SAND2013-2628P

Nuclear Safety
(Week 8/Day 2)

Technical Aspects of Nuclear Safety

Gulf Nuclear Energy Infrastructure Institute – 2013 Fundamentals Course

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Week #8 Learning Objectives

Week 8:

- **Nuclear Safety**

Week 8 Learning Objectives:

- Understand the role of nuclear safety in a responsible nuclear energy program and how it relates to security and safety
- Recognize nuclear safety as a systems concept with basic requirements, design variables, and analysis for both technical and operations objectives
- Understand the concept of defense-in-depth (DID)
- Understand the relationship between Design Basis Accidents and reactor safety systems
- Understand Probabilistic Risk Assessment, fault trees, and event trees

Nuclear Safety
(Week 8/Day 2)

Lecture #1: Reactor Safety Systems and Defense in Depth

Dr. Michael Schuller

Today’s Primary Learning Objective

Primary Day 2 Learning Objective:

- Recognize nuclear safety as a systems concept with basic requirements, design variables, and analysis for both technical and operations objectives

Take away from this lecture:

- Technical design of NPP safety systems is based on the concept of Defense-in-Depth (DID) that includes Inherent Safety.

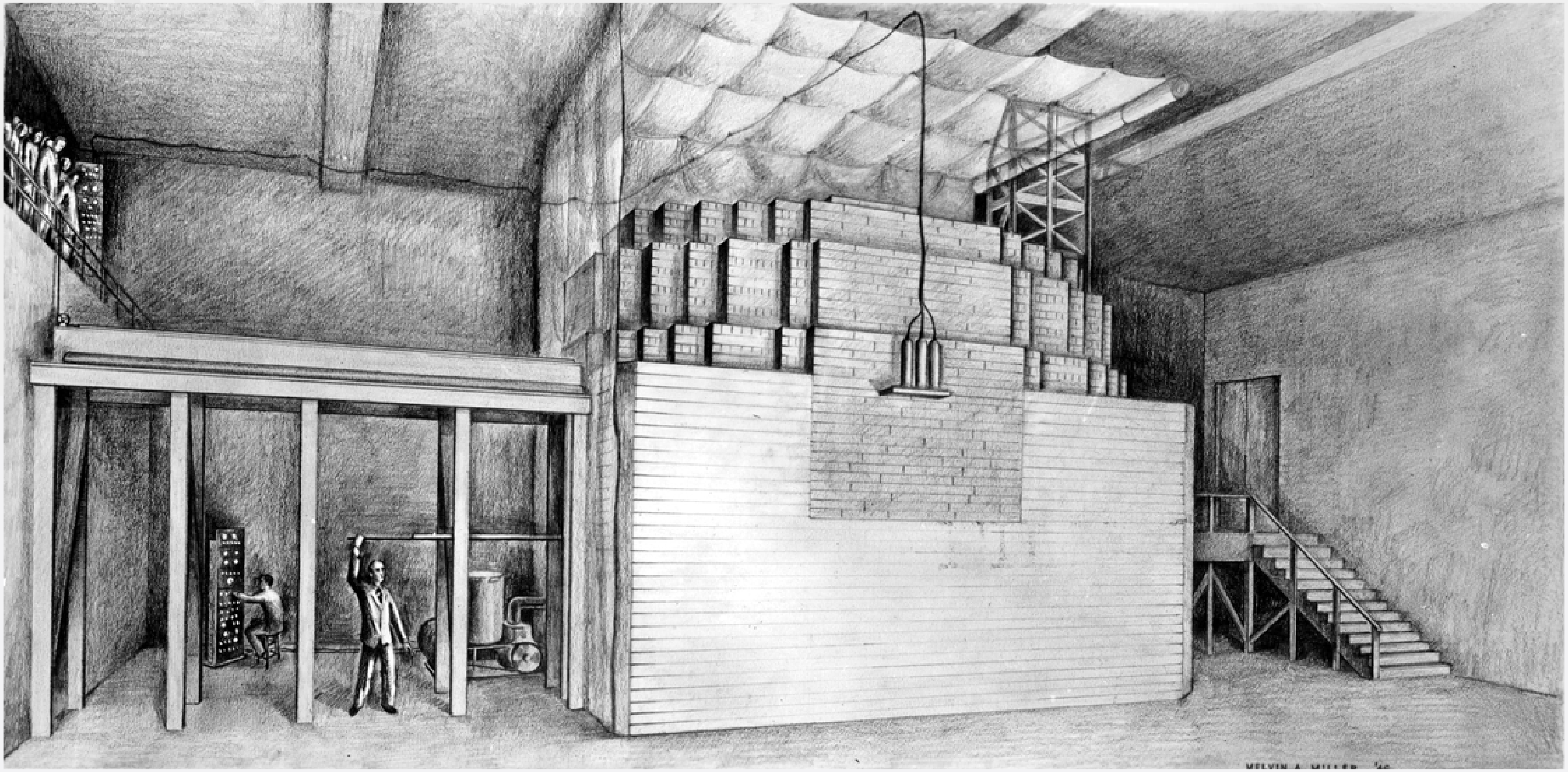
Overview

- History
- Classes of Safety Systems
- Physical Barriers
- Containment
- Safety Criteria

History of Nuclear Safety

- Earliest reactor designs had large safety margins to compensate for the lack of detailed knowledge
  - Incredibly short development period
  - No computers available!
- The idea of “containment” first arose in 1947
  - Now required for licensing in US

Chicago Pile 1 Sketch



http://fermi.lib.uchicago.edu/fermiimages.htm

History of Nuclear Safety

- A high level of safety was achieved in the first generation of NPP’s.
- Safety levels have been raised for new plants
- Measures have been taken to improve the safety of plants already in operation

History of Nuclear Safety

- Extensive testing has been carried out to
  - Understand basic phenomena
  - Develop basis for/verification testbed of analytical models
- Recent improvements to LWR safety have been incremental
  - Move towards simpler, more reliable safety systems

History of Nuclear Safety

- First “severe accident” under operating plant conditions was TMI-2
- Second was Chernobyl
- Third was Fukushima
- Tremendous data bases
- Not instrumented for scientific studies

2 Major Classes of Safety Systems

- Reactor Protection System (RPS)
  - Automatically terminates the nuclear chain reaction and maintains the reactor in a safe condition in response to a system transient or malfunction that might cause core damage
  - Generally consists of sensors, trip logic, electronic data handling, etc.
  - Does not include control rods
  - Actuation set points and priorities set by “safety chain”
    - Reactor shutdown – hydraulic scram and/or fine motion control rod insertion
    - Reactor isolation – closure of reactor containment isolation valves
    - Emergency core cooling – actuation of emergency core cooling system and primary system depressurization

2 Major Classes of Safety Systems

- Engineered Safety Features (ESF)
  - A safety system that provides a safety function to mitigate consequences of accidents that may cause major fuel damage and cannot be accommodated by the RPS alone
  - Prevents the release of radioactive material in excess of NRC requirements
  - Ensures that the radiation exposure to plant personnel does not exceed allowable limits.

