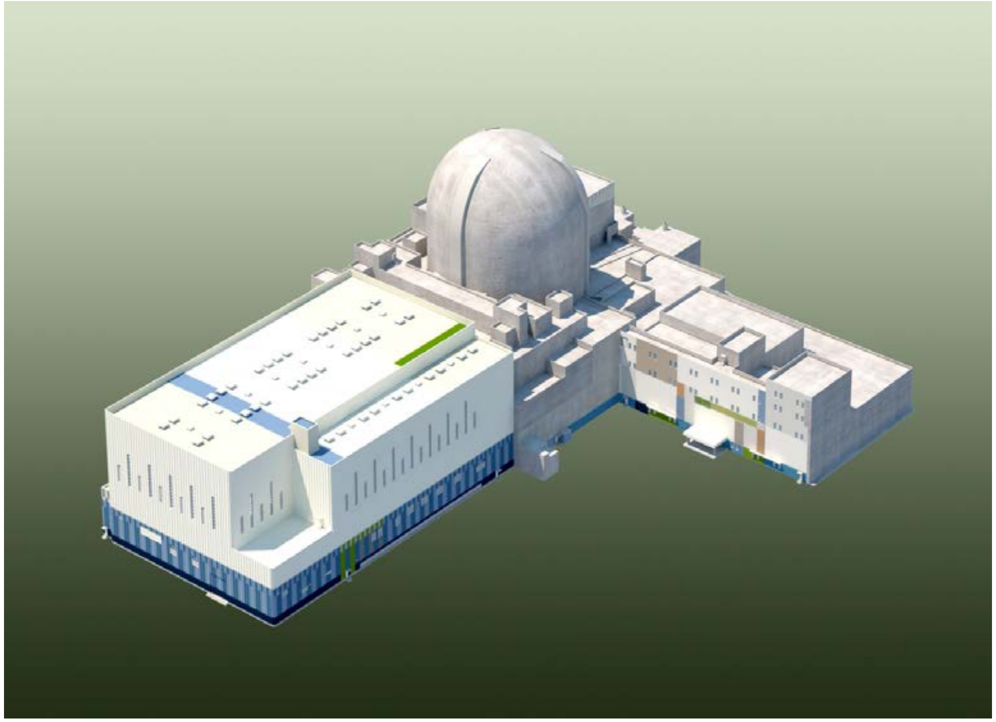
APR1400 Safety Analysis



KEPCO/KHNP

Apr. 20~21. 2016

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    - 15.6.5: Small Break LOCA (SBLOCA)
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Non-PROPRIETARY

Introduction (Regulatory Bases)

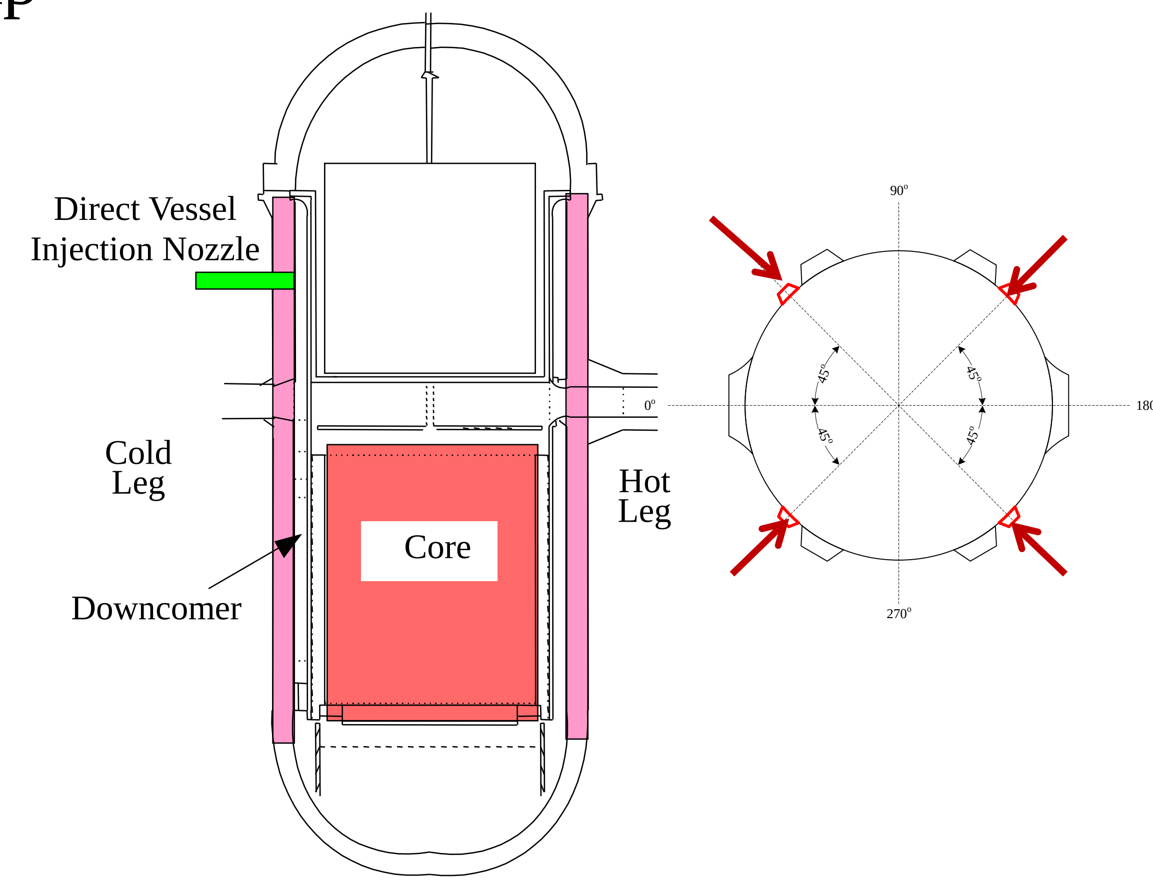
- Code of Federal Regulations
  - ✓ 10 CFR 50.46*
  - ✓ Acceptance Criteria for Emergency Core Cooling Systems (ECCS) for Light Water Nuclear Power Reactors
- Regulatory Bases
  - ✓ Regulatory Guide 1.157, BE Calculations of ECCS Performance
  - ✓ Regulatory Guide 1.206, Combined License Applications for Nuclear Power Plants (LWR Edition)
  - ✓ NUREG-0800, Standard Review Plan for the Review of Safety Analysis Reports for NPP: LWR Edition – Transient and Accident Analysis
  - ✓ NUREG-1230, Compendium of ECCS Research for Realistic LOCA Analysis
  - ✓ NUREG-5249, Quantifying Reactor Safety Margins: Application of Code Scaling, Applicability and Uncertainty Evaluation Methodology to a LBLOCA (CSAU)

