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Our ref: DCP/NRC2291

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Subject: AP1000 Responses to Requests for Additional Information (SRP16)

Westinghouse is submitting responses to the NRC requests for additional information (RAIs) on SRP Section 16. These RAI responses are submitted in support of the AP1000 Design Certification Amendment Application (Docket No. 52-006). The information included in the responses is generic and is expected to apply to all COL applications referencing the AP1000 Design Certification and the AP1000 Design Certification Amendment Application.

Responses are provided for the following RAIs:

RAI-SRP16-CTSB-05	RAI-SRP16-CTSB-44	RAI-SRP16-CTSB-74
RAI-SRP16-CTSB-06	RAI-SRP16-CTSB-45	
RAI-SRP16-CTSB-17	RAI-SRP16-CTSB-57	
RAI-SRP16-CTSB-18	RAI-SRP16-CTSB-59	
RAI-SRP16-CTSB-22	RAI-SRP16-CTSB-61	
RAI-SRP16-CTSB-30	RAI-SRP16-CTSB-72	
RAI-SRP16-CTSB-42	RAI-SRP16-CTSB-73	

Questions or requests for additional information related to the content and preparation of this response should be directed to Westinghouse. Please send copies of such questions or requests to the prospective applicants for combined licenses referencing the AP1000 Design Certification. A representative for each applicant is included on the cc: list of this letter.

Very truly yours,

Robert Sisk

Robert Sisk, Manager
Licensing and Customer Interface
Regulatory Affairs and Standardization

D063
NRO

00532psa.doc

/Enclosure

1. Responses to Requests for Additional Information on SRP Section 16

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	J. Wilkinson	-	Florida Power & Light	1E
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ENCLOSURE 1

Responses to Requests for Additional Information on SRP Section 16

AP1000 TECHNICAL REPORT REVIEW

Response to Request For Additional Information (RAI)

RAI Response Number: RAI-SRP16-CTSB-05
Revision: 0

Question:

TS Section 3.1.5, Shutdown Bank Insertion Limits and related bases.
TS Section 3.1.6, Control Bank Insertion Limits.

Explain and justify the Applicability of each of the above TS with respect to the limitation of the TS applying only when OPDMS is inoperable.

If the Shutdown Bank and Control Bank Insertion Limits are not met the Required Actions and Completion Times of TS 3.1.5 and TS 3.1.6 should apply. By making the TS apply only when the OPDMS is inoperable, the LCO can be not met for an extended/indefinite period of time when the OPDMS is Operable, which is not intended.

Westinghouse Response:

TS Section 3.1.5 and TS Section 3.1.6 and associated Bases will be revised to clarify the exclusion of applicability with respect to OPDMS operability. TS LCO 3.1.5, Shutdown Bank Insertion Limits, will be applicable in MODES 1 and 2. TS LCO 3.1.6, Control Bank Insertion Limits, will be applicable in MODES 1 and 2 with Keff ≥ 1.0. Details are provided below.

TS Section 3.1.5, Shutdown Bank Insertion Limits, are necessary to meet accident analysis pre-conditions in MODE 1 and 2 regardless of OPDMS operability. There will be one RIL in the COLR applicable to the Shutdown Banks.

Technical Specification Section 3.1.6, Control Bank Insertion Limits, are necessary for the AO Control Bank to meet accident analysis pre-conditions regardless of the operability of the OPDMS, however provisions for GRCA bank sequence exchange and the OPDMS monitoring of SDM are necessary for the M-Shim Control Banks. GRCA bank sequence exchange will be necessary periodically in MODES 1 and 2. The two exchanging banks will move out of sequence and overlap limits for several minutes during the sequence exchange. GRCA bank sequence exchange is only allowed with the OPDMS OPERABLE to monitor the parameters of LCO 3.2.5. TS Section 3.1.6 will be revised so that the insertion, sequence and overlap limits of LCO 3.1.6, "Control Bank Insertion Limits," will be applicable in MODES 1 and 2 with keff ≥ 1.0, with two exceptions. The LCO will not be applicable when performing SR 3.1.4.2 and the LCO will not be applicable to GRCA banks when performing a GRCA bank sequence exchange with OPDMS OPERABLE. Note that the term "GRCA swap," is replaced with the more descriptive and specific "GRCA bank sequence exchange." TS Section 3.1.4 is also revised to be consistent with this terminology. Provision for the OPDMS monitoring of SDM will be in the RILs included with the COLR. There will be two RILs applicable to the M-Shim Control Banks, MA, MB, MC, MD, M1 and M2, one for when the OPDMS is operable, the other for OPDMS not operable. There will be one RIL for the AO Control Bank regardless of OPDMS status.

Reference(s):

DCD Section 7.7.1.10
DCD Section 4.3.2.4.16

Design Control Document (DCD) Revision:

See attached markups of DCD Revision 17 pages.

PRA Revision:

None

Technical Report (TR) Revision:

None

3.1 REACTIVITY CONTROL SYSTEMS

3.1.4 Rod Group Alignment Limits

LCO 3.1.4 All shutdown and control rods shall be OPERABLE.

AND

Individual indicated rod positions shall be within 12 steps of their group step counter demand position.

- NOTE -

Not applicable to Gray Rod Cluster Assemblies (GRCAs)~~gray rods~~ during GRCA ~~swap~~ bank sequence exchange with OPDMS OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One or more rod(s) inoperable.	A.1.1	Verify SDM to be within the limits specified in the COLR.	1 hour
	OR		
	A.1.2	Initiate boration to restore SDM within limit.	1 hour
	AND		
	A.2	Be in MODE 3.	6 hours
B. One rod not within alignment limits.	B.1	Restore rod, to within alignment limits.	8 hours with the On-Line Power Distribution Monitoring System (OPDMS) OPERABLE OR 1 hour with the OPDMS inoperable
	OR		

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.4 Rod Group Alignment Limits

BASES

BACKGROUND

The OPERABILITY (e.g., trippability) of the RCCAs is an initial assumption in all safety analyses which assume rod insertion upon reactor trip. Maximum rod misalignment is an initial assumption in the safety analysis that directly affects core power distributions and assumptions of available SDM. Gray Rod Cluster Assemblies (GRCAs) are excluded from this LCO during the planned ~~swap~~ GRCA bank sequence exchange~~of the gray rod banks~~, with OPDMS operable. The ~~swap~~ bank sequence exchange of GRCA banks will be periodically necessary to prevent excessive burnup shadowing of fuel rods near the gray rod assemblies. The bank ~~swap~~ sequence exchange maneuver will purposefully misalign GRCAs from their bank for a short period of time. The exclusion from this LCO is acceptable due to SHUTDOWN MARGIN being calculated exclusive of GRCAs, the relative low worth of individual gray rod assemblies, the short time duration anticipated for the ~~swap~~ bank sequence exchange maneuver and with OPDMS operable, power peaking and xenon redistribution effects will be monitored and controlled.

The applicable criteria for these reactivity and power distribution design requirements are 10 CFR 50, Appendix A, GDC 10, "Reactor Design," GDC 26, "Reactivity Control System Redundancy and Protection" (Ref. 1), and 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Plants" (Ref. 2).

Mechanical or electrical failures may cause a control rod to become inoperable or to become misaligned from its group. Control rod inoperability or misalignment may cause increased power peaking due to the asymmetric reactivity distribution and a reduction in the total available rod worth for reactor shutdown. Therefore, control rod alignment and OPERABILITY are related to core operation in design power peaking limits and the core design requirement of a minimum SDM.

Limits on control rod alignment have been established, and all rod positions are monitored and controlled during power operation to ensure that the power distribution and reactivity limits defined by the design power peaking and SDM limits are preserved.

Rod cluster control assemblies (RCCAs) and GRCAs are moved by their control rod drive mechanisms (CRDMs). Each CRDM moves its RCCA or GRCA one step (approximately 5/8 inch) at a time but at varying rates (steps per minute) depending on the signal output from the Plant Control System (PLS).

BASES

LCO	The limits on shutdown or control rod alignments assure that the assumptions in the safety analysis will remain valid. The requirements on control rod OPERABILITY assure that upon reactor trip, the assumed reactivity will be available and will be inserted. The control rod OPERABILITY requirements (i.e., trippability) are separate from the alignment requirements, which ensure that the RCCAs and banks maintain the correct power distribution and rod alignment. The rod OPERABILITY requirement is satisfied provided the rod will fully insert in the required rod drop time assumed in the safety analysis. Rod control malfunctions that result in the inability to move a rod (e.g., rod lift coil failures), but that do not impact trippability, do not result in rod inoperability. The requirement to maintain the rod alignment to within plus or minus 12 steps is conservative. The minimum misalignment assumed in safety analysis is 24 steps (15 inches), and in some cases a total misalignment from fully withdrawn to fully inserted is assumed. Failure to meet the requirements of this LCO may produce unacceptable power peaking factors and linear heating rates (LHR), or unacceptable SDMs, which may constitute initial conditions inconsistent with the safety analysis. The LCO is modified by a Note to relax the rod alignment limit on ~~gray rods~~ Gray Rod Cluster Assemblies (GRCAs) during GRCA ~~swap~~ bank sequence exchange operations. This operation which occurs frequently throughout the fuel cycle would normally violate the LCO.
APPLICABILITY	The requirements on RCCA OPERABILITY and alignment are applicable in MODES 1 and 2 because these are the only MODES in which neutron (or fission) power is generated, and the OPERABILITY (i.e., trippability) and alignment of rods have the potential to affect the safety of the plant. In MODES 3, 4, 5, and 6, the alignment limits do not apply because the control rods are bottomed and the reactor is shut down and not producing fission power. In the shutdown MODES, the OPERABILITY of the shutdown and control rods has the potential to affect the required SDM, but this effect can be compensated for by an increase in the boron concentration of the RCS. See LCO 3.1.1, "SHUTDOWN MARGIN (SDM)," for SDM in MODES 3, 4, and 5 and LCO 3.9.1, "Boron Concentration," for boron concentration requirements during refueling.

