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SUBJECT: Forwards application for proposed Amend 97 to License NPF-14. Amend rev Tech Specs to allow operation of facility until third refueling & insp outage w/o availability of HPCI sys.

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MAY 14 1987

Director of Nuclear Reactor Regulation
Attention: Dr. W. R. Butler, Project Director
Project Directorate I-2
Division of Reactor Projects
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

SUSQUEHANNA STEAM ELECTRIC STATION
EMERGENCY CHANGE REQUEST - PROPOSED
AMENDMENT 97 TO LICENSE NO. NPF-14
PLA-2859 FILES A17-2, R41-2 Docket No. 50-387

Dear Dr. Butler:

As of May 9, 1987, Susquehanna SES Unit 1 has been in a 14-day Technical Specification Action Statement due to the unavailability of the HPCI system. The purpose of the letter is to provide you the following:

1. Sufficient information to justify operation until the Unit 1 Third Refueling and Inspection Outage (currently scheduled to begin on 9/12/87). We anticipate that this will result in your issuance of either an emergency change to the Technical Specifications (should you deem one required), or an NRC safety evaluation which would relegate what is currently an unreviewed safety question per 10CFR50.59 to a temporarily acceptable mode of operation.

2. A basis for providing immediate relief from the requirements that led us to isolate the HPCI system, so that it can be made available to perform its function. We urge the staff to take prompt action on this part of our request in order to enhance safe operation by restoring HPCI availability.

The information that follows will be presented assuming an emergency Technical Specification change is required. Our proposed change is attached in marked-up form. This information will be followed by a brief discussion as to why interpretation is required to determine whether the issue should be resolved via an amendment to the Technical Specifications or by an NRC safety evaluation in response to an unreviewed safety question. Our basis for the immediate restoration of HPCI availability is provided at the end of this letter.

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A. BASIS FOR EMERGENCY TECHNICAL SPECIFICATION CHANGE

BACKGROUND: The inboard containment isolation valve on the HPCI Steam Supply line (HV-155F002, see attached Figure 1) has been determined to have an improper (low) closing torque switch setting. As is discussed in greater detail in the attached report, the only event of concern given this problem is a HPCI Steamline break outside containment with a single failure which fails the outboard isolation valve (F003) in the open condition. This would allow leakage past the F002 valve, because for this event the F002 valve would stop at the 97% closed position. The isolation signals and time provided in Technical Specification Table 3.6.3-1, "Primary Containment Isolation Valves" exist in support of vessel isolation given this event. Therefore, the F002 valve was declared inoperable and per Specification 3.6.3, the associated penetration was isolated in order to permit continued operation. However, isolation of this line renders HPCI inoperable, and the unit is therefore in a 14-day action (Specification 3.5.1 Action C.1) which expires on May 23, 1987.

JUSTIFICATION FOR CHANGE: See attached "Safety Assessment for Operation with HPCI F002 Steam Supply Isolation Valve with Incorrect Torque Switch Setting".

NO SIGNIFICANT HAZARDS CONSIDERATIONS:

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

As delineated in the attached report, the only event of concern given the current condition of the F002 valve is a HPCI Steamline break outside containment. The probability analysis indicates that this is an unlikely event over the slightly less than four months that the proposed change could be in effect. The radiological consequences were conservatively determined to be slightly worse than the bounding FSAR analysis of a Main Steamline break outside containment, but the increase is insignificant given the implicit error in such calculations and since both numbers are such a small fraction of 10CFR100 limits.

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

No hardware or procedural changes are proposed that would create a new event requiring evaluation.

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

[illegible]

[illegible]

[illegible]

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[illegible]

[illegible]

[illegible]

As delineated in the attached report, the event of concern has been shown to be unlikely, and its consequences, both in terms of affect on safety-related equipment and radiologically, have been shown to be acceptable from a regulatory standpoint. Therefore, the margin of safety is not significantly reduced due to operation with the improper torque switch setting until the Unit 1 Third Refueling and Inspection Outage.

