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

10 CFR 50.90

SUSQUEHANNA STEAM ELECTRIC STATION PROPOSED AMENDMENT TO LICENSES NPF-14 AND NPF-22: REVISE REACTOR STEAM DOME PRESSURE – LOW INSTRUMENTATION FUNCTION ALLOWABLE VALUE
PLA-7950

Docket No. 50-387 and 50-388

Pursuant to 10 CFR 50.90, Susquehanna Nuclear, LLC (Susquehanna), is submitting a request for an amendment to the Technical Specifications (TS) for the Susquehanna Steam Electric Station (SSES), Units 1 and 2, Facility Operating License numbers NPF-14 and NPF-22. The proposed amendment would modify TS 3.3.5.1, Emergency Core Cooling Systems (ECCS) Instrumentation.

The proposed amendment would modify the TS Allowable Values (AVs) for the ECCS Instrumentation, Core Spray and Low Pressure Coolant Injection Reactor Steam Dome Pressure – Low Instrumentation Functions 1.c, 1.d, 2.c, and 2.d, in TS Table 3.3.5.1-1. The proposed amendment is necessary to resolve issues associated with setpoint drift of the installed pressure transmitters which measure the Reactor Steam Dome Pressure, thereby reducing the likelihood of the need to perform an unnecessary shutdown due to compliance with the TS which is not warranted for the plant configuration. The revision to the AV does not affect the overall redundancy and diversity of the ECCS Instrumentation and does not deviate from what is assumed in the accident analyses.

Enclosure 1 provides a description and assessment of the proposed changes along with Susquehanna's determination that the proposed changes do not involve a significant hazard consideration. Enclosure 2 provides the existing TS pages marked to show the proposed changes. Enclosure 3 provides revised (clean) TS pages. Enclosure 4 provides the existing TS Bases pages marked up to show the proposed changes and is provided for information only.

Susquehanna requests NRC approval of the proposed changes and issuance of the requested license amendment by October 31, 2022. Once approved, the amendment shall be implemented within 90 days.

In accordance with 10 CFR 50.91, Susquehanna is providing a copy of this application, with enclosures, to the designated Commonwealth of Pennsylvania state official.

Both the Plant Operations Review Committee and the Nuclear Safety Review Board have reviewed the proposed changes.

There are no new or revised regulatory commitments contained in this submittal.

Should you have any questions regarding this submittal, please contact Ms. Melisa Krick, Manager – Nuclear Regulatory Affairs, at (570) 542-1818.

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

Executed on October 5, 2021.

K. Cimorelli

Enclosures:

1. Description and Assessment
2. Marked-Up Technical Specification Pages
3. Revised (Clean) Technical Specification Pages
4. Marked-Up Technical Specification Bases Pages (Provided for Information Only)

Copy: NRC Region I
Mr. C. Highley, NRC Senior Resident Inspector
Ms. A. Klett, NRC Project Manager
Mr. M. Shields, PA DEP/BRP

Enclosure 1 to PLA-7950

Description and Assessment

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SUSQUEHANNA ASSESSMENT

1. Summary Description

Pursuant to 10 CFR 50.90, Susquehanna Nuclear, LLC (Susquehanna), is submitting a request for an amendment to the Technical Specifications (TS) for the Susquehanna Steam Electric Station (SSES), Units 1 and 2, Facility Operating License numbers NPF-14 and NPF-22. The proposed amendment would modify TS 3.3.5.1, Emergency Core Cooling Systems (ECCS) Instrumentation.

The proposed amendment would modify the TS Allowable Values (AVs) for the ECCS Instrumentation, Core Spray and Low Pressure Coolant Injection (LPCI) Reactor Steam Dome Pressure – Low Instrumentation Functions 1.c, 1.d, 2.c, and 2.d, in TS Table 3.3.5.1-1. The proposed amendment is necessary to resolve issues associated with setpoint drift of the installed pressure transmitters which measure the Reactor Steam Dome Pressure, thereby reducing the likelihood of the need to perform an unnecessary shutdown due to compliance with the TS which is not warranted for the plant configuration. The revision to the AV does not affect the overall redundancy and diversity of the ECCS Instrumentation and does not deviate from what is assumed in the accident analyses.

2. Detailed Description

2.1 System Design and Operation

For both Core Spray and LPCI, the Reactor Steam Dome Pressure—Low signals are initiated from four pressure instruments that sense the reactor dome pressure. Four channels of Reactor Steam Dome Pressure – Low Function are required to be operable only when the ECCS is required to be operable to ensure that no single instrument failure can preclude ECCS initiation.

The pressure instruments are set to actuate between the upper and lower AVs on decreasing reactor steam dome pressure. The upper AV is low enough to ensure that the reactor steam dome pressure has fallen to a value below the Core Spray and LPCI maximum design pressures to preclude piping over pressurization. The lower AV is high enough to ensure that the ECCS injection prevents the fuel peak cladding temperature (PCT) from exceeding the limits of 10 CFR 50.46.

Reactor Steam Dome Pressure – Low signals are used as permissives for the low pressure ECCS subsystems (i.e., Core Spray and LPCI). The low reactor pressure permissive is provided to prevent a high drywell pressure condition which is not accompanied by low reactor pressure, i.e., a false LOCA signal. The low reactor steam dome pressure permissive also ensures that, prior to opening the injection valves of the low pressure ECCS subsystems, the reactor pressure has fallen to a value below these subsystems' maximum design pressure. The Reactor Steam Dome Pressure—Low is one of the Functions assumed to be operable and capable of permitting initiation of the ECCS during the transients analyzed in SSES Updated Final Safety Analysis Report (FSAR), Chapter 15. In addition, the Reactor Steam Dome Pressure—Low Function is directly assumed in the analysis of the recirculation line break. The core cooling function of the ECCS, along with the scram action of the Reactor Protection System, ensures that the fuel PCT remains below the limits of 10 CFR 50.46.

2.2 Current Technical Specifications Requirements

Limiting Condition for Operation (LCO) 3.3.5.1 requires, for each unit, that the ECCS instrumentation for each Function in Table 3.3.5.1-1 be operable in the modes or plant conditions specified within the Table for the instrumentation. As specified in Table 3.3.5.1-1, instrumentation Functions 1.c, 1.d, 2.c, and 2.d each require that four channels be operable in Modes 1, 2, and 3. In order for Functions 1.c, 1.d, 2.c, and 2.d to be operable, their setpoints must be within the AV of ≥ 407 psig (lower) and ≤ 433 psig (upper) (the AV range is equivalent for all four functions).

2.3 Reason for the Proposed Change

The Reactor Steam Dome Pressure instruments that feed into the Core Spray and LPCI initiation logic have a history of setpoint drift at SSES. In 2017, Susquehanna replaced all pressure switches with Cameron-Barton 288A switches, and the setpoint drift issues persisted. Extensive causal analyses, troubleshooting, and testing were performed by Susquehanna, Cameron-Barton, and General Electric personnel. It was ultimately determined that the setpoint drift is caused by temperature and humidity changes within the reactor building.

