BWR: Severe Accident

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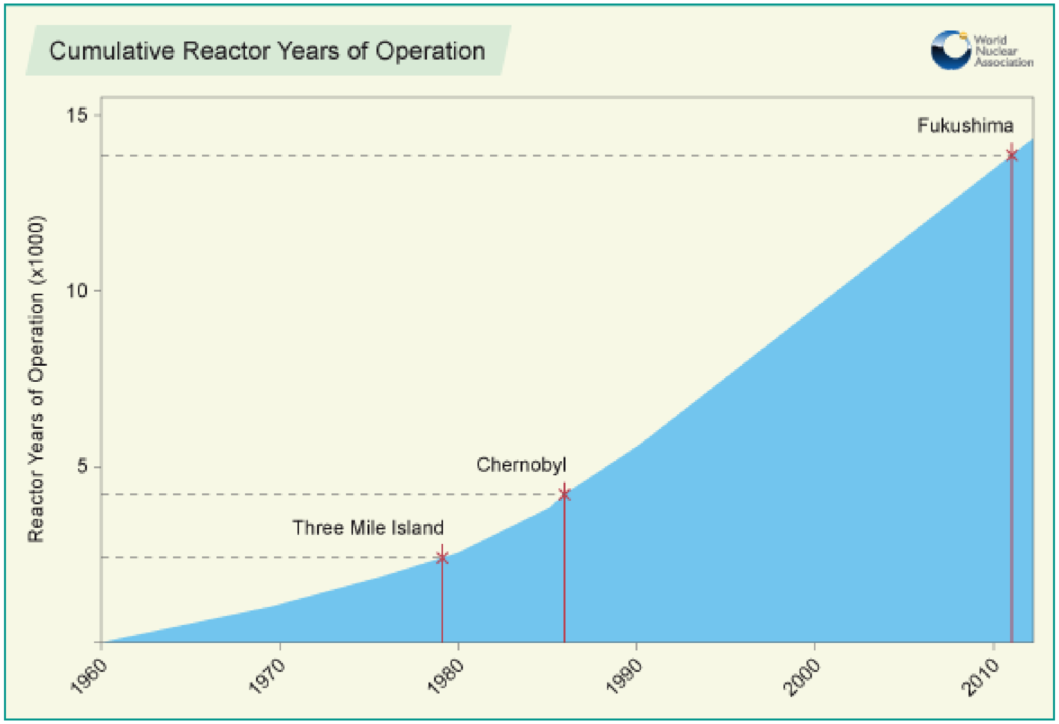
Nuclear Accidents

- Three Mile Island 2 (TMI-2)
  - United States
  - 28 March 1979
- Chernobyl 4
  - Former Soviet Union
  - 26 April 1986
- Fukushima
  - Japan
  - 11 March 2011

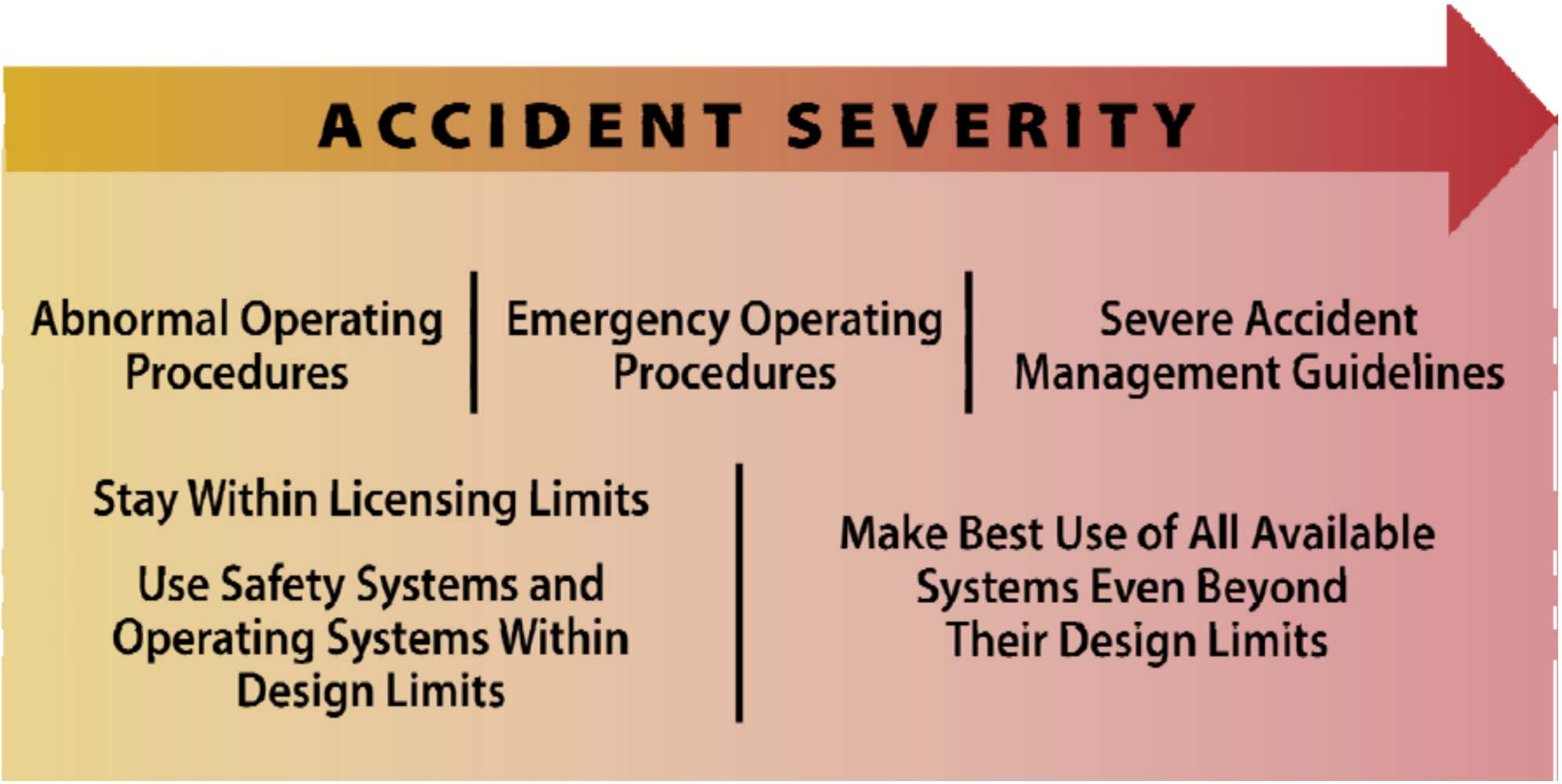


Source: http://www.world-nuclear.org/info/inf06.html. Last accessed: 10/8/2011

Accidents

- **Design-basis accident (DBA)**
  - A postulated accident that a nuclear facility must be designed and built to withstand without loss to the systems, structures, and components necessary to ensure public health and safety.
- **Severe accident (SA)**
  - A type of accident that may challenge safety systems at a level much higher than expected.

DBA → SA



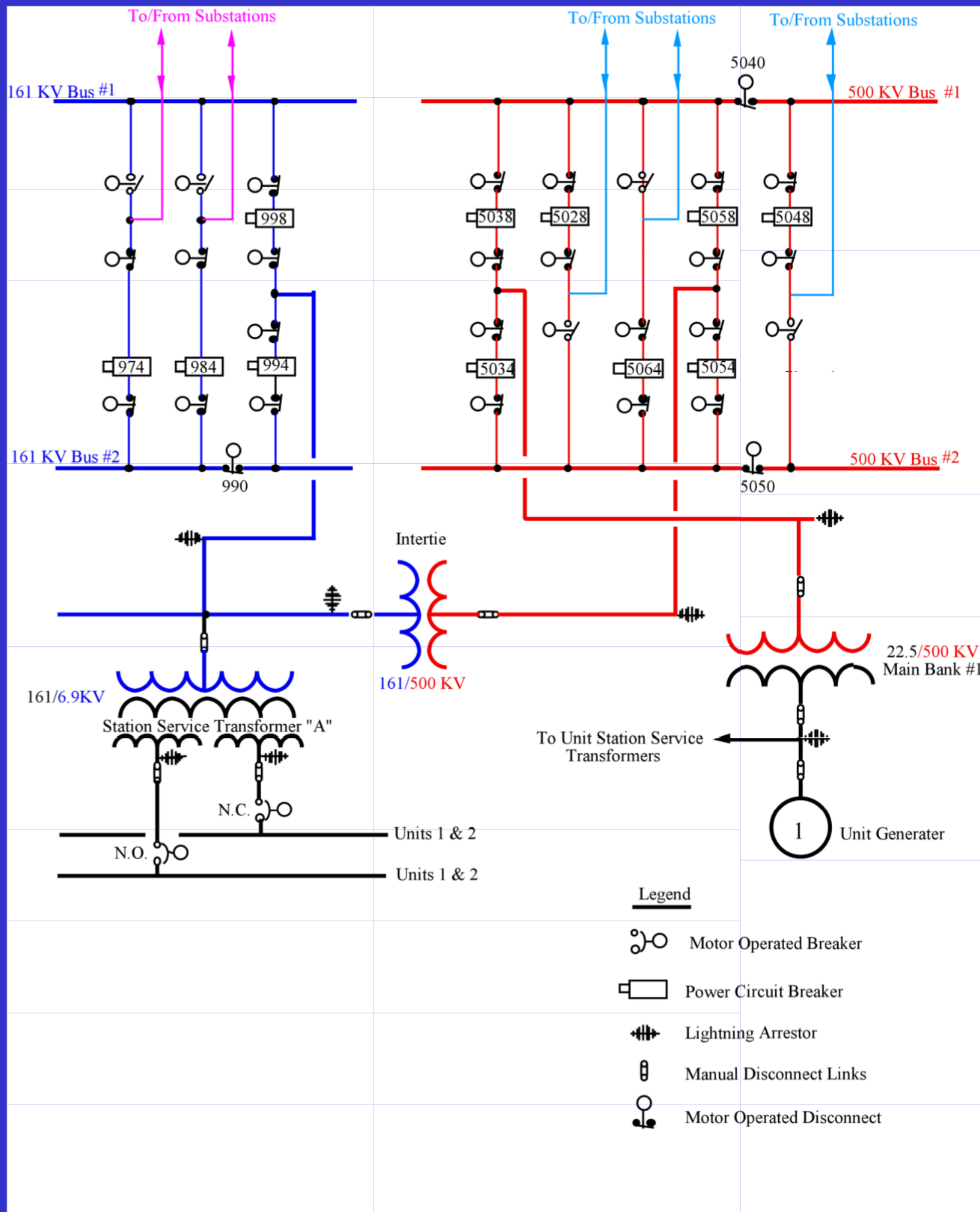
http://nuclear-discussion.uchicago.edu/files/design_basis_sienicki.pdf
Last accessed on 10/24/2011

