REVISED RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

APR1400 Design Certification

Korea Electric Power Corporation / Korea Hydro & Nuclear Power Co., LTD

Docket No. 52-046

RAI No.:	**154-8064**
SRP Section:	**16 - Technical Specification**
Application Section:	**16 - Technical Specification**
Date of RAI Issue:	**08/17/2015**

Question No. 16-42

Paragraph (a)(11) of 10 CFR 52.47 and paragraph (a)(30) of 10 CFR 52.79 state that a design certification (DC) applicant and a combined license (COL) applicant, respectively, are to propose TS prepared in accordance with 10 CFR 50.36 and 50.36a. 10 CFR 50.36 sets forth requirements for technical specifications to be included as part of the operating license for a nuclear power facility. NUREG-1432, "Standard Technical Specifications-Combustion Engineering Plants," Rev. 4, provides NRC guidance on format and content of technical specifications as one acceptable means to meet 10 CFR 50.36 requirements. Staff needs to evaluate all technical differences from standard TS (STS) NUREG-1432, STS Combustion Engineering Plants, Rev. 4, which is referenced by the DC applicant in DCD Tier 2 Section 16.1, and the docketed rationale for each difference because conformance to STS provisions is used in the safety review as the initial point of guidance for evaluating the adequacy of the generic TS to ensure adequate protection of public health and safety, and the completeness and accuracy of the generic TS Bases.

The applicant is requested to describe

(1) The process employed to ensure identification of technical specification (TS) limiting conditions for operation (LCOs) for all structures, systems, and components (SSCs) as required by 10 CFR 50.36(c)(2)(ii) Criteria 1, 2, 3, and 4.

(2) The process employed to ensure the
  (a) accuracy of the "Background," "Applicable Safety Analyses," "LCO," and "Applicability" sections of the generic TS Bases; and
  (b) consistency of the generic TS Bases with the APR1400 DCD.

Response – (Rev.1)

The process employed in APR1400 TS development is described below.

(1) LCOs selection

To select the LCOs for APR1400 TS, the design characteristics of APR1400 that are different to conventional CE plant design are reviewed by each system engineers. Based on the review results, applicability of existing NUREG-1432 LCOs to APR1400 is examined. A full scope comparison for the applicability of each individual SSC to the 10CFR50.36(c)(2)(ii) LCO selection criteria was performed. The results show that most of the LCOs in NUREG-1432 are applicable to APR1400 in respect to the LCO selection criteria of 10CFR50.36(c)(2)(ii). Attachment 1 includes the evaluation table with additional DCD markups where changes to the LCOs selection criteria are needed.

(2) Background, Applicable Safety Analyses, LCO, Applicability Section, and Consistency to DCD

NUREG-1432 Bases sections are examined for applicability of Background, Applicable section, LCOs by each system engineer. Also, safety analysts reviewed to maintain consistency against Applicable Safety Analyses. This process is commonly applicable for whole DCD. No specific process that is used only for DCD 16 Technical Specification exist.

About the CPC auxiliary trip:

The DCD Tier 2 Section 7.2 and Technical Specifications have CPCS trip parameters based on their own inputs/outputs as follows;

- Section 7.2.1.1 e : Core protection calculators
- Section 7.2.1.4 c : High local power density
- Section 7.2.1.4 d : Low Departure from Nucleate Boiling Ratio
- Table 7.2-4 : High LPD, Low DNBR and CPC Auxiliary Trips
- TS Table 3.3.1-1 High LPD, Low DNBR

As shown in the Figure 1 from DCD Tier 2 Figure 7.2-7, the CPC auxiliary trips credited by safety analysis are described additionally with red lines.