* Rule making of 10CFR50.46(C) is on going

BE: Best Estimate

APR1400 Design Features

- SIS: 4 mechanically & electrically independent trains
- Direct Vessel Injection (DVI)
- A safety injection pump and a safety injection tank (SIT-FD) are installed in each train
- All the ECC water is injected into the upper annulus of reactor pressure vessel



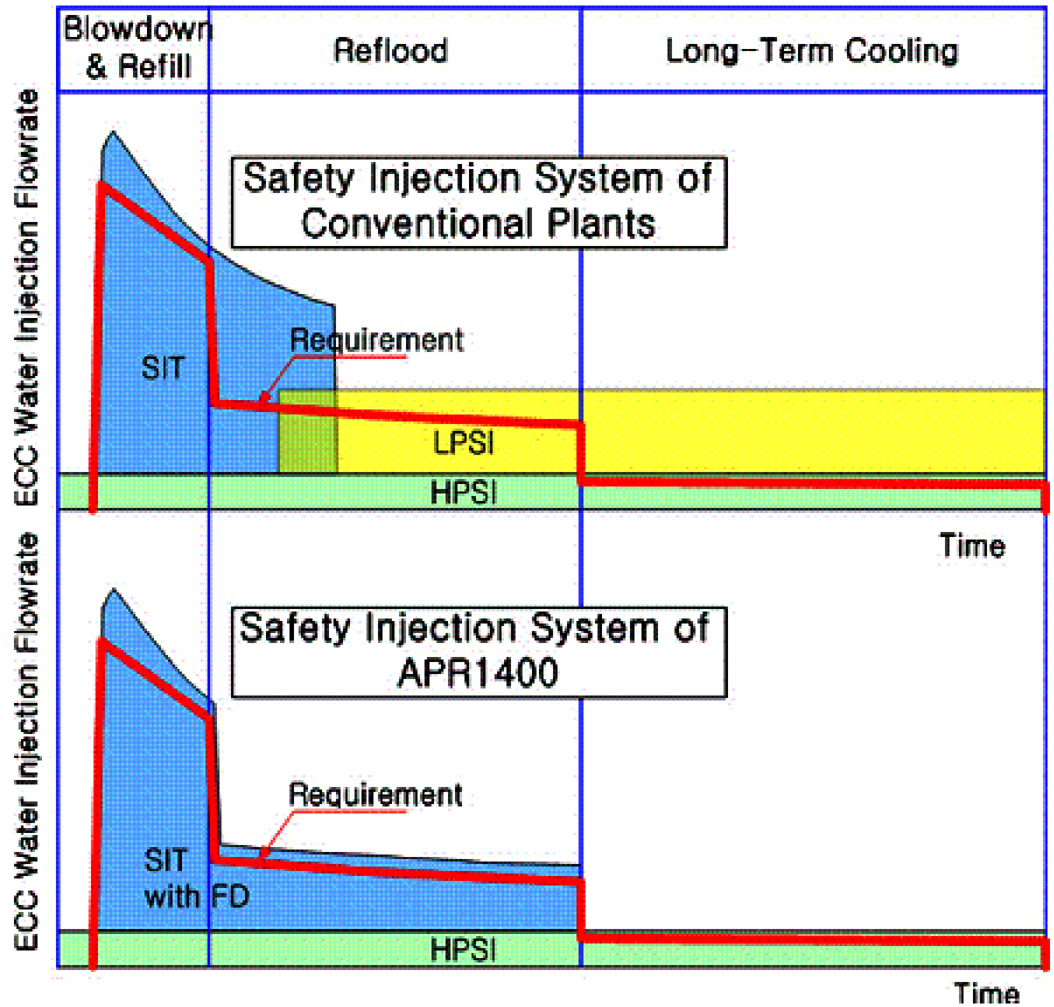
SIS : Safety Injection System
ECC : Emergency Core Cooling

