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THE 2016 WESTINGHOUSE TRAINING PROGRAM

TRAINING SCOPE

In addition to our engineering and site-service activities, Westinghouse Electric Belgium provides training services to customers in the nuclear industry.

Starting from basic courses describing the Pressurized Water Reactor (PWR), the training scope extends to the plant operation field and Emergency Operating Procedures (EOP’s).

Another area of expertise is represented by the training courses on core damage mitigation and severe accident management.

The courses contained in the 2016 training catalog are arranged in four categories:

- Operation and Accident Related Courses
- Engineering Courses
- Accident-Based Seminars
- AP1000

Particular effort is made to eliminate redundancies and repetition between courses, thus avoiding loss of time for attendees participating in several courses.

TEACHING MATERIAL

The materials used to prepare and teach the courses are based on regularly updated Westinghouse documents issued by the Westinghouse Training Organization.

During preparation of the courses, the instructors have free access to any information available in the Belgian Engineering Department or in the US, which assures the high quality of the technical content of our courses.

The teaching material includes PowerPoint presentations and computer software available in the Engineering Department, when needed.

COURSE TEACHERS

The instructors used by the Belgian Training Organization are drawn mostly from our Belgian Engineering Department, or are from the US training organization. All have demonstrated instructional skills and practice. Their technical knowledge is based on years of experience in engineering and/or on-site jobs, as well as on specific instructional training courses.

The excellent feedback received from attendees attests the high level our instructors have attained.
■ GENERAL COURSE ORGANIZATION

Our generic courses are provided on a yearly basis, in English, and are open to all.

All courses are given in our Nivelles offices, in Belgium.

Most of courses run for 6 hours a day, starting at 9:00 am.

In addition to our generic courses we also provide plant specific courses, on site, according to customer request or needs. This allows the customer to enroll a large number of attendees in the classroom which results in a price-per-attendee reduction. In addition, the Belgian Training Organization has a multi-language capability which is to the benefit of attendees who are not totally fluent in English.

■ INFORMATION AND REGISTRATION

Information forms are provided at the end of this catalog. You may use these now or later in 2016 to receive additional information about a course.

Registration forms are also provided at the end of the catalog. You may use them now or later in 2016 to reserve one or several seats on a course.

■ CONTACTS

For any queries about our training program and capabilities, please contact:

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Laetizia Lemaire
Training Administrator
Commercial Integration
Tel. 32-67.28.82.24
e-mail: lemairl@westinghouse.com
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(*) Budgetary prices are only indicative and subject to finalization, based on the number of attendees. They are specifically given to allow budget evaluations.

(**) A 21% VAT has to be added on the indicated value.
The following agenda is liable to modification in case of coincidence between a generic course and a customer specific training session. The courses for which no date is indicated will be organized upon customer request.

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# PWR

## ENGINEERING TRAINING

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## INSTRUMENTATION & CONTROL TRAINING

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(*) For more information about the US specific trainings, you can also consult our web page at [www.training.westinghousenuclear.com](http://www.training.westinghousenuclear.com)
OPERATION AND ACCIDENT RELATED COURSES

OP 161: PWR Plant Systems Description and Operation
OP 162: Transient and Accident Analysis
OP 163: Emergency Response Guidelines
OP 164: Core Damage Mitigation and Severe Accident Management
OP 165: Dedicated TSC/STA Training for EOP Support
PWR PLANT SYSTEMS DESCRIPTION AND OPERATION (OP 161)

Course Objectives

This course is designed to review the design basis and layout of the major PWR plant systems. The interactions between the different systems and the overall plant integration are particularly enhanced.

Exercises are foreseen in order to understand systems operation.

Course Outline

DAY 1

- Course Introduction
- Introduction to Reactor Theory
  - Neutron physics principles and reactor theory
  - Core feedback
  - Thermal hydraulics of the PWR core and heat exchanger

DAY 2

- Reactor Coolant System
  - Reactor vessel and core
  - Steam generator
  - Reactor coolant pump
  - Pressurizer
- Chemical and Volume Control System
  - Make-up
  - Boron recycle
  - Boron thermal regeneration
- Residual Heat Removal System

DAY 3

- Balance of Plant Systems
  - Main steam and turbine
  - Main feedwater and condenser
- Instrumentation & Control Systems
  - Resistance temperature detectors
  - Incore and excore nuclear instrumentation
  - Pressure, DP and level measurements
  - Rod control system
  - Steam dump
  - Pressurizer level and pressure control
  - Steam generator water level control