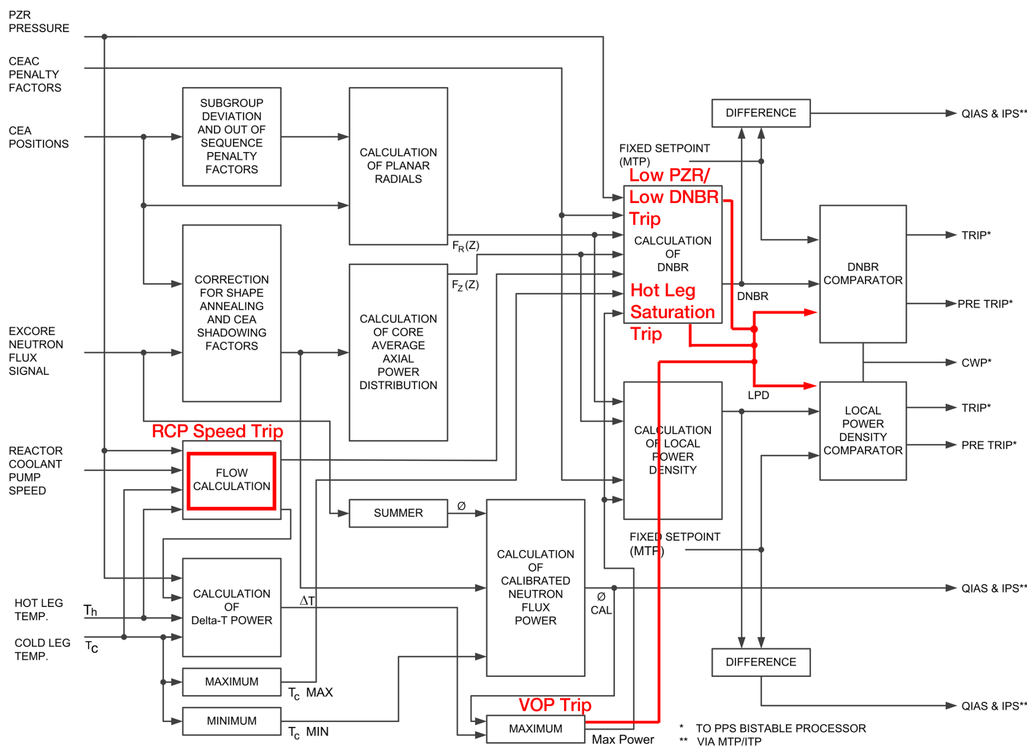

Figure 1 CPC Functional Block Diagram

The auxiliary trips in CPC processor are calculated using the same inputs/outputs and logics with DNBR and LPD trips. Auxiliary trip functions are only software related functions based on the DNBR and LPD measurement channels. All software in the CPC processor is monitored with software cyclic redundancy check. There is no aging and functional degradation issues in the software itself.

To provide more detail requirements, the note (d) in the attachment 2 will be added in the Surveillance Requirement for functions 14 and 15 in TS Table 3.3.1-1 and auxiliary trip parameters in the attachment 2 will be added in the Applicable Safety Analyses for functions 14 and 15 in the BASES. Surveillance Requirement 3.3.1.10 is for the CPC auxiliary trip.

Criteria 4

Impact on DCD

Same as changes described in Impact on Technical Specifications section.

Impact on PRA

There is no impact on the PRA.

Impact on Technical Specifications

Table 3.3.1-1 and Bases 3.3.1, 3.6.7, 3.7.14 and 3.9.6 in TS will be revised as shown in the attached markup.

Impact on Technical/Topical/Environmental Report

There is no impact on any Technical, Topical, or Environment Reports.

Table 1. Evaluation table on the APR1400 TS applicability to LCOs selection criteria

10 CFR 50.36(c)(2)(ii) Criterion 1 :

"Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary."

DCD Section	Installed Instrumentation	TS Section
5.2.5	Containment sump level monitor	3.4.14
5.2.5	Containment air particulate radioactivity monitor	3.4.14
5.2.5	Containment atmosphere humidity monitor	3.4.14

10 CFR 50.36(c)(2)(ii) Criterion 2 :

"A process variable, design feature, or operating restriction that is an initial condition of a design basis accident or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier."

DCD Section	Events	Variable, Feature or Restriction	TS Section
5.4.7	Loss of decay heat removal	· Shutdown Cooling System (SCS) and Coolant Circulation – High Water Level	3.9.4
		· Shutdown Cooling System (SCS) and Coolant Circulation – Low Water Level	3.9.5
5.2.2, 5.3.2.1, 5.4.7, 15.4.4	Overpressure Protection, Pressure-Temperature Limit Curves, Shutdown Cooling System, Startup of an Inactive RCP etc.	· PZR Pressure · Cold Leg Temperatures · Steam Generator Secondary Pressures · SCS Suction Line Relief Valves · SCS Suction Line Isolation Valves	3.4.11
5.4.16, 5A	Intersystem LOCA	· Valve Position Indication · Pressure Indication · Tank Level Indication · Radiation Monitoring	3.4.13
5.2, 5.3 and All relevant sections in Ch.15	All relevant sections including Table 15.0-2, Table 15.0-3	· Reactivity Balance	3.1.2
		· MTC	3.1.3
		· Shutdown CEA Insertion Limit	3.1.5

10 CFR 50.36(c)(2)(ii) Criterion 2 (continued)

DCD Section	Events	Variable, Feature or Restriction	TS Section
5.2, 5.3 and All relevant sections in Ch.15 (Continued)	All relevant sections including Table 15.0-2, Table 15.0-3	· Regulating CEA Insertion Limit	3.1.6
		· Part strength CEA Insertion Limit	3.1.7
		· Planar Radial Peaking Factors (Fxy)	3.2.2
		· Initial RCS Pressure, Temperature, and Flow · Cold Leg Temperature · PZR Pressure · RCS Flow Rate	3.4.1
		· Initial RCS Minimum Temperature for Criticality · Cold Leg Temperature	3.4.2
		· PZR Pressure · Cold Leg Temperature	3.4.3
		· Four RCPs operation in MODES 1 and 2 · RCS Flow Rate · Thermal Power Level	3.4.4
		· Initial Pressurizer Water Level	3.4.9
5.4.10, 15.1, 15.2, 15.4	Increase in Heat Removal by Secondary System etc.	· PZR Pressure · PZR Level	3.4.9
15.1.5, 15.6.3	MSLB / SGTR	· Power · Cold Leg Temperature · PZR Level & Pressure · SG Level · RCS Specific Activity · RCS Operational Leakage Rates	3.4.12, 3.4.15

10 CFR 50.36(c)(2)(ii) Criterion 2 (continued)

DCD Section	Events	Variable, Feature or Restriction	TS Section
15.1.5, 15.6.3 (Continued)	MSLB / SGTR	· Containment Pressure	3.6.4
		· Secondary Specific Activity	3.7.17
15.3.1, 15.3.2, 15.4.3, 15.4.7	Loss of Forced Reactor Coolant Flow /Flow Controller Malfunctions / Control Element Assembly Misoperation / Inadvertent Loading and Operation of a Fuel Assembly in an Improper Position	· POL(Power Operating Limit) conditions	3.2.4
		· ASI	3.2.5
15.4.1	Uncontrolled Control Element Assembly Withdrawal from a Subcritical or Low-Power Startup Condition	· SDM	3.1.1
15.4.2	Uncontrolled Control Element Assembly Withdrawal at Power	· CEA Alignment	3.1.4
		· POL(Power Operating Limit) conditions	3.2.4
		· ASI	3.2.5
		· Linear Heat Rate (LHR)	3.2.1
15.4.4	Startup of an Inactive Reactor Coolant Pump	· SDM	3.1.1
		· PZR Pressure · Cold Leg Temperatures · Steam Generator Secondary Pressures (for Temperatures) · SCS Suction Line Relief Valves · SCS Suction Line Isolation Valves	3.4.11
15.4.6	Inadvertent Decrease in Boron Concentration in the Reactor	· SDM	3.1.1
		· Charging Flow	3.1.8

