ELECTRIC POWER RESEARCH INSTITUTE

Engineering Fundamentals CBT

Printout of CBT Content for Reference Purposes Only

Reference CBT:

Core Protection V 1.0

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PRODUCT DESCRIPTION

Summary

This document provides a printout of the CBT content for use as a reference document only. Students are encouraged to use the CBT as animations, flash video, and interactive features are intended to enhance their learning experience.

Abstract

The Core Protection module of Engineering Fundamentals provides a basic overview of this topic applicable to all engineering disciplines beginning their career in the nuclear power industry.

This module covers basic terms and concepts, and methods used to ensure core protection in nuclear power plants. This course will help new engineers understand the importance of core protection, high equipment reliability and system integrity. This module is intended for use as orientation training for new engineering support personnel.

Platform Requirements

Windows™ 2000/XP

Application, Value and Use

The Core Protection module:

- Allows engineering support personnel to review the content when they desire and at their own pace
- Uses interactive features and graphics to illustrate key concepts & enhance training.

Keywords

Training

Engineering Fundamentals

Core Protection

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CONTENTS

1
INTRODUCTION TO CORE PROTECTION

Introduction

The primary safety concern for nuclear power plant operation is preventing significant radiation exposure to the public during routine and accident conditions. This is achieved by employing fission product barriers and engineered safety features.

To protect the public, it is very important to keep the reactor core covered, cooled, and properly reactive. Failure to achieve this goal can be very detrimental to the entire nuclear industry as well as costly in monetary terms, as demonstrated by the Three Mile Island (TMI) accident. However, despite being the worst US commercial nuclear accident in history, the TMI accident also demonstrated that the core protection design limited the impact to the public.

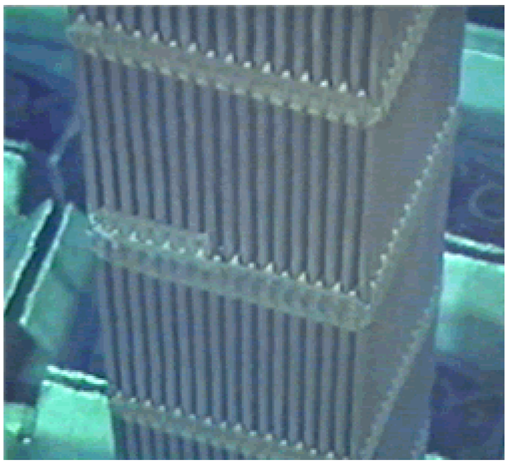
After completing this lesson, you will be able to:

- Identify ways a core can fail
- Define defense-in-depth
- Define a design basis accident
- Explain how defense-in-depth and the plant protection design philosophy are used to protect the reactor

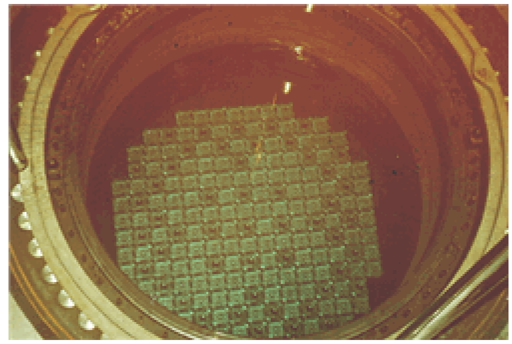
Prerequisite: The Basic Atomic and Nuclear Physics course should be successfully completed prior to taking this course.

Core Geometry

The reactor vessel (RV) is the pressure vessel that contains the fuel and the internal components used to cool the fuel and control the fission process. The fuel is composed of uranium dioxide powder (UO_2) configured in ceramic pellets. These pellets are installed in tubes or cladding, which are sealed and pressurized with helium gas to create fuel pins. The fuel pins are mounted in grids that hold them in place and allow coolant to flow around the fuel. These groupings of pins are called fuel assemblies. There are numerous pins in an assembly and numerous assemblies in a core.



Core Failure



Any loss of core geometry, particularly into a configuration where cooling becomes impractical, is considered a core failure. The primary cause of core failure is an inability to fully remove the heat created by the fuel. This may be due to:

- A failure of the heat removal systems
- An increase of heat created beyond the ability of the heat removal systems
- The inability to transfer heat from the fuel

The following are some examples of these types of failures:

Loss of Cooling Capabilities	Excessive Heat Creation	Inability to Transfer Heat from the Fuel
• Loss of reactor coolant flow • Loss of coolant • Loss of feedwater flow • Loss of decay heat removal	• Failure of the reactor to trip when required (often referred to as anticipated transient without scram (ATWS)) • Rod ejection (PWR) • Inadvertent deboration (PWR)	• Loss of subcooling • Departure from nucleate boiling • Onset of transition boiling

Any of the above could lead to core melt and a loss of core geometry. Nuclear plants are designed to avoid these events. Methods of prevention will be discussed in greater detail in upcoming lessons.

Review Question 1

Which of the following types of failures could result in core failure?

A. Loss of cooling capabilities

B. Uranium dioxide release

C. Excessive heat creation

D. Helium gas pressurization

E. Inability to transfer heat from the fuel

The correct answers are A, C, and E. Loss of cooling capabilities, excessive heat creation, and the inability to transfer heat from the fuel are all types of core damage.

Review Question 2

What is the primary cause of core failure?

A. Inability to remove heat from the core

B. Core melt

C. Loss of core geometry

D. Uranium dioxide powder

The correct answer is A. The primary cause of core failure is the inability to remove heat from the core.

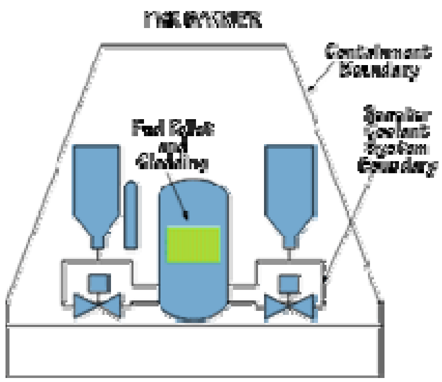
Fission Product Barriers

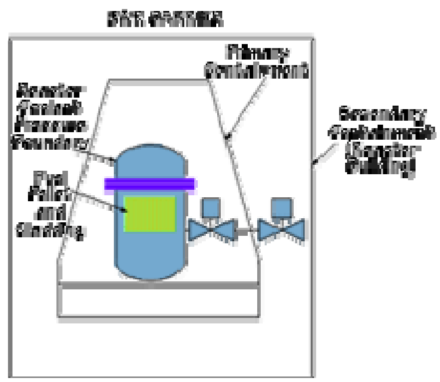
Protecting the public depends on preventing fission products from escaping the plant. There are three fission product barriers: the fuel pellet and cladding; the Reactor Coolant System (RCS) pressure boundary; and containment.

These barriers work in the following ways:

- Fuel pellet and cladding: Prevents the escape of fission products from cladding
- RCS pressure boundary: Contains radionuclides that escaped the cladding or were produced outside the cladding
- Containment: Holds radionuclides that have escaped from the RCS

The images below illustrate fission product barriers in PWR and BWR plants.





Core Melt Accidents

One way to gauge an accident's severity is to determine if core melt occurred. Two examples of core melt accidents are the Three Mile Island (TMI) and Chernobyl (pictured at right) accidents.

At TMI, safety systems were intentionally defeated prior to and during the event contributing to the core melt. There was a loss of core geometry, radionuclides were released to the containment, and hydrogen was created. Although safety features were defeated, containment, the last line of defense, stayed intact and the public was not affected by elevated radiation levels.

Chernobyl also defeated safety features and had a severe core melt. However, the Chernobyl design did not have a containment structure comparable to the U.S. reactors. The absence of an adequate containment structure allowed the radionuclides released from the fuel and RCS to go directly to the atmosphere where thousands of individuals in the public were affected, including fallout on other continents.

Defense-in-Depth

The nuclear industry has adopted a defense-in-depth policy to ensure the core stays covered, is properly reactive, and that adequate heat removal is provided.

The defense-in-depth policy states that a utility should:

- Design to prevent the occurrence of nuclear accidents
- Assume that accidents will occur
- Provide proven capability to meet the range of worst-case accidents

Review Question 3

Which of the following does a defense-in-depth policy state a utility should do?

A. Design to the most frequent common accident

B. Design to prevent the occurrence of nuclear accidents

C. Assume that accidents will occur

D. Provide proven capability to meet the range of worst-case accidents

E. Provide the public with literature about nuclear power plant operation

The correct answers are B, C, and D. A defense-in-depth policy states that a utility should design to prevent the occurrence of nuclear accidents, assume that accidents will occur, and provide proven capability to meet the range of worst-case accidents.

Regulations

The federal government has created several regulations in the code of federal regulations (CFR) to control nuclear power plants. Some of these regulations are listed below.

Click each regulation below to learn more.

10CFR20
Controls the receipt, possession, use, transfer, and disposal of licensed material by any licensee in such a manner that the total dose to an individual does not exceed the standards for protection against radiation prescribed in the regulations in this part.

10CFR50
Governs the licensing of domestic nuclear production and utilization facilities. It also gives notice to all persons who knowingly provide components, equipment, materials, or other goods or services that relate to activities subject to this 10CFR50.

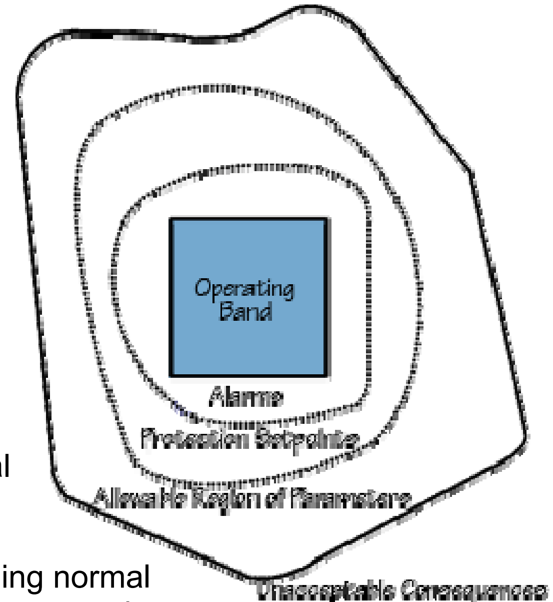
10CFR50.46
Specifies that peak cladding temperature is < 2200°F, local cladding oxidation is < 17% of cladding thickness, and hydrogen generation by cladding oxidation is < 1% available from whole core cladding. Also specifies that core configuration is maintained in a coolable geometry and that it must demonstrate the ability to maintain long term core cooling.

10CFR50 Appendix A
Consists of 55 criteria (numbered through 64 with omissions) arranged in six categories as follows:

- Overall requirements
- Protection by multiple fission product barriers
- Protection and reactor control systems
- Fluid systems
- Reactor containment
- Fuel and radioactivity control

10CFR100
Specifies that a person at any point on the boundary of the exclusion area will not receive a whole body dose in excess of 25 rem or a dose to the thyroid in excess of 300 rem for two hours following an accident.
Radioactive cloud (cloud passage) will not cause a person on the outer border of the area surrounding the utility-controlled property (low population zone (LPZ)) to receive a whole body dose in excess of 25 rem or a dose to the thyroid in excess of 300 rem.

Plant Protection Design Philosophy



The philosophy used when designing a plant is to add several layers of protection. The first of these is specified in the Final Safety Analysis Report (FSAR). Generally, but not always, the FSAR makes the following assumptions:

- The plant is operating in a given band via proper control.
- The plant is assumed to be in steady state prior to the transient.
- The plant is being operated within Technical Specifications.

The first level of protection is provided by maintaining normal plant operation within these conditions or limits (shown as the Operating Band in the figure at right). Plant alarms are set to alert the operator of a developing off-normal condition, such as a failure of one or more of the control systems. The operator backs up the control systems to ensure the previously mentioned assumptions are maintained.

If the plant exceeds specified parameters, the next level of protection is provided by automatic systems. When the plant exceeds certain limits (protection setpoints), these systems act to terminate operations by tripping the reactor. In the event of an accident, additional systems actuate to provide core cooling, negative reactivity insertion, and, if necessary, containment cooling. These systems are designed to ensure the plant systems stay within the equipment's safety limits, which are depicted in the graphic as the allowable region of parameters.

Plant operators provide the final level of protection. In the event of any off-normal event or accident, the operator will ensure the automatic systems are properly functioning. If the automatic systems fail, the operator will manually take action to trip the reactor and provide core cooling.

Design Basis Accidents

The systems in nuclear power plants are designed to mitigate specific accidents. Accidents are defined as hypothesized events that affect one or more of the radioactive material barriers and that are not expected during plant operations. The accident types considered are:

- Mechanical failure of a single component leading to the release of radioactive material from one or more barriers. The components referred to here are not those that act as radioactive material barriers. Examples of mechanical failure are breakage of the coupling between a control rod drive and the control rod, and failure of a spring used to close an isolation valve.
- Arbitrary rupture of any single pipe up to and including complete severance of the largest pipe in the nuclear system process barrier: This type of accident is considered only under conditions in which the nuclear system is pressurized.

By definition, these are the plant's design basis accidents (DBAs). A DBA is typically a worst-case accident of interest. The consequences of DBAs often bound the consequences of other accidents. For example, the DBA large break, Loss of Coolant Accident (LOCA) bounds LOCAs with slightly smaller breaks.

Design Basis Accidents: FSAR Chapter 15

The design for nuclear power plants is documented and approved by the Nuclear Regulatory Commission (NRC) in the form of the plant's Final Safety Analysis Report (FSAR). Chapter 15 of this report documents the DBAs for that plant.

Examples of DBAs	
BWR	**PWR**
LOCA: sudden, complete, circumferential shear of a recirculation pump suction line with simultaneous loss of normal auxiliary power and feedwater flow	LOCA: double-ended rupture of reactor coolant piping with simultaneous loss of off-site power
Main steam line break outside dry well	Steam Generator (SG) tube rupture
Instrument line break	Main steam or feedwater line break
Control rod drop	Control rod drive mechanism rupture (rod ejection and small break LOCA)
Misplaced fuel assembly	Reactor Coolant Pump (RCP) locked rotor
Fuel handling accident	Fuel handling accident
Spent fuel cask drop accident	
Reactor recirculation pump shaft break or seizure	
Radioactive release due to system or component failure	

Review Question 4

When designing a plant, the philosophy documented in the Final Safety Analysis Report (FSAR) contains several layers. Which of the following assumptions are used to ensure core protection?

A. The plant is operating in a given band via proper control.

B. The plant is assumed to be in steady state prior to the transient.

C. The plant is alerting the public when a test is required.

D. The plant is creating regulations that govern how the plant operates.

E. The plant is being operated within Technical Specifications.

The correct answers are A, B, and E. The Final Safety Analysis Report helps ensure core protection by making the following assumptions: the plant is operating in a given band via proper control, the plant is assumed to be in steady state prior to the transient, and the plant is being operated within Technical Specifications.

Review Question 5

A design basis accident is typically a worst-case accident of interest.

True or False?

Answer: A design basis accident is typically a worst-case accident of interest. It is the specification to which a plant is designed.

Conclusion

The primary safety concern for nuclear power plant operation is the protection of the public from the release of fission products. The first line of defense is keeping the core covered, cooled, and properly reactive. In order to do this, the plant is designed using defense-in-depth. This policy realizes that accidents can happen and designs the plant to mitigate them.