DAY 4

- Plant Operation
- Reactor Trip and Protection System Actuation Logic
- Engineered Safety Features Description
  - Emergency core cooling system
  - Auxiliary feedwater
  - Containment systems

DAY 5

- Accident Analysis
  - FSAR
  - Technical Specification
- Introduction to Emergency Operating Procedures (EOP’s)
- Introduction to Severe Accident Management Guidelines (SAMG’s)
- Questions and Answers
Course Objectives
The purpose of this course is to give the attendees the feel for plant behavior during normal, abnormal and accident transients.

Course Outline

FIRST WEEK

DAY 1
- Course Introduction
- Radiological Aspects of Core Damage
- Fundamentals of Reactor Theory

DAY 2
- Fundamentals of Reactor Theory (continued)
- Introduction to Accident Analysis

DAY 3
- Introduction to Accident Analysis (continued)
- Reactivity Addition and Power Distribution Anomaly Accidents

DAY 4
- Increased Heat Removal by the Secondary System Accidents
- Reduced Heat Removal by the Secondary System Accidents

DAY 5
- Reduced Reactor Coolant Flow Accidents
- Loss of Reactor Coolant Accidents

SECOND WEEK

DAY 1
- Loss of Reactor Coolant Accidents (continued)
- Steam Generator Tube Rupture Accidents

DAY 2
- Introduction to Mitigating Core Damage
- Critical Safety Function:
  - Subcriticality
  - ATWS

DAY 3
- Critical Safety Function:
  - Core Cooling
  - Inadequate Core Cooling
- Critical Safety Function:
  - Heat Sink
  - Loss of Secondary Heat Sink

DAY 4
- Critical Safety Function:
  - Primary Integrity
  - Pressurized Thermal Shock
- Critical Safety Function:
  - Containment
- Severe Accident Phenomenology

DAY 5
- Severe Accident Phenomenology (Continued)
- Accident Response Instrumentation
EMERGENCY RESPONSE GUIDELINES (OP 163)

Course Objectives

The purpose of this course is to explain the background of the ERGs Rev. 2, and their use. Emphasis is placed on understanding of phenomena and recovery actions rather than pure description of procedures.

Course Outline

DAY 1
- Philosophy and structure of the ERGs
- E-0 procedure and subprocedures

DAY 2
- LOCA concerns
- E-1, E-2 and subprocedures

DAY 3
- Steam Generator Tube Rupture
- E-3 and Subprocedures

DAY 4
- ATWS
- Inadequate Core Cooling
- Loss of Feedwater

DAY 5
- Pressurized Thermal Shock (PTS)
- Containment Integrity
- RCS Inventory
- Total Loss of AC Power
- Questions and Answers
CORE DAMAGE MITIGATION AND SEVERE ACCIDENT MANAGEMENT (OP 164)

Course Objectives

This course is designed to familiarize plant operation personnel and staff members with severe accident phenomena and accident scenarios highlighting the recovery and mitigation actions to prevent and limit core damage, maintain containment integrity and minimize the fission product releases. The Severe Accident Management Guidelines (SAMGs), developed by the Westinghouse Owners Group, are presented and their link with the Emergency Operating Procedures and the Site Emergency Plan is explained.

Course Outline

- Introduction
  - Definition of a severe accident
  - Description of Chernobyl, Three Mile Island, and Fukushima Accidents
  - Tools for the Study of the Severe Accidents
    - PSA Terminology and Scope
    - PSA Example Results
    - PSA Uses
    - PSA Decision Making Criteria
    - Severe Accident Simulation Models
  - Example Severe Accident Sequence
  - Introduction to Severe Accident Management

- Plant behavior prior to Core Damage: Initiating Events, Emergency Operating Procedures (EOPs)
  - Anticipated Transient Without Scram (ATWS)
  - Loss of Coolant Accidents / Inadequate Core Cooling (ICC)
  - Loss of Feedwater / Loss of Heat Sink (LOHS)
  - Loss of AC Power
  - Severe Overcooling / Pressurized Thermal Shock (PTS)
  - Response of Instrumentation to Core Uncovery