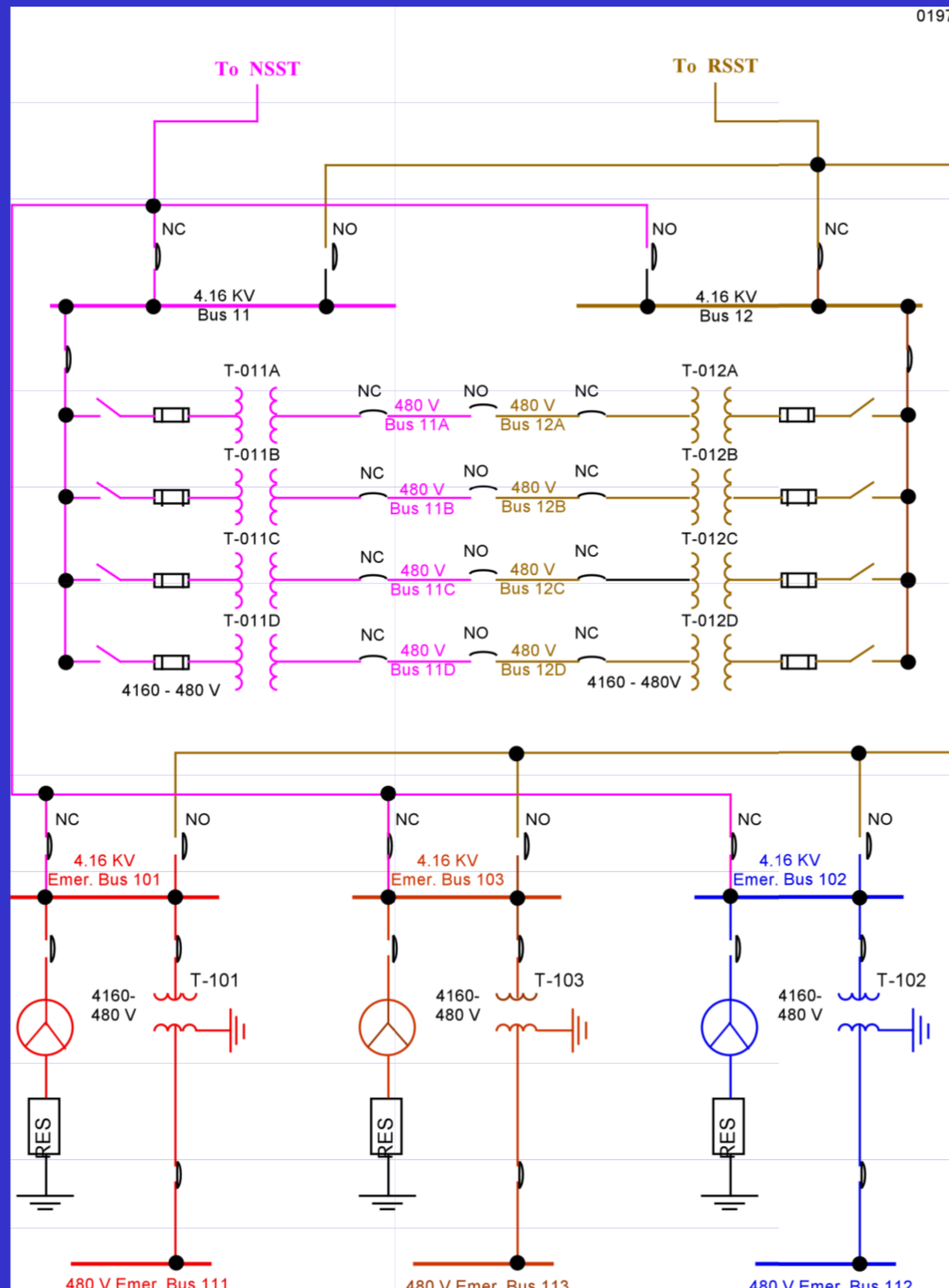
GDC in Appendix A of 10 CFR Part 50

Establishes the necessary design, fabrication, construction, testing and performance requirements for structures, systems, and components important to safety.

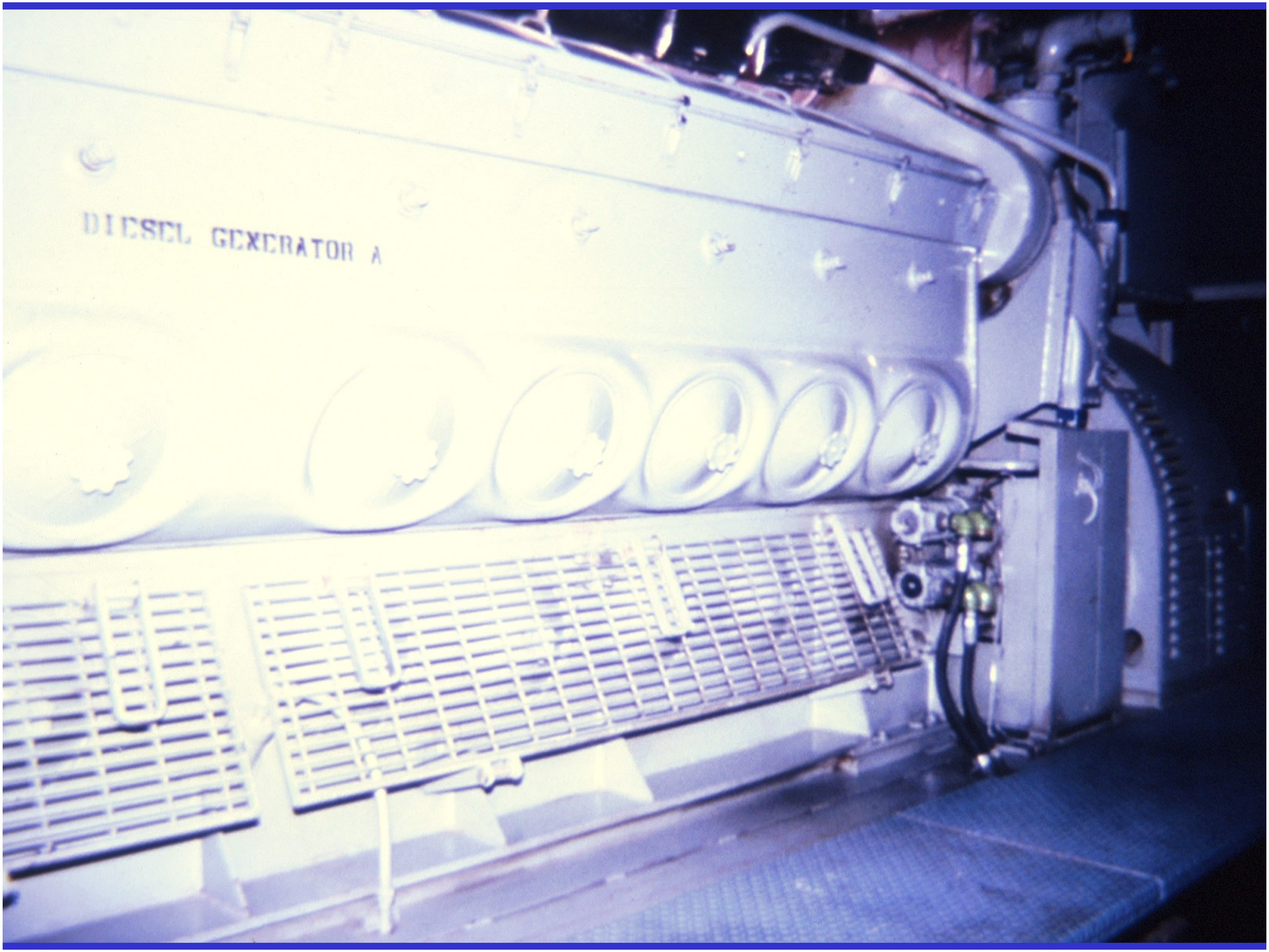
GDC 17 "Electric Power Systems"

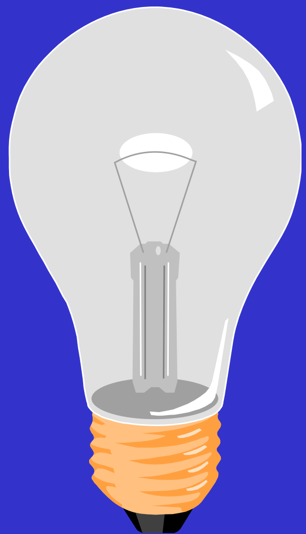
Requires that an onsite and offsite electric power system shall be provided to permit functioning of structures, systems and components important to safety


To/From Substations
To/From Substations
To/From Substations
161 KV Bus #1
5040
500 KV Bus #1
998
5038
5028
5058
5048
974
984
994
5034
5064
5054
161 KV Bus #2
990
500 KV Bus #2
5050
Intertie
161/500 KV
161/6.9KV
Station Service Transformer "A"
22.5/500 KV
Main Bank #1
To Unit Station Service
Transformers
N.C.
N.O.
Units 1 & 2
Units 1 & 2
1
Unit Generater
Legend
Motor Operated Breaker
Power Circuit Breaker
Lightning Arrestor
Manual Disconnect Links
Motor Operated Disconnect

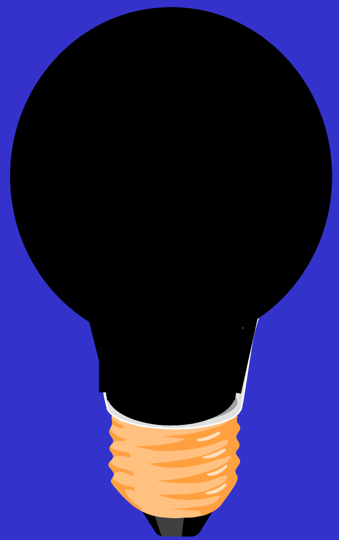


Normal Station Supply Transformer (NSST); Reserve Station Supply Transformer (RSST)


DIESEL GENERATOR A



Station Blackout



Learning Objectives

1. Define the term Station Blackout.
2. Describe the impact station blackout would have when combined with an accident.
3. Describe the primary method available to mitigate the consequences of a station blackout.
4. List the two major classifications Boiling Water Reactors have divided into for discussing station blackouts.