10 CFR 50.36(c)(2)(ii) Criterion 2 (continued)

DCD Section	Events	Variable, Feature or Restriction	TS Section
15.4.6 (Continued)	Coolant System Inadvertent Decrease in Boron Concentration in the Reactor Coolant System	· Unborated Water Source Isolation Valves (MODE 4 and 5)	3.1.12
		· Unborated Water Source Isolation Valves (MODE 6)	3.9.7
		· RCS boron concentration	3.9.1
15.4.7	Inadvertent fuel assembly loading a spent fuel rack	· Spent Fuel Pool Boron Concentration	3.7.15
		· Spent Fuel Assembly Storage	3.7.16
15.4.8	Spectrum of Control Element Assembly Ejection Accidents	· SDM	3.1.1
		· Linear Heat Rate (LHR)	3.2.1
		· Azimuthal Power Tilt (Tq)	3.2.3
		· POL(Power Operating Limit) Conditions	3.2.4
		· ASI	3.2.5
15.6.5	LOCA	· Control Element Assembly (CEA) Alignment	3.1.4
		· Linear Heat Rate (LHR)	3.2.1
		· Axial Shape Index (ASI)	3.2.5
		· Initial RCS Pressure, Temperature, and Flow	3.4.1
		· Two RCS loops and Four RCPs operation in MODES 1 and 2	3.4.4
		· RCS Pressure and Temperature (P/T) Limits - Cooldown Rates	3.4.3
		· Initial Pressurizer Water Level	3.4.9
		· Steam Generator (SG) Tube Integrity	3.4.17
		· Containment Initial Pressure	3.6.4
		· Containment Initial Air Temperature	3.6.5
		· RCS boron concentration	3.9.1
	Inadvertent fuel assembly loading into a spent fuel rack	· Spent Fuel Pool Boron Concentration	3.7.15
		· Spent Fuel Assembly Storage	3.7.16
15.7.4	Fuel Handling Accident	· Refueling Water Level	3.9.6
		· Spent Fuel Pool Water Level	3.7.14

10 CFR 50.36(c)(2)(ii) Criterion 2 (continued)

DCD Section	Events	Variable, Feature or Restriction	TS Section
15.7.4 (Continued)	Fuel Handling Accident	· Spent Fuel Pool Boron Concentration	3.7.15
		· Decay Time	3.9.8

10 CFR 50.36(c)(2)(ii) Criterion 3 :

"A structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier."

DCD Section	Events	Structure, System, or Component (SSC)	TS Section
5.2	Overpressure Protection	• PZR Pressure	3.4.10
5.4.7	Natural Circulation Cooldown	• Reactor Coolant System (RCS) • Steam Generator (SG) • Shutdown Cooling System (SCS) • Reactor Coolant Gas Venting System(RCGVS)	3.4.16
5.4.10 15.1, 15.2, 15.4,	Increase in Heat Removal by Secondary System etc.	• PZR • POSRV	3.4.9, 3.4.10
6.3, 15.1.5,15.6.3,15.6.5	Steam Line Break (SLB) / Steam Generator Tube Rupture (SGTR) / Small break LOCA / Large break LOCA /	• Safety Injection System (SIS) • In-Containment Refueling Water Storage Tank(IRWST)	3.5.1, 3.5.2, 3.5.3, 3.5.4
7.2.1	In MODES 3, 4, and 5 with RTCBs open or the CEAs not capable of withdrawal, this channel monitors core reactivity condition while shut down. In this condition, the information is used for safety-related operator action.	• Ex-core Neutron Flux Monitoring System (Safety Channel)	3.3.13
7.7.1	Boron Dilution Incident Inadvertent Loading and Operation of a Fuel Assembly	• Ex-core Neutron Flux Monitoring System (Startup Channel)	3.9.2
7.3.1, 9.4.1, 9.4.2, 9.4.6, 15.7.4	Fuel handling accident in the containment building / Fuel handling accident outside the containment	• Containment Purge Isolation Actuation Signal (CPIAS)	3.3.8
		• Control Room Emergency Ventilation Actuation Signal (CREVAS)	3.3.9
		• Fuel Handling Area Emergency Ventilation Actuation Signal (FHEVAS)	3.3.10

10 CFR 50.36(c)(2)(ii) Criterion 3 (continued)

DCD Section	Events	Structure, System, or Component (SSC)	TS Section
All relevant sections in Ch.15	All events in which operation of essential chilled water system is required to be operable	· Chillers, Pumps, Tanks, Valves	3.7.10
	All events which do not generate CREVAS and SIAS but operation of control room HVAC system is required to be operable	· Air Handling Units (AHUs)	3.7.11
	All events requiring all loops to be operable	· Reactor Coolant System (RCS) · Steam Generator (SG) - MSSV, ADV, AFWS	3.4.4 3.7.1, 3.7.4, 3.7.5
	Reactivity and Power Distribution Anomalies etc.	· Reactor Coolant System (RCS) · Steam Generator (SG) · Containment · HVAC	3.4.5 3.6.1, 3.6.4, 3.6.5 3.7.1, 3.7.4, 3.7.5
	All events in which Class 1E electric power supply is credited. (Modes 1,2,3, and 4)	· Offsite power circuits · Class 1E emergency diesel generators · Load sequencers	3.8.1
		· Class 1E DC Power System	3.8.4
		· Class 1E Inverters	3.8.7
		· Class 1E onsite AC, DC, and AC vital bus electrical power distribution systems	3.8.9
	All events in which Class 1E electric power supply is	· Offsite power circuits · Class 1E emergency diesel generators	3.8.2