3.1 REACTIVITY CONTROL SYSTEMS

3.1.5 Shutdown Bank Insertion Limits

LCO 3.1.5 Each Shutdown Bank shall be within insertion limits specified in the COLR.

| APPLICABILITY: MODES 1 and 2~~ with OPDMS inoperable~~.

- NOTE -
This LCO is not applicable while performing SR 3.1.4.2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more shutdown banks not within limits.	A.1.1 Verify SDM is within the limits specified in the COLR.	1 hour
	OR	
	A.1.2 Initiate boration to restore SDM to within limit.	1 hour
	AND	
	A.2 Restore shutdown banks to within limits.	2 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	6 hours

BASES

LCO	The shutdown banks must be within their insertion limits any time the reactor is critical or approaching criticality. This in conjunction with LCO 3.1.6, "Control Bank Insertion Limits," and 3.2.5.d, OPDMS Monitored Parameters, "SDM," ensures that a sufficient amount of negative reactivity is available to shut down the reactor and maintain the required SDM following a reactor trip. The shutdown bank insertion limits are defined in the COLR.
APPLICABILITY	The shutdown banks must be within their insertion limits with the reactor in MODE 1 and MODE 2. ~~The LCO is not applicable if OPDMS is OPERABLE since OPDMS ensures that a sufficient amount of negative reactivity is available to shut down the reactor and maintain the required SDM following a reactor trip.~~ The shutdown banks do not have to be within their insertion limits in MODE 3, unless an approach to criticality is being made. In MODE 3, 4, 5, or 6 the shutdown banks are fully inserted in the Core and contribute to the SDM. Refer to LCO 3.1.1 for SDM requirements in MODES 3, 4, and 5. LCO 3.9.1, "Boron Concentration" ensures adequate SDM in MODE 6. The Applicability requirements have been modified by a Note indicating that the LCO requirement is suspended during SR 3.1.4.2. This SR verifies the freedom of the rods to move, and requires the shutdown bank to move below the LCO limits, which would normally violate the LCO.
ACTIONS	A.1.1, A.1.2, and A.2 When one or more shutdown banks is not within insertion limits, 2 hours are allowed to restore the shutdown banks to within the insertion limits. This is necessary because the available SDM may be significantly reduced with one or more of the shutdown banks not within their insertion limits. Also, verification of SDM or initiation of boration within 1 hour is required, since the SDM in MODES 1 and 2 is ensured by the continuous monitoring of SDM by the OPDMS (see LCO 3.2.5) and adhering to the control and shutdown bank insertion limits (see LCO 3.1.1). If shutdown banks are not within their insertion limits, then SDM will be verified by the OPDMS or by performing a reactivity balance calculation, considering the effects listed in the BASES for SR 3.1.1.1. The allowed Completion Time of 2 hours provides an acceptable time for evaluating and repairing minor problems without allowing the plant to remain in an unacceptable condition for an extended period of time.

3.1 REACTIVITY CONTROL SYSTEMS

3.1.6 Control Bank Insertion Limits

LCO 3.1.6 Control banks shall be within the insertion, sequence, and overlap limits specified in the COLR.

APPLICABILITY: MODE 1 and MODE 2 with $k_{eff} \geq 1.0$ ~~with OPDMS inoperable~~.

- NOTES -

1. This LCO is not applicable while performing SR 3.1.4.2.
2. This LCO is not applicable to Gray Rod Cluster Assembly (GRCA) banks during GRCA bank sequence exchange with OPDMS OPERABLE.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. Control Bank insertion limits not met.	A.1.1	Verify SDM is within the limits specified in the COLR.	1 hour
	OR		
	A.1.2	Initiate boration to restore SDM to within limit.	1 hour
	AND		
	A.2	Restore control bank(s) to within limits.	2 hours
B. Control bank sequence or overlap limits not met.	B.1.1	Verify SDM is within the limits specified in the COLR.	1 hour
	OR		
	B.1.2	Initiate boration to restore SDM to within limit.	1 hour
	AND		

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.6 Control Bank Insertion Limits

BASES

BACKGROUND

The insertion limits of the shutdown and control rods are initial assumptions in the safety analyses that assume rod insertion upon reactor trip. The insertion limits directly affect core power and fuel burnup distributions and assumptions of available SDM, and initial reactivity insertion rate.

The applicable criteria for these reactivity and power distribution design requirements are 10 CFR 50, Appendix A, GDC 10, "Reactor Design," GDC 26, "Reactivity Control System Redundancy and Protection," GDC 28, "Reactivity Limits" (Ref. 1) and 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors" (Ref. 2). Limits on control rod insertion have been established, and all rod positions are monitored and controlled during power operation to ensure that the power distribution and reactivity limits defined by the design power peaking and SDM limits are preserved.

The rod cluster control assemblies (RCCAs) are divided among control banks and shutdown banks, gray rod cluster assemblies (GRCAs) are limited to control banks. Each bank may be further subdivided into two groups to provide for precise reactivity control. A group consists of two or more RCCAs or GRCAs that are electrically paralleled to step simultaneously. A bank of RCCAs consists of two groups that are moved in a staggered fashion, but always within 1 step of each other. The AP1000 design has seven control banks and four shutdown banks. See LCO 3.1.4, "Rod Group Alignment Limits," for control and shutdown rod OPERABILITY and alignment requirements, and LCO 3.1.7, "Rod Position Indication," for position indication requirements.

The control bank insertion sequence and overlap limits are specified in the COLR. The control banks are required to be at or above the applicable insertion limit lines. There will be two insertion limit lines. Which is applicable will depend on the operability of the Online Power Distribution Monitoring System (OPDMS)

The control banks are used for precise reactivity control of the reactor. The positions of the control banks are normally controlled automatically by the Plant Control System (PLS), but can also be manually controlled. They are capable of adding reactivity very quickly (compared to borating or diluting).

The power density at any point in the core must be limited so that the fuel design criteria are maintained. Together, LCO 3.1.4, "Rod Group

Alignment Limits," LCO 3.1.5, "Shutdown Bank Insertion Limits,"

BASES

BACKGROUND (continued)

LCO 3.1.6, "Control Bank Insertion Limits," and LCO 3.2.5, "OPDMS – Monitored ~~Powered Distribution~~ Parameters," when the OPDMS is OPERABLE, or LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)," and LCO 3.2.4, "QUADRANT POWER TILT RATIO (QPTR)," when the OPDMS is inoperable, provide limits on control component operation and on monitored process variables which ensure that the core operates within the fuel design criteria.

The shutdown and control bank insertion and alignment limits and power distribution limits are process variables that together characterize and control the three dimensional power distribution of the reactor core. Additionally, the control bank insertion limits control the reactivity that could be added in the event of a rod ejection accident, and the shutdown and control bank insertion limits assure the required SDM is maintained <u>when the OPDMS is inoperable</u>.

Operation within the subject LCO limits will prevent fuel cladding failures that would breach the primary fission product barrier and release fission products to the reactor coolant in the event of a loss of coolant accident (LOCA), loss of flow, ejected rod, or other accident requiring termination by a Reactor Trip System (RTS) trip function.

APPLICABLE SAFETY ANALYSES

The shutdown and <u>applicable</u> control bank insertion limits, AFD and QPTR LCOs are required <u>when the OPDMS is inoperable,</u> to prevent power distributions that could result in fuel cladding failures in the event of a LOCA, loss of flow, ejected rod, or other accident requiring termination by an RTS trip function.

The acceptance criteria for addressing shutdown and control bank insertion limits and inoperability or misalignment are that:

a. There be no violations of:

   1. specified fuel design limits, or

   2. Reactor Coolant System (RCS) pressure boundary integrity; and

b. The core remains subcritical after accident transients.

As such, the shutdown and control bank insertion limits affect safety

BASES

analysis involving core reactivity and power distributions (Ref. 3).

APPLICABLE SAFETY ANALYSES (continued)

The SDM requirement is ensured by the continuous monitoring of the OPDMS and by limiting the control and shutdown bank insertion limits when the OPDMS is inoperable, so that allowable inserted worth of the RCCAs is such that sufficient reactivity is available in the rods to shut down the reactor to hot zero power with a reactivity margin which assumes the maximum worth RCCA remains fully withdrawn upon trip (Ref. 3).

Operation at the insertion limits or AFD limits may approach the maximum allowable linear heat generation rate or peaking factor, with the allowed QPTR present. Operation at the insertion limit may also indicate the maximum ejected RCCA worth could be equal to the limiting value in fuel cycles that have sufficiently high ejected RCCA worth.

The control and shutdown bank insertion limits ensure that safety analyses assumptions for SDM (with OPDMS inoperable), ejected rod worth, and power distribution peaking factors are preserved (Ref. 3).

The insertion limits satisfy Criterion 2 of 10 CFR 50.36©(2)(ii) in that they are initial conditions assumed in the safety analysis.

LCO

The limits on control banks sequence, overlap, and physical insertion as defined in the COLR, must be maintained because they serve the function of preserving power distribution, ensuring that the SDM is maintained (when OPDMS is inoperable), ensuring that ejected rod worth is maintained, and ensuring adequate negative reactivity insertion is available on trip. The overlap between control banks provides more uniform rates of reactivity insertion and withdrawal and is imposed to maintain acceptable power peaking during control bank motion.

