PMSTPCOL PEmails

From: Elton, Loree [leelton@STPEGS.COM]
Sent: Monday, February 22, 2010 4:33 PM
To: Muniz, Adrian; Dyer, Linda; Wunder, George; Tonacci, Mark; Eudy, Michael; Kallan, Paul; Plisco, Loren; Anand, Raj; Foster, Rocky; Joseph, Stacy; Govan, Tekia; Tai, Tom
Subject: Transmittal of Letter U7-C-STP-NRC-100044
Attachments: U7-C-STP-NRC-100044.pdf

Please find attached a courtesy copy of letter number U7-C-STP-NRC-100044, which contains responses to NRC staff questions included in Request for Additional Information (RAI) letter numbers 232, 239, and 314 related to Combined License Application (COLA) Part 2, Tier 2, Section 4.4 and Appendices 6C and 15A.

The official version of this correspondence will be placed in today's mail. Please call Jim Tomkins at 805-215-6129 if you have any questions concerning this letter.

Thank you,

Loree Elton
Licensing, STP 3 & 4
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Hearing Identifier: SouthTexas34Public_EX
Email Number: 2027

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Subject: Transmittal of Letter U7-C-STP-NRC-100044
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February 22, 2010
U7-C-STP-NRC-100044

U. S. Nuclear Regulatory Commission
Attention: Document Control Desk
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Rockville, MD 20852-2738

South Texas Project
Units 3 and 4
Docket Nos. 52-012 and 52-013
Response to Requests for Additional Information

Attached are responses to NRC staff questions included in Request for Additional Information (RAI) letter numbers 232, 239, and 314 related to Combined License Application (COLA) Part 2, Tier 2, Section 4.4 and Appendices 6C and 15A. This completes the response to these letters. The attachments address the responses RAI questions listed below:

RAI 04.04-3 Supplement
RAI 06.02.02-11 Supplement 2
RAI 15.08-3

The COLA changes provided in these responses will be incorporated in the next routine revision of the COLA following NRC acceptance of the RAI response.

There are no commitments in this letter.

If you have any questions regarding these responses, please contact me at (361) 972-7136, or Bill Mookhoek at (361) 972-7274.

I declare under penalty of perjury that the foregoing is true and correct.

Executed on 2/22/10

Scott Head
Manager, Regulatory Affairs
South Texas Project Units 3 & 4

jet

Attachments:

1. RAI 04.04-3 Supplement
2. RAI 06.02.02-11 Supplement 2
3. RAI 15.08-3

cc: w/o attachment except*
(paper copy)

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RAI 04.04-3 Supplement:

<u>QUESTION</u>:

In response to RAI 06.02.02-2, STPNOC in their letter dated Sept 28, 2009 (U7-C-STP-NRC-090141) agreed for a COL license condition to submit an evaluation as part of the license amendment confirming that the fuel for the initial fuel load satisfies the downstream effects of containment debris on the reactor fuel. The acceptance criteria specified in the response are not sufficient.

a) Provide verifiable criteria for the fuel testing. Revise FSAR Section 4.4 to include the details of the acceptance criteria.

b) Confirm that the protective coatings debris characteristics for fuel assembly tests will be consistent with the NRC guidance for operating PWRs (NRC Staff Review Guidance Regarding Generic Letter 2004-02 Closure in the Area of Coatings Evaluation, March 2008, ADAMS Accession No. ML080230462.) For example, if a filtering bed is possible on fuel assembly openings, then fine particles representing coatings debris would be included in the debris load. In addition, show how the test requirements will be included in the application (e.g., license condition wording).

<u>RESPONSE</u>:

This supplemental response to RAI 04.04-3 provides the missing bracketed information as well as the five figures that were referenced in the original RAI response provided in STPNOC letter U7-C-STP-NRC-100015, dated January 18, 2010. The response below replaces the original response in its entirety.

a) The acceptance criteria for the downstream fuel effects testing will ensure that the pressure drop across the fuel assembly inlet is less than a value determined through analysis to provide adequate long term core cooling. This acceptance criterion is a pressure drop across the fuel assembly inlet corrected for flow rate as discussed in the RAI response below.

Appendix 6C and Section 4.4 will be revised as shown in the following COLA markup to describe the details of the acceptance criteria.

b) The protective coatings that will be used in the downstream fuel effects test will be consistent with NRC guidance in the Utility Resolution Guideline (URG), that is, 85 lbs of qualified coatings. (See Response to RAI 06.02.02-8 for additional information on the basis for 85 lbs.) Unlike the URG debris size guidance, the coatings will be assumed to be entirely fine particles so that all coating debris is conservatively assumed to pass through the Emergency Core Cooling

System (ECCS) strainers and reach the fuel assembly. The amount and characteristics of all constituents of the debris are discussed below.

The calculation for determining the acceptance criteria for the downstream fuel effects test has been completed. This calculation, which includes the clad debris fouling factor, is available for audit by the NRC at its convenience. As noted in the previous response, this supplemental response also includes a bounding estimate for Reflective Metal Insulation (RMI) as well as the results of a calculation of the maximum surface area of "latent aluminum" that could be in the suppression pool and not form chemical precipitates. The "latent aluminum" calculation is discussed in the response to RAI 06.02.02-11 Supplement 2.

STPNOC has not taken any departure from the design of the fuel as specified in the reference ABWR Design Control Document (DCD). STPNOC has taken a departure (STD DEP 6C-1) with respect to the ECCS suction strainers in the wetwell of the containment suppression pool. This departure replaces the stacked disk suction strainers in the DCD with state-of-the-art cassette-type suction strainers with a larger strainer surface area and includes prohibiting the use of fibrous material in the primary containment. This limits the amount of fiber entering the ECCS to a very small quantity of "latent fiber". Therefore, the departure will have the effect of improving the performance of the fuel with respect to downstream fuel effects.

Nevertheless, STPNOC will agree to a COL license condition, stating that at least 18 months prior to fuel load, an evaluation will be submitted to the NRC as part of a license amendment request confirming that the fuel for the initial fuel load satisfies the acceptance criteria related to the downstream effects of containment debris on the reactor fuel. The STP 3 & 4 design unique testing will be performed to confirm that downstream effects will not impair the ability to provide adequate flow to provide long term cooling for the fuel. Acceptance criteria for this testing will ensure adequate flow rate through the core region to cool the fuel for an extended period of time post-LOCA. The proposed wording for this license condition is provided at the end of this RAI response.

