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10 CFR 50.90

February 25, 2020

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555-0001

R. E. Ginna Nuclear Power Plant
Renewed Facility Operating License No. DPR-18
NRC Docket No. 50-244

Subject: License Amendment Request to Add a One-Time Note for Use of Alternate Residual Heat Removal Methods

In accordance with 10 CFR 50.90, "Application for amendment of license, construction permit, or early site permit," Exelon Generation Company, LLC (EGC) requests changes to the Technical Specifications (TS) of the R. E. Ginna Nuclear Power Plant (Ginna).

EGC proposes to revise TS 3.4.7 ("RCS Loops - MODE 5, Loops Filled"), TS 3.4.8 ("RCS Loops – MODE 5, Loops Not Filled"), TS 3.9.4 ("Residual Heat Removal (RHR) and Coolant Circulation - Water Level ≥ 23 Ft"), and TS 3.9.5 ("Residual Heat Removal (RHR) and Coolant Circulation - Water Level < 23 Ft") to add an asterisk to allow the use of alternate means for residual heat removal. Detailed descriptions of the alternatives are provided in the Technical Evaluation.

This one-time change is requested to support the station in the shutdown of the reactor during the upcoming refueling outage scheduled to start in April 2020.

The proposed change has been reviewed by the Ginna Plant Operations Review Committee in accordance with the requirements of the EGC Quality Assurance Program.

EGC requests approval of the proposed license amendment by April 5, 2020. Once approved, the amendment shall be implemented within 30 days of receipt. There are no regulatory commitments contained within this letter.

In accordance with 10 CFR 50.91, "Notice for public comment; State consultation," paragraph (b), EGC is notifying the State New York of this application for license amendment by transmitting a copy of this letter and its attachments to a designated State Official.

Should you have any questions concerning this letter, please contact Tom Loomis at (610) 765-5510.

I declare under penalty of perjury that the foregoing is true and correct. This statement was executed on the 25th day of February 2020.

Respectfully,

Shannon Rafferty-Czincila
Director - Licensing
Exelon Generation Company, LLC

Attachments: 1. Evaluation of Proposed Change
2. Markup of Proposed Technical Specifications Pages

cc: NRC Regional Administrator, Region I
NRC Senior Resident Inspector, Ginna
NRC Project Manager, Ginna
A. L. Peterson, NYSERDA

Attachment 1
Evaluation of Proposed Change

1.0 SUMMARY DESCRIPTION

In accordance with 10 CFR 50.90, "Application for amendment of license, construction permit, or early site permit," Exelon Generation Company, LLC (EGC) requests changes to the Technical Specifications (TS) of the R. E. Ginna Nuclear Power Plant (Ginna).

EGC proposes to implement a one-time addition to TS 3.4.7, TS 3.4.8, TS 3.9.4, and TS 3.9.5 to add an asterisk to allow the use of alternate means of residual heat removal.

This one-time change is requested to support the station in the shutdown of the reactor during the upcoming refueling outage scheduled to start in April 2020, due to the possibility of not being able to open the normal RHR suction valve MOV 700.

2.0 DETAILED DESCRIPTION

Due to concerns over the opening of the Motor Operated Valve (MOV) 700 (Figure 1 in the Technical Evaluation) associated with the operation of the Residual Heat Removal (RHR) system, this one-time change is being requested to provide an alternate form of plant cooldown which may be needed during the shutdown of the reactor during the upcoming refueling outage scheduled for April 2020.

LCO 3.4.7 (“RCS Loops-Mode 5, Loops Filled”) requires that one RHR loop shall be operable and in operation, and either: a) one additional RHR loop shall be operable, or b) the secondary side water level of at least one steam generator shall be greater than or equal to 16%. If MOV 700 is not able to be opened, neither RHR loop will be able to be made operable. The Required Action associated with inability to have two loops of RHR operable is to (B.2) “initiate Action to Restore one RHR loop to OPERABLE status and Operation” immediately. The ability to use a water-solid SG loop, or an alternate RHR loop, will allow this action to be met.

LCO 3.4.8 (“RCS Loops – MODE 5, Loops Not Filled”) requires that two RHR loops shall be OPERABLE and one RHR loop be in operation during MODE 5 with RCS loops not filled. As discussed previously, if MOV 700 is not able to be opened, neither RHR loop will be able to be made operable. The Required Action associated with inability to have two loops of RHR operable is to (B.2) “initiate Action to Restore one RHR loop to OPERABLE status and Operation” immediately. The ability to use an alternate RHR loop will allow this action to be met.

LCO 3.9.4 ("Residual Heat Removal (RHR) and Coolant Circulation - Water Level $\geq$ 23 Ft"), applicable during refueling operations, requires among other things, to (A.3) initiate action to satisfy RHR loop requirements. If MOV 700 is not able to be opened, RHR loop requirements could not be met. The ability to use an alternate RHR loop will allow this action to be met.

LCO 3.9.5 ("Residual Heat Removal (RHR) and Coolant Circulation - Water Level < 23 Ft"), applicable during refueling operations, requires that when there are less than the required number of RHR loops operable, (A.1) initiate action to restore RHR loop(s) to operable status, and if there is no RHR loop in operation, (B.2) initiate action to restore one loop to operation. If MOV 700 is not able to be opened, RHR loop requirements could not be met. The ability to use an alternate RHR loop will allow this action to be met.