- Plant behavior during and after core damage: in vessel phase
  - Behavior up to core uncovery
  - Core melt progression
  - Hydrogen generation
  - Natural circulation and creep failure phenomena
  - Reactor vessel failure
  - Importance of EOPs and operator actions

- Plant behavior during and after core damage: ex-vessel phase
  - Containment design
  - Debris dispersal
  - Direct containment heating
  - Vessel thrust
  - Steam explosions
  - Debris coolability
  - Core concrete attack
  - Hydrogen behavior in containment
  - Containment fragility and failure modes

- Radiological Aspects
  - Fission product inventory
  - Fission product release from fuel
  - Fission product transport
  - Source terms

- Severe accident mitigation hardware
  - Filtered containment venting
  - Emergency containment spray system
  - Hydrogen control systems

- Severe Accident Management Guidance – WOG SAMG Overview
  - Background
  - Scope and philosophy
  - Technical basis
  - Goals
  - Structure of SAMG
  - Interface with EOPs and E-plan
  - Control room SAMG
  - TSC SAMG
  - Instrumentation
  - Phenomenology
  - Computational aids
  - Design variations
  - Summary
DEDICATED TSC/STA TRAINING FOR EOP SUPPORT (OP 165)

Course Objectives

The purpose of this course is to provide the necessary information to Technical Support Center (TSC)/Shift Technical Adviser (STA)/Plant Engineering Staff (PES) such that they can provide adequate and effective support to the operators during an accident recovery.

Course Approach

The teaching approach combined classic presentations where the physical aspects of the most important EOP recovery strategies are presented and explained with Case Studies, which put the attendees in situations for which they have to come up with answers to operator initiated questions or advises. The Plant Engineering Staff (PES) Case Studies in the following agenda are intended to provide information on the possible evaluations that a Technical Support Center (TSC), Shift Technical Adviser (STA) or Plant Engineering Staff (PES) would have to perform to support control room operators in case of accidents. This includes evaluations concerning:

- RHR suction alignment
- Need to transfer to hot leg recirculation
- Establishing RCS letdown or not
- Venting RV head or not
- Post-SGTR cooldown method
- Control of sump pH
- SG overfill
- Which SG to use for cooldown
- Reinitialiation of feed to a dry SG
- RCP status
- Long term plant status

Course Outline

DAY 1
- Symptom-Based Emergency Operating Procedures – Introduction
- Diagnostic
- SI Termination & Reinitiation Criteria
- EOP Evaluations by TSC or Plant Engineering Staff
- Loss of Coolant Accident Physics in Relation with Break Size

DAY 2
- RCP Trip/Restart Criteria
- SI Reduction Criteria
- Case Study 1 – Small Break LOCA
- Case Study 2 – Large Break LOCA

DAY 3
- Shutdown LOCA
- Case Study 3 – Stuck Open Safety Valve
- Case Study 4 – LOCA Outside Containment
- Pressurized Thermal Shock Aspects – FR-Ps

DAY 4
- E-3, Steam Generator Tube Rupture
- SGTR Contingencies
- Case Study 5 – SGTR
- Case Study 6 – Faulted & Ruptured SG
- Return to power & ATWS

DAY 5
- Case Study 7 – Anticipated Transient Without Trip
- Total Loss of Feedwater & Bleed and Feed
- Case Study 8 – Loss of Secondary Heat Sink
- Case Study 9 – Degraded and Inadequate Core Cooling
- Instrument Response in Accident Conditions
ENGINEERING COURSES

ENG 161: ASME Code Familiarization
ENG 162: US-Nuclear Standards, Rules and Regulations
ENG 163: Fracture Mechanics Applications
ASME CODE FAMILIARIZATION  
(ENG 161)

Course Objectives

The purpose of the course is to present the background and history of the ASME code for boiler and pressure vessel design and construction. Specific attention is also paid to the organization and use of the code. At the end of the course the attendees will be able to manipulate the ASME code for their own application.

Practical workshops are foreseen to enhance code utilization.

Course Outline

DAY 1

- What is the ASME code?  
  - General introduction to the ASME B&PV Code  
  - Comparison and relationship with other US and European Codes: ANSI, ASTM, AWS, ASNT, French RCCM, German KTA  
  - Administration of the ASME Code and organization of the Code editions and addenda.