It is important to note that, even without the fuel testing that will be performed as part of this license condition, the ABWR design as applied to STP 3 & 4 provides reasonable assurance that downstream effects as a result of debris bypassing the ECCS suction strainers will not have a deleterious effect on the fuel. The basis for this assurance is discussed in the following COLA markups with gray shading showing the changes:

6C. 3.1 Downstream and Chemical Effects Discussion

The ABWR design provides reasonable assurance that downstream effects as a result of debris bypassing the strainers will not have a deleterious effect on critical components such as fuel rods, valves and pumps downstream of the suction strainers. The basis of this assurance is provided in the following:

6C.3.1.1 Latent Debris Generation

Relative to the generation of latent debris, the ABWR contains a number of design features and controls which reduce the likelihood of such debris being generated as compared with operating BWR and PWR plants. Access to the containment during power operation is prohibited as the containment is inerted, thereby eliminating the likelihood of latent debris generation due to work being performed during power operation. In addition, in the unlikely event that latent debris exists in the suppression pool during power operation, the suppression pool cleanup (SPCU) system provides on-going cleanup. This system is run on an intermittent basis during power operation and provides an early indication of any deterioration of the suppression pool water quality. The suction pressure of the SPCU pump is monitored and provides an alarm on low pressure. During refueling outages, when latent debris could be generated by workers inside the containment, temporary filters are used during post-construction systems testing in accordance with plant housekeeping and foreign material exclusion procedures, further reducing the potential for introducing debris to the suppression pool. STP 3 & 4 has an operational program for suppression pool cleanliness, documented in accordance with Section 13.4S of the FSAR, which provides for periodic inspections of the suppression pool for cleanliness during outage periods. This operational program is described in Subsection 6.2.1.7.1. Maintenance procedures provide procedure steps for removing, at periodic intervals, sediment and floating or sunk debris from the suppression pool that is not removed by the suppression pool cleanup system. Quarterly surveillance tests of Residual Heat Removal (RHR), High Pressure Core Flooder (HPCF), and Reactor Core Isolation Cooling (RCIC) systems provide further assurance that there is no blockage due to debris in the pump suction. Finally, the use of a stainless steel liner in the ABWR as opposed to carbon steel, which has been used in earlier version BWR suppression pools, significantly lowers the amount of corrosion products which can accumulate at the bottom of the suppression pool.

6C.3.1.2 LOCA-Generated Debris

Relative to the generation of debris from a postulated pipe break, the ABWR design contains a number of improvements from earlier BWR designs. The elimination of the recirculation piping removes a significant source of insulation debris from the containment and also reduces the likelihood of a large high energy pipe break which could lead to debris generation. For the STP 3 & 4 design, there is no fibrous insulation or calcium silicate on piping systems, including small bore piping, inside the containment. All thermal insulation material is a Reflective Metallic Insulation (RMI) design. RMI breaks up into shards too large to pass through the ECCS suction strainers which have a maximum 2.1 mm (1/12 inch) hole size. Furthermore, the use of fibrous and calcium silicate materials in the STP 3 & 4 Primary Containment is prohibited.

6C.3.1.3 Chemical Effects Debris

The STP 3 & 4 containment will not contain reactive materials such as aluminum, phosphates, or calcium silicate. In addition, the STP 3 & 4 Suppression Pool Cleanliness program (Subsection 6.2.1.7.1) ensures that quantities of latent debris, which might include aluminum or fiber, are kept to a minimum. A solubility calculation indicates that 4.5 square feet of latent aluminum would have to be present in the suppression pool to form aluminum precipitates under bounding conditions post-LOCA. Ensuring that there is less than 4.5 square feet of latent aluminum is within the capability of the containment cleanliness program.

Additionally, there is no exposed concrete inside the containment, i.e. it is covered by stainless steel or carbon steel, or qualified coatings. Even if the qualified coatings were to fail, there are no phosphates in the suppression pool water to form calcium precipitates.

Finally, since there is no exposed concrete there is no potential to form silicon precipitates. Similarly, even if the qualified coatings were to fail, there is no sodium in the suppression pool water to form sodium silicate precipitates.

6C.3.1.4 Debris Transport

The ABWR contains design features which reduce the transport of accident-generated debris to the suction strainers. The wetwell, which is the chamber in direct contact with the suppression pool, is largely empty with the only significant components/structures being an access tunnel, a grated catwalk and the Safety Relief Valve (SRV) discharge piping. There are no normal operating high energy piping systems in the wetwell which could break and lead to debris generation. The high energy piping in the ABWR, which consists largely of the main steam, Reactor Water Cleanup (RWCU) system, and feedwater piping under normal operating conditions, is located in the upper drywell area. Any debris which is generated by a break in these systems would need to pass through a circuitous route involving any one of the ten drywell connecting vents (DCVs) and then through any one of the thirty horizontal vents before reaching the suppression pool. The DCVs have horizontal steel plates located above the openings that prevent any material falling in the drywell from directly entering the vertical leg of the DCVs. A vertically oriented trash rack is installed around the periphery of the horizontal steel plate to intercept debris. In order for debris to enter the DCV, it would have to travel horizontally through the trash rack prior to falling into the vertical leg of the connecting vents. Thus, the ABWR is resistant to the transport of debris from the drywell to the wetwell.

6C.3.1.5 Suction Strainer Design

In addition to these mitigating features, the downstream effects are reduced by the suction strainers themselves. The strainers are designed to protect the ECCS pumps to allow them to function long-term after an accident. As a result, they are designed so that 100% of the ECCS flow is routed through them and filtered such that particles 2.1 mm or larger are captured by the strainer. STP 3 & 4 conforms to Revision 3 of Regulatory Guide 1.82.

6C.3.1.6 Diversity of ECCS Delivery Locations to the Core

The ABWR has diversification of ECCS delivery points which helps to reduce the consequences of downstream blockage. Should any blockage occur in the lower core region, such as the fuel filter, which could limit the effectiveness of systems like RHR, the HPCF will still be effective at providing cooling water because it delivers water through spargers located above the core.