The proposed methods of cooldown to "MODE 5, loops filled" are the water solid Steam Generator cooldown method (for as long as it is effective) or the proposed alternate RHR method. This water solid Steam Generator method involves removing residual heat by filling the steam lines with water and using the Steam Generators as water-to-water heat exchangers. This method was previously discussed in the Reference 2 application which was subsequently approved for mitigating 10 CFR 50.48 fire protection events as part of Ginna's Appendix R and NFPA 805 strategies (Reference 1). When the Steam Generator cooldown method is no longer effective ("MODE 5, loops not filled"), another proposed method will provide an alternate RHR loop using primarily the low-pressure Emergency Core Cooling System flowpath (Figure 2 in the Technical Evaluation). In order to implement this method, additional piping, fittings, connections, and hoses will need to be added. These additional components will be designed, procured, installed, and tested to rigorous standards, consistent with safety-related Class 1 or Class 2 components, as appropriate. Another contingency method has been developed, in the small possibility that MOV 700, when opening is attempted, will only open far enough to come off its seat, i.e., not enough to be a viable cooldown method flowpath but enough that it could no longer be considered an isolation boundary. For this eventuality, alternate cooling will be provided by an alternate loop from the Reactor Coolant System (RCS) to the suction of the RHR pumps and then via low pressure ECCS flowpaths (Figure 3 in the Technical Evaluation).

Following the isolation of the RHR system during heat up from the 2018 refueling outage (RFO), MOV 700 was closed. Normally, the valve is re-opened as part of the interlock and leakage testing subsequent to the isolation; however, during the startup, the testing was delayed. While attempting to reopen the valve, the additional time is likely to have caused heating and distortion of the valve disc resulting in additional thrust required to open the valve. Mitigation strategies to eliminate this potential cause are in place to ensure the valve opens. MOV 701 does not experience heating from the RCS following closure, and as a result is not susceptible to the same mechanism of uneven heating and distortion that prevented opening MOV 700 during heatup from the 2018 refueling outage. In case this is not the only cause for the valve to not open and another failure mechanism is contributing to the increased required thrust to open the valve, other strategies are in place to assist in opening the valve. Some of the potential contributors, e.g. guide rail to disc interference, could result in partial opening of the valve. Although unlikely, a contingency strategy is desired to allow completion of the cooldown and removal of fuel from the reactor to enact repairs on this valve that is un-isolable from the RCS A hot leg. MOV 700 will be refurbished this upcoming outage.

Attachment 2 contains mark-ups of the affected TS pages, respectively, for the proposed change.

3.0 TECHNICAL EVALUATION

As discussed in the Ginna Updated Final Safety Analysis Report (UFSAR), the residual heat removal loop consists of two heat exchangers, two pumps, piping, and the associated valves and instrumentation. After the steam generators have been used to reduce the reactor coolant temperature to 350^0F, decay heat cooling is initiated by aligning the residual heat removal pumps to take suction from the reactor coolant system loop A hot leg through MOVs 700 and 701 and discharge through the residual heat removal heat exchangers to the loop B cold leg through MOVs 720 and 721. With both pumps and heat exchangers in operation, residual heat removal flow is adjusted to maintain a cooldown rate of less than 50°F/hr. If only one pump and heat exchanger are available, cooldown is accomplished at a lower rate. The heat from the

residual heat removal heat exchangers is normally transferred to the Component Cooling Water (CCW) system (UFSAR Section 9.2.2), and from the CCW system to the Service Water (SW) system (UFSAR Section 9.2.1).

The proposed changes will ensure that a safe cooldown, from hot standby to refueling conditions, can be attained if valve MOV 700 is unable to be opened by removing residual heat by filling the steam lines with water and using the SGs as water-to-water heat exchangers. Use of solid water SG cooldown method was previously discussed in the Reference 2 application which was subsequently approved for mitigating 10 CFR 50.48 fire protection events as part of Ginna's Appendix R and NFPA 805 strategies (Reference 1). Detailed analyses of all stressors, including dead, live, and seismic forces have been evaluated to remain within code allowable values. Following use of water-solid cooldown, an alternate RHR method will employ a combination of components within the existing normal RHR loops, additional piping, fittings, hoses, and connections meeting safety-related Class 1 or 2 standards and portions of the low pressure ECCS injection system.

The operation of these RHR methods, and the justification for their use, is provided below.

Water-solid SG cooldown is accomplished by lining up a condensate pump to the main feedwater line and controlling flow to fill the steam lines full of water. The Steam Generators will act as water-to-water heat exchangers with the RCS. One Reactor Coolant Pump will remain operational to maintain circulation, enhancing heat exchange efficiency and maintaining proper boric acid mixing in the RCS. Proper shutdown boric acid concentration will have been achieved prior to initiating water-solid SG cooldown. Throughout this evolution, condensate flow will be controlled, circulating water from the condensate tanks and rejecting heat in the condenser and condensate cooler to the Circulating Water System. One of the three condensate pumps will be used at a time, and the motor-driven Auxiliary Feedwater (AFW) and SFAW pumps will remain available as backups.

Prior to water-solid SG cooling becoming ineffective, an alternate means will be implemented. RCS will be routed from the loop B cold leg, through newly installed piping, fittings, hose, and connections all meeting safety-related Class 1 or 2 criteria as appropriate, to the suction of the RHR pumps via connection between valves 700 and 701. Flow will then be routed to the reactor vessel upper plenum injection nozzles via the normal low pressure ECCS flowpath. In the small possibility that MOV 700, when opening is attempted, will only open far enough to come off its seat (i.e., not enough to be a viable cooldown method flowpath but enough that it could no longer be considered an isolation boundary) RCS will be routed from the loop B cold leg, through newly installed piping, fittings, hoses, and connections to the suction of the RHR pumps via a connection upstream of MOV 851A. Flow will then be routed to the reactor vessel upper plenum injection nozzles via the normal low pressure ECCS flowpath.

