Deterministic Safety Analysis

Prof. S.A. Jonah

Centre for Energy Research and Training,
Ahmadu Bello University, P.M.B. 1014 Zaria, Nigeria

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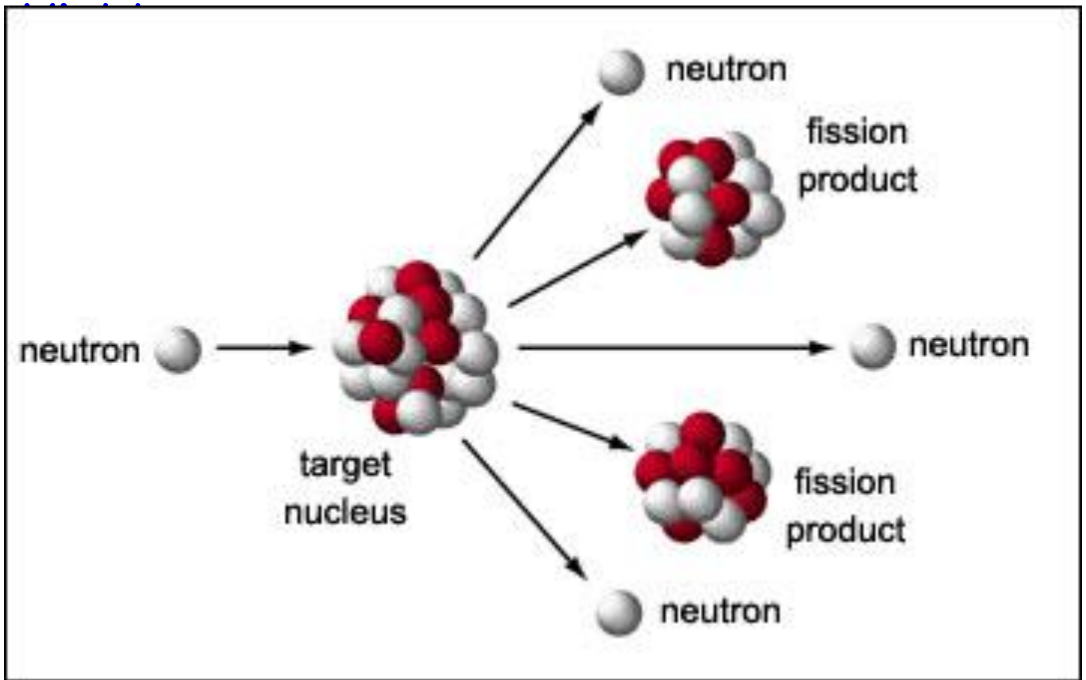
1. Introduction

- Why do we need Safety Analysis ?

  - Nuclear Reactions result in Radioactive Materials
    - As a result of nuclear fission reaction, various Radioactive Fission Product Isotopes are produced

Fission Reaction



1. Introduction

Radioactive Fission Product

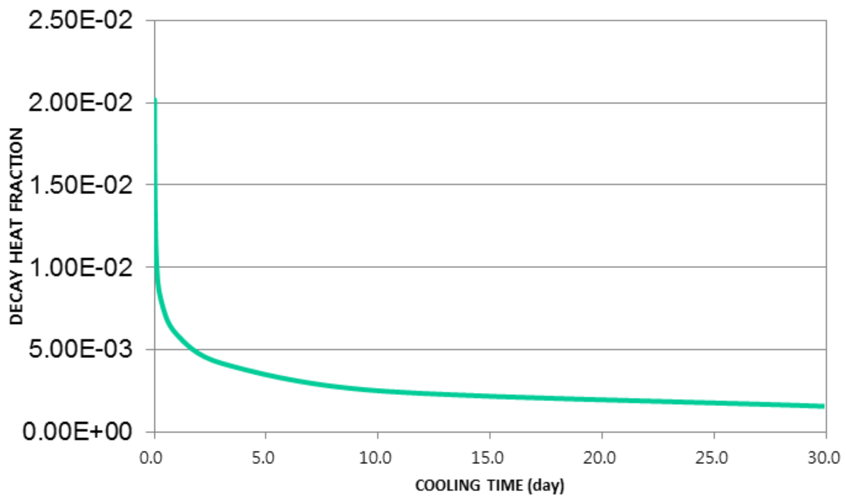
- In the reactor core, Fuel Materials & Fission Products are bombarded by neutrons & other radiations and transformed into other isotopes
- A lot of Fission Product Isotopes are unstable and radioactive
- Structural materials in the core are activated by neutrons and other radiations and become radioactive
- Radioactive Fission Product should be kept inside the fuel rods otherwise it contaminates reactor coolant system and could be leaked out to the environment
- Multiple barriers are used to prevent the release of radioactive material from the Nuclear Reactor