6C.3.1.7 Related Tests

Regarding acceptance criteria for blockage of small clearances, it is noted that there should be no fiber downstream of the STP 3 & 4 suction strainers because the only fiber potentially inside primary containment (latent loose debris) will not be degraded during the pipe break and will not be small enough to pass through the 2.1 mm diameter holes in the CCI cassette-type suction strainers. Preliminary data from testing conducted by Westinghouse (WEC) to resolve GSI-191 has not identified any coagulation of particulate debris until after fiber is introduced to the flow stream. Therefore, blockage of small clearances in downstream components is not likely for the STP 3 & 4 downstream components. The analysis of the effects of debris on downstream components such as pumps, valves and heat exchangers in PWRs was documented in WCAP-16406, which was approved by the NRC. It is expected that the analysis results which showed acceptable performance of these components will apply to BWRs due to similarity in materials and clearances to the PWR components.

6C.3.1.8 Downstream Fuel Effects Test

For the initial fuel cycle, a downstream effects test is performed to ensure that debris bypassing the suction strainers does not impair the flow to the core. The following discusses the test plan, the analysis basis, and the debris assumptions used in this test.

6C.3.1.8.1 Test Plan

A test facility is comprised of a fuel assembly mock-up, a pump, associated recirculation piping, and a mixing tank to add the debris. The test is conducted with a single fuel assembly, including a fuel debris filter, a fuel inlet nozzle, and fuel spacer grids. The cross-section of the fuel is modeled exactly; the length of the fuel assembly is reduced. The fuel assembly is unheated.

The test initial conditions are at a flow rate of 3.326 kg/second, and at atmospheric pressure and ambient temperature. The flow rate is representative of the flow at recirculation conditions. The atmospheric pressure and ambient temperature result in a viscosity that is conservative with respect to pressure drop due to debris blockage. The test is initiated at clean conditions to establish a flow representative of post-LOCA recirculation conditions. The flow is injected at the fuel assembly inlet. Once a steady state has been established, the debris (described in 6C.3.1.8.3) is added to the system. The fibrous debris is added first. The fiber is added slowly and in small amounts. Once all the fibrous debris has been added, the remainder of the debris is added. The particulate debris is added in such a way that it does not coagulate. The pressure drop across the inlet and the entire fuel assembly is monitored. In addition, the flow rate and coolant temperature is monitored. The test is run until all debris has been deposited in the system and or a steady state pressure drop condition has been achieved.

6C.3.1.8.2 Analysis

6C.3.1.8.2.1 Introduction

An analysis determines the acceptable level of blockage in the fuel by LOCA generated debris which bypasses the ECCS suction strainer. This analysis ensures that the long term core cooling per Criterion 5 of 10CFR50.46 is maintained, the calculated core temperature is maintained at an acceptably low value, and decay heat is removed for an extended period of time required by the long-lived radioactivity remaining in the core. The analysis is performed with the LOCA model described in WCAP-17116 (Reference 6C-10). Potential deposition of particulate debris on the fuel and its impact on the heat transfer from the cladding is also included in the evaluation.

The results of the analysis are used to determine the acceptance criteria for the downstream fuel effects test, to be performed at least 18 months prior to initial fuel load.

6C.3.1.8.2.2 Analysis Approach

Although the diversification of ECCS delivery points (injection from the top of the core by the High Pressure Core Flooders and injection from below the core by the Low Pressure Core Flooder and Reactor Core Isolation Cooling) helps reduce the consequences of a blockage in the fuel assembly, for this analysis it is assumed that all the debris is injected from the bottom of the core and therefore, is exposed to the fuel debris filter, which is the most likely place for blockage to occur.

The analysis is performed for a feedwater line break for the following reasons. Following the break and after the blowdown is complete, the water level in the downcomer rises to the feedwater line (i.e. the break elevation). At that point, all the excess flow from the Low Pressure Core Flooder (LPCF) or Reactor Core Isolation Cooling (RCIC), not injected into the core will flow out through the break. The flow rate into the core is dependent upon the natural circulation head of colder water in the downcomer and the hotter water and two-phase mixture in the core region. As the core inlet begins to block, the core flow rate decreases. A steam line break, being at a higher elevation, will produce a higher natural circulation flow and therefore is less limiting than a feedwater line break for establishing the pressure drop limit at the fuel inlet.

For this analysis, the flow area at the fuel inlet is reduced to simulate blockage of the debris filter. All bypass flow paths, except for the inter-assembly bypass holes located in the bottom transition piece, are also assumed to be blocked. The bypass in the bottom nozzle is not likely to be blocked due the large opening size (10.3 mm diameter) which is significantly greater than the strainer hole size (2.1 mm). The reduced flow area at the core inlet decreases the core inlet flow rate and increases the core inlet differential pressure (DP). The minimum flow area is determined to ensure that no point in the core experiences significant cladding heat-up, measured by ensuring that the void fraction remains < 0.95. The corresponding DP at the core inlet, corrected for the changes in the core flow rate, is the parameter monitored and used as the acceptance criterion in the test.

Consistent with the methodology of WCAP-17116, conservative values of the nodal power peaking and pin-to-pin peaking factors for the hot assembly are chosen to place the hot rod at the Thermal Mechanical Operating Limit (TMOL). A core power corresponding to a decay heat at 5 minutes after shutdown is assumed as the debris accumulates at the debris filter and reduces the inlet flow area. This core power corresponding to 5 minute decay heat is conservatively kept constant thereafter. For the reasons stated below, blockage sufficient to reduce core cooling within 5 minutes is not likely:

- As shown in WCAP-17116, the core and the upper plenum retain significant inventory during the blowdown. The void fraction in the upper plenum remains below 1.0 (Figure 4-25). Therefore, additional water injected into the core before a quasi-steady state is established is minimal (i.e., the level in the downcomer increases to the FW line). After the quasi-steady state is achieved, the injection into the core is limited by the natural circulation head and core boil off.
- The debris laden flow from the suppression pool will be injected into the vessel only after the initial inventory of the ECCS piping, which is clean, is swept and injected into the vessel. Therefore, any suppression pool water will be further diluted by this clean initial injection.
- Although not credited in this analysis, the HPCF pumps (and RCIC) initially inject from the condensate storage tank (CST), which is a clean source of water. The LPCF pumps do not start injection until well after 2 minutes.

