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U.S. Nuclear Regulatory Commission
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Perry Nuclear Power Plant
Docket No. 50-440
Emergency Technical Specification
Change Request - T.S. 3.6.4 Action a
for RCIC and RWCU Valves

Gentlemen:

Enclosed is a request for amendment of Facility Operating License NPF-58 for the Perry Nuclear Power Plant (PNPP), Unit 1. This letter requests an emergency change to the requirements of Technical Specification 3.6.4, "Containment Isolation Valves." This emergency change is necessary due to the declaration of the Reactor Core Isolation Cooling (RCIC) and Reactor Water Cleanup (RWCU) Systems outboard containment isolation valves as inoperable for closure under unique circumstances.

While maintaining compliance with all aspects of Technical Specifications, CEI Management discussed a request for a Temporary Waiver of Compliance, via telephone, with NRR and Region III representatives shortly after midnight on September 12, 1992. Verbal authorization for the waiver was granted by NRC Management during the telephone conversation. On September 12, 1992, a written follow-up letter was provided to the NRC. That letter committed to submittal of a Technical Specification Change Request. This letter satisfies that commitment.

Attachment 1 contains a Summary, Background, Safety Analysis, and the Significant Hazards Consideration. Attachment 2 contains the marked up Technical Specification page.

If you have any questions, please feel free to call.

Sincerely,

M. D. Lyster

Michael D. Lyster

MDL:BSF:ss
Attachments

cc: NRC Project Manager
NRC Resident Inspector Office
NRC Region III

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SUMMARY

This Operating License amendment request proposes a change to Technical Specification 3.6.4 to add a footnote to Action a; this footnote would only be effective until startup from the fourth refueling outage. Action a requires that when a containment isolation valve is declared inoperable, the affected penetration is to be isolated if the valve is not restored to Operable status within 4 hours. The Perry Nuclear Power Plant (PNPP) is currently operating under a Waiver of Compliance which provides an exception to the requirements of Action a for the inoperable Reactor Core Isolation Cooling (RCIC) and Reactor Water Cleanup (RWCU) outboard containment isolation valves, E51-F064 and G33-F004, respectively. These valves have been declared inoperable due to revised calculations that show that for very unique and highly improbable circumstances [i.e. a postulated circumferential line break downstream of the outboard valves, with a subsequent failure of the inboard containment isolation valve in the affected line to close, while voltage conditions exist at the Division 1 4.16 kV Class 1E (emergency) bus that have degraded to significantly less than typical values], the RCIC E51-F064 and RWCU G33-F004 valves might not be capable of full closure within the previously analyzed time frame. The proposed Technical Specification (TS) change would provide this exception for the period of time until the valves are restored to operability, with this extension expiring no later than restart from the fourth refueling outage.

BACKGROUND

The RCIC valve (E51-F064) is a normally open valve, which is designed to remain open following design-basis Loss-of-Coolant Accidents (LOCAs) so that the RCIC system can perform its intended functions of injecting cooling water into the reactor vessel and removing decay heat. The primary automatic isolation signals to the E51-F064 valve are for the occurrence of a break in the RCIC steam line downstream of the isolation valves. The RWCU valve (G33-F004) is a normally open valve, which is designed to close following a low water level-2 signal or any of various line break detection signals. During normal plant operation, this open valve allows reactor water to be transported to the RWCU filters for removal of impurities.

During the recent NRC inspection of the Generic Letter 89-10 Motor-Operated Valve program at PNPP, concerns were raised as to the capabilities of E51-F064 to isolate a worst-case complete circumferential line break downstream of the valve, which is assumed to occur when voltage conditions at the Division 1 4.16 kV emergency bus have degraded to significantly less than typical values. The NRC concerns were based on the assumptions utilized by CEI in various calculations used to determine valve capabilities. While CEI continues to maintain the validity of the assumptions utilized in the determination of valve operability, the calculated valve performance under postulated degraded voltage conditions prompted a conservative declaration of the RCIC valve as inoperable, while a more rigorous analysis of valve capability is performed. For similar reasons, G33-F004 was also declared inoperable. The additional analyses of valve capability remain ongoing.