10 CFR 50.36(c)(2)(ii) Criterion 3 (continued)

DCD Section	Events	Structure, System, or Component (SSC)	TS Section
All relevant sections in Ch.15 (Continued)	credited. (Modes 5 and 6)	· Class 1E DC Power System	3.8.5
		· Class 1E Inverters	3.8.8
		· Class 1E onsite AC, DC, and AC vital bus electrical power distribution systems	3.8.10
	All events in which Class 1E electric power supply is credited. (Modes 1,2,3,4,5, and 6)	· Class 1E batteries – battery parameters	3.8.6
	All events in which the EDGs are required to be operable	· Diesel fuel oil (storage tank, transfer pump, and day tank) · Lube oil and starting air subsystems	3.8.3
		· Emergency Diesel Generator (EDG) – Loss of Voltage Start (LOVS)	3.3.7
	All events in which monitoring of AMI variables is required	· Display system · Qualified Indication and Alarm-Non-Safety System · Qualified Indication and Alarm-P System · Accident Monitoring System	3.3.11
15.1.4	Inadvertent opening of an SG relief or safety valve	· RPS on Low DNBR · RPS on High LPD · RPS on Variable overpower · RPS on Low SG pressure	3.3.1, 3.3.4
		· RPS on Low SG pressure	3.3.2
		· MSIS on low SG pressure · SIAS on low PZR pressure	3.3.5, 3.3.6
		· Main Steam Isolation Valves (MSIVs)	3.7.2
		· Main Feedwater Isolation Valves (MFIVs)	3.7.3

10 CFR 50.36(c)(2)(ii) Criterion 3 (continued)

DCD Section	Events	Structure, System, or Component (SSC)	TS Section
15.1.5	Steam system piping failure inside and outside containment	· RPS on Low DNBR · RPS on CPC variable overpower (sets contacts for High LPD and Low DNBR) · RPS on CPC low RCP shaft speed (sets contacts for High LPD and Low DNBR) · RPS on Low SG pressure · RPS on Low PZR pressure · RPS on High containment pressure	3.3.1
		· RPS on Low SG pressure	3.3.2, 3.3.4
		· MSIS on low SG pressure · SIAS on low PZR pressure · AFAS on low SG level	3.3.5, 3.3.6
		· Containment	3.6.1
		· Containment Airlocks	3.6.2
		· Containment Isolation Valves	3.6.3
		· Main Steam Isolation Valves (MSIVs)	3.7.2
		· Main Feedwater Isolation Valves (MFIVs)	3.7.3
		· In-Containment Refueling Water Storage Tank (IRWST)	3.5.4
		· TSP	3.5.5
		· Containment Spray System (CSS)	3.6.6
		· Air Handling Units (AHUs) · Emergency Air Cleaning Units (ACUs)	3.7.11
15.2.3	Loss of condenser vacuum	· RPS on High PZR pressure · RPS on Low SG level · RPS on Low RCP speed	3.3.1, 3.3.4
		· AFAS on low SG level	3.3.5, 3.3.6

10 CFR 50.36(c)(2)(ii) Criterion 3 (continued)

DCD Section	Events	Structure, System, or Component (SSC)	TS Section
15.2.3 (continued)	Loss of condenser vacuum	· Main Steam Safety Valves (MSSVs)	3.7.1
		· Auxiliary Feedwater System (AFWS) · Auxiliary Feedwater Storage Tank (AFWST)	3.7.5, 3.7.6
15.2.8	Feed water system pipe failure	· RPS on High PZR pressure · RPS on Low SG level · RPS on High containment pressure	3.3.1, 3.3.4
		· AFAS on low SG level · CIAS on high containment pressure	3.3.5, 3.3.6
		· Containment	3.6.1
		· Containment Airlocks	3.6.2
		· Containment Isolation Valves	3.6.3
		· Main Steam Safety Valves (MSSVs) · Main Steam Isolation Valves (MSIVs)	3.7.1, 3.7.2
		· Main Feedwater Isolation Valves (MFIVs)	3.7.3
		· Auxiliary Feedwater System (AFWS) · Auxiliary Feedwater Storage Tank (AFWST)	3.7.5, 3.7.6
		· Air Handling Units (AHUs) · Emergency Air Cleaning Units (ACUs)	3.7.11
15.3.1	Loss of Forced Reactor coolant flow	· RPS on CPC low RCP shaft speed (sets contacts for High LPD and Low DNBR) · RPS on High PZR pressure	3.3.1
		· Main Steam Safety Valves (MSSVs)	3.7.1
		· POSRVs	3.4.10
15.3.3	Reactor Coolant Pump Rotor Seizure	· Hot Leg Low Coolant Flow Rate Trip · RPS on Low reactor coolant flow · RPS on High PZR pressure	3.3.1, 3.3.4