In addition, a parametric study is performed to determine the effect of fouling caused by deposition of particulate debris on the cladding. The level of initial fouling on the cladding is increased to represent the effect of uniform deposition of particulate debris on the cladding.

6C.3.1.8.2.3 Analysis Results

Figures 6C-2, 6C-3, and 6C-4 compare the core inlet DP, flow rate and void fractions for the cases with no blockage and with blockage resulting in a reduction of flow area by 90% of inlet flow area. Five minutes is the estimated amount of time required for the debris in the system to start to reach the fuel filters. The models assume clogging begins at 850 seconds because that is when the flow through the core reaches a steady state. In order to ensure that any effects seen are caused by the changes being made, steady flow is required. Despite a very high level of blockage, sufficient flow remains available to the core to ensure that the core void fraction both in the hot assembly and average assembly remain < 0.95.

In the ABWR design, the peak cladding temperature (PCT) occurs very early in the transient during the Reactor Internal Pumps (RIPs) coastdown phase, before ECCS injection occurs. Therefore, the PCT remains unaffected during the RIP coastdown by the subsequent blockage at the fuel inlet because the cladding temperature is maintained low (near the saturation temperature) as the core void fraction, both in the hot and average assemblies, is maintained below < 0.95. Figure 6C-5 provides a comparison of cladding temperature for the blocked and unblocked cases. The low fuel clad temperature also ensures that cladding oxidation does not occur in the long term cooling phase of the accident.

The impact on the clad temperature of fouling caused by an assumed deposition of particulate debris on the cladding is small. The increase in PCT from low fouling to high fouling cases is only approximately 30 deg F.

The results of the analysis provide an acceptable core inlet differential pressure (DP), corrected for the flow rates to account for the fact that the flow rate will decrease differently in the test loop (supplied by a pump) vs. in the analysis (controlled by natural circulation head).

$$\left[\frac{\Delta p_f}{\Delta p_i}\right]_{Test-Measured} = \left[\frac{\Delta p_f}{\Delta p_i}\right]_{LOCA-Aly} * \left(\frac{w_i}{w_f}\right)^2_{Aly} \left(\frac{w_f}{w_i}\right)^2_{Test}$$

Where subscript "*i*" denotes initial (i.e., unfouled conditions), "*f*" indicates fouled conditions, "*Aly*" refers to analysis and "*W*" is the flow rate into the assembly.

6C.3.1.8.3 Debris Assumptions for Downstream Test

The test is conducted using conservative assumptions regarding the debris that would be present in the suppression pool following a LOCA. The following debris types are included: (1) Coatings, (2) Sludge, (3) Dust/Dirt, (4) Rust Flakes, (5) RMI shards, and (6) Latent Fiber. No chemical debris is included since there are no credible sources of chemical debris in STP 3 & 4. The first four debris types are conservatively assumed to be particles smaller than 2.1 mm and are therefore all assumed to pass through the ECCS strainers. For the RMI shards and latent fiber, an assessment of the amount of the debris passing through the strainer is performed. Latent fiber debris upstream of the strainers is assumed to be 1 ft^3 (6C.3 item (6)). The fraction of latent fiber assumed to be small enough to pass through the strainers is 10 % based on conservatively assuming the fraction of bypass is 10 times the amount of destroyed fibrous insulation fibers (which are not credible in the ABWR) that bypassed CCI cassette-type strainers during testing for GSI-191 plants. Based on the size distribution of stainless steel RMI destroyed during jet testing (and shown in Figure 3-7 of NUREG/CR-6808), 2 % of the RMI within the break zone of influence is assumed to be shards smaller than 2.1 mm, and therefore small enough to pass through the strainers.

Since there are 872 fuel assemblies in the STP 3 & 4 core, the above debris amounts are reduced by a factor of 1/872. To account for the possibility of non-uniform debris deposition, a 10 % penalty is assumed. The debris amounts that are used in the test are shown below:

Debris Type	**Debris Assumed in Downstream Fuel Effects Test**
Coatings	0.107 lbs.
Sludge	0.246 lbs
Dust/Dirt	0.189 lbs.
Rust Flakes	0.063 lbs.
Stainless RMI Shards	78.211 in^2
Latent Fiber	0.218 in^3

6C.3.1.9 Summary

In summary, there is reasonable assurance that the downstream effects of material passing through the suction strainers will not adversely affect the fuel or other components. This conclusion is based upon the low potential for generating debris in the ABWR, the tortuous path for any debris to enter the wetwell from the drywell, the cleanup provisions for the water in the wetwell, the low potential for chemical debris , the small size of the holes in the suction strainers that filter out most debris, quarterly/periodic surveillance of HPCF, RHR, and RCIC systems which provides further assurance of the absence of debris which could affect their readiness for water injection capability, diversity of injection points for ECCS into the core, and preliminary data from PWR test results which show little impact on head loss in the fuel region from particulate only debris.

A test will be performed on the fuel to be used in the initial fuel cycle to confirm that debris will not adversely affect the fuel.

6C.6 References

6C-10 Westinghouse BWR ECCS Evaluation Model: Supplement 5 – Application to the ABWR, WCAP-17116-NP, Rev. 0, September 2009.

4.4.6 Testing and Verification

The testing and verification techniques to be used to assure that the planned thermal and hydraulic design characteristics of the core have been provided, and will remain within required limits throughout core lifetime are discussed in Chapter 14.

An analysis is performed to determine the required cooling for a fuel assembly post-LOCA. This analysis is discussed in Appendix 6C and is used to develop acceptance criteria for a downstream fuel effects test performed prior to initial cycle operation.