DESCRIPTION OF EMERGENCY CIRCUMSTANCES

Both valves were declared inoperable at approximately 2235 on September 11, 1992, at which time the plant was in Operational Condition 3, Hot Shutdown, following an unexpected automatic scram on September 10, 1992. With the valves inoperable, Technical Specification 3.6.4, Action a would have required the associated penetrations to be isolated within 4 hours. Such isolation would have prevented the RCIC system from performing its intended function since steam could not be transported to the RCIC turbine, and an isolation of RWCU would have led to an eventual plant shutdown due to buildup of impurities above the Technical Specification limits. Due to the granting of the waiver, plant operation is acceptable for a temporary period; granting of this Technical Specification change request is necessary for plant operation until the valves have been declared Operable.

The situation currently faced could not have been avoided. The assumptions that the NRC staff has asked us to make in our valve capability calculations (thrust required, thrust capability, and degraded voltage considerations) are different than our empirical data would suggest, and are the direct cause of the valves being declared inoperable. Justified by the low safety significance of this very specific concern as detailed below, the issuanc the requested Technical Specification Change will permit these valves to remain open, thereby providing for RCIC and RWCU system availability during plant operation until the valves are declared Operable, which will be accomplished no later than restart from the next refueling outage. Th... change request is necessary to allow plant operation until no later than the next refueling outage, while performing Engineering evaluations and preparing design changes for field implementation. If a reactor shutdown occurs between October 3, 1992 and the start of the next refueling outage, a design change will be implemented to improve the E51-F064 valve capability such that it can be declared Operable again. A design change for the (-F004 valve will also be implemented during such a shutdown, unless structural encroachments are found in the field which would require additional analysis such that startup from the reactor shutdown would be delayed.

SAFETY ANALYSIS

As briefly noted above, in order for the identified calculational concerns for these valves to be of any significance, three circumstances need to exist concurrently. This safety analysis will discuss each of these circumstances, why they are not likely to occur at PNPP, how they would be handled, and mitigating factors that show they are not a concern. It will discuss why the RCIC and RWCU penetrations have a very high probability of isolation. It will also address compensatory actions that will be taken to preclude the concern.

Specifically, the following items will be addressed:

1. Low likelihood of a circumferential pipe break

    a. piping design
    b. low susceptibility to break initiating events
    c. detection of leaks prior to break occurrence

2. Low likelihood of a failure of the inboard valve to close since sufficient thrust is generated by the operator for closure against design-basis differential pressure (dP)

3. Low likelihood of a significantly degraded voltage condition at the outboard valves

4. Compensatory measures to preclude the significantly degraded voltage condition until the valves are declared Operable

5. Radiological releases for the spectrum of leaks/breaks.

Many of the safety analysis discussions presented below were originally developed in response to Generic Letter 89-10, Supplement 3, "Consideration of Results of NRC-Sponsored Tests of Motor-Operated Valves." It requested BWR licensees to assess the applicability of the data from the NRC-sponsored motor operated valve (MOV) tests, to determine the "as-is" capability of the Reactor Core Isolation Cooling and Reactor Water Cleanup MOVs, and to identify any deficiencies in those MOVs. The NRC also requested licensees to perform a plant-specific safety assessment to verify that the generic safety assessments performed by the NRC staff and the BWR Owner's Group are applicable to their plant.

In response to that Generic Letter, CEI performed a plant-specific safety assessment, which was referenced in a letter to the NRC (PY-CEI/NRR-1271 L) dated December 10, 1990. The safety assessment relates directly to the issues at hand, and provides the majority of the basis for this Technical Specification change request. Accordingly the safety assessment conclusions have been updated and included as a part of this Safety Analysis.

The first circumstance that would have to occur in order to raise any concern over the capability of the RCIC or RWCU outboard isolation valves to close unassisted, is the occurrence of a complete circumferential line break. This is unlikely, since materials in these lines were selected for low probability of pipe failure. Also, all of the safety-related piping in RCIC (E51) and RWCU (G33) has been designed to applicable ASME Section III rules. Implicit in the allowable piping stresses of this Code is a substantial built-in margin to the material ultimate strength. The portions of the RCIC and RWCU systems downstream of the isolation valves are fabricated from carbon steel piping and components, utilizing for piping SA106 Grade B or SA333 Grade 6 and for fittings SA105, SA234 Grade WPC or SA420 Grade WPL6 materials.