BASIS FOR EMERGENCY REQUEST: 10CFR50.91 provides guidance on what information the NRC requires in support of an application for an emergency change.

First, it requires the applicant to justify that an emergency exists, i.e. " . . . failure to act in a timely way would result in derating or shutdown of a nuclear power plant . . .". Unit 1 is currently operating at full power. Failure to allow PP&L to restore HPCI to operable status by the end of the 14-day restoration time (May 23, 1987) forces the unit to be placed in hot shutdown within the following 12 hours and reactor steam dome pressure to be reduced to $\leq$ 150 psig within the subsequent 24 hours. Since the torque switch cannot be reset without a containment entry, this clearly meets the NRC criteria.

Secondly, 10CFR50.91 requires the licensee to " . . . explain why this emergency situation occurred and why it could not avoid this situation . . . ". The improper setting was discovered on May 7 during a records search pursuant to IE bulletin 85-03. After some evaluation, the valve was declared inoperable and the penetration isolated on May 9. Application in advance of this situation was impossible since the problem was discovered during full power operation. Based on the time necessary to evaluate the problem and to prepare and review this proposed internally, we believe that this application has been submitted in a timely fashion.

B. DISCUSSION OF TECHNICAL SPECIFICATION CHANGE VS. UNREVIEWED SAFETY QUESTION

As stated previously, the F002 valve was originally declared inoperable due to its existence in Technical Specification Table 3.6.3-1, "Primary Containment Isolation Valves". The Bases section for this specification clearly indicates that its purpose is to validate LOCA analysis assumptions. The attached report points out that the F002 valve, in its current configuration, will fully close under LOCA conditions if required. Therefore it could be stated that the F002 valve is operable in support of this requirement, and no

Technical Specification actions were required. PP&L took the conservative position that although the LOCA function is protected, the isolation time and isolation signals indicated in the table analytically support the evaluation of a HPCI Steamline Break, and therefore the table was validating this function as well.

Should this position be deemed inappropriate by the NRC, PP&L would still consider the issue an unreviewed safety question pursuant to 10CFR50.59. This is primarily because the radiological consequences of the HPCI Steamline break have been shown to be slightly worse (although still well within regulatory limits) than the current evaluation presented in FSAR Section 15.6. In this light, we would request that the NRC consider the attached safety assessment as our justification that operation with the improper torque switch setting is safe until the Unit 1 Third Refueling and Inspection Outage.

C. <u>BASIS FOR IMMEDIATE RESTORATION OF HPCI AVAILABILITY</u>

As stated in Part B above, PP&L feels that we took a conservative approach to compliance with the Technical Specifications under the circumstances presented. However, we firmly believe that in this case, our request for immediate restoration of HPCI availability is a safe and prudent one. In order to attempt to quantify our engineering judgement, we reviewed the Individual Plant Evaluation (IPE) for Susquehanna SES to compile a few important facts relative to HPCI importance from a risk perspective. Two events of particular interest arise from this perspective, the "isolation" (i.e., turbine bypass is unavailable) ATWS and Station Blackout. The expected frequency of this ATWS event with HPCI unavailable is 6.5×10^{-5} per year. This is on the same order as the expected frequency of the HPCI Steamline break over the four month period (see attached Safety Assessment). However, no core damage is expected with the HPCI Steamline break event, while complicating the isolation ATWS by rendering HPCI unavailable results in a factor of 7 increase in core damage frequency (CDF). The Station Blackout event is expected at a frequency of 2×10^{-4} per year. HPCI would not be able to be recovered in this event since the F002 valve is AC powered. A factor of 10 increase in CDF results from HPCI unavailability during this event.

The information above has led the HPCI system to be identified as one of the most important systems to maintain available at Susquehanna. Prior to the occurrence of the torque switch problem, administrative controls were in place to ensure that in the event HPCI was declared inoperable, around-the-clock efforts would be made to restore HPCI availability rather than to use the 14-day action time provided. Given the current circumstances and the justification provided above, we again urge the NRC to take immediate action to restore HPCI availability.