In support of the transition to ATRIUM 11 fuel, the design basis accident analyses have been re-performed and demonstrated that a lower analytical limit is justified; thus, a decreased lower AV is also justified. Further, the Core Spray and LPCI system designs inherently provide overpressure protection irrespective of the upper AV setting (see Section 3.1 for additional discussion). By decreasing the overly conservative lower AV and eliminating the unnecessary upper AV, the instrument setpoints can be adjusted to account for expected instrument drift due to environmental conditions, mainly temperature and relative humidity. The impact of environmental conditions on instrument drift is included in the revised setpoint calculation using data provided by Cameron-Barton. The proposed change will reduce the likelihood of the need to perform a plant shutdown due to compliance with the TS which is not warranted for the plant configuration.

2.4 Description of the Proposed Change

Two changes are proposed to the AVs for Functions 1.c, 1.d, 2.c, and 2.d in TS Table 3.3.5.1-1. First, the upper AV (i.e., ≤ 433 psig) is deleted. Second, the lower AV (i.e., ≥ 407 psig) is revised to require the AV be greater than or equal to 387 psig. The proposed TS pages are provided in Enclosures 2 and 3. Conforming changes are made to the TS Bases, and are provided in Enclosure 4 for information only.

3. Technical Evaluation

3.1 Deletion of Upper Allowable Value

In NUREG-0931 (Reference 1) and NUREG-1042 (Reference 2), the NRC issued the initial TS for SSES, Units 1 and 2, respectively. In Table 3.3.3-2 of References 1 and 2, only a lower AV of ≥ 416 psig was established for the Reactor Vessel Steam Dome Pressure – Low Functions of Core Spray and LPCI. The AV for these functions remained unchanged through the conversion of the SSES, Units 1 and 2 TS to a format consistent with the Improved Standard TS (ISTS) in NUREG-1433, Revision 1 (Reference 3), which was approved in Reference 4.

Subsequently, Susquehanna submitted license amendment requests (approved in References 5 and 6 for Units 2 and 1, respectively) to implement an upper AV and modify the lower AV to its current value of 407 psig. During the application process for these amendments, Susquehanna stated the incorporation of the upper AV was required to provide over pressurization protection for the Core Spray and LPCI system piping.

3.1.1 Core Spray System Design

The Core Spray System is a low pressure injection system connected to the reactor vessel. See FSAR Figure 6.3-4 for the piping and instrumentation diagram (P&ID) for the Core Spray System. The Core Spray piping is rated for high pressure (approximately 1230 psig) until the high/low pressure interface. Core Spray Injection Valve HV152F005A for Unit 1, Division 1, Valve HV152F005B for Unit 1, Division 2, Valve HV252F005A for Unit 2, Division 1, and Valve HV252F005B for Unit 2, Division 2 (hereafter referred to collectively as HV1(2)52F005A/B with a similar convention used for all identified components) are normally closed and provide overpressure protection to the low pressure piping which has a design pressure of 500 psig. Core Spray Injection Valve HV1(2)52F005A/B is normally closed and receives an open signal when the system pressure is less than or equal to 433 psig (i.e., the TS upper AV).

The Core Spray System contains Swing Check Valve HV1(2)52F006A/B on the high pressure side of the piping to provide protection of the low pressure Core Spray System piping. Once the check valve is closed, the differential pressure between the high and low side piping ensures that the low pressure piping is protected since the check valve will remain closed. The check valve cannot open until the low side piping pressure is higher than the high side (reactor) pressure. This check valve provides adequate single failure proof pressure protection of the low pressure piping. There is also a one inch equalizing line around the check valve which contains Isolation Valve HV1(2)52F037A/B. The equalizing line isolation valve is normally closed with its handwheel removed to prevent any override of a containment isolation signal. These Core Spray System valves provide adequate protection against over-pressurization of the Core Spray System.

The low pressure piping also contains Pressure Relief Valve PSV1(2)52F012A/B to prevent over-pressurization of the piping system. The pressure relief valve has a setpoint that matches the design pressure of the piping per the American Society of Mechanical Engineers (ASME) Code and provides additional protection of the Core Spray piping. Between the presence of the check valve, check valve bypass line protection, and pressure relief valve, the Core Spray System will not be over-pressured during any design basis event with single failure even without crediting the pressure permissive on Core Spray Injection Valve HV1(2)52F005A/B currently required by TS 3.3.5.1.

During plant start-up, general operating procedures require confirmation that the check valves are confirmed closed ensuring protection of the low-pressure piping. In addition to this testing, the check valves, equalizing line isolation valves, and pressure relief valves are tested in accordance with the Inservice Testing (IST) Program.

The high pressure setpoint that is being removed from the TS is not being physically removed or deleted from the plant. The design feature of the setpoint will be maintained and still protect the Core Spray System from over-pressurization consistent with a defense in depth philosophy, but it is not explicitly credited in the plant safety analyses and therefore can be removed from the plant TS.

3.1.2 LPCI System Design

The Residual Heat Removal (RHR) System has multiple functions at SSES (e.g., suppression pool cooling, drywell spray, reactor vessel head spray, fuel pool cooling and emergency low pressure injection). During emergency operation of RHR in the LPCI mode, coolant injection occurs into the reactor vessel. See FSAR Figures 5.4-13-1 through 5.4-13-4 for P&IDs for the RHR System. The RHR LPCI piping is rated for high pressure (approximately 1500 psig) until the high/low pressure interface. RHR LPCI Injection Valve HV1(2)51F015A/B is normally closed and provides overpressure protection to the low pressure piping which has a design pressure of 450 psig. RHR LPCI Injection Valve HV1(2)51F015A/B is normally closed and receives an open signal when the system pressure is less than or equal to 433 psig (i.e., the TS upper AV).

The RHR LPCI piping contains Swing Check Valve HV1(2)51F050A/B on the high pressure side of the piping to provide protection of the low pressure RHR LPCI System piping. Once the check valve is closed, the differential pressure between the high and low side piping ensures that the low pressure piping is protected since the check valve will remain closed. The check valve cannot open until the low side piping pressure is higher than the high side (reactor) pressure. This check valve provides adequate single failure proof pressure protection of the low pressure piping. There is also a one inch equalizing line around the check valve which contains Isolation Valve HV1(2)51F122A/B. The equalizing line isolation valve is normally closed with its handwheel removed to prevent any override of a containment isolation signal. These RHR LPCI System valves provide adequate protection against over-pressurization of the RHR LPCI System.

The low pressure piping also contains Pressure Relief Valve PSV1(2)51F025A/B to prevent over-pressurization of the piping system. The pressure relief valve has a setpoint that matches the design pressure of the piping per the ASME Code and provides additional protection of the RHR LPCI piping. Between the presence of the check valve, check valve bypass line protection, and pressure relief valve, the RHR LPCI System piping will not be over-pressured during any design basis event with single failure even without crediting the pressure permissive on RHR LPCI Injection Valve HV1(2)51F015A/B currently required by TS 3.3.5.1.

During plant start-up, general operating procedures require confirmation that the check valves are confirmed closed ensuring protection of the low-pressure piping. In addition to this testing, the check valves, equalizing line isolation valves, and pressure relief valves are tested in accordance with the IST Program.

The high pressure setpoint that is being removed from the TS is not being physically removed or deleted from the plant. The design feature of the setpoint will be maintained and still protect the RHR LPCI System piping from over-pressurization consistent with a defense in depth philosophy but it is not explicitly credited in the plant safety analyses and therefore can be removed from the plant TS.