The systems of the plant are designed with several layers of protection including automatic controls, alarms, and automatic protective systems. Nuclear power plants are designed to mitigate design basis accidents (DBAs), which are severe enough that they bound the consequences of other less severe accidents. The NRC approves the DBAs and the method the plant will use to mitigate these accidents when they approve the plant’s FSAR. The focus of accident mitigation is protecting the reactor core. The remaining lessons in this course will describe how these various systems work together to protect the core, thereby protecting the public.

Now that you have completed this lesson, you can:

- Identify ways a core can fail
- Define defense-in-depth
- Define a design basis accident
- Explain how defense-in-depth and the plant protection design philosophy are used to protect the reactor

In the next lesson, we will discuss methods used to design nuclear power plant systems: the deterministic and the probabilistic methods.

2
PROBABILISTIC RISK ASSESSMENT

Introduction

The purpose of this lesson is to introduce Probabilistic Risk Assessment (PRA) as it relates to core protection, and explain the differences between PRA and Deterministic methods. Engineers need to be able to identify that modifications to plant systems, procedures, etc. will impact PRA, and should consider PRA results in evaluating system modifications and procedure changes to support engineering work.

After completing this lesson, you will be able to:

- Describe the attributes of PRA
- Differentiate between probabilistic and deterministic analysis
- Identify how PRA supports various engineering activities, including examples

Probabilistic Risk Assessment

Probabilistic Risk Assessment (PRA), also known as Probabilistic Safety Assessment (PSA), is a quantitative assessment of the risk associated with plant operation and maintenance. This risk is measured by how often different events that lead to severe core damage occur. The original goal of PRA was to analyze and understand severe accident behavior, and then identify and fix the plant vulnerabilities.

Some of the attributes of a PRA are:

- It realistically models plant design, plant procedures, and human performance
- It is a best estimate
- It is not a safety-related tool
- It is a mathematical model of all important plant systems using plant-specific data for critical components
- It models the plant response for various initiating events using calculated system reliabilities
- It takes into account realistically usable mitigating equipment

PRA takes into account internal initiating events such as a LOCA and station blackout. Some plant PRAs will also include external events such as earthquakes, flooding, or a hurricane. This leads to a calculated probability of a specific event that leads to core damage.

To quantitatively measure risk, two parameters are calculated: Core Damage Frequency (CDF) and Large Early Release Frequency (LERF). The NRC's minimum safety goal for CDF and LERF is 1E-4 and 1E-5, respectively. This means, for these values, the probability of core damage for a given plant would be 1 in 10,000 years of reactor operation for CDF, and 1 in 100,000 years of reactor operation for LERF. Most plant CDFs and LERFs are typically an order of magnitude smaller than these goals, decreasing their probability of reactor core damage.

Review Question 6

Which of the following describes the attributes of a PRA?

A. It realistically models plant design, plant procedures, and human performance.

B. It is based on Technical Specifications.

C. It is a safety-related tool.

D. It is a mathematical model of all important plant systems using plant-specific data for critical components.

E. It models the plant response for various initiating events using calculated system reliabilities.

The correct answers are A, D, and E. A PRA realistically models plant design, plant procedures, and human performance, is a mathematical model of all important plant systems using plant-specific data for critical components, and models the plant response for various initiating events using calculated system reliabilities.

The original basis for plant design and licensing uses a deterministic approach.

Several conservative (worst case) assumptions are used to set the bounds for the design:

- Worst case single failure of equipment
- Non-safety equipment not available during an accident
- Equipment providing no capability if not fully operable

Licensing Basis

Licensing Basis analyses are based on conservative assumptions. They use Technical Specification's Limiting Condition for Operation (LCO) as a baseline for plant configuration to mitigate the various design basis events that will be described later. This plant design basis along with the core protection design concepts predict whether or not the plant will be able to mitigate the accident.

PRA

On the other hand, PRA is based on realistic analyses. Success criteria in PRA uses all available methods and equipment to satisfy mitigation of the events that can lead to core damage, and thus provides a quantitative analysis to predict success.

Deterministic vs. Probabilistic Table

The following table describes the differences between deterministic analyses and PRAs.

Deterministic	Probabilistic
Used in design and licensing	Used in licensing
YES or NO answer	Estimates the likelihood of the YES/NO answer
Uses conservative techniques	Uses realistic techniques
Evaluates worst case single failure	• Considers the Probability of a Failure • Considers Multiple Failures
Assumes non-safety related equipment is not available	Credits all potentially available equipment. Any possible plant configuration may be included.
Adverse consequence is typically defined as cladding damage	Adverse consequence defined as Core Damage that is generally taken as loss of structural integrity of the fuel
Operable	Functional

Deterministic vs. Probabilistic Accident Analysis: Examples

The following is an example of how deterministic and probabilistic analyses are applied in evaluating simultaneous maintenance on different components in a nuclear power plant.

Deterministic

In deterministic analysis, simultaneous maintenance on an auxiliary feedwater pump, an emergency diesel generator, and switchyard breakers could show that the core would be protected for design basis accidents.

PRA

In PRA, simultaneous maintenance on those same pieces of equipment would show a significant increase in the CDF and could make this a configuration beyond acceptable risk. Plant operators and plant management would use the increased CDF value to make a conservative decision and not allow all three maintenance activities to take place at the same time, if possible, or would take compensatory actions such as placing backup equipment in service or reducing reactor power to gain back some PRA margin, thereby managing the risk.

Review Question 7

A key difference between deterministic analysis and PRA is that deterministic analysis is based on conservative assumptions, while PRA uses realistic analyses based on all available methods and equipment.

True or False?

Answer: Licensing Basis analyses are based on conservative assumptions and use Technical Specification's Limiting Condition for Operation (LCO) as a baseline for plant configuration to mitigate the various design basis events. On the other hand, PRA is based on realistic analyses. Success criteria in PRA uses all available methods and equipment to satisfy mitigation of the events that can lead to core damage, and thus provides a quantitative analysis to predict success.

How PRA Supports Engineering Work

PRA is used extensively to predict the potential impact of:

- Plant modifications
- Plant configuration changes
- License changes
- Equipment unavailability

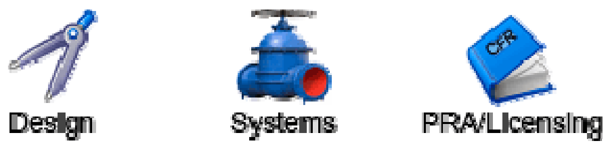
PRA models rank components and systems in their order of risk/importance to the plant. This modeling may be used to evaluate the risk to the plant, such as for changing plant configurations or performing maintenance on-line rather than during an outage. Engineers can also use this information to aid them in various work activities.

PRA is a tool used to evaluate risk/safety significance and is useful when making decisions regarding plant operation. However, while it models the design and operation of the plant and is required by the Maintenance Rule (10CFR50.65) Program, it is not intended to constitute a design or licensing basis analysis. An item defined as low-risk significant by the PRA might represent a significant regulatory issue. It is the responsibility of those using the PRA results to identify any non-PRA requirements that may apply to their specific situations.

In addition to the plant and site engineering using PRA, the Nuclear Regulatory Commission (NRC) also uses PRA as part of the NRC Reactor Safety Cornerstone in the Significance Determination Process and their performance index, the Mitigating Systems Performance Index (MSPI).

Engineers can use PRA during their everyday activities.

Click each icon below to see some examples of how various engineering groups can use PRA. (Note: These activities for use of PRA may be different at your plant.)



Design Engineer
Use PRA when:

- Installing new safety-related equipment to the plant
- Adding a backup cooling supply to a piece of safety-related equipment
- Changing control room indications
- Replacing an AOV with an MOV
- Changing failure mode of equipment

Systems Engineer
Use PRA when:

- Supporting justifications for continued operation
- Incorporating Maintenance Rule (10CFR50.65)
- Using the Mitigating Systems Performance Index (MSPI)
- Determining risk factors for on-line work activities
- Providing input for plant testing

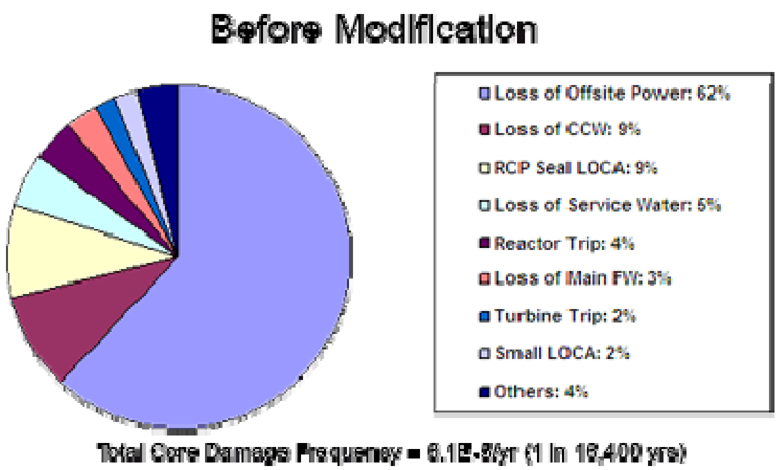
Licensing/PRA Engineer
Use PRA when:

- Using the Significance Determination Process
- Using the Mitigating Systems Performance Index (MSPI)
- Using the Regulatory Oversight Process
- Providing basis for Equipment out of Service (EOOS)
- Providing basis for risk-informed Technical Specifications

How PRA Supports Engineering Work: Example

The following illustration shows how PRA is used to help evaluate if a potential plant modification will benefit reactor safety.

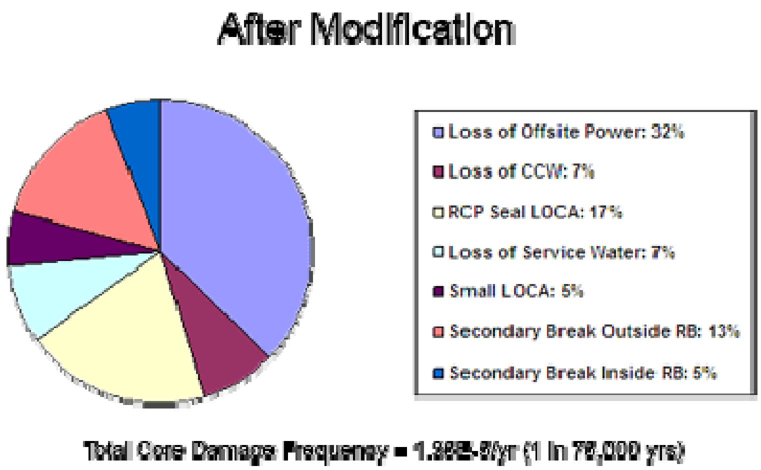
At XYZ plant, the CDF was 6.1 E-5/year, or 1 CDF event for every 16,400 years of reactor operation. The initiating events that can lead to a core damage event, and their relative percentage as a contributor to CDF, are listed next to the chart. For example, accidents that start with a Loss of Service Water contribute 5% to the overall CDF number.



How PRA Supports Engineering Work: Example, continued

A plant modification was installed that added an additional Emergency Safeguards Feature (ESF) AC power source. This additional AC source provided backup power in loss of ESF AC Power situations, thereby adding additional success paths for reducing plant events that lead to core damage. The CDF was reduced by a factor of 4.5 due to this installation.

Qualifying this value, the modification decreased the chance of any event causing core damage from 1:16,400 to 1:75,000 years. The plant's reactor is now 4.5 times safer. Looking at the plant initiating events, a Loss of Service Water now contributes 7% to CDF, but since the CDF is smaller, the chance of this event causing core damage is reduced due to the plant modification.



Review Question 8

Plant modifications have no affect on a plant's CDF.

True or False?

Answer: False. Any plant modification can have an effect on a plant's CDF. For instance, adding an additional power source to mitigate the loss of off-site power can greatly reduce the chance that a core damage event will occur at the power plant.

Review Question 9

Match each type of engineer with his or her possible use of a PRA.

Design Engineer A. Installing new safety-related equipment to the plant

Systems Engineer B. Providing basis for risk-informed Technical Specifications

Licensing/PRA Engineer C. Incorporating Maintenance Rule (10CFR50.65)

The correct matching sequence is ACB.

Conclusion

PRA is a tool that measures the quantitative risk associated with plant operation and maintenance. Although it is not the method used for a plant's design basis, it is very important when analyzing changes to plant operations and configuration. Engineers will use PRA in many aspects of their jobs, from system modifications to system operation and testing, to determine the impact of these changes as they relate to core protection.

Now that you have completed this lesson, you can:

- Describe the attributes of PRA
- Differentiate between probabilistic and deterministic analysis
- Identify how PRA supports various engineering activities, including examples

In the next lesson, we will examine a core's limits and the conditions that could lead to its failure, as well as the protections in place to prevent failure.

3
SAFETY LIMITS AND MARGINS

Introduction

In the first lesson, we discussed the importance of core protection, and the regulations that govern the protection of cores in industrial nuclear power production. In the second lesson, we discussed the methods used to design protection systems: deterministic, which is used for the majority of original plant design, and probabilistic, which analyzes the equipment or failure paths most likely to occur.

In this lesson, we will examine a core's limits and the conditions that could lead to its failure. Then, we will discuss the layers of control methods, how they are used for system design, and how they apply to core protection.

After completing this lesson, you will be able to:

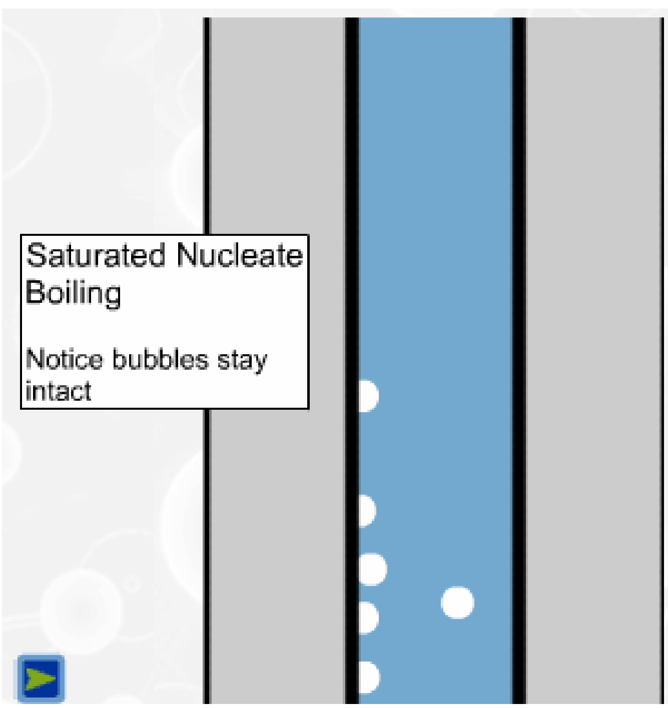
- Define the following terms: nucleate boiling, subcooled nucleate boiling, bulk (saturated nucleate) boiling, departure from nucleate boiling, critical heat flux, minimum critical power ratio, onset of transitional boiling, and film boiling
- Define safety limits
- Define limiting conditions for operation (LCOs)
- State the hierarchy of limits on safety parameters

Boiling: Subcooled Nucleate Boiling

In order to maintain the core's integrity, the departure from nucleate boiling (DNB) or onset of transition boiling (OTB) should be avoided. First, we will define departure from nucleate boiling.