APR1400 Design Features

- Fluidic Device in SIT regulates the injection flow rate and enhances removal of decay heat in early reflood phase

- Topical Report;

  'Fluidic Device Design'
  (APR1400-Z-M-TR-12003, Dec. 2012)



SIT-FD Injection Flow Rate

Design Review Status

- Large Break LOCA
  - ✓ Topical Report; 'Realistic Evaluation Methodology for Large-Break LOCA of the APR1400'
  - ✓ 99 Audit Issues for Discussion Related to CAREM, August 2015.
  - ✓ NRC Large Break LOCA Audit, January 12~14, 2016.
  - ✓ Request for Additional Information (RAI) Process
- Small Break LOCA
  - ✓ Analysis Results Confirmed the Satisfaction of Acceptance Criteria
  - ✓ Application of CENPD-137P for APR1400 Proceeds
- LTC / Non-LOCA
  - ✓ Analysis Results Confirmed the Satisfaction of Acceptance Criteria
  - ✓ RAI Process

CAREM : Code Accuracy based Realistic Evaluation Model

Overview of LBLOCA Analysis

- License Information for LBLOCA
  - ✓ Topical Report Submission

    ; 'Realistic Evaluation Methodology for Large-Break LOCA of the APR1400' (APR1400-F-A-TR-12004), Dec. 2012
  - ✓ Code Accuracy based Realistic Evaluation Model (CAREM)
  - ✓ Codes
    - RELAP5/MOD3.3K: Thermal-hydraulic analysis
    - CONTEMPT4/MOD5: Containment back pressure calculation
    - Two codes are consolidated to exchange P & M/E data
  - ✓ Licensing process for CAREM is currently under way

Overview of LBLOCA Analysis

- Code & Methodology
  - ✓ RELAP5/MOD3.3K & CONTEMPT4/MOD5
    - Two codes exchange mass & energy (RELAP5) and pressure (CONTEMPT4) as boundary conditions for each other
  - ✓ CAREM developed based on the CSAU
    - Uncertainties are quantified by non-parametric statistics and SRS calculation
    - Introduce Experimental Data Covering (EDC) for confirmation of uncertainty parameters and their ranges & distributions

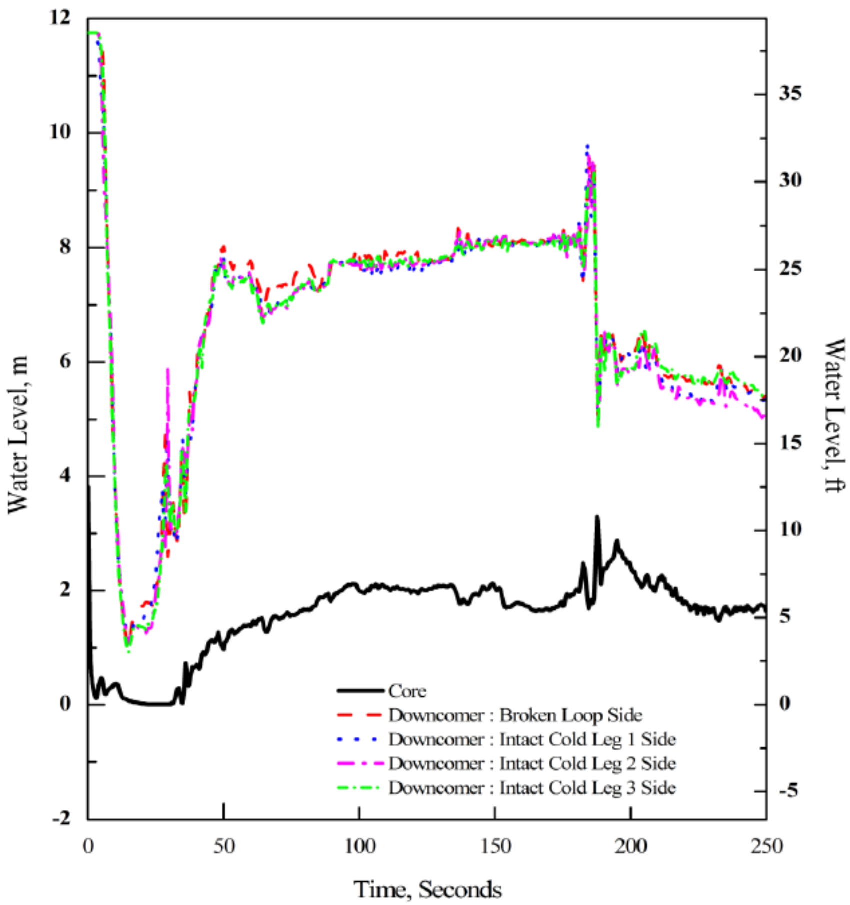
CSAU: Code Scaling, Applicability and Uncertainty (NUREG-5249)
SRS: Simple Random Sampling
EDC: Experimental Data Covering