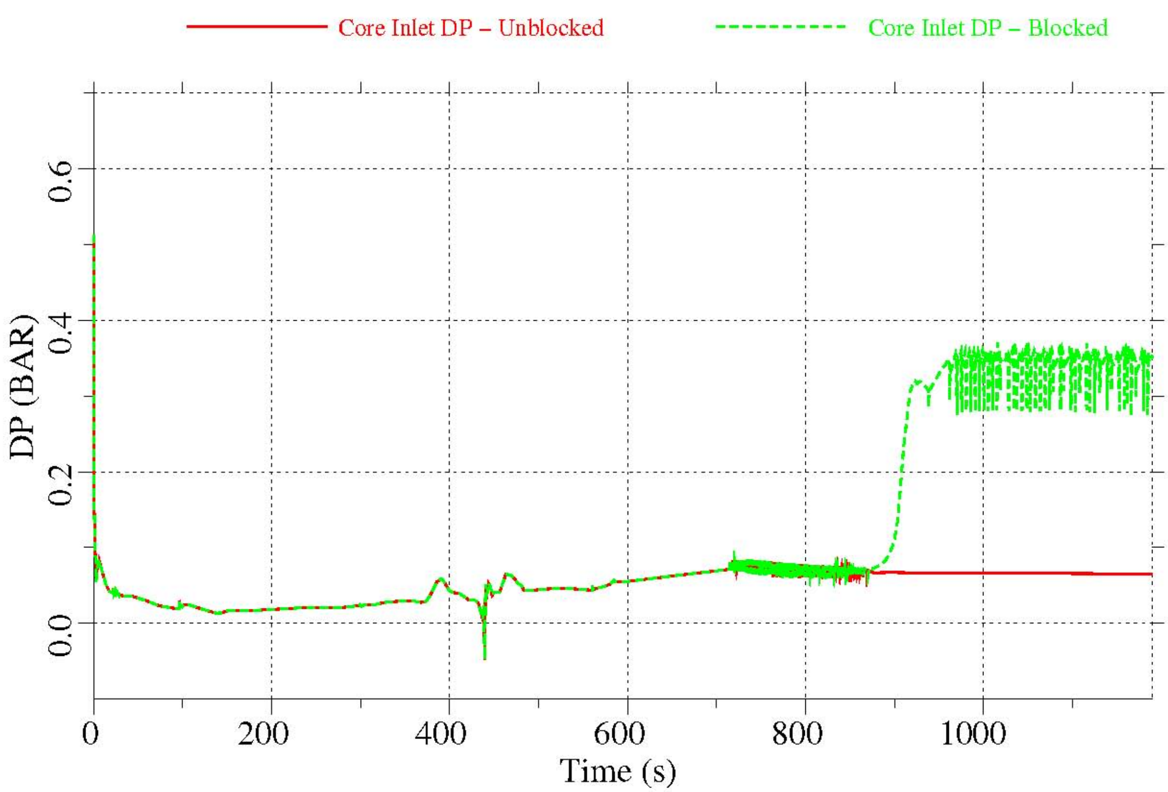


Figure 6C-2 Core Inlet Delta-P for Blocked and Unblocked Cases

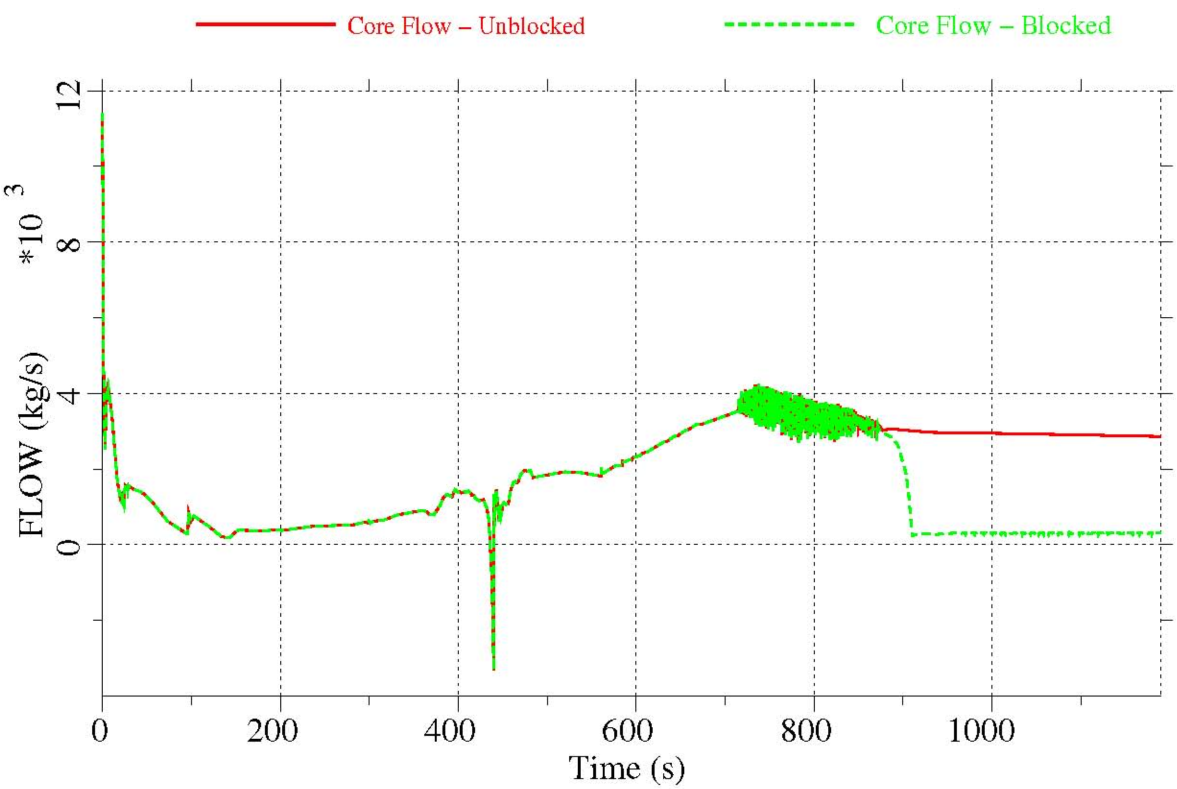


Figure 6C-3 Total Core Flow Rate for Blocked and Unblocked Cases

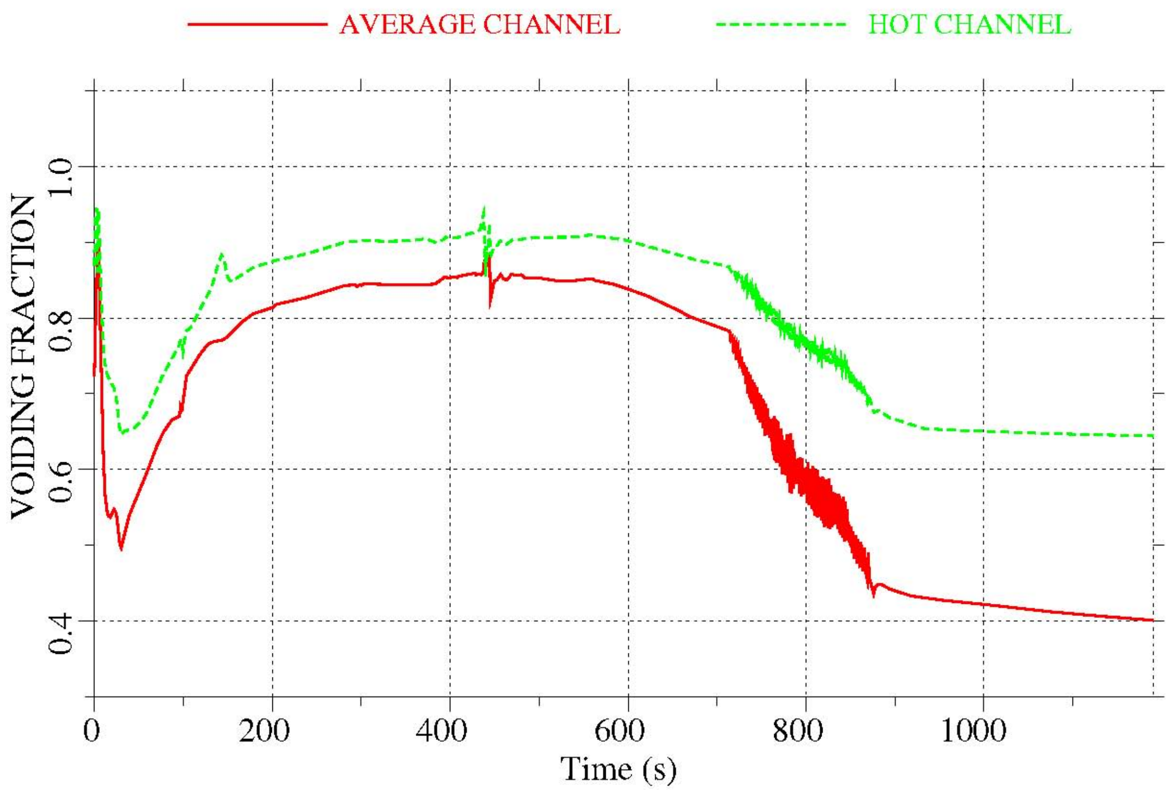


Figure 6C-4a Void Fraction for Unblocked Case

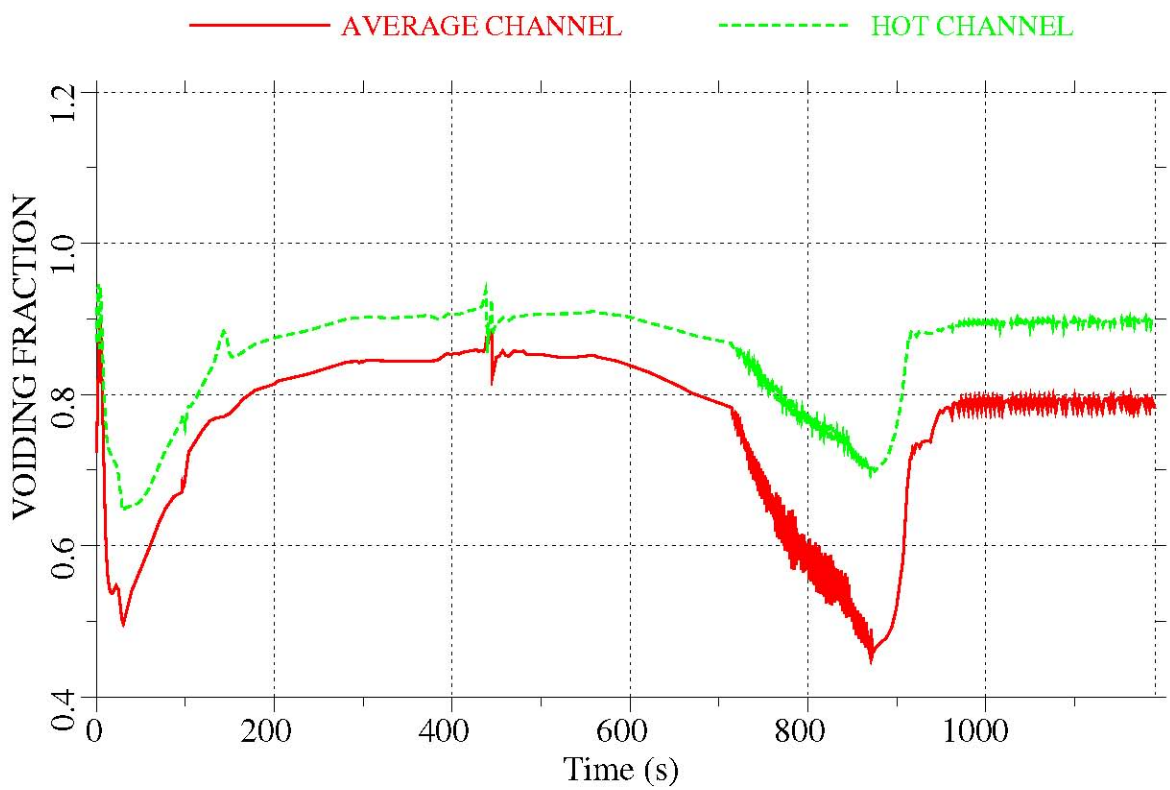


Figure 6C-4b Void Fraction for Blocked Case

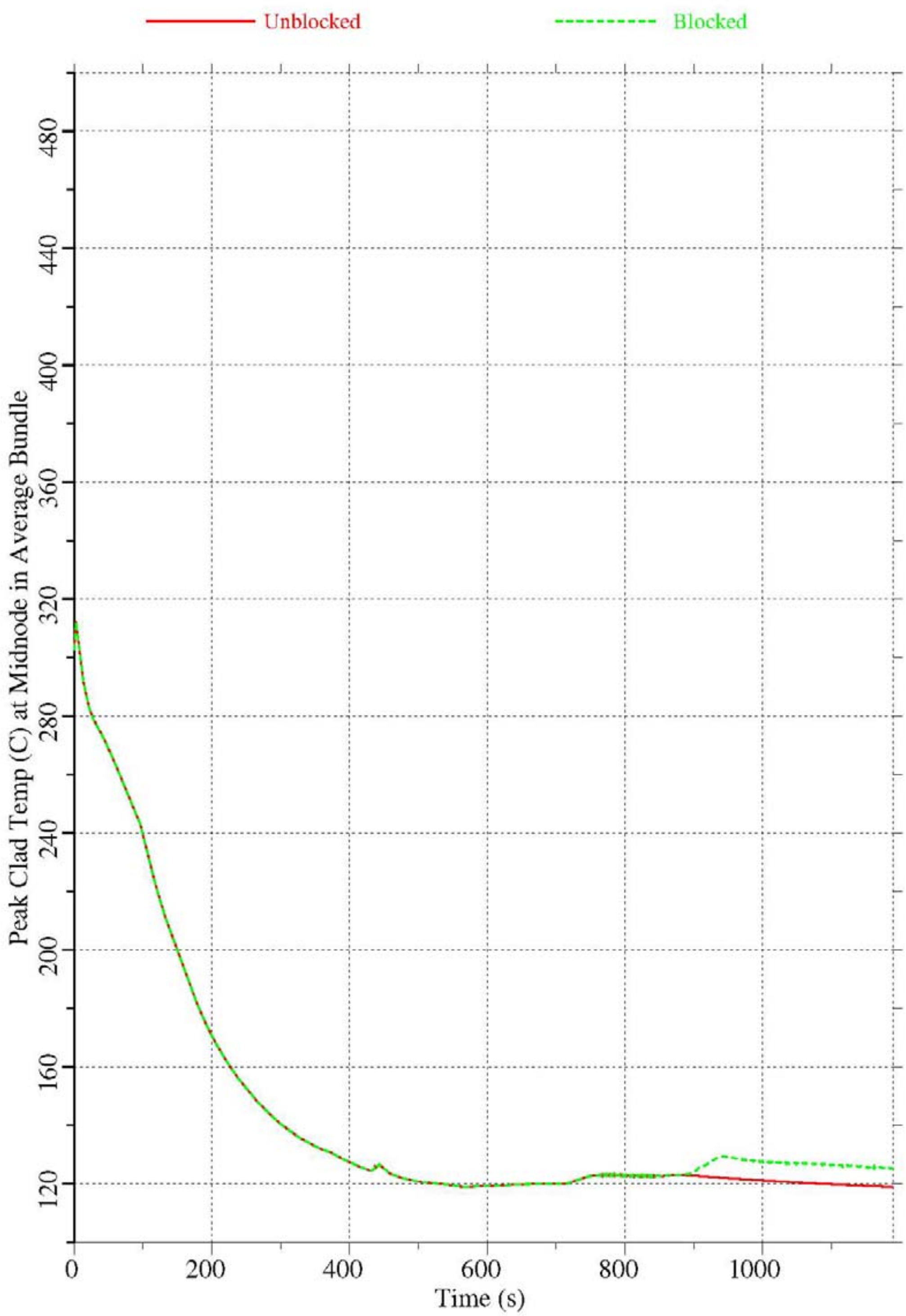


Figure 6C-5 Peak Clad Temperature for Blocked and Unblocked Cases

PROPOSED LICENSE CONDITION

A downstream fuel effects test will be conducted and the results provided to the NRC no later than 18 months prior to fuel load. The test plan, analysis basis, and debris assumptions are described in Appendix 6C.3.1.8. The test procedure will be provided to the NRC no later than 24 months prior to fuel load. The acceptance criteria for this test will be a fuel assembly inlet steady-state pressure drop less than 5.076 psid.

RAI 06.02.02-11 Supplement 2:

QUESTION:

Please discuss how the chemical effects evaluation for STP 3&4 was conservative with respect to dissolution and precipitation over the full range of environmental conditions (e.g., pH and temperature) that may exist over the 30-day ECCS mission time. For example, based on the September 28, 2009, response to RAI 06.02.02-9, it appears dissolution testing was conducted only in the pure water environment, but the dissolution rate of aluminum would likely increase when the pH is raised with sodium pentaborate. The response to Item b of RAI 06.02.02-9 suggests the suppression pool pH-temperature-time profile may not have been calculated to determine, for example, the maximum pH and corresponding corrosion rate expected for aluminum. In your response, please address the materials that may be in containment and have been shown to form chemical precipitates, such as aluminum, calcium (e.g., from concrete), and silicon.

RESPONSE:

In the original response to RAI 06.02.02-11 provided in STPNOC Letter No. U7-C-STP-NRC-090226, dated December 21, 2009, it was stated that STPNOC is preparing a calculation of the maximum surface area of “latent” aluminum that could be in the suppression pool, corrode, and then dissolve over the 30-day post-LOCA period, and still remain in solution in the suppression pool (i.e., not form precipitates). That calculation has been completed. The calculation used aluminum dissolution rates from the NRC contractor (Argonne National Laboratories) test report, “Aluminum Solubility in Boron-Containing Solution as a Function of pH and Temperature,” ADAMS accession Number ML091610696 dated September 19, 2008.

From DCD Tier 2 Section 3I.3.2.3, reactor water pH for the design basis LOCAs inside primary containment range from a pH of 5.3 to 8.9. For aluminum dissolution and precipitation, these bounding values of pH, the normal pH of 7.0, and an intermediate pH of 6.0, coupled with the temperature profile over the 30-day post-LOCA period, produced the following maximum surface areas of aluminum that would stay in solution and not form precipitates:

pH:	8.9	7.0	6.0	5.3
sq. ft. aluminum:	3000	138	20	4.5

The results show that for the most conservative assumptions for pH, latent aluminum surface area up to 4.5 square feet can be present for the 30 day post-LOCA period, and the resulting dissolved aluminum would stay in solution and not form precipitates.

