Lecture 12: Safety codes (Like, programs and shit)

Tuesday, April 2, 2024 18:13

Nuclear Power Plant Design
MECH 4106 A

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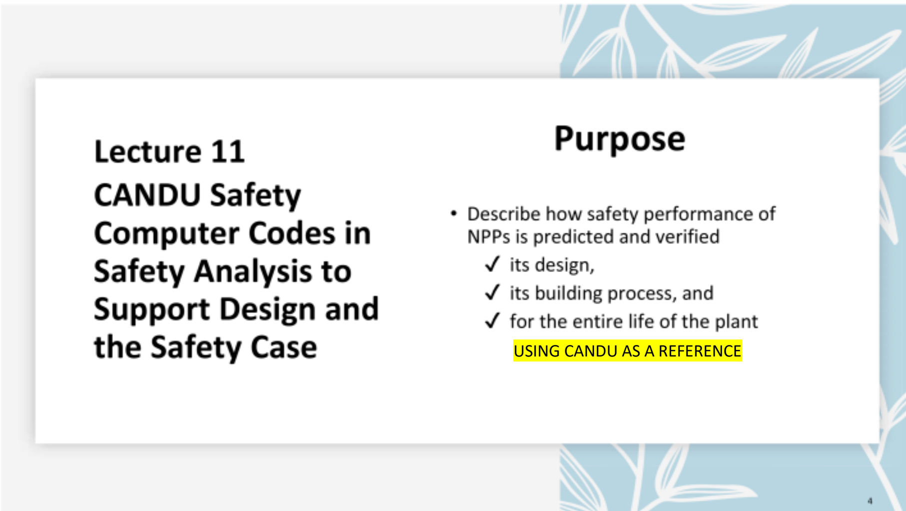
Lecture 11
CANDU Safety Computer Codes in Safety Analysis to Support Design and the Safety Case


Lecture 11
CANDU Safety Computer Codes in Safety Analysis to Support Design and the Safety Case
Purpose
• Describe how safety performance of NPPs is predicted and verified
✓ its design,
✓ its building process, and
✓ for the entire life of the plant
USING CANDU AS A REFERENCE
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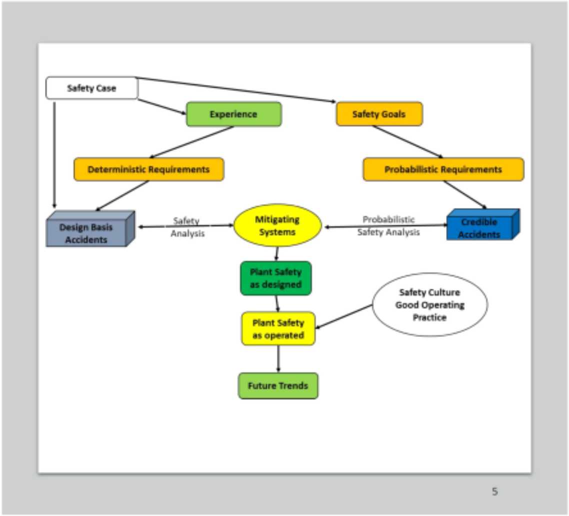
How Safe is the Safety Case?

- General Overview
  - REGDOC 2.5.2 and plant envelope
  - Presented in another form of a road map.
  - Analyst has performed it to achieve the end goal of complying with REGDOC 2.5.2



Recap: Plant Design Envelope (REGDOC 2.5.2)

Plant state	Plant Design Envelope				
	NO	AOO	DBA	Design extension conditions (DECs)	
					Severe accidents
Classification Frequency	1	1 to 10^{-2}	10^{-2} to 10^{-5}	$<10^{-5}$	
Public Radiological Acceptance Criteria	Effective dose limits as per the Radiation Protection Regulations lowest	Dose acceptance criteria 0.5 mSv	Dose acceptance criteria 20 mSv	Safety goals (CDF < 10^{-5}, SRF < 10^{-5}, LRF < 10^{-6}) and deterministic requirements	
Structures, Systems, Components	Normal operation and control systems		Safety systems	Complementary design features	Complementary design features
Safety Analysis	Deterministic safety analysis, probabilistic safety assessment and hazards analysis are performed				
Design Rules	Design basis Rules			Reasonable level of confidence	
Operator procedures	Operating manuals		Emergency operating procedures		Severe accident management guidelines
			Emergency management procedures		
Off-Site Response	None required		Graded response		Fully mobilised

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How Safe is the Safety Case?

- **Safety Case**
  - ✓ the risk from nuclear reactors
  - ✓ radioactive material is in the reactor fuel/core
  - ✓ risk can be quantified
  - ✓ society can tolerate a level of risk that is reasonable
  - ✓ Epidemiological studies
  - ✓ Dose limits

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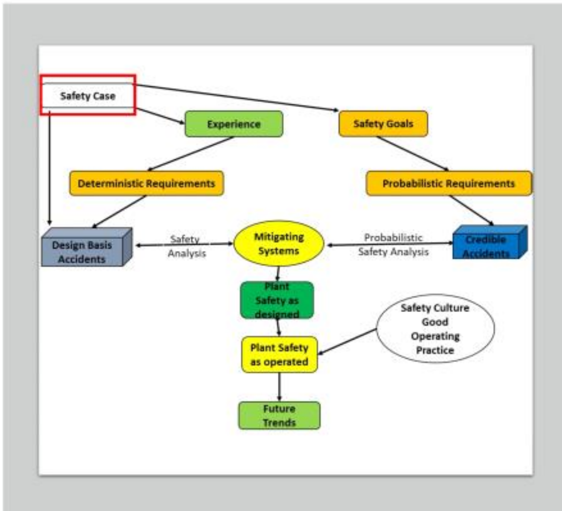

How Safe is the Safety Case?

