For design purposes, the maximum rate (5%/min) and maximum magnitude (25% to 100%) ramp load increase event is considered. Normal control system automatic action will cause an increase in feedwater flow, helium flow, reactor power, and turbine load.

5.1.6. Normal Load Increase (0.5% Per Minute) (25% to 100%)

This event encompasses all normal ramp load increases from any load between 25% and 100% of rated load at a rate up to and including 0.5% of rated load per minute. The load increase is accomplished the same as the rapid load increase, except the maximum rate is less than or equal to 0.5% of rated load per minute.

5.1.7. Rapid Load Decrease (5% Per Minute) (100% to 25%)

This event includes all rapid ramp load decreases from any load between 100% and 25% of rated load at a rate between 0.5% and 5% of rated load per minute. This event is the inverse of the rapid load increase. For design purposes, the maximum rate (5%/min) and maximum magnitude (100% to 25%) ramp load decrease event is considered.

5.1.8. Normal Load Decrease (0.5% Per Minute) (100% to 25%)

This event encompasses all normal ramp load decreases from any load between 100% and 25% of rated load at a rate up to and including 0.5% of rated load per minute. This event is the inverse of the normal load increase. For design purposes, the maximum rate (0.5%/min) and maximum magnitude (100% to 25%) ramp load decrease event is considered.

5.1.9. Step Load Increase (+10%)

Step load changes may be imposed on the plant largely due to grid upsets. The plant shall be designed to accommodate a step increase in electric load of up to and including +10% of rated load. This increase can occur at any operating load between 25% and 100%, but the load after the step change does not exceed 100% of rated load. Normal automatic control system actions will cause an increase in feedwater flow, helium flow, and reactor power in response to the step load increase.

5.1.10. Step Load Decrease (-10%)

The plant shall accommodate a step decrease in electric load of up to and including -10% of rated load. This decrease can occur between 25% and 100% of rated load, but the load after the step change is not less than 25% of rated load. Normal automatic control system actions will cause a decrease in feedwater flow, helium flow, and reactor power in response to the step load decrease.

5.1.11. Grid Frequency Variations

This event encompasses the range of small amplitude (up to ±1% of rated) load changes which are induced by the slight frequency variations of the grid during steady-state operation. The frequency variation is caused by the ever changing load demand (consumption by users) on the electrical grid. This is a normal power production operating condition for which a high (1×10^6) number of cycles is specified.

5.2. OFF-NORMAL/FORCED OUTAGE EVENTS

These off-normal events result in a reduction or complete loss of power production and are anticipated to occur within the plant lifetime. Recovery to full power production is expected following repair of the initiator of event.

5.2.1. Main Loop Trip

The loss of HTS (main loop trip) may be due to either total loss of feedwater flow to the module or due to a circulator trip.

Upon circulator trip, or loss of feedwater flow, the affected module is tripped; i.e., reactor trip, circulator trip, feedwater and main steam isolation valve closure and helium shutoff valve closure. Turbine load is reduced to a level which the remaining modules can support. If any load pickup margin is available in the remaining modules, then their output will be increased.

Following main loop trip, the shutdown cooling system is started in order to provide decay heat removal.

About 18% (15 out of 85) of the main loop trip events involve pumpdown of the primary coolant to enable repair of internal failures in the main circulator.

5.2.2. Reactor Trip From 100% Load

The plant shall be designed to accommodate a reactor trip in one or more of the modules while other reactor modules remain in operation. This transient imposes a rapid cooldown of reactor and HTS components in the tripped module(s). The tripped module is decoupled from the turbine by closing the module main steam isolation valve. Steam from the tripped module is directed to the module startup flash tank and then to the condenser. At the time of reactor trip, the turbine-generator load is reduced to a level which the remaining modules can support. If any load pickup margin is available in the remaining modules, then their output will be increased.

The NI control subsystem provides post-trip control actions to mitigate this transient cooldown. Accordingly, feedwater flow in the tripped module is ramped down to 15% of nominal. The main steam temperature setpoint for the affected module is ramped linearly down to saturation temperature, and the circulator speed control of main steam temperature remains active until saturation is reached. Thus, the steam temperature is brought smoothly to saturation. Decay heat removal continues with forced circulation using the main loop. About 10% of the reactor trip occurrences result in a transition to SCS cooling to facilitate repair of initiator of event.