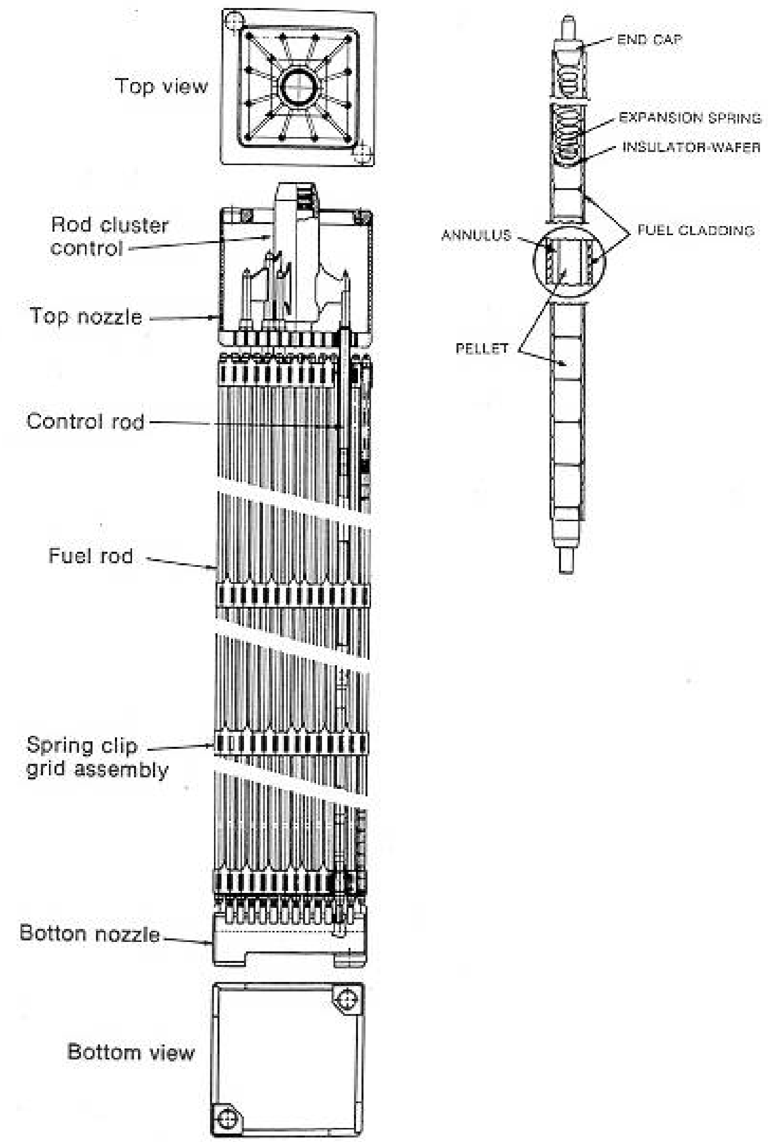
LWR ESF’s

- Overpressure protection systems
- Emergency Core Cooling System (ECCS)
- Emergency boration system
- Main steam isolation system
- Long-term cooling system
- In-core neutron monitoring system
- Water cleanup systems
- Many others!

Physical Barriers to Release of Radioactivity

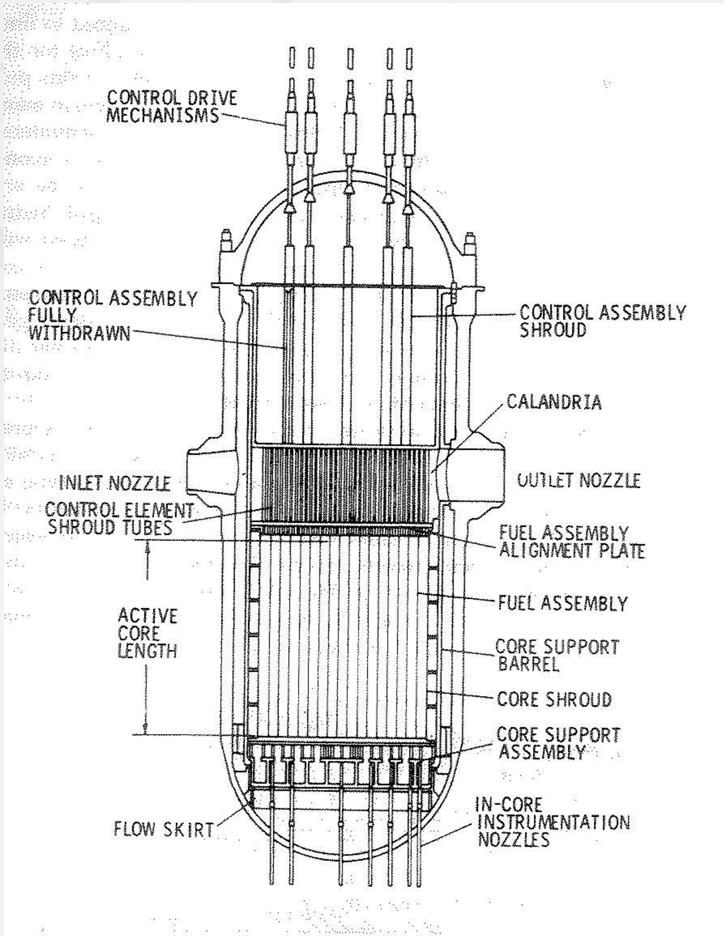
- Fuel pellet
- Fuel cladding
- Reactor coolant
- Reactor pressure boundary
- Containment building

Physical Barriers to Release of Radioactivity



A.V. Nero, Jr., *A Guidebook to Nuclear Reactors*, 1979.

PWR Fuel Assembly



ig. 1.5 Vertical cross section of large PWR vessel (courtesy Combustion Engineering).

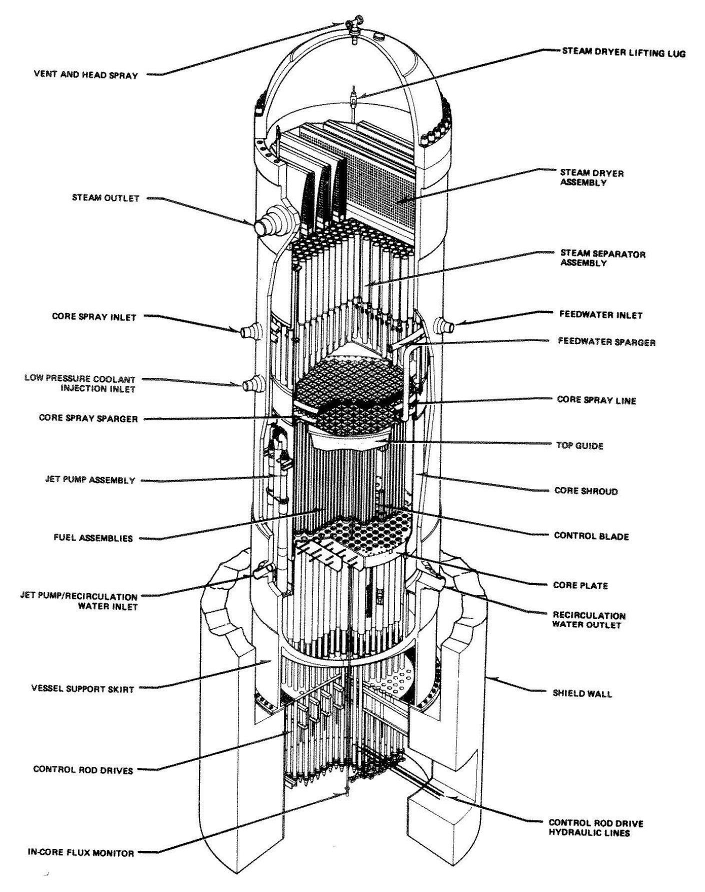


Figure 2-1. Reactor Assembly

Ref. Tong and Weisman, Thermal Analysis of Pressurized Water Reactors, 3rd ed., 1996.