APPLICABILITY

The control bank sequence, overlap, and physical insertion limits shall be maintained with the reactor in MODES 1 and 2 with $k_{eff} \geq 1.0$. ~~The LCO is not applicable if OPDMS is OPERABLE since OPDMS ensures that the limits are met.~~ There will be two sets of insertion limits applicable to the control banks depending on OPDMS operability. With OPDMS inoperable, ~~T~~these limits must be maintained since they preserve the assumed power distribution, ejected rod worth, SDM, and reactivity rate insertion assumptions. With OPDMS operable and continuously monitoring power distribution and SDM, the applicable insertion limits must be maintained since they preserve the accident analysis assumptions.

Applicability in MODES 3, 4, and 5 is not required, since neither the power distribution nor ejected rod worth assumptions would be exceeded in these MODES.

The applicability requirements are modified by a Note indicating the LCO requirements are suspended during the performance of SR 3.1.4.2. This SR verifies the freedom of the rods to move, and requires the control bank to move below the LCO limits, which would violate the LCO.

<u>The second Note suspends LCO applicability during GRCA bank sequence exchange operations. The two exchanging banks will move out of sequence and overlap limits for several minutes during the sequence exchange. This operation which occurs frequently throughout the fuel cycle would normally violate the LCO. GRCA bank sequence exchange is only allowed with the OPDMS OPERABLE to monitor the parameters of LCO 3.2.5, "OPDMS Monitored Parameters."</u>

ACTIONS

<u>A.1.1, A.1.2, A.2, B.1.1, B.1.2, and B.2</u>

When the control banks are outside the acceptable insertion limits, they must be restored to within those limits. This restoration can occur in two ways:

a. Reducing power to be consistent with rod position; or

b. Moving rods to be consistent with power.

Also, verification of SDM or initiation of boration to regain SDM is required within 1 hour, since <u>with OPDMS inoperable,</u> the SDM in MODES 1 and 2, ensured by adhering to the control and shutdown bank insertion limits (see LCO 3.1.1, "SHUTDOWN MARGIN (SDM)"), has been upset. If control banks are not within their insertion limits, then SDM will be verified by <u>the OPDMS or if the OPDMS is inoperable, by</u> performing a reactivity balance calculation, considering the effects listed in the BASES for SR 3.1.1.1.

Similarly, if the control banks are found to be out of sequence or in the wrong overlap configuration, they must be restored to meet the limits.

Operation beyond the LCO limits is allowed for a short time period in order to take conservative action because the simultaneous occurrence of either a LOCA, loss of flow accident, ejected rod accident, or other accident during this short time period, together with an inadequate power distribution or reactivity capability, has an acceptably low probability.

The allowed Completion Time of 2 hours for restoring the banks to within the insertion, sequence and overlap limits provides an acceptable time for evaluating and repairing minor problems without allowing the plant to

BASES

remain outside the insertion limits for an extended period of time.

C.1

If Required Actions A.1 and A.2, or B.1 and B.2 cannot be completed within the associated Completion Times, the plant must be brought to MODE 2 with k_{eff} < 1.0, where the LCO is not applicable. The allowed Completion Time of 6 hours is reasonable based on operating experience for reaching the required MODE from full power condition in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.1.6.1

This Surveillance is required to ensure that the reactor does not achieve criticality with the control banks below their insertion limits.

BASES

SURVEILLANCE REQUIREMENTS (continued)

The estimated critical position (ECP) depends upon a number of factors, one of which is xenon concentration. If the ECP was calculated long before criticality, xenon concentration could change to make the ECP substantially in error. Conversely, determining the ECP immediately before criticality could be an unnecessary burden. There are a number of unit parameters requiring operator attention at that point. Performing the ECP calculation within 4 hours prior to criticality avoids a large error from changes in xenon concentration, but allows the operator some flexibility to schedule the ECP calculation with other startup activities.

SR 3.1.6.2

Verification of the control banks insertion limits at a Frequency of 12 hours is sufficient to detect control banks that may be approaching the insertion limits since, the insertion limits are monitored and alarms will occur on approach to and/or the exceeding of the limit and normally, very little rod motion occurs in 12 hours.

SR 3.1.6.3

When control banks are maintained within their insertion limits as checked by SR 3.1.6.2 above, it is unlikely that their sequence and overlap will not be in accordance with requirements provided in the COLR. A Frequency of 12 hours is consistent with the insertion limit check above in SR 3.1.6.2.

REFERENCES

1. 10CFR50, Appendix A, GDC 10, GDC 26, and GDC 28.
2. 10CFR50.46.
3. Chapter 15, "Accident Analysis."

AP1000 TECHNICAL REPORT REVIEW

Response to Request For Additional Information (RAI)

RAI Response Number: RAI-SRP16-CTSB-06
Revision: 0

Question:

TS Section 3.2.1, Heat Flux Hot Channel Factor (FQ(Z)) (FQ Methodology); and related TS:
TS Section 3.2.2, Nuclear Enthalpy Rise Hot Channel Factor (NFΔH)) ;
TS Section 3.2.3, AXIAL FLUX DIFFERENCE (AFD) (Relaxed Axial Offset Control (RAOC) Methodology);
TS Section 3.2.4, QUADRANT POWER TILT RATIO (QPTR); and,
TS Section 3.2.5, OPDMS-Monitored Parameters.

Explain and justify the Applicability of each of the above TS with respect to the limitation of the TS applying only when OPDMS is inoperable.

By relying on TS Section 3.2.5, OPDMS-Monitored Parameters, when the OPDMS is Operable, the Completion Times for restoring the parameters to within limits are not consistent with the related TS. In addition, the difference with the STS in using this approach results in SR Frequency changes.

The limits of TS 3.2.1, TS 3.2.2, TS 3.2.3, and TS 3.2.4, apply whether or not the OPDMS is operable; it appears that these TS need to be revised, and that TS 3.2.5 may not be necessary.

Westinghouse Response:

TS Sections 3.2.1 and 3.2.2 provide measurable parameters with limits related to peak linear heat rate and DNBR. They are necessary only in the absence of direct monitoring of those parameters. TS 3.2.3 and 3.2.4 limit reactor operation so that operation until the next surveillance of the parameters of TS Sections 3.2.1 and 3.2.2 will not exceed the limits. The OPDMS utilizes the Westinghouse BEACON™ Core Monitoring System technology to continuously directly monitor the margin to the limits on peak linear heat rate and DNBR. TS Section 3.2.5 replaces the functions of TS Sections 3.2.1, 3.2.2, 3.2,3 and 3.2.4 by requiring the monitored parameters of the OPDMS to be within limits. The OPDMS provides alarms to the reactor operator if a monitored parameter approaches or exceeds a limit. See reference 1 for information on the approved methodology of the BEACON core monitoring system. See reference 2 for present Technical Specifications representative of methodology similar to that of the OPDMS.

The completion time of Action C of TS Section 3.2.5 (15 minutes) is applicable to LCO parameter 3.2.5.d (SDM) and is consistent with other Technical Specification Action completion times where the SDM is not within the limit (eg. 3.1.1 Action A.1). The completion time for Action A of one hour is applied to the power distribution parameters of peak linear heat rate, DNBR and $F_n\Delta h$. TS sections 3.2.1 and 3.2.2 action completion times are based in part on the fact that these limits are measured only periodically on a frequency of many days. In the absence of continuous online monitoring it is possible that by the time these parameters are confirmed to exceed limits, it could be many hours after the core power distribution was actually measured. With continuous online monitoring of the OPDMS, the margin to the limit is known to operations personnel on a continuous basis. Furthermore, these limits protect accident pre-conditions such that significant consequences of exceeding these limits would require a coincident condition 2, 3 or 4 accident. The time of one hour is appropriate as it is extremely unlikely an accident would occur in any one hour of operation coincident with exceeding these limits. One hour allows for operations personnel to evaluate the reactor situation, plan and execute an orderly response (eg. a power distribution change or power reduction) where less time practically allows for only a fast reaction to the condition.

The surveillance frequencies associated with TS Section 3.2.5 are based on alarms and continuous quantitative displays of margins to these limits available to control room operators.

Reference(s):

1. WCAP-12472-P-A, BEACON™ Core Monitoring and Operations Support System, C. Beard and T. Morita, June, 1994
2. Technical Specifications, Docket Number STN 50-456, Braidwood Unit 1

Design Control Document (DCD) Revision:

None

PRA Revision:

None

Technical Report (TR) Revision:

None

AP1000 TECHNICAL REPORT REVIEW

Response to Request For Additional Information (RAI)

RAI Response Number: RAI-SRP16-CTSB-17
Revision: 0

Question:

Bases 3.1.5; Page B.3.1.5-2 and associated Bases (B.3.1.5; Page B.3.1.5-4):

Provide clarification/justification regarding conflicting statements defining SDM with respect to gray rod cluster assemblies (GRCAs). State specifically which control rods are to be inserted during a scram in order to maintain SDM.

The Bases for Section 3.1.5 appears to conflict with Sections 3.1.1 and 3.1.6 in defining SDM. The Bases for Section 3.1.1 states " ..SDM defines the degree of subcriticality that would be obtained immediately following the insertion or scram of all Rod Cluster Control Assemblies (RCCAs), assuming that the single rod cluster assembly of highest reactivity worth is fully withdrawn." This appears to conflict with the statement in the Bases for Section B 3.1.5, "On a reactor trip, all RCCAs (shutdown banks and control banks exclusive of GRCAs), except the most reactive RCCA, are assumed to insert into the core." The discrepancy in these two statements involves the GRCAs (a subgroup of the Control Bank).