During normal power operations, the fuel cladding temperature is higher than the coolant saturation temperature. Once the energy in the cladding is high enough, steam bubbles will begin to form on the cladding. As these bubbles form, they break away from the cladding wall and condense back into water. This phenomenon can be observed when boiling water on the stove. At the onset of boiling, bubbles form on the bottom of the pan and break loose; but, as they rise through the water, they condense before reaching the surface. This is called subcooled nucleate boiling.

Click the arrow at right to see what happens when subcooled nucleate boiling, saturated nucleate boiling, and film boiling occur. (Note: This will not work in this Word document.)



Departure from Nucleate Boiling

As heat energy is added, the bubbles form at a faster and faster rate. When the temperature of the coolant reaches saturation temperature, the bubbles will form and release from the cladding, but they will not immediately condense. Instead, they will combine to form larger bubbles that travel with the flow. This is called bulk or saturated nucleate boiling.

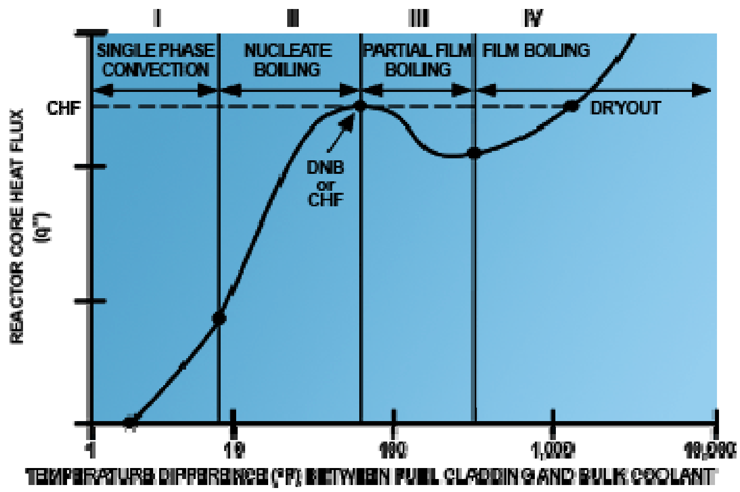
As long as the bubbles break loose from the wall as they form, there is an efficient transfer of heat from the wall to the coolant. Once heat flux increases to the point where bubbles form faster than they can be removed, a film of steam starts forming on the clad surface, known as film boiling. Heat transfer through this film layer is poor, so the overall ability to remove heat from the cladding, and therefore the fuel, drops drastically. This is considered departure from nucleate boiling (DNB).

DNB must be avoided. The proximity of the coolant to DNB is measured by a quantity called the Departure from Nucleate Boiling Ratio (DNBR). It is defined as:

$$\text{DNBR} = \frac{\text{DNB Heat Flux predicted by applicable correlation}}{\text{Actual Local Heat Flux}}$$

The heat flux, which produces this rapid drop in heat transfer ability, is called the critical heat flux (CHF). When CHF is reached, the core is said to be experiencing DNB. If this continues for any length of time, the fuel and cladding will overheat. This is a local condition dependent on the local heat flux at the clad surface.

The figure below shows a representation of the CHF versus ΔT, the difference between the surface and liquid bulk temperature. The curve is for a pool boiling with no flow. Note that the scaling on the graph is a log scale.



Onset of transition boiling (OTB) in a BWR is equivalent to departure from nucleate boiling in a PWR. OTB results in a decrease in heat transfer from the cladding, elevated clad temperature, and the increased probability of clad failure. The margin to boiling transition can be calculated from plant operating parameters such as core power, core flow, feedwater temperature, and core power distribution.

The margin for each fuel assembly is characterized by the Critical Power Ratio (CPR), which is equal to the bundle power at OTB divided by the actual bundle power, as shown in the equation below.

$$CPR = \frac{\text{Bundle Power}_{OTB}}{\text{Bundle Power}_{Actual}}$$

The minimum value of this ratio for any bundle in the core for each class of fuel is called the Minimum Critical Power Ratio (MCPR). As seen in the mathematical expression for CPR, we want to keep the MCPR > 1.0 such that OTB is never experienced anywhere in the core.

Review Question 10

Match each term with its definition.

Subcooled nucleate boiling	A. The point where the steam produced by the fuel creates a film on the heat surface or fuel cladding causing a drastic drop in the heat removal capability of a liquid
Bulk (saturated) nucleate boiling	B. At onset of boiling, bubbles form on the heated surface and break loose; but then, as they flow through the water, they condense.
Film boiling	C. The point when the water is at saturation so the bubbles don't immediately condense.

The correct matching sequence is BCA.

Review Question 11

Match each term with its definition.

Departure from nucleate boiling	A. The point where the steam produced by the fuel creates a film on the heat surface or fuel cladding causing a drastic drop in the heat removal capability of a liquid.
Critical heat flux	B. Any bundle in the core will be approaching the onset of transition boiling as this value approaches 1.
Minimum critical power ratio	C. The flux at which departure from nucleate boiling (DNB) occurs. This is a local condition and depends on the flux at the clad surface.

The correct matching sequence is ACB.

Safety Limits

Safety limits are limits on the specific nuclear process variables imposed by technical specifications (the actual values at your plant may differ). These limits are set to ensure the integrity of the fission product barriers.

Examples of safety limits for a BWR are:

- Maintain Thermal Power less than 25% Reactor Thermal Power (RTP) with low pressure (< 785 psig) and low core flow (<10%)
- Minimum Critical Power Ratio (MCPR) greater than the limit with higher pressure (> 785 psig) and core flows greater than 10%
- Reactor water level greater than the top of active irradiated fuel
- Reactor steam dome pressure less than the limit (< 1325 psig)

Examples of safety limits for a PWR in modes 1 or 2 are:

- Fuel pin centerline temp < 4642°F
- Departure from Nucleate Boiling Ratio (DNBR) > 1.18 (this value is calculated according to the specific fuel assemblies used)
- RCS pressure < 2750 psig

Limiting Conditions for Operation (LCO) specify the minimum acceptable levels of system performance necessary to assure safe operation of the facility. A typical LCO has four major sections.

Click each LCO section below to learn more about it.

Statement
This is the requirement. An example statement is: "A recirculation loop Flow Control Valve (FCV) shall be OPERABLE in each operating recirculation loop."

Applicability
This section lists the reactor modes (defined elsewhere) when the requirements of the LCO must be met.

Actions
- Actions that must be met
- The time limits for completing those actions when limited conditions of operation (LCOs) are not satisfied

Surveillance Requirements
Surveillance requirements are testing or verification activities performed on a specified frequency that ensure the LCO requirements continue to be met.

An example LCO statement would be that two high pressure injection (HPI) pumps must be operable whenever the plant is in modes 1 or 2 (when the reactor is critical). The applicability is modes 1 and 2.

The action statement requires that if only one pump is operable, the plant must begin reducing power within 72 hours. The action statement further requires that if no HPI pumps are operable, the unit must begin reducing power in one hour.

An example surveillance requirement is "the automatic start on HPI pumps must be tested every 18 months."

Review Question 12

Safety limits are set to ensure the integrity of the fission product barriers.

True or False?

Answer: True. Safety limits are set to ensure the integrity of the fission product barriers. They are limits on the specific nuclear process variables imposed by technical specifications.

Review Question 13

Limiting Conditions for Operation (LCO) specify the maximum acceptable levels of system performance necessary to assure safe operation of the facility.

True or False?

Answer: False. Limiting Conditions for Operation (LCO) specify the minimum acceptable levels of system performance necessary to assure safe operation of the facility. A typical LCO has four major sections: Statement, Applicability, Actions, and Surveillance Requirements.

Levels of Protection

All plant equipment is designed to survive specific conditions. For tanks or piping, this could be pressure, temperature, and substance contained. For example, if a tank is designed for 2500 psig, then the tank is expected to withstand at least 2500 psig. The actual point at which the tank will rupture is above this pressure. When a system as complicated as the RCS is analyzed, the design pressure should be the pressure limit of the weakest component in that system. When considering plant equipment, there are several layers of protection for the safety-significant parameters.

Click each layer of protection's label to learn more about it.

Automatic controls
Most plant parameters have controls that should keep them within established setpoints.

Alarms/operator action
Typically there will be alarms to let the operator know the parameter has deviated from where it should be maintained by the automatic controls. This will allow the operators to take action before the trip or relief setpoint if it is required.

Automatic trips/relief
If safety significant parameters deviate too far from normal, the equipment can take two actions to protect itself. In the case of over temperature, voltage, pressure, etc. the equipment can automatically trip to stop the addition of more energy. In the case of pressure, the equipment can additionally relieve the energy through the use of relief valves. It is possible for the equipment to be designed to either relieve or trip first. Then if the condition doesn't improve, it could perform the other.

Emergency operating procedures/Operator action
The operators are scanning the safety significant parameters to make sure they are not moving too far out of the setpoints without action being taken. If the parameter passes the point where an automatic trip or relief should have taken place, operating procedures will have the operators take action.

Technical specification limit
All alarms or actions taken before this point should prevent the equipment from reaching the tech spec limit. The plant is analyzed to survive a design basis accident with the parameter less than that value. This is the plant's licensing basis.

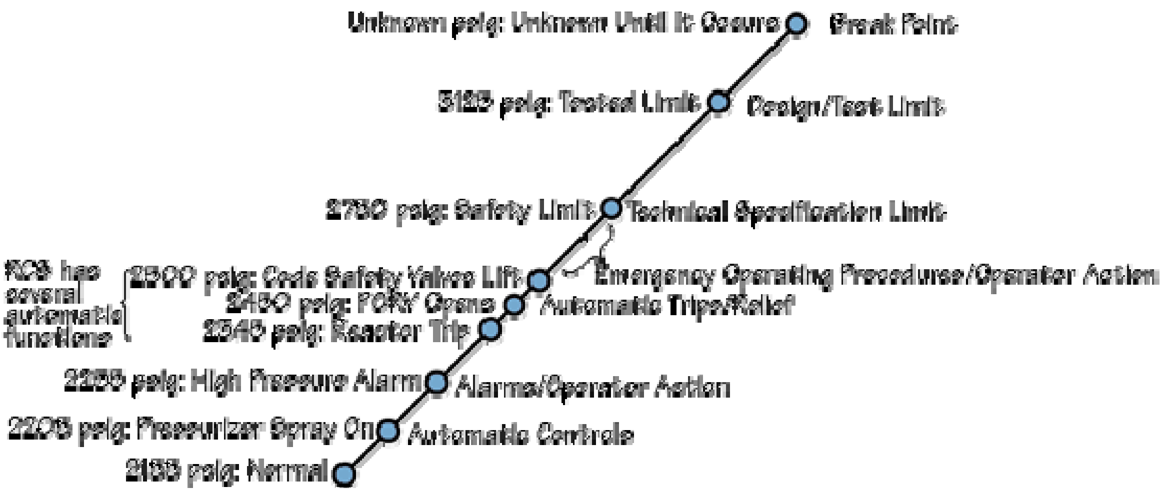
Design/test limit
This is the highest value for a specified parameter that a piece of equipment is designed to survive and remain operational. In some cases, this is a value where the equipment was tested.

Break point
The point where the equipment actually fails is somewhere above the design limit. There is no way to know this value without an actual failure.

PWR RCS Example

One example of how various layers of protection are used to protect the core is when high RCS system pressure occurs. Pressure in the RCS is controlled with a pressurizer, which is a tank that uses heaters to increase pressure and a spray to decrease pressure as needed. Shown below is the sequence of events as RCS pressure increases. The values shown are examples only and will vary among plants.

Click each pressure value below to learn more about what happens during its event.



2155 psig
Normal pressure. This pressure is maintained by variable pressurizer heaters.
2205 psig
Pressurizer spray on. The pressurizer spray will come on attempting to collapse the pressurizer steam bubble and reduce pressure.
2255 psig
High pressure alarm in control room. If the operator can take normal action, he or she will.
2345 psig
Reactor trip. This is an attempt to remove the addition of more energy. Even if the reactor trips, thermal inertia can still increase pressure.
2450 psig
The Power Operated Relief Valve (PORV) opens. The PORV is the first extraordinary action to remove pressure.
2500 psig
Code safety valves lift. These valves are completely mechanical and offer another layer of protection.
2750 psig
This is the technical specification limit and operator actions will be taken to limit pressure to lower values.
3125 psig
Tested limit. This is the pressure used for the hydrostatic test prior to startup.

Protecting the Fission Product Barriers

In a nuclear power plant, automatic systems are designed to protect the barriers (fuel pellet and cladding, reactor coolant system, and containment) to prevent the release of fission products. These systems can perform several functions, one of which is to trip the reactor if there is any indication of conditions that are detrimental to the core. The monitored conditions may include any that could damage the system or indicate an approach to DNB or the onset of transition boiling (OTB) in the core.

This system protects the core, but other systems are designed to protect the reactor coolant system and containment. These systems will mitigate a loss of reactor coolant pressure and inventory. As the reactor coolant pressure drops, systems would align to add inventory at pre-arranged setpoints in an attempt to replenish the coolant. At the same time, systems would be initiated to cool containment and condense any resultant steam.

These systems align and start automatically to protect the public.

Review Question 14

Match each level of protection to its description.

Automatic controls	A. The operators are scanning the safety significant parameters to make sure they are not moving too far out of the setpoints without action being taken.
Emergency operating procedures/Operator action	B. Most plant parameters have controls that should keep them within established setpoints.
Automatic trips/relief	C. If safety significant parameters deviate too far from normal, the equipment can take two actions to protect itself: trip or relieve.
Design/test limit	D. The highest value that piece of equipment is designed to survive and remain operational. In some cases, this is a value where the equipment was tested.

The correct matching sequence is BACD.

Conclusion

In this lesson, we discussed the mechanisms of boiling and the importance of avoiding the departure from nucleate boiling or the onset of transition boiling in the core. We also gave examples of the safety limits and protection theories used to design systems and protect the fission product barriers for nuclear power plants, which include the commitments made during the licensing process and the systems designed to control, alarm, and activate during emergencies.

Now that you have completed this lesson, you can:

- Define the following terms: nucleate boiling, subcooled nucleate boiling, bulk (saturated nucleate) boiling, departure from nucleate boiling, critical heat flux, minimum critical power ratio, onset of transitional boiling, and film boiling
- Define safety limits
- Define limiting conditions for operation (LCOs)
- State the hierarchy of limits on safety parameters

Now that you have learned about the basics of boiling mechanics and core protection, you will learn about the methods used to keep the core cooled and safe in the next lesson.

4
CORE COOLING MECHANISMS

Introduction

The purpose of this lesson is to describe core cooling mechanisms used in PWRs and BWRs during normal and accident conditions. This lesson will help lay the foundation for understanding system response during design basis accidents in the next lesson.