There are very few failure mechanisms for these types of piping systems. Nuclear and fossil power plant experience has indicated that large breaks have resulted from either large water hammer events or undetected significant pipe wall erosion. CEI believes that there is a low probability of these mechanisms occurring in the subject piping. The technical findings relevant to the resolution of Unresolved Safety Issue A-1, Water Hammer, were contained in NUREG-0927, Revision 1, "An Evaluation of Water Hammer Occurrence in Nuclear Power Plants." In this NUREG the safety significance of water hammer in the RCIC and RWCU systems is classified as low. Therefore, the probability of a large pipe break in any of the subject lines due to water hammer should be low. With respect to pipe wall erosion, the RCIC steam lines are used only intermittently during pump testing, leading to insignificant erosion/corrosion

occurring in this line. For the RWCU system, the evaluation performed to establish erosion/corrosion monitoring points and the comprehensive erosion/corrosion monitoring program established for the Perry Nuclear Power Plant pursuant to Bulletin 87-01 and GL 88-08 provide reasonable assurance of the continued structural integrity of the RWCU system.

As indicated above, the portions of PNPP's RCIC and RWCU systems downstream of the isolation valves are fabricated from carbon steel piping and components. Intergranular Stress Corrosion Cracking (IGSCC) is not a concern for carbon steel piping systems, therefore this type of failure is not a concern.

Even if leakage from these systems should occur, it would not likely be an instantaneous circumferential break of the line. It is industry experience that high energy pipes experience leaks long before a pipe break condition develops. Industry has referred to this phenomena as Leak-Before-Break (LBB). In general terms, the LBB concept is based on the fact that piping is fabricated from tough ductile materials which can tolerate large through-wall cracks without complete fracture under service loadings. By monitoring for leakage from such cracks, they can be detected and isolated before the margin to rupture is challenged, and therefore before the isolation valves can be subjected to full differential pressures. With respect to leak detection, the Perry Nuclear Power Plant has been designed for compliance to General Design Criterion (GDC) 54. Piping systems penetrating primary reactor containment are provided with leak detection, isolation and containment capabilities (refer to USAR Section 5.2.5 and 7.6.1.3).

The PNPP design has incorporated multiple channel, redundant leak detection monitoring of the high energy lines external to the containment. This monitoring is sensitive to small leaks and causes both an alarm in the control room and, for somewhat larger leaks, an automatic isolation signal to the leaking system's isolation MOVs. Should a leak develop, it will be detected by area temperature and floor drain sump level monitors. In addition for RWCU, a differential flow device will detect a small break. The monitoring instrumentation provides alarms in the Control Room which cause entry into annunciator response instructions. These instructions would direct the operators to determine the cause of the alarm, and for these types of small leaks, would lead to closure of the MOV in the leaking pipe before the leakage could cause any significant flow change, fluid loss, or radiation releases, and before any significant long term environmental challenge to the MOVs could occur. For a wide spectrum of these smaller leaks, the closing valves would not experience the maximum differential pressures that are postulated in the NRC concern, and valve closure capability and resultant leak isolation is not in question.

Based on the discussions above, it can be seen that the likelihood of a full circumferential break of either line is very remote. To support this conclusion, a review of the frequency of a circumferential line break at PNPP for either the RCIC or RWCU line was performed. This review considered the number of pipe segments of concern downstream of the outboard valve. For RCIC, the frequency of such a circumferential break was determined to be 6.31E-5/yr. For RWCU, the frequency of such a circumferential break was determined to be 2.42E-4/yr.

