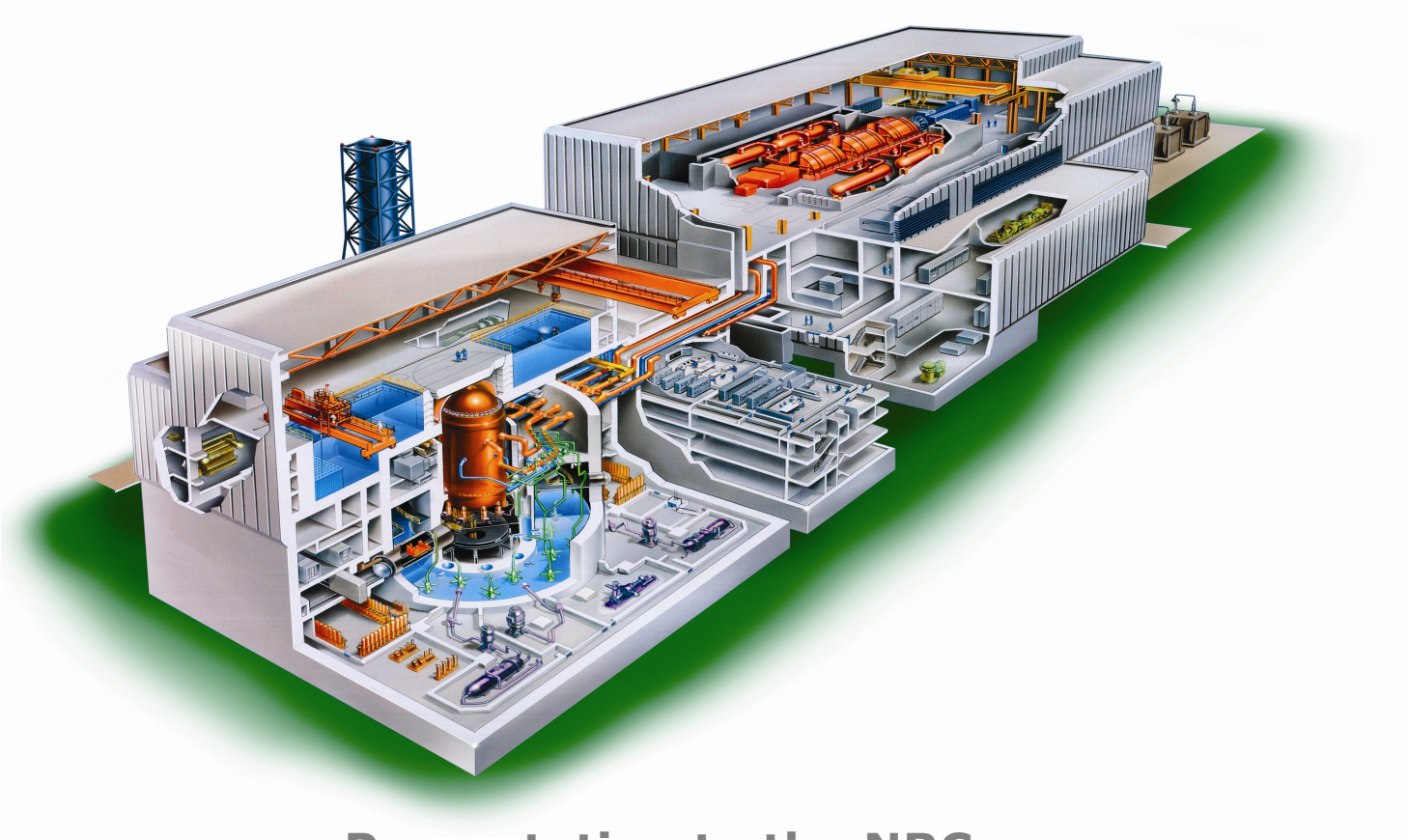
SOUTH TEXAS PROJECT
UNITS 3 & 4



Presentation to the NRC

January 25, 2007

Enclosure 2

Agenda

• Objectives & Opening Remarks	Greg Gibson
• Overview of Construction Era Lessons Learned	John Crenshaw
• Modularization Plans	John Crenshaw
• Quality Assurance Program Implementation	Marion Smith
• Units 3 & 4 Probabilistic Risk Assessment (PRA)	Bill Stillwell
• Summary	Greg Gibson