After completing this lesson, you will be able to:

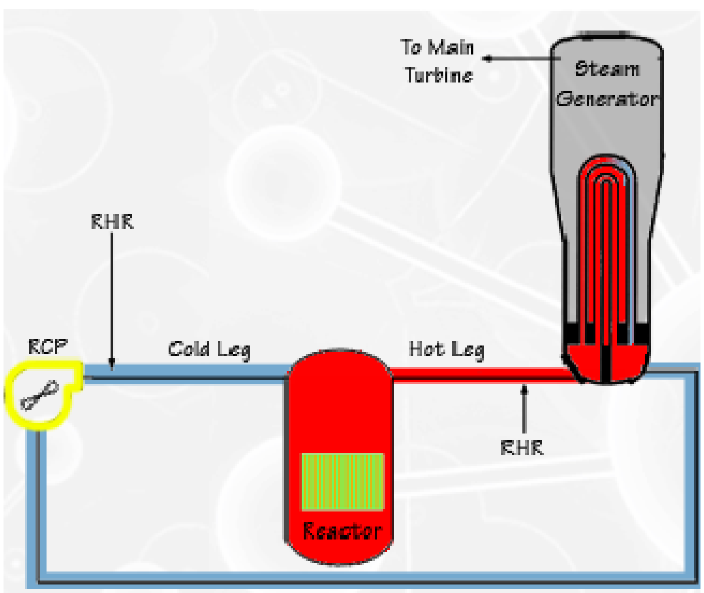
- Describe core cooling during power operation
- Describe issues associated with decay heat
- Describe core cooling mechanisms while shutdown
- Identify mechanisms required for natural circulation to occur
- Describe how the Emergency Core Cooling System works to provide core cooling during accident conditions

Core Cooling Mechanisms at Power

The function of a nuclear power plant is to convert nuclear energy to electrical energy or power. The nuclear fission process, covered in great detail in the Basic Atomic and Nuclear Physics course, describes how nuclear fuel produces heat, which will ultimately create steam to turn the main turbine generator and produce power for the utility's grid.

The heat energy produced by the fission process in the fuel is transferred out of the fuel pin into the primary water via convective heat transfer, causing the temperature of the water to increase. The primary water is often called the "reactor coolant," or more simply, the "coolant." In effect, primary water cools the fuel rods and, in doing so, initiates an energy transfer. The reactor coolant pump forces the heated coolant through the piping network of the primary system.

Click the diagram at right to see how heat produced in the reactor is used to generate steam, and how the reactor coolant pump (RCP) provides coolant to the reactor.



PWR

In a PWR, the heat energy produced by the fission process is transferred to the highly pressurized, subcooled coolant until the water reaches the steam generators. The hot coolant then transfers its heat via conduction across the steam generator heat exchanger tubes and turns the fluid on the secondary side into steam, which then drives the turbine generator. This transfer continually takes place: the fuel transfers heat to the circulated primary coolant; the circulated primary coolant heats the fluid in the secondary system to produce steam; the steam turns the turbine generator. This process removes the heat produced in the nuclear fuel and prevents it from exceeding its design temperature limits.

BWR

In a BWR, the heat energy, produced by the fission process, is transferred out of the fuel pin and into the reactor coolant via convective heat transfer, which increases the water's temperature. The reactor recirculation pumps force water from the reactor vessel's downcomer region through the reactor core, where it is changed to a steam/water mixture. Steam quality is then improved to 90% in the steam separators and then to 99% in the steam dryers, after which it is sent out to the turbine generator to produce electricity. The main condenser condenses the steam and returns it to the reactor vessel downcomer region by the condensate and feedwater system. The water removed from the steam is directed to the reactor vessel's downcomer region and preheats the feedwater before it is forced into the reactor core.

Review Question 15

In a PWR, the heat energy produced by the fission process is transferred to the highly pressurized, subcooled coolant until the water reaches the steam generators, where it is transferred by conduction to create steam in the secondary system to drive the turbine generator.

True or False?

Answer: True. This process removes the heat produced in the nuclear fuel and prevents it from exceeding its design temperature limits.

Review Question 16

In a BWR, the heat energy produced by the fission process is transferred out of the fuel pin and into the reactor coolant via convective heat transfer, increasing the water's temperature. The reactor recirculation pumps force water from the reactor vessel's downcomer region through the reactor core, where it is changed to a steam/water mixture.

True or False?

Answer: True. Afterwards, the main condenser condenses the steam and returns it to the reactor vessel downcomer region by the condensate and feedwater system.

Decay Heat

As you learned in the Basic Atomic and Nuclear Physics course, one unique aspect of nuclear power is that the fuel continues to generate heat as fission products decay after the reactor is shut down and the fission chain reaction has stopped. The amount of decay heat decreases over time since it is proportional to the radioactive decay process.

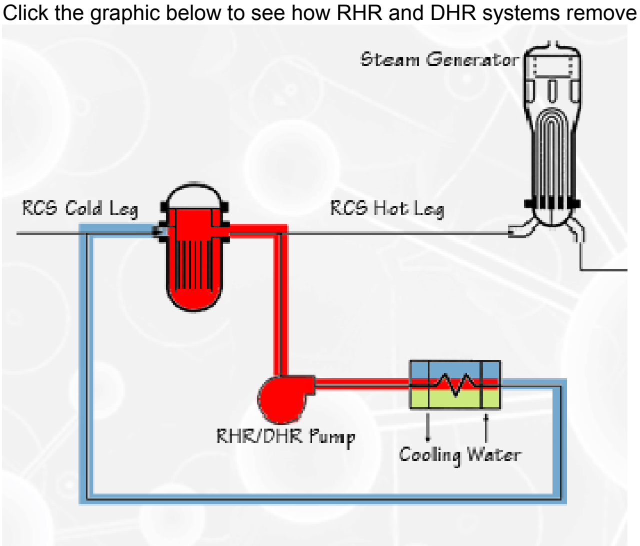
Time After Shutdown	Percent of Full Power
1 second	6.00%
1 minute	4.50%
30 minutes	2.00%
1 hour	1.60%
8 hours	1.00%
1 day	0.70%
2 days	0.60%
5 days	0.30%

Decay heat generation impacts spent fuel handling and storage, waste management, and reactor safety at a nuclear plant. Long after the reactor is shut down, adequate cooling must be available to remove the decay heat to prevent fuel damage. Reactor designs have specifically engineered safeguards to provide such cooling. Without cooling, there is enough thermal energy to boil the core dry and reduce heat transfer to the point of melting the fuel. Nuclear power plants use decay heat removal systems to remove decay heat from the core to a heat sink.

Normally, after a reactor shutdown or reactor trip, the reactor coolant system (RCS) continues to operate to remove the decay heat. The reactor coolant pumps transport the primary fluid through the system (including the reactor), removing the decay heat from the fuel. The coolant then transfers this heat to the secondary fluid in the steam generator, where it produces steam. While the reactor is shut down, the heat sink for this steam is typically either the condenser or atmosphere. This process is similar to power operation except that steam is not sent to the turbine to generate electricity.

When primary fluid temperatures are too low to support adequate steam formation in the steam generators, other options must be employed to remove heat from the RCS coolant. The residual heat removal (RHR) system [decay heat removal (DHR) system (Babcock and Wilcox (B&W) plants) or shutdown cooling systems in Combustion Engineering (CE) plants] removes decay heat when plant conditions cannot support decay heat removal through the steam system, such as when the plant is cooled down. These systems take reactor coolant from the RCS and pass it through a heat exchanger, where the decay heat is transferred to a separate cooling water system. The RHR and DHR systems remove heat from the RCS to cool the plant to an ambient temperature for maintenance or refueling operations and maintain the RCS temperature below saturation conditions.

Click the graphic below to see how RHR and DHR systems remove heat from the core.



Review Question 17

What system in a PWR is normally used immediately after a unit shutdown to remove decay heat?

A. Residual Heat Removal System
B. Decay Heat Removal System
C. Reactor Coolant System
D. All of the above

Answer: The Reactor Coolant System is used immediately after a unit shutdown to remove decay heat in a PWR.

Review Question 18

Which of the following are other options to remove decay heat when primary fluid temperatures are too low to support adequate steam formation in the steam generators?

A. Residual heat removal (RHR) system
B. Steam separator
C. Fission process
D. Decay heat removal (DHR) system

Answer: RHR and DHR systems are used to remove decay heat when primary fluid temperatures are too low to support adequate steam formation in the steam generators to further remove heat from the RCS coolant.

Natural Circulation: PWR

The reactor coolant pumps may become unavailable for decay heat removal at any time during normal operation. So, it’s critical that you understand the concept of natural circulation, how its flow can be verified, and what phenomena can promote or degrade its effectiveness.

Natural circulation plays an important role in the steam generator's heat transfer process, as well as in removing decay heat from a shutdown reactor core, if normal reactor coolant flow or RHR is not available. Natural circulation can typically remove generated heat at rates up to 10% of rated thermal power. Natural circulation becomes even more important in an accident situation, where continuous core cooling is essential to prevent further plant safety degradation.

Natural circulation flow is caused by the pressure differential between two columns of water of different densities. These different densities produce a differential pressure (ΔP) that generate a thermal driving head for flow.

Four conditions must be present for natural circulation to occur:

1. A heat source
2. A heat sink to which the secondary fluid transfers its heat
    - Without the heat sink, the secondary fluid's temperature continues to rise until it is equal to that of the heat source.
    - At this point, natural circulation ceases due to constant fluid density.
3. A difference in elevation between the heat sink (higher) and the heat source (lower)
    - If the positions of the reactor and steam generator were reversed, natural circulation would not be possible.
4. A continuous, unobstructed flowpath
    - A break in the path or blockage in the steam generator tubes will restrict natural circulation flow.

Once these conditions are met and natural circulation is taking place, the continuous flow of coolant without the aid of a pump will remove heat from the fuel.

Approximately 10-15 minutes after an initial loss of forced flow, natural circulation conditions are established and the conditions in the primary system begin to stabilize. The decay heat generation in the fuel will heat the coolant, which decreases its density. In the steam generator, the coolant transfers its heat to the secondary fluid making the cooler primary fluid more dense. The steam produced by the hot primary fluid is dumped, preferably to the condenser or atmosphere. The steam generators are fed via the Emergency (Auxiliary) Feedwater System. These actions establish the steam generators as operable heat sinks for the RCS.

Since the steam generators are at a higher elevation than the reactor vessel, the denser coolant tends to flow down into the vessel, displacing the hotter, less dense coolant from the core. As coolant flows through the core and becomes heated and less dense, it tends to rise into the steam generator. It is then effectively pushed along by cooler, denser water flowing out of the steam generator into the reactor vessel. Thus, a natural circulation is established, based upon the thermal driving head and the difference in height between the steam generator and the reactor vessel.

The following indications determine if natural circulation is taking place in a standard Westinghouse PWR. These indications may vary based on PWR design:

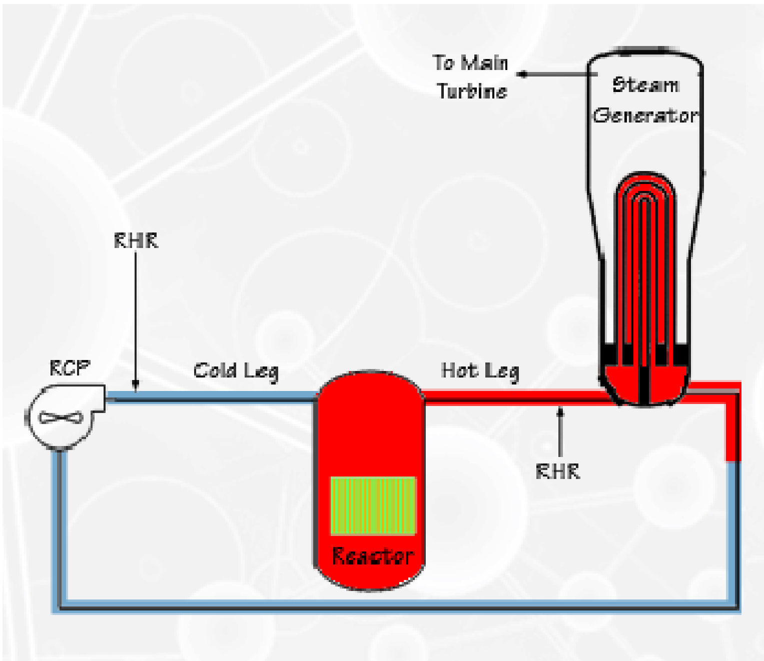
- RCS hot leg temperatures are stable or decreasing
- RCS core exit thermocouple temperatures are stable or decreasing
- Steam generator pressures are stable or decreasing
- RCS cold leg temperature is at saturation temperature for existing S/G pressure
- RCS loops have at least 30ºF of subcooling

Natural circulation can be retarded if boiling occurs or if gaseous voids or non-condensable gases accumulate in the system's high points, such as in the steam generator tubes or reactor vessel head.

Natural Circulation Example: PWR

The example below shows what happens when natural circulation is used to cool the reactor vessel.

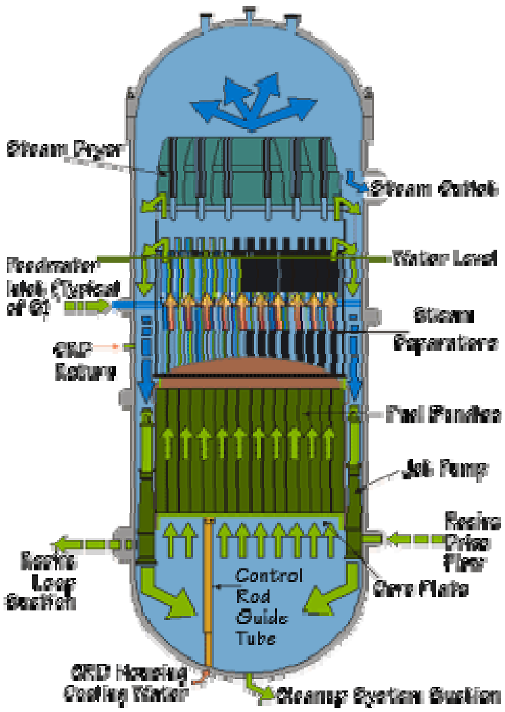
Click the diagram below to see an animation of when natural circulation is used to help cool the reactor vessel.



Natural Circulation: BWR

Natural circulation readily takes place in a BWR since the cold water column is outside the shroud, and the heated water column is inside the shroud and throughout the reactor core. Water heated in the core is displaced by cooler water entering from the downcomer region at the bottom of the core. As the water is heated and rises through the reactor, it reaches the spill-over point at the steam separators where the water flows into the downcomer region.

The water level must be maintained above the bottom of the dryer skirt to ensure enough head pressure is generated to force water through the steam separators. The process of natural circulation can happen in a BWR, but is not the normal mode of operation.



BWR Reactor Pressure Vessel

Review Question 19

Which of the following are indications that natural circulation is occurring?

A. RCS hot leg temperatures are increasing
B. RCS core exit thermocouple temperatures are stable or decreasing
C. Steam generator pressures are stable or decreasing
D. RCS cold leg temperature is at saturation temperature for existing steam generator pressure
E. RCS loops have less than 25ºF of subcooling

Answer: Indications that natural circulation is occurring include: RCS core exit thermocouple temperatures are stable or decreasing, stable or decreasing steam generator pressures, and RCS cold leg temperature is at saturation temperature for existing steam generator pressure.

Review Question 20

Which of the following are required for natural circulation to take place?

