"Collective Significance Chart"

USQ	TITLE	INTERIM CONTROLS	COMMENT	
TRA-USQ-SE-2003-145	ATR Firewater Supply System Modeling Issues	Interim resolution EDF-5614 R1	Concern is the delivery of firewater to the core. It very unlikely that this concern would exacerbate other PISA/USQs	x
~~TRA-USQ-2004-176 (ATR-2004-004)~~	~~Reactor Shutdown System Reactor Vessel Level Limiting Control Setting~~	~~Do not change current in-plant settings (JCC-02-05)~~		x
TRA-USQ-2004-214 R1 (ATR-2004-004)	ATR Seismic Safety Basis Deterioration	Interim resolution; EDF-5614 R1	Concern is the delivery of firewater to the core. It very unlikely that this concern would exacerbate other PISA/USQs	x
TRA-USQ-2004-385 R1 (ATR-2004-004) ESS submitted to DOE 3/27/10; resubmitted 6/14/10	ATR Surge Tank Level Instrument Limiting Control Settings	Surge tank level 50%+/-5%	Concern is uncertainty in surge tank level; control decreases the allow band for reactor operation. Safety analysis assumes worse case level. Decreasing the allow band has no effect on the safety analyses. It is very unlikely that this concern could have any effect on other PISA/USQs.	x
TRA-USQ-2004-396 (ATR-2004-004)	ATR Seismic Primary Coolant System Break Size and TRA Support Building Vulnerabilities	Interim resolution; EDF-4334 (incorporated into EDF-5614 R1)	Concern is the delivery of firewater to the core. It very unlikely that this concern would exacerbate other PISA/USQs	x
TRA-USQ-2004-413 (ATR-2004-004)	ATR Seismic Primary Coolant System Break Size Contribution from Letdown Valves	Interim resolution; EDF-5614 R1	Concern is the delivery of firewater to the core. It very unlikely that this concern would exacerbate other PISA/USQs	x
RTC-USQ-2005-248 (ATR-2004-004)	GAP-005-05 – Radiation Monitoring and Seal System Setpoint Derivation	Do not change current in-plant settings (JCC-02-05)	Concern is the basis of the RMSS RMS-1 setpoint. Any accident that relies on the confinement to mitigate the dose consequence would be affected by this concern. With the exception of the analysis used to set the setpoint, all other analysis, however, assume RMS-1 trip at the initiation of the accident. The accident analyses will not be affected by any change in the RMS-1 setpoint.	x
RTC-USQ-2006-376 (ATR-2006-0006)	Discrepancy with high inlet pressure setpoint derivation	No interim controls; maintain current plant settings will increase margin [385 psig] (JCC-02-05)	Concern is the accounting of the inputs to the over-pressure analysis. Current PPS setpoints will maintain reactor operation within the safety analyses. Using current in plant setting has no effect on other	x

USQ	TITLE	INTERIM CONTROLS	COMMENT	
			USQs/PISAs.	
RTC-USQ-2006-577 (ATR-2006-0015)	Discrepancies in the Analysis of Rupture of a Heat Exchanger Tube and the Bases for the Secondary Coolant System (SCS) Activity Level	Enter AOP 2.4 if SCS activity is 100 cpm above background	Concern is basis for identifying PCS to SCS boundary fault and when to Enter the AOP. PCS pressure boundary faults are analyzed in the accident analyses. The accident analyses assume the worst fault location. In addition, these LOCA analyses did not rely upon entering this AOP. It is very unlikely that this concern would exacerbate other PISA/USQs.	x
RTC-USQ-2006-655 (ATR-2006-0024)	Inconsistencies in the Maximum Hypothetical Accident Analysis, GAP-025-06	Interim Controls reduce effective plate power limit to 87% of TSR	Concern is the basis of the dose consequence analyses. Since the MHA is used to bound the consequence of the analyzed accidents, this concern could affect the consequences of the accidents. The possible effect, however, is only analytical and would not exacerbate any accident sequence. The interim control reduces the consequence of all accidents initiated at full allowable power.	x
RTC-USQ-2006-656 (ATR-2006-0023)	Section 15.12 Maximum Hypothetical Accident Analysis Used to Bound Canal Accidents, GAP-026-06	No Interim Control	Concern is the basis for the dose consequence of a severe canal accident. Although this concern could affect the consequences of the canal accident, the effect is only analytical and would not exacerbate any accident sequence.	
RTC-USQ-2007-726	Evaluation of Confinement Pressure Transient Capability (GAP-013-07)	Interim controls to perform building leak rate test	Any accident that relies on the confinement to mitigate the dose consequence would be affected by this concern. Interim controls ensure confinement is returned to its integrity as assumed in the accident analyses, prior entering operational modes that require confinement.	x
RTC-USQ-2008-137	Response Time and Safety Rod Drop Time Modeling and Requirements Discrepancy for NI Scram Function	Interim controls for response time	Any accident that relies on the RSS for mitigation would be affected by this concern. Interim controls ensure power operations are enveloped by the accident analyses.	x
RTC-USQ-2008-451	Canal Bulkhead Seal Leakage	TEV-284 Rev. 1; dual independent seals or two	Concern is leakage of a damaged canal	x