Any questions on this request should be directed to Mr. R. Sgarro at (215) 770-7916. Pursuant to 10CFR170, the appropriate fee is enclosed.

Very truly yours,

B D Kenyon

Bruce D. Kenyon
Senior Vice President-Nuclear

Attachments

cc: NRC Document Control Desk (original)
NRC Region I
Mr. L. R. Plisco, NRC Resident Inspector
Mr. M. C. Thadani, NRC Project Manager
Mr. T. M. Gerusky, Pa. DER

SAFETY ASSESSMENT FOR

OPERATION WITH HPCI F002

STEAM SUPPLY ISOLATION VALVE

WITH

INCORRECT TORQUE SWITCH SETTING

SAFETY ASSESSMENT FOR OPERATION
WITH HPCI F002 STEAM SUPPLY ISOLATION
VALVE WITH INCORRECT TORQUE SWITCH SETTING

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I. Introduction And Problem Statement

On May 7, 1987, it was discovered that the "closing" torque switch on the Unit #1 HPCI F002 Steam Supply Isolation Valve was incorrectly set to 1 instead of the required setting of 3 (see NCR 87-0187). The setting of 3 is required to close the valve fully under the maximum ΔP that the valve would experience (i.e. 1000 psid) and corresponds to a thrust load of 27,000 lbs. The thrust load generated by a torque switch setting of 1 would be 8,000 lb, which would correspond to a ΔP across the valve of 141 psid. A momentum/inertial load of approximately 3,000 lb would be present under all conditions experienced by the valve and torque switch settings.

The torque and limit switch logic is such that the torque switch setting does not trip the motor until the limit switch is tripped. The limit switch is set to trip when the valve is 97% closed. Once the limit switch has tripped, the motor is then controlled by the torque switch. In the case of the torque setting of 1, the motor would be tripped immediately following the limit switch reaching 97% closed for all situations where the valve would be experiencing a ΔP greater than 140 psid. Based on this, engineering judgement was used to estimate the resulting flow area that would be present with the valve 97% closed. This was done in the absence of firm data with regard to the actual position of the valve at 97% closed but took into account closing/seating data from similar valves. Based on this, the resulting flow area would be approximately 0.25 in^2, which would be equivalent to the valve disc being 1/8" off the seats.

Consequently, a flowrate of 24,550 lbm/hr of steam would be possible with a 1000 psi ΔP across the valve at 97% closed, as would be the case in a HPCI line break accident. This would result in a unisolatable leak of steam outside primary containment in the unlikely event of the outboard steam supply isolation valve (F003) failing to close. This paper will show that the consequences of this unlikely event do not exceed 10CFR100 limits for off-site dose and safe operation of the reactor is possible, until the torque switch setting can be corrected during the Unit #1 3rd RIO.

II. Functional/Safety Requirements Of F002 Valve

The HPCI Steam Supply Isolation Valves F002 (Inboard) and F003 (Outboard) are normally open valves. This is to allow steam to fill the lines up to the turbine steam admission valves which is just upstream of the HPCI turbine. This allows the steam lines to remain warm and reduces the chances of a steam hammer when the HPCI system is called upon in an emergency. Therefore, these valves must remain open at all times during reactor operation in order for the HPCI system to be available for use to mitigate an accident.

There are two situations which require the valves to be closed: 1) To provide containment isolation for a LOCA inside containment where HPCI does not receive a start signal; 2) To automatically isolate the reactor in the event of a HPCI steam line rupture. A discussion of the consequences of the HPCI F002 valve torque switch setting of 1 with regard to the two closure requirements will be provided in Section III.

III. Assessment Of Failure To Fully Close On Demand

For the case where the F002 valve must close for a LOCA inside the containment, the torque switch setting of 1 would not prevent the valve from fully closing since the ΔP would be significantly less than 140 psid. Additionally, the LLRT from the 2nd RIO indicates that when seated under a "No-Load Condition" the valve passed the LLRT. For the situation of a HPCI steam line break, if the break were to be one where the pipe developed a crack (i.e., leak before break) the valve would be able to fully close since the flow would be choked at the crack and not the valve. However, for a large break (i.e., double-ended guillotine), the valve would not close due to the high ΔP. This condition is discussed in the following sections.