Ref. General Electric Company, BWR/6 General Description of a Boiling Water Reactor, 1980.

Containment Safety Functions

- Limit offsite releases
- Maintain structural integrity (within specified leakage) during design basis accidents (DBAs)
- Reduce post-DBA pressure/temperature to allow recovery
- Protect RCS from external hazards

Containment Heat Removal

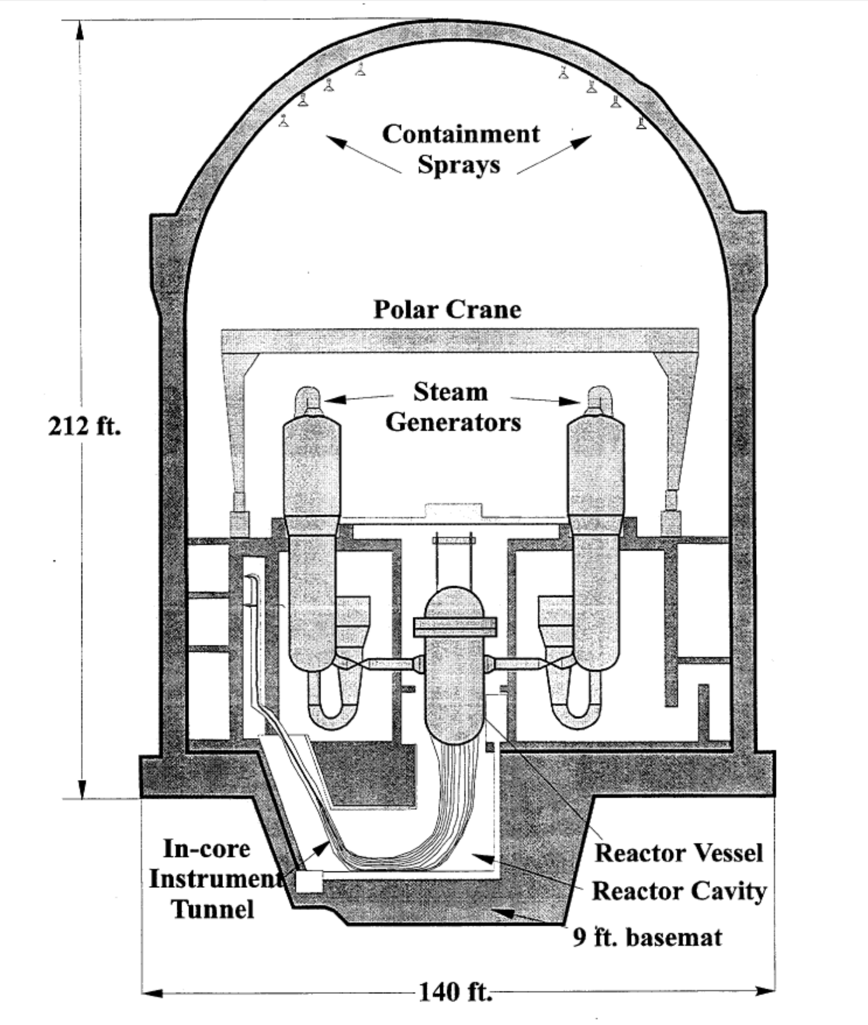
- Principal heat removal mechanisms
  - BWR
    - Pressure suppression pool receives blowdown steam and is cooled by the RHR system in the “pool cooling mode”
  - PWR
    - In recirculation mode ECCS pumps sump water through the shutdown cooling heat exchanger (cooled by the service water system) back to the RCS; containment fans and spray systems are also designed to remove all decay heat

Containment Safety Systems

- The first three containment safety functions depend on
  - Emergency Core Cooling System(ECCS)
  - Residual Heat Removal (RHR) system
  - Recirculation system
  - CIS (Containment Isolation System)
  - CSS (Containment Spray System)
  - Containment heat removal system
  - Combustible gas control system
  - Support systems including
    - Electric power supplies
    - Service water systems
    - Component cooling water systems

Containment Safety Systems

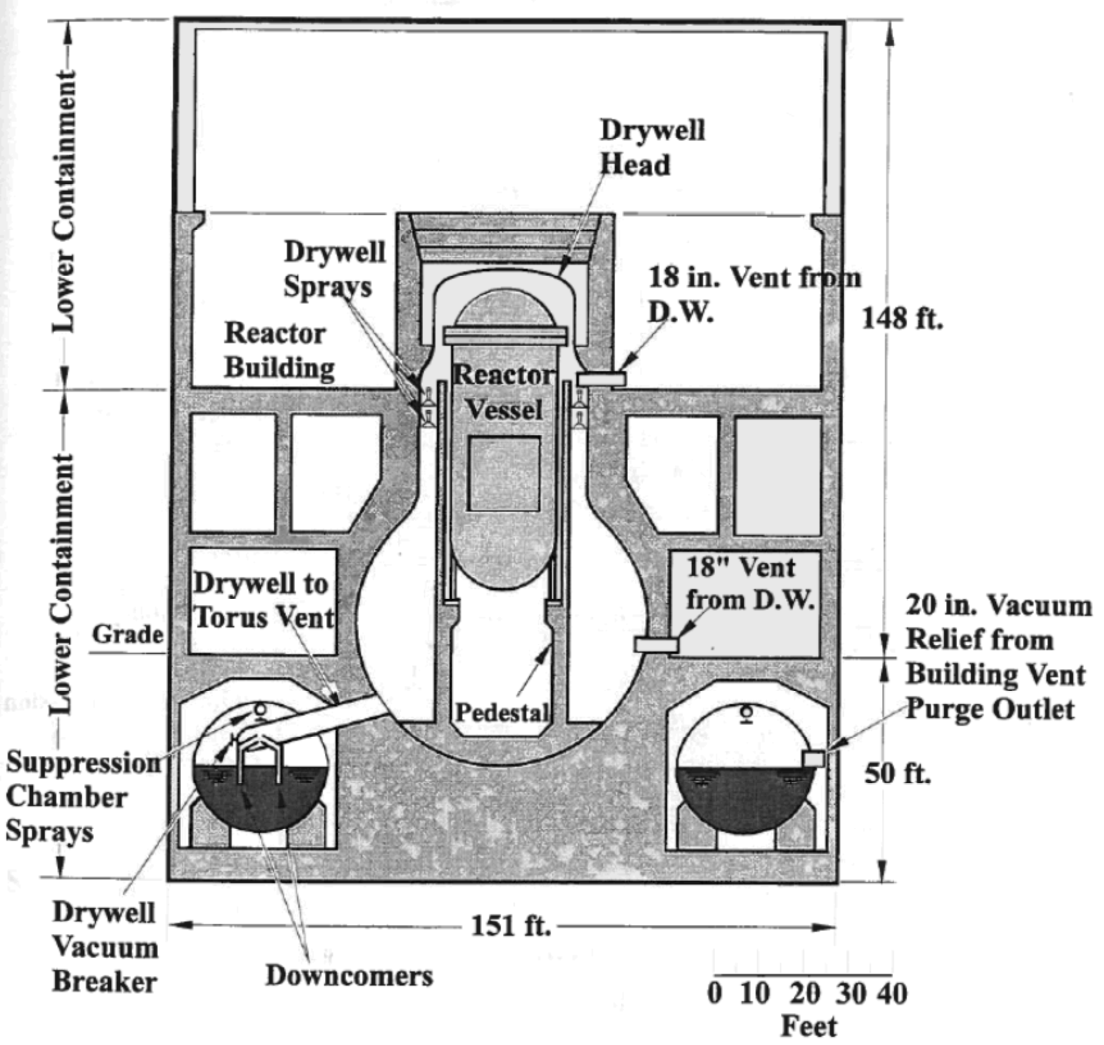
PWR Containment



Ref.: Perspectives on Reactor Safety, NUREG/CR-6042, Rev. 2, SAND 93-0971, 2002, http://www.nrc.gov/reading-rm/doc-collections/nuregs/contract/cr6042/