10 CFR 50.36(c)(2)(ii) Criterion 3 (continued)

DCD Section	Events	Structure, System, or Component (SSC)	TS Section
15.3.3 (Continued)	Reactor Coolant Pump Rotor Seizure	· AFAS on low SG level	3.3.5, 3.3.6
		· EDG – Loss of Voltage Start	3.3.7
		· POSRVs	3.4.10
		· Main Steam Safety Valves (MSSVs) · Main Steam Atmospheric Dump Valves (MSADVs)	3.7.1, 3.7.4
		· Auxiliary Feedwater System (AFWS) · Auxiliary Feedwater Storage Tank (AFWST)	3.7.5, 3.7.6
		· Air Handling Units (AHUs) · Emergency Air Cleaning Units (ACUs)	3.7.11
15.4.1	Uncontrolled Control Element Assembly Withdrawal from a Subcritical or Low-Power Startup Condition	· RPS on High logarithmic power level · RPS on Variable overpower · RPS on CPC variable overpower (sets contacts for High LPD and Low DNBR) · RPS on High PZR pressure	3.3.1, 3.3.2, 3.3.3, 3.3.4
15.4.2	Uncontrolled Control Element Assembly Withdrawal at Power	· RPS on Variable overpower · RPS on CPC variable overpower (sets contacts for High LPD and Low DNBR) · RPS on Low DNBR · RPS on High LPD · RPS on High PZR pressure.	3.3.1, 3.3.3, 3.3.4
15.4.3	Control Element Assembly Misoperation	· RPS on Low DNBR · RPS on High LPD	3.3.1, 3.3.3, 3.3.4
		· Required Power Reduction	3.1.4
		· Control Element Assembly Calculators (CEACs)	3.3.3, 3.3.4
15.4.4	Startup of an Inactive Reactor Coolant Pump	· RCS Loops – MODE 3	3.4.5

10 CFR 50.36(c)(2)(ii) Criterion 3 (continued)

DCD Section	Events	Structure, System, or Component (SSC)	TS Section
15.4.6	Inadvertent Decrease in Boron Concentration in the Reactor Coolant System	· Boron Dilution Alarm System (BDAS)	3.3.14
15.4.8	Spectrum of Control Element Assembly Ejection Accidents	· RPS on Variable overpower · RPS on CPC variable overpower (sets contacts for High LPD and Low DNBR) · RPS on Low DNBR · RPS on High LPD · RPS on High PZR pressure.	3.3.1, 3.3.3, 3.3.4
		· AFAS on low SG level	3.3.5, 3.3.6
		· EDG – Loss of Voltage Start	3.3.7
		· POSRVs	3.4.10
		· Containment	3.6.1
		· Containment Airlocks	3.6.2
		· Containment Isolation Valves	3.6.3
		· Main Steam Safety Valves (MSSVs)	3.7.1
		· Auxiliary Feedwater System (AFWS) · Auxiliary Feedwater Storage Tank (AFWST)	3.7.5, 3.7.6
		· Air Handling Units (AHUs) · Emergency Air Cleaning Units (ACUs)	3.7.11

10 CFR 50.36(c)(2)(ii) Criterion 3 (continued)

DCD Section	Events	Structure, System, or Component (SSC)	TS Section
15.5.2	CVCS malfunction that increases reactor coolant inventory	· RPS on High PZR pressure	3.3.1, 3.3.4
		· Main Steam Safety Valves (MSSVs)	3.7.1
		· Auxiliary Feedwater System (AFWS) · Auxiliary Feedwater Storage Tank (AFWST)	3.7.5, 3.7.6
15.6.2	Letdown line break	· RPS on CPC low PZR pressure coincident with low DNBR (≤ 1.45) (LPLD) (sets contacts for High LPD and Low DNBR)	3.3.1, 3.3.4
		· SIAS on low PZR pressure	3.3.5, 3.3.6
		· Air Handling Units (AHUs) · Emergency Air Cleaning Units (ACUs)	3.7.11
15.6.3	SGTR	· RPS on Low DNBR · RPS on Low PZR pressure · RPS on High SG level · RPS on Low PZR pressure and low DNBR (LPLD) (sets contacts for High LPD and Low DNBR) · RPS on CPCS hot leg saturation temperature (sets contacts for High LPD and Low DNBR)	3.3.1, 3.3.4
		· MSIS on high SG level · SIAS on low PZR pressure · AFAS on low SG level	3.3.5, 3.3.6
		· Main Steam Safety Valves (MSSVs) · Main Steam Isolation Valves (MSIVs)	3.7.1, 3.7.2
		· Main Feedwater Isolation Valves (MFIVs)	3.7.3
		· Air Handling Units (AHUs) · Emergency Air Cleaning Units (ACUs)	3.7.11

10 CFR 50.36(c)(2)(ii) Criterion 3 (continued)

DCD Section	Events	Structure, System, or Component (SSC)	TS Section
15.6.5	LOCA	· RPS on Low PZR pressure	3.3.1
		· ESFAS · SIAS on low PRZ pressure	3.3.5, 3.3.6
		· Emergency Diesel Generator (EDG) auto start on low bus voltage · EDG auto start on SIAS, AFAS, and CSAS	3.3.7 3.3.6
		· Safety Injection Tanks (SITs)	3.5.1
		· Safety Injection System (SIS)	3.5.2
		· In-Containment Refueling Water Storage Tank (IRWST)	3.5.4
		· TSP	3.5.5
		· Containment	3.6.1
		· Containment Airlocks	3.6.2
		· Containment Isolation Valves	3.6.3
		· Containment Spray System · CSAS on Containment Pressure – High High signal	3.6.6 3.3.5, 3.3.6
		· Main Steam Safety Valves (MSSVs) · Main Feedwater Isolation Valves (MFIVs)	3.7.1, 3.7.2
		· Auxiliary Feedwater System (AFWS) · Auxiliary Feedwater Storage Tank (AFWST)	3.7.5, 3.7.6
		· CCWS	3.7.7
		· ESWS	3.7.8
		· UHS	3.7.9
		· Air Handling Units (AHUs) · Emergency Air Cleaning Units (ACUs)	3.7.11, 3.7.12
15.7.4	Fuel handling accident	· Spent Fuel Pool Water Level (SFPWL)	3.7.14
		· Air Handling Units (AHUs) · Emergency Air Cleaning Units (ACUs)	3.7.11, 3.7.13
		· Containment Penetrations	3.9.3

