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Docket No. 50-213
B11730

Director of Nuclear Reactor Regulation
Attn: Mr. John A. Zwolinski, Chief
Operating Reactors Branch #5
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

References: (1) W. G. Counsil letter to D. M. Crutchfield, dated May 19, 1980.

(2) D. M. Crutchfield letter to W. G. Counsil, dated October 5, 1982.

(3) D. E. Vandenburgh, Chairman External Review Group, letter to T. E. Murley, dated September 6, 1985.

Gentlemen:

Haddam Neck Plant
Auxiliary Feedwater System Status

Introduction

Earlier this month, a number of questions had been posed by the Staff regarding the Auxiliary Feedwater System. Following discussions with NRR personnel and the Resident Inspector, we have agreed to provide this letter to address the concerns raised, and to document the basis for our continued position that the Auxiliary Feedwater System ensures that Technical Specifications and design basis requirements for the Haddam Neck Plant are fulfilled. Three areas of concern raised are addressed, in turn, below.

Design Basis Analysis

Reference (1) provided the design basis analysis for the Auxiliary Feedwater System in response to TMI Action Plan Item II.E.1.2 and to support a plant modification which installed automatic initiation of the auxiliary feedwater system. This was found acceptable as documented in NRC's SER (Reference (2)) The analysis reevaluated the "Loss of Feedwater Event", the limiting event used as the design basis of the AFW auto initiation system, to support the modification of the Auxiliary Feedwater System. The analysis showed that given a "Loss of Feedwater Event", with automatic Auxiliary Feedwater (AFW) initiation or manual AFW initiation within 10 minutes, there were acceptable consequences in terms of the maximum pressure the RCS would attain. One of

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the acceptance criteria was based on a pressure value that would not challenge the PORV setpoint of 2285 psia. This is more restrictive than the SRP acceptance criterion of 110% of the design pressure, or 2750 psia. The more restrictive maximum pressure criterion was chosen to minimize the challenges to the PORV's. Additionally, CYAPCO noted that given the worst single failure, determined to be loss of one AFW pump, only two steam generators were required to be operable post trip. This statement was based upon a hand calculation which showed that 318.2 gpm of AFW flow is needed to remove decay heat and heat from four Reactor Coolant Pumps (RCP's) 15 minutes after reactor trip. This statement was submitted primarily to provide an indication of the conservatism in the overall design, not to indicate that the design basis requires consideration of three random failures.

Subsequent to this submittal, additional analyses were performed to evaluate the effects of:

(1) The tripping of two Reactor Coolant Pumps post-reactor trip,

(2) a reduction in AFW flow resulting from a design change on the steam driven pumps, and

(3) three-loop operation.

The assumption made in Reference (1), and in the original FDSA analysis was that following a reactor trip all four RCP's continue to run. The actual plant configuration is such that following a turbine trip caused by a reactor trip there is a 52 second delay before station power buses transfer to offsite power.[1] At that time two RCP's are tripped leaving only two RCP's running. The trip of two RCP's following reactor trip has an effect on the loss of feedwater transient since it results in reverse flow through the steam generators in the idle loops. This results in less effective heat transfer to the generators in the idle loops. The AFW flow reduction was caused by a change to the inlet steam pressure to the AFW turbine as a result of a test of the auto-AFW system.

The analyses subsequently performed to reflect the correct RCP trip scheme predict higher RCS peak pressures, but without lift of pressurizer safety valves. Although these analyses predict higher system pressure than stated in Reference (1), they showed that the major conclusions of Reference (1) were not altered and all applicable acceptance criteria continue to be met. The NRC was not specifically informed of these new analyses nor was any Licensee Event Report submitted, since there was nothing to report. Further details regarding these subsequent analyses are summarized below.

<u>Four Loop Operation</u>

In this case, most of the decay heat is removed by the two "active" steam generators in the loops having forced flow. The following observations are made when comparing these results to the results reported in Reference (1).

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[1] This matter was discussed briefly in the J. F. Opeka letter to J. A. Zwolinski, dated August 18, 1985.