History

- 1975 Reactor Safety Study (WASH 1400)
- 1979 Commission Issued USI-A44 SBO
  - (Unresolved Safety Issue)
- 1988 NUREG 1032 Study of SBO
  - 50.63 SBO Rule
  - REG Guide 1.155
    - DG reliability programs with target values
- 1997 NUREG 1560 Individual Plant Exam

Station Blackout (SBO)

- "the complete loss of alternating current (ac) electric power to the essential and nonessential switchgear buses in a nuclear power plant (i.e., loss of offsite electric power system concurrent with turbine trip and unavailability of the onsite emergency ac power system). .." – 10 CFR 50.2
- SBO is one of the most challenging severe accidents for a boiling water reactor (BWR)
  - many of the safety systems required for Rx core cooling, decay heat removal, & containment heat removal depend on ac power
    - consequences of SBO could be severe

Plant Response

- Immediate consequences of SBO are not severe unless accompanied by an accident such as a LOCA
- If the condition continues for a prolonged period, the potential consequences to the plant & public health and safety can be serious
- Combination of core damage and containment over-pressurization could lead to significant offsite FP releases
- Any DBA in conjunction with a SBO reduces the time until core damage and release will occur

Plant Response (cont.)

- w/o systems designed to operate independently of ac power, the only way to mitigate the consequences of a SBO is to take steps to minimize the loss of reactor vessel inventory & quickly restore electrical power to replenish the lost inventory
  - This will ensure the ability to remove decay heat from the core and prevent fuel damage.
- Primary method for mitigating SBO
  - Initiate a controlled Rx cooldown (EOP – guidelines)

NRC's Interim Response

- Following license hearings for the operation of St. Lucie Unit 2 (southern FLA) interest in loss of all ac power intensified (mid-1980)
  - Concern:
    - Plant being located in an area subject to periodic severe weather conditions (hurricanes) and questionable grid stability
    - Probability of LOOP would be much higher than normal
- St. Lucie Unit 2, considered SBO as a DB event

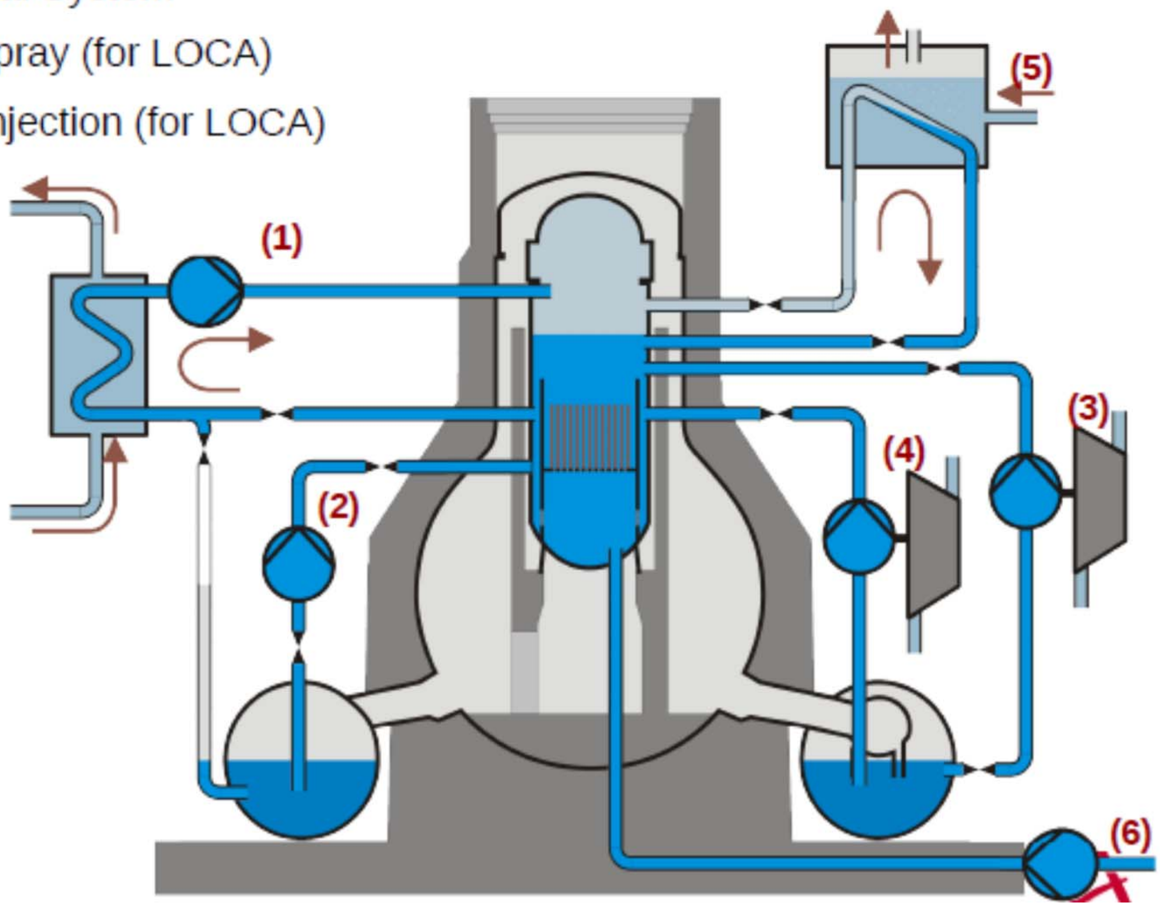
Regulation Changes

- Rule change based on USI A-44 results
  - SBO definition
  - All NPPs required to be capable of coping with SBO for plant specific period of time
    - Time depends on existing plant capabilities & factors identified as main contributors to core melt risk from LOOP
    - Factors: redundancy and reliability on onsite emergency ac power sources, LOOP frequency, and probable time needed to restore offsite power
  - Time duration of either 4 or 8 hours would be designated depending on the specific plant design and site related characteristics

BWR APPLICATION

- Emergency Core Cooling Systems

1) Residual Heat Removal System
2) Low-Pressure Core Spray (for LOCA)
3) High-Pressure Core Injection (for LOCA)
4) Reactor Core isolation cooling (Unit 2,3 [BWR4])
5) Isolation Condenser (Unit 1 [BWR3])
6) Borating System



BWR SBO assessment

- Functional classes:
  - BWRs that use isolation condensers for decay heat removal but do not have makeup capability independent of ac power (BWR-2 and 3 designs)
  - BWRs with a reactor core isolation cooling (RCIC) system and either a high pressure coolant injection (HPCI) system or high pressure core spray (HPCS) system with a dedicated diesel, either of which is adequate to remove decay heat from the core and control water inventory in the reactor vessel, independent of ac power (BWR-4, 5, and 6 designs)

Suppression Pool (SP)

- Decay heat (DH) removal by discharging to SP through relief valves; lost Rx coolant made up with RCIC and HPCI or HPCS
  - DH not discharged to the environment (stored in SP)
- Long term heat removal is by SP cooling mode of the RHR system
- Time Duration – adequate core cooling & cover determined in part by SP T_{max}

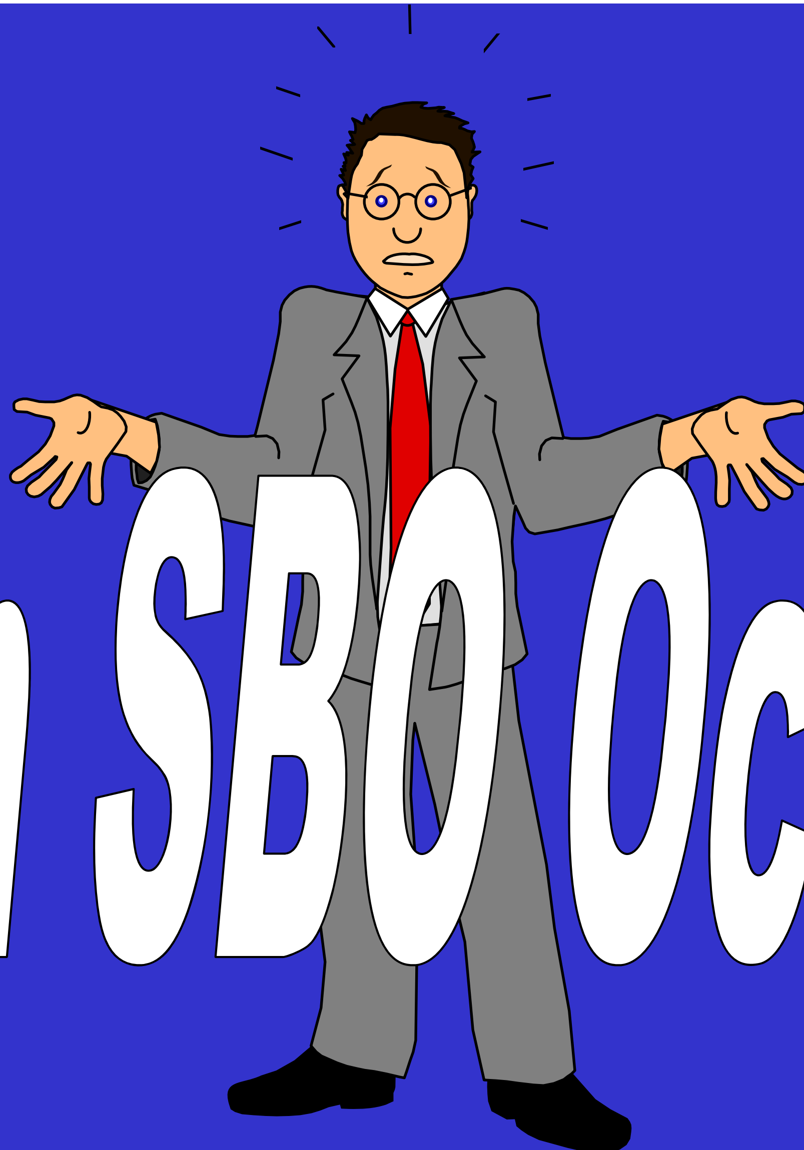
DC Power Systems

- All LWR designs remove decay heat for some period of time
  - Time depends on the capabilities and availability of support systems (e.g., sources of makeup water, compressed air, and dc power supplies)
  - During SBO, unless special emergency systems are provided, the battery charging capability is lost
    - dc power systems (most important support system) are generally designed to provide specific load carrying capacity in the event of a DBA w/battery charging unavailable
    - most dc power systems in operation today have the capacity to last longer during a SBO than during a DBA