Objectives and Opening Remarks

Greg Gibson
Manager - Regulatory Affairs
STP Units 3 & 4

Objectives

- Discuss lessons learned from previous construction
- Discuss modularization concepts
- Discuss Quality Assurance Program development and implementation
- Discuss impact of modularization on ITAAC
- Discuss Unit 3 & 4 PRA

Overview of Construction Era Lessons Learned and Modularization Plans

John W. Crenshaw
Vice President, Engineering and Construction
STP Units 3 & 4

Introduction

STP Construction Era Lessons Learned

- Inputs
- Lessons Learned
- Actions to address Lessons Learned

Lessons Learned Inputs

- Input from 37 STP employees who were here during construction phase of Units 1 & 2

- Review of
  - INPO 83-011
  - NUREG-1055

Lessons Learned

1. Initial Engineering/Procurement/Construction (EPC) Contractor lacked nuclear experience
2. Owner/Operator experience, involvement, and preparedness were weak and the operational transition plan lacked rigor
3. Management/Oversight lacked rigor/integration
4. Project planning was weak – infrastructure was developed "on the fly"
5. Lack of skilled craft resulted in quality and training issues

Actions to Address Lessons Learned

- Develop a team using experienced contributors with strong nuclear experience [LL1]
- Apply operating and construction experience from ABWR fleet [LL1, LL2]
- Complete construction planning and design prior to construction [LL1]
- Scheduling: Plan the work & work the plan [LL1, LL2]
- Implement strong inspection and oversight processes [LL2, LL3]
- Implement "plant management" organization early to prepare effective O&M strategies, organizations, transition, and turnover plans [LL2, LL3]

Actions to Address Lessons Learned

- Develop solid organizational and individual contributor roles and responsibilities [LL3]
- Develop strong Safety Conscious Work Environment [LL3]
- Apply and integrate state-of-the-art project management tools, techniques, and infrastructure [LL4]
- Assess work force knowledge and skill gaps - train before work [LL5]
- Implement staffing strategy consistent with nuclear experience [LL5]

Introduction

Modularization

- Description
- Advantages
- Requirements
- Types of modules
- Sequence for module fabrication and installation
- Conclusions

Modularization Details from Hitachi

Description

What is modularization

- Rebar/embeds
- Structural (e.g. RCCV wall with liner, penetration sleeves, hatches)
- Pumps/valves/piping
- Supports

Description (Cont)

What modularization is not

- Electrical cabling and equipment
- I&C components (transmitters, tubing, wiring)

Current Uses

- US petrochemical industry
- US ships and submarines
- Japanese nuclear plants

Advantages

Allows parallel construction efforts

- Shorter construction duration
- Less downtime due to craft unavailability
- Offsite and onsite fabrication

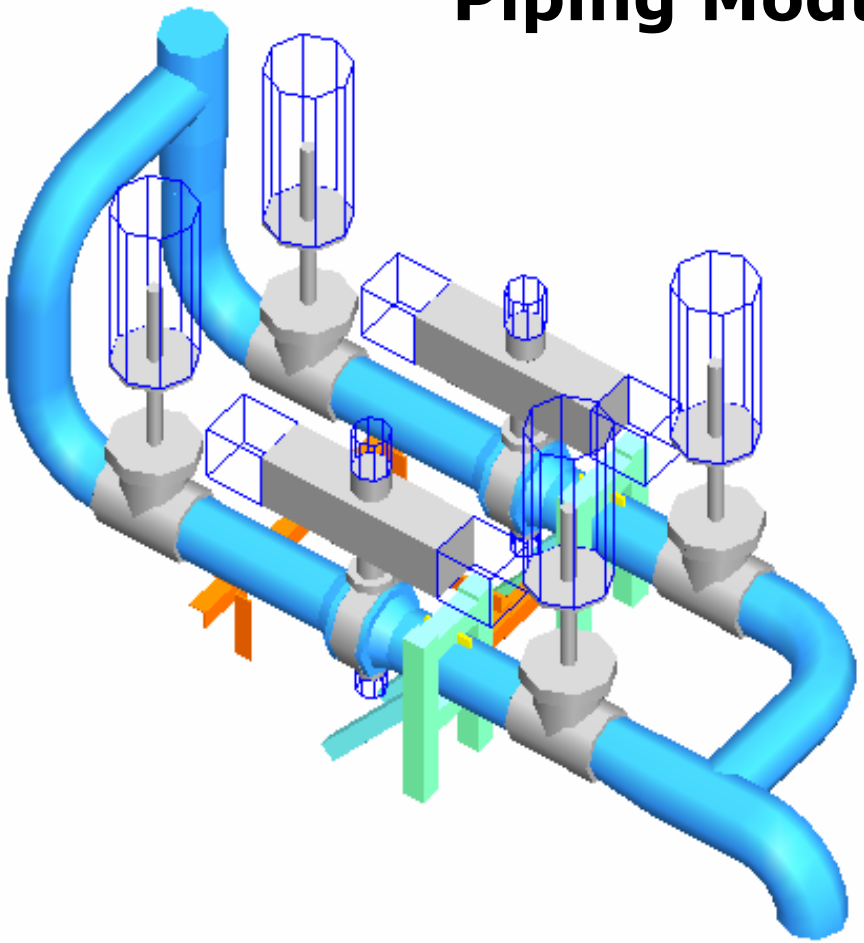
Easier construction (more room to work in)