- o the inventories in the "active" generators boil down sooner, and
- o the primary system pressure increase is larger.

The effects of the RCP trip and lowered AFW flow rate produce acceptable results for either automatic or manual initiation of AFW at 10 minutes. Results of analyses performed assuming the lower AFW flow rates indicate that two steam generators may not be sufficient to remove decay heat and pump heat after 15 minutes even if this feedwater were supplied to two "active" steam generators. Detailed analyses of this scenario were not performed in that it would be equivalent to assuming three random failures (one auxiliary feed pump and two steam generators lost).

Three Loop Operation

For the case of RCP trip following reactor trip while in three loop operation, the most severe loss of feedwater event occurs when the isolated loop contains a RCP that would normally be running following reactor trip (i.e. loop 2 or 4). When this is the case, there is a forward flow through the loop with the operating RCP and reverse flow through loops 1 and 3. As a result, most of the decay heat load is placed on the single steam generator in the loop with the running RCP. Because of the heat load, the inventory in that steam generator boils down rapidly. The inventories in the steam generators in the loops 1 and 3, (containing the idle RCPs) boil down much more slowly. Since the auto-initiation signal for AFW utilizes two-out-of-three logic, auto-initiation was predicted to occur later than 10 minutes. As a result, the auto-initiation of AFW results in higher calculated peak pressures, (PORVs were predicted to lift) than a similar case assuming operator action to start AFW at 10 minutes. However, even without crediting PORV operation, calculated peak pressures remain below the acceptance criterion of 110% of the design pressure and safety valves do not lift.

Number of Steam Generators Required to Remove Decay Heat

As a result of the reduction in the AFW flow rate and the effects of tripping two of the reactor coolant pumps, the statement made as an aside in Reference (1), that with one AFW pump operable, only two steam generators are required to be operable to ensure that enough AFW flow is delivered to remove decay heat and pump heat 15 minutes after reactor trip, can no longer be made. For 4-loop operation, the present design basis analyses show that injecting AFW into all four steam generators produces acceptable results. The effects of having fewer than four operating steam generators with both AFW pumps have not been specifically analyzed.

Similarly in 3-loop operation, the present analyses show that with one AFW pump operable, injecting into the three operable steam generators produces acceptable results. The effects of having fewer than three operating steam generators with both AFW pumps have not been specifically analyzed. This is consistent with the single failure assumption of loss of one AFW pump made in the FDSA, whereby other potential single failures concurrent with the availability of both AFW pumps were not considered.

The availability of fewer than four steam generators while in four loop operation and fewer than three steam generators while in three loop operation with two operable AFW pumps is being evaluated as part of the FDSA Chapter 10 reanalysis effort. This corresponds to assuming that a single random failure (other than an AFW pump) will reduce the number of steam generators available for decay heat removal.

Reanalysis Philosophy

As documented in NUREG-0826, the Integrated Plant Assessment Report for the Haddam Neck Plant, CYAPCO is committed to the performance of a reanalysis upgrade for all the FDSA Chapter 10 analyses. This effort is far more extensive than that required by the Staff as a result of the SEP review, and will ultimately represent a substantial upgrade in the analysis status for this plant. In the interest of providing some perspective, the number of analyses expected to be conducted at the conclusion of this effort will be an order of magnitude increase as compared to the number of analyses performed as part of the original plant licensing effort during the 1960s. The analyses involve conformance to a much larger percentage of the standard review plan criteria and assumptions, which substantially increase the conservatism associated with analyses involved.

Given the scope and depth of this reanalysis, it is not surprising that the calculated consequences of certain accidents will more closely approach the acceptance criteria. The analyses completed to date have demonstrated that on some occasions, this in fact occurs. It is emphasized however, that a reanalysis which shows more severe consequences but still within acceptance criteria do not automatically represent either significant safety issues or cause for reportability to the NRC under 10CFR50.72 and 50.73. As we have communicated via discussions with the NRC, changes in calculated consequences which are more limiting than the licensing bases analyses but are still within acceptance criteria are not planned to be submitted prior to completion of the total effort. It is only when these calculations reveal the existence of errors in licensing bases analyses or exceedance of acceptance criteria that reporting and corrective actions are necessary. Changes in calculated consequences which are the direct result of different, SRP-type initial conditions and assumptions are not planned to be submitted prior to completion of the total effort which we plan to submit in mid-1986.

