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PY-CEI/NRR-1547 L

U.S. Nuclear Regulatory Commission
Document Control Desk
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Perry Nuclear Power Plant
Docket No. 50-440
Request For Temporary Waiver of
Compliance - Technical Specification
3.6.4 Action a.2 for RCIC and RWCU Valves

Gentlemen:

A Temporary Waiver of Compliance to the requirements of Technical Specification 3.6.4, "Containment Isolation Valves," is hereby requested by The Cleveland Electric Illuminating Company (CEI), operators of the Perry Nuclear Power Plant (PNPP). This waiver 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 the request, 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. This letter satisfies the requirements for prompt written follow-up to the verbal request and authorization.

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. It is only for the occurrence of a break in the RCIC steam line downstream of the isolation valves that E51-F064 is called upon to automatically close. 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. In the open position, this 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, concurrent with electrical grid conditions that have degraded to 95% of

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rated voltage. The NRC concerns were based on the assumptions used by CEI in various calculations used to determine valve capabilities. While CEI continues to maintain the validity of the assumptions utilized in determination of valve operability, the calculated valve performance under postulated degraded voltage conditions has prompted us to conservatively declare the valve inoperable, while a more rigorous analysis of valve capability is performed. For simila reasons, G33-F004 has also been declared inoperable.

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.2 requires the associated penetrations to be isolated within 4 hours. Such isolation would prevent the RCIC system from performing its intended function since steam could not be transported to the RCIC turbine; with RCIC inoperable, startup of the plant is precluded by Technical Specification 3.7.3. Technical Specifications do not specifically preclude plant startup with RWCU isolated; however, RWCU is necessary for water level control during startup, and eventual plant shutdown would be required due to buildup of impurities above the Technical Specification limits.

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 valve being declared inoperable. Justified by the low safety significance of this very specific concern as detailed below, the issuance of the requested Waiver of Compliance permits these valves to remain open, thereby providing for RCIC and RWCU system availability until such time that a Technical Specification change can be processed.

Justification for Continued Operability (JCO)

On October 25, 1990, the NRC issued 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 discussed 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 JCO. Accordingly the safety assessment is provided as an attachment to this letter. The attachment has been annotated appropriately to reflect current conditions. All annotations are enclosed in brackets [ ] and identified by the date of this letter. Additional justification is provided in the following paragraphs.

Both E51-F064 and G33-F004 remain fully capable of performing intended functions in the open position, and also remain capable of closing to isolate a wide spectrum of conceivable line breaks in the RCIC and RWCU systems, respectively, provided that adequate supply voltage is provided. The occurrence of the set of circumstances necessary to create an event that would both require closure of either of the subject valves and call into question its capability to fully close is extremely remote. First, a line break in the safety class piping outside of the containment would have to occur. We have determined that the expected frequency of occurrence for both RCIC and RWCU line breaks is 4.2 E-5/yr. Also, the probability of degraded voltage at PNPP has been calculated to be 1.76 E-4/yr. The combined probability of a line break occurring simultaneously with a degraded voltage condition is 7.39 E-9/yr. In order for non-closure of the E51-F064 or G33-F004 valve to be of any notable concern, as explained in the attached Safety Assessment, the other isolation valve in the penetration must also fail to close. Valves E51-F0063 and G33-F001 are the inboard isolation valves for the RCIC and RWCU systems, respectively, and the capability of these valves to close even under degraded voltage conditions is not in question. Taking into account the frequency of failure of the inboard valve, the frequency of a degraded voltage condition coincident with an unisolable RCIC or RWCU line break is 2.17 E-11/yr. Thus it can be seen that this is an extremely low probability occurrence.

In order to decrease the probability of the occurrence of such a combination of events even further, administrative controls are being placed into effect which will require starting and transfer of loads to the Division 1 diesel-generator (which supplies power to the safety bus that feeds E51-F064 and G33-F004), if the Bus voltage decreases to the degraded voltage setpoint as specified in Technical Specifications. The transfer of loads to the Emergency Diesel Generator would remove the possibility of reduced voltage conditions as a result of further degradation of the non-safety related power supply.