Requirements

- More onsite lay-down area
- Onsite and offsite fabrication facilities
- Special cranes
- Engineered rigging and picks
- More front-end construction planning and engineering
- Modularization must be carefully planned to avoid difficulties during transportation and installation

Types of Modules

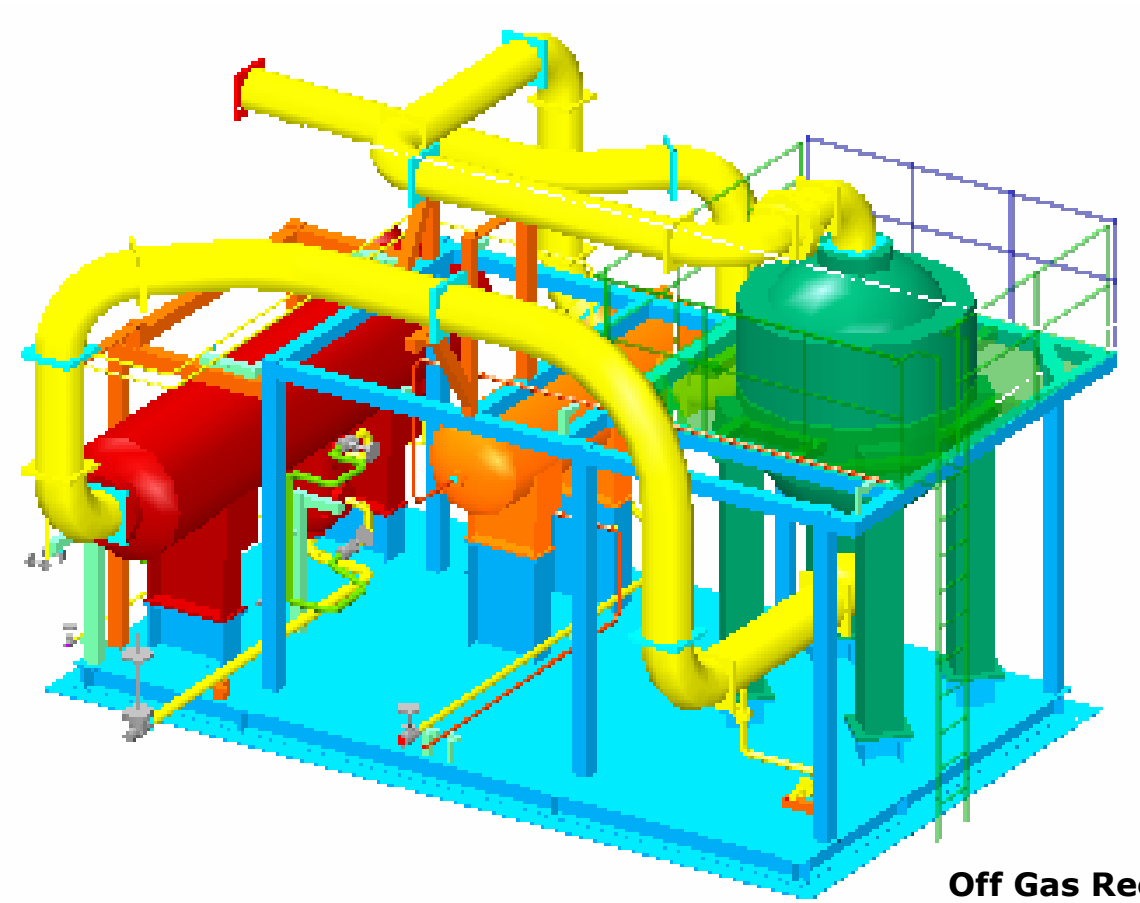
Piping Module



Consists of piping, valves, and supports

Types of Modules (Cont)