Overview of LBLOCA Analysis

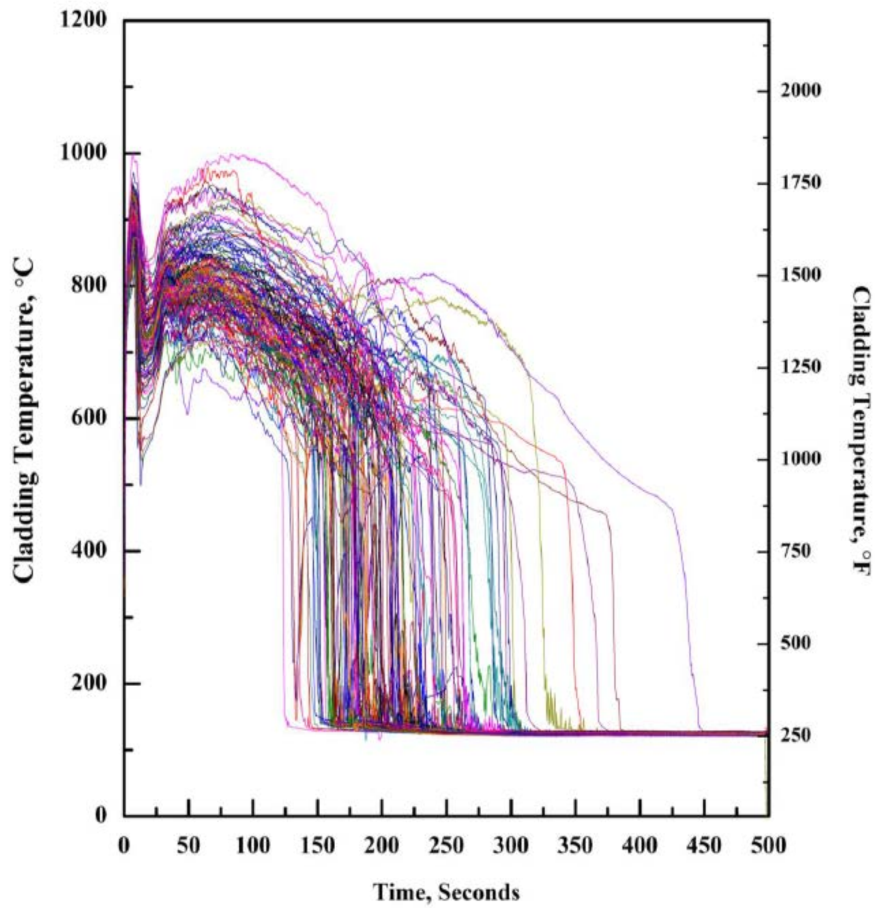
TS

LBLOCA Analysis Results

➢ 100% Double-ended Guillotine Break in Pump Discharge Leg



Water Levels in Core and Downcomer



SRS Peak Cladding Temperatures

LBLOCA Analysis Results

Peak Cladding Temperature (PCT)		Value, ℃
SRS Results	Highest PCT	1,000.3
	Highest Reflood PCT	999.8
Scale BIAS Evaluation Results	Final BIAS Reflood PCT	1,002.9
	Max. BIAS Case Reflood PCT	999.8
	- ECC Bypass BIAS	+3.1
	- Steam Binding BIAS	+0.0
Final PCT (w/ BIAS)		1,002.9 (1)
Max. Cladding Oxidation		Value, %
SRS Results	Max. Cladding Oxidation	3.85
Scale BIAS Evaluation Results	Final BIAS Max. Cladding Oxidation	3.86
	Max. BIAS Case Cladding Oxidation	3.85
	- ECC Bypass BIAS	+0.01
	- Steam Binding BIAS	+0.0
Final Max. Cladding Oxidation (w/ BIAS)		3.86