10 CFR 50.36(c)(2)(ii) Criterion 4 :
"A structure, system, or component which operating experience or probabilistic risk assessment has shown to be significant to public health and safety"

DCD Section	Structure, System, or Component (SSC)	TS Section
7.8	• Engineered Safety Features Actuation System (ESFAS) - Diverse Manual ESF Actuation Switches	3.3.6
5.4.7	• Shutdown Cooling System, Component Cooling Water System, Essential Service Water System	3.4.6, 3.4.7, 3.4.8 3.7.7, 3.7.8
9.2.1, 9.2.2	• Shutdown Cooling System, Component Cooling Water System, Essential Service Water System	3.4.6 3.4.7, 3.7.7, 3.7.8
7.4.1	• Remote Shutdown Console (RSC)	3.3.12
15.4.4	• RCS Loops - MODE 4	3.4.6
	• RCS Loops - MODE 5 (Loops Filled)	3.4.7
	• RCS Loops - MODE 5 (Loops Not Filled)	3.4.8
19.1.6	• Containment Penetration – Shutdown Operations	3.6.7

※ As described in LCO 3.0.7, compliance with Special Test Exception (STE) LCOs is optional, and therefore no criteria of 10 CFR 50.36(c)(2)(ii) applied to LCOs in Section 3.1.9, 3.1.10, and 3.1.11.

BASES

BACKGROUND (Cont.)	In MODE 6 during shutdown operations, large air exchanges may be required to conduct refueling operations. The high volume purge system is used for this purpose and all valves are closed by the ESFAS such as containment purge isolation actuation signal (CPIAS) and containment isolation actuation signal (CIAS) in accordance with LCO 3.3.5, “Engineered Safety Feature Actuation System (ESFAS) Instrumentation." The containment penetrations that provide direct access from containment atmosphere to outside atmosphere must be isolated on at least one side. Isolation may be achieved by an OPERABLE automatic isolation valve, or by a manual isolation valve, blind flange, or equivalent. Equivalent isolation methods must be approved and may include use of a material that can provide a temporary, atmospheric pressure ventilation barrier for the other containment penetrations during fuel movements.
APPLICABLE SAFETY ANALYSES	Release of fission products to the environment from containment is limited by 10 CFR 50.34. If the LCO requirements are adhered to, then no release exceeding the 10 CFR 50.34 limits can occur (Ref. 1). Shutdown operations satisfy ~~Criterion 3~~ Criteria 4 of 10 CFR 50.36(c)(2)(ii).
LCO	This LCO minimizes the release of radioactivity from containment. The LCO requires the equipment hatch be closed and held in place by [four bolts], one door in each airlock be closed, and each penetration providing direct access to the outside environment to be closed with the exception of the containment purge.
APPLICABILITY	The LCO is applicable during MODE 5 with RCS loops not filled or MODE 6 with the water level < 7.0 m (23 ft) above the top of reactor vessel flange. The equipment hatch keeps closed during these applicability MODEs, because the equipment hatch is administratively closed before the manway of the pressurizer opens.

B 3.7 PLANT SYSTEMS

B 3.7.14 Spent Fuel Pool Water Level (SFPWL)

BASES

BACKGROUND

The minimum water level in the spent fuel pool meets the assumptions of iodine decontamination factors following a fuel handling accident. The specified water level shields and minimizes the general area dose when the storage racks are at their maximum capacity. The water also provides shielding during the movement of spent fuel.

A general description of the spent fuel pool design is found in Subsection 9.1.2 (Ref. 1). A general description of the Spent Fuel Pool Cooling and Cleanup System design is found in Chapter 9.1.3 (Ref. 2). The assumptions of the fuel handling accident are found in Chapter 15.7.4 (Ref. 3).

APPLICABLE SAFETY ANALYSES

The minimum water level in the spent fuel pool meets the assumptions of the fuel handling accident described in Nuclear Regulatory Commission (NRC) Regulatory Guide (RG) 1.195 (Ref. 4). The resultant two hour dose to a person at the exclusion area boundary (EAB) is well within the 10 CFR 50.34 (Ref. 5) limits.

According to Reference 4, there is 7 m (23 ft) of water between the top of the damaged fuel bundle and the fuel pool surface for a fuel handling accident. With 7 m (23 ft) water level, the assumptions of Reference 4 can be used directly. In practice, this Limiting Condition for Operation (LCO) preserves this assumption for the bulk of the fuel in the storage racks. In the case of a single bundle, dropped and lying horizontally on top of the spent fuel racks, however, there could be < 7 m (23 ft) above the top of the fuel bundle and the surface by the width of the bundle. To offset this small non-conservatism, the analysis assumes that all fuel rods fail.

The spent fuel pool water level satisfies ~~Criterion 3~~ Criteria 2 and 3 of 10 CFR 50.36(c)(2)(ii).

LCO

The specified water level preserves the assumptions of the fuel handling accident analysis (Ref. 3). As such, it is the minimum required for irradiated fuel storage within the spent fuel pool.