IV. Probability Of An Unisolatable HPCI Line Break

As previously discussed, the only time the F002 valve will fail to completely isolate is when a high ΔP exists across it. This situation is indicative of a HPCI steam line break. If the F003 valve closes, then the break is isolated and no hazard exists to the general public.

The case of concern then involves a HPCI line break outside the containment with a failure of the F003 valve to isolate. This can occur in two ways. First, the line break could cause the F003 valve to also fail. Second, the pipe could break and the F003 valve could fail independently of the pipe break. Each is considered below.

There are three ways of having a HPCI line break coincident with a failure of the F003 valve: the valve could rupture, the weld connecting the valve to the containment penetration could fail, or the pipe section contiguous with the valve could fail in a manner which would also fail the valve. Failure frequencies for each of these cases are available in the Reactor Safety Study, WASH-1400, and they are listed below.

Valve Rupture	1.0×10^{-8}/hr
Weld Failure	3.0×10^{-9}/hr
Pipe Section Rupture	1.0×10^{-10}/hr
Total	1.3×10^{-8}/hr

There are about 2900 hours until the outage. Thus, the chance of a coincident HPCI pipe and F003 valve failure becomes 1.3×10^{-8}/hr x 2900 hr = 3.8×10^{-5}.

If the pipe failure occurs downstream of the F003 valve, then it must fail randomly to cause the HPCI line break to be unisolated. There are 31 pipe sections and 32 welds between the F003 valve and the HPCI turbine. The chance of an unisolated HPCI line break becomes the product of the line break and the valve failures. This becomes:

Weld Failure	32 welds x 3.0×10^{-9} $\frac{\text{weld}}{\text{hr}}$	$= 9.6 \times 10^{-8}$/hr
Pipe Ruptures	31 sections x 1×10^{-10}/hr	$= 3.1 \times 10^{-9}$/hr
	Total	$= 9.9 \times 10^{-8}$/hr

The random failure of the F003 valve is assessed at 5.6×10^{-3}/demand (NUREG/CR-1363). Therefore, the chance of a pipe break and a random failure of the F003 valve causing an unisolated HPCI line break becomes:

$$(9.9 \times 10^{-8}/\text{hr} \times 2900 \text{ hr}) \quad 5.6 \times 10^{-3}/\text{demand} = 1.6 \times 10^{-6}$$

The total chance from both failure mechanism becomes:

$$3.8 \times 10^{-5} + 1.6 \times 10^{-6} = 4.0 \times 10^{-5}.$$

This is on the same order as a large LOCA.

Therefore, this can be considered an unlikely event.

V. HPCI Steam Line Break Evaluation

A. Break Scenario

The 10" DBB-114 HPCI steam supply line penetrates the primary containment in the RHR equipment/valve room on Floor Elevation 683' and Area 28 of the Reactor Building. From this room, it proceeds downs into the HPCI pump/turbine room on Elevation 645' of the Reactor Building. Both rooms have direct access to the Reactor Building vent via blowout panels which are designed to relieve pressure at 0.5 psig. The failure of the HPCI steam supply line will be a double-ended guillotine break, concurrent with a failure of the Division II 250 VDC power supply. The failure of the power supply will not allow the HPCI outboard steam supply valve (F003) to close. The failure of Division II power supply will also cause the "B" loops (Division II) of RHR and Core Spray Systems to be unavailable for shutting down the reactor. This would place the full burden of isolation on the F002 valve, resulting in the maximum leakage, and disable only Division II ECCS Systems. Division I ECCS Systems are protected from failure of Div. II 250 VDC, and are still available for mitigating the consequences of the accident.