Detailed analyses performed to ensure that these activities will not result in a significant increase in the probability or consequences of any accident previously evaluated are described herein.

Since this LAR is requesting alternate cooling methods to be employed to cool the RCS from 350^0F to refueling conditions, the following accidents correlating to a shutdown reactor are evaluated:

Mode 4 LOCA - This event would involve a Loss of Coolant Accident in Mode 4. The addition of the new equipment will not impact the ability of the plant to mitigate the consequences of a LOCA in Mode 4 as previously analyzed in the UFSAR.

Startup of an Inactive Loop - This event would involve the startup of an inactive RCS loop. This event is bounded by the UFSAR analyzed scenario. The temperature of the inactive loop is much lower than the analyzed condition, and since the RCS pressure response is directly proportional to the temperature change, the peak RCS pressure will be lower than the analyzed scenario.

Seismic Event - All Systems, Structures, and Components (SSCs) required to provide safe shutdown capability are designed to seismic Category I requirements. These include only the SSCs required to shut down from Mode 1 to Mode 3, and to maintain in Mode 3 indefinitely. Ginna subsequently performed seismic analyses for the equipment required to attain water solid SG cooldown, using the approved Ground Motion Response Spectra (GMRS), and all code allowable stresses were met. If any non-seismic equipment is in use during this evolution and is affected by an actual seismic event (e.g., the condensate pump), a seismic method to accomplish the function, such as use of Auxiliary Feedwater, will be utilized.

Tornado Event – All components in the steam and feedwater systems required for safe shutdown, including water solid SG cooldown, have been successfully analyzed for tornado effects, including tornado missiles. If any non-tornado protected equipment which could be used for this operation, such as the condensate pumps, were affected by a tornado, safe shutdown could be maintained with tornado-protected equipment, such as the SAFW pumps. All components needed to effect alternate residual heat removal are protected from tornado forces.

As a part of the proposed alternate alignments, steam line water hammer was evaluated as a potentially different kind of event from that previously evaluated. The steam lines have been analyzed to be filled with water and remain within allowable stresses. The potential for water hammer in the feedwater lines was also previously evaluated in response to Unresolved Safety Issue A-1, in a letter dated October 19, 1982 (Reference 3). The potential for water hammer is limited by the flow rate and capacity of the condensate pumps, which have a higher capacity but much lower discharge pressure than the AFW pumps. To ensure that condensate flow does not significantly increase during filling of the SGs, the recirculation line to the condenser will be blocked open to accept the majority of the flow, and the remaining flow will be throttled using a manual valve to maintain the flow rate to below 375 gpm to each steam generator (flow will be controlled to about 325 gpm/SG to maintain margin). Calibrated clamp-on instrumentation will be used to verify flowrate. The use of manual valves rather than power operated valves reduces the potential for inadvertent flow rate increases. If the AFW or the SAFW pumps were required to be called into operation during water solid SG cooling, i.e. if the condensate pumps became unavailable, then these would also be limited to a flow rate of less than 325 gpm to each SG.

Operation of an alternate RHR path, principally using equipment associated with the low pressure ECCS, will not result in a new or different kind of accident. Any additional piping, fittings, connections, and hoses will be procured and installed to the high standards needed to operate in these conditions. Procedural guidance for this operational mode will be developed, reviewed, and carried out to the same high standards used for plant safety applications.

The projected operational mode of the water-solid SG cooldown system will include the use of one of the three condensate pumps, as well as portions of the feedwater and condensate piping, with discharge to the blowdown flash tank which will then return water to the condenser providing a suction source to the condensate pumps. Other means of SG makeup include the GE Betz system and the Service Water System. Additionally, safety-related motor-driven AFW pumps can draw from the Condensate Storage tanks, and the SAFW pumps can draw from the Deionized Water Storage tank and the Service Water System.

The RHR/ECCS is capable of being isolated by a series of two in-series valves from the RCS in containment. As part of this change, there will be two isolation valves on the suction side and an ECCS check valve and an MOV on the discharge side; therefore, redundancy of isolation is maintained.

The proposed alternate residual heat removal means, to include a water solid secondary loop, as well as an alternate RHR loop using primarily low pressure ECCS flowpath components and engineered equivalents, does not result in a reduction in the margins of safety. All operational characteristics and constraints, analytical design considerations, and equipment quality and installation will ensure original safety margins are maintained.