B 3.9 REFUELING OPERATIONS

B 3.9.6 Refueling Water Level

BASES

BACKGROUND

The movement of irradiated fuel assemblies or performance of CORE ALTERATIONS, except during latching and unlatching of control rod drive shafts, within containment requires a minimum water level of 7 m (23 ft) above the top of the reactor vessel flange. During refueling this maintains sufficient water level in the containment, the refueling canal, the fuel transfer canal, the refueling pool, and the spent fuel pool. Sufficient water is necessary to retain iodine fission product activity in the water in the event of a fuel handling accident (Refs. 1 and 2). Sufficient iodine activity would be retained to limit offsite doses from the accident to under 25% of 10 CFR 50.34 limits, as provided by the guidance of Reference 3.

APPLICABLE SAFETY ANALYSES

During CORE ALTERATIONS and during movement of irradiated fuel assemblies, the water level in the refueling pool and refueling canal is an initial condition design parameter in the analysis of the fuel handling accident in containment postulated by Nuclear Regulatory Commission (NRC) Regulatory Guide (RG) 1.183 (Ref. 1). A minimum water level of 7 m (23 ft) allows a decontamination factor of 200 to be used in the accident analysis for iodine. This relates to the assumption that 99.5% of the total iodine released from the pellet to cladding gap of all the dropped fuel assembly rods is retained by the refueling pool water.

The fuel handling accident analysis inside containment is described in Reference 2. With a minimum water level of 7 m (23 ft) and a minimum decay time of 100 hours prior to fuel handling, the analysis and test programs demonstrate that the iodine release due to a postulated fuel handling accident is adequately captured by the water and offsite doses are maintained within allowable limits (Ref. 4).

Refueling water level satisfies ~~Criterion 4~~ of 10 CFR 50.36(c)(2)(ii).

Criteria 2

LCO

A minimum refueling water level of 7 m (23 ft) above the reactor vessel flange is required to ensure that the radiological consequences of a postulated fuel handling accident inside containment are within acceptable limits (Ref. 3).

Table 3.3.1-1 (Page 3 of 3)
Reactor Protection System Instrumentation – Operating

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITION	SURVEILLANCE REQUIREMENTS
13. Reactor Coolant Flow, Steam Generator #2 Water Level – Low	1, 2	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.9 SR 3.3.1.13
14. Local Power Density – High(c) (d)	1, 2	SR 3.3.1.1 SR 3.3.1.2 SR 3.3.1.3 SR 3.3.1.4 SR 3.3.1.5 SR 3.3.1.7 SR 3.3.1.9 SR 3.3.1.10 SR 3.3.1.11 SR 3.3.1.12 SR 3.3.1.13
15. Departure From Nucleate Boiling Ratio (DNBR) – Low(c) (d)	1, 2	SR 3.3.1.1 SR 3.3.1.2 SR 3.3.1.3 SR 3.3.1.4 SR 3.3.1.5 SR 3.3.1.7 SR 3.3.1.9 SR 3.3.1.10 SR 3.3.1.11 SR 3.3.1.12 SR 3.3.1.13

(c) Trip may be manually bypassed when THERMAL POWER is < 1E-4%. Operating bypass shall be automatically removed when THERMAL POWER is ≥ 1E-4%. During testing pursuant to LCO 3.1.9, trip may be bypassed below 5% RTP. Operating bypass shall be automatically removed when THERMAL POWER is > 5% RTP.

(d) The OPERABILITY of the Local Power Density - High and DNBR - Low Functions include the CPC auxiliary trips.

APPLICABLE SAFETY ANALYSES (continued)

- Primary sample or instrument line break (accident); and
- Steam generator tube rupture (accident).

CPC auxiliary trips

The CPC auxiliary trip parameters are calculated using the same process sensor inputs used to calculate the DNBR and LPD trips. There are no separate dedicated inputs for the auxiliary trip parameters. The CPC output signals of the DNBR - Low and LPD - High reactor trip Functions are sent to the PPS through DNBR trip contact and LPD trip contact, respectively. When any of the calculated auxiliary trip parameters exceeds the trip setting value specified in the SCP, the DNBR trip contact and LPD trip contact are both activated at the same time. This results in the CPC channel sending DNBR - Low and LPD - High reactor trip signals to the PPS.

The CPC auxiliary trip signal parameters and setpoints are the following:

- RCS Cold Leg Temperature, High and Low setpoints

- Hot Pin AXIAL SHAPE INDEX, Positive and Negative setpoints

- Pressurizer Pressure, High and Low setpoints

- Integrated One Pin Radial Peaking Factor, High and Low setpoints

- RCP Shaft Speed, Low setpoint

- CPC Variable Overpower, Ceiling, Increasing Rate, Decreasing Rate, and Step setpoints

- Hot Leg Temperature Saturation Margin, temperature difference setpoint below saturation temperature

- Asymmetric Steam Generator Transient, Loop 1 and Loop 2 cold leg temperature difference setpoint

- Pressurizer Pressure, Low setpoint, Coincident with DNBR, Low setpoint

from a 2-out-of-4 into a 2-out-of-3 logic configuration. The all-bypass function for bypassing all parameters in one channel is interlocked in the LCL algorithm to prevent simultaneous bypass of two or more channels. The all-bypass interlock is implemented based on an analog circuit with hardwired cable between the LCLs of all channels. The purpose of the all-bypass function is to support testing and maintenance of bistable processor (BP) racks whereas the trip channel bypass is used to support repair and testing of failed sensors.