A rupture of the HPCI steam supply line in both the RHR equipment room and the HPCI pump has been evaluated in Section 3.6A of the FSAR. This evaluation was performed for structural integrity purposes and only considers the release of steam for the period until the isolation valves go closed. However, this would still be applicable to the event considered here, since the F002 valve would stroke closed but not fully seat.

Based on Section 3.6A of the FSAR, the amount of steam released during the period from break initiation until the valve achieves 97%

closed would be approximately 22,000 lbm. During this blowdown, the peak pressure would be 2.12 psig which would rupture the blowout panel in the room. The blowdown for the HPCI pump room, as evaluated in Section 3.6A of the FSAR, is slightly less at 15,000 lbm with a peak pressure of 4.11 psig. Once the valve achieves 97% closed, the blowdown rate is estimated to be 24,000 lbm/hr at 1000 psig across the valve, thus resulting in an unisolatable leak of steam outside of primary containment. The leak would continue until the reactor pressure is reduced below 140 psig or until the valve can be closed manually by overriding the torque switch at the local MCC. A discussion of the consequences of the leak from a mitigation and radiological standpoint is addressed below.

B. Leak Mitigation

1. HPCI Pump Room

As indicated earlier, the break could occur in one of two rooms; the RHR equipment room or the HPCI pump room. If the break were to occur in the HPCI pump room, the affected equipment would be localized to the HPCI system. Sufficient break detection exists to automatically close the F002 valve and alert the Control Room operator of a break in the HPCI pump room. However, since the HVAC system serving the HPCI room does not have isolation dampers, a portion of the leaking steam will enter the Reactor Building HVAC system and could be spread to adjacent areas, assuming a duct failure in the vicinity of the HPCI pump room. Lack of HVAC (BDID) isolation dampers in the HPCI pump room is consistent with the design criteria governing use of BDIDs for protection of ECCS Systems. The objective of the BDID System is to protect one division's electrical and mechanical equipment from the effect of a high energy pipe break outside of the primary containment which may have disabled electrical or mechanical equipment of the other division. This lack of BDIDs in the HPCI pump room concurrent with an assumed duct failure in an adjacent room has been evaluated with the conclusion that HPCI (ECCS Div. II) equipment may be disabled due to the subject pipe break, but ECCS Div. I equipment in adjacent rooms would not be affected. The steam would be diluted and should not pose an environmental threat to equipment required for safe shutdown of the plant. During the period of steam leakage from the F002, the Control Room will be receiving indication in the Control Room from the HPCI steam flow meters and would indicate an unisolatable leak from the HPCI steam line. This would direct the operator to the EOP and result in a manual reactor scram. A personnel radiation hazard could also exist which may prevent plant personnel from reaching the MCC that would permit overriding the torque switch and manually closing the F002 valve (Note: MCC is located just outside the door to RHR equipment room). Therefore, it may be necessary to manually initiate the ADS system to depressurize the RPV so that the break could be isolated. It should be noted that the emergency procedures would quickly direct the operator to initiating ADS if HPCI were.

unavailable and the leak could not be isolated. It is estimated from simulator training of a HPCI steam line break that the operator would identify the HPCI line break and initiate ADS in 10 to 20 minutes from the time the line break occurs. The total blowdown during this period would be approximately 4,000-6,000 lbm. The depressurization would take place over an additional 3-4 minute period and would release an additional 500 lbm of steam. Once the RPV is depressurized below 140 psig, the valve could be closed, thus isolating the leak. The vessel could then safely achieve cold shutdown since all normal and emergency low pressure sources of water would be available, as well as the main condenser for heat removal. No fuel failure would occur since the reactor core would never become uncovered during the event. An assessment of the off-site doses consequences will be addressed below in Section V.C.

2. <u>RHR Equipment Room</u>

For a break in the RHR equipment room, the leak mitigation and off-site doses would be the same as discussed above except that the total release of steam would be slightly larger and equipment required for safe shutdown other than HPCI related would be affected by the leak. The off-site dose assessment is presented in Section V.C. and is for a leak in the RHR equipment room since it would bound the HPCI pump room leak. The operator response times indicated above would also apply since similar detection devices are present for a leak in the RHR equipment room.