3.2 Revision to Lower Allowable Value

3.2.1 <u>Revised Analytical Limit</u>

The low pressure permissive for Core Spray and RHR is set to ensure that injection of the Core Spray and LPCI Systems occurs within the assumed reactor pressure in the FSAR Chapter 15 transient analyses. Opening of these valves and initiation of the pumps at values greater than that assumed in the design basis accident analyses ensures that the reactor fuel PCT remains within regulatory limits. The proposed change will reduce the lower AV from ≥ 407 psig to ≥ 387 psig.

Susquehanna is introducing a new fuel type, ATRIUM-11, into the SSES Unit 1 and 2 reactor cores in reload quantities. Therefore, there will be a time period that both ATRIUM-11 (new fuel type) and ATRIUM-10 (previous fuel type) will reside in the core. An initial batch of ATRIUM-11 fuel was loaded into the Unit 2 reactor in spring 2021 and ATRIUM-11 fuel will be initially loaded into the Unit 1 reactor during refueling outage scheduled to begin in spring 2022. Because ATRIUM-10 and ATRIUM-11 fuel are co-resident in the SSES reactor cores, the impact of the setpoint change on the FSAR Chapter 15 transient analyses for both the ATRIUM-11 and ATRIUM-10 fuel types are discussed.

3.2.1.1 ATRIUM-11

Enclosure 15a to Reference 7 provided Framatome Report ANP-3784P, Revision 0, "Susquehanna Units 1 and 2 LOCA Analysis for ATRIUM 11 Fuel." As stated in Tables 4.5 and 4.6 of ANP-3784P, the assumed reactor pressure permissive for opening valves in the LPCI and Core Spray Systems was 380 psig. Further, as stated in Section 2.0 of ANP-3784P, the limiting PCT for ATRIUM-11 fuel was calculated to be 1784°F, which is substantially lower than the limit of 2200°F in 10 CFR 50.46(b)(1).[1] The NRC approved use of Advanced Framatome Methodologies at SSES, including those methods discussed in ANP-3784P, in Reference 8. Since the FSAR Chapter 15 transient analyses for ATRIUM-11 fuel were performed using an analytical limit of 380 psig, as approved by the NRC in Reference 8, an analytical limit of 380 psig for Functions 1.c, 1.d, 2.c, and 2.d in TS Table 3.3.5.1-1 is justified for ATRIUM-11 fuel.

3.2.1.2 ATRIUM-10

As stated above, Susquehanna is transitioning from ATRIUM-10 to ATRIUM-11 fuel in reload quantities. The limiting PCT for ATRIUM-10 fuel (i.e., for a fresh assembly) was calculated to be 1876°F when the assumed reactor pressure permissive for opening valves in the LPCI and Core Spray Systems was 380 psig. There is significant margin to the PCT limit of 2200°F in 10 CFR 50.46(b)(1). Additionally, upon introduction of ATRIUM-11 fuel into the SSES Unit 1 and 2 reactor cores, no new ATRIUM-10 fuel is expected to be loaded into the cores. Thus, all remaining ATRIUM-10 fuel in the reactor cores will have exposures consistent with at least one cycle of operation. Since the "once burned" fuel will produce less energy than new fuel bundles, the Linear Heat Generation Rate will be lower for the ATRIUM-10 fuel than that assumed in the design analysis. Since the heat generation within an individual fuel bundle or rod is lower, the calculated PCT will be lower than the results calculated for a fresh ATRIUM-10 assembly. This provides further margin between the actual expected PCT for a once-burned or twice-burned ATRIUM-10 fuel assembly (i.e., less than 1876°F) and the PCT limit of 2200°F in 10 CFR 50.46(b)(1). Since there is adequate margin, an analytical limit of 380 psig for Functions 1.c, 1.d, 2.c, and 2.d in TS Table 3.3.5.1-1 is justified for ATRIUM-10 fuel.

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[1] In Reference 12, Susquehanna submitted the annual report of changes to evaluation models pursuant to 10 CFR 50.46. Due to errors and corrections identified therein, the limiting PCT for ATRIUM 11 fuel has been calculated to be 1785°F.

3.2.2 Revised Allowable Value

Using the revised analytical limit of 380 psig as described above, Susquehanna calculated the revised AV and Nominal Trip Setpoint (NTSP) using General Electric NEDC-31336P-A (Reference 9). This NRC-approved topical report ensures that NTSPs and AVs are determined using consistent methods and provides the controls to ensure that the calculations and basis for these values are documented and retrievable. The same methodology was used to determine the current TS AVs as approved in References 5 and 6 for Units 2 and 1, respectively. The proposed change lowers the analytical limit for Functions 1.c, 1.d, 2.c, and 2.d in TS Table 3.3.5.1-1. No field changes will be made to implement the requested change; i.e., the same instruments and calibration procedures will be used. Therefore, the previously established instrument, loop and calibration accuracies, and the calculated instrument drift will be the same as those used for the current TS setpoints and are simply applied to the new analytical limit. Table 1 displays the changes to the calculated parameters.

Table 1 – Changes to Calculated Parameters

Parameter	Current (psig)	Proposed (psig)
Upper Analytical Limit	440	NONE
Upper Allowable Value	433	NONE
Upper Nominal Trip Setpoint	427	NONE
Setpoint	420	420
Lower Nominal Trip Setpoint	413	393
Lower Allowable Value	407	387
Lower Analytical Limit	400	380

The setpoint will remain unchanged at 420 psig. This provides additional margin to the TS AV while also ensuring that the upper AV, which is requested to be removed from the TS, continues to be met.

4. Regulatory Evaluation

4.1 Applicable Regulatory Requirements/Criteria

General Design Criteria

During the applicable period of this proposed license amendment, SSES will maintain the ability to meet the applicable General Design Criteria (GDC) as described in FSAR Section 3.1.

GDC-13, Instrumentation and Control

Instrumentation shall be provided to monitor variables and systems over their anticipated ranges for normal operation, for anticipated operational occurrences, and for accident conditions as appropriate to assure adequate safety, including those variables and systems that can affect the fission process, the integrity of the reactor core, the reactor coolant pressure boundary, and the containment and its associated systems. Appropriate controls shall be provided to maintain these variables and systems within prescribed operating ranges.

GDC-20, Protection System Functions

The protection system shall be designed (1) to initiate automatically the operation of appropriate systems including the reactivity control systems, to assure that specified acceptable fuel design limits are not exceeded as a result of anticipated operational occurrences and (2) to sense accident conditions and to initiate the operation of systems and components important to safety.

GDC-35, Emergency Core Cooling

A system to provide abundant emergency core cooling shall be provided. The system safety function shall be to transfer heat from the reactor core following any loss of reactor coolant at a rate such that (1) fuel and clad damage that could interfere with continued effective core cooling is prevented and (2) clad metal-water reaction is limited to negligible amounts.

Suitable redundancy in components and features, and suitable interconnections, leak detection, isolation, and containment 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.

Conclusion

The proposed change to the TS will not permanently impact any installed components at SSES. The design of the Core Spray and LPCI Systems will continue to inherently provide overpressurization protection for the system piping by way of the installed check valves. Thus, the Reactor Steam Dome Pressure – Low Functions are not impacted by the proposed change, and the Core Spray and LPCI systems will continue to be able to perform their intended function to mitigate the consequences of an event.