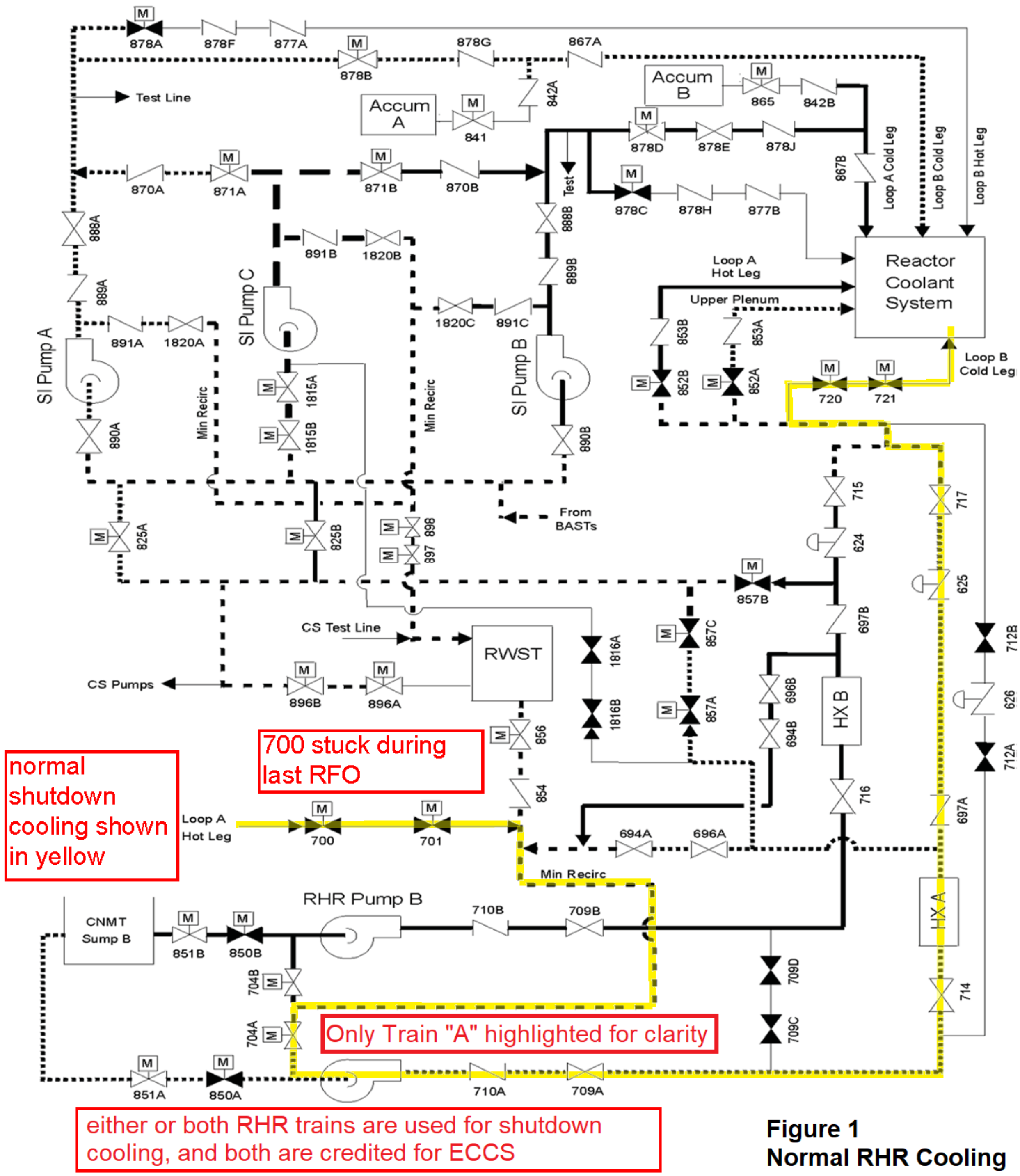


Figure 1
Normal RHR Cooling