- Licensing PCT

  $= PCT_{95/95} + \Delta PCT_{Bias\ results} + \Delta PCT_{additional\ (10\ °C)}$

  $= 1{,}012.9$ °C (1,855.2 °F) ≤ 1,204.4 °C (2,200 °F)

- The satisfaction of acceptance criteria is confirmed for APR1400 design

Overview of SBLOCA Analysis

- Small Break Loss Of Coolant Accident (SBLOCA)
  - ✓ Technical Report Submission; ‘Small Break LOCA Evaluation Model’ (APR1400-F-A-NR-14001), Sep. 2014
  - ✓ C-E SBLOCA Evaluation Model (CENPD-137P)
  - ✓ Codes
    - CEFLASH-4AS: T/H behavior of RCS during blowdown phase
    - COMPERC-II: T/H behavior of RCS during reflood phase
    - PARCH and STRIKIN-II: Fuel rod heat-up calculation (PCT, PLO)
  - ✓ Methodology & Codes approved by the NRC for existing US PWRs are Applied

SBLOCA Analysis Results

➢ Peak Cladding Temperature and Oxidation Percentage

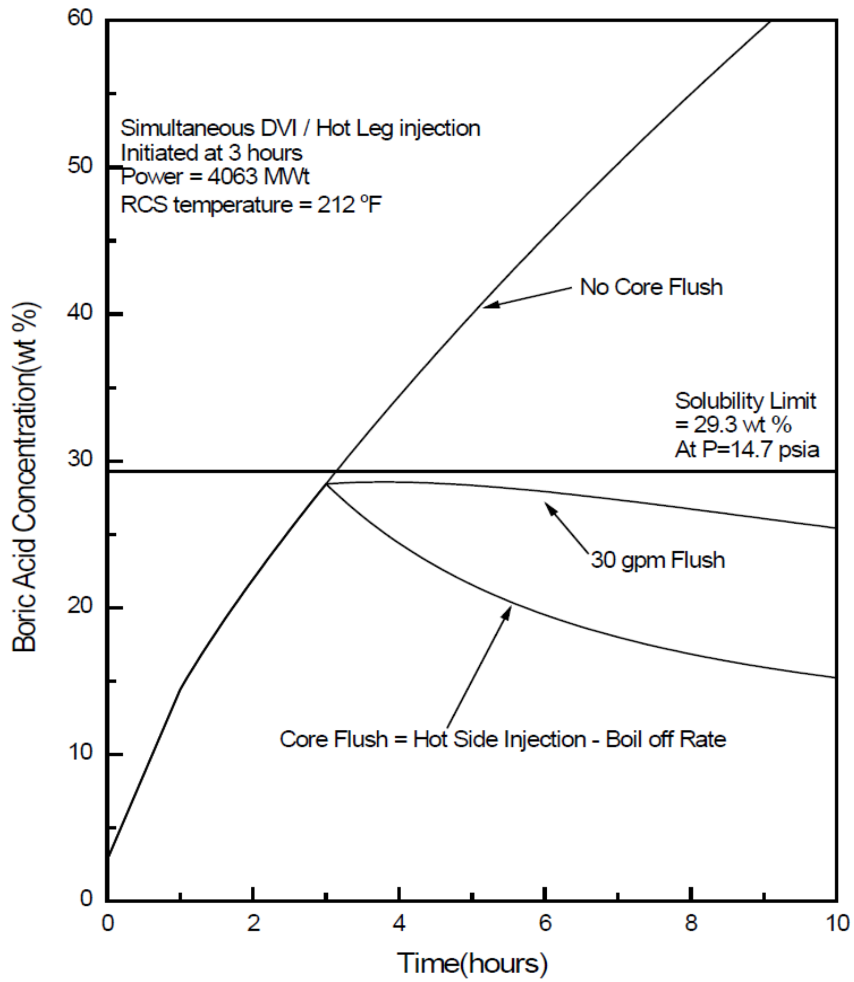
Break	Peak Cladding Temperature, °C (°F)	Maximum Cladding Oxidation, %	Maximum Core-Wide Oxidation, %
465 cm^2/PD	498 (929)	0.0017	< 0.0003
325 cm^2/PD	492 (917)	0.0015	< 0.0002
93 cm^2/PD	565 (1,049)	0.0010	< 0.0001
46.5 cm^2/PD	568 (1,054)	0.0008	< 0.0002
372 cm^2/DVI	624 (1,156)	0.0195	< 0.0029
93 cm^2/DVI	569 (1,056)	0.0069	< 0.0009
46.5 cm^2/DVI	571 (1,059)	0.0018	< 0.0003
18.6 cm^2/DVI	616 (1,140)	0.0029	< 0.0006
27.9 cm^2/HL	568 (1,055)	0.0006	< 0.0002