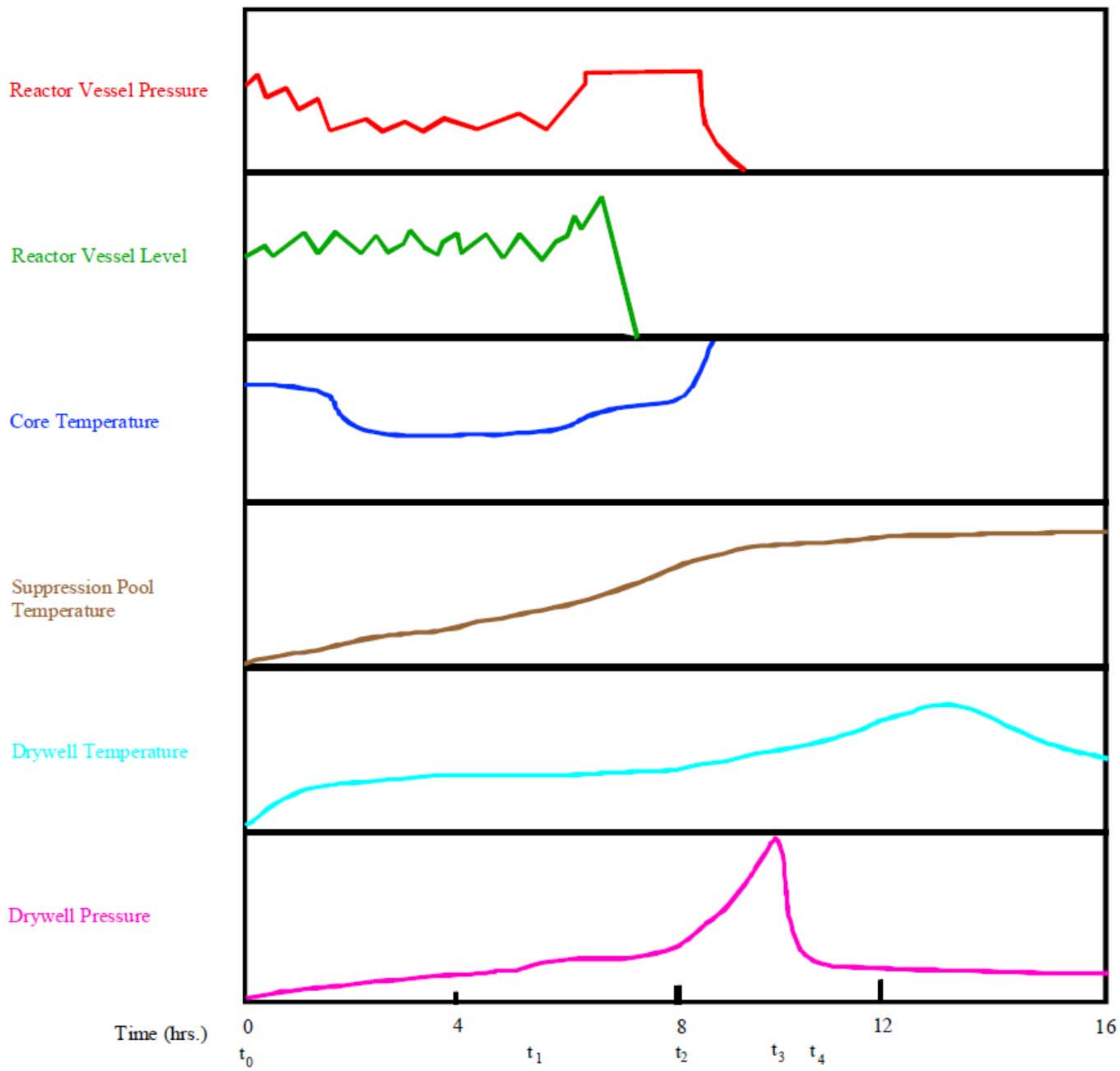
Actions

- actions necessary to operate systems during a SBO would not be routine.
- operator has less information & operational flexibility than is normally available during most other transients requiring a reactor cooldown

Can SBO Occur?



ACCIDENT SEQUENCE



Time	Sequence of Events
t_0	Loss of all AC power
t_1	DC power (batteries) depleted
t_2	Core uncovery begins
t_3	Reactor vessel penetration
t_4	Containment failure

Containment

- Mark I & II offer some pressure suppression capability during SBO
  - after core melt, they may fail by one of two modes:
    - mechanical or electrical fixtures in penetrations fail
    - overpressure & subsequent containment rupture
- Mark III
  - Inerted (N_2); no H_2 burn
  - Failure: overpressurization

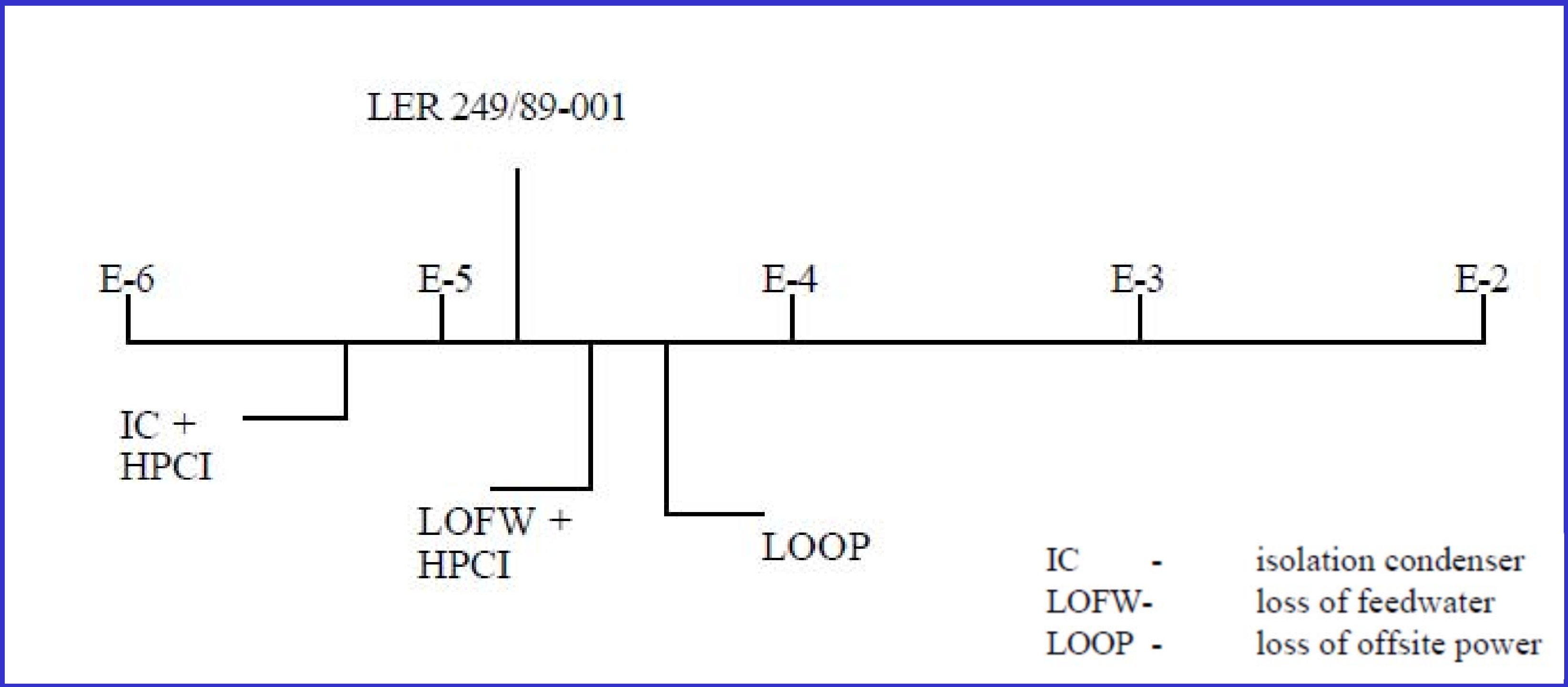
Summary

- SBO is one of the largest contributors to core damage frequency at BWRs
- All LWR operators are prepared to deal with the effects of a loss of and restoration of ac power
- Extensive studies are ongoing to better understand & cope with the effects if a total loss of ac power

Plant staff have typically considered the low probability of numerous failures occurring at the same time as an incredible situation.

PRA INSIGHTS

Dresden Unit 3 (3/25/89)



- for over 1 hour, lost LPCI, instrument air (IA), one division of the containment cooling water system

Brunswick 2 (6/17/89)

- LOOP
  - control room received a ground fault annunciator alarm on the Standby Auxiliary Transformer (SAT)
  - per procedure, recirculation pumps were being powered from the SAT
    - minimize pump seal failure caused by frequent tripping of the recirculation pumps
  - technician shorted out the transformer
    - caused a loss of SAT & eventually, dual recirculation pump trip