- **Experience**
  - ✓ Risk from nuclear power reactors from research and operation of power reactors.
  - ✓ Events concerning reactor safety is important

Past Events:
- 3 Mile Island (TMI)
- Chernobyl
- Fukushima

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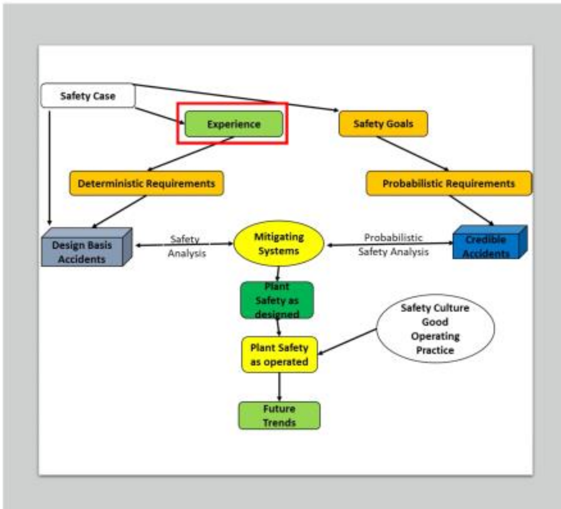

3 Mile Island: They thought the reactor could not fail, but were in an un-analyzed situation
Chernobyl: Design flaw and malpractice
Fukushima:

How Safe is the Safety Case?

- **Deterministic Requirements**
  - ✓ Lessons learned from experiences are used to form deterministic requirements, which describe the accidents to be designed for and the assumptions used in showing the safety systems were effective
  - ✓ became design basis accidents. (DBAs).

Design Upgrades:
- Fail safe designs
- Inherent - walk away safety

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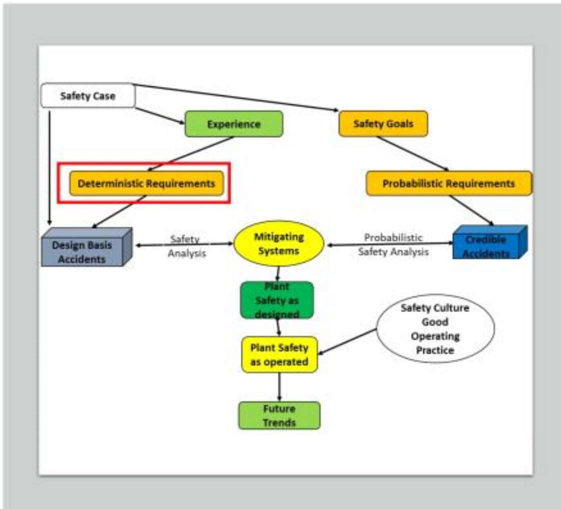

How Safe is the Safety Case?

- **Design Basis Accidents (DBA)**
  - ✓ Accidents are postulated
  - ✓ allow radioactivity to escape and
  - ✓ design systems (called mitigating systems or safety systems) to prevent or control such postulated accidents.
- ✓ Approach limits risk
- ✓ does not quantify the risk
- ✓ Deterministic requirements and the DBAs are chosen "conservatively",

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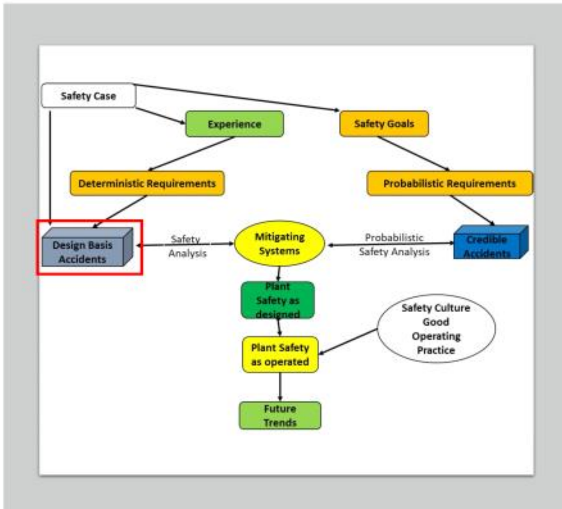

Conservatism:
- Accidents are postulates as those that challenge the system the most
- Large margins to wailure

Design Basis Accidents

- Concept of stylized accidents (DBA) used in the plant
- DBAs are the set of accidents for which the designer makes explicit provision (defence), while remembering that more severe or peculiar accidents can occur and ensuring that his/her design has some capability to deal with them.
  - ✓There is no way of identifying possible accidents beforehand
  - ✓Technology is replete with unpleasant surprises, especially at the beginning
  - ✓Technologies could have their accidents early on (Boeing 737 MAX)
  - ✓Nuclear Power Plants have had their share too
    - TMI-3, Chernobyl, Fukushima

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Question from Glenn: What is the difference between a frequency and a probability?

It's the units.

Conjoined probabilities: You can multiply the probability of different events

YOU CAN NOT MULTIPLY THE FREQUENCY OF DIFFERENT EVENTS.

How Safe is the Safety Case?

- **Safety Goals**
  - ✓ Set of numerical risk targets for the plant as a whole (safety goals).
  - ✓ Possible accidents are identified and classified using a frequency-based approach.
  - ✓ parallel approach to deterministic requirements

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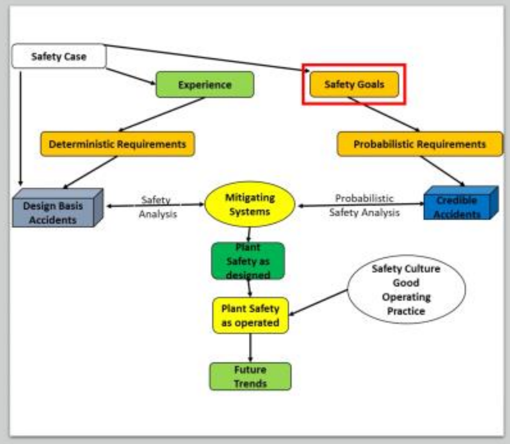

CNSC Safety Goals and Risk Assessment

Safety Goals

- Quantifying risk, and present probabilistic safety assessment tools used to determine if plant meets safety goals.

Safety Goals - "how safe is safe enough?"

- "safe as possible" mean different when no guidance is given to designer.
- "The reactor must never have a severe accident" - physically impossible and expectations that cannot be complied with

CANDU -

- Recall: the 3 safety goals (REGDOC 2.5.2)
- (1)) Core damage (2) Large Release Frequencies (3) Small Release Frequencies

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How Safe is the Safety Case?