Containment Safety Systems

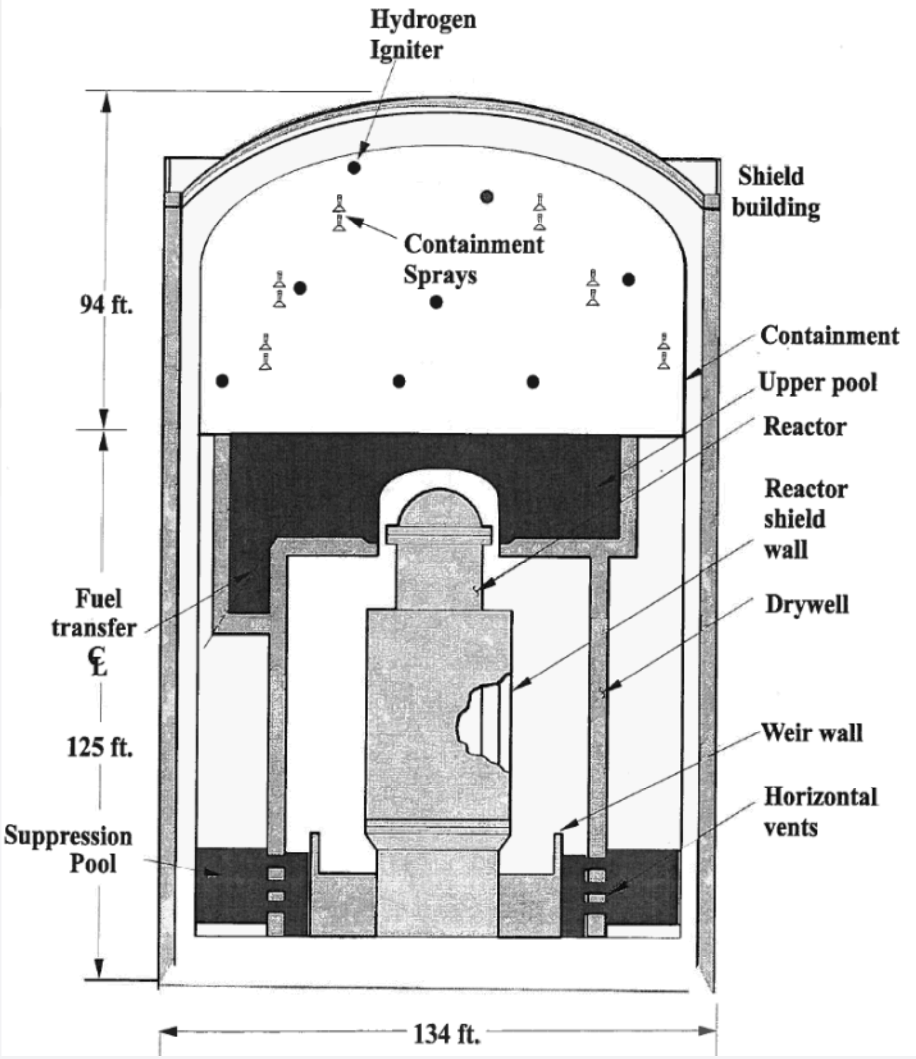
BWR Mark I Containment



Ref.: Perspectives on Reactor Safety, NUREG/CR-6042, Rev. 2, SAND 93-0971, 2002, http://www.nrc.gov/reading-rm/doc-collections/nuregs/contract/cr6042/