- Why was the Code introduced and how is it applied in the nuclear industry?  
  - Significant events in the USA  
  - References to US Code of Federal Regulations (CFR) and NRC Regulatory Guides  
  - Design Basis and the Code  
  - Adaptation and use of the ASME Code in European countries. Position of the respective National Regulatory Bodies.

- How is the ASME code organized?  
  - Detailed content of the ASME B&PV code  
  - Definitions  
  - Organization in sections  
  - Organization in subsections and articles  
  - Interpretations and Code cases  
  - Relationship between the different parts of the Code  
  - Nuclear Power Plant components: definition of and relationship between ANSI Safety Classification and ASME Code Class

- Workshop on the general Code structure and use.

DAY 2

- Specific presentation of the Code sections of main interest to the Nuclear Power Industry  
  - Section II Material Specifications (Ferrous, Nonferrous and Welding Materials)  
  - Section III Subsection NCA  
    - Scope of each Subsection  
    - Structure in Articles (1000 to 8000)  
    - Subsection NB: Class 1 Components  
      - Article 2000 (Material)  
      - Article 3000 (Design)  
      - Article 4000 (Fabrication and Installation)  
      - Article 5000 (Examination)  
      - Article 6000 (Testing)  
    - Subsections NC, ND: Class 2 and 3 Components  
    - Subsection NF: Component Supports  
    - Subsection NG: Core Support Structures

- Specific presentation of the Code section III Article NB-3000 (Class 1 Components Design)  
  - This presentation will address the concepts and background applied in NB-3000: General Design, Design by Analysis (i.e. NB-3200, stress categories and respective limits), Design by Rules (e.g. NB-3600 Piping)  
  - Purpose, meaning and details of specific equations will be presented (e.g. NB-3600 Class 1 fatigue analysis)

- Workshop on the use and application of ASME III subsection NB.

DAY 3

- Specific presentation of the Code sections  
  - Section V: Nondestructive Examination  
    - NDE methods and actual records will be presented to the attendees  
  - Section VIII: Pressure Vessel, relation and difference with Section III  
  - Section IX: Welding and Brazing Qualifications

- Introduction to Fracture Mechanics concepts and their application in ASME III Appendix G, ASME XI, and 10CFR50 Appendix G  
  - Section XI: In-service Inspection of NPP Components.

- Exercises on Section XI application  
- Summary overview of the Code usage through the component life (design, inspection, repair, fatigue monitoring)  
- Current development of the Code, NRC position, European Regulators position.
Course Objectives

This course will provide an overview of the nuclear rules, regulations and standards, currently applicable in the US and followed in many other Western countries.

Their application in licensing during construction and operation of a nuclear power plant will be discussed.

Aspects of safety classifications, qualification, quality assurance programs, maintenance and inspection, and rules of backfitting are included in the program.

The US rules presented will be compared samplewise with other international regulations.

This course is addressed to people looking for an introduction to Western nuclear standards. It can serve as a preparation for specialized courses (for example on the ASME-Code).

Course Outline

DAY 1
- Safety objectives
- Licensing process
- Introduction to US rules and regulations
- Policy statements
- Regulations
- Regulatory guides and standard review plan
- National standards
- Industry practice
- Other NRC documents
- Compliance with regulatory requirements
- Safety classification

DAY 2
- Seismic classification
- QA classification
- Classification for electrical and I&C equipment
- Application of safety classes and ASME code for mechanical equipment
- Application of safety classes and IEEE standards for electrical equipment
- Qualification of electrical and I&C equipment
- Software qualification
- QA program requirements
- Backfitting and upgrading
FRACTURE MECHANICS (ENG 163)

Course Objectives

This course is designed to give engineers involved in design, analysis, maintenance and inspections a working knowledge of basic fracture mechanics. The subject matter presented will include the basic theory of fracture mechanics, as well as its application to fatigue crack growth and stress corrosion cracking. The course also includes a workshop, with examples of solved problems, as well as problems to be solved in class by the students. After these problems are solved, the proper answers will be reviewed and questions answered before proceeding. Another important aspect of the workshop will be an introduction to stress analysis techniques, which may include available programs for use on personal computers. Both theory and applications will be aimed at developing an understanding of the technology, to allow simple checks of others’ work as well as capability to solve problems directly. Also included will be sources of material properties information and references for further study.