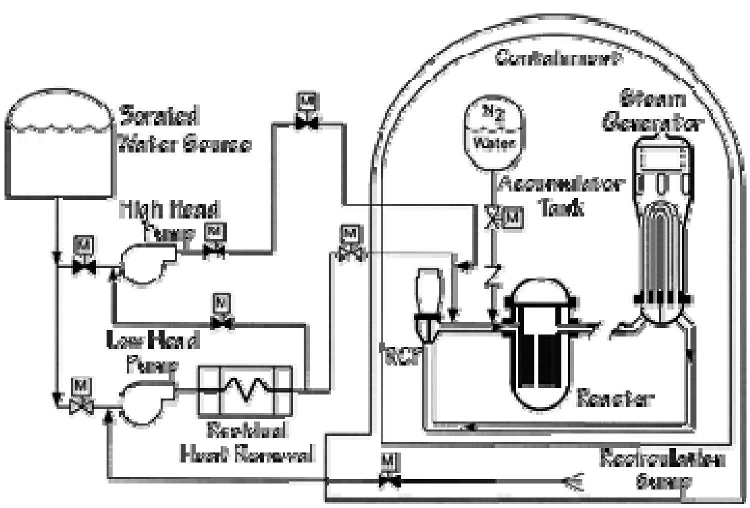
A. A heat sink
B. A Reactor Coolant Pump in operation
C. The heat sink above the heat source
D. Flow restriction in the S/G tubes

Answer: A heat sink and a heat sink above the heat source are required for natural circulation to take place.

PWR Emergency Core Cooling Systems: Safety Injection

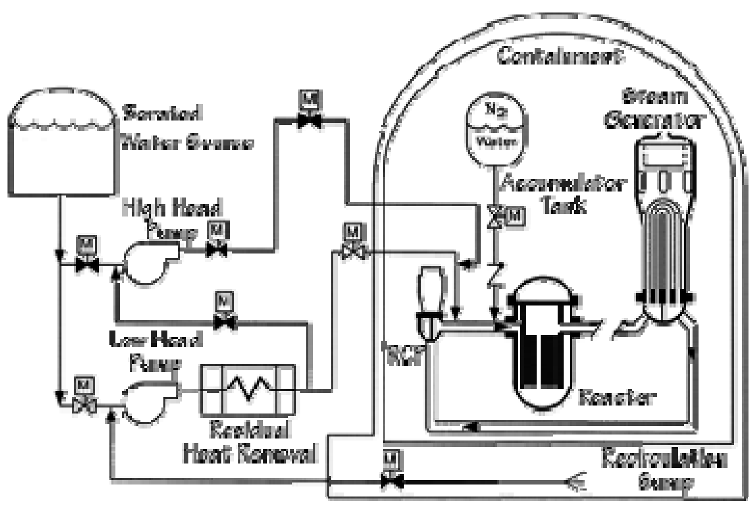
Safety Injection (SI) is the method for injecting large amounts of makeup water into the reactor core during accident/emergency conditions, when the normal method of core heat removal is insufficient or unavailable. A safety injection can be initiated automatically by the Emergency Safeguards Features Actuation System (ESFAS) when certain system setpoints are reached, or initiated manually by the control room operators. Once initiated, an SI signal will automatically start pumps and reposition valves in the Emergency Core Cooling System (ECCS) to direct the makeup water into the Reactor Coolant System piping and ultimately to the reactor vessel and the fuel. An SI signal will also activate other emergency equipment, such as emergency power sources and important support systems, which help mitigate a potential accident.

In order to better understand the plant response to a DBA-LOCA, as well as other accidents, it is important to comprehend the major subsystems in the ECCS. In a standard Westinghouse PWR, the ECCS consists of four primary subsystems: water sources, High Head SI pumps, Low Head SI pumps, and accumulators. Minor differences at individual plants may exist, but they will not affect the overall ECCS concept.



ECCS uses two primary water sources for accident mitigation. The first is a large, ready reserve of cool, borated water, which is used as the initial water source for the pumps used for safety injection. Borated water is used to add negative reactivity to the reactor, aiding in its shutdown during an accident. Typically, the water supply's volume is between 350,000 and 500,000 gallons and is housed in a large supply tank. The SI pumps are directly aligned with this water source and pump into the Reactor Coolant System (RCS) on a safety injection signal.

Once the borated water source is exhausted, a second source must be available. This secondary source is from the coolant that leaked from the RCS during the accident (e.g., a LOCA). This water accumulates at the bottom of the containment building and is collected in sumps, which are aligned to the SI pumps so that the flow of makeup water to the fuel is not interrupted. Because this sump water from the RCS was previously cooling water for the fuel, its temperature is elevated, often greater than 200°F. This sump water must be cooled before it is injected into the RCS. Typically, this water is cooled by the RHR and DHR systems in heat exchangers, which was previously discussed.



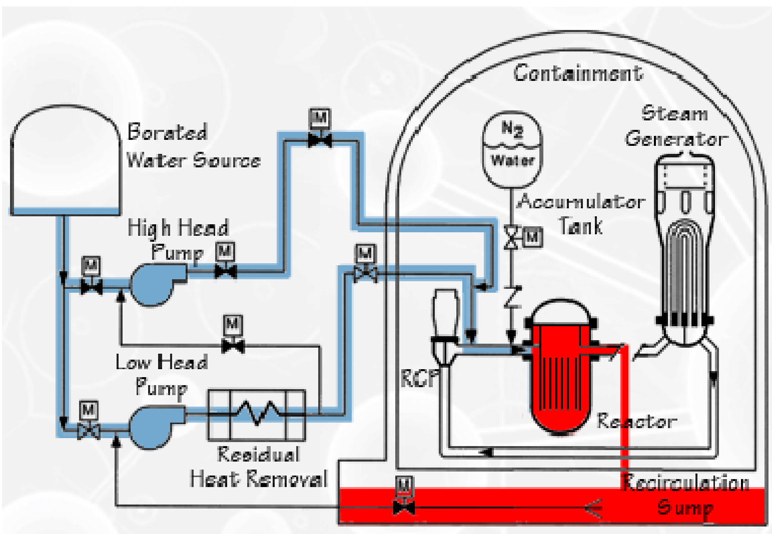
High Head (Pressure) SI pumps take suction from the large supply tank of cool, borated water and inject it into the RCS during a safety injection. These pumps are designed to pump a relatively low volume of water at a very high pressure, hence the name "High Head" or "High Pressure." They are the first line of defense for a reactor since they supply the emergency makeup water when the RCS is at high pressures.

Typically, at least two High Pressure pumps supply cooling water on an SI, each with a different power source. In the deterministic accident analysis, only one train of high pressure injection is required to mitigate the effects of a LOCA. When the tank of water is empty, High Pressure SI pumps will obtain their water directly from the containment sumps or through another system. These pumps usually have a dual purpose: they also supply smaller amounts of makeup water to the RCS during normal operation. You will see an example of how these pumps work on the next page.

PWR ECCS: Low Head SI Pumps

Low Head (Pressure) SI pumps also take suction from the large supply tank of cool, borated water and inject it into the RCS on a safety injection. These pumps are designed to pump a high volume of water at a low pressure, hence the name "Low Head" or "Low Pressure." They supply emergency makeup water when the RCS is depressurized. As with the High Pressure SI pumps, typically at least two pumps supply cooling water on an SI, each with a different power source, with only one train of low pressure injection required to mitigate the effects of a LOCA.

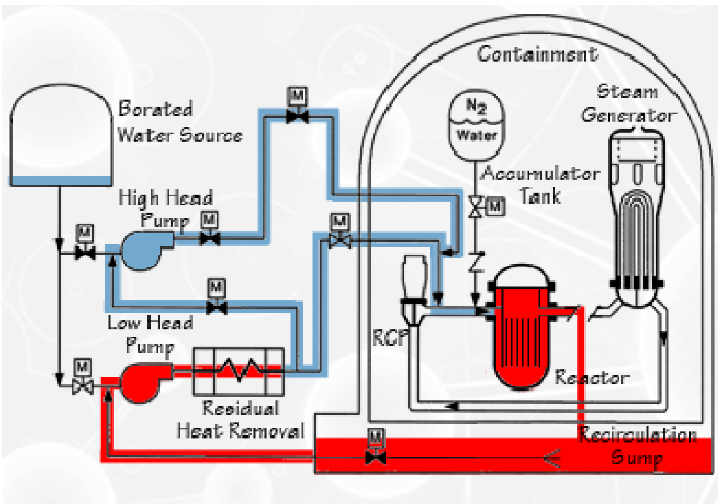
Click the animation at right to see how high head and low head SI pumps use a borated water source to cool the core during a LOCA.



PWR ECCS: Low Head SI Pumps, continued

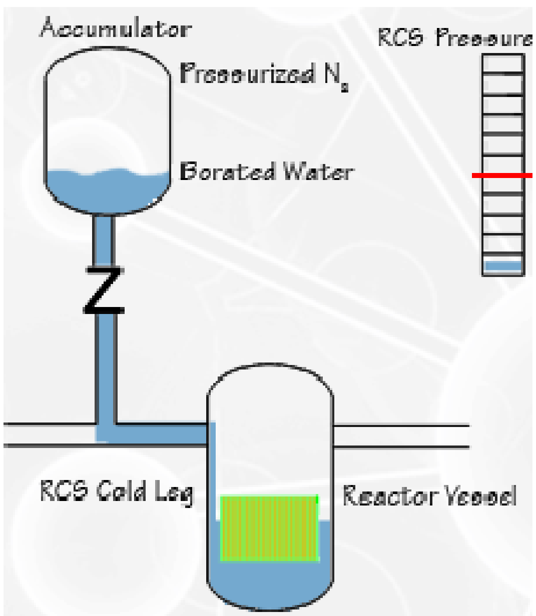
When the water tank is empty, Low Pressure SI pumps will obtain their water source from the containment sumps. These low pressure pumps are typically the RHR or DHR pumps discussed earlier. Some plants also have Intermediate Head SI pumps, which provide water to the reactor when slightly elevated RCS pressures exist.

Click the animation at right to see how Low Head SI pumps obtain water from containment sumps to cool the core.



One accumulator tank (core flood tank for a B&W plant and safety injection tanks for a CE plant) is directly connected to each loop of the RCS to aid core cooling on a large break LOCA. Each tank is partially filled with borated water and is pressurized with nitrogen gas to between 600 psig and 700 psig. Check valves prevent the higher pressure RCS water from back-filling the accumulator tanks. Because the accumulators are directly connected to the RCS and use basic mechanical principles, no signal is required for their operation and they will start to instantaneously inject their contents into the RCS piping once RCS pressure is below the tank pressure. Unlike the two SI pump types, ECCS accumulators do not have dual functionality.

Click the animation at right to see how accumulators work to inject borated water into the RCS as RCS pressure decreases.



Review Question 21

Match the PWR Emergency Core Cooling System term with its description.

Safety injection	A. Used for accident mitigation by the ECCS: the first is a large, ready reserve of cool, borated water, the second is from coolant that leaks from the RCS during an accident
Water systems	B. Injects high volume of water at a low pressure into the RCS during an SI
High Head SI pumps	C. Method for injecting large amounts of makeup water into the reactor core during accident/emergency conditions
Low Head SI pumps	D. Directly connected to each loop of the RCS to aid with core cooling on a large break LOCA
Accumulators	E. Injects relatively low volume of water at a very high pressure into the RCS during an SI

The correct matching sequence is CAEBD.

In a BWR, there are typically four Emergency Core Cooling Systems.

Click each Emergency Core Cooling System below to learn more about them.

High Pressure Core Spray (HPCS - BWR 5&6 designs)
The high pressure core spray system (HPCS) provides emergency core cooling for small breaks in the reactor coolant pressure boundary that do not depressurize the reactor vessel. The system assist in removing decay heat from the reactor and is used as a backup for the reactor core isolation cooling system (RCIC).

High Pressure Coolant Injection (HPCI - BWR 1-4 designs)
The high pressure coolant injection system (HPCI) is used to provide emergency core cooling for small breaks in the reactor coolant pressure boundary that do not depressurize the reactor vessel. The system will assist in removing decay heat from the reactor pressure vessel and is used as a backup for the reactor core isolation cooling system (RCIC).

Low Pressure Coolant Injection (LPCI)
The low pressure coolant injection system (LPCI) is used to restore and maintain the reactor pressure vessel water level after a LOCA event where the reactor vessel has been depressurized. This system is part of the residual heat removal system (RHR).

Core Spray (CS – BWR 2-4) or Low Pressure Core Spray (LPCS – BWR 5&6)
The low pressure core spray system (LPCS) is used to restore and maintain reactor pressure vessel water level after a LOCA event where the reactor vessel has been depressurized.

Automatic Depressurization System (ADS)
The automatic depressurization system is used to depressurize the reactor after a pipe break in the reactor core pressure boundary. The system is used when the pressure cannot be reduced and the ECCS cannot maintain the water level above the very low level mark. This system will only depressurize the vessel and will not replace any reactor pressure vessel water.

The emergency core cooling system is designed to protect the reactor vessel during all LOCA events, small, intermediate, and large.

For a small LOCA (e.g., small instrument line break), the reactor pressure will remain at high pressure and the HPCI/HPCS is sufficient to maintain the water level in the reactor vessel and adequate core cooling can be achieved.

For an intermediate LOCA (e.g., small break in recirculation piping), the reactor vessel pressure stays high and the HPCI/HPCS can inject water into the reactor but cannot maintain the water level due to the pipe break. The low pressure systems that could raise the water level in the reactor begin operating, but cannot inject water into the system because the reactor pressure is too high. To allow the water level to reach normal levels, the ADS will start operating and automatically reduce the reactor vessel pressure to allow the low pressure systems (CS/LPCS and LPCI) to inject water into the vessel and achieve adequate core cooling in the reactor vessel.

For a large LOCA (e.g., design basis accident), the reactor vessel depressurizes and the HPCI will start injecting water into the reactor vessel and then be isolated to start the HPCS injection. Once pressure has been reduced to the operational pressure of the CS/LPCS and LPCI systems, the low pressure systems will begin injecting water into the reactor vessel and allow for adequate cooling of the reactor vessel.

Each of the systems that make up the ECCS has a flow path that either receives cooling water from the condensate storage tank, the suppression pool, or both.

HPCI

This system is started by a DC-powered pump that draws the steam out of the reactor vessel and into a steam driven turbine pump. Once the turbine is operational the steam from the reactor will operate the turbine and will flow to the suppression pool. Once operational, the turbine-driven pump will pump water from the condensate storage tank into the feedwater lines going to the reactor vessel. The suppression pool water can be used as a backup for this system when the condensate storage tank level is too low.

HPCS

This system is operated by an AC-powered pump that draws water from the condensate storage tank and will inject water into the high pressure core spray spargers that are located above the core in the reactor vessel. This system can draw water from the suppression pool if the level in the condensate storage tank gets too low.

CS/LPCS

The low pressure core spray system is operated by an AC-powered pump that pulls water from the suppression pool and injects water into the low pressure core spray spargers that are located above the core in the reactor vessel.

Residual Heat Removal System (RHR)

This system is used to remove heat from the suppression pool as other emergency systems are injecting water into the pool. The system pumps water out of the pool into heat exchangers that have service water on the shell side of the shell and tube heat exchanger. The heat from the suppression pool water is transferred to the service water and the cooled suppression pool water is pumped into the reactor where it will assist in cooling the reactor vessel.