- **Mitigating Systems**
  - Systems can mitigate both DBAs and accidents identified by the probabilistic approach.
  - Methods confirm the effective systems are:
    - ➢ Probabilistic Safety Analysis (PSA)
    - ➢ Deterministic Safety Analysis (DSA)

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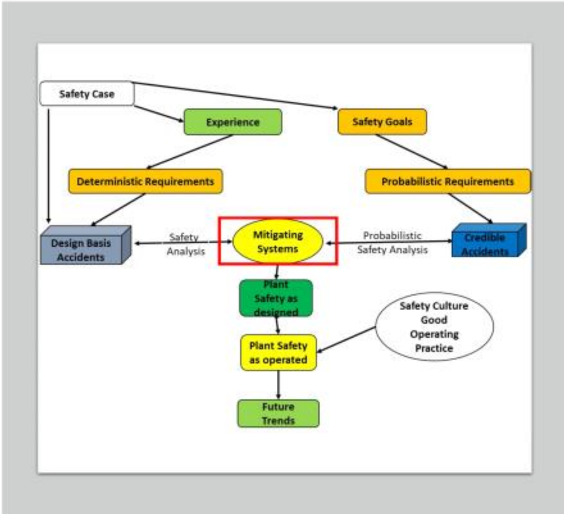

Mitigating Systems

CANDU reactors for our examples, although other reactor types have similar systems.

- ✓ How much redundancy?
- ✓ What independence for each safety function?
- ✓ Defence-in-depth is a key in nuclear safety**, Four safety functions** required in a nuclear reactor:

- Mitigate
  - ✓ shut down
  - ✓ remove decay heat
  - ✓ contain radioactive material
  - ✓ monitor state of the plant

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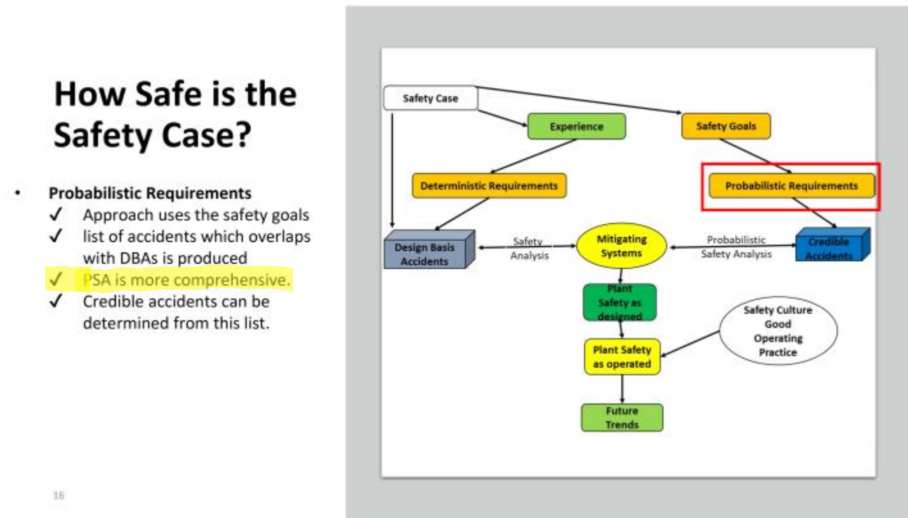

Lame example: Preventing the case where you open a door and hit somebody with it, then mitigate that probability by, say, adding a window to it.

In a plant, if you identify a vulnerability, you add redundancies upon redundancies and safety systems among safety systems.

Defence in Depth focuses on preventing and remediating cases where you have a Single Point of Vulnerability (SPV)

BUT sometimes it's too expensive, and it's worth increasing maintenance.

How Safe is the Safety Case?

- **Probabilistic Safety Analysis (PSA)**
  - ✓ Drives design requirements
  - ✓ verifies the reliability of, normal and mitigating systems.
  - ✓ It also ensures to the extent practical, that an accident which requires a mitigating system does not also impair it
    - it uses Boolean algebra
    - it combines these frequencies with the reliability of mitigating systems to determine the frequency of severe accidents,
    - Shows safety goals have been achieved.

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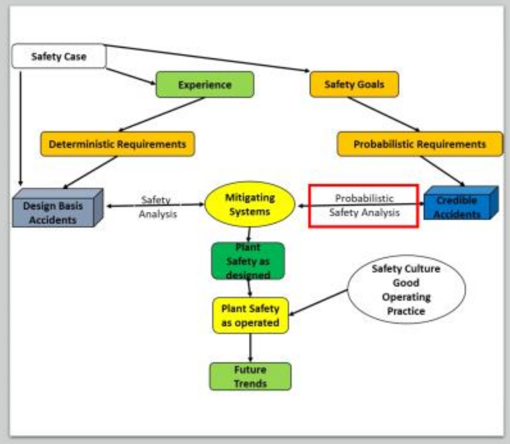

Note: Turns the frequencies into probabilities, AND THEN combines them.

Uses CAFTA Model???
(Computer Assisted Fall Tree Analysis

How Safe is the Safety Case?

- **Deterministic Safety Analysis (DSA)**
  - ✓ "Safety analysis", drives design requirements
  - ✓ verifies the performance of, mitigating systems.
  - ✓ It uses computer codes/analytical tools to model all the key systems in a plant
  - ✓ DSA demonstrates deterministic requirements have been met.

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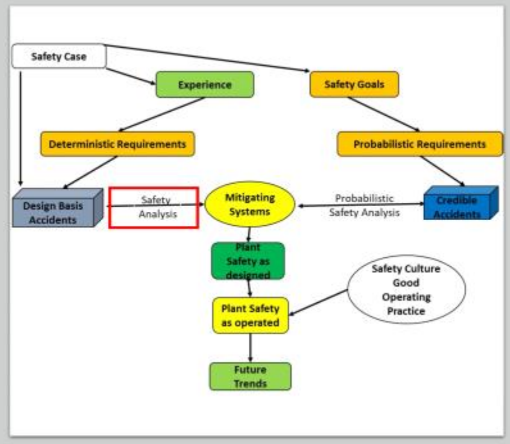

How Safe is the Safety Case?