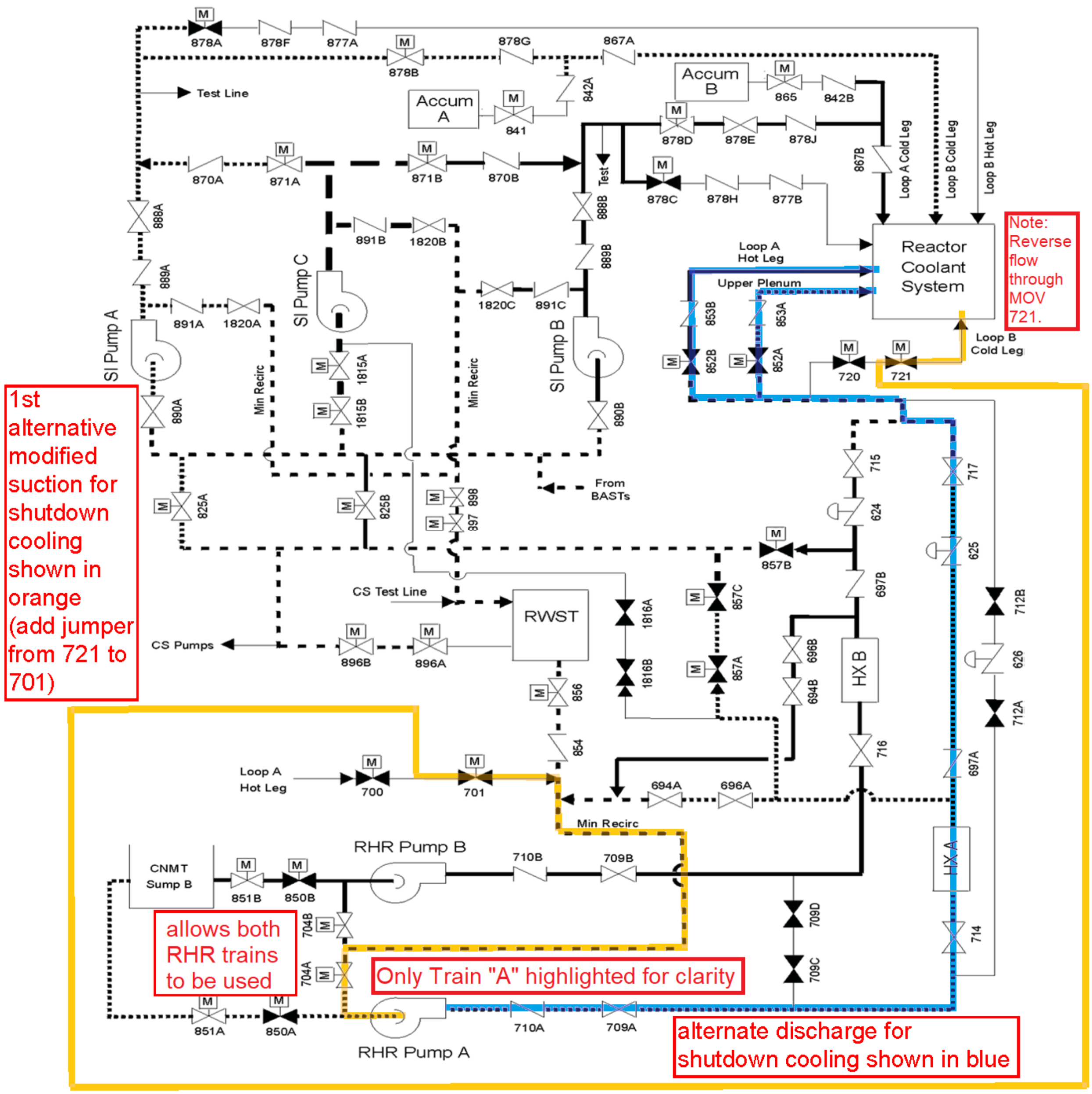
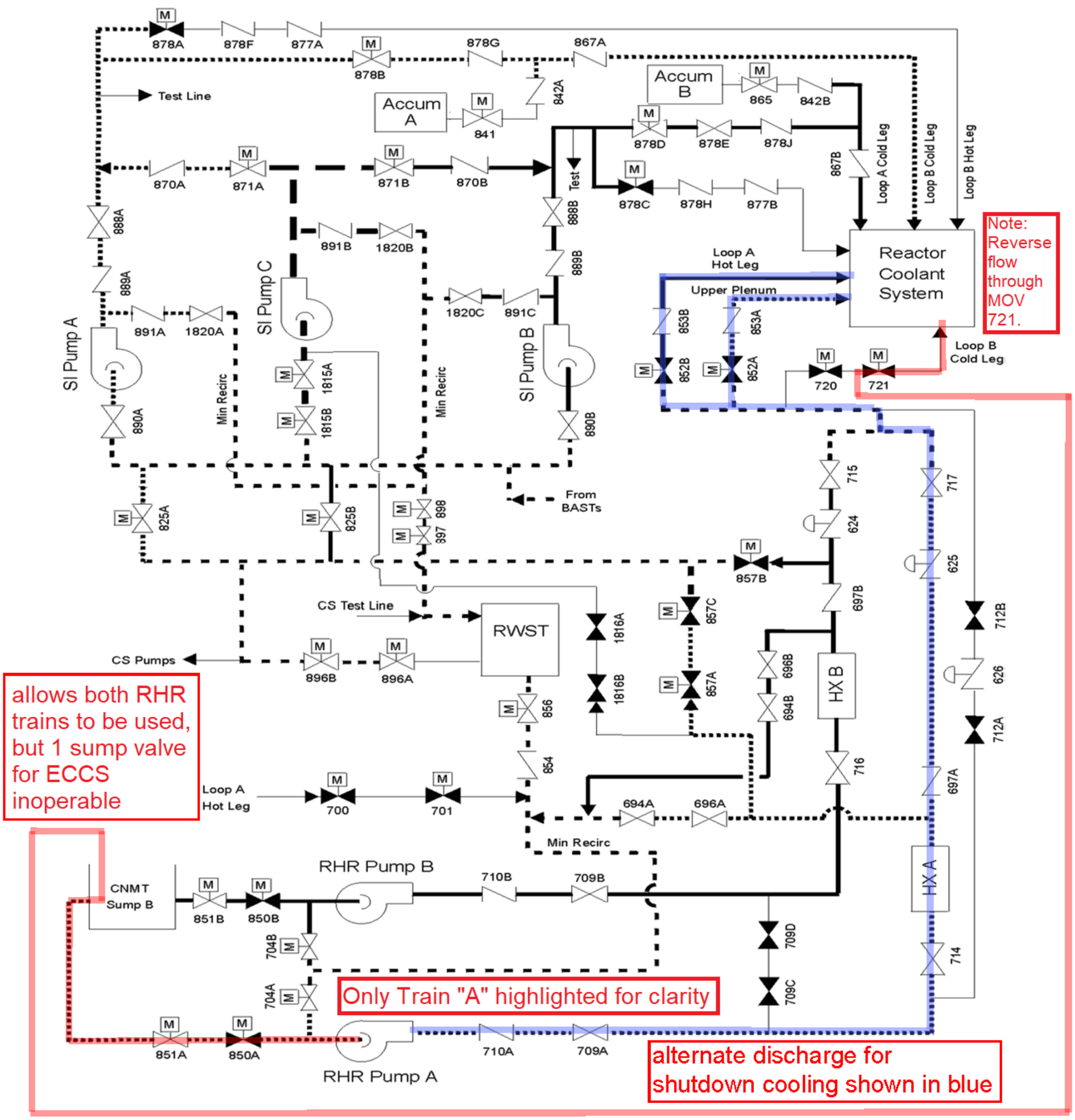