Containment Safety Systems

BWR Mark III Containment

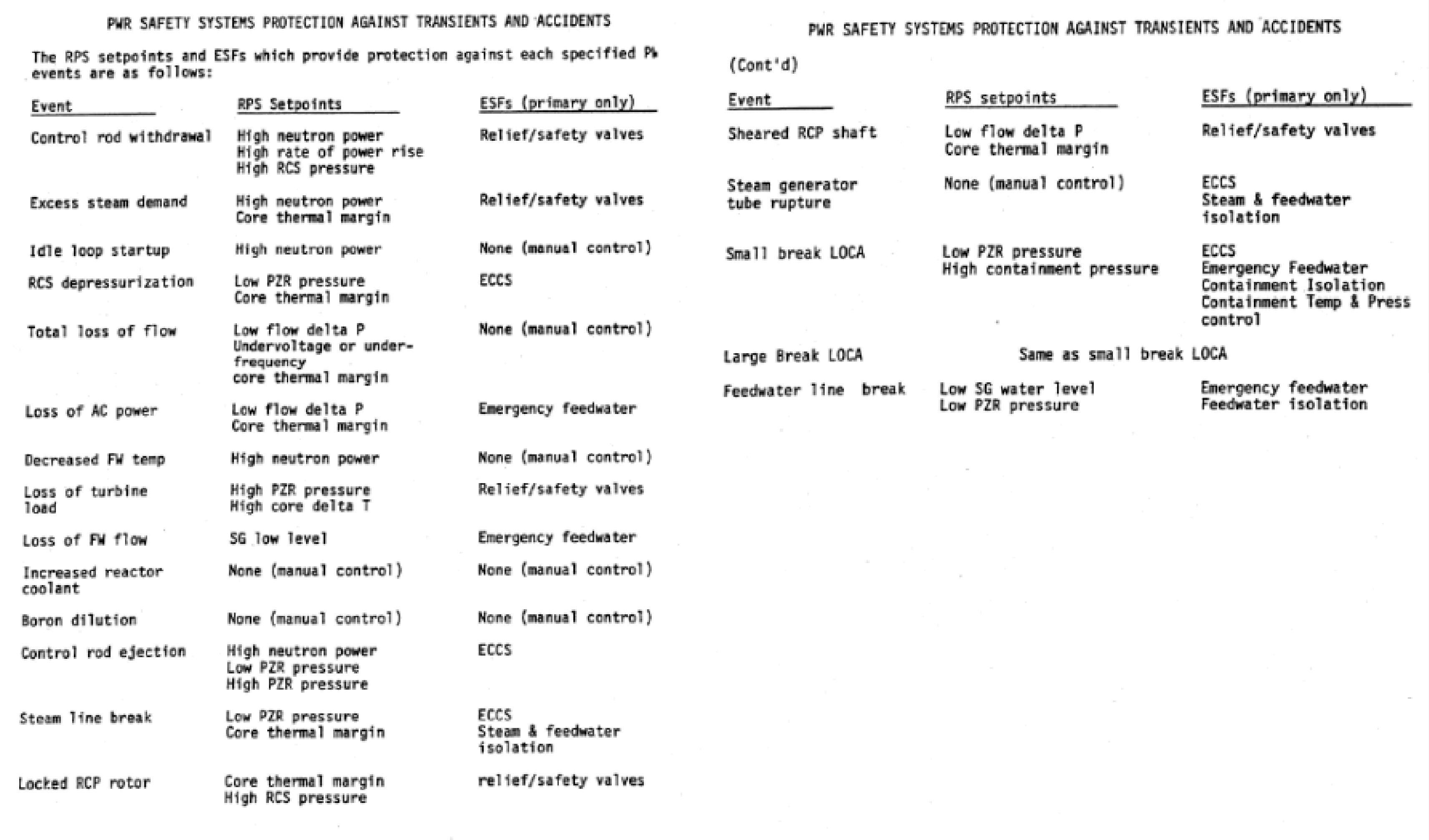


Ref.: Perspectives on Reactor Safety, NUREG/CR-6042, Rev. 2, SAND 93-0971, 2002, http://www.nrc.gov/reading-rm/doc-collections/nuregs/contract/cr6042/

Containment Failure Modes

1. Over-Pressure
   - Blowdown and coolant injection
   - Direct containment heating
   - Core-concrete interaction
   - Hydrogen combustion
   - Anticipated Transients Without SCRAM heating
2. Over-Temperature
   - Penetration failures (electrical)
   - Concrete creep
3. Bypass

PWR Safety Systems Protection



PWR SAFETY SYSTEMS PROTECTION AGAINST TRANSIENTS AND ACCIDENTS

The RPS setpoints and ESFs which provide protection against each specified PW events are as follows:

Event	RPS Setpoints	ESFs (primary only)
Control rod withdrawal	High neutron power High rate of power rise High RCS pressure	Relief/safety valves
Excess steam demand	High neutron power Core thermal margin	Relief/safety valves
Idle loop startup	High neutron power	None (manual control)
RCS depressurization	Low PZR pressure Core thermal margin	ECCS
Total loss of flow	Low flow delta P Undervoltage or under-frequency core thermal margin	None (manual control)
Loss of AC power	Low flow delta P Core thermal margin	Emergency feedwater
Decreased FW temp	High neutron power	None (manual control)
Loss of turbine load	High PZR pressure High core delta T	Relief/safety valves
Loss of FW flow	SG low level	Emergency feedwater
Increased reactor coolant	None (manual control)	None (manual control)
Boron dilution	None (manual control)	None (manual control)
Control rod ejection	High neutron power Low PZR pressure High PZR pressure	ECCS
Steam line break	Low PZR pressure Core thermal margin	ECCS Steam & feedwater isolation
Locked RCP rotor	Core thermal margin High RCS pressure	relief/safety valves

PWR SAFETY SYSTEMS PROTECTION AGAINST TRANSIENTS AND ACCIDENTS

(Cont'd)

Event	RPS setpoints	ESFs (primary only)
Sheared RCP shaft	Low flow delta P Core thermal margin	Relief/safety valves
Steam generator tube rupture	None (manual control)	ECCS Steam & feedwater isolation
Small break LOCA	Low PZR pressure High containment pressure	ECCS Emergency Feedwater Containment Isolation Containment Temp & Press control
Large Break LOCA	Same as small break LOCA	
Feedwater line break	Low SG water level Low PZR pressure	Emergency feedwater Feedwater isolation

Acceptance Criteria - Types

- Fuel Temperature
- Fuel Energy Deposition
- Clad Heat Transfer
- Clad Temperature
- Clad Integrity
- Radioactivity In Coolant
- Primary Or Secondary System Pressure
- Containment Pressure
- Radioactivity In Containment
- Dose In Containment
- Dose To Site Workers
- Dose At Site Boundary
- Dose In Low Population Zone
- Dose To The Public
- Public Health Effects

Acceptance Criteria

Acceptance Criteria Dependence on Event Frequency

Ref. ANSI/ANS-51.1-1983

EVENT FREQUENCY (F) *(per Reactor-Year)*	*ACCEPTANCE CRITERIA* *(Dose Limit)*
Normal Operation	*10CFR50, App. I*
$F \geq 10^{-1}$	*10CFR50, App. I*
$10^{-1} > F \geq 10^{-2}$	*10% 10CFR100*
$10^{-2} > F \geq 10^{-4}$	*25% 10CFR100*
$10^{-4} > F \geq 10^{-6}$	*100% 10CFR100*

Conclusions

- Nuclear safety has always been a high priority
- The two major classes of safety systems in LWRs are the Reactor Protection System and Engineered Safety Features
- Several physical barriers to release of radioactivity are designed into reactors
- The containment has several safety functions
- Several types of safety criteria exist
  - 10CFR50 – Appendix A documents safety requirements

References

- Enrico Fermi Image Gallery, The University of Chicago Library Digital Activities & Collections, http://fermi.lib.uchicago.edu/fermiimages.htm
- Nero, Anthony V., A Guidebook to Nuclear Reactors, University of California Press, Berkeley, 1979.
- Tong, L. S., and J. Weisman, Thermal Analysis of Pressurized Water Reactors, 3rd ed., 1996.
- General Electric Company, BWR/6 General Description of a Boiling Water Reactor, 1980.
- The Westinghouse Pressurized Water Reactor Nuclear Power Plant, Westinghouse Electric Co. LLC, Pittsburgh, PA (2006)
- *Perspectives on Reactor Safety*, NUREG/CR-6042, Rev. 2, SAND 93-0971, 2002, http://www.nrc.gov/reading-rm/doc-collections/nuregs/contract/cr6042/
- PWR Safety Systems Protection – RPS Setpoints, from Bill Burchill's NUEN 609 lecture notes, 2005
- Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants, ANSI/ANS-51.1-1983
- 10 CFR Appendix A to Part 50—General Design Criteria for Nuclear Power Plants, US NRC, http://www.nrc.gov/reading-rm/doc-collections/cfr/part050/part050-appa.html#6_appa

Assignments, Activities, and Exercises

- Differentiate between the major classes of safety systems in light water reactors
- List the physical barriers to release of radioactivity from nuclear fuel in a reactor and discuss why each is needed.
- Explain the main conceptual difference between containments for PWRs and BWRs
- Look up allowable dose limits specified by regulators
  - Discuss any points of contention that you may see in these limits.
- Summarize the safety criteria in Appendix A to 10 CFR Part 50

Nuclear Safety
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Lecture #2: Passive Safety

Dr. Michael Schuller

Today's Primary Learning Objective

Primary Day 2 Learning Objective:

- Recognize nuclear safety as a systems concept with basic requirements, design variables, and analysis for both technical and operations objectives

Take away from this lecture:

- Technical design of Nuclear Power Plant (NPP) safety systems is based on the concept of Passive Safety Systems.