USQ	TITLE	INTERIM CONTROLS	COMMENT	
(NE-ID-BEA-ATR-2008-0011)	With Air-pillow Seal Deflated	bulkheads	isolation bulkhead seal. Draining the irradiated fuel section of the canal is considered beyond design basis. Interim controls ensure this accident analysis assumption.	
RTC-USQ-2008-553 (NE-ID-BEA-ATR-2008-0018) ESS submitted to DOE 3/18/10	Error in RELAP5 Reactor Kinetics Model	No interim controls	Concern is an error in the RELAP5 kinetics model. Any analyses that used RELAP5 kinetic model could be affected. The effect, however, is only analytical and would not exacerbate any accident sequence.	N A
RTC-USQ-2008-622 R1	Essential Test Requirements for the Manual RSS Functional Test Procedure (Manual SATS)	No interim controls	Concern is SAR-153 did not adequately provide the specific requirements to manually provide the necessary RSS functional testing. Although any accident that relies on the RSS for mitigation could be affected by this concern. Although not implemented into the safety basis, i.e., PISA/USQ still listed in LST-100, the conditions of PISA are being corrected.	N A
ATR Complex-USQ-2009-208	Pump Motor Room Flooding Following a Cask Drop Onto the Canal Parapet	No interim controls	Concern is an initiating event of a different type for the CLOFA within the first 30 minutes of initiation. The sequence, however, remains beyond design basis and consistent with the accident analysis assumptions.	N A
ATR Complex-USQ-2009-320	Emergency Coolant Pump Discrepancies in SAR-153	670-M-10 must be running ECP	Concern is the power operations ECP alignment. Interim controls ensure ECP alignment is consistent with the accident analysis assumptions.	x
ATR Complex-USQ-2009-631	ATR Primary Coolant System pH Control	5.0 ≤PCS pH ≤ 5.4	Concern is the lower limit of the PCS pH. The consequence of low pH could degrade the thermal conductivity of the aluminum oxide. This concern could affect many of the analyzed accidents. Interim controls, however, protect the accident analysis assumptions.	x
ATR Complex-USQ-2009-714	Evacuation System Seismic Requirements not Supported	Condition 4 seismic safety basis analyses	Evaluations demonstrate that the radiological exposures to ATR Complex	

USQ	TITLE	INTERIM CONTROLS	COMMENT
ESS submitted to DOE 6/14/10	by Seismic Design of Buildings		collocated personnel (not considered directly at the location of the accident) continue to be met without reliance on the evacuation system sirens to initiate a complex-wide evacuation (See PISA). However, the Condition 4 PPC for reactor accidents allow significant melting and fission product release from the fuel. Hence, the PPC do not ensure on a continuing basis that the calculated consequences of postulated events (i.e., future analyses) do not require evacuation. Therefore, an interim control is required to ensure that Condition 4 events do not result in Significant fuel damage or that an accident-specific radiological analysis be performed (without reliance on the evacuation system sirens) if significant fuel damage is postulated.
ATR Complex-USQ-2010-497	Seismic-Induced Experiment Loop Leakage	No interim control or immediate action required	Consequence to ATR Complex worker could be as high as 10 rem. Condition 4 PCS LOCA may also have occurred. LOCA water management will be necessary. Potential operator exposure should be a concern.
ATR Complex-USQ-2010-607	Application of Revised Surge Tank Level Analyses to Interim Operating Controls	EFIS firewater pump Response Time ≤ 60 s.	Adds reduced EFIS Response Time to TRA-USQ-2004-385 R1