This supplemental response provides a COLA markup, which is shown below. Changes from Rev. 3 of the COLA are shown with gray shading.

6C.3 RG 1.82 Improvement

(7) STP 3&4 design specifications prohibit aluminum inside primary containment. Despite that prohibition, it is conservatively assumed that there is 4.5 sq. ft. of aluminum in the primary containment; however, this quantity of aluminum is not expected to form aluminum precipitates. Analysis has shown that, under conservative conditions, the maximum surface area of latent aluminum that could be present in the primary containment, corrode over the 30-day post-LOCA period and not precipitate out of the suppression pool solution is 4.5 sq. ft. The implementation of the STP 3 & 4 suppression pool cleanliness and FME programs will ensure that latent aluminum quantities would be less than this amount. Therefore, chemical precipitates due to presence of latent aluminum in primary containment will not be generated.

RAI 15.08-3:

QUESTION:

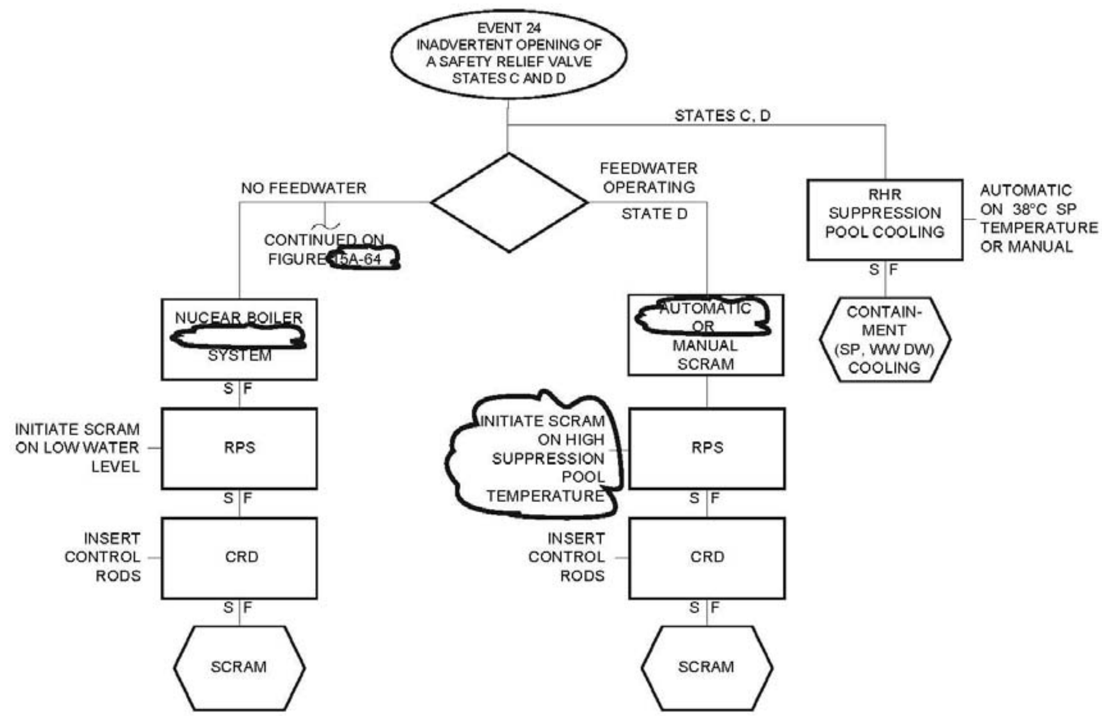
The applicant identified the change to FSAR Figure 15A-29 as an administrative change. The suppression pool temperature set point for the automatic initiation of RHR is changed from 38 degrees C to 35 degrees C. Provide justification for this change.

RESPONSE:

ABWR DCD Tier 2, Subsection 6.2.1.1 discusses the analyzed response of the Reactor Coolant System and the Containment System during the short-term blowdown period of the Feedwater Line Break accident. The analysis assumes a Suppression Pool operating and wetwell airspace initial temperature of 35°C. The service water temperature is conservatively assumed at 35°C to maximize the suppression pool water temperature. Also, Technical Specification (TS) LCO 3.6.2.1a specifies that the Suppression Pool average temperature shall be ≤ 35°C when thermal power is > 1% RTP, and no testing that adds heat to the Suppression Pool is being performed. Because of this, the change from 38°C to 35°C was considered an administrative departure (STD DEP Admin) to align the figure with the analysis and TS requirements for consistency.

The ABWR is unique with respect to other BWRs because it is designed with an automatic start of Suppression Pool cooling. This design provision is an operator aid, a back-up for operator manual action to start the Suppression Pool cooling, and is not directly credited in the analysis. Because the design includes an alarm at 35°C, a TS requirement of ≤ 35°C, and operator capability to initiate cooling, the change from 38°C to 35°C should not have been made. Therefore, the automatic start of Suppression Pool cooling will remain at 38°C as shown on DCD Figure 15A-29.

STP 3 & 4 COLA Revision 3 will be revised as shown below.


EVENT 24
INADVERTENT OPENING OF
A SAFETY RELIEF VALVE
STATES C AND D
STATES C, D
NO FEEDWATER
FEEDWATER
OPERATING
STATE D
CONTINUED ON
FIGURE 15A-64
RHR
SUPPRESSION
POOL COOLING
AUTOMATIC
ON 38°C SP
TEMPERATURE
OR MANUAL
S F
CONTAIN-
MENT
(SP, WW DW)
COOLING
NUCEAR BOILER
SYSTEM
AUTOMATIC
OR
MANUAL
SCRAM
INITIATE SCRAM
ON LOW WATER
LEVEL
RPS
INITIATE SCRAM
ON HIGH
SUPPRESSION
POOL
TEMPERATURE
INSERT
CONTROL
RODS
CRD
SCRAM