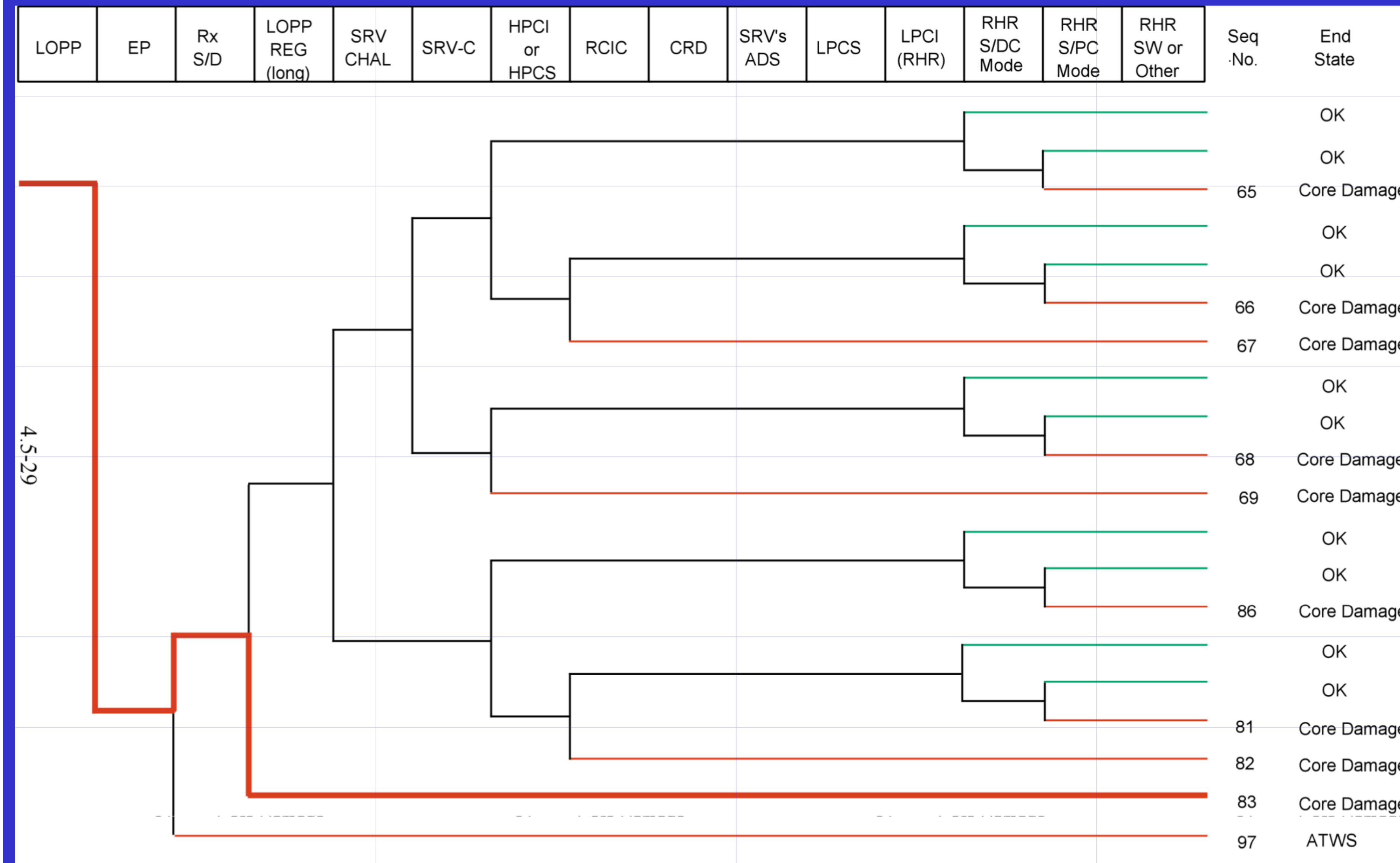


Figure 4.5-8 Dominant Core Damage Sequence for LER 324/89-009

4.5-29

Outage Events

June 17 1989 - Brunswick 2

August 10, 1996 - Western United States

August 11, 1999 - Callaway nuclear plant

August 15, 2003 - Northeastern United States and Canada

CURRENT EVENTS

Summit: Nuclear Safety Post-Fukushima

- **Nuclear Safety Post-Fukushima** will bring regulators and policy-makers together with nuclear operators and safety experts to explore the path forward for enhancing design and operational safety. They will bring you the latest information on critical safety areas: seismic and flooding hazard analyses, enhanced capabilities for prolonged SBO events, emergency prep and training, multiunit event safety, spent fuel storage safety, and new containment vessel and exclusion standards.
- ***Tuesday, December 6, 2011 8:00 am-5:00 pm, and Wednesday, December 7, 2011 8:00 am-12:30 pm***
- ***Renaissance Dupont Circle Hotel***
- ***1143 New Hampshire Avenue, NW • Washington, DC 20037***

Learning Objectives:

1. Define the term Station Blackout.

"the complete loss of alternating current (ac) electric power to the essential and nonessential switchgear buses in a nuclear power plant (i.e. loss of the offsite electric power system concurrent with turbine trip and unavailability of the onsite emergency ac power system)."

2. Describe the impact station blackout would have when combined with an accident.

Any design basis accident in conjunction with a station blackout reduces the time until core damage and release will occur.

Learning Objectives:

3. Describe the primary method available to mitigate the consequences of a station blackout.

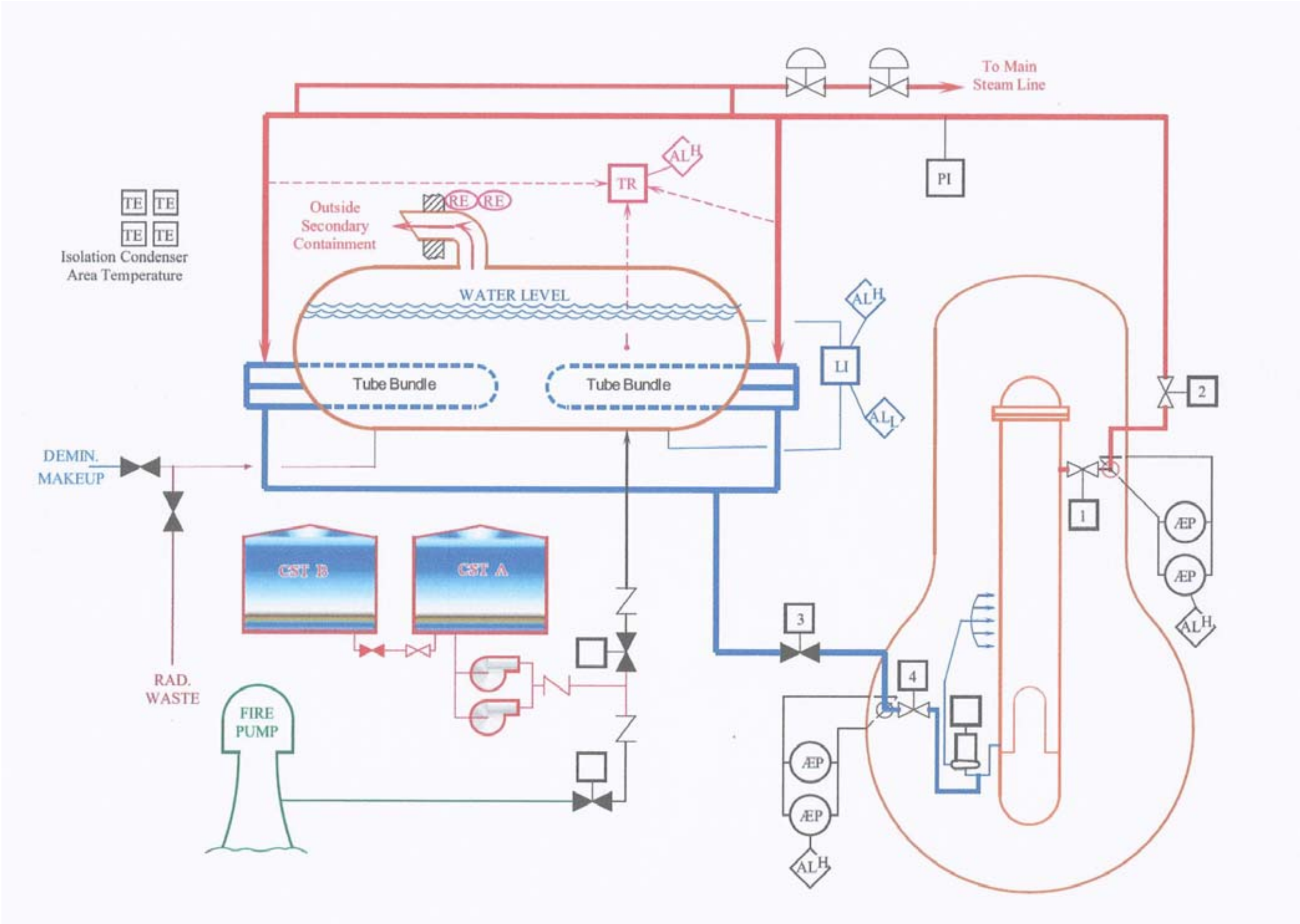
The primary method available to mitigate a station blackout with current plant design features is to initiate a controlled cooldown of the reactor. This evolution is covered in the existing Emergency Procedure Guidelines.

4. List the two major classifications Boiling Water Reactors have divided into for discussing station blackouts.

(1) those that use isolation condensers for decay heat removal but do not have makeup capability independent of ac power (BWR-2 and 3 designs), and (2) those with a reactor core isolation cooling (RCIC) system and either a high pressure coolant injection (HPCI) system or high pressure core spray (HPCS) system with a dedicated diesel, either of which is adequate to remove decay heat from the core and control water inventory in the reactor vessel, independent of ac power (BWR-4, 5, and 6 designs).

Questions?

Isolation Condensers



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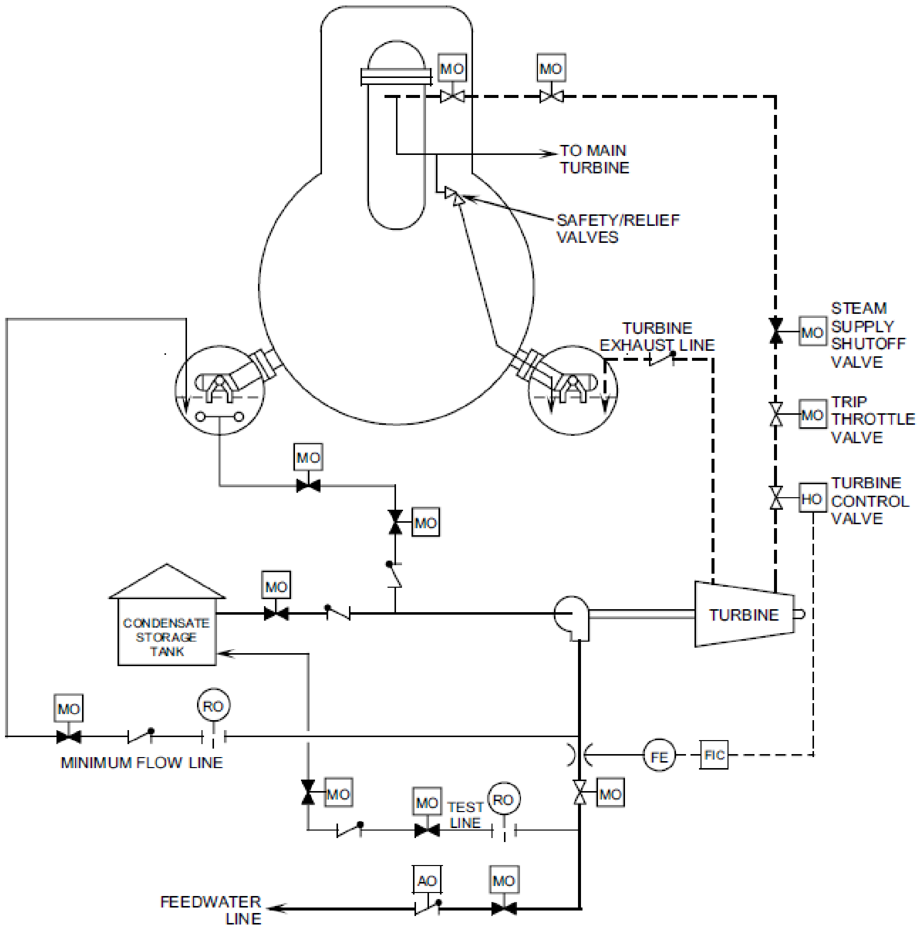
Reactor Core Isolation Cooling (RCIC)