5.2.3. Reactor Trip From 25% Load

This event is a reactor trip from 25% load. The transient effect on various plant components may be sensitive to initial power level. The low power trip is more severe for some components, and the full power trip is more severe for other components. Therefore, two sets of reactor trip design number of occurrences are shown in Table 4-1. The second set (see footnote c in Table 4-1) is applicable for component design when reactor trip from 25% load is worse. Similar to the trip from 100% load, the breakdown is 90% main loop and 10% SCS decay heat removal occurrences.

5.2.4. Turbine Trip With Recovery

Although most of the turbine trips involve only one turbine, and therefore have a smaller load reduction impact than both turbines trip, the design requirements imposed on system components are different depending upon the number of turbines tripped. For the case of both turbines trip, the plant control system commands each reactor module to run back to a standby condition at 25% of rated feedwater flow and rated steam conditions. Turbine bypass valves and steam vents control the pressure excursion and provide an alternate flow path to the condenser following turbine trip. This standby condition is maintained for a time until the turbine can be restarted.

For the case of a single turbine trip, the plant steam load is reduced to a level commensurate with the output capability of the remaining turbine. Coordination of the main steam pressure control and the turbine bypass valve control is required in order to effect a smooth transition to single turbine operation.

Turbine recovery/reloading is generally expected to occur within about 2 h. The plant shall be capable of recovery to full load at rates up to 5% of rated load per minute similar to the rapid load increase event described in Section 5.1.5. In situations where the turbine cannot be restarted in a reasonable time, the modules are shutdown in an orderly manner.

5.2.5. Turbine Trip With Plant Shutdown

As indicated in Section 5.2.4, turbine recovery/reloading is generally expected to occur within about 2 h following a turbine trip. If recovery cannot be achieved within this time frame, the modules are shutdown in an orderly manner similar to the scheduled shutdown described in Section 5.1.4.

5.2.6. Loss of Condenser Vacuum

The loss of both condenser vacuum will result in the total loss of feedwater flow after a few minutes due to lack of condensation flows to the deaerator tank. The plant response is similar to the turbine trip until the deaerator tank level reaches the minimum. Thereafter, the plant will respond similar to the total loss of feedwater (from 25% power) with main loop trip (Section 5.2.1), and will result in the cooldown of the modules on the SCS.

5.2.7. Loss of Offsite Power With Turbine Trip

The loss of offsite power will result in the loss of all main loops. During this event, all reactors will trip due to high power-to-flow ratio. The reactor decay heat removal is accomplished by the SCS, utilizing the standby power, until offsite power is restored. At that time the main loops may be restarted to take over the reactor decay heat removal.

5.2.8. Grid Load Reject With Hold of House Load and Recovery

The event is initiated by either loss of offsite power and/or by a trip of the main transformer breaker. The loss of the grid load results in a turbine speed increase. This, in turn, causes the main turbine inlet valve to close rapidly. The turbine load is reduced to house load, while the module output is reduced to 25%, with most of the steam bypassed to the condenser. As soon as the grid is reestablished, the load on the generator can be increased to the desired level at rates up to 5% of rated load per minute similar to the rapid load increase event described in Section 5.1.5.

5.2.9. Main Loop Undercooling

This transient is essentially due to the same failures as the main loop overcooling event (Section 5.2.10) except that, in this case the circulator overspeeds or the feedwater flow decreases or the feedwater heating is excessive. For significant circulator overspeed or feedwater flow decrease without commensurate circulator speed decrease, the IPS initiates reactor trip and main loop trip. Subsequent decay heat removal is provided by the SCS.

For limited amounts of undercooling caused by small mismatches in circulator speed-to-feedwater flow or feedwater heating rate, the NI control is capable of maintaining steam delivery at rated conditions without incurring a protective system trip. An orderly plant/module shutdown is initiated by the plant operator if the undercooling causes plant parameters to approach technical specification limits.