Lecturer Annotation in Lieu of Live lecture

- **Safety Analysis**
  - ✓ It describes the phenomenon in accidents
  - ✓ Summarizes the mathematical tools used to predict how they evolve (explained later)

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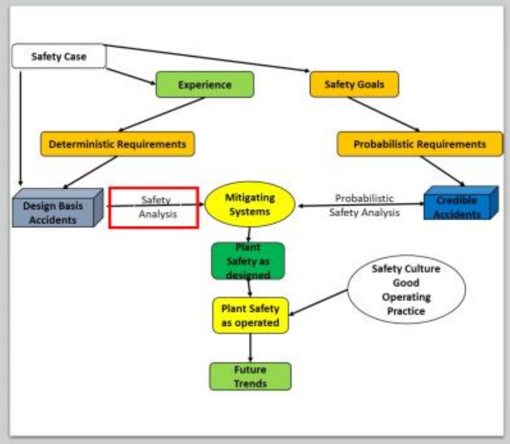

How Safe is the Safety Case?

- **Plant Safety As Designed**
  - ✓ Safety depends on the people who run it.
    - safety culture is implemented
    - Innovative future designs to deliver increased safety to plant operation

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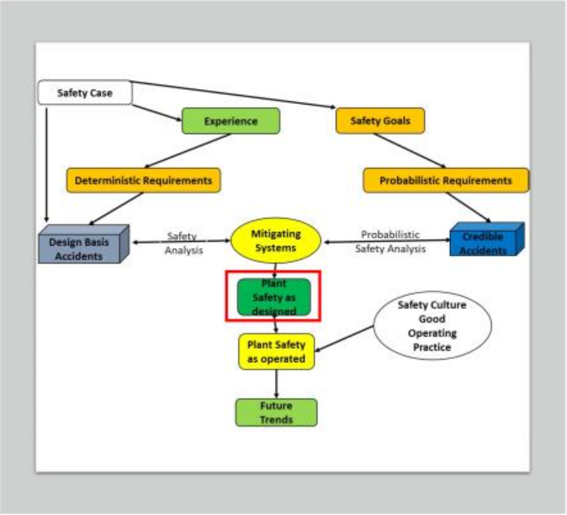

Safety Analysis–Accident Phenomenology

Some Accidents by Phenomena

- Reactivity accidents
- reactor coolant inventory
  - ✓Typically called LOSS OF COOLANT ACCIDENTS (LOCA)
- reactor coolant pressure transients
- Pressure control (increase)
- secondary-side heat removal transients
- Moderator and shield-cooling system failures
- Fuel-handling accidents

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Elements of Safety Analysis for a Safety Case

Main Elements

- Development of Trip Set Points
- Initial Conditions
- Initiating Event
- Event Sequence
- Barrier Protection

The Last 3 is - "Accident Walk Through"

Discussion of the elements next

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Safety Analysis: Trip Set Points

- Automatic Instruments monitor reactor conditions:
  - ✓heat transport system pressure,
  - ✓reactor power and coolant flow.
- Any measurement with unsafe operating condition, triggers a reactor shutdown
  - **A shutdown by a protective system is called a reactor trip.**
- Trip set points
  - have conservative safety margins exist.
  - ✓a trip occurs automatically whenever a trip parameter exceeds its trip set point (a limit).
  - ✓There is manual trips called operator actions

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Safety Analysis: Initial Conditions

- A number of key parameters are chosen in a "conservative" direction for analysis.
  - Example reactor core properties

Parameter	Conservative Direction	Rationale
Reactor thermal power	**High**	**Minimize time to use up cooling water inventory, minimize margins to critical heat flux, etc.**
Shutdown system	**Back-up trip on less effective shutdown system using the last of three instrumentation channels to trip**	**Delay shutdown system effectiveness**
SDS2 injection nozzles	**Most effective nozzle unavailable**	**Reduce shutdown system reactivity depth**
SDS1shut-offrods	**Two most effective rods unavailable**	**Reduce shutdown system reactivity "bite" and depth**
Reactor decay power	**High**	**Minimize time to use up cooling water inventory**
Containment leak rate	**1. High ~2× to 10×design leak rate); 2. Low**	**1. Maximize public dose; 2. Maximize containment pressure**

Based on
- ✓ Initial plant conditions,
- ✓ system performance measures, and
- ✓ assumptions on unavailability of mitigating systems or portion of it.

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Safety Analysis: Initiating Event

- **Initiating Event**
  - ✓assumed that the **pipe break is instantaneous**,
    - ✓ bears little relationship to reality
  - ✓selected to **ensure conservatism in that it maximizes the predicted coolant voiding rate and hence the coolant void-reactivity insertion** and reactor power pulse.
  - ✓**greatest challenge to shutdown-system effectiveness.**
- Accident Walk-Through: Large Loss of Coolant Accident (LOCA)
  - ✓Large Heat Transport System pipe break.
  - ✓A large LOCA in a CANDU
    - ✓ break area is larger than twice the cross-sectional area of the largest feeder pipe. (Since being changed)
  - ✓A large LOCA can be located only in the large piping above the core. There are three representative locations (think of largest challenge)
    - reactor inlet header (RIH),
    - reactor outlet header (ROH), and
    - pump suction line (PSH).

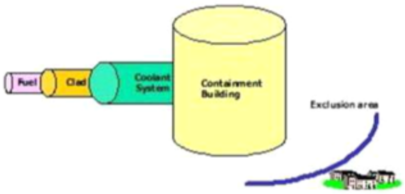
Safety Analysis: Event Sequence

- Estimate the approximate time of the following
  - ✓ reactor trip
  - ✓ start of fuel overheating
  - ✓ failure of first channel
  - ✓ core collapse
  - ✓ shield-tank failure (CANDU specific)
  - ✓ containment behaviour
- Event sequence is usually best represented by the concept of stylized accidents (design basis accidents- DBA) and run using computer codes/analytical tools
  - ✓ described later in **Analytical Models and Computer Tools**

Safety Analysis - Barrier Protection

Safety functions required in a nuclear reactor:

1. **control** by shutting down the reactor in case of transient
2. remove decay heat to **cool**
3. **contain** any radioactive material
4. monitor the state of the plant.