1. Introduction

Fission Product generates Decay Heat

- Various Fission Product Nuclides are unstable and tend to decay into other nuclides until the daughter nuclides are stable
- Unstable Nuclides are emitting various energetic radiations
- In the fuel rods, most of the Radiation Energy from Fission Product turns into heat which should be removed by proper cooling for a prolonged period even after the reactor shutdown

Fission Product Decay Heat



1. Introduction

- Decay Heat Cooling
  - The evaluation of the decay heat cooling capability is important in Safety Analysis
  - Loss of reactor coolant, loss of ultimate heat sink or loss of cooling power results in insufficient decay heat cooling and can lead to the fuel failure
  - Fuel failure means loss of the first barrier against the release of radioactive material

Core of TMI-2 →

1. Introduction

- What is Safety Analysis ?
  - Safety analyses are analytical evaluations of physical phenomena occurring at nuclear power plants, made for the purpose of demonstrating that safety requirements, such as the requirement for ensuring the integrity of barriers against the release of radioactive material and various other acceptance criteria, are met for all postulated initiating events that could occur over a broad range of operational states, including different levels of availability of the safety systems. [IAEA SSG-2]

1. Introduction

- Deterministic Safety Analysis versus Probabilistic Safety Analysis
  - Deterministic Safety Analysis is to assess the consequence of postulated initiating events in nuclear facilities
  - Probabilistic Safety Analysis is to identify the major contributors to the risk of damaging the plant and exposing the workforce and the public to radiation and demonstrate that the design provides a well balanced protection against possible accidents

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1. Introduction

- The objective of **DSA** is to confirm that **safety** functions can be fulfilled and that the necessary SSCs, in combination with operator actions, are effective in keeping the releases of radioactive material from the plant below acceptable limits.

- While **PSA** is an additional tool that was introduced. Also known as a **probabilistic risk assessment**, the **PSA** uses **probabilities** to analyze the overall **risk** to a **nuclear power plant** under abnormal conditions

1. Introduction

- Deterministic Safety Analysis includes
  - Evaluation of the performance of reactor protection system, engineered safety features, and adequacy of emergency operating procedures
  - Dynamic Behavior of Reactor, Fuel Rod Integrity, Thermal-hydraulics of Core, RCS and Containment for the Postulated Initiating Events
  - Analysis to confirm the Operability and Integrity of SSC Contributing to the Safety Function
  - Analysis should extend up to the moment when the plant reaches a safe and stable end state

1. Introduction

- Plant Characteristics Considered

  - Range of values for plant parameters should be representative and sufficiently broad to cover all cycles to the extent practical
  - Parameters : core power, core inlet temperature, reactor system pressure, core flow, axial and radial power distribution, fuel and moderator temperature coefficient, void coefficient, reactor kinetics parameters, available shutdown rod worth, and control rod insertion characteristics, etc

2. Accident Categorization

- Categorization of Transients and Accidents:
  - Initiating events and the consequential transients should cover all possible scenarios including operator error, equipment failure and natural events
  - Categorization by Frequency of Occurrence
  - Anticipated Operational Occurrence (AOO)
    - Expected to occur once or more during the plant lifetime
    - Loss of normal power
    - Loss of power to the reactor coolant pump
    - Turbine trip
    - Failure of control equipment
    - Loss of normal feed water
    - Loss of condenser cooling

2. Accident Categorization

- Categorization (cont)

- Postulated Accidents
  - Not expected to occur during the plant lifetime
    - Major rupture of a pipe containing reactor coolant up to and including double-ended rupture of the largest pipe in the reactor coolant pressure boundary
    - Ejection of a control rod assembly
    - Control rod drop accident (BWR)
    - Major secondary system pipe rupture up to and including double-ended rupture
    - Single reactor coolant pump locked rotor

2. Accident Categorization

- Categorization by Type of Phenomena

  (1) Increase in heat removal by the secondary system

  (2) Decrease in heat removal by the secondary system

  (3) Decrease in RCS flow rate

  (4) Reactivity and power distribution anomalies

  (5) Increase in reactor coolant inventory

  (6) Decrease in reactor coolant inventory

  (7) Radioactive release from a subsystem or component

2. Accident Categorization

- Computational Analysis on limiting cases for each Category
  - Bounding or enveloping scenario from each category
  - Greatest possible challenge to the acceptance criteria
  - Limiting case for the performance of safety related equipment
- BDBA/DEC are treated separately
  - Results of DEC analysis help to determine the necessary measures to prevent severe accidents and to mitigate the radiological consequences

3. Acceptance Criteria

- Defined as limits and conditions set by a regulatory body to achieve an adequate level of safety
- The individual/collective doses to workers and the public are required to be within prescribed limits and as low as reasonably achievable (ALARA) in all operational states by mitigating the radiological consequences of any accident
- The integrity of barriers against the release of radioactive material (fuel itself, fuel cladding, primary/secondary reactor coolant system, containment) should be maintained, depending on the plant states
- The capabilities of systems and operators intended to perform a safety function, directly or indirectly, should be ensured for the accidents for which safety function is required.