5.2.10. Main Loop Overcooling

This transient results from either a failure of the circulator speed control system (which causes the circulator to underspeed) or a failure of the feedwater flow control system (which causes the feedwater flow to increase). Another cause for an overcooling is the loss of deaerator heating.

For a significant circulator underspeed, the circulator speed-to-feedwater flow mismatch causes an IPS reactor trip and a main loop trip. A significant feedwater flow increase without commensurate circulator speed increase will also trip the main loop and the reactor.

Subsequent decay heat removal is provided by forced circulation flow on the SCS.

For limited amounts of overcooling such as loss of deaerator heating or small mismatches in circulator speed-to-feedwater flow the NI control is capable of maintaining steam delivery at rated conditions without incurring a protective system trip. An orderly plant/module shutdown is initiated by the plant operator if the overcooling causes plant parameters to approach technical specification limits.

5.2.11. Accidental Control Rod Group Insertion

The plant design shall accommodate the transient resulting from the inadvertent insertion of a control rod group or due to inadvertent release of reserve shutdown control material (RSCM). An inadvertent control rod insertion can result from operator error, neutron flux controller malfunction, loss of rod holding power, or rod cable break.

As the rod or reserve shutdown material inserts, the reactor power decreases which causes a decrease in primary coolant and steam temperatures. When the steam temperature drops to the low setpoint, a reactor trip and main steam isolation are initiated by the IPS. The sequence of events is similar to the reactor trip (Section 5.2.2). System cooldown is more rapid due to the initial reactor overcooling condition for this transient.

Three occurrences each of RSCM and control rod insertion are specified. Primary coolant pumpdown to permit cleanup/repair is specified for these events.

5.2.12. Small Primary Coolant Leak

A small primary coolant leak is defined as a breach in the primary coolant pressure boundary, resulting in an opening with equivalent area of 1.0 $in.^2$ or less. For design purposes, the maximum opening of 1.0 $in.^2$ is used. The opening results in a slow loss of helium inventory from the primary coolant system to the reactor confinement and a resultant decrease in the primary coolant pressure and density. Main loop cooling continues, but the reduced density results in reduced helium flow. The NI control causes an increase in reactor power and circulator speed in order to maintain the module outlet steam temperature. Eventually, a reactor trip is initiated by the RPS on low primary coolant pressure. Decay heat removal continues to be supplied by the HTS. The depressurization of the primary coolant to ambient results in release of circulating activity to the reactor confinement and a limited amount of air ingress into the primary circuit.

5.2.13. Small Steam Generator Tube Leak

Plant and reactor module components shall be designed to accommodate the moisture ingress and resulting transient initiated by a small steam generator tube leak (up to 0.1 lbm/s) into the primary coolant.

Upon detection of high moisture level, the IPS initiates reactor trip, main loop trip, and steam generator dump. Decay heat removal is provided by forced circulation on the SCS. Total moisture ingress is about 35 lbm.

Pumpdown of the primary coolant is performed to permit repair of the steam generator.

5.2.14. Small Earthquake

An earthquake with a ground acceleration of up to 0.06 g does not adversely affect operation of the plant. The seismic loads corresponding to this event do not result in damage which would normally cause a departure from power production. Should module trips occur during this event (e.g., trip due to operator action), the plant is designed to be restarted using normal plant start up procedure once the seismic event has ended. One out of five occurrences involves operator-initiated plant shutdown with decay heat removal by the main loops.

5.2.15. Shutdown Cooling Heat Exchanger Tube Leak

This event involves a tube leak in the shutdown cooling heat exchanger (SCHE) tube bundle. If such a leak occurs, the response of the system depends on the primary coolant pressure. When the primary coolant pressure is greater than the SCS secondary system pressure (as during normal operation), then primary coolant will leak into the secondary side of the SCS. If the primary coolant pressure is less than the SCS secondary system pressure (e.g., when the reactor is depressurized for maintenance), then water from the SCS secondary side will enter the primary coolant.

In either case, once the leak has been detected, the secondary side of the SCHE is isolated. If the SCHE was being used for decay heat removal, the main loop will be restarted, assuming that it is available.