4.2 Precedent

In Reference 10, the NRC granted approval to Detroit Edison for the conversion of the TS of Fermi-2 to a format consistent with the ISTS in NUREG-1433. With that approval, the Fermi-2 TS retained only a lower AV for Reactor Steam Dome Pressure – Low instrumentation functions in TS Table 3.3.5.1-1. During the application process, in Reference 11, Detroit Edison stated an upper AV was not required. Fermi-2 is a Boiling Water Reactor 4 (BWR/4) design, consistent with SSES, Units 1 and 2. Thus, the design and function of the Core Spray and LPCI systems (including the initiation logic) are substantially similar at the two sites. Further, as described in Section 3 of this Enclosure, the Core Spray and LPCI system design inherently prevents overpressurization of the system piping and an upper AV is not required. Therefore, the approval for the Fermi-2 TS is applicable to the SSES, Units 1 and 2, TS.

4.3 No Significant Hazards Considerations Analysis

In accordance with the requirements of 10 CFR 50.90, Susquehanna Nuclear, LLC (Susquehanna), requests an amendment to the Technical Specifications (TS) for the Susquehanna Steam Electric Station (SSES), Units 1 and 2. The proposed amendment would modify the TS Allowable Values (AVs) for the Emergency Core Cooling System (ECCS) Instrumentation, Core Spray and Low Pressure Coolant Injection (LPCI) Reactor Steam Dome Pressure – Low Instrumentation Functions 1.c, 1.d, 2.c, and 2.d, in TS Table 3.3.5.1-1. The proposed amendment is necessary to resolve issues associated with setpoint drift of the installed pressure transmitters which measure the Reactor Steam Dome Pressure, thereby reducing the likelihood of the need to perform an unnecessary shutdown due to compliance with the TS which is not warranted for the plant configuration.

Susquehanna has evaluated the proposed amendment against the standards in 10 CFR 50.92 and has determined that the operation of SSES in accordance with the proposed amendment presents no significant hazards. Susquehanna's evaluation against each of the criteria in 10 CFR 50.92 follows.

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

   Response: No

   The proposed change will modify the TS AV for Reactor Steam Dome Pressure – Low Instrumentation Functions in Table 3.3.5.1-1. The modified AV continues to ensure Core Spray and LPCI operation prevent the fuel peak cladding temperature from exceeding the limits of 10 CFR 50.46 while the system designs continue to inherently provide overpressure protection for Core Spray and LPCI piping. The proposed change does not modify any structures, systems, or components (SSCs) installed at SSES, nor does it alter the manner in which any SSCs are operated. Thus, the capability of performing the design functions of the Core Spray and LPCI systems are not affected. The revision to the AV does not affect the overall redundancy and diversity of the ECCS Instrumentation and does not deviate from what is assumed in the accident analyses. Thus, the proposed change has no impacts on previously evaluated accidents.

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

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

   Response: No

   The proposed change will modify the TS AV for Reactor Steam Dome Pressure – Low Instrumentation Functions in Table 3.3.5.1-1. The modified AV continues to ensure Core Spray and LPCI operation prevent the fuel peak cladding temperature from exceeding the limits of 10 CFR 50.46 while the system designs continue to inherently provide overpressure protection for Core Spray and LPCI piping. The change does not involve a physical alteration of the plant (i.e., no different SSCs will be installed) or a change in the methods governing normal plant operations. The proposed change does not create any new credible failure mechanisms of the ECCS initiation instrumentation. As such, the proposed change does not introduce new failure mechanisms, malfunctions, or accident initiators not considered in the design and licensing basis.

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

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

   Response: No

   The proposed change will modify the TS AV for Reactor Steam Dome Pressure – Low Instrumentation Functions in Table 3.3.5.1-1. The modified AV continues to ensure Core Spray and LPCI operation prevent the fuel peak cladding temperature from exceeding the limits of 10 CFR 50.46 while the system designs continue to inherently provide overpressure protection for Core Spray and LPCI piping. The proposed change does not alter the manner in which safety limits or limiting conditions for operation are determined. The safety analysis assumptions and acceptance criteria are not affected by this change.

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

Based on the above evaluation, Susquehanna concludes that the proposed amendment does not involve a significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and accordingly, a finding of "no significant hazards consideration" is justified.

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. Environmental Consideration

Susquehanna 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 effluents 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. References

1. NRC NUREG-0931, "Technical Specifications for Susquehanna Steam Electric Station, Unit No. 1 Docket No. 50-387 Appendix 'A' to License No. NPF-14," dated July 1982 (Legacy ADAMS Accession No. 8208040344).

2. NRC NUREG-1042, "Technical Specifications for Susquehanna Steam Electric Station, Unit No. 2 Docket No. 50-388 Appendix 'A' to License No. NPF-22," dated March 1984 (Legacy ADAMS Accession No. 8404100519).

3. NRC NUREG-1433, "Standard Technical Specifications General Electric Plants, BWR/4," Volumes 1 through 3, Revision 1, dated April 1995 (ADAMS Accession Nos. ML13196A477, ML13196A483, and ML13196A485).

4. NRC letter to Susquehanna, "Susquehanna Steam Electric Station, Units 1 and 2 (TAC Nos. M96327 and M96328)," dated July 30, 1998 (ADAMS Accession No. ML010160119).

5. NRC letter to Susquehanna, "Susquehanna Steam Electric Station, Unit 2, Technical Specification Changes on Reactor Steam Dome Pressure – Low Allowable Value (TAC No. MA2447)," dated March 4, 1999 (ADAMS Accession No. ML010160354).

6. NRC letter to Susquehanna, "Susquehanna Steam Electric Station, Unit 1 – Issuance of Amendment Re: Technical Specification Changes on Reactor Steam Dome Pressure – Low Allowable Value (TAC No. MA4982)," dated May 25, 1999 (ADAMS Accession No. ML010160341).

7. Susquehanna letter to NRC, "Proposed Amendment to Licenses NPF-14 and NPF-22: Application of Advanced Framatome Methodologies and TSTF-535 (PLA-7783)," dated July 15, 2019 (ADAMS Accession Nos. ML19196A270 [Non-Proprietary] and ML19196A271 [Proprietary]).

8. NRC letter to Susquehanna, "Issuance of Amendment Nos. 278 and 260 to Allow Application of Advanced Framatome ATRIUM 11 Fuel Methodologies (EPID L-2019-LLA-0153)," dated January 21, 2021 (ADAMS Accession Nos. ML20164A181 [Proprietary] and ML20168B004 [Non-Proprietary]).

9. General Electric Topical Report NEDC-31336P-A, "General Electric Instrument Setpoint Methodology," dated September 1996 (ADAMS Accession Nos. ML072950103 [Proprietary] and ML073450560 [Non-Proprietary]).

10. NRC letter to Detroit Edison, "Fermi 2 – Issuance of Amendment Re: Conversion of Current Technical Specifications to Improved Standard Technical Specifications (TAC No. MA1465)," dated September 30, 1999 (ADAMS Accession Nos. ML20217H401, ML20217H414, and ML20217H431).

11. Detroit Edison letter to NRC, "Transmittal of Revision 6 to Fermi 2 Improved Technical Specification Submittal (TAC No. MA1465)," dated June 2, 1999 (ADAMS Accession No. ML20195G122).