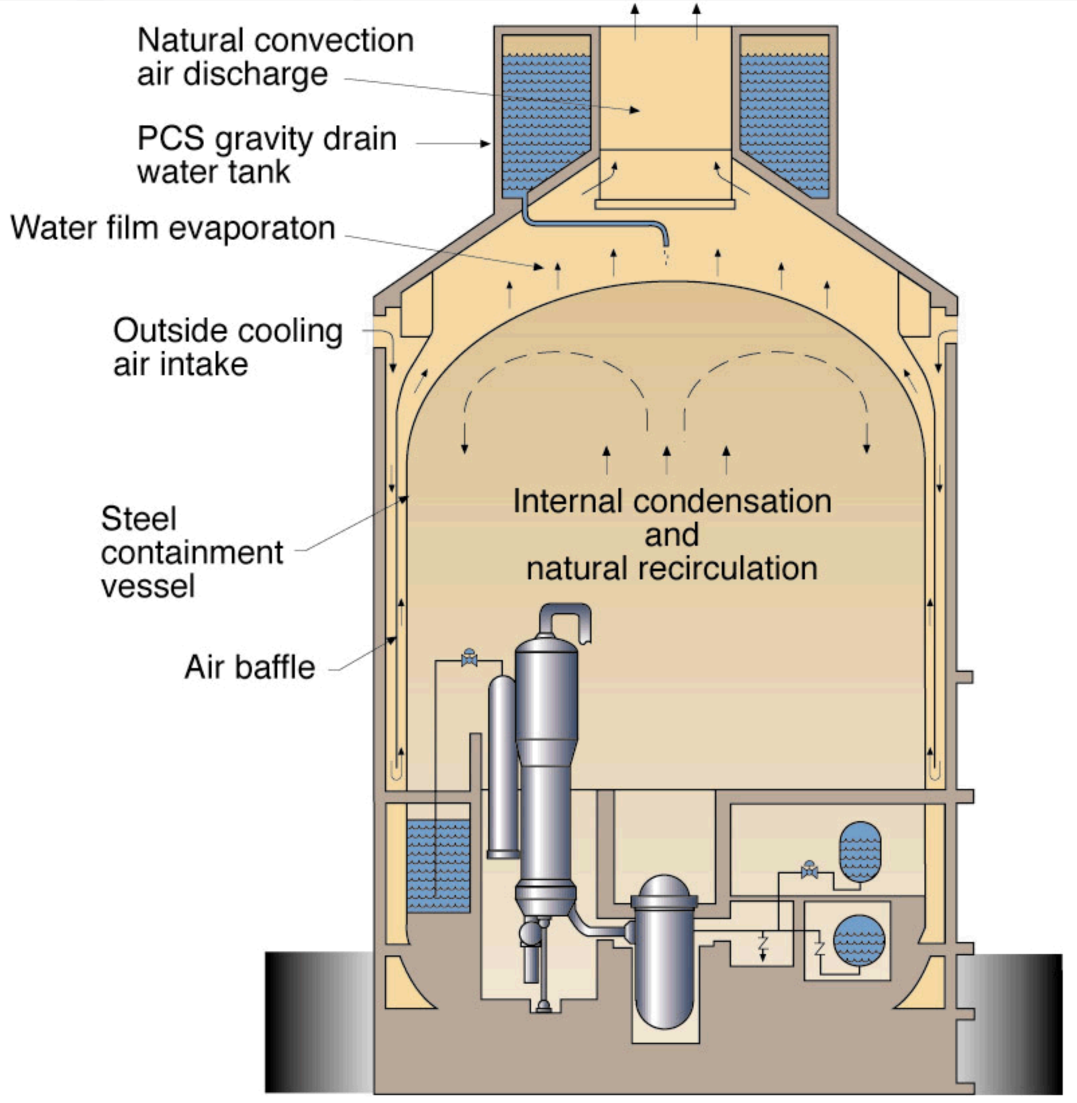
Overview

- Definitions
- Natural Circulation
- AP600 Containment Passive Systems

Definition of Passive Safety

- In response to increasingly higher demands on nuclear power plant safety systems, the nuclear community is responding by developing “passive” safety systems that are inherently safe.
  - Passive systems are those that do not rely on:
    - Power supplies
    - Moving parts
    - Human intervention
  - Passive systems provide failsafe safety mechanisms

AP600 Containment Passive Safety Systems



http://www.westinghousenuclear.com/images/news_room/AP600PassiveContainment.jpg

Definition of Passive Safety

- Condensation heat transfer is a primary driving force for many of long-term heat removal systems.
  - Passive condensation heat transfer systems are reliable and inherently safe.
  - They rely on basic physics forces to transport energy and maintain the plant within design specifications.
    - Gravity
    - Buoyancy
    - Small pressure differences
  - The development of innovative systems for future reactors promises to further increase the safety, reliability and economic competitiveness of nuclear power.

Conclusions

- Passive safety systems operate in a failsafe manner by the very nature of their design
  - Reliance on natural forces
  - No need for intervention or active systems
- Some advanced LWR designs feature passive systems as replacements for active, more complicated safety systems.

References

- Beard, Alan J., "ESBWR Overview," Nuclear Power 2010 Training Session Materials, September 15, 2006, http://www.ne.doe.gov/np2010/pdfs/esbwrOverview.pdf
- A. S. Rao, "ESBWR Design Technology and Program Plan Overview", NRC-Staff-GE Meeting, Rockville, MD, June 20-21, 2002.
- R. Gamble, GE's ESBWR Overview, March 9, 2005, NRC Accession # ML050770285, http://adamswebsearch2.nrc.gov/idmws/doccontent.dll?library=PU_ADAMS^PBNTAD01&ID=050770508
- http://www.westinghousenuclear.com/images/news_room/AP600PassiveContainment.jpg
  - Permission to use this image is stated here: http://www.westinghousenuclear.com/News_Room/image_gallery.shtm

Assignments, Activities, and Exercises

- Discuss the benefits and costs of passive safety systems.

Nuclear Safety
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Lecture #3: Reactor Transients and Accidents

Dr. Michael Schuller

Today’s Primary Learning Objective

Primary Day 2 Learning Objective:

- Recognize nuclear safety as a systems concept with basic requirements, design variables, and analysis for both technical and operations objectives

Take away from this lecture:

- Technical design of NPP safety systems is based on the concept of Reactor Transients and Accidents.

Lecture Overview

- Event Classifications
- Reactor Transients
- Severe Accidents

Event Classification

- NRC Definitions
  - Anticipated Operational Occurrences (AOO)
    - “Conditions of normal operation which are expected to occur one or more times during the life of the nuclear power unit and include but are not limited to loss of power to all recirculation pumps, tripping of the turbine generator set, isolation of the main condenser, and loss of all offsite power.”
    - Ref. for all definitions: NUREG-0800, US NRC Standard Review Plan, Chapter 15

Event Classification

- NRC Definitions
  - Postulated accidents
    - "Unanticipated occurrences (i.e., they are postulated but not expected to occur during the life of the nuclear power plant)."

Event Classification

- Accident Categories
  - Design Basis Accident (DBA)
    - "Postulated accidents that are used to set design criteria and limits for the design and sizing of safety-related systems and components."
  - Design Basis Events (DBE)
    - "Conditions of normal operation, including AOOs, design-basis accidents, external events, and natural phenomena, for which the plant must be designed to ensure functions of safety-related electric equipment that ensures the integrity of the reactor coolant pressure boundary; the capability to shut down the reactor and maintain it in a safe shutdown condition; or the capability to prevent or mitigate the consequences of accidents that could result in potential offsite exposures."