INTERIM CONTROLS

TSR-186, Section 3.1.1:

Table 3.1.1-1 response times are reduced by 2 ms to account for uncertainty in the RCCC release time measurement. (response time)

TSR-186. Section 3.2.1.2:

The 688-M-1 and 688-M-2 firewater pumps with automatic start setpoint ≥ 55 psig, based on an accuracy of ± 5 psi, and start time of ≤ 1 min must be operable when irradiated fuel is in the reactor vessel. These firewater pumps must be seismically qualified and not rely on commercial power. (EDF-5614 R1)

Comment [11]: ATR Complex-USQ-2010-607 reduces Response Time

A minimum water level of ≥ 227 in. in the TRA-781 tank (indicated from LI-10-100) must be maintained with irradiated fuel in the reactor vessel. The level indication instrumentation for the TRA-781 tank (LI-10-100) must be calibrated to support operability of the TRA-781 tank with its required EFIS supply inventory. (EDF-5614 R1)

The crossover line check valves must be operable with irradiated fuel in the reactor vessel. Operability shall be supported by functional testing, that includes verification that the valves fully open and seat on demand, and that the back leakage through each individual valve does not exceed 100 gpm. (EDF-5614 R1)

The overhead raw water storage tank (TRA-718) must have an inventory of ≥ 58,000 gal during power operation, low power operation, and pressurized standby following power operation until the fuel cooling time is met. (EDF-5614 R1)

One seismically-qualified deep-well pump with diesel emergency-power source must be operable with irradiated fuel in the reactor vessel. (EDF-5614 R1)

Two deep-wells must be operable with irradiated fuel in the reactor vessel. The operable emergency-powered deep-well required above qualifies as one of the two required operable deep-wells. (EDF-5614 R1)

TSR-186, Section 3.3.3:

M-10 is the running ECP during power operations and for 30 minutes following power operations. (M-10 must be running ECP)

TSR-186 Section 3.3.4:

To ensure the consequences of a LOCA with one primary coolant pump continuing to run do not exceed analyzed conditions, the minimum procedural operating range must be ensured by treating this limit as an interim TSR level control. Therefore, the existing TSR [LCO] 3.3.4 (Surge Tank Level) applicabilities, action statements, completion times, and surveillance requirements for the surge tank water level, associated with a water level of ≤ 28%, should be invoked for a water level of ≤ 45% of instrument span. The remaining TSR 3.3.4 applicabilities, actions, etc., associated with water levels ≥ 73%, ≥ 10%, < 10%, and with instrument operability, are not affected and remain valid. (Surge tank level 50%±5%)

TSR-186, Section 3.3.5:

Primary coolant pH shall be ≥ 5.0 but ≤ 5.4 during power operation. (5.0 ≤PCS pH ≤ 5.4)

TSR-186, Section 3.3.6:

Reactor primary coolant system total leakage at system pressure (identified and unidentified) may not exceed 50 gpm. (EDF-5614 R1)

TSR-186, Section 3.5.5:

A required water level of ≥: 227 in. in the TRA-781 tank (indicated from LI-1 0-1 00) must be maintained with irradiated fuel in the ATR canal when the 688-M-1 or 688- M-2 firewater pump is relied on to provide canal makeup. (EDF-5614 R1)

A required water level ≥ 19 ft 1 in. in the TRA-719A, Band C tanks must be maintained with irradiated fuel in the A TR canal when the 633-1 firewater pump is relied on to provide canal makeup. (EDF-5614 R1)

A single firewater pump (688-M-1, 688-M-2 or 633-1) must be operable and able to start by operator action or automatic start mode with irradiated fuel in the ATR canal. (EDF-5614 R1)