Common Base Module



Off Gas Recombiner

Consists of equipment, piping, and support structures

Often applied to comparatively narrow areas where system components are intensively arranged

Types of Modules (Cont)

Building Structure Module

Reactor Vessel Pad

Consists of anchor bolts and embedded piping with floor/beam rebar

Shortening of construction period would be expected

Interference avoidance among embedded components must be well coordinated

Types of Modules (Cont)

Building Structure Module

Upper Drywell

Module Fabrication Sequence

Using a Platform Module as an example

Piping & valve subassembly fabrication

- Component receiving inspection
- Component fit-up
- Fit-up inspection
- Component welding
- Welding inspection

Fabrication Sequence (Cont)

Platform subassembly

- Assemble platform plane
- Platform inspection

Piping support and temporary structure

- Assemble piping support
- Assemble temporary structure
- Piping support inspection

Fabrication Sequence (Cont)

Module assembly

- Assemble piping, valve, and platform subassemblies
- Temporary support for transportation
- Final module fabrication inspection

Transportation

Installation

Module Installation

- Receipt inspection
- Module carry-in (treated as “equipment”)
- Module installation
- Remove temporary supports
- Perform module final inspection
  [After this step all modules are handled and inspected the same as non-module (stick-built) components]

Installation (Cont)

- Install non-module components (tubing, conduit, wiring, instruments, controls, etc.)
- Inspect non-module components
- Perform system piping flushing and hydro
- Painting/insulation
- System pre-service inspection
- System commissioning

Conclusions

- Modularization is a demonstrated technology
- Modularization can be an effective tool to reduce construction time
- Modules can be inspected and accepted with existing QA/QC techniques

STP Units 3 & 4
Quality Assurance Program Implementation

Marion E. Smith
Quality Assurance Supervisor
STP Units 3 & 4

Introduction

- Quality Assurance Program Description (QAPD)
- Quality Assurance Program Implementation Plan
- ITAAC Interface with Modules
- Conclusions

Quality Assurance Program Description (QAPD)

- QAPD standardized with New Plant Utilities NEI 06-014
- Utilizes NQA-1 1994
- NQA-1 combines old N 45.2 standards into one Standard
- Program to include Quality Requirements for Non Safety Related Components that are “Safety Significant”

QA Program Implementation Plan Phase 1

Initial COLA development

Modified Unit 1 and 2 Quality Assurance Program

- Organization specific for Units 3 & 4
- Specific procedures modified to include Units 3 & 4
- Site characterization
- FSAR development and review
- Initial contracts for services

QA Program Implementation Plan Phase 2

Transition to NEI 06-14 based QAPD for Units 3 & 4

COLA completion activities
- FSAR completion and review

Materials Contracts

Component Fabrication

Oversight plan at Supplier facilities developed from
- Lessons learned
- Component significance
- Supplier history
- Supplier performance

Initial Site Work

QA Program Implementation Plan Phase 3

After LWA/COL - Construction

Component Fabrication and Installation

- Constructors with 10CFR50 Appendix B program
- Oversight provided by Constructor QA/QC personnel

Oversight of Constructor by STP personnel with performance based oversight skills

QA Program Implementation Plan Phase 4

Operational Program

- Programs to ensure safe operation in place
- ITAACS complete
- Implemented through Inspections, Performance Monitoring and Audits
- NQA-1 Quality Standard

Transition plan

- Combine all four units into single Quality Standard

ITAAC Interface with Modules

General ITAAC Requirements for Fluid Systems

- As Built conforms to Design
- System Integrity
- Physical Separation of Safety Trains
- Preoperational Tests
  - Safety Signal Initiation
  - System meets design flow requirements
  - System meets NPSH requirements