The Safety Case for Design

- Recall
  - ✓the risk from nuclear reactors comes from accidental **release of radioactive material.**
    - radioactive material is the **reactor fuel**.
  - ✓therefore, one can postulate accidents (safety analyses) which might allow radioactivity to escape
  - ✓design systems (called mitigating systems or safety systems) to prevent or control such accidents to support the safety case
  - ✓Example – hydrogen mitigating systems
    - Passive autocatalytic recombiners

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Nuclear Plant Safety Assured

Reliable safety is knowledge based

Supported by

- regulations and
- Design engineering and
- computer codes and analytical work (Tools)

CANDU reactors

- safety analysis (computer codes and analytical tools) to support the safety case
- 4 special safety systems (mitigating systems);
- principles of Defence in Depth

Nuclear Plant Safety Assured

- **Reactor containment**
  - ✓A Special safety systems
  - ✓contains the reactor vessel/calandria
  - ✓Contains radioactive fuel.
- The containment the last barrier left preventing large releases
- Requirements are formed from CNSC regulations and
- Fulfilments of requirements must be verified with safety analyses.

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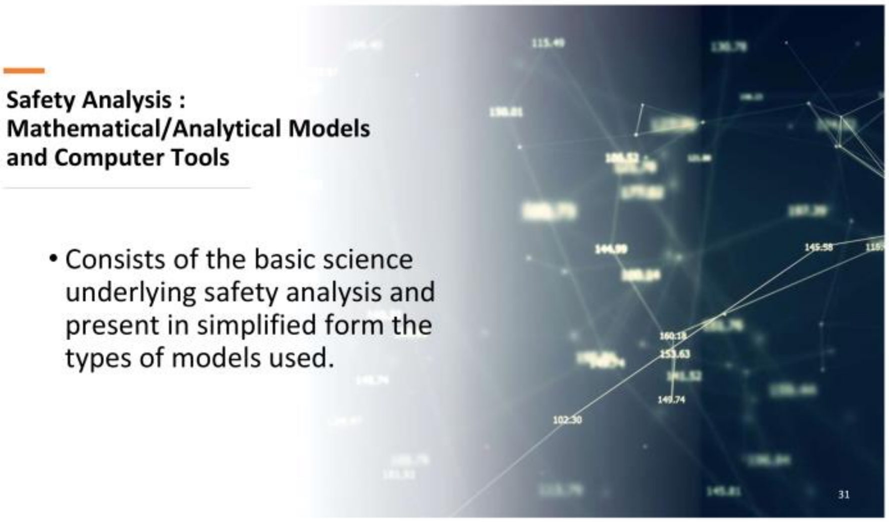

Note:
For the ICS, natural convection is the word!!!
Note 2: They don't have a pressure regulation system to control the pressure once the ICS isolated the reactor

Computer Codes/Tools

• Used to model the accident sequence to ensure safety of plants

Code Validation and Verification

• The codes are validated against

Nuclear Plant and

to ensure safety of plants

Nuclear Plant and Safety Analysis

Code Validation and Verification

- The codes are validated against experiments
  - ✓ demonstrate they can predict the reality
  - ✓ plant can operate as designed (NO) and
  - ✓ respond in the event of abnormal events (AAOs, DBAs) and beyond its design

Analytical tools used in CANDU Reactors next:

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Mathematical/Analytical Tools and Computer Codes

Used in **safety analysis and assessments**

Consists of the basic science underlying safety analysis utilizing analytical correlations and models

These tools/codes/models allow analysts to effectively and efficiently meet the following objectives:

- Evaluation acceptance criteria
- understanding of the system behaviour
- licen**see (vendor/operator)** reactor licen**ce**
- Covers various disciplines from the upstream analysis of reactor physics to the downstream analysis of dose
- User's requirements - fed back to the code developer; therefore, an excellent/strong interface is established

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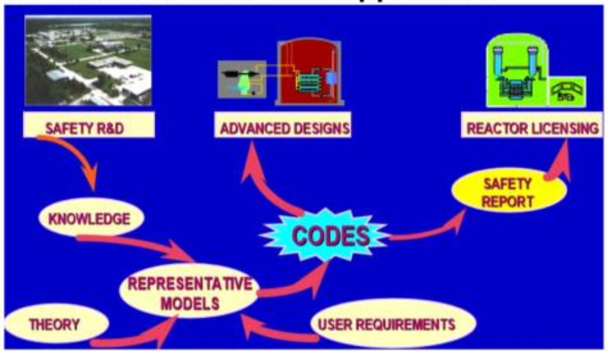
Computer Code Evolution and Application



Source: AECL

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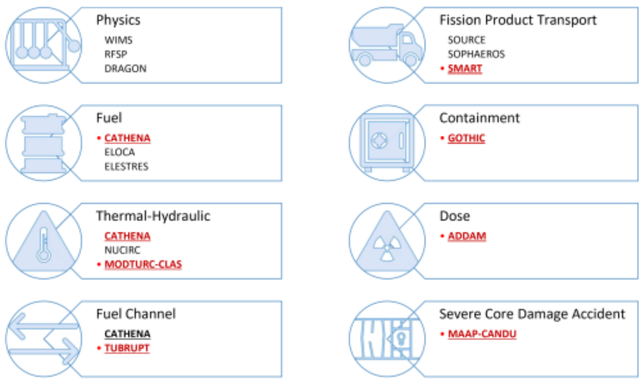
Key Computer Codes for CANDU Safety Analysis



- The ones in **RED** will be discussed in lecture

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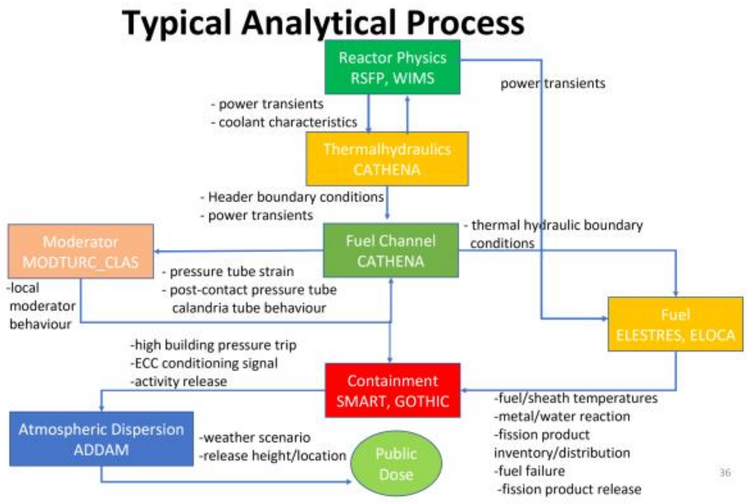

SUPER IMPORTANT SLIDE

These codes have been developed for 60 years and are incredibly impressive.