BASES

LCO (continued)

3. CEAC1 processor module failure – this failure is addressed in LCO 3.3.3.

4. CEAC2 processor module failure – this failure is addressed in LCO 3.3.3.

5. CPP1 processor module failure – this failure is addressed in LCO 3.3.3.

6. CPP2 processor module failure – this failure is addressed in LCO 3.3.3.

The CPC channels may be manually bypassed below 1E-4% as sensed by the logarithmic nuclear instrumentation. This bypass is enabled manually in all four CPC channels when plant conditions do not warrant the trip protection. The bypass effectively removes the DNBR – Low and LPD – High trips from the RPS automatically removed when enabling bypass conditions are no longer satisfied.

This operating bypass is required to perform a plant startup, since both CPC generated trips will be in effect whenever shutdown CEAs are inserted. It also allows system tests at low power with pressurizer pressure – low or RCPs off.

PHYSICS

During TESTS pursuant to LCO 3.1.9, the trip may be manually bypassed to make ~~this test possible without reactor trip in condition ≤ 5% RTP.~~

these tests possible without tripping the reactor as long as THERMAL POWER is ≤ 5% RTP.

The OPERABILITY of an LPD - High reactor trip Function channel requires that the channel's CPC auxiliary trip Functions also be OPERABLE.

15. Departure from Nucleate Boiling Ratio (DNBR) – Low

This LCO requires four channels of DNBR – Low to be OPERABLE. The LCO on the CPCs ensures that the SLs are maintained during all AOOs and the consequences of accidents are acceptable.

The CPC channel has many redundant features designed to improve channel reliability. A minimum subset of features must be functional in order for the CPC to be capable of performing its safety related trip function. Therefore, the channel can remain OPERABLE in the presence of a subset of channel failures, while maintaining the ability to provide the DNBR – Low trip function. On-line CPC channel diagnostics make use of redundant features to maintain channel OPERABILITY to the extent possible, and provide alarm and annunciation of detectable failures.

BASES

LCO (continued)

A CPC is not considered inoperable if CEAC inputs to the CPC are inoperable. The Required Actions required in the event of CEAC channel failures ensure the CPCs are capable of performing their safety function.

The CPC channels may be manually bypassed below 1E-4% , as sensed by the logarithmic nuclear instrumentation. This bypass is enabled manually in all four CPC channels when plant conditions do not warrant the trip protection. The bypass effectively removes the DNBR – Low and LPD – High trips from the RPS logic circuitry. The operating bypass is automatically removed when enabling bypass conditions are no longer satisfied.

- This operating bypass is required to perform a plant startup, since both CPC generated trips will be in effect whenever shutdown CEAs are inserted. It also allows system tests at lower power with Pressurizer Pressure – Low or RCPs off.

PHYSICS

During TESTS pursuant to LCO 3.1.9, the trip may be manually bypassed to make ~~this test possible without reactor trip in condition ≤ 5% RTP~~.

these tests possible without tripping the reactor as long as THERMAL POWER is ≤ 5% RTP.

The OPERABILITY of an DNBR - Low reactor trip Function channel requires that the channel's CPC auxiliary trip Functions also be OPERABLE.

Interlocks/Bypasses

The LCO on operating bypass permissive removal channels requires that the automatic operating bypass removal feature of all four operating bypass channels be OPERABLE for each RPS function with an operating bypass in the MODES addressed in the specific LCO for each Function. All four operating bypass removal channels must be OPERABLE to ensure that none of the four RPS channels are inadvertently bypassed.

This LCO applies to the operating bypass removal feature only. If the bypass enable Function is failed so as to prevent entering a bypass condition, operation may continue. In the case of the Logarithmic Power Level – High trip (Function 2), the absence of a bypass will limit maximum power to below the trip setpoint.

The interlock function Allowable Values are based upon analysis of functional requirements for the bypassed Functions. These are discussed above as part of the LCO discussion for the affected Functions.

SURVEILLANCE REQUIREMENTS (continued)

The Surveillance is modified by a Note to indicate that the excore neutron excore detectors are excluded from CHANNEL CALIBRATION because they are passive devices, with minimal drift, and because of the difficulty of simulating a meaningful signal. Slow changes in detector sensitivity are compensated for by performing the daily calorimetric calibration (SR 3.3.1.4) and the monthly linear subchannel gain check (SR 3.3.1.6). In addition, the associated MCR indications are monitored by the operators.

SR 3.3.1.10

Every 18 months, a CHANNEL FUNCTIONAL TEST is performed on the CPCs. The CHANNEL FUNCTIONAL TEST shall include the injection of a signal as close to the sensors as practicable to verify OPERABILITY including alarm and trip Functions.

This surveillance also includes the CPC Auxiliary trip functions.

The basis for the 18 month Frequency is that the CPCs perform a continuous self-monitoring function that eliminates the need for frequent CHANNEL FUNCTIONAL TESTS. This CHANNEL FUNCTIONAL TEST essentially validates the self -monitoring function and checks for a small set of failure modes that are undetectable by the self-monitoring function.

SR 3.3.1.11

The three excore neutron detectors used by each CPC channel for axial flux distribution information are far enough from the core to be exposed to flux from all heights in the core, although it is desired that they only read their particular level. The CPCs adjust for this flux overlap by using the predetermined shape annealing matrix elements in the CPC software.

After refueling, it is necessary to re-establish the shape annealing matrix elements for the excore neutron detectors based on more accurate incore detector readings. This is necessary because refueling could possibly produce a significant change in the shape annealing matrix coefficients.

Incore detectors are inaccurate at low power levels. THERMAL POWER should be significant but < 80% to perform an accurate axial shape calculation used to derive the shape annealing matrix elements.

By restricting power to ≤ 80% until shape annealing matrix elements are verified, excessive local power peaks within the fuel are avoided. Operating experience has shown this Frequency to be acceptable.