Westinghouse Response:

Following a reactor trip, all RCCAs and GRCAs will be released from their Control Rod Drive Mechanisms (CRDMs) and will be free to fall to the bottom of the reactor. For the purpose of meeting the SDM definition in MODES 1 and 2 with keff ≥ 1.0 however, the worth of the GRCAs is conservatively ignored in the calculated inserted rod worth. In MODES 2 with keff < 1.0, 3, 4 and 5, after GRCAs are confirmed to have inserted, the worth of the GRCAs may be credited in the calculation of SDM.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

Technical Report (TR) Revision:

None

AP1000 TECHNICAL REPORT REVIEW

Response to Request For Additional Information (RAI)

RAI Response Number: RAI-SRP16-CTSB-18
Revision: 0

Question:

LCO 3.1.5; Page 3.1.5-1 and associated Bases (B.3.1.5; Page B.3.1.5-3) and LCO 3.2.5; Page 3.2.5-1 and associated Bases (B.3.1.5; Page B.3.1.5-3):

Discuss the methodology and process used to calculate SDM using the OPDMS or provide the reference for the applicable approved method. Provide justification for why LCO 3.1.5 is not applicable when OPDMS is operable. LCOs 3.1.5 and 3.2.5 both relate to the operability of the online power distribution monitoring system (OPDMS) and its capability to calculate SDM.

The staff needs to understand the methodology and/or process of how shutdown margin (SDM) is calculated using OPDMS (e.g.: how the various parameters, in-core neutronics and actual flux distribution, are measured and if values are being extrapolated, etc. Assuming instrumentation is octal symmetric - is the algorithm and placement of the instruments sensitive to individual rod position?).

This RAI also applies to Section 3.1.6, "Control Bank Insertion Limits," Section 3.1.7, "Rod Position Indication," and Section 3.2.5, "OPDMS-Monitored Parameters."

Westinghouse Response:

TS Section 3.1.5 and associated BASES will be revised to clarify the exclusion of applicability when the OPDMS is operable as discussed in the response to RAI-SRP 16-CTSB-05. TS LCO 3.1.5 will be applicable in MODES 1 and 2 with the only exemption being for performance of SR 3.1.4.2.

TS Section 3.1.6 and associated BASES will be revised to clarify the exclusion of applicability when the OPDMS is operable as discussed in the response to RAI-SRP 16-CTSB-05. TS Section LCO 3.1.6 will be applicable in MODES 1 and 2 with keff ≥ 1.0, with exemptions for performance of SR 3.1.4.2 and for GRCA bank sequence exchange.

TS Section 3.1.7 is related to the OPDMS in that several action statements refer to using the OPDMS (or the incore detectors) to "verify the position of the rods." This use of the OPDMS is only applicable when in the action statements of TS Section 3.1.7. The OPDMS will rely on bank demand primarily, and digital rod position indication as a contingency, for rod position information in support of the calculation of power distribution and SDM. This is consistent with the presently approved methodology per reference 1. In the event of an inoperable or misaligned rod, the capability will exist for manual input of the rod position and it operability status to the OPDMS, to allow for appropriate monitoring of SDM.

The OPDMS will continuously monitor SDM in MODES 1 and 2 with keff ≥ 1.0. The OPDMS will utilize the approved technology of the BEACON™ core monitoring system to monitor power distribution and SDM based on plant operation data (reference 1). BEACON uses the methods of PHOENIX-P/ANC (Reference 2) and/or NEXUS/ANC (reference 3), which are approved to calculate core reactivity in a wide variety of conditions, to continuously calculate the SDM condition of the reactor as follows:

> BEACON will calculate the RCS critical boron concentration of the reactor in the present condition, CBi.
> BEACON then calculates the RCS boron concentration required following a reactor trip from the present condition to meet SDM criteria accounting for the most reactive rod stuck out, CB1. GRCAs are conservatively excluded from this calculation by the assumption that they do not move on trip.
>
> The difference between CBi and CB1 is calculated and a factor of safety (FS) applied to account for uncertainties. This difference is defined as "Excess Shutdown Margin (ESDM)," like:
>
> ESDM = CBi – [CB1 + FS]

Compliance with the SDM requirements of LCO 3.2.5 is demonstrated by confirmation of ESDM greater than zero. The value of ESDM will be continuously calculated by the OPDMS and available for display to operations personnel in MODES 1 and 2. ESDM will be calculated in units of RCS boron concentration but may be converted to other reactivity units for display to operations personnel.

If the OPDMS is not operable, SHUTDOWN MARGIN is assured in MODES 1 and 2, keff ≥ 1.0, by operation in compliance with TS LCOs, 3.1.5, "Shutdown Bank Insertion Limits," and 3.1.6, "Control Bank Insertion Limits." There will be one RIL included in the COLR for the Shutdown Banks consistent with accident analysis and SDM requirements. There will two RILs applicable to the M-shim control banks MA, MB, MC, MD, M1 and M2. One M-shim RIL will be applicable with OPDMS OPERABLE, which will not include SDM considerations (because the OPDMS is monitoring SDM). Another more restrictive RIL to include SDM considerations will be included in the COLR for when OPDMS is inoperable. Also there will be one RIL applicable to the AO Bank that is independent of OPDMS operability. WCAP-9272-P-A (reference 4) describes that the Rod Insertion Limits for SDM are calculated such that SDM is ensured with the control banks positioned above the limits. The methodology for this calculation is as follows:

SDM = Power Defect – [Inserted RCCA worth assuming most reactive RCCA stuck out + FS]

Power Defect is the reactivity difference between the tripped reactor condition and the present reactor condition.

In the OPDMS calculation of ESDM, the difference in core conditions between the two calculated boron concentrations (CBi and CB1) includes the power defect, and the control rod worth currently available for insertion. The credited available control rod worth excludes the highest worth stuck control rod (as required) and also conservatively excludes the worth potentially available from insertion of the GRCAs. Since the required shutdown margin components are built into the calculation of CB1, the method employed in the OPDMS continuous online shutdown margin calculation for Mode 1 and 2 with keff ≥1.0, is functionally equivalent to the bounding method currently used to validate a particular rod insertion limit and verify SDM in off-line design calculations.

Reference(s):

1. WCAP-12472-P-A, BEACON™ Core Monitoring and Operations Support System, C. Beard and T. Morita, June, 1994 (through addendum 3)
2. WCAP-11596-P-A, "Qualification of the PHOENIX-P/ANC Nuclear Design System for Pressurized Water Reactor Cores," June 1988.
3. WCAP 16045-P-A, Addendum 1-A, "Qualification of the NEXUS Nuclear Data Methodology," August 2007.)
4. WCAP-9272-P-A, "Westinghouse Reload Safety Analysis Methodology, July, 1985.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

Technical Report (TR) Revision:

None

AP1000 TECHNICAL REPORT REVIEW

Response to Request For Additional Information (RAI)

RAI Response Number: RAI-SRP16-CTSB-22
Revision: 0

Question:

LCO 3.2.1; Page 3.2.1-4 (SR 3.2.1.2):

Resolve discrepancy between "Note a." in SR 3.2.1.2 and "Note a." in the associated Bases for SR 3.2.1.2 (Surveillance Requirements). Guidance provided in the Bases is not included in LCO 3.2.1 "Note a." which ensures the last FQW (Z) measurement is to be increased. Also, provide an explanation for reverifying that FQW(Z) is within limits (this reasoning is not provided in the Bases).

SR 3.2.1.2 Note states:
If FQW(Z) measurements indicate maximum over zFQC(Z) has increased since the previous evaluation of FQC(Z) :

a. Increase FQW(Z) by the greater of a factor of 1.02 or by an appropriate factor specified in the COLR and reverify FQW(Z) is within limits; or

b. Repeat SR 3.2.1.2 once per 7 EFPD until two successive flux maps indicate maximum over zFQC(Z) has not increased.

Bases SR 3.2.1.2 states:
If the two most recent FQ(Z) evaluations show an increase in FQC(Z) , it is required to meet the FQ (Z) limit with the last FQW(Z) increased by the greater of a factor of 1.02 or by an appropriate factor as specified in the COLR or to evaluate FQ(Z) more frequently, each 7 EFPDs. These alternative requirements will prevent FQ(Z) from exceeding its limit for any significant period of time without detection.

The Bases specifically states to use the last FQW(Z) when increasing by a factor of 1.02 or the appropriate COLR factor, however, "Note a" of LCO 3.2.1 does not specifically provide this guidance.

Westinghouse Response:

SR 3.2.1.2 reads:

SR 3.2.1.2 --

- NOTE -

If $F_Q^W(Z)$ measurements indicate

maximum over z $[F_Q^C(Z)]$

has increased since the previous evaluation of $F_Q^C(Z)$:

a. Increase $F_Q^W(Z)$ by the greater of a factor of 1.02 or by an appropriate factor specified in the COLR and verify $F_Q^W(Z)$ is within limits; or

b. Repeat SR 3.2.1.2 once per 7 EFPD until two successive flux maps indicate maximum over z $[F_Q^c(Z)]$ has not increased.

Verify F_Q^W (Z) within limits.

SR 3.2.1.2 of the AP1000 Technical Specifications and Bases sections is of the same form as SR 3.2.1.2 of Reference 1. It appears slightly different because the value of K(z) will always be 1.0 for the AP1000 and is removed from the expression.

Part a of the note modifying SR 3.2.1.2 is intended to provide additional assurance that reactor operation until the next surveillance of $F_Q^C(Z)$ will not result in exceeding limits. The note instructs that if the present measured value of $F_Q^C(Z)$ is greater than the value measured in the previous surveillance, then the present measured value of $F_Q^W(Z)$ is to be increased by 1.02, or greater factor, and this increased value compared to the limits. If the increased value of $F_Q^W(Z)$ meets the limits, operation can continue until the next required surveillance. If the increased value of $F_Q^W(C)$ does not meet the limits, than compliance with part b of the note is required. Part b of the note reasonably requires more frequent surveillance until the trend in $F_Q^C(Z)$ starts to decrease.

The Bases describing this surveillance refer to the present measured value of $F_Q^W(Z)$ as the "last" value.