Effects of Modular Construction on ITAACs

Modular construction will have minimal affect on ITAAC Implementation

- Most ITAAC requirements will be completed after final installation

STP UNITS 3 & 4
Probabilistic Risk Assessment

Bill Stillwell
PRA Supervisor
STP Units 3 & 4

Introduction

ABWR Design-Specific PRA

As-Built Plant PRA

ABWR Design-Specific PRA

ABWR Design-specific PRA technical tasks includes:

- Level 1 Analysis for Internal Events
- Level 2 Analysis for Internal Events
- Level 3 Analysis for Internal Events
- Sensitivity, Importance, and Uncertainty Analysis for Internal Events
- Shutdown Risk Assessment
- External Events Risk Analysis

Scope of Design-Specific PRA

Level 1 analysis included:

- Internal initiating events evaluation
- Event tree and success criteria analysis
- Plant systems analysis using fault tree models
- Common cause failure and human reliability analysis
- Data analysis
- Fault tree and event tree quantification to calculate the core damage frequency

Scope of Design-Specific PRA (Cont)

Level 2 analysis included:

- An evaluation of severe accident phenomena and fission product source terms
- Modeling of the containment event tree and associated success criteria

Level 3 analysis included:

- an offsite dose evaluation of the representative release categories

Internal Event PRA CDF by Type of Event

Type of Event	CDF (per reactor-year)	Percentage of Total
• Transients	3.7E-08	23
• Loss of offsite power	1.2E-07	75
• LOCAs	6.9E-10	<1
• ATWS	2.7E-10	<1
• **Total**	**1.6E-07**	**100**
• ALWR Goal	1.0E-5	

Scope of Design-Specific PRA (Cont)

Level 1 shutdown assessment for low power and shutdown analysis

External events analysis included:

- Internal fire assessment
- Internal flooding assessment
- Seismic margin assessment

Shutdown Assessment

Loss of Decay Heat Removal

- Purpose
  - Determine minimum equipment sets that meet core damage frequency criterion
- Methodology
  - Operating RHR system assumed to randomly fail with a frequency of 0.1/yr
  - Time-dependent success criteria for preventing core damage identified
  - Fault trees and event trees developed similar to full power PRA
  - Minimum sets of systems identified to not be in maintenance (all remaining systems assumed to be unavailable)
  - Conditional core damage frequency calculated to meet criterion of 1.0E-5 which combined with the loss of RHR frequency of 0.1 corresponds to the USNRC “large release” goal of 1.0E-6/yr

Internal Flooding PRA Results

Building	Core Damage Frequency (per reactor-year)
Turbine Building	3.7E-9
Control Building	1.1E-8
Reactor Building	8.5E-9
– Inside ECCS Rooms	2.2E-9
– Corridor	4.6E-9
– Outside secondary Containment	1.7E-9

Internal Fire PRA Results

Building	Bounding CDF Screening Values (per reactor-year)
Safety-Related Building	
Division 1	4.3E-8
Division 2	1.1E-7
Division 3	5.6E-8
Control Room	8.9E-7
Turbine Building	2.2E-7

Seismic Margins Analysis

Methodology

- HCLPF ground Acceleration is High Confidence (95%) of Low probability (<5%) of Failure for that acceleration

  HCLPF = $A_m \exp(-2.326\, b_c)$

  A_m = Median Capacity

  b_c = Composite log standard deviation

- USNRC Goal: HCLPF > 0.5g (1.67*SSE)

- Two analytical methods used:
  - Convolution
  - Min-Max

  Results :HCLPF > 0.5g

Overall Containment Performance

Conditional containment failure probability goal of 0.1 established in the ABWR Licensing Review Basis

Potential for containment failure modes examined for all core damage sequences

Two definitions for containment failure considered:

- Large release (assumed to be greater than 25 rem at the site boundary)

  CCFP = 0.002

- Pressure Integrity ("Integrity as a pressure boundary can no longer be controlled" — SECY-90-016)

  CCFP = 0.005

Comparison with Safety Goals

Final Safety Evaluation Report, Chapter 19.1.3.8.3 Comparison with NRC and EPRI Safety Goals

- Individual risk of early fatality – Goal: < 5E-07
  - *"Meets by a large margin"*
- Individual risk of cancer fatality – Goal: < 2E-06
  - *"Meets by a large margin"*
- Core damage frequency – Commission Goal: <1E-04, EPRI Goal: < 1E-05
  - *"Meets both goals for internal events by a large margin"*
- Probability of Large Early Release (one or more early fatalities) – Goal: < 1E-06
  - *"Meets by a considerable margin"*

Comparison with Safety Goals
(Cont)

- Conditional containment failure probability (structural integrity definition) – Goal: < 0.1
  - *"Did not 'strictly' meet" (staff's uncertainty bridged the goal)*

- Conditional containment failure probability (dose definition) - Goal: < 0.1
  - *"Did not 'strictly' meet" (staff's uncertainty bridged the goal).*

- Staff and GE agreed that the mean values of the risk estimates meet the Commission's safety goals.