Overview of LTC Analysis

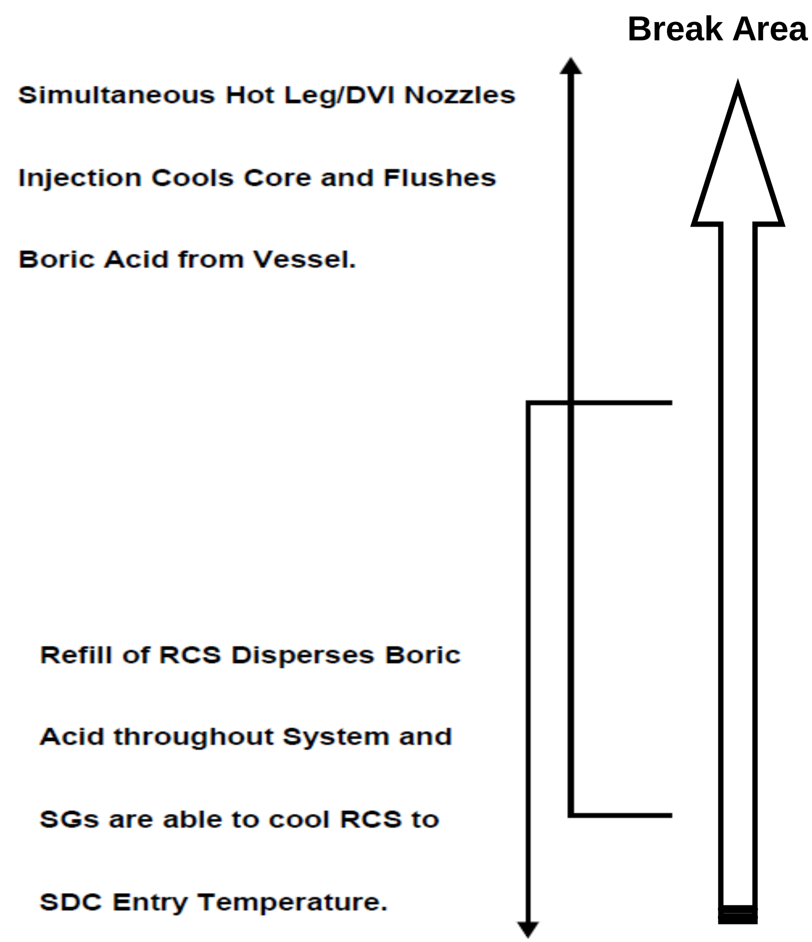
- Long Term Cooling (LTC)
  - ✓ Technical Report Submission

    ; 'Post-LOCA Long Term Cooling Evaluation Model' (APR1400-F-A-NR-14003), Sep. 2014
  - ✓ C-E Post-LOCA Long Term Cooling Evaluation Model (CENPD-254P-A)
  - ✓ Codes
    - CELDA: Long term depressurization and refill of the RCS
    - NATFLOW: Flowrates, pressure and temperature in primary system
    - CEPAC: SG cooldown performance
    - BORON: Transient boric acid concentration in the core
  - ✓ Adopting the Interim Method (Waterford Unit 3, ML050490396)

LTC Analysis Results



Inner Vessel Boric Acid Concentration



Overlap of Acceptable LTC Modes

Overview of Non-LOCA Analyses

- License Information for Non-LOCA
  - ✓ Technical Report Submission

    ; 'Non-LOCA Safety Analysis Methodology of the APR1400' (APR1400-Z-A-NR-14006-P), Sep. 2014
  - ✓ CESEC-Digital Simulation of a CE NSSS (CENPD-107)
  - ✓ CENPD-188, 190, 98, 135, 161, 206, 183, CEN-214, etc.
  - ✓ Codes
    - CESEC-III: NSSS thermal hydraulic transient simulation
    - COAST: Reactor coolant flow coastdown simulation
    - STRIKIN-II: Fuel performance evaluation (temperature & enthalpy)
    - CETOP-D/TORC: Core T/H performance evaluation (DNBR)
    - HERMITE: Reactor core thermal hydraulic transient simulation

Non-LOCA Analysis Results

➢ 15.1 Increase in Heat Removal by the Secondary System (1/2)

Section	Event	Class	Acceptance Criteria	Analysis Results
15.1.1	Decrease in Feedwater Temperature	AOO	Maximum RCS pressure < 110% design pressure	< 2,750 psia
			Maximum SG pressure < 110% design pressure	< 1,320 psia
			Minimum DNBR > 95/95 DNBR Limit	Satisfied
15.1.2	Increase in Feedwater Flow	AOO	Maximum RCS pressure < 110% design pressure	< 2,750 psia
			Maximum SG pressure < 110% design pressure	< 1,320 psia
			Minimum DNBR > 95/95 DNBR Limit	Satisfied
15.1.3	Increase in Steam Flow	AOO	Maximum RCS pressure < 110% design pressure	< 2,750 psia
			Maximum SG pressure < 110% design pressure	< 1,320 psia
			Minimum DNBR > 95/95 DNBR Limit	Satisfied