Additionally, various options for the restoration of operability of the valves are being evaluated. Prior to proposing modifications to the existing system equipment, engineering evaluations will be performed to establish additional margin to design requirements. In the event that adequate additional margin cannot be effectively demonstrated, possible design options for the existing systems include modification of piping and valve components, replacement of the valve operator, changes in the overall gear ratio of the operator, reduction in length or increase in size of the power supply cable, and changes to the undervoltage setpoint. 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.

Based on the discussions provided in the Justification for Continued Operation, CEI has concluded that the requested waiver does not involve a significant hazard consideration. A detailed restatement of the significant hazard consideration, in accordance with 10CFR50.92, will be provided in a Technical Specification Change Request. The requested waiver has been reviewed against the criteria of 10CFR51.22 for environmental considerations. As shown above, the proposed change does not involve a significant hazards consideration, does not increase the types and amounts of effluents that may be released offsite, and does not significantly increase individual or cumulative occupational radiation exposures. Based on the foregoing, it has been concluded that the proposed waiver of compliance meets the criteria given in 10CFR51.22(c)(9) for a categorical exclusion from the requirement for an Environmental Impact Statement.

This Waiver of Compliance is requested only for the interim period until a Technical Specification change can be processed. A formal Technical Specification Change Request is expected to be submitted for NRC staff review on or about September 18, 1992.

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

Sincerely,

Michael D. Lyster

MDL:HLH:ss

Attachments

cc: NRC Project Manager
NRC Resident Inspector
NRC Region III

PERRY NUCLEAR POWER PLANT SAFETY ASSESSMENT RELATIVE TO THE ISOLATION FUNCTION OF MOVs FOR RCIC STEAM SUPPLY LINE AND RWCU WATER SUPPLY LINE

NOTE: This attachment has been annotated appropriately to reflect upon conditions. All annotations are enclosed in brackets [ ] and identified by the date of this letter. Attachments to this Safety Assessment are not provided, but are available for review at the site.

Likelihood of Pipe Break

Piping Stress Levels

All of the safety-related piping in RCIC (E51) and RWCU (G33) have been designed to applicable ASME Section III rules. Implicit in the allowable stresses of this Code is a substantial built-in margin below the material ultimate strength. The RCIC and RWCU systems are fabricated primarily 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 (refer to USAR and E51 and G33 Piping Design Specifications).

Failure Mechanisms

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 augmented inspections being performed for detection of erosion are discussed below. 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.

RCIC and RWCU Low Erosion/Corrosion and Inter-granular Stress Corrosion Cracking Susceptibility

Perry's response to NRC Bulletin 87-01 and Generic Letters 88-01 and 89-08 are outlined in the attached memo from C. Frank to E. Ortalan/R. Parker dated December 7, 1990.

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, PNPP's RCIC and RWCU systems are fabricated primarily from carbon steel piping and components. IGSCC is not a concern for carbon steel piping systems. The review of RWCU performed in response to Generic Letter 88-01 for IGSCC showed only two austenitic stainless steel welds in the RWCU supply lines. These welds are located in the drywell upstream of the RWCU containment isolation valves and therefore do not contribute to the potential for a high energy line break downstream of the RWCU containment isolation valves. Furthermore, both welds are categorized as IGSCC Category A (IGSCC resistant) and are included within PNPP's GL 88-01 IGSCC inspection program [During RFO-3, RWCU piping inside the drywell was replaced with austenitic stainless steel that is considered resistant to IGSCC; all of this new piping is also upstream of the RWCU containment isolation valves -- 9/12/92]

Plant Mitigative Features

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). However, 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).

Perry plant 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 is likely to be detected by area temperature and floor drain sump level monitors. These monitors alarm in the Control Room and cause entry into annunciator response procedures. These procedures would direct the operators to determine the cause of the alarm and 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. Refer to attached memo from G. Chasko to C. J. Frank dated November 30, 1990 for details of the operator actions to leak detection annunciator response. See also, "Closure After Depressurization" discussion below for description of additional operator action in the event a system containing a pipe break cannot be isolated.