12. Susquehanna letter to NRC, "10 CFR 50.46 Annual Report (PLA-7947)," dated June 10, 2021 (ADAMS Accession No. ML21161A005).

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Marked-Up Technical Specification Pages

Revised Technical Specifications Pages

Unit 1 TS Pages
3.3-42 and 3.3-43

Unit 2 TS Pages
3.3-43 and 3.3-44

Table 3.3.5.1-1 (page 1 of 6)
Emergency Core Cooling System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER FUNCTION	CONDITIONS REFERENCED FROM REQUIRED ACTION A.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Core Spray System					
a. Reactor Vessel Water Level – Low Low Low, Level 1	1, 2, 3	4[(a)]	B	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.4 SR 3.3.5.1.5	≥ -136 inches
b. Drywell Pressure – High	1, 2, 3	4[(a)]	B	SR 3.3.5.1.2 SR 3.3.5.1.3 SR 3.3.5.1.5	≤ 1.88 psig
c. Reactor Steam Dome Pressure – Low (initiation)	1, 2, 3	4	B	SR 3.3.5.1.2 SR 3.3.5.1.3 SR 3.3.5.1.5	≥ ~~407~~ 387 psig ~~(lower)~~ ~~≤ 433 psig (upper)~~
d. Reactor Steam Dome Pressure – Low (injection permissive)	1, 2, 3	4	C	SR 3.3.5.1.2 SR 3.3.5.1.3 SR 3.3.5.1.5	≥ ~~407~~ 387 psig ~~(lower)~~ ~~≤ 433 psig (upper)~~
e. Manual Initiation	1, 2, 3	2 1 per Subsystem	C	SR 3.3.5.1.5	NA
2. Low Pressure Coolant Injection (LPCI) System					
a. Reactor Vessel Water Level – Low Low Low, Level 1	1, 2, 3	4[(b)]	B	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.4 SR 3.3.5.1.5	≥ -136 inches

(a) Also required to initiate the associated diesel generator (DG), initiate Drywell Cooling Equipment Trip, and Emergency Service Water (ESW) Pump timer reset.

(b) Also required to initiate the associated DGs, ESW Pump timer reset and Turbine Building and Reactor Building Chillers trip.

Table 3.3.5.1-1 (page 2 of 6)
Emergency Core Cooling System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER FUNCTION	CONDITIONS REFERENCED FROM REQUIRED ACTION A.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
2. LPCI System (continued)					
b. Drywell Pressure – High	1, 2, 3	4(b)	B	SR 3.3.5.1.2 SR 3.3.5.1.3 SR 3.3.5.1.5	≤ 1.88 psig
c. Reactor Steam Dome Pressure – Low (initiation)	1, 2, 3	4	B	SR 3.3.5.1.2 SR 3.3.5.1.3 SR 3.3.5.1.5	≥ ~~407~~ 387 psig ~~(lower)~~ ~~≤ 433 psig (upper)~~
d. Reactor Steam Dome Pressure – Low (injection permissive)	1, 2, 3	4	C	SR 3.3.5.1.2 SR 3.3.5.1.3 SR 3.3.5.1.5	≥ ~~407~~ 387 psig ~~(lower)~~ ~~≤ 433 psig (upper)~~
e. Reactor Steam Dome Pressure – Low (Recirculation Discharge Valve Permissive)	1(c), 2(c), 3(c)	4	C	SR 3.3.5.1.2 SR 3.3.5.1.3 SR 3.3.5.1.5	≥ 216 psig

(b) Also required to initiate the associated DGs, ESW pump timer reset and Turbine Building and Reactor Building Chiller trip.

(c) With either associated recirculation pump discharge or bypass valves open.

Table 3.3.5.1-1 (page 1 of 5)
Emergency Core Cooling System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER FUNCTION	CONDITIONS REFERENCED FROM REQUIRED ACTION A.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Core Spray System					
a. Reactor Vessel Water Level – Low Low Low, Level 1	1, 2, 3	4[(a)]	B	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.4 SR 3.3.5.1.5	≥ -136 inches
b. Drywell Pressure – High	1, 2, 3	4[(a)]	B	SR 3.3.5.1.2 SR 3.3.5.1.3 SR 3.3.5.1.5	≤ 1.88 psig
c. Reactor Steam Dome Pressure – Low (initiation)	1, 2, 3	4	B	SR 3.3.5.1.2 SR 3.3.5.1.3 SR 3.3.5.1.5	≥ ~~407~~ 387 psig ~~(lower)~~ ~~≤ 433 psig (upper)~~
d. Reactor Steam Dome Pressure – Low (injection permissive)	1, 2, 3	4	C	SR 3.3.5.1.2 SR 3.3.5.1.3 SR 3.3.5.1.5	≥ ~~407~~ 387 psig ~~(lower)~~ ~~≤ 433 psig (upper)~~
e. Manual Initiation	1, 2, 3	2 1 per Subsystem	C	SR 3.3.5.1.5	NA
2. Low Pressure Coolant Injection (LPCI) System					
a. Reactor Vessel Water Level – Low Low Low, Level 1	1, 2, 3	4[(b)]	B	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.4 SR 3.3.5.1.5	≥ -136 inches

(a) Also required to initiate the associated diesel generator (DG), initiate Drywell Cooling Equipment Trip, and Emergency Service Water (ESW) Pump timer reset.

(b) Also required to initiate the associated DGs, ESW Pump timer reset and Turbine Building and Reactor Building Chillers trip.

Table 3.3.5.1-1 (page 2 of 5)
Emergency Core Cooling System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER FUNCTION	CONDITIONS REFERENCED FROM REQUIRED ACTION A.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
2. LPCI System (continued)					
b. Drywell Pressure – High	1, 2, 3	4(b)	B	SR 3.3.5.1.2 SR 3.3.5.1.3 SR 3.3.5.1.5	≤ 1.88 psig
c. Reactor Steam Dome Pressure – Low (initiation)	1, 2, 3	4	B	SR 3.3.5.1.2 SR 3.3.5.1.3 SR 3.3.5.1.5	≥ ~~407~~ 387 psig ~~(lower)~~ ~~≤ 433 psig (upper)~~
d. Reactor Steam Dome Pressure – Low (injection permissive)	1, 2, 3	4	C	SR 3.3.5.1.2 SR 3.3.5.1.3 SR 3.3.5.1.5	≥ ~~407~~ 387 psig ~~(lower)~~ ~~≤ 433 psig (upper)~~
e. Reactor Steam Dome Pressure – Low (Recirculation Discharge Valve Permissive)	1(c), 2(c), 3(c)	4	C	SR 3.3.5.1.2 SR 3.3.5.1.3 SR 3.3.5.1.5	≥ 216 psig
f. Manual Initiation	1, 2, 3	2 1 per Subsystem	C	SR 3.3.5.1.5	NA

(b) Also required to initiate the associated DGs, ESW pump timer reset and Turbine Building and Reactor Building Chiller trip.

(c) With either associated recirculation pump discharge or bypass valves open.