Figure 2
Alternate RHR Cooling #1



2nd alternative modified suction for shutdown cooling shown in red with jumper from 721 to 851A (only used if 700 stuck partially open but less than 10% open)

Figure 3
Alternate RHR Cooling #2

4.0 REGULATORY EVALUATION

4.1 Applicable Regulatory Requirements/Criteria

EGC proposes to revise TS 3.4.7 ("RCS Loops – MODE 5, Loops Filled"), TS 3.4.8 ("RCS Loops - MODE 5, Loops Not Filled"), TS 3.9.4 ("Residual Heat Removal (RHR) and Coolant Circulation - Water Level ≥ 23 Ft"), and TS 3.9.5 ("Residual Heat Removal (RHR) and Coolant Circulation - Water Level < 23 Ft") to add an asterisk to allow the use of alternate means for residual heat removal.

This one-time change is requested to support the station in the shutdown of the reactor during the upcoming refueling outage scheduled to start in April 2020. The proposed method of cooldown to MODE 5 is the water solid Steam Generator cooldown method. This method involves removing residual heat by filling the steam lines with water and using the Steam Generators as water-to-water heat exchangers. This method was previously discussed in the Reference 2 application which was subsequently approved for mitigating 10 CFR 50.48 fire protection events as part of Ginna's Appendix R and NFPA 805 strategies (Reference 1). Prior to this method becoming ineffective (MODE 5, loops not filled), an alternate RHR loop will be employed consisting of portions of the normal RHR loop, additional piping, fittings, hoses, and connections, and portions of the low pressure ECCS system. The following regulatory requirements/criteria apply:

1) General Design Criterion 34 (Residual Heat Removal) - A system to remove residual heat shall be provided. The system safety function shall be to transfer fission product decay heat and other residual heat from the reactor core at a rate such that specified acceptable fuel design limits and the design conditions of the reactor coolant pressure boundary are not exceeded. Suitable redundancy in components and features, and suitable interconnections, leak detection, and isolation capabilities shall be provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure.

   The alternate system provides a temporary method to transfer fission product decay heat and other residual heat from the reactor core at a rate such that specified acceptable fuel design limits and the design conditions of the reactor coolant pressure boundary are not exceeded. Redundancy, interconnections, leak detection, and isolation capabilities are provided when the alternate RHR loops are in use.

2) General Design Criterion 35 (Emergency Core Cooling) - This criterion requires that an ECCS with the capability for accomplishing abundant emergency core cooling shall be provided. The system safety function shall be to transfer heat from the core following any loss of reactor coolant such that fuel and clad damage that could interfere with continued effective core cooling is prevented and clad metal-water reaction is limited to negligible amounts.

   The alternate RHR means are designed so that their operation will not interfere with the emergency core cooling function to limit the clad metal-water reaction to acceptable amounts for all sizes of breaks in the reactor coolant piping up to the equivalent of a double-ended rupture of the largest pipe.

3) General Design Criteria 37 (Testing of Emergency Core Cooling System (ECCS)) - The emergency core cooling system shall be designed to permit appropriate periodic pressure and functional testing to assure (1) the structural and leak-tight integrity of its components, (2) the operability and performance of the active components of the system, and (3) the operability of the system as a whole and, under conditions as close to design as practical, the performance of the full operational sequence that brings the system into operation, including operation of applicable portions of the protection system, the transfer between normal and emergency power sources, and the operation of the associated cooling water system.

   The design and the use of alternate residual heat removal means will not prevent testing of the ECCS, when the plant is in a mode where ECCS is required.

4) General Design Criteria 54 (Piping Systems Penetrating Containment) - Piping systems penetrating primary reactor containment shall be provided with leak detection, isolation, and containment capabilities having redundancy, reliability, and performance capabilities which reflect the importance to safety of isolating these piping systems. Such piping systems shall be designed with a capability to test periodically the operability of the isolation valves and associated apparatus and to determine if valve leakage is within acceptable limits.

   All valves used as containment isolation valves will meet all containment isolation valve criteria for operation and testing.

4.2 Precedent

The use of water-solid steam generator cooldown was previously discussed in the Reference 2 application which was subsequently approved for mitigating 10 CFR 50.48 fire protection events as part of Ginna's Appendix R and NFPA 805 strategies (Reference 1). Alternate cooldown methods, where valve MOV 700 was postulated to be inoperable, were also discussed in the NRC SER for SEP Topics V-10.B, V-11.A, V-11.B, VII-3, and IX-3 (Safe Shutdown Systems), November 14, 1980.

4.3 No Significant Hazards Consideration

Exelon Generation Company, LLC (EGC) has evaluated whether or not a significant hazards consideration is involved with the proposed amendment by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:

1. Does the proposed amendment involve a significant increase in the probability or consequences of an accident previously evaluated?

   Response: No.

   This one-time change is requested to support the station in the shutdown of the reactor during the upcoming refueling outage scheduled to start in April 2020. The proposed method of cooldown during Mode 5 is the water solid Steam Generator cooldown method. This method involves removing residual heat by filling the steam lines with water and using the Steam Generators as water-to-water heat exchangers. The proposed method to

achieve Mode 5, loops not filled, utilizes portions of the normal RHR loop, additional piping, fittings, hoses, and connections meeting to safety-related Class 1 or 2 criteria, and portions of the low pressure ECCS system. These proposed alternative methods will not act as a precursor or an initiator for any transient or design basis accident; therefore, the proposed change does not significantly increase the probability of any accident previously evaluated.