- **Purpose**:
  - To provide makeup water to the reactor vessel for core cooling when: the main steam lines are isolated; and the condensate and feedwater system is not available.
- **Components**:
  - Steam supply isolation valves; turbine trip throttle valve; turbine control valve; turbine; turbine driven pump; and flow controller.
- **System Interfaces**:
  - Main Steam System; Primary Containment System; Condensate and Feedwater System

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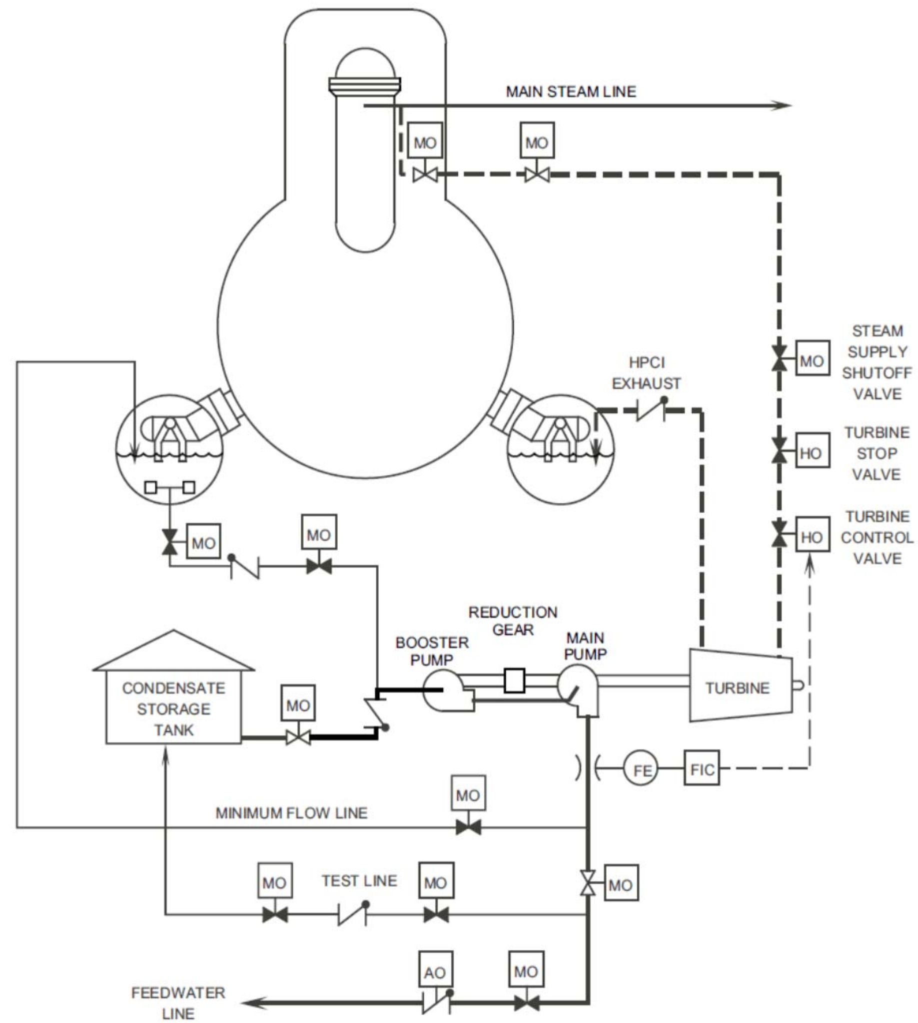
High Pressure Coolant Injection (HPCI)

- Purpose:
  - To provide makeup water to the reactor vessel for core cooling under small and intermediate sized loss of coolant accidents.
  - To backup the RCIC system.
- Components:
  - Steam supply isolation valves;
  - steam supply shutoff valve;
  - turbine stop valve;
  - turbine control valves;
  - turbine;
  - suction path;
  - pump assembly;
  - flow controller.



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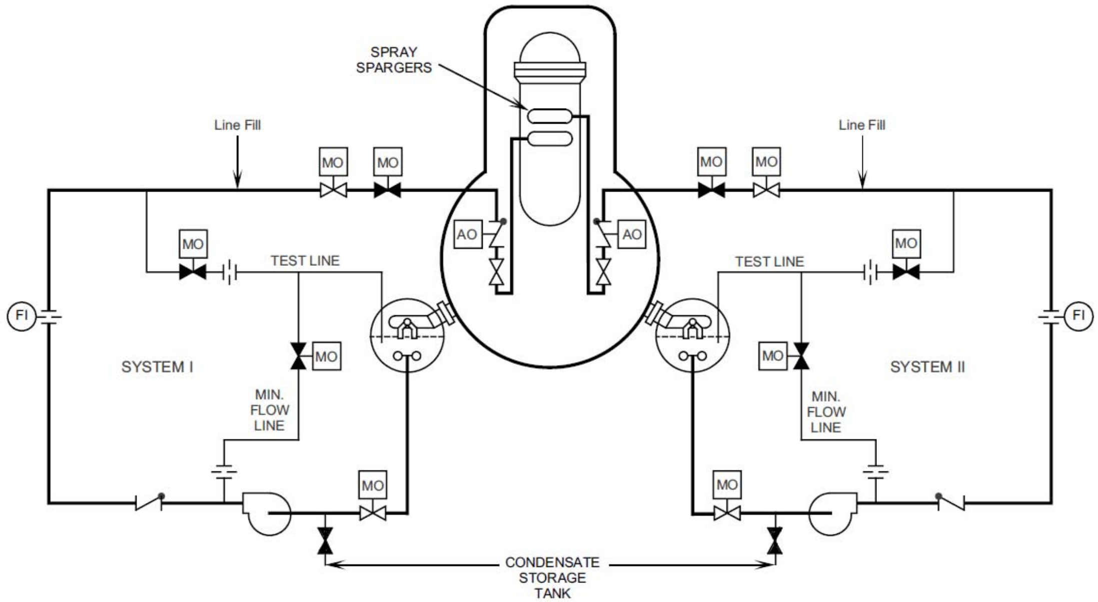
Core Spray

- **Purpose**: To provide low pressure makeup water to the reactor vessel for core cooling under loss of coolant accident (LOCA) conditions.
- **Components**: Suction path; core spray pumps; motor operated valves; testable check valves; keep full line.
- **System Interfaces**: Primary Containment System; Reactor Vessel System; Standby Auxiliary Power System.



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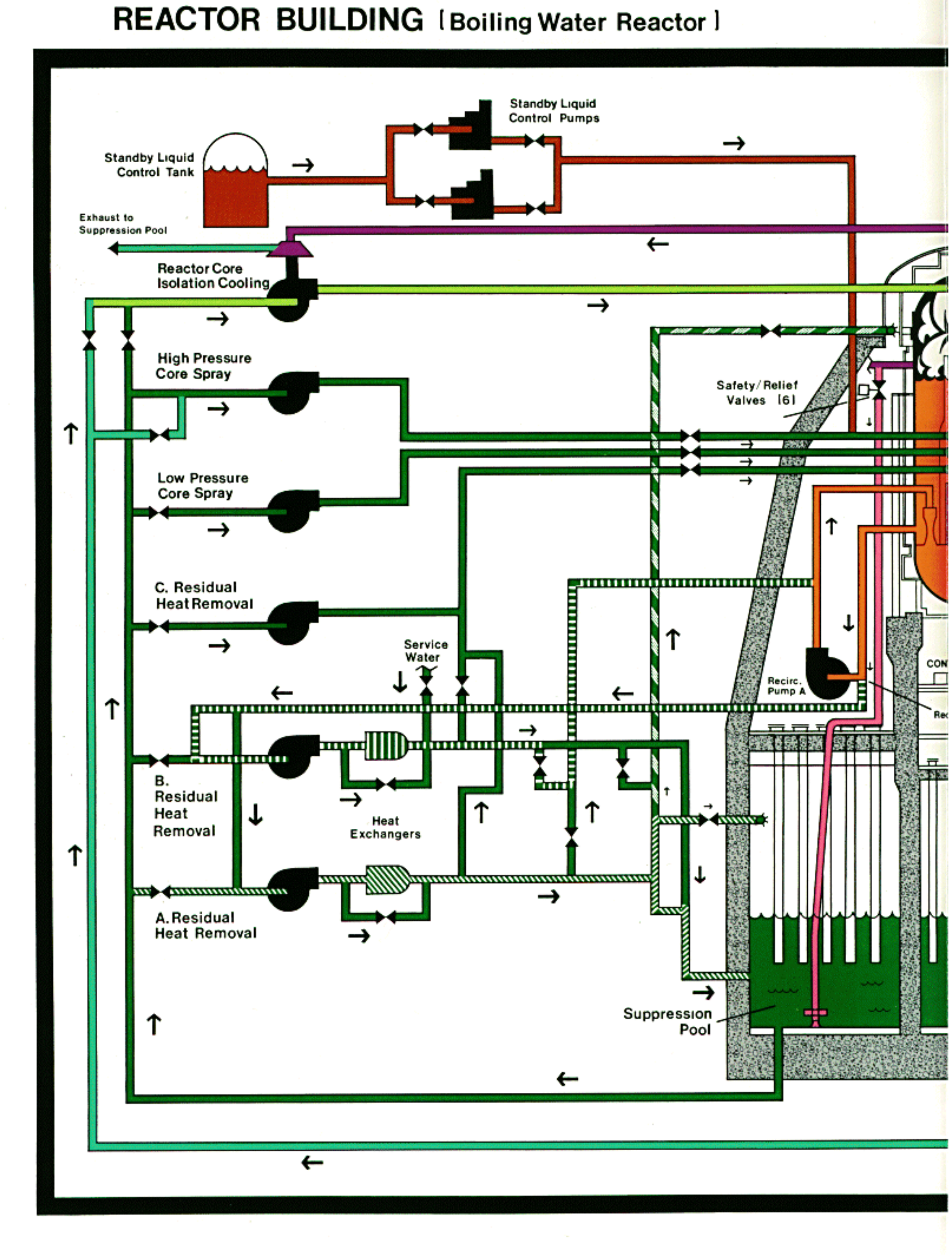

High Pressure Core Spray (HPCS)

RETURN

Residual Heat Removal (RHR)

- Purpose:
  - Low Pressure Coolant Injection Mode: To provide low pressure makeup water to the reactor vessel for core cooling under loss of coolant accident conditions.
  - Containment Spray and Cooling Mode:
    - 1. To reduce primary containment pressure following a loss of coolant accident.
    - 2. To remove heat from the suppression pool.
  - Shutdown Cooling and Head Spray Mode:
    - 1. To remove decay heat from the reactor core following a reactor shutdown.
    - 2. To remove residual heat from upper reactor vessel internals during a cooldown.