Reference(s):

1. NUREG-1431, "Standard Technical Specifications Westinghouse Plants," Volumes 1 and 2.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

Technical Report (TR) Revision:

None

AP1000 TECHNICAL REPORT REVIEW

Response to Request For Additional Information (RAI)

RAI Response Number: RAI-SRP16-CTSB-30
Revision: 0

Question:

TS 3.7.1, Main Steam Safety Valves (MSSVs).

Revise TS 3.7.1 and related information in TS bases B 3.7.1 to provide a completion time (CT) for the Required Action A.1 including a justification for the selected value.

A CT was missing for the Required Action A.1. In the Westinghouse STS 3.7.1, a CT of 4 hours is posted for a similar action.

Westinghouse Response:

See response to identical RAI - RAI-SRP-16-CTSB-12.

Design Control Document (DCD) Revision:

See response to identical RAI - RAI-SRP-16-CTSB-12.

PRA Revision:

None

Technical Report (TR) Revision:

None

AP1000 TECHNICAL REPORT REVIEW

Response to Request For Additional Information (RAI)

RAI Response Number: RAI-SRP16-CTSB-42
Revision: 0

Question:

TS 3.3.1; Table 3.3.1-1, Page 2; Table 3.3.1-1, Note 1 (overtemperature ΔT) and Note 2 (overpower ΔT):

Provide the technical bases and derivation of the revised overtemperature ΔT and overpower ΔT reactor trip functions and submit a specific reference to a supporting analytical method that has been approved by the staff for these functions, or submit an appropriate methodology to the staff for further review. Per Generic Letter 88-16, include the approved method for these equations in TS Section 5.6.5. The current overtemperature ΔT and overpower ΔT equations (Rev. 15) were taken from WCAP-8745-P-A. However, Revision 16 makes a change to these equations but does not provide a revision to WCAP-8745 or a reference to another approved method containing the technical bases for the proposed equations.

In addition, provide definitions for f1(ΔI) and f2(ΔI). The definitions of these terms were not presented or referenced in Table 3.3-1. The definitions should be consistent with those provided in DCD 7.2.1.1.3 and the supporting analytical methods.

Per Generic Letter 88-16, "Removal of Cycle-Specific Parameter Limits from Technical Specifications," the analytical methods used to determine the core operating limits must be those previously reviewed and approved by the staff. The revised calculation formulas presented for these functions differ from those previously submitted in Revision 15 of DCD 7.2.1.1.3 and Technical Specification Table 3.3-1. Reference is also made in Note 1 of Table 3.3.1-1 to interpolation from tables of allowable core thermal power as a function of core inlet temperature at various pressures. The bases for these formulas and for development of the tables were not referenced to methods approved by the staff, either in the Bases for LCO 3.3.1 or in Technical Specification 5.6.5, "Core Operating Limits Report (COLR)."

Westinghouse Response:

Westinghouse does not consider the change in the OTDT/OPDT equations as a change in methodology within the intent of Generic Letter 88-16. In summary, the digital formulation provides a simpler and direct means of converting the core thermal design limits to protection system setpoints than the previous analog formulation.

The analog formulation for the OTDT/OPDT reactor trips dates from 1966-68 when it was first applied to W-PWRs. At that time, a novel design was proposed to minimize errors in measuring ΔT. To eliminate the errors in obtaining an absolute temperature for T-hot and T-cold and then subtracting (thus adding the individual errors in measuring T-hot and T-cold), a four-RTD bridge was proposed. With two hot leg RTDs and two cold leg RTDs, ΔT would be obtained directly across two points of the bridge and Tavg obtained across the other two points. That design was soon obsolete as a result of (a) improvements in instrumentation; and (b) discovery that errors caused by hot leg streaming were large compared to instrument measurement errors. However, the use of Tavg and ΔT was already frozen into the protection system design despite the fact that the core thermal design limits are computed as functions of core inlet temperature (T-inlet) and power.

The conversion of T-inlet and power to Tavg and ΔT is a non-trivial process considering the non-linearities in the density and heat capacity of water. That led to the OPTOWAX code and was one of the two major reasons for generation of the topical report, WCAP-8745. (The other reason for WCAP-8745 was to document the nuclear-thermal-design, or N-T-H, basis for generation of the core thermal design limits. That basis has not been changed.)

With improved instrumentation for converting RTD resistance to temperature signals, high accuracy is obtained by measuring T-cold and T-hot directly. Thus, the OTDT/OPDT equations can be re-arranged such that the setpoint is computed as a function of T-cold (equal to T-inlet in the N-T-H design) rather than as a function of Tavg. And with the advent of digital systems, the core thermal power can be more accurately calculated than with the approximations inherent in using a ΔT signal. Thus, the ΔT power signal, that uses Thot, Tcold, and pressure to compute cold leg density and hot leg and cold leg enthalpies, is directly analogous to the ΔT signal but is a much more accurate measure of core thermal power.

Note that the ΔT power signal can be directly compared to the measured calorimetric power, and is thereby easier to monitor and trend errors. No correction for RCS temperature, pressure, or power level is needed.

Regarding f1(ΔI) and f2(ΔI): The protection system instrumentation for all operating W-PWRS with OPDT/OPDT uses a signal of axial flux difference, ΔI, to generate a reduction in the OPDT or OPDT trip setpoints. That continues to be the case for AP1000, with the only difference being that the trip setpoint is ΔT power instead of ΔT. The shape of the penalty function (zero if ΔI is within some range, then a linear increase if ΔI increases outside that range) is also unchanged. The definitions for f1(ΔI) and f2(ΔI) as they are used in Table 3.3.-1 of the Technical Specifications has been provided in the Westinghouse response to RAI-SRP16-CTSB-69.

Separation of Tavg and ΔT into Thot and Tcold also simplifies the dynamic compensation for instrument lags and transient time from the cold leg RTD to the core (to adjust for the difference between T-inlet and measured T-cold), or from the core to the hot leg RTD. The dynamic compensation need account only for the time difference in each independent signal.

Hot leg boiling is a measurement limit rather than a safety limit. As is now well known, hot leg streaming can cause large variations between local hot leg temperatures. Hence, one RTD could approach saturation – and become invalid for measurement -- well before the bulk fluid in the hot leg approaches saturation. (In the AP1000, the temperature signals from three separate

RTDs, located approximately 120° apart around the hot leg circumference, are combined to obtain a hot leg temperature signal in each protection system division.) The digital design permits setting an individual RTD temperature signal to a trip value if it approaches saturation. This minimizes the need to assign uncertainties to the variation between local RTD temperature and bulk hot leg fluid temperature.

Additional information on the digital OTDT/OPDT reactor trips and comparison with the analog formulation is provided in the Westinghouse response to RAI-SRP16-CTSB-03.

In conclusion, The digital OTDT and OPDT reactor trip formulation substantially simplifies the conversion from the core thermal design limits (which remain unchanged from those described in WCAP-8745) to reactor trip setpoints, and therefore meets the intent of Generic Letter 88-16.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

Technical Report (TR) Revision:

None

AP1000 TECHNICAL REPORT REVIEW

Response to Request For Additional Information (RAI)

RAI Response Number: RAI-SRP16-CTSB-44
Revision: 0

Question:

TS 3.3.1 Reactor Trip System Instrumentation

For Table 3.3-1, function 6 (over temperature ΔT) and function 7 (overpower ΔT), clarify the "required channels" column entry by adding appropriate notation that would make Table 3.3-1 consistent with DCD Table 7.2-2, sheet 1, and would be consistent with descriptions of other required channels presented on Table 3.3.1-1.

Westinghouse Response:

DCD Table 7.2-2 Overtemperature ΔT and Overpower ΔT Functions specify "4 (2/loop)" number of channels. Specification Table 3.3-1, Functions 6 and 7 and the associated Bases are marked up to be consistent with Table 7.2-2 with the addition of "(2/loop)."

Design Control Document (DCD) Revision:

Markups of the affected DCD Revision 17 pages are attached.

Note that the change to Function 12 is included in the attached markups according to the RAI-SRP16-CTSB-45 response, since the changes also affect page 3.3.1-13.

PRA Revision:

None

Technical Report (TR) Revision:

None

Table 3.3.1-1 (page 2 of 5)
Reactor Trip System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	TRIP SETPOINT
6. Overtemperature ΔT	1,2	4 (2/loop)	E	SR 3.3.1.1 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 3.3.1.12	Refer to Note 1 Table 3.3.1-1 (Page 5 of 5)	Refer to Note 1 Table 3.3.1-1 (Page 5 of 5)
7. Overpower ΔT	1,2	4 (2/loop)	E	SR 3.3.1.1 SR 3.3.1.3 SR 3.3.1.7 SR 3.3.1.9 SR 3.3.1.12	Refer to Note 2 Table 3.3.1-1 (Page 5 of 5)	Refer to Note 2 Table 3.3.1-1 (Page 5 of 5)
8. Pressurizer Pressure						
a. Low Setpoint	1(f)	4	K	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.9 SR 3.3.1.12	≥ 1809.9 psig	1810.3 psig
b. High Setpoint	1,2	4	E	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.9 SR 3.3.1.12	≤ 2420.7 psig	2420.3 psig
9. Pressurizer Water Level – High 3	1(f)	4	K	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.9 SR 3.3.1.12	≤ 71.05%	71%
10. Reactor Coolant Flow – Low	1(f)	4 per hot leg	K	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.9 SR 3.3.1.12	≥ 89.96%(i)	90%(i)
11. Reactor Coolant Pump (RCP) Bearing Water Temperature – High	1,2	4 per RCP	E	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.9 SR 3.3.1.12	≤ 190.4°F	190°F
12. RCP Speed – Low	1(f)	4 (1/pump)	K	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.9 SR 3.3.1.12	≥ 90.9%	91%

(f) Above the P-10 (Power Range Neutron Flux) interlock.