The proposed change provides an alternate means to remove decay heat and is intended to mitigate the consequences of an initiating event within the assumed acceptance limits. This alternative method has been analyzed to ensure that it does not involve a significant increase in the probability or consequences of an accident previously evaluated.

Implementation of this method does not affect the integrity of the fission product barriers utilized for mitigation of radiological dose consequences as a result of an accident. Plant response as modeled in the safety analyses is unaffected. Hence, the releases used as input to the dose calculations are unchanged from those previously assumed.

Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the proposed amendment create the possibility of a new or different kind of accident from any accident previously evaluated?

   Response: No.

   The proposed alternative methods do not affect accident initiation sequences or response scenarios as modeled in the safety analyses. This method will not create a new failure scenario. In addition, no new failure modes are being created for any plant equipment. The proposed alternative methods have been designed to applicable regulatory and industry standards. Fault conditions, failure detection, reliability and equipment qualification have been considered. The new methods do not result in any new or different accident scenarios. The types of accidents defined in the UFSAR continue to represent the credible spectrum of events to be analyzed which determine safe plant operation.

   Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

3. Does the proposed amendment involve a significant reduction in a margin of safety?

   Response: No.

   No safety analyses were changed or modified as a result of the proposed TS changes. The proposed change does not alter the manner in which safety limits, limiting safety system settings, or limiting conditions for operation are determined. Margins associated with the current safety analyses acceptance criteria are unaffected. The current safety analyses remain bounding since their conclusions are not affected by the new method.

   Therefore, the proposed change does not result in a significant reduction in a margin of safety.

4.4 Conclusions

In conclusion, based on the considerations discussed above, (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

5.0 ENVIRONMENTAL CONSIDERATION

A review has determined that the proposed amendment would change a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, or would change an inspection or surveillance requirement. However, the proposed amendment does not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluent that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

6.0 REFERENCES

1. Letter from D. Render (U.S. Nuclear Regulatory Commission) to B. Hanson (Exelon Generation Company, LLC), "R. E. Ginna Nuclear Power Plant - Issuance of Amendment Regarding Transition to a Risk Informed, Performance-Based Fire Protection Program in Accordance with the Title 10 of the Code of Federal Regulations Section 50.48(c) (CAC No. MF1393)," dated November 23, 2015 (ML13093A064).

2. Letter from J. Pacher (Constellation Energy Nuclear Group) to U.S. Nuclear Regulatory Commission, "License Amendment Request Pursuant to 10 CFR 50.90: Adoption of NFPA 805, Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants (2001 Edition)," dated March 28, 2013 (ML13093A064).

3. Letter from J. Maier (Rochester Gas and Electric) to D. Crutchfield (U.S. Nuclear Regulatory Commission), "Unresolved Safety Issue Status," dated October 19, 1982 (ML17256A352).

Attachment 2
Markup of Proposed Technical Specifications Pages

TECHNICAL SPECIFICATIONS PAGES

3.4.7-1 (information only)
3.4.7-2
3.4.8-1
3.4.8-2 (information only)
3.9.4-1
3.9.4-2 (information only)
3.9.5-1
3.9.5-2 (information only)

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.7 RCS Loops - MODE 5, Loops Filled

LCO 3.4.7 One residual heat removal (RHR) loop shall be OPERABLE and in operation, and either:

a. One additional RHR loop shall be OPERABLE; or

b. The secondary side water level of at least one steam generator (SG) shall be ≥ 16%.

- NOTE -

1. The RHR pump of the loop in operation may be de-energized for ≤ 1 hour per 8 hour period provided:

    a. No operations are permitted that would cause introduction of coolant into the RCS with boron concentration less than required to meet the SDM of LCO 3.1.1; and

    b. Core outlet temperature is maintained at least 10°F below saturation temperature.

2. One required RHR loop may be inoperable for ≤ 2 hours for surveillance testing provided that the other RHR loop is OPERABLE and in operation.

3. No reactor coolant pump shall be started with one or more RCS cold leg temperatures less than or equal to the LTOP enable temperature specified in the PTLR unless:

    a. The secondary side water temperature of each SG is ≤ 50°F above each of the RCS cold leg temperatures; or

    b. The pressurizer water volume is < 324 cubic feet (38% level).

4. All RHR loops may be removed from operation during planned heatup to MODE 4 when at least one RCS loop is in operation.

APPLICABILITY: MODE 5 with RCS loops filled.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One RHR loop * inoperable. AND Both SGs secondary side water levels not within limits.	A.1 Initiate action to restore a second RHR loop to OPERABLE status. OR A.2 Initiate action to restore required SG secondary side water levels to within limits.	Immediately Immediately
B. Both RHR loops * inoperable. OR No RHR loop in * operation.	B.1 Suspend operations that would cause introduction of coolant into the RCS with boron concentration less than required to meet the SDM of LCO 3.1.1. AND B.2 Initiate action to restore one RHR loop to OPERABLE status and operation.	Immediately Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.7.1	Verify one RHR loop is in operation.	In accordance with the Surveillance Frequency Control Program
SR 3.4.7.2	Verify SG secondary side water level is ≥ 16% in the required SG.	In accordance with the Surveillance Frequency Control Program
SR 3.4.7.3	Verify correct breaker alignment and indicated power are available to the required RHR pump that is not in operation.	In accordance with the Surveillance Frequency Control Program
SR 3.4.7.4	Verify required RHR loop locations susceptible to gas accumulation are sufficiently filled with water.	In accordance with the Surveillance Frequency Control Program

* Beginning April 1, 2020, an alternate means of RHR may be provided until June 30, 2020.