Margin on Assumed Differential Pressure

The differential pressures assumed in the design phase for the establishment of the operating capability of the MOVs are greater than would actually occur during a high energy line break blowdown event. This provides additional thrust for valve closure during a blowdown from a line break. As described in detail in PNPP's letter to the NRC, PY-CEI/NRR-1271L, the RCIC and RWCU containment isolation valves have this margin. [This conclusion is the current issue under discussion with NRC -- 9/12/92]

Valve Redundancy

RCIC steam supply lines and the RWCU letdown lines are all equipped with two AC powered motor-operated isolation valves, one inside containment and one outside containment. These pairs of valves receive coincident signals to close and have the same static load closing stroke times.

The total differential pressure between the reactor and the break would be shared across the two valves. The reduced differential pressure therefore experienced by each individual valve would increase the likelihood of one or both valves closing due to the reduced thrust necessary to close each valve.

Closure After Depressurization

For a large RCIC steam line break, if the isolation valves fail to close, the reactor will depressurize below the low pressure injection systems shutoff head before the core would begin to uncover. These systems would therefore respond to provide abundant water makeup to the vessel. With offsite power still available, the feedwater/condensate pumps can likewise provide abundant cooling to maintain core integrity. After depressurization the load on the isolation valves would be greatly reduced. For MOVs that failed to close completely because of high friction factor caused by high blowdown flow, the likelihood of closure on the second try following depressurization would be greater.

For the case of an RWCU system break, if the isolation valves fail to close, HPCS and RCIC would be available to provide make up while the reactor coolant system depressurizes. However, if these systems alone could not keep up with the loss from a RWCU line break, depressurization through the break or by ADS would result in the low pressure systems functioning; thus, core damage should not occur. As discussed above, after depressurization the load on the isolation valves would be greatly reduced and the likelihood of closure of an undamaged MOV on the second try following depressurization would be greater.

In the event a pipe break cannot be isolated, plant operating procedures direct the operators to depressurize the reactor and shutdown the plant by either a unit shutdown or fast reactor shutdown depending on the severity of the break.

Equipment Qualification/Flooding

The Equipment Qualification program at PNPP established the capability of the plant safety related electrical equipment to perform their design basis safety functions under the limiting environmental conditions postulated for that equipment. The attached memo from S. Patel/S. W. Litchfield to E. Ortalan, et.al. dated 12/5/90 details the qualification of the subject RCIC and RWCU valves.

The design of the ECCS pump rooms are compartmentalized such that they are water tight so that any potential flooding in one room cannot impact the other ECCS rooms. Floor drains inside the ECCS pump rooms are also isolated from each other. Also the ECCS pump room and the Auxiliary Building corridors are equipped with level switches which alarm in the control room. The outboard isolation valves are located in the steam tunnel which is equipped with floor drains routed to the turbine power complex precluding submergence of the valves.

The likelihood of valve closure/reclosure in any of these lines is high at PNPP as outlined in our letter PY-CEI/NRR-1271L, which showed that valve motors are not undersized. [This conclusion is the current issue under discussion with NRC -- 9/12/92]

Consequence Mitigation

The primary symptom in the emergency procedures for a break in one of the lines under consideration is water level in the reactor pressure vessel (RPV). The mitigative systems for supplying make-up are HPCS and/or RCIC, main feedwater, low pressure coolant injection and low pressure core spray systems that can be used in an accident management capacity. Core cooling would continue without serious offsite consequences even if isolation of the broken line is delayed until much later in the scenario. ECCS components have spacial separation such that the impact of the postulated high energy line break of RCIC steam supply piping or the RWCU piping will not affect safe shutdown of the plant (refer to USAR Chapter 3.6).

Radiological Consequences

The radiological release from the High Energy Line Break (HELB) of the RCIC steam supply line and the RWCU supply lines is bounded by that of the main steam line break. These smaller lines do not depressurize the reactor vessel as fast as the main steam line. The reactor inventory release for these breaks is mostly steam. The dose from steam loss through an outside line break is small, therefore, the offsite release from the RWCU and RCIC line break will still meet requirements of 10CFR100.