Containment Cooling Systems

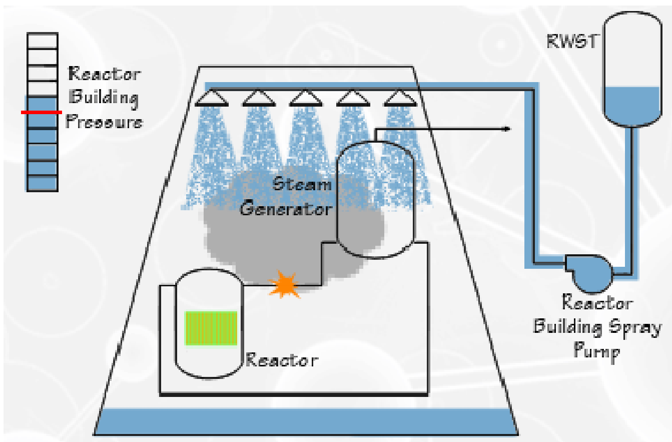
The reactor building forms a final physical barrier, by preventing the release of radioactive fission products to the general public. In the event of a LOCA, the fuel's cladding and the RCS could fail and the radioactive fission products from the nuclear fuel may be released into the reactor building atmosphere. However, as long as the reactor building remains intact, fission products will not be released to the general public. For instance, because reactor building integrity was maintained at Three Mile Island, fission products were contained. On the other hand, when reactor integrity was breached at Chernobyl, which lacked a containment structure, radioactivity was released into the atmosphere. Therefore, maintaining the integrity of the reactor building is very important.

The reactor building is a reinforced concrete structure designed to withstand earthquakes and pressures expected from design accidents. The rupture of a main steam line, or a LOCA, inside the reactor building will release very hot water that immediately turns to steam in the building's atmosphere.

To prevent the pressure and temperature inside the building from exceeding their design values, the spray system, located at the top of the structure, sprays cool water through its nozzles into the building's atmosphere. This spray absorbs heat from the steam, condenses the steam, and lowers the reactor building's pressure and temperature. The condensed steam and spray water fall into sumps at the bottom of the reactor building. During normal power plant operation, a separate containment cooling system cools the reactor building. During an accident, this same cooling system works in conjunction with the spray system to maintain temperatures below design values.

In a BWR, the primary containment consists of the dry well and suppression pool. The dry well is used to contain a pipe break outside the reactor building. When dry well pressure increases, the increase alarms the reactor safety systems and SCRAMs the reactor. The suppression pool cools the reactor and dry well after a LOCA, and is the location where the relief and safety valves from the pressure vessel will blow down if the reactor pressure level increases above normal operating limits. There are three active containment designs (Mark I, II, and III), which will be discussed in your Systems Training. The reactor building is considered the secondary containment and has provisions for containing any releases from the primary containment. Refer to your Systems Training for more information.

Click the graphic at right to see how reactor building pressure is maintained during a LOCA.



Review Question 22

The reactor building forms the last physical barrier preventing the release of radioactive fission products to the general public.

True or False?

Answer: True. As long as the reactor building remains intact, fission products will not be released to the general public.

Conclusion

Providing cooling to the fuel is essential in order to remove the heat generated, both from power operation or decayed heat. Because of the importance of maintaining core cooling, plants are designed with multiple methods to provide this cooling so that heat is removed from the fuel for all situations that can be encountered for normal, abnormal, or emergency operations.

In addition to stressing the importance of maintaining cooling to the core, this lesson described the operation of the ECCS, which will be the foundation of how nuclear plants mitigate many of the design basis accidents that require analysis.

Now that you have completed this lesson, you can:

- Describe core cooling during power operation
- Describe issues associated with decay heat
- Describe core cooling mechanisms while shutdown
- Identify mechanisms required for natural circulation to occur
- Describe how the Emergency Core Cooling System works to provide core cooling during accident conditions

In the next lesson, we will review design basis accidents and how they are used in BWRs and PWRs.

5
DESIGN BASIS ACCIDENTS

Introduction

The largest threats to the Core Protection systems in a nuclear plant are transients and accidents. Of these, the most challenging are design basis accidents. In this section, we will explore the worst case design basis accident, the large break LOCA, explore the acceptance criteria that determine the design requirements for this accident, and look at critical safety functions and the role they play in ensuring core protection. We will also speak briefly to other design basis accidents and plant transients.

After completing this lesson, you will be able to:

- Describe DBA-LOCA
- Identify the ECCS Acceptance Criteria IAW 10CFR50.46
- Identify the sources and dangers of hydrogen
- Describe the four accident classifications
- Identify the six (6) critical safety functions
- Identify critical parameters monitored during accident conditions

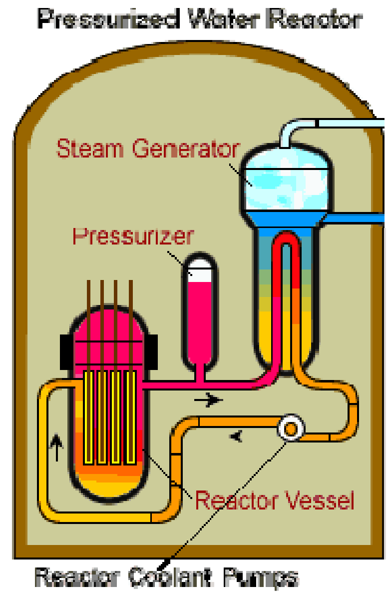
A Design Basis Accident (DBA) is typically a worst-case accident of interest used to design plant components. DBAs often bound (have more severe consequences than) other accidents so the latter do not need to be analyzed further. For example, the Design Basis Large Break Loss of Coolant Accident (LOCA) bounds (has a higher containment pressure than) other LOCAs with slightly smaller break areas.

A DBA-LOCA can occur in both Pressurized and Boiling Water reactors.

Pressurized Water Reactor (PWR) DBA-LOCA
A PWR DBA-LOCA is defined as a double-ended guillotine piping rupture with a simultaneous loss of off-site power. The single worst case active failure is also assumed. This accident has major implications on Nuclear Steam Supply System (NSSS) design, particularly the ECCS.

Boiling Water Reactor (BWR) DBA-LOCA
A BWR DBA-LOCA is defined as an instantaneous guillotine severance of the recirculation line with a simultaneous loss of off-site power. The single worst case active failure is also assumed. “Instantaneous guillotine” means that the pipe is postulated to crack completely through and around its circumference and immediately separates so that blowdown flow comes out of both ends across the full inside diameter. The specific recirculation line location, suction or discharge, is typically fuel cycle-dependent.

Symptoms and indications: PWR-LOCA

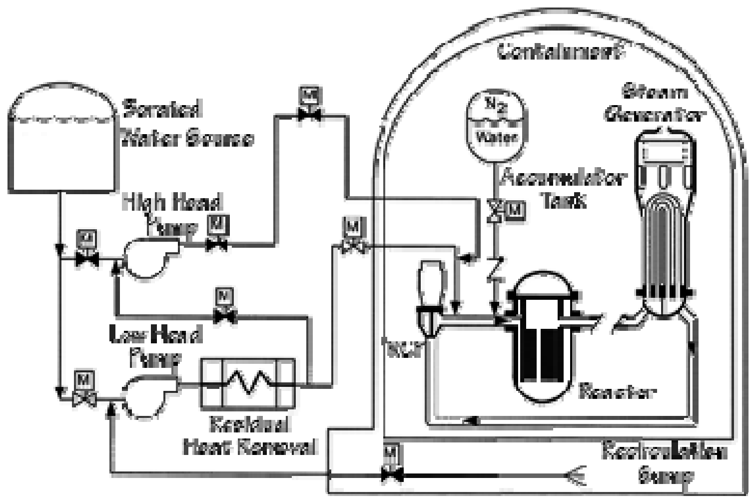


A break of the RCS piping or any line connected to that system up to the first closed valve would result in a LOCA. The charging pumps, however, can make up for ruptures of a small cross section, thus permitting an orderly shutdown. The coolant released would remain in the containment.

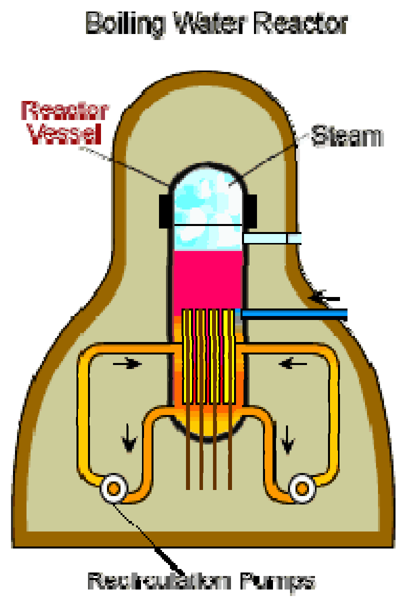
For a postulated break, a reactor trip is initiated when the reactor coolant's low pressure setpoint is reached while the SI signal is actuated by reactor coolant low pressure or containment high pressure. The consequences of the accident are limited in two ways:

1. Reactor trip and borated water injection causes a rapid reduction of nuclear power to a residual level corresponding to the delayed fission product decay
2. Injection of cool, borated water ensures sufficient core flooding to prevent excessive temperatures

Before the reactor trip occurs, the reactor is in equilibrium (i.e., the heat generated in the core is being removed via the secondary system where the energy is transferred to the turbine generator). After a reactor and/or turbine trip, heat from the core, hot internals, and the vessel continues to be transferred to the RCS fluid and then to the secondary system; however, the turbine has been tripped. Accordingly, the secondary system pressure increases and steam dump may occur. Steam is dumped to either the condenser or the atmosphere. The secondary flow helps reduce RCS pressure, which will allow the accumulators (core flood tanks) to inject borated water when pressure falls below 600 psia (pressure may vary at each plant).



BWR-LOCA initial symptoms and indications include:



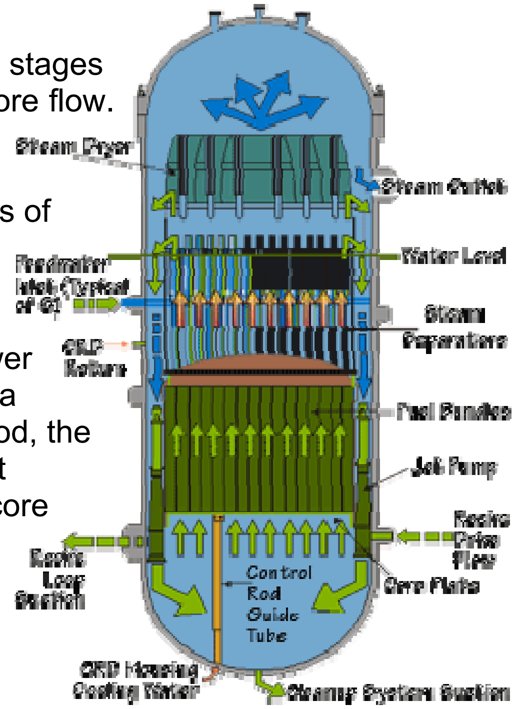
- Rapid rise in dry well (Primary Containment) pressure
- Decrease in reactor water level
- All ECCS pumps start
- Reactor empty and depressurized (inventory to suppression pool)

Immediately after the line break, vessel pressure and core flow begin to decrease. The initial pressure response is governed by the three events: closure speed of main steam isolation valves, values of decay heat, and energy removed from the system by the fluid from the break.

The core flow initially decreases rapidly because the broken loop's recirculation pump ceases to pump almost immediately due to lost suction. The intact loop's pump slows governing the core flow response for the next several seconds. As the level decreases, core flow will near zero. The reactor vessel's pressure decreases significantly when a break occurs with fluid leaving the break changing from water to steam. Subcooled water in the lower plenum saturates and flashes up through the core, momentarily increasing the core flow.

Symptoms and Indications: BWR-LOCA, continued

Heat transfer rates on the fuel cladding during the early stages of the accident are governed primarily by changes in core flow. When the break uncovers the fuel, convective heat transfer is assumed to cease. The water level in the core area remains relatively high during the early stages of the accident (jet pumps maintain flooding of 2/3 of the core), but the reactor core will go completely dry before ECCS is actuated to reflood the core. The cladding temperature initially decreases due to low power and relatively high core flow, then rapidly increases for a short time when the core is uncovered. During this period, the clad is almost the same temperature as the fuel so heat transfer is governed by the amount of decay heat and core spray heat transfer rate. The temperature increase is terminated by reflood of the core.



Review Question 20

A design basis accident (DBA) is typically an average-type accident used to design plant components.

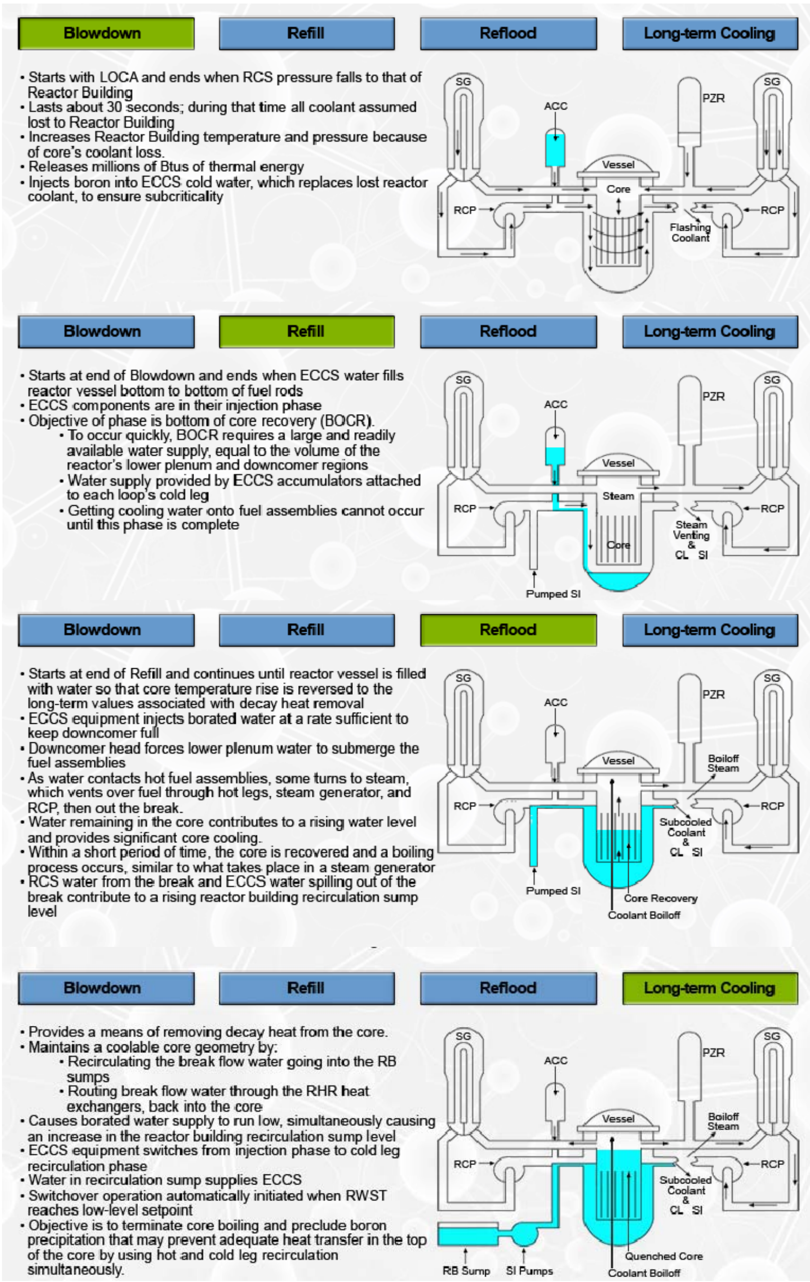
True or False?