The equipment in the penetration room was reviewed for adverse impact of HPCI steam leakage. All safety related equipment in the room is qualified for High Energy Line Break (HELB) environmental conditions. These conditions endure for 60 seconds until the line break is isolated. The only equipment required to operate after 60 seconds to safely shut down the reactor are motor operators on ECCS valves. These valve operators were all qualified for environmental conditions in excess of HELB requirements. The environmental conditions expected in the penetration area following 60 seconds do not exceed the qualification parameters for the operators. This is due to the large volume of the room and the sensible heat capacity of surfaces, structures and equipment located inside. The surface temperature of equipment in the room should not exceed 212°F except in the immediate area of the break. However, equipment in this area is already postulated to fail due to the loss of Division II of 250 VDC power. Therefore, equipment supplied by the remaining division will be available for safe shutdown.

The RHR equipment room is equipped with isolation dampers; however, the design pressure for two of the ten dampers are not consistent with the peak pressure seen in the room. However, these two dampers were tested to at least 4 times the design pressure and showed no structural damage or excessive leakage. Therefore, credit can be taken for the damper to isolate. As identified above, safe shutdown of the reactor would be achieved.

C. Dose Associated With An Unisolated Pipe Break

The dose assessment was performed as follows:

1. Estimate the reactor coolant activity and assume this activity is homogenously mixed in both liquid and steam.

2. Estimate the mass expelled during the line break.

3. Estimate the mass expelled through the break during the depression phase. An iodine spike will be applied during depressurization.

4. Based on 1-3, estimate the iodine source term.

5. Compute the site boundary 2-hour thyroid dose.

Each of this is discussed below.

The coolant activity was assumed to be the same as that assumed in the main steam line break. All the iodine is assumed in the liquid phase. As the liquid phase flashes, the iodine becomes airborne.

After determine the coolant activity, the mass expelled must be estimated. For the 10 minutes prior to the depressurization, it is estimated that 24,500 lbm is released through the break. Since a rapid depressurization is not occurring, iodine spiking was not considered in this phase. During the depressurization phase, iodine spiking was assessed at 500 times the coolant activity (NUREG-0803). The mass flow was estimated using moody critical flow of saturated steam and a flow area of 0.25 square inches. It requires 3.3. minutes to depressurize to 100 psia (NEDO-24951). The mass flow during the depressurization was assumed to be the average of the maximum and minimum flow. The F002 valve is assumed to be closed 20 minutes after the break occurs and thus release is terminated. This release is assumed to be released instantaneously and directly to the environment.

The 2-hour site boundary dose is computed using standard health physics equations as discussed in the FSAR. The 2-hour site boundary dose is 6.9 Rem, with ninety percent of the dose coming from the iodine spike. It was also conservatively assumed that, following initiation of ADS at 10 minutes into the accident, depressurization and closing of the F002 valve took 10 minutes. The iodine spiking was assumed to occur during this 10 minute interval. Through this assumption, the operator is allowed 6.7 minutes to close the F002 valve. This allows the operator more than enough time to close the valve once the RPV is depressurized, knowing the condition which exists on the F002 valve.

VI. Conclusions/Recommendations

Based on the above discussions, safe operation of the plant with the HPCI F002 valve torque switch set at 1 is justified until the Unit #1 3rd RIO with the F002 and F003 valves opened to support HPCI availability. Furthermore, based on the probability of the pipe break occurring vs. the likelihood of needing the HPCI system, it is less safe to have HPCI inoperable by closing the F002 and F003 valves than to leave the isolation valves open.

In addition to the above conclusions, NPE makes the following recommendations for operation until the Unit #1 3rd RIO.

1. If the unit is shutdown for a containment outage prior to the Unit #1 3rd RIO, the torque switch should be reset to the proper value (i.e. a setting of 3).

2. The Control Room operators should be made aware of the problem with the F002 valve, so that in the unlikely event of a HPCI steam line break, they could take prompt action to depressurize the reactor to allow closure of the valve.

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