They're not only complex behind the scenes, but they're also vera complex to use.

MOST PROFESSIONALS ONLY KNOW HOW TO USE ONE OF THESE CODES EFFECTIVELY. THE PROF ONLY KNOWS GOTHIC, ADDAM, AND MAAP-CANDU

CATHENA

- Brief code description
  - ✓ Canadian Algorithm for THErmal-hydraulics Network Analysis(CATHENA)
  - ✓ one-dimensional, two-fluid system code
    - Steam (Gas) and Liquid
  - ✓ flow regime dependent relations coupled with the two-phase model
  - ✓ full network, user defined by input file
  - ✓ modeling of heat transfer in pin bundles
  - ✓ heat transfer correlations for entire boiling curve
  - ✓ built-in temperature dependent property tables
  - ✓ variety of component models available
    - generalized tank, valves, discharge break, pump, etc

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CATHENA

- Code general application
  - ✓ multi-purpose reactor cooling system thermal-hydraulics and thermo-mechanical analysis
  - ✓ full reactor cooling system network transient analysis
  - ✓ reactor fuel channel transient analysis
  - ✓ secondary side transient analysis
  - ✓ reactor fuel bundle transient analysis
  - ✓ fuel channel thermo-mechanical transient analysis
  - ✓ ECCS system operation analysis
  - ✓ auxiliary system thermal-hydraulics transient analysis
  - ✓ capability to be coupled with other codes (e.g. ELOCA, etc.)

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MODTURC_CLAS

- Brief code description
  - ✓ **MOD**erator **TUR**bulent **Cir**culation **C**o-**L**ocated **A**dvanced **S**olution
  - ✓ consists of coupling CANDU moderator related specific modules
  - ✓ volume-based porosity & distributed hydraulic resistance
  - ✓ two-equation k-epsilon model of turbulence
  - ✓ code capabilities include
    - calculation of pressure losses in the calandria tube array
    - calculation of the volumetric heat load distribution in the calandria vessel from steady-state and transient neutronic power and radioactive decay distributions
    - simulation of the moderator temperature control system
    - modeling of the moderator heat exchangers and associated control valves
    - the setting up of transient boundary conditions (inlet/outlet mass flows, transient poison concentration and other scalar inlet/outlet conditions, and restart capability)

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MODTURC_CLAS

- Code general application
  - ✓ used in nuclear safety analysis to predict velocity, temperature and/or poison concentration distributions (corresponding to SDS2 activation)
  - ✓ this information is used to determine:
    - moderator subcooling availability for the following postulated accident scenarios:
    - large break LOCA's with and without ECC(involving PT/CT contact heat loads to the moderator)
    - loss of moderator circulation
    - loss of moderator cooling
    - the poison distribution corresponding to:
      - ❑ in-core, single-channel breaks in an over-poisoned guaranteed shutdown state

The (CANDU) moderator cooling handles 7% of the power

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TUBRUPT

- Brief code description
  - ✓ **TUB**e **RUPT**ure
  - ✓ used to determine the pressure transients within calandria vessel due to the injection of fuel channel content during in-core break accidents
  - ✓ phenomena modeled
    - flashing coolant hydrodynamic transient in moderator
    - high temperature channel debris interaction with water
    - ruptures channel projectile formation and impact on the calandria vessel, shutoff rods guide tubes, and other fuel channels

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THIS CODE ONLY MODELS ONE TUBE FAILURE. THEY DO NOT HAVE CODE FOR 2 TUBE FAILURES.

2 OR MORE IS JUST STRAIGHT UP TOO COMPLEX.

TUBRUPT

- ✓Code general application
  - ✓estimate the extent of in-core damage due to a single fuel channel rupture caused by either of the following scenarios
    - spontaneous pressure tube /calandria tube rupture
    - severe flow blockage
    - feeder stagnation break
  - ✓code calculates the following parameters
    - moderator pressure transient
    - damage mapping of
      - ❑adjacent channels to the broken fuel channel
      - ❑shut-off rods guide tubes
      - ❑calandria vessel

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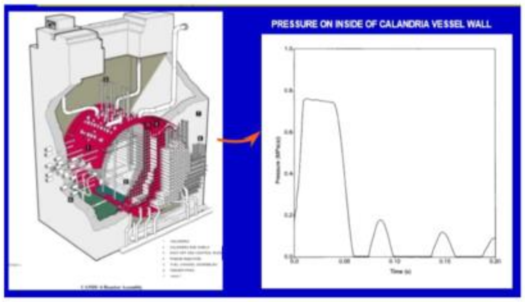
TUBRUPT Application



- Used to determine the peak pressure pulse in the calandria vessel when a PT/CT ruptures
- Estimate the extent of in-core damage due to a single fuel channel rupture

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SMART

- Brief code description
  - ✓ Simple Model for Activity Removal and Transport
  - ✓ calculates radionuclide behavior in containment
  - ✓ the code is composed of a set of one-dimensional, partial differential equations that describe the aerosol and fission product behavior
  - ✓ an aerosol general dynamics equation is solved to calculate aerosol size distribution as a function of space and time
  - ✓ mass conservation equations are solved to predict fission product concentrations in containment

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SMART

- Code general application
  - ✓calculates aerosol-fission product behavior within containment under accident conditions and releases to the outside atmosphere
  - ✓provides details about the concentration of individual isotopes present in various parts of the containment
  - ✓provides input to the ADDAM code for calculating dose to public and station staff

ADDAM

- Brief code description
  - ✓ **A**tmospheric **D**ispersion and **D**ose **A**nalysis **M**ethod
  - ✓ it is a Gaussian Dispersion model
  - ✓ used for analysis of hypothetical accident releases of radioactive material to the atmosphere from CANDU stations
  - ✓ conceptual model is based on modeling 15 atmospheric dispersion phenomena (covered in the ADDAM validation matrix)
  - ✓ ADDAM calculates concentrations of radioactivity in the air and on the ground and doses to members of the public following an atmospheric release

ADDAM

- Code general application
  - ✓Simulates the dispersion of aerosol-fission product release into the environment
    - ✓considers release height
    - ✓condisers time and duration of release
    - ✓Considers weather pattern(s)

✓ADDAM calculates concentrations of radioactivity in the air and on the ground and doses to members of the public following an atmospheric release

MAAP

- Brief code description
  - ✓ **M**odular **A**ccident **A**nalysis Program
  - ✓ a family of integrated computer codes designated for Severe Accident Analysis in nuclear plants, used by more than 40 international utilities
  - ✓ intended for PSA Level 2 analysis (out of scope for this course)
  - ✓ name of code is MAAP-CANDU in CANDU Reactors
  - ✓ MAAP_CANDU has models for horizontal CANDU-type fuel channels and CANDU-specific systems, such as:
    - calandria vessel,
    - reactor vault,
    - reactor cooling system,
    - containment systems (dousing), etc.