(i) 90% of loop specific indicated flow.

BASES

APPLICABLE SAFETY ANALYSES, LCOs, and APPLICABILITY (continued)

compensation to account for cold leg-to-core transit time;

- pressurizer pressure – the Trip Setpoint is varied to correct for changes in system pressure; and
- axial power distribution – the Trip Setpoint is varied to account for imbalances in the axial power distribution as detected by the PMS upper and lower power range detectors. If axial peaks are greater than the design limit, as indicated by the difference between the upper and lower PMS power range detectors, the Trip Setpoint is reduced in accordance with Note 1 of Table 3.3.1-1.

Dynamic compensation of the ΔT power signal is included for system piping delays from the core to the temperature measurement system. The Overtemperature ΔT trip Function is calculated for each loop as described in Note 1 of Table 3.3.1-1. This Function also provides a signal to generate a turbine runback prior to reaching the Trip Setpoint. A turbine runback will reduce turbine power and reactor power. A reduction in power will normally alleviate the Overtemperature ΔT condition and may prevent a reactor trip. No credit is taken in the safety analyses for the turbine runback.

The LCO requires four channels <u>(two per loop)</u> of the Overtemperature ΔT trip Function to be OPERABLE in MODES 1 and 2. Four channels are provided to permit one channel in trip or bypass indefinitely and still ensure no single random failure will disable this trip Function. Note that the Overtemperature ΔT Function receives input from channels shared with other RTS Functions. Failures that affect multiple Functions require entry into the Conditions applicable to all affected Functions.

In MODE 1 or 2, the Overtemperature ΔT trip must be OPERABLE to prevent DNB. In MODE 3, 4, 5, or 6, this trip Function does not have to be OPERABLE because the reactor is not operating and there is insufficient heat production to be concerned about DNB.

7. <u>Overpower ΔT</u>

The Overpower ΔT trip Function ensures that protection is provided to ensure the integrity of the fuel (i.e., no fuel pellet melting and less than 1% cladding strain) under all possible overpower conditions. This trip Function also limits the required range of the Overtemperature ΔT trip function and provides a backup to the Power Range Neutron Flux – High Setpoint trip. The Overpower ΔT

APPLICABLE SAFETY ANALYSES, LCOs, and APPLICABILITY (continued)

trip Function ensures that the allowable heat generation rate (kW/ft) of the fuel is not exceeded. It uses the same ΔT power signal generated for the Overtemperature ΔT. The setpoint is automatically varied with the following parameter:

- Axial power distribution – the Trip Setpoint is varied to account for imbalances in the axial power distribution as detected by the PMS upper and lower power range detectors. If axial peaks are greater than the design limit, as indicated by the difference between the upper and lower PMS power range detectors, the Trip Setpoint is reduced in accordance with Note 2 of Table 3.3.1-1.

The Overpower ΔT trip Function is calculated for each loop as per Note 2 of Table 3.3.1-1. The Trip Setpoint reflects the inclusion of both steady state and adverse environmental instrument uncertainties as the detectors provide protection for a steam line break and may be in a harsh environment. Note that this Function also provides a signal to generate a turbine runback prior to reaching the Trip Setpoint. A turbine runback reduces turbine power and reactor power. A reduction in power normally alleviates the Overpower ΔT condition and may prevent a reactor trip.

The LCO requires four channels (two per loop) of the Overpower ΔT trip Function to be OPERABLE in MODES 1 and 2. Four channels are provided to permit one channel in trip or bypass indefinitely and still ensure no single random failure will disable this trip Function. The Overpower ΔT Function receives input from channels shared with other RTS Functions. Failures that affect multiple Functions require entry into the Conditions applicable to a affected Functions.

In MODE 1 or 2, the Overpower ΔT trip Function must be OPERABLE. These are the only times that enough heat is generated in the fuel to be concerned about the heat generation rates and overheating of the fuel. In MODE 3, 4, 5, or 6, this trip Function does not have to be OPERABLE because the reactor is not operating and there is insufficient heat production to be concerned about fuel overheating and fuel damage.

APPLICABLE SAFETY ANALYSES, LCOs, and APPLICABILITY (continued)

The LCO requires four RCP Speed – Low channels (one per pump) to be OPERABLE in MODE 1 above P-10. Four channels are provided to permit one channel in trip or bypass indefinitely and still ensure no single random failure will disable this trip Function.

In MODE 1 above the P-10 setpoint, the RCP Speed – Low trip must be OPERABLE. Below the P-10 setpoint, all reactor trips on loss of flow are automatically blocked since no power distributions are expected to occur that would cause a DNB concern at this low power level. Above the P-10 setpoint, the reactor trip on loss of flow in two or more RCS cold legs is automatically enabled.

13. Steam Generator Water Level – Low

The SG Water Level – Low trip Function ensures that protection is provided against a loss of heat sink. The SGs are the heat sink for the reactor. In order to act as a heat sink, the SGs must contain a minimum amount of water. A narrow range low level in any steam generator is indicative of a loss of heat sink for the reactor. The Trip Setpoint reflects the inclusion of both steady state and adverse environmental instrument uncertainties as the detectors provide primary protection for an event that results in a harsh environment. This Function also contributes to the coincidence logic for the ESFAS Function of opening the Passive Residual Heat Removal (PRHR) discharge valves.

The LCO requires four channels of SG Water Level – Low per SG to be OPERABLE in MODES 1 and 2. Four channels are provided to permit one channel in trip or bypass indefinitely and still ensure no single random failure will disable this trip Function.

In MODE 1 or 2, when the reactor requires a heat sink, the SG Water Level – Low trip must be OPERABLE. The normal source of water for the SGs is the Main Feedwater System (non-safety related). The Main Feedwater System is normally in operation in MODES 1 and 2. PRHR is the safety related backup heat sink for the reactor. During normal startups and shutdowns, the Main and Startup Feedwater Systems (non-safety related) can provide feedwater to maintain SG level. In MODE 3, 4, 5, or 6, the SG Water Level – Low Function does not have to be OPERABLE because the reactor is not operating or even critical.

AP1000 TECHNICAL REPORT REVIEW

Response to Request For Additional Information (RAI)

RAI Response Number: RAI-SRP16-CTSB-45
Revision: 0

Question:

TS 3.3.1 Reactor Trip System Instrumentation

For Table 3.3-1, function 12 (RCP speed - low), clarify the "required channels" column entry by adding the appropriate notation that would make Table 3.3-1 consistent with DCD Table 7.2-2, sheet 1, and would be consistent with descriptions of other required channels presented on Table 3.3.1-1.

Westinghouse Response:

DCD Table 7.2-2 Reactor Coolant Pump Underspeed Function specifies "4 (1/pump)" number of channels. Specification Table 3.3-1, Function 12 and the associated Bases are marked up to be consistent with Table 7.2-2.

Design Control Document (DCD) Revision:

The required DCD markups are attached to the RAI-SRP16-CTSB-44 response, which also affected the same Table 3.3-1 page.

PRA Revision:

None

Technical Report (TR) Revision:

None

AP1000 TECHNICAL REPORT REVIEW

Response to Request For Additional Information (RAI)

RAI Response Number: RAI-SRP16-CTSB-57
Revision: 0

Question:

TS 3.4.6, Pressurizer Safety Valves. Technical Report (TR) 74A, Revision 1.

Revise DCD Section 15.2.2 and Table 1.6-1 and/or TS bases B 3.4.6 to cite the same reference concerning overpressure protection.

In the TS bases B 3.4.6, the preliminary document listed as Reference 2, [WCAP-7769, "Topical Report on Overpressure Protection", October 1971.] is being replaced with a final document WCAP-16779, "AP1000 Overpressure Protection Report", April 2007, under TR 74A, Revision 1, however DCD Section 15.2.2 and Table 1.6-1 (Sheets 14 and 17) are not being revised to reflect the new referenced document.

This is required to ensure consistency between the TS bases and referenced information provided in the AP1000 DCD.

Westinghouse Response:

WCAP-16779 is a plant specific supplement to WCAP-7769 and not a replacement. The most appropriate solution is to add WCAP-16779 in both Section 15.2.2 and Table 1.6-1 rather than to replace WCAP-7769 with WCAP-16779. Westinghouse will revise the DCD to add WCAP-16779 in both Section 15.2.2 and Table 1.6-1.

Design Control Document (DCD) Revision:

DCD markups are shown below.