Even the occurrence of a full circumferential break is not of any notable concern as long as the inboard valve on the broken line performs its function and isolates the penetration. The likelihood that the inboard valve will randomly fail upon receipt of the isolation signal is 2.93E-3/yr. The combined probability of an unisolable line break (i.e. a line break occurring simultaneously with inboard valve closure failure) is 1.84E-7/yr for RCIC and it is 7.08E-7/yr for RWCU. If the inboard valve does not suffer such a random mechanical failure, it is fully expected to close. These valves have demonstrated their capabilities to close against normal reactor pressure as a result of plant operation. The RWCU valves have closed against reactor pressure during various false water level 2 signals and high differential flow signals. The inboard RCIC valve was tested for Bulletin 85-03 by successfully closing it against a reactor pressure of 920 psig. Also, the valve capability calculations performed for the inboard valves on both lines showed adequate margin to close even against the higher than normal reactor pressures assumed for the design-basis case, even utilizing the extremely conservative assumptions considered appropriate by the NRC staff. The closure of the valve isolates the penetration and terminates the blowdown event of concern.

Even in the event of the above postulated combination of events, the capability of the E51-F064 and/or G33-F004 valves to close is not in question unless a significantly degraded voltage condition exists at the emergency bus while the valve is trying to close. The frequency of a degraded voltage condition sufficient to result in the valves' performance being affected to the point that it might not fully close was also reviewed and determined to be low. It is CEI's position that, at the typical voltages provided to the emergency bus, the outboard containment isolation valves' capability to isolate a circumferential line break with an inboard valve failure is sufficient. This sufficiency is also maintained if the emergency bus voltage is being supplied by the diesel.

To illustrate this point, it is noted that even considering a voltage at the emergency bus that has degraded all the way to the Analytical Limit for the Technical Specification "degraded voltage" value (the Analytical Limit is approximately 2% below the Tech Spec setpoint of 3800 volts = 3730 volts), and assuming other currently justifiable valve and operator factors, including values considered appropriate by the NRC staff, the RCIC outboard valve operator capability is less than 2.5% below the calculated thrust requirements for full valve closure against the design-basis differential pressures. In order for the valve calculations to show that the RCIC valve is able to produce the thrust necessary to fully close the valve, the voltage at the emergency bus merely needs to be within the Technical Specification alarm setpoint limits of 3800±20 volts (or worst case equal to 3780 volts). At all higher voltages, additional margin to the design basis thrust requirement is available. The Tech Spec degraded voltage instrumentation is calibrated during every refueling outage, and the setpoint is verified. The as-left values for the instrument setpoints on the Division 1 emergency bus, which supplies both the RCIC and RWCU outboard valves, were all equal to or above 3800 volts following the most recent calibration. Therefore, although it is necessary to assume the Analytical limit of 3730 volts for the Operability determination, it is unlikely that voltages lower than 3780 could actually ever exist without being alarmed by the Tech Spec instrumentation, at which point actions would be taken to maintain voltage or raise it above the alarm setpoint. Therefore, the RCIC valve should always have sufficient voltage to

generate adequate thrust. To the best of CEI's knowledge, this alarm has never been received. It should be noted that the calculations for the cases where sufficient thrust is generated for valve closure do not assume that "locked rotor" current exists, because there is no reason to believe that such a condition will exist. [Future calculations performed as part of the revised response to Generic Letter 89-10 Supplement 3 (which will be submitted prior to restart from the fourth refuel outage) will further address the approp ite assumptions for current in their conclusions.] For the G33-F004 valve, using valve and operator factors consistent with those for the E51-F064 valve (appropriate for the specific valve and operator configuration used for G33-F004) and consistent calculational assumptions, the G33-F004 valve has been shown to have positive margin for closure. However, this valve has not yet been declared Operable pending further discussions with the NRC.

The above discussions have shown that in all but the most remote circumstances, the penetration will be isolated early in the leak/break scenario. In order for the penetration to be postulated to not fully isolate, the break must be a very large break, with a subsequent failure of the inboard containment isolation valve in the broken line to close due to a random mechanical failure, while a degraded voltage condition exists at the emergency bus far enough below the typical levels that the valve operator's capability to fully close the valve might be questioned. The likelihood of this overall scenario was reviewed and is significantly less than the unisolable RCIC or RWCU line break numbers of 1.84E-7/yr and 7.08E-7/yr, and for purposes of discussion, it will be considered to be in the E-11/yr range.