If the main loop is unavailable, then decay heat will be transmitted through the reactor vessel to the RCCS. If the reactor was operating at the time of SCHE isolation, it is shut down in order to allow primary coolant depressurization and repair of the SCHE tube bundle.

5.2.16. Loss of Shutdown Cooling Heat Exchanger Cooling Water Flow During Standby

The event assumes that during normal power production, the SCHE cooling water supply is interrupted. This results in an increase in the SCHE tube temperatures due to the continued small bypass flow of primary coolant from the reactor vessel bottom plenum through the SCHE to the core outlet plenum. Repair and restoration of the SCHE cooling water supply is accomplished while the reactor module remains on line. Should shutdown of the module become necessary to allow repair access, an orderly shutdown would be used.

5.2.17. Pressurized Conduction Cooldown With Prior Cooling on SCS

The initiating event is similar to the main loop trip (Section 5.2.1) except the SCS fails to continue to operate after a period of time. The subsequent decay heat removal is accomplished by conduction and radiation to the RCCS. For pressurized conduction cooldowns on the RCCS, prior forced convection cooling using the SCS will result in a milder transient than for the case with no prior SCS cooling.

In this event, after the main loop is lost, the SCS provides 6 h of cooldown before the SCS is also lost. Six hours of SCS cooling is the minimum prior forced circulation cooling required to ensure vessel temperature remains below Service Level B limits during pressurized conduction cooldown on the RCCS. The resulting pressurized conduction cooldown is a subset of the pressurized conduction cooldown given in Section 5.3.3a, where there is no prior cooling using the SCS. As expected, this particular transient with prior cooling results in lower temperatures than for the conduction cooldown with no prior cooling.

System transient response for the initial 6 h of SCS cooling is similar to that described in Section 5.2.1.

5.2.18. Accidental Control Rod Group Withdrawal

This event involves the improper withdrawal of a control rod group. Reactor power increases significantly until a reactor trip is initiated by the RPS. Two separate cases are considered.

Event 5.2.18a considers the continuous withdrawal of a maximum worth control rod group at the design control rod movement speed. Such an event may result from failure of the reactor power flux controller. The core power level rises rapidly until a reactor trip is initiated due to high power-to-flow ratio. Decay heat removal is provided by the main loop similar to the reactor trip event (Section 5.2.2).

Event 5.2.18b considers the gradual, discontinuous withdrawal of a control rod group. Such an event might result from certain types of reactor control system failures. Both core power and core outlet temperature rise gradually. Temperatures significantly higher than those experienced during normal operation occur before the high steam generator inlet helium temperature trip setpoint is reached. The high power-to-flow ratio trip is ineffective for this event. The subsequent post-trip reactor cooldown using the main loop is more limiting than for a normal reactor trip because the initial temperatures are significantly higher.

5.2.19. Moderate Steam Generator Tube Leak

This moisture ingress event encompasses the range of steam generator (SG) tube leaks between the small (0.1 lbm/s) SG tube leak (Section 5.2.13) up to a double offset tube rupture (12.5 lbm/s). Upon detection of high moisture in the primary coolant, the IPS initiates reactor trip, main loop trip, and SG dump. The SCS provides decay heat removal. Total moisture ingress into the primary coolant is about 600 lbm.

Pumpdown of the primary coolant is performed to permit repair of the SG.

5.2.20. Small Feedwater or Main Steam Pipe Break

A small feedwater or main steam pipe break may be a partial rupture (e.g., through-wall crack of a principal pipe) or a double offset rupture of a small diameter line which results in release of secondary coolant into the reactor confinement or turbine building. The IPS detects the leakage and initiates main loop trip, reactor trip, and startup of the SCS. Reactor decay heat removal using the SCS is similar to that of the main loop trip described in Section 5.2.1.

5.3. OFF-NORMAL/LONG OUTAGE EVENTS

These off-normal events have a low probability of occurrence (one or two design occurrences per event in the plant lifetime) and may result in plant or module outages of greater than six months prior to recovery to full power operation. In addition to repair of initiator of event, inspection/repair of concomitant damage may be required. As indicated in Table 3-1, this category of events is driven by the top-level requirement which restricts the forced outage allocation for long outage events.