3. Acceptance Criteria

- Acceptance criteria should be established for the entire range of operational states and accident conditions.
- Acceptance criteria may be related to the frequency of the event. Events that occur frequently, such as anticipated operational occurrences, should have acceptance criteria that are more restrictive than those for less frequent events such as design basis accidents.
- Acceptance criteria should be set in terms of the variable or variables that directly govern the physical processes that challenge the integrity of a barrier. Surrogate variables can also be used as acceptance criterion that, if not exceeded, will ensure the integrity of the barrier (DNBR, Pellet Enthalpy Rise, etc.)
- Compliance with the single failure criterion should be evaluated for each safety system in the plant

4. Safety Analysis Model

- Protective Actions and Safety Systems Actions
  - Listing of setting of all protection or safety system function used (reactor trips, isolation valve closures, ECCS initiation, etc)
  - Inclusion of the most limiting single failure
  - Limiting delay time for protection safety system function used (calibration error, drift, instrumentation error, etc)
- Single Failure Criterion
  - Redundancy in safety system is essential to minimize the possibility of loss of the safety function
  - Single Failure is assumed in accident analysis
  - One control rod with maximum worth is assumed to be stuck out of the reactor core, in spite of reactor trip signal

4. Safety Analysis Model

- Initiating Events

- Examples of similar events
  - Decrease in feedwater temperature,
  - Increase in feedwater flow
  - Increase in steam flow (Limiting)
  - Inadvertent opening of a steam generator relief or safety valve

Increase heat transfer from the primary side to secondary

4. Safety Analysis Model

- Event Evaluation
  - Sequence of Events and Systems Operation
    - Step-by-step from initiation to finalized condition (e.g. occurrence, sensor trip, insertion of control rods, attainment of safety valve setpoint, opening/closing of safety valve, generation of containment isolation signal, containment isolation, operator action credited, etc)
    - Extent to which normal operating plant I&C assumed and reactor protection system required
    - Credited operation of engineered safety systems
    - Use only safety-related system

4. Safety Analysis Model

- Parameters
  - Nuclear design
    - control rod worth, rod insertion time, shutdown margin
    - control temperature feedback coefficients (fuel, moderator)
    - power distribution (radial, axial)
    - decay heat
    - fission product inventory
    - delayed neutron fraction
  - Fuel
    - thermal conductivity (pellet, gap, cladding)
    - gap fraction of fission product
    - fuel and cladding dimension

4. Safety Analysis Model

- Parameters (cont)
  - RCS
    - coolant pressure/temperature
    - coolant inventory (Pressurizer level, charging flow, letdown flow)
    - Pressurizer safety valve open/close setpoints
  - Main steam system
    - coolant inventory (SG water level, feedwater flow rate)
    - Steam pressure/temperature
    - main steam safety valve open/close setpoints
  - Instrumentation and control system
    - Process time including delay in instrumentation and actuation