Course Outline

DAY 1

- Introduction and Fracture Mechanics CoP
- The Stress Intensity Factor Concept
- Sub-critical Crack Growth: Fatigue Crack Growth

DAY 2

- Stress Corrosion Crack Growth
- Introduction to ASME Appendix G
- Hands-on Solutions to Fracture Problems

DAY 2

- Section XI Component Flaw Evaluation
- Development of Reactor System P-T Curves
- Section XI – Pipe Flaw Evaluation
- The Flaw Evaluation Handbook Concept
- Alloy 600 Applications in PWRs
- Flaw Evaluation of V.C. Summer
- Special Topics: Component Cooling Water System Flaw Evaluation
- Hands-on Solutions to Fracture Problems

DAY 3

- Probabilistic Fracture Mechanics Analysis
- Pressurized Thermal Shock: The Reactor Vessel Issue
- Special Topics: Low Shelf Fracture Toughness
- Leak Before Break Criteria and Analysis
- Application: Reactor Coolant Pump Flywheel Integrity
- Thermal Stratification Issues in Piping Systems
- French RCC-M ZG & ZD and RSE-M
- Summary and Wrap-up
ACCIDENT-BASED SEMINARS

ABS 161: Loss of Coolant Accident and Loss of All AC Power Accidents
ABS 162: Steam Generator Tube Rupture
ABS 163: Pressurized Thermal Shock
ABS 164: Fission Products Behavior During a Severe Accident

Seminar Objectives and Organization

The purpose of these seminars is to give the attendees a detailed overview and a deep understanding of the different concerns related to accidents. The seminars will include a review of safety concerns and FSAR type analysis, the comparison with best estimate transients, the extension to multiple failure or potential damaging situations and a detailed explanation of recovery actions. The seminars are independent from each other and may be followed separately.
**Seminar Objectives**

See statement on page 21.

**Seminar Outline**

**DAY 1**

- **Classroom**
  - Origin of the accidents: failure mechanisms in the reactor coolant system
  - Nature and classification of the events
  - Safety concerns associated with the LOCA
    - Core integrity. The FQ concern
    - Containment integrity
  - Review of protection systems and ECCS design criteria
  - Overview of regulations and technical specifications associated with the LOCA
  - The physics of the LOCA in relation with the break size
  - The six modes of Michelson for a 2" break
  - Core integrity analyses (FSAR type analyses)
    - History
    - Assumptions
    - Codes
    - Transient results
    - Interpretation
  - Containment integrity analyses
    - Mass/energy releases
    - H2 concern

**DAY 2**

- **Classroom**
  - Optimal recovery guidelines. Review of background and major steps of
    - Diagnosis E-0
    - RCP trip criteria
    - SI termination/reinitiation criteria
    - Loss of primary coolant E-1
    - SI termination ES-1.1
    - Post-LOCA cooldown ES-1.2
    - Cold leg recirculation ES-1.3
    - Hot leg recirculation ES-1.4
    - Loss of emergency coolant recirculation ECA-1.1
    - LOCA outside containment ECA-1.2

**DAY 3**

- **Classroom**
  - Potential damaging situation
    - Small break loss of coolant accidents without safety injection
    - Core cooling integrity function
    - FR-C function restoration guidelines
  - Core uncovering and core degradation evaluation
  - Environmental impact and dose assessment
  - The LOCA and emergency planning

**DAY 4**

- **Classroom**
  - LOCA induced by a total loss of AC power
    - Origin of the accident and sequence of events
  - Recovery action
    - Detailed review of the background and major steps of ECA-0.0, ECA-0.1 and ECA-0.2
STEAM GENERATOR TUBE RUPTURE (ABS 162)

Seminar Objectives
See statement on page 21.

Seminar Outline

DAY 1
- Classroom
- Origin of the accident: failure mechanism of SG tubes
- Nature of the event and classification of the accident
- Symptoms of the steam generator tube leak and tube rupture before RT
- Technical specifications associated with the tube leak and related actions
- Physical phenomena associated with the SGTR: Iodine behavior and steam generator stratification
- Conservative FSAR analysis
  - Assumptions
  - Codes
  - Transient results and interpretation
  - Evaluation of releases
  - Impact on environment
- Philosophy of the SGTR mitigation
  - Detailed review of the E-3 procedure

DAY 2
- Post SGTR cooldown methods
  - Backfill, blowdown, dump
  - Detailed review of
- ES-3.1 Post SGTR cooldown using backfill
- ES-3.2 Post SGTR cooldown using blowdown
- ES-3.3 Post SGTR cooldown using dump
- Case studies
  - The Ginna and North Anna SGTR analysis of operator actions impact
  - Effect of the RCP trip criteria

DAY 3
- Combination of a SGTR event and a LOCA or SLB
  - Recovery actions ECA-3.1, ECA-3.2, ECA-3.3
PRESSURIZED THERMAL SHOCK
(ABS 163)

Seminar Objectives
See statement on page 21.