Event Classification

- Classification
  - Initiating events may be categorized according to expected frequency of occurrence and by type.
    - Categorization by frequency of occurrence provides a basis for selection of the applicable analysis acceptance criteria for each initiating event.
    - Categorization by type provides a basis for comparison between events.

Event Classification

- Example of categorization of events by frequency
  - Anticipated Operational Occurrence (AOO):
    - A frequent event
    - Expected mean frequency of occurrence > 10 - 2 per plant-year
  - Design Basis Accident (DBA):
    - An infrequent event that might occur once during the collective lifetimes of a large number of plants
    - The plant is specifically designed to mitigate the event using only equipment classified as safety grade
    - Typically associated with events having a mean frequency between 10 - 2 and 10 - 4 per plant-year.

Event Classification (2)

- Beyond Design Basis Accident (BDBA):
  - Very low-probability event not expected to occur within the collective lifetimes of a large number of similar plants
  - Plant design would mitigate the consequences, taking credit for
    - Available safety-related equipment
    - Operator actions
    - Any existing non-safety related equipment
    - Accounting for long time periods available for corrective actions
  - BDBAs typically have a mean frequency between 10–4 and 5 × 10–7 per plant-year.

Ref: Ball, S. J., S. E. Fisher, Next Generation Nuclear Plant Phenomena Identification and Ranking Tables (PIRTs) Volume 1: Main Report, NUREG/CR-6944, Vol. 1, ORNL/TM-2007/147, Vol. 1, March 2008.

Event Classification

- Categorization of events by type
  - AOO’s and postulated accidents can be grouped into the following seven types:
    - Increase in heat removal by the secondary system
    - Decrease in heat removal by the secondary system
    - Decrease in Reactor Coolant System (RCS) flow rate
    - Reactivity and power distribution anomalies
    - Increase in reactor coolant inventory
    - Decrease in reactor coolant inventory
    - Radioactive release from a subsystem or component

Ref: NUREG-0800, US NRC Standard Review Plan, Chapter 15

Reactor Transients

- Reactor transients fall under the category of AOO.
- Definition
  - Occurrences expected to occur frequently or regularly in the course of power operation, refueling, maintenance or plant maneuvering
    - Accommodated by the margin between a plant parameter and the value of that parameter requiring protective action
    - Or, results, at worst, in a reactor trip with the plant being able to return to operation.
  - Not expected to cause fuel rod failures, reactor coolant system failures or secondary system overpressurization in PWRs.

Example of PWR transient

- Increase in heat removal from primary system
  - Feedwater system malfunctions causing a reduction in feedwater temperature
    - Example: Low pressure or high pressure heater train is bypassed or out of service
    - Core power increases
    - Large thermal capacity of RCS and secondary diminishes the transient
    - Increased subcooling creates an increased load demand on the RCS
  - Net effect: Reactor reaches a new equilibrium condition at a power level corresponding to the new steam generator ΔT

Ref.: AP1000 Design Control Document, p. 15.1

Severe Accident Phenomena Outline

- Source Term
- In-vessel phenomena
- Ex-vessel phenomena

Ref.: material from here on is heavily referenced from Perspectives on Reactor Safety, NUREG/CR-6042, Rev. 2, SAND 93-0971, 2002, http://www.nrc.gov/reading-rm/doc-collections/nuregs/contract/cr6042/

Source Term

- The source term is the radionuclide released to the environment
- Quantities, characteristics and radionuclide transport must be known to assure that public safety is protected
- If the energy in the core is not controlled, damage can be done to the barriers containing radionuclides

Source Term

- Volatile isotopes are most likely to be released from the plant to the environment during an accident
- Most volatile isotopes: Krypton & Xenon (noble gases)
- Second most volatile isotopes: Iodine, Caesium
  - Radioiodine concentrates in the thyroid → hazard
  - Radioactive cesium is a source of long-term offsite dose (e.g. from Chernobyl and Fukushima)

Source Term

- Magnitude also affects source
- Handout shows source term characteristics

Ref.: *Perspectives on Reactor Safety,* NUREG/CR-6042, Rev. 2, SAND 93-0971, 2002, http://www.nrc.gov/reading-rm/doc-collections/nuregs/contract/cr6042/ (page 5.1-12)

Source Term

- Release forms
  - Gases
    - Kr, Xe, I_2
  - Aerosol particles of water-soluble substances
    - CsI, CsOH, $Sr(OH)_2$
  - Slightly soluble oxides
    - Te, Ru, La
  - Major release can be considered a plume of radioactive gases, aerosol particles and water vapor

Source Term

- The source term is characterized by
  - Fractions of core inventory that are released
  - Start time and duration of the release
  - Size distribution of aerosols
  - Elevation of the release
  - Energy released

In-Vessel Accident Stages

- In-vessel accident progression is divided into successive stages
  - Initiating event and failures leading to insufficient core cooling
  - Sustained core uncovering
    - → Core heatup
  - Onset of exothermic clad oxidation
    - Hydrogen production
      - Clad failure
      - Release of gaseous fission products

In-Vessel Accident Stages

- Onset of clad melting and fuel liquefaction
- Increased radionuclide release from fuel
- Relocation of molten core material into reactor vessel lower plenum
- Reactor vessel head lower failure
- Discharge of hot core debris (corium) into containment

Ref: Perspectives on Reactor Safety, NUREG/CR-6042, Rev. 2, SAND 93-0971, 2002, http://www.nrc.gov/reading-rm/doc-collections/nuregs/contract/cr6042/ (page 3.1-5)

Accident Initiator

- Loss of Offsite Power + other failures (Fukushima)
- Normal transient + other failures (TMI-2)
- Intentional safety violations (Chernobyl)
- TMI-2 and Chernobyl scenarios have been designed out of the possibility of reoccurrence; working on Fukushima

Start of Core Uncovery

- “Core uncovery”
  - Water level drops below the top of the core
  - Important because fuel is not expected to be damaged by overheating as long as it is covered

Start of Core Uncovery

- Coolant boil off in core region
  - Assume reactor is scrammed (usually is)
  - Core decay heat goes towards boiling coolant
  - As water level decreases
    - Less decay heat goes toward boiling water
    - More decay heat goes toward fuel heat up

Clad Oxidation

- Steam/water reaction becomes important at Tclad > 1000° C
  - Highly exothermic (6.5 MJ/kg of Zr reacting)
  - Produces hydrogen

$$Zr + 2H_2O \rightarrow ZrO_2 + 2H_2$$

Ref: Perspectives on Reactor Safety, NUREG/CR-6042, Rev. 2, SAND 93-0971, 2002, http://www.nrc.gov/reading-rm/doc-collections/nuregs/contract/cr6042/ (page 3.3-6)

Clad Oxidation

- Clad damage due to oxidation
  - Thermal stresses
  - Pressure stresses (Pin from fission gas → Pout)
    - Ballooning and rupture
  - Embrittlement, spallation of ZrO_2
    - Expose more metallic Zr to H_2O
  - During reflood (coolant addition), clad fractures

Clad Melting/Fuel Liquefaction

- Decay heat distribution determines heat up rate
  - Decay heat is proportional to thermal power during operation
- Clad melting
  - Zr-4 melts at about 1760° C
  - Ni-Zr eutectic (spacer-clad interaction) may melt earlier
  - Ag-In-Cd control rod material melts at 800° C
  - Stainless steel melts at 1450° C