RHR (cont.)

  - Standby Coolant Supply Mode: To provide a means of flooding the primary containment.
  - Fuel Pool Cooling Mode: To provide additional heat removal capability for spent fuel in the event the Fuel Pool Cooling & Cleanup (FPCC) System is inadequate or unavailable
- Components:
  - Suction path; pumps; heat exchangers; motor operated valves; testable check valves; containment spray spargers
- System Interfaces:
  - Recirculation System; Reactor Vessel System; Primary Containment System; Reactor Building Service Water System; Fuel Pool Cooling and Cleanup System; Standby Auxiliary Power System

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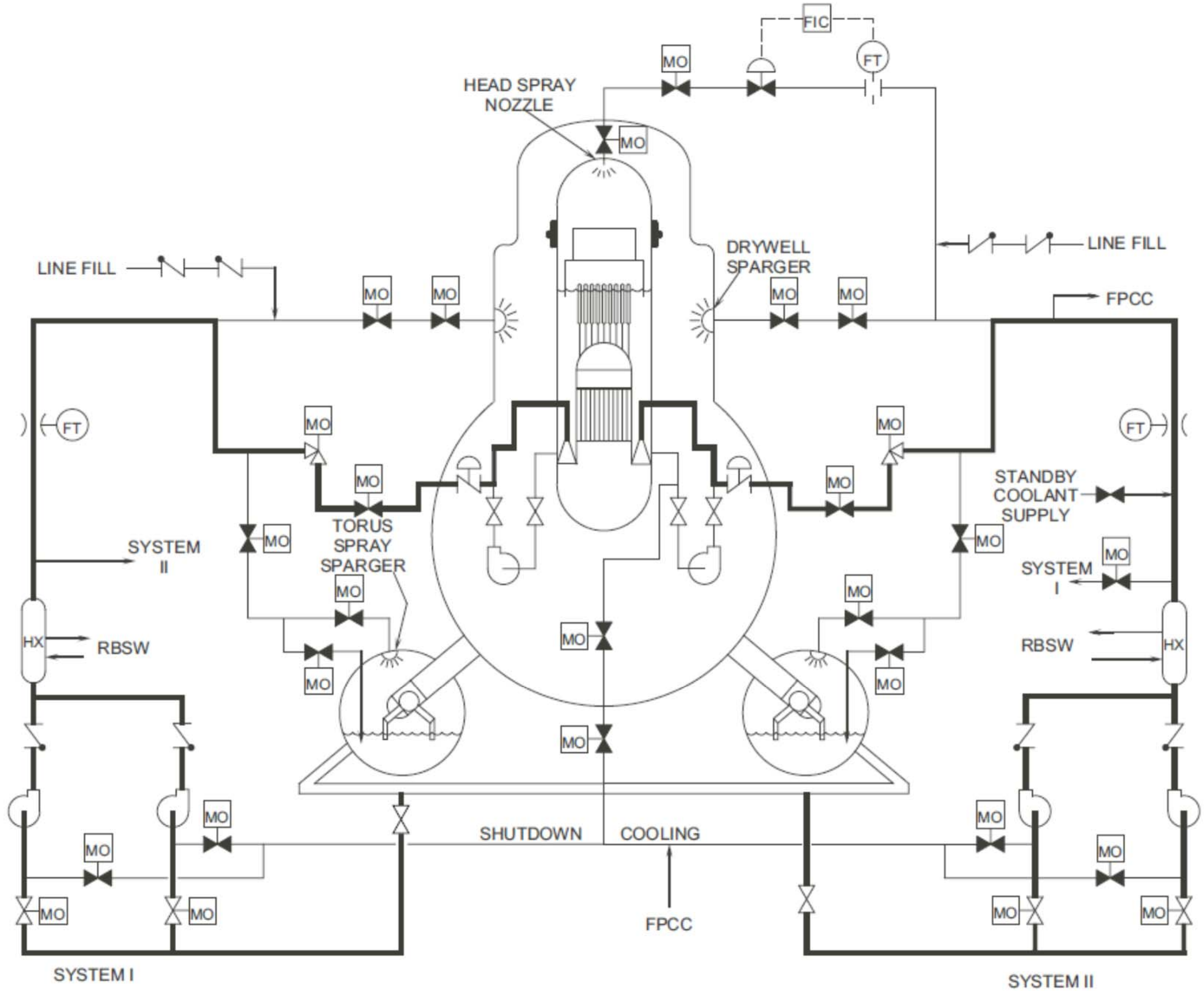

FIC
FT
MO
HEAD SPRAY NOZZLE
LINE FILL
DRYWELL SPARGER
LINE FILL
FPCC
FT
STANDBY COOLANT SUPPLY
SYSTEM II
TORUS SPRAY SPARGER
SYSTEM I
HX
RBSW
SHUTDOWN
COOLING
FPCC
SYSTEM I
SYSTEM II

RETURN

§50.63 Loss of all alternating current power.

(a) *Requirements*.

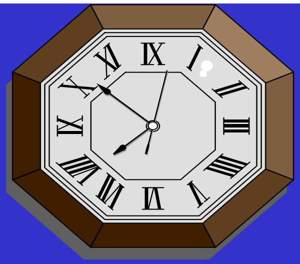
(1) Each light-water-cooled nuclear power plant licensed to operate must be able to withstand for a specified duration and recover from a station blackout as defined in §50.2. The specified station blackout duration shall be based on the following factors:

(i) The redundancy of the onsite emergency ac power sources;

(ii) The reliability of the onsite emergency ac power sources;

(iii) The expected frequency of loss of offsite power; and

(iv) The probable time needed to restore offsite power.



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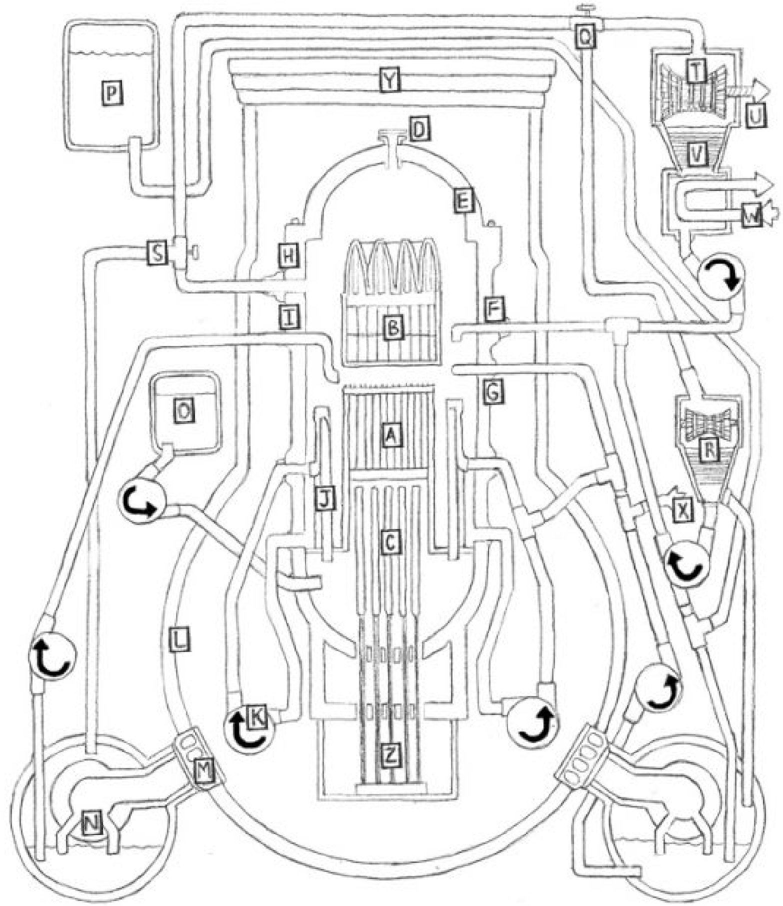
(2) The reactor core and associated coolant, control, and protection systems, including station batteries and any other necessary support systems, must provide sufficient capacity and capability to ensure that the core is cooled and appropriate containment integrity is maintained in the event of a station blackout for the specified duration. The capability for coping with a station blackout of specified duration shall be determined by an appropriate coping analysis. Licensees are expected to have the baseline assumptions, analyses, and related information used in their coping evaluations available for NRC review.