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MAAP Does everything! It's not as in-depth as the others, but it puts it all together for full-system analysis!

MAAP

- Code general application
  - ✓ MAAP_CANDU calculates severe accident progression starting from normal operating conditions for a set of plant system faults and initiating events leading to:
    - reactor cooling system inventory blow-down or/and boil-off
    - core heat-up and melting
    - fuel channel failure and core disassembly
    - calandria vessel failure
    - shield tank / reactor vault failure
    - containment failure
  - ✓ Physical processes modeled
    - thermal-hydraulics processes in reactor cooling system, calandria vessel, reactor vault and shield tank, end-shield, and containment components
    - core heat-up, melting and disassembly
    - zirconium oxidation by steam and hydrogen generation
    - material creep and possible rupture of reactor cooling system components, calandria vessel and shield tank walls
    - ignition of combustible gases
    - energetic and molten corium-coolant interactions
    - fission product release, transport and deposition

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GOTHIC

- Brief code description
  - ✓ Generation Of Thermal-Hydraulic Information for Containment
  - ✓ multi-dimensional thermal-hydraulic code specialized for containment analysis
    - code allows for hybrid modeling of containment volumes, ie, combinations of lumped parameter, 1D, 2D or 3D volumes
    - conservation equations are solved for 3 fields
      - ❑ steam/gas mixture
      - ❑ continuous liquid
      - ❑ liquid droplet
    - thermal non-equilibrium is allowed between phases and unequal phase velocities

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GOTHIC

- Code general application
  - ✓ treatment of momentum transport in multi-dimensional models
    - With optional models for turbulent, mass and energy diffusion
  - ✓ hydrogen combustion
  - ✓ engineering model options include
    - pumps, fans, valves, doors, heat exchangers, fan coolers
    - vacuum breakers, spray nozzles
    - coolers, heaters, volumetric fans
    - Hydrogen recombiners and ignitors
    - pressure relief valves
  - ✓ modeling of solid structures (thermal conductors) for flat plate (e.g., walls), cylindrical tube, solid rod

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Typical GOTHIC capabilities

- It is a versatile, general purpose thermal-hydraulics software package.
- GOTHIC™ solves the conservation equations for mass, momentum and energy for multicomponent, multi-phase compressible flow in lumped parameter and/or multi-dimensional (1, 2, or full 3D) geometries.
- The ability to combine these different nodalization options in a single model allows GOTHIC to provide computationally efficient solutions for multiscale application in **NPP Containments**

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GOTHIC and Operating Equipment

- ***GOTHIC has extensive models including:***
  - ✓**Pumps and fans,**
  - ✓**Valves and doors,**
  - ✓Heat exchangers and fan coolers,
  - ✓Vacuum breakers,
  - ✓Spray nozzles,
  - ✓**Coolers and heaters**,
  - ✓Volumetric fans (annular fans, deck fans, *etc.*),

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GOTHIC and Operating Equipment

- ***GOTHIC have extensive models :***
  - ✓**Hydrogen recombiners (forced and natural convection),**
  - ✓**Ignitors (spark device used to ignite hydrogen burns),**
  - ✓**Pressure relief valves (PRVs),**
  - ✓Filter and sump strainer,
  - ✓Dryer/Demisters and
  - ✓Charcoal filters

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GOTHIC and NPP Response

- ***Extensive control system, table function and trip capability***
  - ✓This feature can be used to modify nominal parameter values or dynamically control modeling elements based on predicted conditions.
  - ✓This allows GOTHIC™ to emulate plant response and provides a great deal of flexibility for modeling.

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Typical GOTHIC capabilities
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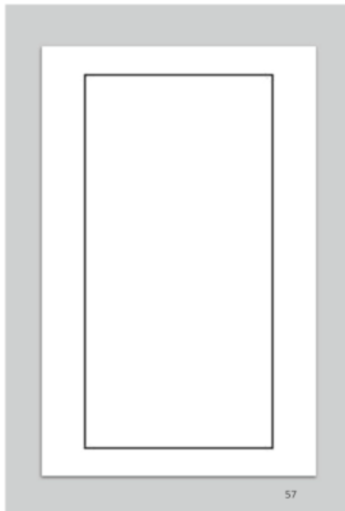
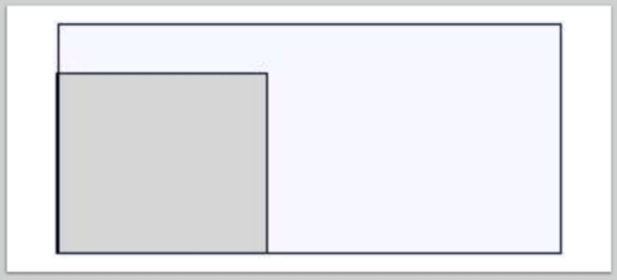
GOTHIC filling of container

- GOTHIC calculation of a container being filled with water.
  - ✓ The container is initially filled with air and is open at the top.
  - ✓ Water enters from the bottom as a vertical jet
  - ✓ As the water rises it spreads out, splashes against the sides of the container and falls back to the bottom.
  - ✓ This action traps air bubbles, seen rising to the surface as the container continues to fill.
  - ✓ The container (15 cm by 30 cm) takes about one second to fill to the top.



GOTHIC and Waterfall

- GOTHIC calculation - waterfall.
  - ✓ Water enters from the left, gradually filling the upstream channel and spilling over into the cavity below.
  - ✓ The falling stream takes on the familiar parabolic shape and entrains air as it accelerates downward.
  - ✓ As a steady flow condition is approached, the depth in the lower cavity behind the waterfall gradually increases.