Core Slumping

- Movement of molten fuel-bearing debris into the lower plenum
- Relocation is highly situation-specific
- TMI-2
  - 19 metric tons of molten core relocated
    - Total of about 150 metric tons of fuel in core
  - Crust failure appears to be near peripheral region of upper half of consolidated region

Vessel Breach

- Failure of reactor pressure vessel and discharge of fuel debris to the containment
- Possible failure mechanisms
  - Ablation of penetrations
  - Weakening of vessel head itself
  - Energetic interaction of molten fuel with residual coolant in lower plenum

In-Vessel Fuel-Coolant Interactions

- Contact of liquid water (coolant) with molten core materials
  - Quiescent boiling
  - Explosive corium fragmentation with rapid steam generation
    - Steam explosion
    - If melt contains unoxidized metals, exothermic metal-water reactions can take place
      - More energy release
      - Additional hydrogen generation

Ex-Vessel Phenomena

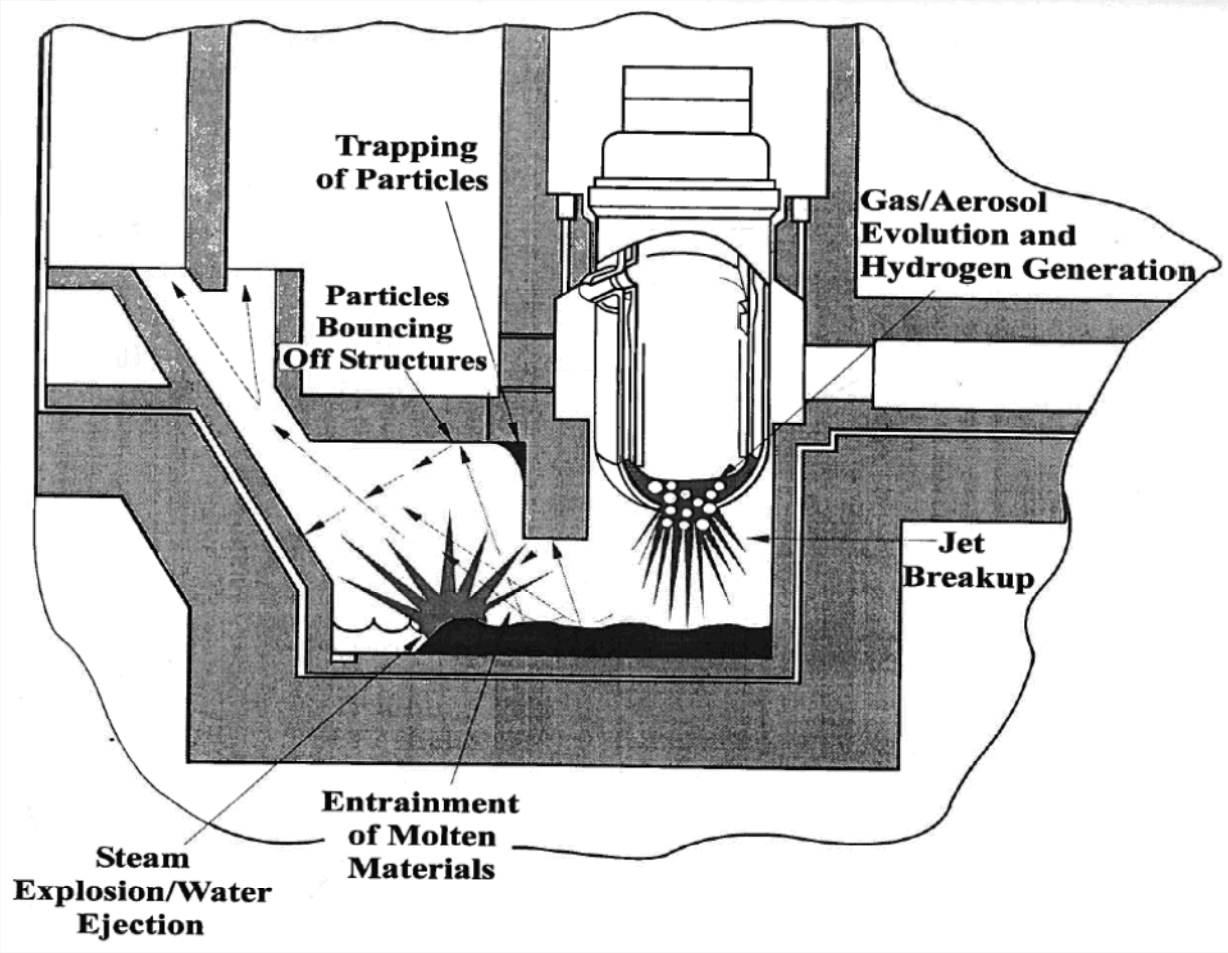
- Vessel fails and scenario progresses to the containment

Containment Strategies

- Containment design criteria
  - Design must ensure that dose guidelines are not exceeded for the most limiting accident identified in the Saftey Analysis Report
  - "…the containment structure and its internal compartments can accommodate, without exceeding the design leakage rate and with sufficient margin, the calculated pressure and temperature conditions resulting from any loss-of-coolant accident."

Ref: Perspectives on Reactor Safety, NUREG/CR-6042, Rev. 2, SAND 93-0971, 2002,
http://www.nrc.gov/reading-rm/doc-collections/nuregs/contract/cr6042/ (page 4.1-11)

Direct Containment Heating



Perspectives on Reactor Safety, NUREG/CR-6042, Rev. 2, SAND 93-0971, 2002, http://www.nrc.gov/reading-rm/doc-collections/nuregs/contract/cr6042/ (page 4.3-6)

Hydrogen Combustion

- TMI-2 experienced a 28 psig peak pressure due to H_2 combustion
  - Raised awareness of the potential threat
  - Threat of over-pressurization and damage to safety equipment
- $2H_2 + O_2 \rightarrow 2\, H_2O$ + energy
  - Release of 5.2 x104 BTU/lb mol of H_2 burned
  - Deflagration: subsonic combustion waves resulting in nearly steady state loads on the containment
  - Detonation: supersonic waves producing impulsive loads in addition to quasi-steady loads

Hydrogen Combustion

- Post-TMI-2 measures were implemented to address hydrogen combustion
  - BWR's are fitted with equipment such as hydrogen igniters or hydrogen recombiners
  - PWR's have larger containment volume
    - Focus is on avoiding local accumulations of hydrogen
    - Conditions for combustion locally are avoided
- Fukushima not equipped with these devices

Conclusions

- Transients are expected to happen during the reactor lifetime
  - Low consequence for safety
  - Reactor can usually override these events
- Design basis accidents are not expected to occur during the life of the reactor
  - Reactors are designed to be able to achieve a safe end-state following a design basis accident
- Severe accidents are highly unlikely
  - Less than once every 100,000 years of reactor life
  - Safety consequences are large
  - Large amount of research and countermeasures

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Assignments, Activities, and Exercises

- Read through the handouts
  - Distinguish between transients, design basis accidents and severe accidents, noting the severity (or lack of) in the outcome
  - Provide examples of LWR transients
  - Based on your knowledge of severe accident phenomena, explain why they have a very low frequency of occurrence