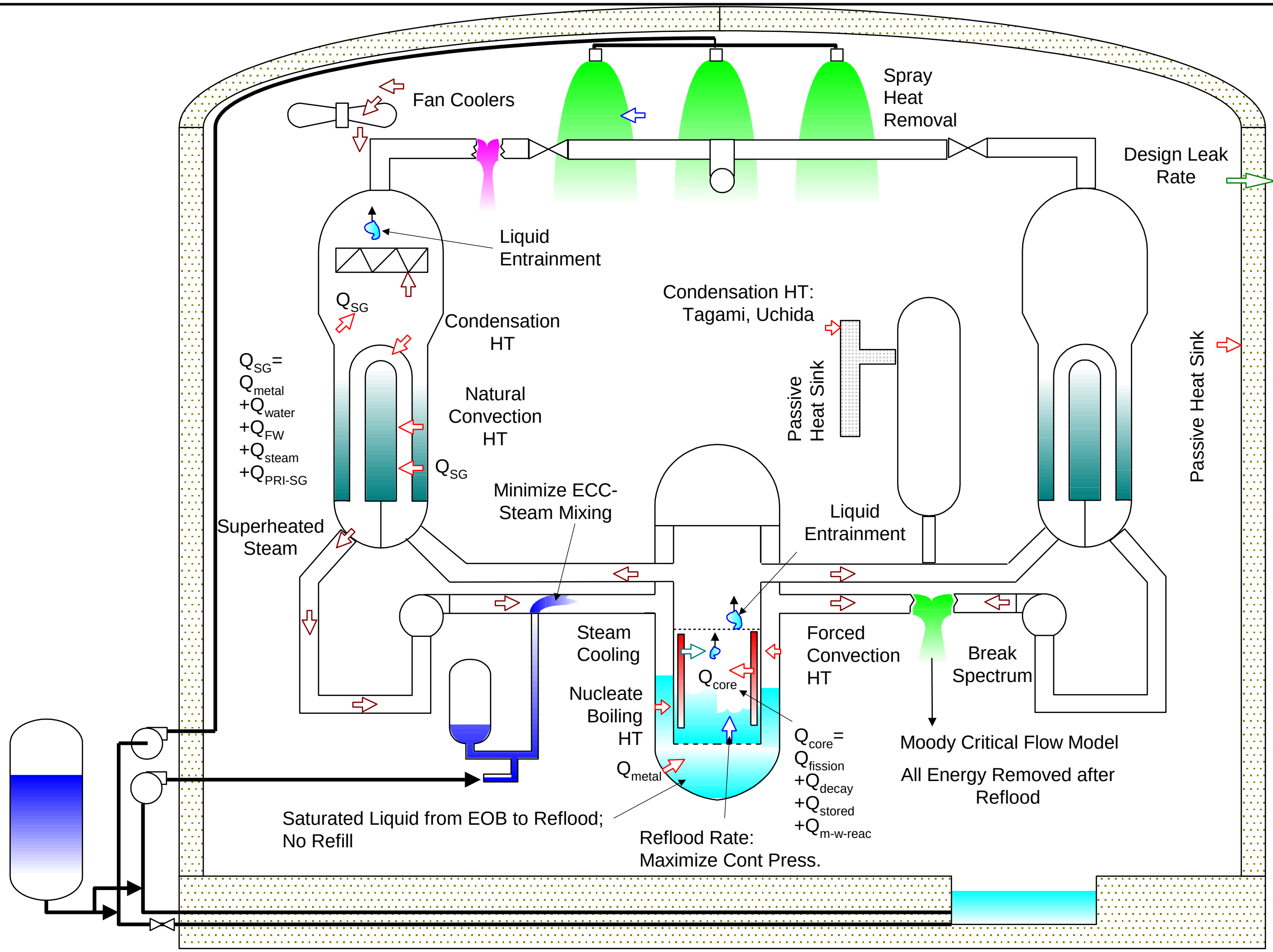
4. Safety Analysis Model

- Mitigating Systems include:
  - Reactor protection system
  - Safety injection system
  - Auxiliary feedwater system
  - Overpressure protection system
  - Main steam/feedwater isolation system
  - Emergency diesel generators
  - Reactor containment system

5.Containment Analysis

- Analysis long-term behavior of Containment Systems following LOCA
- Calculates the time variation of compartment pressures, temperatures, mass and energy inventories, heat structure, temperature distributions, and energy exchange with adjacent compartments.
- Calculates the effects of leakage on containment response, fan cooler and cooling spray systems.
- compartments can be modeled
- Computer Code for Korean LWR : CONTEMPT-LT/028 Idaho National Engineering Lab. Sponsored by USNRC, NUREG/CR-0255, 1975

5.Containment Analysis



6. Radiological Consequence Analysis

- Radiological consequences of an accident
  - Licensing Criteria : 25 rem for whole body, 300 rem for thyroid at Exclusion Area Boundary
  - Quantity of the radioactive material that escapes to the environment or enters the control room.
  - Credit for several natural and engineered removal mechanisms. (sprays, natural deposition, leakage, natural and forced convection, filters)
  - Atmospheric Dispersion depends on the meteorological condition at the time of accident

6. Radiological Consequence Analysis

- Fission Product Inventory
  - maximum full-power operation plus uncertainty to maximize FPI
  - ORIGEN2, ORIGEN-ARP
  - DBA LOCA : all fuel assemblies in the core assumed to be affected
  - Others: damaged fuel rod inventory
- Release Fractions (Reg. Guide 1.195)
  - DBA LOCA or Non-LOCA Fuel Fail : Noble Gases 1.0, Iodines 0.5
  - Non-LOCA : I-131(0.08), Kr-85(0.1),
    Other Iodines(0.05), Other Noble Gases (0.05)
- Timing of Release Phases
  - DBA LOCA : Immediately, Non-LOCA DBA : Time of Fuel failure
- Radionuclide Composition
  - Noble Gases(Xe, Kr) and Iodines: I(Par. 5%, Ele. 91%, Org. 4%)

6. Radiological Consequence Analysis

- Control Room Dose
  - whole body, thyroid, and skin dose
  - Sources:
    - Contamination of the control room envelope atmosphere by the intake or infiltration of the radioactive material contained in the radioactive plume released from the facility
    - Contamination of the control room envelope atmosphere by the intake or infiltration of airborne radioactive material from areas and structures adjacent to the control room envelope

Please Stay Safe &
Thanks for your attention !