Scope of STP PRA in COLA

- STP COLA, Part 9 will evaluate site and plant specific information to determine any differences and impact on the design-specific PRA.
- STP COLA will update, as necessary, the design-specific PRA internal events, internal fire, internal flood, seismic events at full power and shutdown assessment to address STP site characteristics or changes from the Design Control Document (DCD)
- Site-specific external events such as high wind and external flooding will be addressed in the design analyses.

Scope of STP PRA in COLA (Cont)

- The methods used in the ABWR design-specific DCD severe accident analyses and phenomenological studies will be retained for COLA

- Radiological consequences will be performed with MACCS2 computer code for STP COLA

Identified Differences Between DCD and STP

- RCIC turbine and pump design (Weir Turbine)

- Electrical Distribution: ABWR DCD design was a single 6.9kV switchgear per division, STP will have two medium voltage switchgear (13.8kV and 4.16kV) per division

- Ultimate Heat Sink: ABWR DCD design was a Cooling Pond, STP will be designed with mechanical draft cooling tower

Identified Differences Between DCD and STP (Cont)

- ABWR DCD is designed based on a single unit plant, STP will be designed for dual units with some common facilities

- Loss of offsite power frequency and recovery probability (NUREG/CR-6890)

COLA PRA Development

- Collect design-specific PRA information, and convert models to Windows CAFTA
- Identify the differences between ABWR design-specific and STP plant design
- Incorporate/Evaluate STP characteristics or design changes into design PRA model
- Compare COL Application PRA results and insights to ABWR design-specific PRA results and insights

COLA PRA Development (Cont)

- Update and document the ABWR design-specific PRA for the COLA, as necessary to reflect STP specific features that are not bounded by the ABWR design-specific PRA

- Update commitments for COL items needed to be updated or evaluated when as-built, as-operated information becomes available before fuel loading (DCD 19.9).

NRC Review of the Design-Specific PRA

NUREG-1503, Final Safety Evaluation Report Related to the Certification of the Advanced Boiling Water Reactor Design, July 1994, Section 19.1.1

"…the ABWR is of a robust design, that the design is an improvement over existing designs, and that the design meets the Commission's safety goals described in 51 FR 28044 and 51 FR 30028…for internal events…"

"Although direct comparison of external-event results to these goals is not possible, the ABWR design has significant margins above the design bases for seismic, fire, and internal flood initiating events…"

"The staff believes that the ABWR design meets the Commission's safety goals."

As-Built Plant PRA

- STP will have an all-modes, Level 1 and 2 PRA that includes internal and external events
  - At least 6 months prior to fuel load
- PRA will model the as-built plant
- As-Built PRA will satisfy, to the extent possible, codes and standards covering PRA quality and applicable NRC Regulatory Guides
  - ASME Capability Category 2
- The PRA will be peer reviewed prior to use

As-Built Plant PRA (Cont)

- As-Built PRA will be maintained as a living document
  - Periodic update
  - Incorporate industry and plant experience
  - Maintain awareness of changing requirements

- As-Built PRA will be capable of supporting current and potential future Risk-Informed applications

High Level Schedule

The As-Built PRA will be performed in two stages

Stage 1: COLA submittal to COLA Approval

- Level 1 and LERF (Level 2) Internal Events, limited external events, qualitative treatment of shutdown, etc.

Stage 2: Construction Start to Operation

- All modes of operation, Internal and External Events, Level 1 and Level 2

High Level Schedule (Cont)

At the completion of stage 1

- PRA will be capable of supporting risk-informed decision making during construction.
- Component importance, Important operator actions, Dominant sequences, etc.

At the completion of stage 2

- PRA will be capable of supporting risk-informed decision making for plant operation
- A(4), RITS, MSPI, SDP, etc.