In order to preclude the significantly degraded voltage condition until the v s are declared Operable, and therefore remove the concern, the voltage conditions on the emergency bus will be monitored and appropriate actions taken for voltages well above those of concern. An alarm has been added to the Emergency Response Information System (ERIS) computer point which monitors the emergency bus voltage such that it will alert the operator whenever voltage drops below 4000V. If this occurs, instructions will require the plant Operators to verify the validity of the alarm by checking alternate plant instrumentation, and if it is valid, will require starting and transfer of loads to the Division 1 Emergency Diesel-Generator (which supplies power to the Division 1 emergency bus that feeds E51-F064 and G33-F004). It should be noted that while these actions are being taken, if bus voltage continues to drop and reaches the "degraded voltage" setpoint (nominally set at 3800V), the diesel would automatically start after only 5 minutes and would supply power to these loads even without operator action. If voltage continues to drop to the "undervoltage" setpoint, the diesel would immediately auto start and supply power to the bus. The manual or automatic transfer of loads to the Emergency Diesel-Generator would remove the possibility of reduced voltage conditions existing on the emergency bus. In the event that ERIS should be unavailable, a non-licensed plant Operator or other technically qualified member of the plant staff will be promptly stationed at the alternate plant instrument that provides a reading of the emergency bus voltage; with the assignment to continuously monitor the voltage and report to the Control Room Operator if a degraded voltage condition of 4000V is noted. The resultant Operator actions would be the same as described above. The incorporation of these administrative controls ensures that there is not a concern for the

limited time until startup from the next refueling outage, by which time the valves will have been declared Operable again.

The radiological release from a leak smaller than the circumferential break in the RCIC or RWCU line would be bounded by the radiological analysis for the main steam line break since the line would clearly be capable of being isolated rapidly. The radiological release from even a circumferential break would also clearly be bounded by the main steam line break for the situation without a significantly degraded voltage condition since the RCIC and RWCU outboard valves would be assumed to close within the previously analyzed time frames. The main steam line break release would also bound the situation when the inboard valve does not suffer a random mechanical failure, and is therefore available to close to isolate the penetration.

Future Actions

Various options for the restoration of operability of the valves are being evaluated. Engineering evaluations are being performed to establish additional margin to design requirements. Design options for the existing systems currently under consideration include modification of valve components, replacement of the valve operator, changes in the overall gear ratio and/or motor of the operator, reduction in length or increase in size of the power supply cable, and changes to the undervoltage setpoint. If a reactor shutdown occurs between October 3, 1992 and the start of the next refueling outage, a design change will be implemented to improve the E51-F064 valve capability such that it can be declared Operable again. A design change for the G33-F004 valve will also be implemented during such a shutdown, unless structural encroachments are found in the field which would require additional analysis such that startup from the reactor shutdown would be delayed. Whichever option is utilized to restore the operability classification of these valves, they will be restored to an OPERABLE status prior to restart from the next refueling outage.

SIGNIFICANT HAZARDS CONSIDERATION

The standards used to arrive at a determination that a request for amendment involves no significant hazards considerations are included in the Commission's Regulations, 10CFR50.92, which state that the operation of the facility in accordance with the proposed amendment would not (1) involve a significant increase in the probability or consequences of an accident previously evaluated, (2) create the possibility of a new or different kind of accident from any previously evaluated, or (3) involve a significant reduction in a margin of safety.

The proposed amendment has been reviewed with respect to these three factors and it has been determined that the proposed change does not involve a significant hazard because:

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

   The proposed change is to provide a footnote to Technical Specification 3.6.4 "Containment Isolation Valves" Action a, to permit the penetrations associated with the Reactor Core Isolation Cooling (RCIC) steam supply

line and Reactor Water Cleanup (RWCU) suction line outboard containment isolation valves to remain open to permit the associated systems to perform their primary functions, although these valves have been declared inoperable due to revised calculations of their closing capability under a very unique set of circumstances.