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Revised (Clean) Technical Specification Pages

Revised Technical Specifications Pages

Unit 1 TS Pages
3.3-42 and 3.3-43

Unit 2 TS Pages
3.3-43 and 3.3-44

Table 3.3.5.1-1 (page 1 of 6)
Emergency Core Cooling System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER FUNCTION	CONDITIONS REFERENCED FROM REQUIRED ACTION A.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Core Spray System					
a. Reactor Vessel Water Level – Low Low Low, Level 1	1, 2, 3	4[(a)]	B	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.4 SR 3.3.5.1.5	≥ -136 inches
b. Drywell Pressure – High	1, 2, 3	4[(a)]	B	SR 3.3.5.1.2 SR 3.3.5.1.3 SR 3.3.5.1.5	≤ 1.88 psig
c. Reactor Steam Dome Pressure – Low (initiation)	1, 2, 3	4	B	SR 3.3.5.1.2 SR 3.3.5.1.3 SR 3.3.5.1.5	≥ 387 psig
d. Reactor Steam Dome Pressure – Low (injection permissive)	1, 2, 3	4	C	SR 3.3.5.1.2 SR 3.3.5.1.3 SR 3.3.5.1.5	≥ 387 psig
e. Manual Initiation	1, 2, 3	2 1 per Subsystem	C	SR 3.3.5.1.5	NA
2. Low Pressure Coolant Injection (LPCI) System					
a. Reactor Vessel Water Level – Low Low Low, Level 1	1, 2, 3	4[(b)]	B	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.4 SR 3.3.5.1.5	≥ -136 inches

(a) Also required to initiate the associated diesel generator (DG), initiate Drywell Cooling Equipment Trip, and Emergency Service Water (ESW) Pump timer reset.

(b) Also required to initiate the associated DGs, ESW Pump timer reset and Turbine Building and Reactor Building Chillers trip.

Table 3.3.5.1-1 (page 2 of 6)
Emergency Core Cooling System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER FUNCTION	CONDITIONS REFERENCED FROM REQUIRED ACTION A.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
2. LPCI System (continued)					
b. Drywell Pressure – High	1, 2, 3	4[(b)]	B	SR 3.3.5.1.2 SR 3.3.5.1.3 SR 3.3.5.1.5	≤ 1.88 psig
c. Reactor Steam Dome Pressure – Low (initiation)	1, 2, 3	4	B	SR 3.3.5.1.2 SR 3.3.5.1.3 SR 3.3.5.1.5	≥ 387 psig
d. Reactor Steam Dome Pressure – Low (injection permissive)	1, 2, 3	4	C	SR 3.3.5.1.2 SR 3.3.5.1.3 SR 3.3.5.1.5	≥ 387 psig
e. Reactor Steam Dome Pressure – Low (Recirculation Discharge Valve Permissive)	1[(c)], 2[(c)], 3[(c)]	4	C	SR 3.3.5.1.2 SR 3.3.5.1.3 SR 3.3.5.1.5	≥ 216 psig

(b) Also required to initiate the associated DGs, ESW pump timer reset and Turbine Building and Reactor Building Chiller trip.

(c) With either associated recirculation pump discharge or bypass valves open.

Table 3.3.5.1-1 (page 1 of 5)
Emergency Core Cooling System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER FUNCTION	CONDITIONS REFERENCED FROM REQUIRED ACTION A.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Core Spray System					
a. Reactor Vessel Water Level – Low Low Low, Level 1	1, 2, 3	4[(a)]	B	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.4 SR 3.3.5.1.5	≥ -136 inches
b. Drywell Pressure – High	1, 2, 3	4[(a)]	B	SR 3.3.5.1.2 SR 3.3.5.1.3 SR 3.3.5.1.5	≤ 1.88 psig
c. Reactor Steam Dome Pressure – Low (initiation)	1, 2, 3	4	B	SR 3.3.5.1.2 SR 3.3.5.1.3 SR 3.3.5.1.5	≥ 387 psig
d. Reactor Steam Dome Pressure – Low (injection permissive)	1, 2, 3	4	C	SR 3.3.5.1.2 SR 3.3.5.1.3 SR 3.3.5.1.5	≥ 387 psig
e. Manual Initiation	1, 2, 3	2 1 per Subsystem	C	SR 3.3.5.1.5	NA
2. Low Pressure Coolant Injection (LPCI) System					
a. Reactor Vessel Water Level – Low Low Low, Level 1	1, 2, 3	4[(b)]	B	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.4 SR 3.3.5.1.5	≥ -136 inches

(a) Also required to initiate the associated diesel generator (DG), initiate Drywell Cooling Equipment Trip, and Emergency Service Water (ESW) Pump timer reset.

(b) Also required to initiate the associated DGs, ESW Pump timer reset and Turbine Building and Reactor Building Chillers trip.

Table 3.3.5.1-1 (page 2 of 5)
Emergency Core Cooling System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER FUNCTION	CONDITIONS REFERENCED FROM REQUIRED ACTION A.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
2. LPCI System (continued)					
b. Drywell Pressure – High	1, 2, 3	4[(b)]	B	SR 3.3.5.1.2 SR 3.3.5.1.3 SR 3.3.5.1.5	≤ 1.88 psig
c. Reactor Steam Dome Pressure – Low (initiation)	1, 2, 3	4	B	SR 3.3.5.1.2 SR 3.3.5.1.3 SR 3.3.5.1.5	≥ 387 psig
d. Reactor Steam Dome Pressure – Low (injection permissive)	1, 2, 3	4	C	SR 3.3.5.1.2 SR 3.3.5.1.3 SR 3.3.5.1.5	≥ 387 psig
e. Reactor Steam Dome Pressure – Low (Recirculation Discharge Valve Permissive)	1[(c)], 2[(c)], 3[(c)]	4	C	SR 3.3.5.1.2 SR 3.3.5.1.3 SR 3.3.5.1.5	≥ 216 psig
f. Manual Initiation	1, 2, 3	2 1 per Subsystem	C	SR 3.3.5.1.5	NA

(b) Also required to initiate the associated DGs, ESW pump timer reset and Turbine Building and Reactor Building Chiller trip.

(c) With either associated recirculation pump discharge or bypass valves open.

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Marked-Up Technical Specification Bases Pages

Revised Technical Specification Bases Pages

Unit 1 TS Bases Pages
3.3-102, 3.3-103, 3.3-110, and 3.3-111

Unit 2 TS Bases Pages
3.3-102, 3.3-103, 3.3-110, and 3.3-111

(Provided for Information Only)

BASES

BACKGROUND (continued)

Core Spray System (continued)

Once an initiation signal is received by the CS control circuitry, the signal is sealed in until manually reset.

The logic can also be initiated by use of a manual push button (one push button per subsystem). Upon receipt of an initiation signal, the CS pumps are started 15 seconds after initiation signal if normal offsite power is available and 10.5 seconds after diesel generator power is available.

The CS test line isolation valve, which is also a primary containment isolation valve (PCIV), is closed on a CS initiation signal to allow full system flow assumed in the accident analyses and maintain primary containment isolated.