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.8 RCS Loops - MODE 5, Loops Not Filled

LCO 3.4.8 Two residual heat removal (RHR) loops shall be OPERABLE and one RHR loop shall be in operation.

- NOTE -

1. All RHR pumps may be de-energized for ≤ 15 minutes when switching from one loop to another provided:

   a. No operations are permitted that would cause introduction of coolant into the RCS with boron concentration less than required to meet the SDM of LCO 3.1.1; and

   b. Core outlet temperature is maintained at least 10°F below saturation temperature; and

   c. No draining operations to further reduce the RCS water volume are permitted.

2. One RHR loop may be inoperable for ≤ 2 hours for surveillance testing provided that the other RHR loop is OPERABLE and in operation.

APPLICABILITY: MODE 5 with RCS loops not filled.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One RHR loop inoperable. *	A.1 Initiate action to restore RHR loop to OPERABLE status.	Immediately
B. Both RHR loops inoperable. * <u>OR</u> No RHR loop in operation. *	B.1 Suspend operations that would cause introduction of coolant into the RCS with boron concentration less than required to meet the SDM of LCO 3.1.1. <u>AND</u>	Immediately

* Beginning April 1, 2020, an alternate means of RHR may be provided until June 30, 2020.

CONDITION	REQUIRED ACTION		COMPLETION TIME
	B.2	Initiate action to restore one RHR loop to OPERABLE status and operation.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.8.1	Verify one RHR loop is in operation.	In accordance with the Surveillance Frequency Control Program
SR 3.4.8.2	Verify correct breaker alignment and indicated power are available to the RHR pump that is not in operation.	In accordance with the Surveillance Frequency Control Program
SR 3.4.8.3	Verify RHR loop locations susceptible to gas accumulation are sufficiently filled with water.	In accordance with the Surveillance Frequency Control Program

3.9 REFUELING OPERATIONS

3.9.4 Residual Heat Removal (RHR) and Coolant Circulation - Water Level ≥ 23 Ft

LCO 3.9.4 One RHR loop shall be OPERABLE and in operation.

- NOTE -

The required RHR loop may be removed from operation for ≤ 1 hour per 8 hour period, provided no operations are permitted that would cause introduction of coolant into the Reactor Coolant System (RCS) with boron concentration less than that required to meet the minimum required boron concentration of LCO 3.9.1.

APPLICABILITY: MODE 6 with the water level ≥ 23 ft above the top of reactor vessel flange.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. RHR loop requirements not met. *	A.1	Suspend operations that would cause introduction of coolant into the RCS with boron concentration less than required to meet the boron concentration of LCO 3.9.1.	Immediately
	<u>AND</u>		
	A.2	Suspend loading irradiated fuel assemblies in the core.	Immediately
	<u>AND</u>		
	A.3	Initiate action to satisfy RHR loop requirements.	Immediately
	<u>AND</u>		

* Beginning April 1, 2020, an alternate means of RHR may be provided until June 30, 2020.

CONDITION	REQUIRED ACTION		COMPLETION TIME
	A.4	Close all containment penetrations providing direct access from containment atmosphere to outside atmosphere.	4 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.4.1	Verify one RHR loop is in operation and circulating reactor coolant.	In accordance with the Surveillance Frequency Control Program
SR 3.9.4.2	Verify required RHR loop locations susceptible to gas accumulation are sufficiently filled with water.	In accordance with the Surveillance Frequency Control Program

3.9 REFUELING OPERATIONS

3.9.5 Residual Heat Removal (RHR) and Coolant Circulation - Water Level < 23 Ft

LCO 3.9.5 Two RHR loops shall be OPERABLE, and one RHR loop shall be in operation.

APPLICABILITY: MODE 6 with the water level < 23 ft above the top of reactor vessel flange.

ACTIONS

CONDITION		REQUIRED ACTION		COMPLETION TIME
A.	Less than the required number of RHR loops OPERABLE. *	A.1	Initiate action to restore RHR loop(s) to OPERABLE status.	Immediately
		<u>OR</u>		
		A.2	Initiate action to establish ≥ 23 ft of water above the top of reactor vessel flange.	Immediately
B.	No RHR loop in operation. *	B.1	Suspend operations that would cause introduction of coolant into the RCS with boron concentration less than required to meet the boron concentration of LCO 3.9.1.	Immediately
		<u>AND</u>		
		B.2	Initiate action to restore one RHR loop to operation.	Immediately
		<u>AND</u>		
		B.3	Close all containment penetrations providing direct access from containment to outside atmosphere.	4 hours

* Beginning April 1, 2020, an alternate means of RHR may be provided until June 30, 2020.

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.5.1	Verify one RHR loop is in operation and circulating reactor coolant.	In accordance with the Surveillance Frequency Control Program
SR 3.9.5.2	Verify correct breaker alignment and indicated power available to the required RHR pump that is not in operation.	In accordance with the Surveillance Frequency Control Program
SR 3.9.5.3	Verify RHR loop locations susceptible to gas accumulation are sufficiently filled with water.	In accordance with the Surveillance Frequency Control Program