AOO: Anticipated Operational Occurrence
DNBR: Departure from Nucleate Boiling Ratio

Non-LOCA Analysis Results

➢ 15.1 Increase in Heat Removal by the Secondary System (2/2)

Section	Event	Class	Acceptance Criteria	Analysis Results
15.1.4	Inadvertent Opening of a Steam Generator Relief or Safety Valve	AOO	Maximum RCS pressure < 110% design pressure	< 2,750 psia
			Maximum SG pressure < 110% design pressure	< 1,320 psia
			Minimum DNBR > 95/95 DNBR Limit	Satisfied
15.1.5	Steam System Piping Failure Inside and Outside the Containment	PA	Maximum RCS pressure < 110% design pressure	< 2,750 psia
			Maximum SG pressure < 110% design pressure	< 1,320 psia
			Radiological consequences	Satisfied

PA: Postulated Accident

Non-PROPRIETARY

Non-LOCA Analysis Results

➤ 15.2 Decrease in Heat Removal by the Secondary System (1/3)

Section	Event	Class	Acceptance Criteria	Analysis Results
15.2.1	Loss of External Load	AOO	Maximum RCS pressure < 110% design pressure	< 2,750 psia
			Maximum SG pressure < 110% design pressure	< 1,320 psia
			Minimum DNBR > 95/95 DNBR Limit	Satisfied
15.2.2	Turbine Trip	AOO	Maximum RCS pressure < 110% design pressure	< 2,750 psia
			Maximum SG pressure < 110% design pressure	< 1,320 psia
			Minimum DNBR > 95/95 DNBR Limit	Satisfied
15.2.3	Loss of Condenser Vacuum	AOO	Maximum RCS pressure < 110% design pressure	< 2,750 psia
			Maximum SG pressure < 110% design pressure	< 1,320 psia
			Minimum DNBR > 95/95 DNBR Limit	Satisfied

Non-LOCA Analysis Results

- 15.2 Decrease in Heat Removal by the Secondary System (2/3)

Section	Event	Class	Acceptance Criteria	Analysis Results
15.2.4	Closure of the Main Steam Isolation Valve	AOO	Maximum RCS pressure < 110% design pressure	< 2,750 psia
			Maximum SG pressure < 110% design pressure	< 1,320 psia
			Minimum DNBR > 95/95 DNBR Limit	Satisfied
15.2.6	Loss of Nonemergency AC Power to the Station Auxiliaries	AOO	Maximum RCS pressure < 110% design pressure	< 2,750 psia
			Maximum SG pressure < 110% design pressure	< 1,320 psia
			Minimum DNBR > 95/95 DNBR Limit	Satisfied
15.2.7	Loss of Normal Feedwater Flow	AOO	Maximum RCS pressure < 110% design pressure	< 2,750 psia
			Maximum SG pressure < 110% design pressure	< 1,320 psia
			Minimum DNBR > 95/95 DNBR Limit	Satisfied

Non-PROPRIETARY

Non-LOCA Analysis Results

- 15.2 Decrease in Heat Removal by the Secondary System (3/3)

Section	Event	Class	Acceptance Criteria	Analysis Results
15.2.8	Feedwater System Pipe Break Inside and Outside the Containment	PA	Maximum RCS pressure < 110% design pressure (low) < 120% design pressure (very low)	< 2,750 psia < 3,000 psia
			Maximum SG pressure < 110% design pressure (low) < 120% design pressure (very low)	< 1,320 psia < 1,440 psia
			Radiological consequences	Satisfied

Non-LOCA Analysis Results

➢ 15.3 Decrease in Reactor Coolant System Flow Rate

Section	Event	Class	Acceptance Criteria	Analysis Results
15.3.1	Loss of Forced Reactor Coolant Flow	AOO	Maximum RCS pressure < 110% design pressure	< 2,750 psia
			Maximum SG pressure < 110% design pressure	< 1,320 psia
			Minimum DNBR > 95/95 DNBR Limit	Satisfied
15.3.3	Reactor Coolant Pump Rotor Seizure	PA	Maximum RCS pressure < 110% design pressure	< 2,750 psia
			Maximum SG pressure < 110% design pressure	< 1,320 psia
			Radilogical consequences	Satisfied
15.3.4	Reactor Coolant Pump Shaft Break	PA	Maximum RCS pressure < 110% design pressure	< 2,750 psia
			Maximum SG pressure < 110% design pressure	< 1,320 psia
			Radilogical consequences	Satisfied

Non-LOCA Analysis Results

➢ 15.4 Reactivity and Power Distribution Anomalies (1/2)