PRA Revision:

None

Technical Report (TR) Revision:

None

Table 1.6-1 (Sheet 14 of 20)

MATERIAL REFERENCED

DCD Section Number	Westinghouse Topical Report Number	Title
15.0	WCAP-7979-P-A (P) WCAP-8028-A	TWINKLE – A Multi-Dimensional Neutron Kinetics Computer Code, January 1975
	WCAP-10698-P-A (P) WCAP-10750-A	SGTR Analysis Methodology to Determine the Margin to Steam Generator Overfill, August 1987
	WCAP-14234 (P) WCAP-14235	LOFTRAN and LOFTTR2 AP600 Code Applicability Document, Revision 1, August 1997
	WCAP-15644-P (P) WCAP-15644-NP	AP1000 Code Applicability Report, Revision 2, March 2004
15.1	WCAP-7907-P-A (P) WCAP-7907-A WCAP-11397-P-A (P) WCAP-11397-A	LOFTRAN Code Description, April 1984 Revised Thermal Design Procedure, April 1989
	WCAP-9226 (P) WCAP-9227	Reactor Core Response to Excessive Secondary Steam Releases, January 1978
	WCAP-15644-P (P) WCAP-15644-NP	AP1000 Code Applicability Report, Revision 2, March 2004
	WCAP-7908-A	FACTRAN – A FORTRAN-IV Code for Thermal Transients in a UO_2 Fuel Rod, December 1989
15.2	WCAP-7769	Overpressure Protection for Westinghouse Pressurized Water Reactors, Revision 1, June 1972
	WCAP-16779-NP	Overpressure Protection Report for AP1000 Nuclear Power Plant
	WCAP-7907-P-A (P) WCAP-7907-A	LOFTRAN Code Description, April 1984
	WCAP-9230 (P) WCAP-9231	Report on the Consequences of a Postulated Main Feedline Rupture, January 1978
	WCAP-11397-P-A (P) WCAP-11397-A	Revised Thermal Design Procedure, April 1989
	WCAP-15644-P (P) WCAP-15644-NP	AP1000 Code Applicability Report, Revision 2, March 2004
	WCAP-7908-A	FACTRAN - A FORTRAN-IV Code for Thermal Transients in a

(P) Denotes Document is Proprietary

If a safety limit is approached, protection is provided by high pressurizer pressure, high pressurizer water level, and overtemperature ΔT trips. Voltage and frequency relays associated with the reactor coolant pump provide no additional safety function for this event. Following a complete loss of external electrical load, the maximum turbine overspeed is not expected to affect the voltage and frequency sensors. Any increased frequency to the reactor coolant pump motors results in a slightly increased flow rate and subsequent additional margin to safety limits. For postulated loss of load and subsequent turbine-generator overspeed, an overfrequency condition is not seen by the protection and safety monitoring system equipment or other safety-related loads. Safety-related loads and the protection and safety monitoring system equipment are supplied from the 120-Vac instrument power supply system, which in turn is supplied from the inverters. The inverters are supplied from a dc bus energized from batteries or by a regulated ac voltage.

If the steam dump valves fail to open following a large loss of load, the steam generator safety valves may lift and the reactor may be tripped by the high pressurizer pressure signal, the high pressurizer water level signal, or the overtemperature ΔT signal. This would cause steam generator shell side pressure and reactor coolant temperature to increase rapidly. However, the pressurizer safety valves and steam generator safety valves are sized to protect the reactor coolant system and steam generator against overpressure for load losses, without assuming the operation of the turbine bypass system, pressurizer spray, or automatic rod cluster control assembly control.

The steam generator safety valve capacity is sized to remove the steam flow at the nuclear steam supply system thermal rating from the steam generator, without exceeding 110 percent of the steam system design pressure. The pressurizer safety valve capacity is sized to accommodate a complete loss of heat sink, with the plant initially operating at the maximum turbine load, along with operation of the steam generator safety valves. The pressurizer safety valves can then relieve sufficient steam to maintain the reactor coolant system pressure within 110 percent of the reactor coolant system design pressure.

A discussion of overpressure protection can be found in WCAP-7769, Revision 1 (Reference 1) and WCAP-16779 (Reference 8).

A loss-of-external-load event is classified as a Condition II event, fault of moderate frequency.

A loss-of-external-load event results in a plant transient that is bounded by the turbine trip event analyzed in subsection 15.2.3. Therefore, a detailed transient analysis is not presented for the loss-of-external-load event.

The primary side transient is caused by a decrease in heat transfer capability, from primary to secondary, due to a rapid termination of steam flow to the turbine, accompanied by an automatic reduction of feedwater flow (should feedwater flow not be reduced, a larger heat sink is available and the transient is less severe). Reduction of steam flow to the turbine following a loss-of-external load event occurs due to automatic fast closure of the turbine control valves. Following a turbine trip event, termination of steam flow occurs via turbine stop valve closure, which occurs in approximately 0.15 seconds. The transient in primary pressure, temperature, and water volume is less severe for the loss-of-external-load event than for the turbine trip due to a slightly slower loss of heat transfer capability.

15.2.9 Combined License Information

This section has no requirement for additional information to be provided in support of the Combined License application.

15.2.10 References

1. Cooper, L., Miselis, V., and Starek, R. M., "Overpressure Protection for Westinghouse Pressurized Water Reactors," WCAP-7769, Revision 1, June 1972. (Also letter NS-CE-622, C. Eicheldinger (Westinghouse) to D. B. Vassallo (NRC), additional information on WCAP-7769, Revision 1, April 16, 1975).

2. Burnett, T. W. T., et al., "LOFTRAN Code Description," WCAP-7907-P-A (Proprietary) and WCAP-7907-A (Nonproprietary), April 1984.

3. "American National Standard for Decay Heat Power in Light Water Reactors," ANSI/ANS-5.1-1979, August 1979.

4. Lang, G. E., and Cunningham, J. P., "Report on the Consequences of a Postulated Main Feedline Rupture," WCAP-9230 (Proprietary) and WCAP-9231 (Nonproprietary), January 1978.

5. Friedland, A. J., and Ray, S., "Revised Thermal Design Procedure," WCAP-11397-P-A (Proprietary) and WCAP-11397-A (Nonproprietary), April 1989.

6. "AP1000 Code Applicability Report," WCAP-15644-P (Proprietary) and WCAP-15644-NP (Nonproprietary), Revision 2, March 2004.

7. Hargrove, H. G., "FACTRAN – A FORTRAN-IV Code for Thermal Transients in a UO_2 Fuel Rod," WCAP-7908-A, December 1989.

8. Matthys, C., "Overpressure Protection Report for AP1000 Nuclear Power Plant," WCAP-16779-NP, April 2007.

AP1000 TECHNICAL REPORT REVIEW

Response to Request For Additional Information (RAI)

RAI Response Number: RAI-SRP16-CTSB-59
Revision: 0

Question:

TS 3.9.5 Containment Penetrations

Provide an explanation of (or provide a reference in the Bases) where the surveillance requirement, SR 3.9.5.3 requires that one VFS system can maintain a negative pressure of less than or equal to -0.125 inches water gauge, relative to outside atmospheric pressure in the fuel handling building, and if this is the slight negative pressure mentioned in Section 9.4.7.2.3, Abnormal Operation of the VFS.

The VAS is required to perform this function when the normal ventilation (VBS) isolates due to high radiation. Section 9.4.7.2.3, Abnormal Operation for the VFS does not mention this specific pressure. It only notes that a slight negative pressure is maintained for this configuration. The value of -0.125 inches water gauge pressure should be evaluated to verify that this is the "slight negative pressure" intended as described in Section 9.4.7.2.3. The DCD, Table 3.9-17 mentions this pressure for testing. (This RAI applies to LCO 3.9.6, SR 3.9.6.3).

Also applicable to TS 3.9.6.

Westinghouse Response:

The following answer is based on the first sentence of the second paragraph being "The VFS is required to perform this function when the normal ventilation (VAS) isolates due to high radiation." The VAS and VBS are not interconnected in the AP1000.

The negative pressure of -0.125 inches of water is the slight negative pressure from section 9.4.7.2.3. This pressure was chosen based on ASHRAE Applications, which recommends at least 0.05 to 0.06 inches of water across boundaries when exfiltration or infiltration is minimized. Conservatively, Westinghouse chose a higher pressure difference of 0.125 inches of water.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

Technical Report (TR) Revision:

None

AP1000 TECHNICAL REPORT REVIEW

Response to Request For Additional Information (RAI)

RAI Response Number: RAI-SRP16-CTSB-61
Revision: 0

Question:

TS Bases 3.6.1 Containment

Provide the specific sections of Chapter 15 that discuss the specific accident in the APPLICABLE SAFETY ANALYSIS section on bases page B 3.6.1-4. Note: This RAI also applies to LCO's 3.6.2 and 3.6.3 pages B 3.6.2-7 and B 3.6.3-9.

For LCO 3.6.1, on Bases page B 3.6.1-4, REFERENCES, Ref. 2 is stated as Chapter 15, "Accident Analysis". In the body of the Bases on page 3.6.1-2, APPLICABLE SAFETY ANALYSIS section, second paragraph, the LOCA and rod ejection accidents are mentioned. The specific sections of Chapter 15 are 15.6.5 and 15.4.8. Note: Several other listings of REFERENCES at the end of the Bases section, list specific sections of Chapter 15 that apply.

Westinghouse Response:

The level of detail provided by the 3.6.1, 3.6.2 and 3.6.3 Bases references to Chapter 15 is consistent with the STS. For example, the STS 3.6.1 and 3.6.3 Bases discussions of applicable accidents are accompanied by reference to Chapter 15. The specific Chapter 15 sections are not listed in the STS.

No Bases changes are required.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

Technical Report (TR) Revision:

None

AP1000 TECHNICAL REPORT REVIEW

Response to Request For Additional Information (RAI)

RAI Response Number: RAI-SRP16-CTSB-72
Revision: 0

Question:

1. Provide a reference to Reference 3, on Bases page B 3.6.9-4 in the appropriate section of the Bases.

2. LCO 3.6.9, Bases page B 3.6.9-4, Reference 3 is not quoted in the body of the Bases. (See also LCO 3.7.2, 3.7.6 and 3.9.1 Bases)

Westinghouse Response:

The information in reference 3 is not discussed in the Bases and is not needed. The attached markup shows the deletion of reference 3.

Design Control Document (DCD) Revision:

A markup of the affected DCD Revision 17 page is attached.

PRA Revision:

None

Technical Report (TR) Revision:

None

BASES

SURVEILLANCE REQUIREMENTS (continued)

before the required pH is achieved. This would ensure compliance with the Standard Review Plan requirement of a pH ≥ 7.0 by the onset of recirculation after a LOCA.