The CS System design ensures the system is not subjected to overpressurization events.~~The CS System also monitors the pressure in the reactor to ensure that, before the injection valves open, the reactor pressure has fallen to a value below the CS System's maximum design pressure. The variable is monitored by four redundant instruments. The instrument outputs are connected to relays whose contacts are arranged in a one-out-of-two taken twice logic.~~

Low Pressure Coolant Injection System

The LPCI is an operating mode of the Residual Heat Removal (RHR) System, with two LPCI subsystems. The LPCI subsystems may be initiated by automatic or manual means. Automatic initiation occurs for conditions of Reactor Vessel Water Level Low, Low, Low, Level 1 or Drywell Pressure - High concurrent with Reactor Pressure - Low. Each of these diverse variables is monitored by four instruments in two divisions. Each division is arranged in a one-out-of-two-twice network using level and pressure instruments which will generate a signal when:

(1) both level sensors are tripped, or

(2) two high drywell pressure sensors and two low reactor vessel pressure sensors are tripped, or

(3) a combination of one channel level sensor and one of the other channel of high drywell pressure sensor together with its associated low reactor vessel pressure sensor. (i.e. Channel A level sensor and Channel C high drywell and low reactor vessel pressure sensor).

BASES

BACKGROUND (continued)

Low Pressure Coolant Injection System (continued)

The initiation logic is cross connected between divisions (i.e., either start signal will start all four pumps and open both loop's injection valves). Once an initiation signal is received by the LPCI control circuitry, the signal is sealed in until manually reset. The cross division start signals for the pumps affect both the opposite division's start logic and the pump's 4KV breaker start logic. The cross division start signal to the opposite division's start logic is for improved reliability. The cross division start signals to the pump's 4KV breaker start logic is needed to ensure specific control power failures do not prevent the start of an adequate number of LPCI pumps.

Upon receipt of an initiation signal, all LPCI pumps start after a 3 second time delay when normal AC power is lost and standby diesel generator power is available. If normal power is available, LPCI pumps A and B will start immediately and pumps C and D will start 7.0 seconds after initiation signal to limit loading of the offsite sources.

The RHR test line and spray line are also isolated on a LPCI initiation signal to allow the full system flow assumed in the accident analyses and for those valves which are also PCIVs maintain primary containment isolated.

The LPCI System design ensures the system is not subjected to overpressurization events.~~The LPCI System monitors the pressure in the reactor to ensure that, before an injection valve opens, the reactor pressure has fallen to a value below the LPCI System's maximum design pressure. The variable is monitored by four redundant instruments. The instrument outputs are connected to relays whose contacts are arranged in a one-out-of-two taken twice logic.~~

Logic is provided to close the recirculation pump discharge valves to ensure that LPCI flow does not bypass the core when it injects into the recirculation lines. The logic consists of an initiation signal (Low reactor water level and high drywell pressure in a one out of two taken twice logic) from both divisions of LPCI instruments and a pressure permissive. The pressure variable is monitored by four redundant instruments.

The instrument outputs are connected to relays whose contacts are arranged in a one-out-of-two taken twice logic.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

1.b, 2.b. Drywell Pressure—High

High pressure in the drywell could indicate a break in the reactor coolant pressure boundary (RCPB). The low pressure ECCS (provided a concurrent low reactor pressure signal is present) and associated DGs, without a concurrent low reactor pressure signal, are initiated upon receipt of the Drywell Pressure—High Function in order to minimize the possibility of fuel damage. The Drywell Pressure—High Function, along with the Reactor Water Level—Low Low Low, Level 1 Function, is directly assumed in the analysis of the recirculation line break (Ref. 1). The core cooling function of the ECCS, along with the scram action of the RPS, ensures that the fuel peak cladding temperature remains below the limits of 10 CFR 50.46.

High drywell pressure signals are initiated from four pressure instruments that sense drywell pressure. The Allowable Value was selected to be as low as practical and be indicative of a LOCA inside primary containment. The Drywell Pressure—High Function is required to be OPERABLE when the ECCS or DG is required to be OPERABLE in conjunction with times when the primary containment is required to be OPERABLE. Thus, four channels of the CS and LPCI Drywell Pressure—High Function are required to be OPERABLE in MODES 1, 2, and 3 to ensure that no single instrument failure can preclude ECCS and DG initiation. In MODES 4 and 5, the Drywell Pressure—High Function is not required, since there is insufficient energy in the reactor to pressurize the primary containment to Drywell Pressure—High setpoint. Refer to LCO 3.5.1 for Applicability Bases for the low pressure ECCS subsystems and to LCO 3.8.1 for Applicability Bases for the DGs.

1.c, 1.d, 2.c, 2.d Reactor Steam Dome Pressure—Low

Low reactor steam dome pressure signals are used as permissives for the low pressure ECCS subsystems. The low reactor pressure permissive is provided to prevent a high drywell pressure condition which is not accompanied by low reactor pressure, i.e. a false LOCA signal, from disabling two RHR pumps on the other unit. The low reactor steam dome pressure permissive also ensures that the injection valves open prior to pressure falling below the Allowable Value~~, prior to opening the injection valves of the low pressure ECCS subsystems, the reactor pressure has fallen to a value below these subsystems' maximum design pressure~~. The Reactor Steam Dome Pressure—Low is one of the Functions assumed to be OPERABLE and capable of permitting initiation of the ECCS during the transients analyzed in Reference 2. In addition, the Reactor Steam Dome Pressure—Low Function is directly assumed in the analysis of the recirculation line break (Ref. 1). The core cooling function of the ECCS, along with the scram action of the RPS, ensures that the fuel peak cladding temperature remains below the limits of 10 CFR 50.46.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

1.c, 1.d, 2.c, 2.d Reactor Steam Dome Pressure—Low (continued)

The Reactor Steam Dome Pressure—Low signals are initiated from four pressure instruments that sense the reactor dome pressure.

The pressure instruments are set to actuate at or above~~between the Upper and Lower~~the Allowable Value~~s~~ on decreasing reactor dome pressure.

~~The Upper Allowable Value is low enough to ensure that the reactor dome pressure has fallen to a value below the Core Spray and RHR/LPCI maximum design pressures to preclude piping over pressurization.~~

The ~~Lower~~ Allowable Value is high enough to ensure that the ECCS injection prevents the fuel peak cladding temperature from exceeding the limits of 10 CFR 50.46. The CS and RHR/LPCI system designs ensure the systems are not subjected to overpressurization events.

DGs C and D which are initiated from the LPCI LOCA initiation are cross connected such that both DGs receive an initiation signal from both Divisions of the LPCI LOCA initiation circuitry. This cross connected logic is only required in MODES 1, 2, and 3. In MODES 4 and 5, redundancy in the DG initiation circuitry is not required. Therefore, in MODES 4 and 5 for DGs C and D only one division of ECCS initiation logic is required.

Four channels of Reactor Steam Dome Pressure—Low Function are required to be OPERABLE only when the ECCS is required to be OPERABLE to ensure that no single instrument failure can preclude ECCS initiation.

1.e, 2.f. Manual Initiation

The Manual Initiation push button channels introduce signals into the appropriate ECCS logic to provide manual initiation capability and are redundant to the automatic protective instrumentation. There is one push button for each of the CS and LPCI subsystems (i.e., two for CS and two for LPCI).

The Manual Initiation Function is not assumed in any accident or transient analyses in the FSAR. However, the Function is retained for overall redundancy and diversity of the low pressure ECCS function as required by the NRC in the plant licensing basis.

BASES

BACKGROUND (continued)

Core Spray System (continued)

Once an initiation signal is received by the CS control circuitry, the signal is sealed in until manually reset. The logic can also be initiated by use of a manual push button (one push button per subsystem). Upon receipt of an initiation signal, the CS pumps are started 15 seconds after initiation signal if normal offsite power is available and 10.5 seconds after diesel generator power is available.