Isolation bulkheads with dual independent inflated seals OR two isolation bulkheads each with an inflated seal shall be used to isolate irradiated fuel storage from heavy load handling. (TEV-284 Rev. 1)

TSR-186. Section 5.3.2:

The ATR required shift complement must support the ability of the operations staff to complete operator actions to isolate the ATR firewater loop and establish long-term firewater supply as discussed in the SAR Commitment below. (EDF-5614 R1)

TRS-186, Section 3.6.1:

Until the seismic LOCA with an aged reflector is evaluated, reactor operation is not allowed when <2σ from Beryllium cracking, or if cracks in the Beryllium are observed. (<2σ Be cracking)

Operations of the ATR with the 13% reduced EPP will result in the worst case Condition 4 accident meeting the Condition 2 plant protection criteria, and consequently, will eliminate any credible possibility of fuel damage from the worst case design basis LOCA. Other postulated design basis ATR fuel damage events occur with an intact primary PCS. (reduce effective plate power limit to 87% of TSR)

TSR Section 3.7.1:

Table 3.7.1-1 release times are reduced by 2 ms to account for uncertainty in the RCCC release time measurement. The drop times are reduced by 5 ms for pressurized operation and 8 ms for depressurized operation to account for uncertainty in the RCCC release time measurement (2 ms) and the discrepancy in the release time requirement (3 ms pressurized, 6 ms depressurized). (response time)

TSR-186, Section 3.8.1:

Any failure of HVS-1 which result in a building under-pressure will require the following: If the reactor is in an outage, a building leak-rate test is to be performed prior to entering confinement applicability. If the reactor is in operation, walk-down of the accessible areas to inspect for damaged confinement will be performed; perform a building leak-rate test prior to the next scheduled reactor startup. (perform building leak rate test)

SAR Commitments:

The instrumentation that controls the automatic start and stop features on the deep-well pumps must be calibrated and have nominal setpoints ≥ 22 ft for pump start and ≤ 26 ft for the pump shutoff for the deep-well pump selected to start first when the TRA-719 tanks are relied on to provide canal makeup. (EDF-5614 R1)

At least one of the 633-1, 619-10, and 619-12 pumps must be available with automatic start capability and supply inventory of ≥ 19 ft 1 in. in the TRA-719A, B and C tanks during weekly Fire Protection pump testing of the 688-M-1 and 688-M-2 pumps when EFIS Supply OPERABILITY is required by the TSR. (EDF-5614 R1)

The instrumentation that measures water level in the overhead tank (TRA-718) must be calibrated and have a nominal low level alarm setpoint of ≥ 24.5 ft. (EDF-5614 R1)

The ability for operators to ensure that only one 688 firewater pump is running within 50 minutes of the seismic scram, followed by aligning and starting a deep-well pump in order to establish long-term firewater supply within an additional 75 minutes must be supported by facility procedures and initiated from the EPNs. A seismically robust flow path providing EFIS to the reactor vessel is also required. In addition, the running firewater pump must not be shutdown prior to 2 hours following the seismic scram. (EDF-5614 R1)

TRA-671 firewater piping must be isolated (or seismic issues remedied) in order to prevent unacceptable firewater supply losses during EFIS operability or TRA-781 inventory requirements applicability (TSR 3.2.1.2). (EDF-5614 R1)

A program to track safety related equipment failure experience, particularly TSR equipment, and to track the duration of time spent in action statements will be developed and implemented. (EDF-5614 R1)

Do not change current in-plant settings: (JCC-02-05)

Establish an interim control for the SCS activity of no more than 100 cpm above background. With the radioactivity greater than 100 cpm above background perform a 10 minute background per DOP-6.1.1 (ATR Canberra Well Counting System), take a second, and if necessary a third sample, within 30 minute

intervals. If two samples show radioactivity greater than 100 cpm above background, boundary fault leakage is possible. Enter AOP 2.4 - Primary To Secondary Leak. (SCS activity ≤ 100 cpm)

For Condition 4 postulated accidents, initiated by a seismic events, new or revised accident analyses implemented into the safety basis shall demonstrate that potential fuel damage is bounded by existing safety basis analyses.(Condition 4 seismic analyses)