Answer: A design basis accident (DBA) is typically a worst-case accident of interest used to design plant components. DBAs often bound (have more severe consequences) other accidents so the latter do not need to be analyzed further.

DBA LOCA Phases: PWR

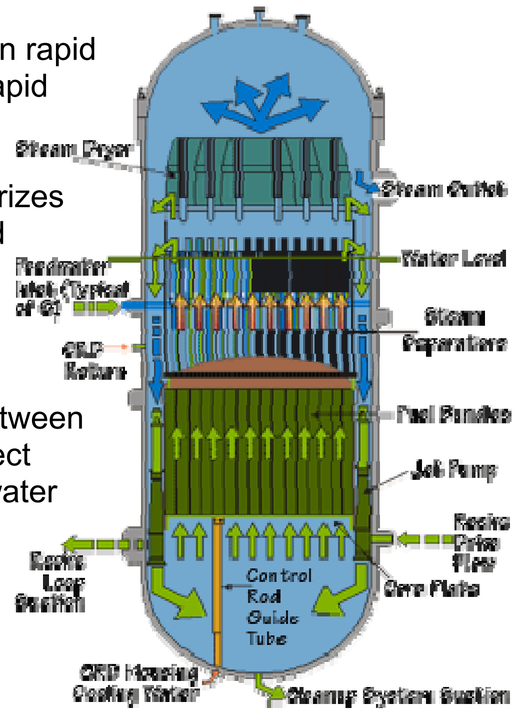
A large-break LOCA has four characteristic stages: blowdown, refill, reflood, and long-term cooling. Prior to a LOCA, the primary coolant is subcooled to about 587°F with a pressure near 2250 psia (plant dependent values). When the rupture occurs, the differential pressure between the RCS and the reactor building atmosphere rapidly expels fluid from the system.

Click each button below to learn more about the LOCA's four stages.



DBA LOCA Phases: BWR

A larger break (e.g., a recirculation loop shear) results in rapid depressurization and a loss of coolant inventory. The rapid discharge of coolant causes large and rapid pressure and temperature increases and a large downward loading on the dry well floor. Once the vessel depressurizes and the blowdown is over, the ECCS pumps will reflood the vessel. The cooler suppression pool water will rapidly reduce the pressure and temperature in the dry well and may cause the vacuum breakers to cycle.



When the vessel has reflooded, a closed loop forms between the reactor and the containment. The ECCS pumps inject water from the suppression pool into the reactor. The water picks up decay heat as it flows through the core. The heated water flows out of the break, spills onto the dry well floor, and flows back to the suppression pool.

The suppression pool water is cooled by the RHR heat exchangers. Over the next few hours, the temperature of the suppression pool water and the pressure and temperature of the dry well will slowly increase until the decay heat drops below the cooling capacity of the heat exchangers. The timing of this temperature drop depends upon reactor power history and RHR heat exchanger cooling capacity, which varies with cooling water temperature and heat exchanger fouling.

ECCS Acceptance Criteria

Under accident conditions, the Emergency Core Cooling System (ECCS) ensures that core cooling is maintained and provides for the safe shutdown of the reactor. Its design basis is to meet the acceptance criteria of 10CFR50.46.

Click each acceptance criterion on the left side of the table to learn about its limit and purpose.

Cladding temperature	≤ 2200°F	Prevents loss of clad integrity
Cladding oxidation	< 17 percent of clad thickness	Prevents loss of clad integrity
Hydrogen generation	< 1 percent of amount if all clad surrounding the fuel were to react	Prevents accumulation of flammable or explosive mixture of H_2
Geometry	Core remains intact and in a geometry conducive to cooling	Prevents clad failures which could block coolant channels
Long-term cooling	Core remains in configuration, which can be cooled for long period of time	Ensures that decay heat can be removed to prevent core damage

Review Question 21

Which of the following are ECCS Acceptance Criteria IAW 10CFR50.46?

A. Long-term heating
B. Cladding temperature
C. Hydrogen generation
D. Cladding neutralization
E. Geometry

The correct answers are B, C, and E. Cladding temperature, hydrogen generation, and geometry are three acceptance criteria.

Hydrogen Hazards during Accidents

The hydrogen generated during accident and post-accident conditions is a concern because of the potential for a hydrogen burn or explosion. If the burn or explosion is inside the containment, the design limit for containment pressure may be approached or exceeded, thus challenging the containment barrier.

When airborne, hydrogen has both explosive and flammable ranges:

- Explosive range: 18%-59%
- Flammable ranges: ~4%-18%, and 59%-75%

There are six major sources of hydrogen generation:

Click each source of hydrogen below to learn more.

Radiolytic Decomposition of Water
- Occurs in high gamma flux
- The most prevalent hydrogen source after a design basis LOCA (chronic source of hydrogen)

Zinc Paint Corrosion and Zinc Corrosion
- Found in paint, conduit, and junction boxes
- Second most prevalent source

Aluminum-corrosion
- Extreme reaction with NaOH
- Because of this, there are limits on aluminum in Containment

Zirc–water Reaction
The zirconium reaction with water forms zirc-oxide, represented by the formula, $Zr+2H2O \rightarrow ZrO2+2H2+heat$. The Zirc-H2O reaction:
- ~1800°F: Starts
- ~2200°F: Becomes significant
- ~2800°F: Becomes self-sustaining
- ~3375°F: Zirc-oxide melts (at ~5080°F, the fuel melts)

ZrO2 is very brittle and the H2 gas produced is a potential explosive hazard.

RCS Hydrogen Inventory
This occurs since radiolytic decomposition of water occurs even at low neutron flux levels.

Pressurizer Bubble
This occurs only in PWRs. This is primarily due to Radiolytic Decomposition of Water.

Boiling Water Reactors use nitrogen in their dry wells to prevent a flammable mixture of O_2 and H_2. Pressurized Water Reactors and Boiling Water Reactors have hydrogen recombiners to remove hydrogen from the containment and keep the concentration below the flammable limit.

Other Design Basis Accidents: PWRs

There are other DBAs to be considered in PWRs.

Click each PWR-DBA below to learn more.

Control Rod Drive Mechanism (CRDM) Rupture (rod ejection & small break LOCA)
If a control rod were to be ejected through the reactor vessel head penetration assembly where the corresponding drive mechanism is located, a leak path for the reactor coolant system would result creating a small break LOCA.

Fuel Handling Accident
- Radioactive fuel dropped
- Fission products released to atmosphere or refueling area tanks

Main Steam (MS) or Feedwater (FW) Line Break
- Main Steam break decreases moderator temperature entering the reactor (positive reactivity added) and potential power transient
- Feedwater line break removes the heat sink for the reactor causing increase in temperature and pressure

Reactor Coolant Pump (RCP) Locked Rotor
- Rapid reduction in flow through the steam generator and core
- May cause inadequate core cooling

Steam Generator (SG) Tube Rupture (SGTR)
- Radioactive water leaks into the secondary side of the plant
- Loss of Reactor Coolant Inventory

Other Design Basis Accidents: BWRs

There are other DBAs to be considered in BWRs.

Click each BWR-DBA below to learn more.

Main Steam Line Break Outside Dry Well (Primary Containment)
- Radioactive steam outside the primary containment
- Reduction in Reactor Vessel Inventory

Control Rod Drop
- Control rod stuck at the top of core, uncoupled from its withdrawn drive mechanism
- At a later time, control rod drops out of core, rapidly inserting positive reactivity
- Reactivity excursion limited by design of control rods

Refueling Accident
- Radioactive fuel dropped and fission products released to the atmosphere or refueling area tanks
- Misplaced fuel in reactor core (not in its design location)

Recirculation Pump Shaft Seizure or Break
- Rapid decrease in core flow
- Reduced core cooling

Radioactive Release Due to System/Component Failure
Radioactive tanks, piping, or components leak or break allowing release of radioactivity outside the plant boundaries

Review Question 22

What is the most common source of hydrogen generation after a DBA-LOCA?

A. Zirc–water Reaction
B. Aluminum-corrosion
C. Zinc Paint Corrosion and Zinc Corrosion
D. Radiolytic Decomposition of Water

The correct answer is D. The most common source of hydrogen generation is radiolytic decomposition of water after a design basis LOCA.

Accident Classifications: PWRs

There are four types of accident classifications for PWRs: normal operations, faults of moderate frequency, infrequent faults, and limiting faults.

Click each accident classification on the left side of the table to see a description and example of it.

Normal Operations | Faults of Moderate Frequency | Infrequent Faults | Limiting Faults

Description	Examples
Occurrences, which are expected frequently or regularly in the course of power operation, refueling, maintenance, or maneuvering of the plant.	• Power Operation Startup • Heatup and cooldown up to 110 F per hour • Step load changes (up to 10%)

Normal Operations | Faults of Moderate Frequency | Infrequent Faults | Limiting Faults

Description	Examples
Also called Anticipated Transients, Condition 2 events result in a reactor trip, at worst. In any case, there is no fuel damage and any radiation release is less than 10CFR20 limits.	• Excessive increase in secondary steam flow • Feedwater malfunctions that decrease feedwater temperature or increase feedwater flow • Loss of non-emergency AC power • Inadvertent operation of ECCS during power operations

Normal Operations | Faults of Moderate Frequency | Infrequent Faults | Limiting Faults

Description	Examples
These are faults that may occur infrequently during the life of the plant, may be accompanied by the failure of only a small fraction of the fuel rods, although sufficient fuel damage might occur to preclude resumption of the operation for a considerable outage time. The release of radioactivity will not be sufficient enough to interrupt or restrict public use of areas beyond the exclusion radius of the site. An ANS Condition 3 (infrequent faults) occurrence will not by itself generate an ANS Condition 4 (limiting faults) occurrence or result in a consequential loss of function of the RCS or containment barriers.	• Minor steam system piping failure • Complete loss of forced reactor coolant flow • Rod cluster control assembly misalignment • Small break LOCA • Spent fuel cask drop accidents

Normal Operations | Faults of Moderate Frequency | Infrequent Faults | Limiting Faults

Description	Examples
These faults are not expected to occur. They are postulated because their consequences include the potential release of significant amounts of radioactive material. These are the most drastic occurrences that must be designed against and represent limiting design scenarios. ANS Condition 4 occurrences shall not cause: • A fission product release to the environment resulting in a radiation exposure to the public in excess of the guidelines in 10CFR100 • A consequential loss of required functions of systems needed to cope with the fault including those of the ECCS and the containment	• Major Steam system piping failure • Feedwater pipe break • Reactor coolant pump shaft seizure or break • Rod cluster ejection • Large break LOCA • Design basis fuel handling accidents

Accident Classifications: BWRs

The same four types of accident classifications exist for BWRs.

Click each accident classification on the left side of the table to see a description and example of it.

Normal Operations
Faults of Moderate Frequency
Infrequent Faults
Limiting Faults

Description	Examples
High frequency events expected during the regular course of power operation, refueling, maintenance, or maneuvering the plant. These events do not result in: • Release of radioactive material to the environs that exceed 10 CFR 20 or 10 CFR 50 limits • Fuel failure such that 10 CFR 20 limits would be exceeded via normal discharge paths • Nuclear system stresses in excess of that allowed by applicable industry codes or result in an event not considered by the plant safety analyses.	• Refueling • Achieving criticality • Heatup (while critical) • Power operation • Achieving shutdown • Cooldown (while sub-critical)

Normal Operations
Faults of Moderate Frequency
Infrequent Faults
Limiting Faults

Description	Examples
These are incidents that may occur as often as once a year up to once every 20 years. These events are referred to as an "anticipated (expected) operational transient" and do not result in: • Release of radioactive material to the environs that exceed 10 CFR 20 limits • Operation-induced fuel failure • Nuclear system stresses in excess of that allowed for the transient classification by applicable industry codes • Containment stresses in excess of that allowed by applicable industry codes	• Loss of instrument air • Inadvertent startup of HPCS or idle recirculation loop • Inadvertent MSIV closure • Loss of all feedwater flow • Feedwater controller failure • Loss of main condenser vacuum

Normal Operations
Faults of Moderate Frequency
Infrequent Faults
Limiting Faults

Description	Examples
These are incidents that may occur once during the life of a plant. These events are referred to as an "abnormal (unexpected) operational transient" and do not result in: • Release of radioactive material to the environs that exceeds a small fraction of 10 CFR 100 • Fuel damage that would preclude resumption of normal operation after restart • A condition that results in consequential loss of reactor coolant system function • A condition that results in a consequential loss of containment barrier function	• Main generator or main turbine trip with bypass system failure • Inadvertent loading and operation of a fuel assembly in an improper position

Normal Operations
Faults of Moderate Frequency
Infrequent Faults
Limiting Faults

Description	Examples
These are incidents that are not expected to occur but are postulated because their consequences may result in the release of significant amounts of radioactive material. These events are referred to as "design basis (postulated) accidents" and do not result in: • Release of radioactive material to the environs that exceeds the guideline values of 10 CFR 100 • Fuel damage that would cause changes in core geometry such that core cooling would be inhibited • Nuclear system stresses in excess of those allowed for the accident classification by applicable industry codes • Containment stresses in excess o those allowed for the accident classification by applicable industry codes when containment is required • Radiation exposure to plant operations personnel in the main control room in excess of 5 Rem whole body, 30 Rem inhalation, and 30 Rem skin	• Fuel handling accident • LOCA inside the containment • Feedwater line break outside the containment • Liquid radwaste system storage tank failure

Review Question 23

Match each accident classification to its description.

Condition 1: Normal Operations	A. Events referred to as an "anticipated (expected) operational transient."
Condition 2: Faults of Moderate Frequency	B. These are incidents that are not expected to occur but are postulated because their consequences may result in the release of significant amounts of radioactive material.
Condition 3: Infrequent Faults	C. These events are referred to as an “abnormal (unexpected) operational transient." The release of radioactivity will be insufficient to interrupt or restrict public use of areas beyond the site's exclusion radius.
Condition 4: Limiting Faults	D. Expected regularly in the course of power operation, refueling, maintenance, or maneuvering the plant.

The correct matching sequence is DACB.

Critical Safety Functions (CSF)

Critical Safety Functions were identified to focus protection on the fission product barriers and associated monitoring. They are, in order of priority:

1. Sub-criticality
2. Core Cooling
3. Heat Sink
4. RCS Integrity
5. Containment Integrity
6. RCS Inventory

Each fission product barrier is protected by one or more of the six Critical Safety Functions.

Barrier Being Protected	Critical Safety Functions	
Fuel pellet and cladding	**Sub-criticality:**	Minimizes energy production in fuel
	Core Cooling:	Removes adequate heat from the fuel
	Heat Sink:	Removes heat from the coolant
	RCS Inventory:	Provides enough coolant to keep the fuel covered and for effective heat removal
RCS pressure boundary	**Sub-criticality:**	Minimizes energy production in fuel
	Heat Sink:	Removes adequate heat from the RCS
	RCS Integrity:	Designed for high pressures and pressurized thermal shock
	RCS Inventory:	Assures sufficient coolant is available for effective heat removal and pressure control
Containment barrier	**Containment Integrity:**	Reactor building and containment vessel are designed for elevated pressures

Critical Parameter Monitoring during Accident Conditions

As a result of the accident at Three Mile Island (TMI) Unit 2, each UFSAR has an Appendix that states requirements for additional monitoring instrumentation.