Section	Event	Class	Acceptance Criteria	Analysis Results
15.4.1	Uncontrolled Control Element Assembly Withdrawal form a Subcritical or Low-Power Start up Condition	AOO	Peak centerline temperature < melting point	< 20 kW/ft
			Minimum DNBR > 95/95 DNBR Limit	Satisfied
15.4.2	Uncontrolled Control Element Assembly Withdrawal at Power	AOO	Peak centerline temperature < melting point	< 20 kW/ft
			Minimum DNBR > 95/95 DNBR Limit	Satisfied
15.4.3	Control Element Assembly Misoperation	AOO	Peak linear heat generation rate	< 20 kW/ft
			Minimum DNBR > 95/95 DNBR Limit	Satisfied
15.4.4	Startup of an Inactive Reactor Coolant Pump	AOO	Maximum RCS pressure < 110% design pressure	< 2,750 psia
			Maximum SG pressure < 110% design pressure	< 1,320 psia
			Minimum DNBR > 95/95 DNBR Limit	Satisfied

Non-LOCA Analysis Results

➢ 15.4 Reactivity and Power Distribution Anomalies (2/2)

Section	Event	Class	Acceptance Criteria	Analysis Results
15.4.6	Inadvertent Decrease in Boron Concentration in the Reactor Coolant System	AOO	Maximum RCS pressure < 110% design pressure	< 2,750 psia
			Maximum SG pressure < 110% design pressure	< 1,320 psia
			Minimum DNBR > 95/95 DNBR Limit	Satisfied
			Operator action time > 15 minutes (MODEs 1,2,3,4,5)	> 30 minutes
			Operator action time > 30 minutes (MODE 6)	Prohibit (TS3.9.7)
15.4.7	Inadvertent Loading and Operation of a Fuel Assembly in an Improper Position	AOO	Maximum RCS pressure < 110% design pressure	< 2,750 psia
			Maximum SG pressure < 110% design pressure	< 1,320 psia
			Minimum DNBR > 95/95 DNBR Limit	Satisfied
15.4.8	15.4.8 Spectrum of Control Element Assembly Ejection Accidents	PA	Maximum RCS pressure < Service Limit C	Satisfied
			Maximum SG pressure < 110% design pressure	< 1,320 psia
			Maximum fuel rod enthalpy < 230 cal/g	Satisfied
			Radilogical consequences	< well within

Non-LOCA Analysis Results

➢ 15.5 Increase in Reactor Coolant Inventory

Section	Event	Class	Acceptance Criteria	Analysis Results
15.5.1	Inadvertent Operation of the Emergency Core Cooling System that Increases the Reactor Coolant Inventory	AOO	Maximum RCS pressure < 110% design pressure	< 2,750 psia
			Maximum SG pressure < 110% design pressure	< 1,320 psia
			Minimum DNBR > 95/95 DNBR Limit	Satisfied
15.5.2	Chemical and Volume Control System Malfunction that Increases the Reactor Coolant Inventory	AOO	Maximum RCS pressure < 110% design pressure	< 2,750 psia
			Maximum SG pressure < 110% design pressure	< 1,320 psia
			Minimum DNBR > 95/95 DNBR Limit	Satisfied

Non-LOCA Analysis Results

➢ 15.6 Decrease in Reactor Coolant Inventory

Section	Event	Class	Acceptance Criteria	Analysis Results
15.6.1	Inadvertent Opening of a PWR Pressurizer Pressure Relief Valve	PA	Maximum RCS pressure < 110% design pressure	< 2,750 psia
			Maximum SG pressure < 110% design pressure	< 1,320 psia
15.6.2	Failure of Small Lines Carrying Primary Coolant Outside Containment	AOO	Maximum RCS pressure < 110% design pressure	< 2,750 psia
			Maximum SG pressure < 110% design pressure	< 1,320 psia
			Minimum DNBR > 95/95 DNBR Limit	Satisfied
			Radiological consequences	Satisfied
15.6.3	Steam Generator Tube Failure	PA	Maximum RCS pressure < 110% design pressure	< 2,750 psia
			Maximum SG pressure < 110% design pressure	< 1,320 psia
			Radiological consequences	Satisfied

Summary

- APR1400 Design Features and APR1400 Safety Analyses Design Review Status
- Safety Analyses are Conducted in Accordance with the NRC Regulations: LBLOCA/SBLOCA/LTC/Non-LOCA
- Methodologies
  - ✓ LBLOCA: Realistic Evaluation Methodology (CAREM)
  - ✓ SBLOCA: C-E SBLOCA Evaluation Model (CENPD-137P)
  - ✓ LTC: C-E Post-LOCA Long Term Cooling Evaluation Model (CENPD-254P-A)
  - ✓ Non-LOCA: Deterministic Evaluation Methodology (CENDP-107)
- Based on the results of safety analyses, codes and methodologies are applicable to the APR1400 DCD Chapter 15
- APR1400 design is confirmed the satisfaction of acceptance criteria in Safety Analyses