With respect to the probability of previously evaluated accidents, the only initiating event of concern is a circumferential line break in the RCIC or RWCU lines downstream of the E51-F064 or G33-F004 outboard containment isolation valves, respectively. The proposed allowance to have these valves open (although declared inoperable) does not increase the probability of occurrence of a line break in the downstream lines above that which is already possible when the valves are open (and declared Operable) during power operation of the plant. The probability of occurrence is based on the number of piping segments downstream of the valves and their likelihood of failure, which is completely unaffected by this proposed change. This probability is estimated to remain at approximately 6.31E-5/yr for RCIC and 2.42E-4/yr for RWCU. This request does not propose changes to the number of pipe segments or to their capability to withstand imposed stresses, nor does it change the probability of a water hammer event, nor the potential for erosion/corrosion of the piping systems, nor the low susceptibility of the piping to intergranular stress corrosion cracking effects.

With respect to consequences of a previously evaluated accident, it is necessary to examine the event of concern. The previously evaluated accident is the Steam Line Break Outside Containment. The only event of concern for which the valves were declared inoperable was for a very unique and highly improbable combination of events [i.e. a postulated initiating event of a circumferential line break downstream of the RCIC or RWCU outboard containment isolation valves E51-F064 or G33-F004, with the subsequent random failure of the inboard containment isolation valve in the affected line to close early in the scenario, wh a voltage condition exists at the Division 1 4.16 kV Class 1E (emergency) bus that is far enough below typical voltages that the closure capability of the outboard valve might be questioned]. For all other line leak or break scenarios, the capability of the outboard RCIC and RWCU valves to close and isolate the leakage is not in question, and the associated radiological consequences would clearly be bounded by the existing USAR evaluation of the main steam line break outside containment. In order to preclude the significantly degraded voltage condition (until the valves are declared Operable) and therefore remove the concern, the voltage conditions on the emergency bus will be monitored and appropriate actions will be taken for voltages well above those of concern. The outboard valve operator capability will therefore not be in question, and although the valves will still be declared inoperable for an interim time period, the valve will be capable of closure within its previously analyzed time frame if called upon. Therefore, the consequences of the event remain bounded by the existing Steam Line Break Outside Containment analysis. In conclusion, the proposed change does not involve a significant increase in the consequences of an accident previously evaluated.

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

   This proposed change simply allows normally open valves to remain open to permit continued functioning of the associated systems, although the valves have been declared inoperable (due to calculational assumptions) upon a circumferential line break downstream of the outboard containment isolation valves. Breaks in these RCIC and RWCU lines downstream of the outboard containment isolation valves have been previously evaluated, therefore the initiating event to which these valves respond is not considered to be a new or different kind of accident. The proposed change does not involve a new or different operating mode of the associated systems, nor does it involve any design change to the plant.

3. The proposed change does not involve a significant reduction in a margin of safety.

   Even without the compensatory actions which will be in place, the valves E51-F064 and G33-F004 remain able to fully close under a wide range of postulated piping leaks and breaks for which the resulting differential pressure the valve would close against would be less than the design basis differential pressure condition. The valves are only considered inoperable for the unique set of circumstances of a circumferential RCIC or RWCU line break occurring while the voltage at the emergency bus is significantly degraded such that the valve operators capability for closure might be questioned. Even under these highly improbable circumstances, there is not a concern provided the inboard valves, which have adequate design margin for closure in these circumstances against the full design-basis differential pressure, close to isolate the penetration. The closure of the valve terminates the blowdown. The penetration will also be isolated by the outboard valves, due to the compensatory actions to maintain voltages at the emergency bus well above those that lead to a concern over valve operator capability. The Bases for the Containment Isolation Valve Specification state that the intent is to limit releases of radioactive material to the environment consistent with the assumptions used in the analysis for a LOCA. These Bases are maintained, since even for the circumstances of a RCIC or RWCU line break with single failure of the inboard or the outboard valve, the remaining valve will close to isolate the break. The resultant doses will be less than the existing evaluations for the bounding Main Steam Line Break Outside Containment, and therefore will be significantly less than the 10CFR100 guidelines for a LOCA. Therefore the margin of safety for the Containment Isolation Valve Specification is not reduced.

Based on the above, it is determined that no significant hazards considerations are involved with the proposed change.