Topics for Further Consideration

- Codes and Standards changes to support new/advanced designs

- Level 2 PRA
  - Standard developed?

- Level 3 PRA
  - Required? Standard developed?

- Specific Risk-Informed applications (?)

Summary of Design-Specific and As-Built PRAs

- The ABWR design-specific PRA will be updated as necessary to support the COLA

- The As-Built PRA will satisfy, to the extent possible, codes and standards applicable to an all modes, internal and external events, Level 1 and Level 2 PRA

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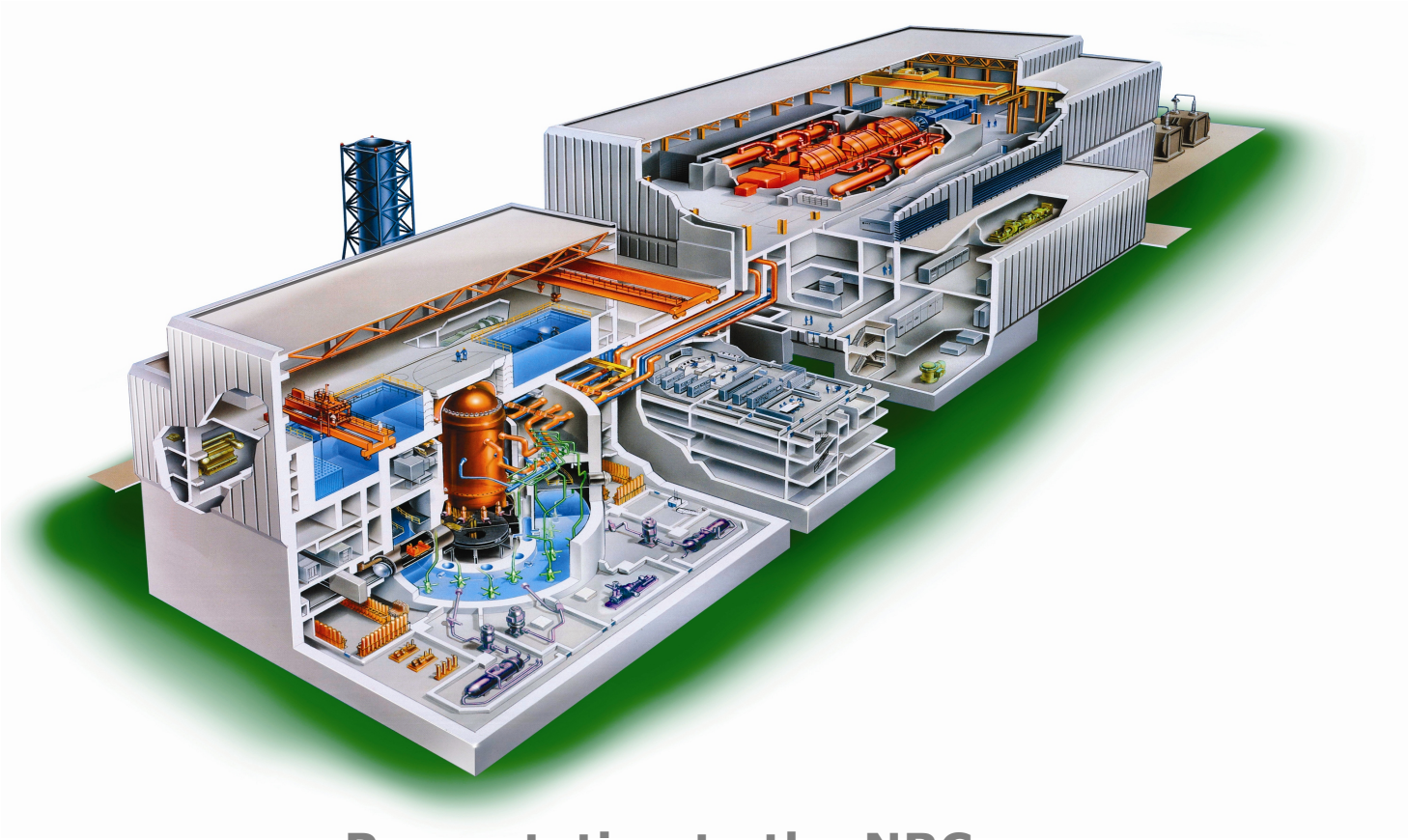

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