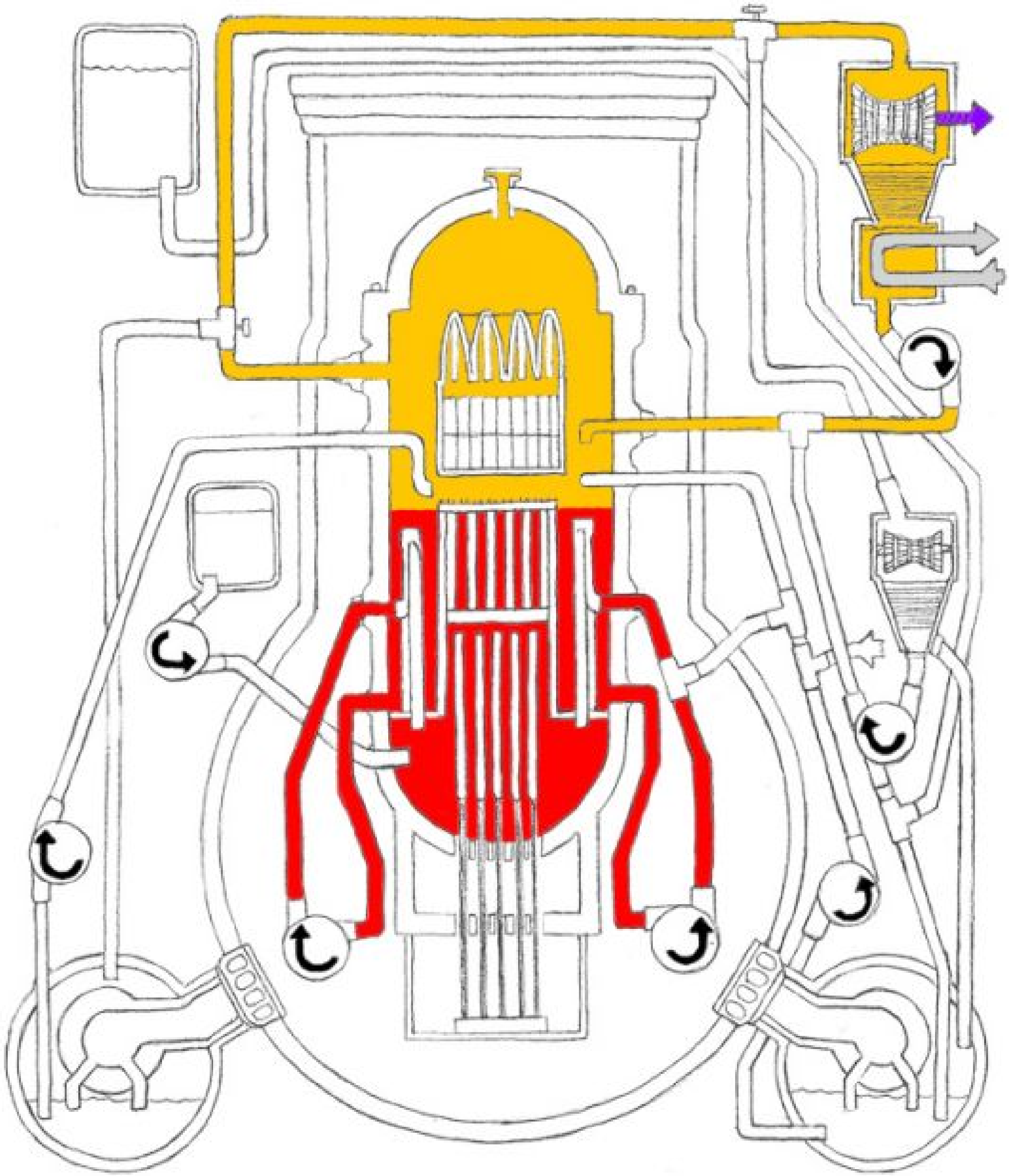
(c)(i) A proposed station blackout duration to be used in determining compliance with paragraph (a) of this section, including a justification for the selection based on the four factors identified in paragraph (a) of this section;

MARK I

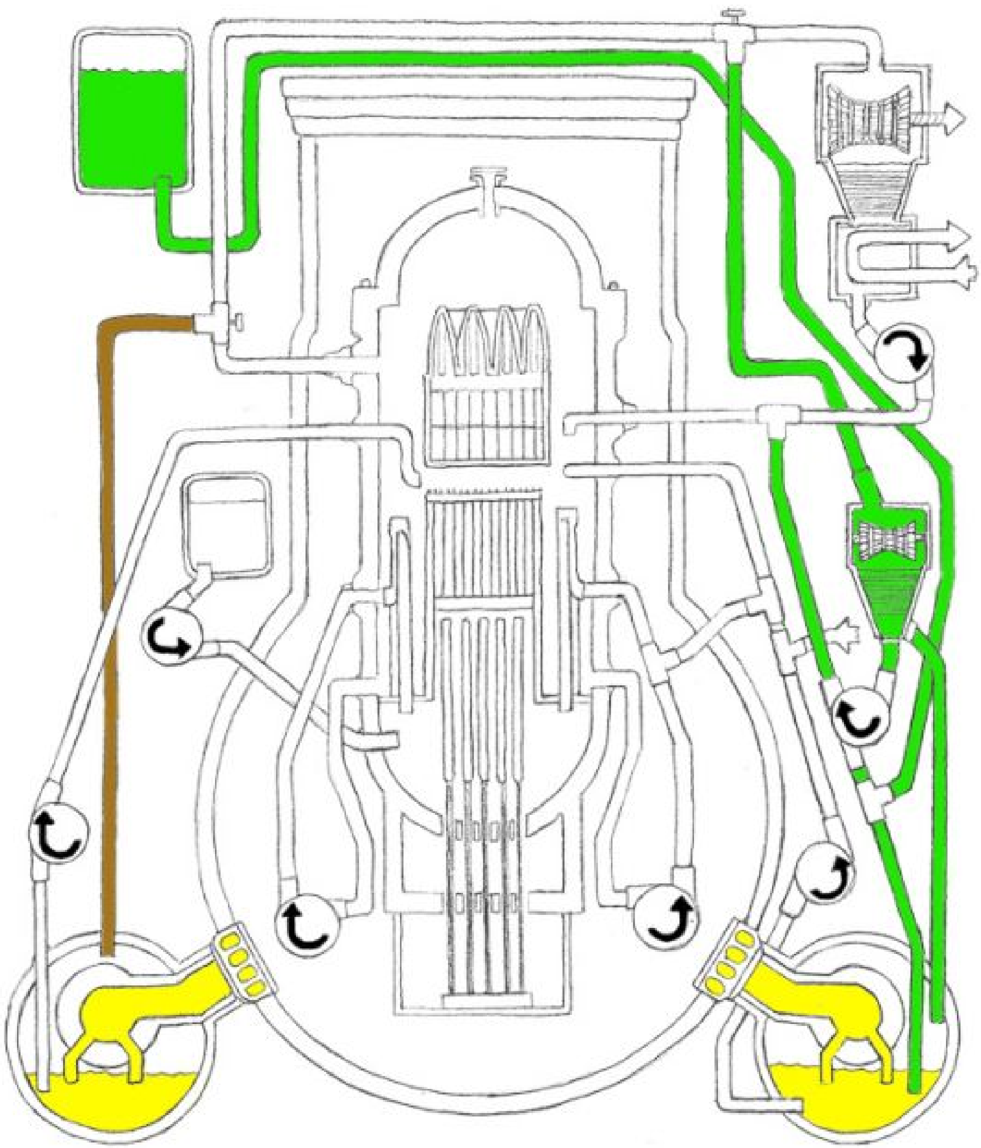


Mark I Reactor Components

(A) Uranium fuel rods; (B) Steam separator and dryer assemblies (C) Graphite control rods; (D) Vent and head spray; (E) Reactor vessel; (F) Feedwater inlet; (G) Low pressure coolant injection inlet; (H) Steam outlet; (I) Core spray inlet; (J) Jet pump; (K) Recirculation pump; (L) Concrete shell "drywell"; (M) Venting system; (N) Suppression pool; (O) Boron tank; (P) Condensate storage tank; (Q) High pressure coolant injection system; (R) HCIS turbine; (S) Automatic depressurization system; (T) Main turbine; (U) Connection to generator; (V) Condenser; (W) Circulating water; (X) Connection to outside service water; (Y) Concrete shield plug; (Z) Control rod drives.

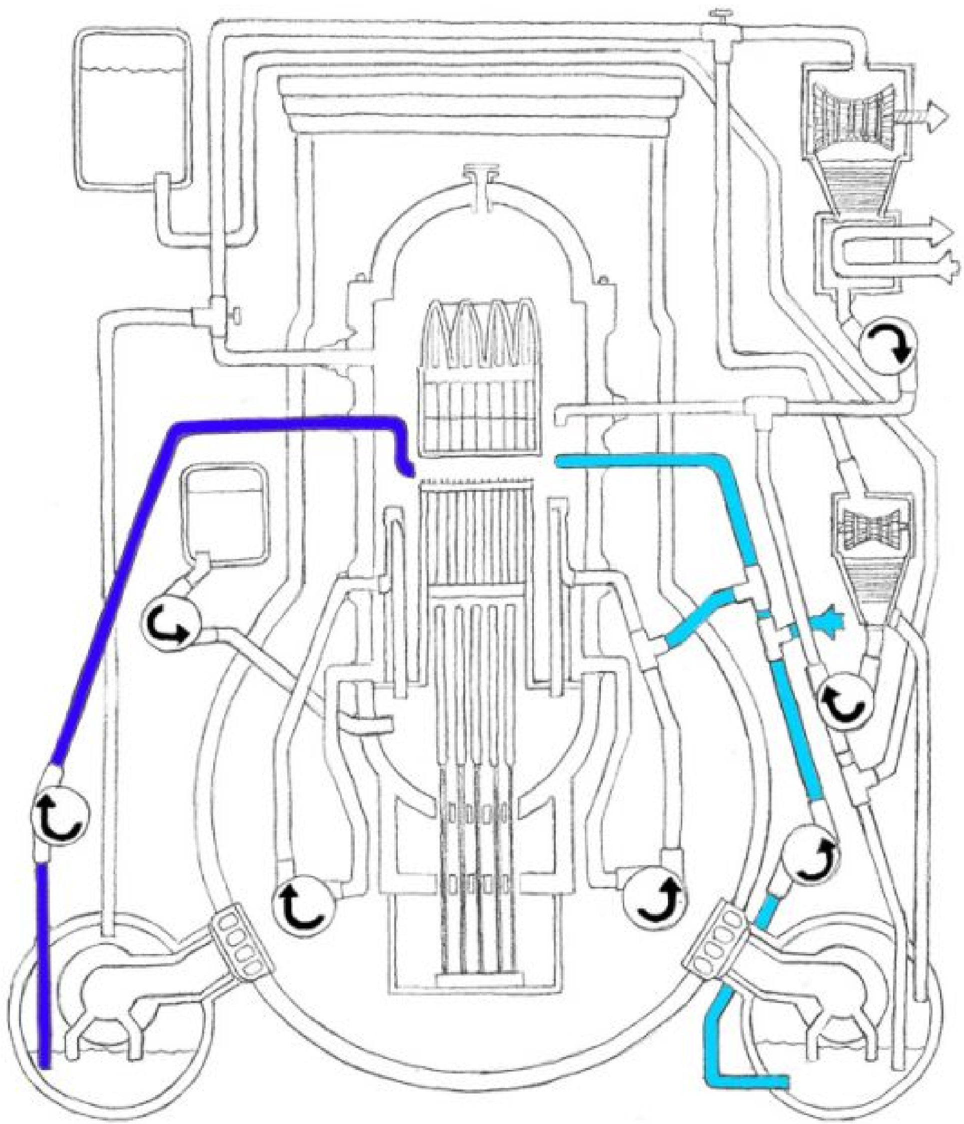


Mark I Reactor Running Normally



Mark I Reactor High Pressure Emergency Core Cooling System

Mark I Reactor Low Pressure Emergency Core Cooling System



Mark I Reactor Standby Liquid Control System