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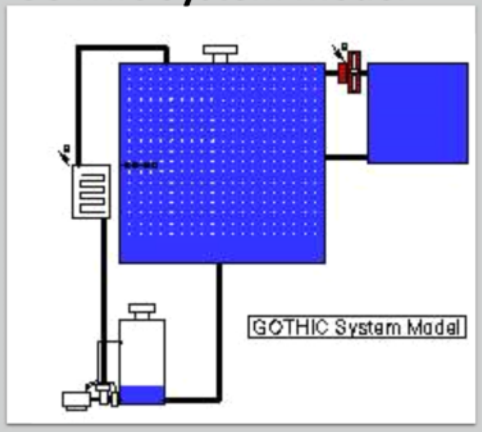
GOTHIC System Model



- Modeling capabilities
  - ✓ subdivided and lumped parameter volumes
  - ✓ mechanical equipment
  - ✓ Interconnection of rooms/tanks

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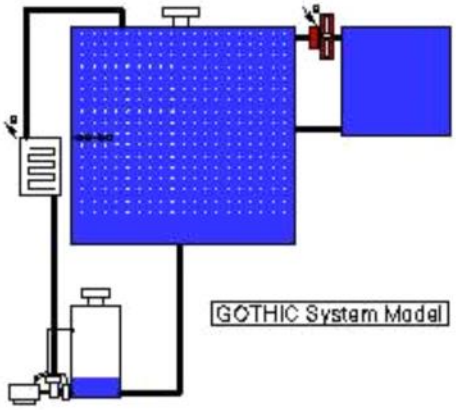
GOTHIC System Model



- ✓ A fan located in ducts connects the two rooms circulates air between them.
- ✓ The air being drawn through the fan is heated before it enters the lumped parameter (LP) volume.
- ✓ Water in the bottom of the subdivided volume is continually draining into the tank located below it. A pump starts when the water level reaches a predetermined set point, returning the water from the tank to the subdivided volume.
- ✓ The return piping contains a heat exchanger to cool the water and a spray nozzle to inject it into the room as small drops.
- ✓ After the tank has been drained, a low level trip shuts the pump off.
- ✓ Subdivided room contains four heater components, located near the left wall, which continually heats the room atmosphere throughout the calculation

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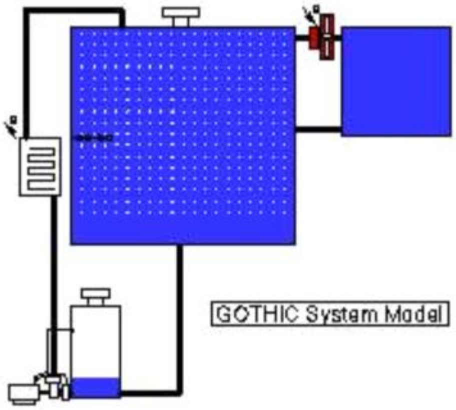
GOTHIC System Model



- ✓ The temperature is initially uniform throughout the model, but a buoyant plume quickly develops around the heater components.
- ✓ The average temperature for the LP volume increases more gradually as heat is added by the fan component.
- ✓ As the temperature in the lumped volume increases, the discharge into the subdivided volume changes from a momentum dominated jet to a buoyant plume.
- ✓ The pump periodically cycles on and off as indicated by the changing water level in the tank.
- ✓ When the pump is on the effect of the cold spray is evident from the changing temperature distribution and velocity vectors in the subdivided room.

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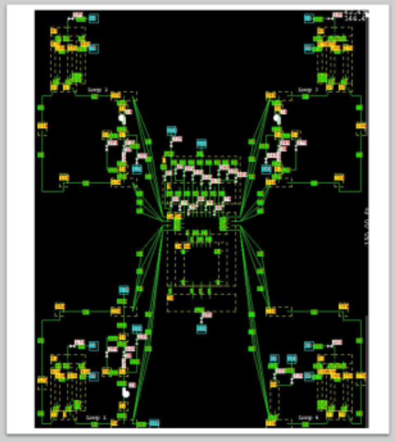

GOTHIC™ Model of 4-loop PWR
Each corner represents a loop

•Source – Numerical Associates Incorporated, USA

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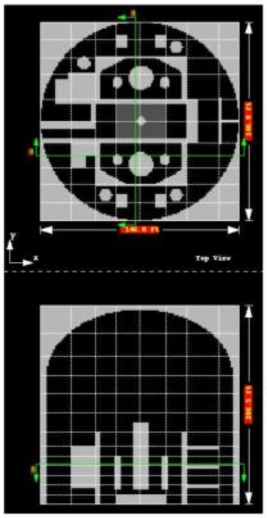
3-D GOTHIC™ Model of Containment



Viewing from Top of dome looking down in Containment

Side viewing of containment showing components in model

Source – Numerical Associates Incorporated, USA

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GOTHIC CODE in NPP Design

Can find operators of interesting NPPs that has been using GOTHIC for various design and operation of reactors in the world at the link

http://www.applied-analysis.com/gothic.htm

Other Computer Codes for SMR Safety Analysis

Physics
- TBA

Fuel
- TBA

Thermal-Hydraulic
- TRACG (BWRX-300)

Fuel Channel
- Not applicable

Fission Product Transport
- TBA

Containment
- GOTHIC

Dose
- RASCAL

Severe Core Damage Accident
- MAAP
- MELCOR

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TRACG

- Brief code description
  - ✓TRACG is a General Electric (GE) proprietary version of the Transient Reactor Analysis Code (TRAC)
  - ✓It is a best estimate code for analysis of boiling water reactor (BWR) transients ranging from
    - simple operational transient to design basis loss-of-coolant accidents (LOCA),
    - stability and anticipated transients without scram (ATWS).
  - ✓ TRACG is based on
    - a multi-dimensional two-fluid model for the reactor thermal hydraulics and
    - a three-dimensional neutron kinetics model.
  - ✓Playing the role of CATHENA in CANDU reactors

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MELCOR

- Brief code description
  - ✓MELCOR is a fully integrated, engineering-level computer code
  - ✓Primary purpose
    - model the progression of accidents in light water reactor nuclear power plants.
  - ✓Analyses reactor plant systems and their response to off-normal or accident conditions including
    - A broad spectrum of severe accident phenomena in both boiling and pressurized water reactors is treated in MELCOR in a unified framework.
    - estimation of fission product source terms
  - ✓Similar expected inputs and responses as in MAAP for CANDU reactors

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End Lecture 11

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