5.3.1. Moderate Earthquake With Main Loop Trip

A moderate earthquake is defined as a seismic event with ground acceleration between 0.06 and 0.18 g. In this event sequence the maximum earthquake acceleration is considered and results in a main loop trip due to either loss of feedwater flow or circulator trip. One or more modules may be affected and subsequent pressurized decay heat removal is provided by the SCS similar to the main loop trip event (Section 5.2.1).

5.3.2. Small Steam Generator Tube Leak With Moisture Monitor Failure

This event is similar to the small SG tube leak described in Section 5.2.13 except the moisture monitors fail to detect high primary coolant moisture level. Due to the small leak rate (0.1 lbm/s), the primary coolant pressure increases gradually. If plant operator action is ignored in the event detection process, the RPS will trip the reactor and main loop within about 5 h due to high primary coolant pressure. Control rod and RSCM trips are initiated by the high pressure trip signal. The SCS is started by the IPS to provide decay heat removal. The operator dumps the SG within 20 min after main loop trip. The integrated moisture ingress (about 1850 lbm) is much greater than that of SG tube leak events where the moisture monitors work properly (Sections 5.2.13 and 5.2.19).

Pumpdown of the primary coolant is performed to permit repair of the SG.

5.3.3a. Pressurized Conduction Cooldown

This event is a main loop trip and the SCS either fails to start or fails to continue to operate after 6 h. Reactor decay heat removal is provided by conduction and radiation to the RCCS.

For design purposes, the limiting condition of no prior cooling by the SCS provides the maximum temperature transient to the reactor and vessel systems.

During the pressurized conduction cooldown, decay heat removal is accomplished by conduction and radiation to the RCCS. Most of the decay heat is removed in the radial direction to the RCCS. After about 50 h, the overall heat removal rate exceeds heat generation at which time system temperatures begin to level off and decrease.

5.3.3b. Pressurized Conduction Cooldown With Restart of SCS

The event is a pressurized conduction cooldown on the RCCS similar to event 5.3.3a except the SCS is repaired/restarted at the peak fuel temperature condition (~36 h). The event creates the largest heat load on the SCS.

5.3.4. Main Loop Trip With Helium Shutoff Valve Failure

This event begins with a main loop trip identical to that described in Section 5.2.1. However, as the helium flow approaches zero, the main circulator helium shutoff valve fails in the open position. Followup action by the plant operator to close the valve using the jet assist is ignored. When the SCS is started to provide decay heat removal, the fraction of helium which backflows through the steam generator is much larger than normal, due to the stuck valve. Coolant flow provided to the reactor core is diminished, and core outlet temperatures are higher than in the normal SCS cooldown following main loop trip. A higher than normal rate of cooldown of the steam generator is also expected during this event.

Pumpdown of the primary coolant is performed to permit repair of the failed shutoff valve.

5.3.5. Plant Feedwater or Main Steam Pipe Rupture

A main steam pipe of feedwater pipe double offset rupture is assumed for this event and may affect one or more modules depending upon rupture location. Analysis of this event is required for the environmental qualification of system components. Rupture within the reactor confinement and rupture within the turbine building are considered independently as appropriate for the system being qualified. Following detection of the pipe rupture, the affected modules are tripped and decay heat removal is provided by the SCS.

5.3.6. Moderate Primary Coolant Leak

A moderate primary coolant leak is defined as a rupture in the primary coolant pressure boundary with equivalent area between 1.0 and 13 in.^{2}. For design purposes, the maximum opening of 13 in.^{2} is used. the moderate leak area results in reactor trip by the RPS on low primary coolant pressure within 20 s of the leak initiation. Post-trip decay heat removal is provided by the main loop or the SCS if the main loop becomes unavailable.

Primary coolant circulating activity is released to the reactor confinement and a limited amount of air ingresses into the primary circuit.

5.3.7. Depressurized Conduction Cooldown

Primary coolant leak events described in Sections 5.2.12 and 5.3.6 involve continued forced circulation cooling of the reactor by either the main loop or the SCS if the main loop becomes unavailable. In the unlikely event that both the main loop and SCS are unavailable, the decay heat is removed by conduction and radiation to the RCCS.