For PWRs, instruments monitor:	For BWRs, instruments monitor:
• RCS Wide Range T_{hot} And T_{cold} Temperatures • RCS Pressure • Pressurizer (PZR) Water Level • Steam Line Pressure • Steam Generator (SG) Water Level • Containment Pressure • Refueling Water Storage Tank Level • Reactor Vessel Levels Indicating System (RVLIS) • Core Exit Thermocouple Temperatures	• Reactor Vessel Water Level • Reactor Vessel Pressure • Dry Well Pressure and Temperature • Suppression Chamber Water Temperature and Level • Suppression Chamber Air Temperature • Source Range Neutron Levels

Technical Specifications also require monitoring of post-accident parameters to help diagnose the accident and/or indicate the status of core cooling. This vital instrumentation must meet Environmental Qualification (EQ) and Seismic Qualification (SQ) standards.

Review Question 24

Match the Critical Safety Function (CSF) to the component it protects.

Core Cooling	A. Cladding
RCS Integrity	B. RCS Piping
Containment Integrity	C. Containment

The answer sequence is ABC.

Conclusion

In this lesson, you learned about design basis accidents (DBAs) in PWRs and BWRs. We also discussed the ECCS Acceptance Criteria used to ensure that core cooling is maintained and provide for the safe shutdown of the reactor. We identified the sources and dangers surrounding hydrogen, as well as the four accident classifications. We also identified the six critical safety classifications and described the critical parameter monitoring during accident conditions. You should now have an appreciation of the effort required to protect the reactor core from the effects of transients and accidents.

Now that you have completed this lesson, you can do the following:

- Describe DBA-LOCA
- Identify the ECCS Acceptance Criteria IAW 10CFR50.46
- Identify the sources and dangers of hydrogen
- Describe the four accident classifications
- Identify the six (6) critical safety functions
- Identify critical parameters monitored during accident conditions

In the next lesson, you will learn about the plans made to ensure on-site and off-site health and safety.

6
SITE EMERGENCY PLANS

Introduction

In general, for an accident to occur, weaknesses or shortcomings must occur in the plant's design, construction, or operations. The primary concern in any core protection scheme is ensuring the health and safety of the public and minimizing radiological doses to radiation workers. To accomplish these two goals, Emergency Planning Organizations have been established to function on and off-site.

After you have completed this lesson, you will be able to:

- Describe radiological hazards and monitor response
- Describe the effects of elevated radiation levels and radioactive releases following an accident on the protection of the public and site workers
- Describe the recovery process for design basis accidents

A radiation accident or incident, as it is sometimes called, may be defined as an unintentional or unexpected event resulting in an individual's radiological exposure or physical injury or physical damage to property.

The relative magnitude of radiological incidents can be classified on the basis of human dose resulting from the type and concentration of radioactive material involved. An incident involving radiation and radioactive material may result in:

- Environmental contamination
- Human irradiation
- Human inhalation or ingestion of radioactive materials

All emergency planning assumes that such an accident would release radioactive material produced in the reactor into the atmosphere over a wide area. Possible radiation exposure paths include:

- Inhalation: exposure to radioactive materials via the atmosphere
- Ingestion: exposure to radioactive materials via the food chain
- Absorption: exposure to radioactive materials when in contact with the individual

The responsibility for protecting the public during accident conditions falls on the Emergency Response Organization at each nuclear site, which is responsible for implementing its own emergency plan.

A radiological accident can have two major effects on the environment and its inhabitants:

Surface contamination
The deposition of radioactive material on the land and water surfaces, which can result in:
- Contamination of land used for agriculture
- Contamination of food chains, of which humans are a part
- Contamination of water reservoirs or watersheds, requiring the abandonment of their use by humans

Personal injury
Released radioactive material may cause radiation injuries from external exposure or internally from the ingestion, inhalation, or absorption of the radionuclides. The extent of exposure (external and internal) and injury (long term [cancer] or short term) depends on the nature of the released fission products.

Review Question 25

Which of the following are potential effects of a radiological accident?

A. Surface contamination
B. Personal injury
C. Loss of core geometry
D. Emergency planning

The correct answers are A and B. Surface contamination and personal injury are two major effects of a radiological accident.

Emergency Planning

As a result of the Three Mile Island accident, emergency planning requirements were greatly expanded, with the local and state emergency plans becoming an intimate part of the overall emergency preparedness requirements. There are two identified zones used in emergency planning:

- Plume exposure zone: Extending out to 10 miles from the reactor, it is concerned with **direct radiation exposure to the public**.
- Ingestion zone: Extending out to 50 miles from the reactor, it is concerned with **ingestion of radioactive material through the food chain**.

Emergency Classification Levels (ECLs) are entered by meeting the Emergency Action Level (EAL) Threshold Values or a potential or actual radiological sabotage event. There are four different ECLs:

- Notification of Unusual Event
- Alert
- Site Area Emergency
- General Emergency

Notification of Unusual Event

- Potential degradation of the plant's level of safety and escalation to more serious emergencies
- No releases of radioactive material requiring off-site response or monitoring
- Can lead to escalation of a more serious event

Alert

- Actual or potential substantial degradation of the level of safety of the plant
- Any releases are expected to be limited to small fractions of the EPA Protective Action Guideline exposure levels
- Lowest level at which the Technical Support Center (TSC) will be staffed
- For example, an alert may be declared due to a confirmed fire in a Safe Shutdown Building, where plant personnel reported visible damage to the building or the equipment contained within it.

Site Area Emergency

- Actual or likely major failures of plant functions needed for protection of the public
- Any releases are not expected to exceed EPA Protective Action Guideline exposure levels beyond the site boundary
- A DBA-LOCA is an example of an event that could initiate a site area emergency
- Lowest level at which a site assembly and accountability of site personnel and relocation of non-essential personnel

General Emergency

- Actual or imminent substantial core degradation or melting with potential for loss of containment integrity
- Releases reasonably expected to exceed EPA Protective Action Guideline exposure levels off-site beyond the immediate site area

Review Question 26

Rank order these Emergency Classification Levels from the least to most severe with 1 being the least severe and 4 being the most severe.

Site Area Emergency
Alert
Notification of Unusual Event
General Emergency

The least severe Emergency Classification Level is Notification of Unusual Event, followed by Alert, Site Area Emergency, and finally, the most severe ECL, General Emergency.

Potential Radiation Levels Outside Exclusion Zone

The effect of every accident on radiation levels outside the Exclusion Area Boundary (EAB) is different. The job of the accident analysis in the UFSAR is to evaluate the worst of these accidents and ensure that the radiation effects are within acceptable limits.

For all transients of moderate and infrequent frequencies, the Nuclear Regulatory Commission has defined unacceptable (or non-acceptable) safety results, one of which relates to off-site radiation doses:

- Release of radioactive material to environs in excess of 10CFR20 limits

There are three additional non-acceptable results for transients that apply to engineering work, but not dose rates:

- Reactor operation-induced fuel clad failures
- Nuclear system stresses in excess of those allowed for in the transient classification by applicable industry codes
- Containment stresses in excess of those allowed for in the transient classification by applicable codes

Unusual Event and Alert actions are taken before reaching the off-site dose limits of 10CFR20.

Potential Radiation Levels Outside Exclusion Zone: DBAs

For design basis accidents (limiting faults), the NRC has also defined unacceptable safety results. One of these again relates to off-site dose:

- Release of radioactivity resulting in dose consequences in excess of 10CFR100 values

Four more non-acceptable results apply to engineering work, but not off-site dose rates. These are:

- Failure of fuel cladding sufficient to cause changes in core geometry such that core cooling would be inhibited
- Nuclear system stresses in excess of those allowed for the accident (faulted) classification by applicable industry codes
- Containment stresses in excess of those allowed for the accident classification by applicable codes when containment is required as a barrier
- Radiation exposure to plant operations personnel in the main control room in excess of 5 rem whole body, 30 rem inhalation dose, and 75 rem skin dose

Review Question 27

What is the job of the accident analysis in the UFSAR?

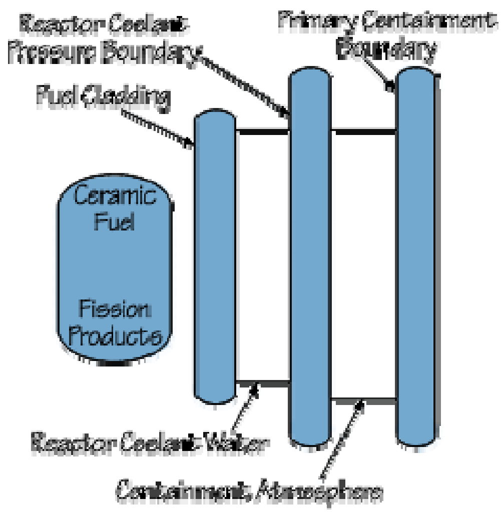
A. To check core temperature increases and decreases
B. To check core geometry
C. To evaluate the worst radiological accidents and ensure that radiation effects are within acceptable limits
D. To ensure that emergency plans are in place for an accident

The correct answer is C. The job of the accident analysis in the UFSAR is to evaluate the worst radiological accidents and ensure that radiation effects are within acceptable limits.

Evacuation Criteria: Public

Evacuation of the public will only occur if a General Emergency is declared. The public will be alerted within the Emergency Planning Zone (typically, a 10 mile radius for inhalation and 50 mile radius for ingestion) to take shelter and monitor local radio stations.

Evacuation will be based on the status of the three fission product barriers: fuel pellet and cladding, Reactor Coolant Pressure Boundary, and Containment.



Evacuation Criteria: Personnel

Site Area Emergency sets the evacuation criteria for on-site personnel. The radiation levels to enter this Emergency Classification level are, like the General Emergency, based on off-site dose levels.

All on-site personnel will assemble and any personnel not part of the Emergency Response Organization will evacuate to an area designated by the Emergency Director (usually based on wind direction, et al.) and await further direction. These areas will generally be in contact with the site to ensure that personnel necessary to complete recovery activities are available.

Each nuclear power plant emergency plan specifies the on-site emergency organization of plant staff personnel for all shifts and their relationship to the responsibilities and duties of the normal staff component. Specific positions and major tasks to be performed by personnel assigned to the functional areas of emergency response are clearly defined. Specified types of individuals are qualified for emergency positions based on expertise, experience, and training.

In general, the utility emergency organization is comprised of individuals housed in one of the following four centers:

Control Room
The initial emergency response organization is comprised of Control Room and other shift coverage staff personnel. The Shift Manager serves as the Emergency Coordinator, while control room staff personnel are responsible for initial first aid, fire fighting, rescue, damage control, and radiation monitoring.

Technical Support Center (TSC)
The TSC is activated for an Alert or any higher emergency classification. The Emergency Coordinator will relieve the Shift Manager of emergency coordination duties and reside in the TSC until the Emergency Operations Facility (EOF) is fully activated, allowing the Shift Manager to concentrate on plant operations. The TSC will provide necessary assistance to the Control Room staff on matters of recovery from the emergency.

Operations Support Center (OSC)
The OSC provides support in the form of corrective actions and recovery efforts as directed by the Control Room, TSC, or EOF.

Emergency Operations Facility (EOF)
The EOF houses the support personnel whose primary responsibility is to assume administrative burdens from the Control Room and TSC allowing them to focus on recovery efforts. The Emergency Coordinator resides in the EOF and is responsible for emergency response.

Anticipated Radiation Levels: Off-Site

The radiological consequence limits to the public are specific to 10CFR20 and within the guidelines of 10CFR100, depending on the severity of the incident.

Some plants have adopted a more realistic analysis called the “Alternate Source Term.” It takes into account more than 40 years of experience in setting limits. Off-site dose limits will be further limited by 10CFR67. If you become a member of the Emergency Response Organization, additional training will be provided.

Anticipated Radiation Levels: On-Site

In the event of a large scale LOCA and subsequent fuel damage, radiation levels throughout the plant will increase. Fission product release into the containment will result in extremely high radiation levels. Radiation levels in other site buildings will also be extremely high due to reactor coolant running through systems.

US Nuclear Regulatory Commission Regulation (NUREG) 0737 requires that reactor coolant and containment samples be obtained and analyzed within hours after an accident. Installed radiation monitors – liquid, process, and area - and manual sampling will meet these requirements.

Post-Accident Monitoring Instrumentation is installed and maintained to assist in post-accident assessment of plant and radiological conditions. NUREG 0737 requires that these instruments be ranged for the worst anticipated accident and that they be environmentally qualified to withstand the heat, humidity, and pressure of the post-LOCA containment atmosphere.

Following the Three Mile Island accident, three major areas of the Auxiliary Building were at radiation levels of 1000 R/hour.

Containment

Containment is the fission product barrier assumed to fail last in the DBA-LOCA. Containment leakage or failure may release radioactive materials to the environment.

Even without leakage, there is a potential for significant dose rates from the containment. For example, a typical PWR containment building wall reduces dose by a factor of 10,000. If the radiation level inside the containment is 16,000 R/hour (one plant's LOCA estimate with 1% melted fuel), the dose immediately outside the containment would be about 1.6 R/hour. This example assumes a case beyond the design basis of the plant. For most nuclear sites, the DBA-LOCA does not produce a fuel failure. In this case, dose rates in the containment would be a few rem per hour with negligible doses outside the containment.

Recovery Process

Recovery is the final phase of the Emergency Plan when major repairs are performed to return the plant to an acceptable condition while no possibility of an emergency condition exists. Once the plant has been stabilized, and issues contained and controlled, the Recovery classification can be declared. Corporate will head the recovery organization.

Recovery is a phase of the emergency planning process entered into by meeting emergency termination criteria included in Emergency Response Procedures. Emergency Response Procedures are developed to provide:

- Event-related actions to address the accident that is occurring
- Function-related responses to ensure the fission product barriers are maintained

The optimal recovery concept is based on minimizing radiation release and equipment damage by implementing an event-related recovery strategy that consists of:

- General Procedures and Operating Procedures for normal operations
- Abnormal Procedures exist to address alarms
- Emergency Procedures are used at a minimum when a reactor trip has occurred

Review Question 28

Recovery is the final phase of the Emergency Plan when major repairs are being performed to return the plant to an acceptable condition while no possibility of an emergency condition exists.

True or False?

Answer: True. Recovery is the final phase of the Emergency Plan when major repairs are being performed to return the plant to an acceptable condition while no possibility of an emergency condition exists.

Conclusion

As you can see from this module, nuclear power plants are designed to mitigate the effects of design basis accidents to ensure the health and safety of the public as well as nuclear workers. As part of that protection, worst case accidents (based on radiological release) are analyzed and the Emergency Preparedness Organization is in place to ensure the dose to the public and nuclear workers is minimized.

Accidents are never supposed to happen. Three Mile Island and Chernobyl have taught us that we must be prepared, just in case.

Now that you have completed this lesson, you can:

- Describe radiological hazards and monitor response
- Describe the effects of elevated radiation levels and radioactive releases following an accident on the protection of the public and site workers
- Describe the recovery process for design basis accidents

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