REFERENCES

1. Section 6.3.2.1.4, "Containment pH Control."
2. Section 6.3.2.2.4, "pH Adjustment Baskets."
3. ~~Section 15.6.5.3.1, "Identification of Cause and Accident Description."~~

AP1000 TECHNICAL REPORT REVIEW

Response to Request For Additional Information (RAI)

RAI Response Number: RAI-SRP16-CTSB-73
Revision: 0

Question:

1. Bases 3.7.11
Delete the revision number (DCD Rev. 15) for Reference 1 on bases page 3.7.11-3.

2. Bases 3.7.12
Delete the revision number (DCD Rev. 15) for Reference 3 on bases page 3.7.12-4 and correct the typographical error for Reference 3 on page B 3.7.12-4. (Section 9.12 to Section 9.1.2)

The revision numbers are not necessary as the most recent revision is the one implied.

3. In Bases section 3.7.2, provide a reference to Reference 4 in the appropriate section of the Bases.

4. In Bases section 3.7.6, provide references to References 2 and 5 in appropriate sections of the Bases.

Westinghouse Response:

1. & 2. The identified Bases 3.7.11, Reference 1 and Bases 3.7.12, Reference 3 changes are required. The attached markup shows the deletion of the revision number as well as changes for consistency with other Tech Spec references to DCD sections.

3. The Tech Spec 3.7.2 Bases do not discuss DCD Section 10.2; therefore, reference 4 is not needed. The attached markup shows the deletion of reference 4.

4. The Bases 3.7.6 Background section discusses the VBS. The attached markup shows the addition of reference 2 in the first paragraph.

   The Tech Spec 3.7.6 Bases do not discuss a chemical release; therefore, reference 5 is not needed. The attached markup shows the deletion of reference 5.

Design Control Document (DCD) Revision:

Markups of the affected DCD Revision 17 pages are attached.

PRA Revision:

None

Technical Report (TR) Revision:

None

BASES

SURVEILLANCE REQUIREMENTS	SR 3.7.11.1 This SR verifies that the concentration of boron in the fuel storage pool is within the required limit. As long as this SR is met, the analyzed accidents are fully addressed. The 7 day Frequency is appropriate because no major replenishment of pool water is expected to take place over such a short period of time.
REFERENCES	1. ~~AP1000 Design Control Document, Rev. 15~~DCD, Sections 9.1.2, "Spent Fuel Storage" and 15.7.4, "Fuel Handling Accident." 2. Double contingency principle of ANSI N16.1-1975, as specified in the April 14, 1978 NRC letter (Section 1.2) and implied in the proposed revision to Regulatory Guide 1.13 (Section 1.4, Appendix A). 3. APP-GW-GLR-029, "AP1000 Spent Fuel Storage Racks Critically Analysis," June 2006.

BASES

ACTIONS (continued)

A.1

The LCO is not met if spent fuel assemblies stored in Region 2 "All Cell," "1-out-of-4 5.0 weight-percent fresh" or interface spent fuel assembly storage locations do not meet the applicable initial enrichment, burnup and decay time limits in accordance with Figure 3.7.12-1 or 3.7.12-2.

Additionally, LCO is not met if fuel, required to be stored in the New Fuel location of the "1-out-of-4 5.0 weight-percent fresh" storage configuration, is misplaced. When the LCO is not met, action must be initiated immediately to make the necessary fuel assembly movement(s) in Region 2 to bring the storage configuration into compliance with Figures 3.7.12-1 and 3.7.12-2 or to move fuel to Region 1 or the defective fuel cells.

SURVEILLANCE REQUIREMENTS

SR 3.7.12.1

This SR verifies by administrative means that the initial enrichment, burnup and decay time of the fuel assembly is in accordance with Figure 3.7.12-1 or 3.7.12-2 as applicable for "All Cell," "1-out-of-4 5.0 weight-percent fresh" and interface spent fuel assembly storage locations. Fuel stored in Region 2 that does not meet the Figure 3.7.12-1 or 3.7.12-2 limits shall be stored in Figure 4.3-1 "1-out-of-4 5.0 weight-percent fresh" New Fuel location.

REFERENCES

1. Double contingency principle ANSI N16.1-1975, as specified in the April 14, 1978 NRC letter (Section 1.2) and implied in the proposed revision to Regulatory Guide 1.13 (Section 1.4, Appendix A).
2. APP-GW-GLR-029, "AP1000 Spent Fuel Storage Racks Criticality Analysis," June 2006.
3. ~~AP1000 Design Control Document, Rev. 15~~DCD, Sections 9.1.2, "Spent Fuel Storage" and 15.7.4, "Fuel Handling Accident."

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.2.2

This SR verifies that the turbine stop, turbine control, turbine bypass, and moisture separator reheater 2nd stage steam isolation valves' closure time is ≤ 5.0 seconds, on an actual or simulated actuation signal. These alternate downstream isolation valves must meet the MSIV isolation time assumed in the accident and containment analyses. This Surveillance is normally performed upon returning the unit to operation following a refueling outage. The alternate downstream valves should not be tested at power, since even a part stroke exercise increases the risk of a valve closure when the unit is generating power. As the alternate downstream valves are not tested at power, they are exempt from the ASME OM Code (Ref. 7) requirements during operation in MODE 1 or 2.

The Frequency is in accordance with the Inservice Testing Program.

This test is conducted in MODE 3 with the unit at operating temperature and pressure. This SR is modified by a Note that allows entry into and operation in MODE 3 prior to performing the SR. This allows a delay of testing until MODE 3, to establish conditions consistent with those under which the acceptance criterion was generated.

REFERENCES

1. Section 10.3, "Main Steam System."
2. Section 6.2.1, "Containment Functional Design."
3. Section 15.1, "Increase in Heat Removal by Secondary System."
4. ~~Section 10.2, "Turbine Generator."~~Not used.
5. NUREG-138, Issue 1, "Staff Discussion of Fifteen Technical Issues Listed in Attachment to November 3, 1976 Memorandum from Director NRR to NRR Staff."
6. Section 10.4, "Other Features of Steam and Power Conversion Systems."
7. ASME OM Code, "Code for Operation and Maintenance of Nuclear Power Plants."

B 3.7 PLANT SYSTEMS

B 3.7.6 Main Control Room Emergency Habitability System (VES)

BASES

BACKGROUND

The Main Control Room Habitability System (VES) provides a protected environment from which operators can control the plant following an uncontrolled release of radioactivity. The system is designed to operate following a Design Basis Accident (DBA) which requires protection from the release of radioactivity. In these events, the Nuclear Island Non-Radioactive Ventilation System (VBS) would continue to function if AC power is available (Ref. 2). If AC power is lost or a High-2 main control room (MCR) radiation signal is received, the VES is actuated. The major functions of the VES are: 1) to provide forced ventilation to deliver an adequate supply of breathable air (Ref. 4) for the MCR occupants; 2) to provide forced ventilation to maintain the MCR at a 1/8 inch water gauge positive pressure with respect to the surrounding areas; and 3) to limit the temperature increase of the MCR equipment and facilities that must remain functional during an accident, via the heat absorption of passive heat sinks.

The VES consists of compressed air storage tanks, two air delivery flow paths, associated valves, piping, and instrumentation. The tanks contain enough breathable air to supply the required air flow to the MCR for at least 72 hours. The VES system is designed to maintain CO_2 concentration less than 0.5% for up to 11 MCR occupants.

Sufficient thermal mass exists in the surrounding concrete structure (including walls, ceiling and floors) to absorb the heat generated inside the MCR, which is initially at or below 75°F. Heat sources inside the MCR include operator workstations, emergency lighting and occupants. Sufficient insulation is provided surrounding the MCR pressure boundary to preserve the minimum required thermal capacity of the heat sink. The insulation also limits the heat gain from the adjoining areas following the loss of VBS cooling.

In the unlikely event that power to the VBS is unavailable for more than 72 hours, MCR envelope habitability is maintained by operating one of the two MCR ancillary fans to supply outside air to the MCR envelope.

The compressed air storage tanks are initially pressurized to 3400 psig. During operation of the VES, a self contained pressure regulating valve maintains a constant downstream pressure regardless of the upstream pressure. An orifice downstream of the regulating valve is used to control the air flow rate into the MCR. The MCR is maintained at a 1/8 inch water gauge positive pressure to minimize the infiltration of airborne contaminants from the surrounding areas.

BASES

SURVEILLANCE REQUIREMENTS (continued)

delivery flow path is tested on an alternating basis. The system performance test demonstrates that the MCR pressurization assumed in dose analysis is maintained.

REFERENCES

1. Section 6.4, "Main Control Room Habitability Systems."
2. Section 9.4.1, "Nuclear Island Non-Radioactive Ventilation System."
3. SECY-95-132, "Policy and Technical Issues Associated With The Regulatory Treatment of Non-Safety Systems (RTNSS) In Passive Plant Designs (SECY-94-084)," May 22, 1995.
4. ASHRAE Standard 62-1989, "Ventilation for Acceptable Indoor Air Quality."
5. ~~Regulatory Guide 1.78, "Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release," Revision 1, December 2001.~~

AP1000 TECHNICAL REPORT REVIEW

Response to Request For Additional Information (RAI)

RAI Response Number: RAI-SRP16-CTSB-74
Revision: 0

Question:

Correct the following editorial errors:

1. (Page 3.9.5-2) In SR 3.9.5.2, change from "LCO 3.9.4.d.1" to "LCO 3.9.5.d.1"

2. In Bases section 3.9.1, provide references to References 1 and 2 in appropriate sections of the Bases.

Westinghouse Response:

1. The LCO reference in SR 3.9.5.2 was corrected to "LCO 3.9.5.d.1" in DCD Revision 17.

2. The REFERENCES section of the TS Bases provides a list of documents where more detailed information pertinent to the Specification can be located. There is no requirement to cite each reference. In the STS and AP1000 Tech Specs, references are usually cited; however, some references are only listed in the Reference section as a source of additional information related to the requirements.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

Technical Report (TR) Revision:

None