The CS test line isolation valve, which is also a primary containment isolation valve (PCIV), is closed on a CS initiation signal to allow full system flow assumed in the accident analyses and maintain primary containment isolated.

The CS System design ensures the system is not subjected to overpressurization events.~~The CS System also monitors the pressure in the reactor to ensure that, before the injection valves open, the reactor pressure has fallen to a value below the CS System's maximum design pressure. The variable is monitored by four redundant instruments. The instrument outputs are connected to relays whose contacts are arranged in a one-out-of-two taken twice logic.~~

Low Pressure Coolant Injection System

The LPCI is an operating mode of the Residual Heat Removal (RHR) System, with two LPCI subsystems. The LPCI subsystems may be initiated by automatic or manual means. Automatic initiation occurs for conditions of Reactor Vessel Water Level Low, Low, Low, Level 1 or Drywell Pressure - High concurrent with Reactor Pressure - Low. Each of these diverse variables is monitored by four instruments in two divisions. Each division is arranged in a one-out-of-two-twice network using level and pressure instruments which will generate a signal when:

(1) both level sensors are tripped, or

(2) two high drywell pressure sensors and two low reactor vessel pressure sensors are tripped, or

(3) a combination of one channel of level sensor and one of the other channel of high drywell pressure sensor together with its associated low reactor vessel pressure sensor (i.e., Channel A level sensor and Channel C high drywell and low reactor vessel pressure sensor).

BASES

BACKGROUND (continued)

Low Pressure Coolant Injection System (continued)

The initiation logic is cross connected between divisions (i.e., either start signal will start all four pumps and open both loop's injection valves). Once an initiation signal is received by the LPCI control circuitry, the signal is sealed in until manually reset. The cross division start signals for the pumps affect both the opposite division's start logic and the pump's 4KV breaker start logic. The cross division start signal to the opposite division's start logic is for improved reliability. The cross division start signals to the pump's 4KV breaker start logic is needed to ensure specific control power failures do not prevent the start of an adequate number of LPCI pumps.

Upon receipt of an initiation signal, all LPCI pumps start after a 3 second time delay when normal AC power is lost and standby diesel generator power is available. If normal power is available, LPCI pumps A and B will start immediately and pumps C and D will start 7.0 seconds after initiation signal to limit loading of the offsite sources.

The RHR test line and spray line are also isolated on a LPCI initiation signal to allow the full system flow assumed in the accident analyses and for those valves which are also PCIVs maintain primary containment isolated.

The LPCI System design ensures the system is not subjected to overpressurization events.~~The LPCI System monitors the pressure in the reactor to ensure that, before an injection valve opens, the reactor pressure has fallen to a value below the LPCI System's maximum design pressure. The variable is monitored by four redundant instruments. The instrument outputs are connected to relays whose contacts are arranged in a one-out-of-two taken twice logic.~~

Logic is provided to close the recirculation pump discharge valves to ensure that LPCI flow does not bypass the core when it injects into the recirculation lines. The logic consists of an initiation signal (Low reactor water level and high drywell pressure in a one out of two taken twice logic) from both divisions of LPCI instruments and a pressure permissive. The pressure variable is monitored by four redundant instruments. The instrument outputs are connected to relays whose contacts are arranged in a one-out-of-two taken twice logic.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

1.b, 2.b. Drywell Pressure-High

High pressure in the drywell could indicate a break in the reactor coolant pressure boundary (RCPB). The low pressure ECCS (provided a concurrent low reactor pressure signal is present) and associated DGs, without a concurrent low reactor pressure signal, are initiated upon receipt of the Drywell Pressure-High Function in order to minimize the possibility of fuel damage. The Drywell Pressure-High Function, along with the Reactor Water Level-Low Low Low, Level 1 Function, is directly assumed in the analysis of the recirculation line break (Ref. 1). The core cooling function of the ECCS, along with the scram action of the RPS, ensures that the fuel peak cladding temperature remains below the limits of 10 CFR 50.46.

High drywell pressure signals are initiated from four pressure instruments that sense drywell pressure. The Allowable Value was selected to be as low as practical and be indicative of a LOCA inside primary containment. The Drywell Pressure-High Function is required to be OPERABLE when the ECCS or DG is required to be OPERABLE in conjunction with times when the primary containment is required to be OPERABLE. Thus, four channels of the CS and LPCI Drywell Pressure-High Function are required to be OPERABLE in MODES 1, 2, and 3 to ensure that no single instrument failure can preclude ECCS and DG initiation. In MODES 4 and 5, the Drywell Pressure-High Function is not required, since there is insufficient energy in the reactor to pressurize the primary containment to Drywell Pressure-High setpoint. Refer to LCO 3.5.1 for Applicability Bases for the low pressure ECCS subsystems and to LCO 3.8.1 for Applicability Bases for the DGs.

1.c, 1.d, 2.c, 2.d Reactor Steam Dome Pressure-Low

Low reactor steam dome pressure signals are used as permissives for the low pressure ECCS subsystems. The low reactor pressure permissive is provided to prevent a high drywell pressure condition which is not accompanied by low reactor pressure, i.e. a false LOCA signal, from disabling two RHR pumps on the other unit. The low reactor steam dome pressure permissive also ensures that the injection valves open prior to pressure falling below the Allowable Value~~, prior to opening the injection valves of the low pressure ECCS subsystems, the reactor pressure has fallen to a value below these subsystems' maximum design~~

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

1.c, 1.d, 2.c, 2.d Reactor Steam Dome Pressure-Low (continued)

~~pressure~~. The Reactor Steam Dome Pressure—Low is one of the Functions assumed to be OPERABLE and capable of permitting initiation of the ECCS during the transients analyzed in Reference 2. In addition, the Reactor Steam Dome Pressure—Low Function is directly assumed in the analysis of the recirculation line break (Ref. 1). The core cooling function of the ECCS, along with the scram action of the RPS, ensures that the fuel peak cladding temperature remains below the limits of 10 CFR 50.46.

The Reactor Steam Dome Pressure-Low signals are initiated from four pressure instruments that sense the reactor dome pressure.

The pressure instruments are set to actuate at or above the~~between the Upper and Lower~~ Allowable Value~~s~~ on decreasing reactor dome pressure.

~~The Upper Allowable Value is low enough to ensure that the reactor dome pressure has fallen to a value below the Core Spray and RHR/LPCI maximum design pressures to preclude overpressurization.~~

The ~~Lower~~ Allowable Value is high enough to ensure that the ECCS injection prevents the fuel peak cladding temperature from exceeding the limits of 10 CFR 50.46. The CS and RHR/LPCI system designs ensure the systems are not subjected to overpressurization events.

DGs C and D which are initiated from the LPCI LOCA initiation are cross connected such that both DGs receive an initiation signal from both Divisions of the LPCI LOCA initiation circuitry. This cross connected logic is only required in MODES 1, 2, and 3. In MODES 4 and 5, redundancy in the DG initiation circuitry is not required. Therefore, in MODES 4 and 5 for DGs C and D only one division of ECCS initiation logic is required.

Four channels of Reactor Steam Dome Pressure—Low Function are required to be OPERABLE only when the ECCS is required to be OPERABLE to ensure that no single instrument failure can preclude ECCS initiation.