For this event, primary coolant leak sizes up to 13 in.^{2} are considered. The RPS trips the reactor due to low primary coolant pressure. The main loop becomes unavailable and the SCS fails to start following loss of the main loop. The subsequent depressurized conduction cooldown results in higher reactor and vessel temperatures than for a pressurized conduction cooldown (Section 5.3.3a) due to a lack of significant natural circulation within the core when the primary coolant becomes depressurized. System cooldown is slower than for a pressurized conduction cooldown. Primary coolant circulating activity is released into the reactor confinement and a limited amount of air ingresses into the primary circuit.

5.4. OFF-NORMAL/SAFETY DESIGN BASIS EVENTS

This category of off-normal events is comprised of a subset of the LBEs for which the controlling requirement is to restrict radiological releases to levels within the guidelines of 10CFR100/PAGs using only safety-related SSCs to perform the required functions. Unlike the events of Section 5.3 which must also comply with 10CFR100/PAGs guidelines, the long outage requirement (and subsequent return to service) is not controlling the event selection in this category.

5.4.1. Pressurized Conduction Cooldown With Large Earthquake

This event considers the occurrence of a 0.35 g earthquake followed by main loop trip, reactor trip, and failure of the SCS to operate. Decay heat removal is provided by conduction and radiation to the RCCS. Except for the added seismic loadings the thermal and pressure transient is similar to that for pressurized conduction cooldown (Section 5.3.3a).

5.4.2. Depressurized Conduction Cooldown With Moderate Moisture Ingress

This event considers a moderate SG tube leak (12.5 lbm/s), reactor trip and main loop trip on high primary coolant pressure, failure of the SCS to operate, and failure of the primary coolant pressure relief valve to reclose. This event results in the greatest integrated moisture ingress and dose consequence because the mitigating actions of nonsafety-related SSCs and the plant operator were ignored. Failure of the primary coolant pressure relief valve to reclose results in a depressurized conduction cooldown similar to event 5.3.7.

5.4.3. Pressurized Conduction Cooldown With Failure of Control Rod Trip

This event considers main loop trip with failure of control rod trip and failure of the SCS to operate. Loss of the main loop results in decrease in reactor heat removal rate and decrease in reactor power due to the negative temperature coefficient of reactivity. Reactor shutdown is assured by a trip of the reserve shutdown control material due to high power-to-circulator speed ratio. Reactor decay heat removal is provided by conduction and radiation to the RCCS. In spite of failure of the control rod trip, the thermal/pressure transient for this event is similar to the pressurized conduction cooldown event (Section 5.3.3a).

5.4.4. Pressurized Conduction Cooldown With Control Rod Withdrawal

This event considers the spurious withdrawal of a control rod group, loss of main loop cooling, and failure of the SCS to operate. Similar to event 5.2.18a, the control rod withdrawal results in reactor trip due to high power-to-flow ratio. Loss of main loop cooling and failure of the SCS to operate results in reactor decay heat removal by conduction and radiation to the RCCS. In spite of the initial reactivity transient caused by control rod withdrawal, the thermal/pressure transient is similar to the pressurized conduction cooldown event (Section 5.3.3a).

6. REFERENCES

1. "Overall Plant Design Specification Modular High Temperature Gas-cooled Reactor," Document DOE-HTGR-86004, Rev. 7, August 1988.
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3. "Licensing Basis Events for the Standard MHTGR," Document DOE-HTGR-86034, Rev. 1, February 1987.
4. "Plant Scheduled Outage Report," Update 4 x 350 MW(t) Modular HTGR Plant," Document DOE-HTGR-86046, Rev. 0, August 1986.
5. "Forced Outage Assessment for the MHTGR," Document DOE-HTGR-86069, Rev. 1, July 1987.
6. "Modular High Temperature Gas-cooled Reactor Plant Design Basis Transient Analysis," Document DOE-HTGR-86121, Rev. 2, November 1988.
7. "Status of Investment Risk Studies," Document DOE-HTGR-88235, Rev. 0, November 1988.
8. American Society of Mechanical Engineers, Boiler and Pressure Vessel Code, Section III, 1986 Edition, through 1986 Addenda.