Seminar Outline
DAY 1
• Introduction: Material and Fracture Mechanics Fundamentals
• The PTS concern: origin of the problem
• PTS analysis methodology
  − The deterministic approach
  − The probabilistic approach
• PTS concerns of Major Plant Accidents
• Influence of Plant Subsystems on PTS
• Instrumentation

DAY 2
• The NRC position and regulations
• The reactor vessel integrity safety function and status tree
• Detailed review of FR-P function restoration guidelines: background and major steps
• Use of ERGs for Major Plant Accidents

DAY 3
• Case studies
FISSION PRODUCT BEHAVIOR DURING A SEVERE ACCIDENT (ABS 164)

Seminar Objectives

In case of a severe accident on a nuclear power plant, the principal concern is that the engineered safety systems will fail, resulting in a large release of radioactive material (Fission Products). The severity of the accident depends on the degree to which the fuel is damaged and the degree to which the containment integrity is lost.

Radiation doses following the accident depend ultimately on the way fission products are released outside the containment.

The goal of this training course is to provide a global overview of fission products behavior from the production in the fuel pellet to release outside the containment during a severe accident.

Course Outline

DAY 1

- Introduction and course objectives

- Review of nuclear physics basis
  - Different types of radioactivity
  - Radioactivity decreases
  - Generation of FP
  - Fission reactions
  - FP reaction characteristics
  - Fission products

- FP inventory and classification
  - FP mass and activity in the core
  - Representative nuclide
  - Volatile and non volatile classification

- FP generation and retention in the fuel
  - FP behavior in the fuel pellets
  - Predicted chemical behavior of I and Cs in UO2

- Release from fuel pellets
  - Release rate mechanisms
  - Burst release
  - Diffusion release of the fuel pellet to cladding gap
  - Grain boundary release
  - Diffusion from UO2 grain
  - Release from molten material

DAY 2

- Transport in RCS
  - Aerosols sizes and shapes
  - Major processes governing the transport of FP
  - Condensation-evaporation
  - Agglomeration
    - Brownian agglomeration
    - Sedimentation agglomeration
    - Turbulent agglomeration
    - Inertial agglomeration
  - Deposition mechanisms
    - Brownian diffusion
    - Turbulence
    - Diffusionphoresis
    - Thermophoresis
    - Sedimentation
    - Comparison

- FP released from primary circuit, transport and retention in the containment
  - Aerosol sources
  - Release from fuel melt concrete interactions
  - Characteristics of aerosols generation
  - Physics of aerosols into the containment
  - Behavior of I in the containment
    - Source
    - In gas
    - In water

- FP release out of the containment
  - Source term definition
  - Influence of containment failure
  - Loss of containment integrity
  - Source term estimation
AP 1000

Course Objectives

The purpose of the course is to give the attendees an overall knowledge of the AP1000 passive design. It will also allow the student to understand the key differences in the AP1000 design and licensing bases when compared to the operating pressurized water reactors. The participants will be informed of the status of the various AP1000 licensing and construction activities and experience in the world.

Course Outline

DAY 1
- Overview of the AP1000 design
- Reactor Coolant System
- Passive Safety Systems

DAY 2
- Piping
- Modular Construction
- Seismic and Structural Design
- Inspection & Maintenance, Technical Specifications, Short Term Availability Controls & Reliability Assurance Programs

DAY 3
- Chemical and Waste Processing Systems
- Auxiliary Cooling Water Systems
- Steam & Power Conversion Systems
- Instrumentation and Control
- AP1000 Project Updates & Lessons Learned from Construction Activities

DAY 4
- HVAC Systems
- Electrical Systems
- Fuel and Reactor Control
- Deterministic Safety Analyses
- Probabilistic Safety Analyses
- Design Extension Conditions

DAY 5
- AP1000 Licensing
- Final wrap-up
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