Postulated Loss of Control Air to the AFW System

The loss of control air on the automatic initiation system of the auxiliary feedwater system was identified by the Connecticut Yankee Plant Design Change Task Group (CYPDCTG) as a potential deficiency. The nature of the concern raised is that loss of control air may result in overspeed of the steam-driven turbines. CYAPCO recognizes its obligation to address the issue raised, and notes that it has not yet been substantiated that upon loss of control air, overspeed would occur. Hence, no immediate safety problem exists. For the concern to materialize, an automatic AFW demand would have to occur simultaneously with a loss of control air event. We currently view this to be unlikely but sufficiently plausible (e.g. seismic event) to warrant additional consideration. Further, even if the event were to occur, ample time exists to

take corrective measures long before decay heat removal capability is threatened. The judgement we have exercised in not taking more prompt action on this potential deficiency reflects the fact that the postulated event will not prevent the auxiliary feedwater system from being manually initiated and performing its decay heat removal function.

The PDCTG had in place procedures to assure prompt action was taken on issues of more immediate safety significance. To illustrate this point, we call your attention to the containment isolation issues raised in LER #85-17(2). However, in this particular instance, prompt reporting and implementation is not warranted for reasons summarized above. Hence, we plan to continue to conform to the requirements of the December, 1984 Order(3). As required by the Order:

"Within two months after the date of issuance of the final appraisal report required by Section IV.B. of this Order, the licensee shall submit to the Regional Administrator, Region I, for review and approval, a plan for improvements based on evaluation of the appraisal findings and recommendations. This plan shall include (1) action items to be performed and (2) a schedule for completion of each specific action item. This plan shall also provide justification if any of the recommendations of the appraisal report are not adopted."

The final appraisal report required by Section IV.B of the Order was transmitted to you via Reference (3). Accordingly, our evaluation and plan for corrective actions on this deficiency will be forwarded on or before November 6, 1985.

AFW Wiring Deficiency

On November 2, 1984 during the surveillance test for the auxiliary feedwater system, 2 main feedwater bypass valves failed to open. The cause was determined to be sticking solenoid operated valves (SOV's). To alleviate the problem, per manufacturer's recommendations more frequent cycling of the valves was proposed. This was accomplished by implementing Preventative Maintenance Procedure (PMP)-9.2-45. (Note: As a result of the failed bypass valves, LER #85-005 was written.)

During the performance of this preventative maintenance procedure, fuses protecting circuits which actuate the SOV's for the feedwater bypass valves and AFW Terry Turbine system steam admission valves were individually removed to deenergize the circuits and cause a change in position of the SOV's and corresponding opening of the feedwater bypass valves and steam admission valves. An error was discovered when a fuse thought to be associated with the (A) Terry Turbine steam admission valve was pulled resulting in startup of the (B) Terry Turbine. Further investigation revealed that a similar cross-connection problem existed in two of the four AFW system bypass valves. Additionally, during the performance of the procedure, one of the four solenoid

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(2) LER #85-17 discusses our evaluation of two newly identified potential release paths from containment, a concern originating from work performed by the PDCTG.

(3) James M. Taylor letter to W. G. Counsil, dated December 13, 1984.

operated valves (SOV's) for the feedwater bypass valves failed to change position and allow the bypass valve to open. This later problem is related to our LER #85-005 described event. The attached wiring diagram (sketch) illustrates the details of the error which caused the improper valve operations.

The following is a chronology of events from implementation of the PDCR which added the automatic initiation feature to the auxiliary feedwater system through to the final dispositioning of interim splicing.

- o The original plant design change was a result of a post-TMI requirement to incorporate auto-initiation of the auxiliary feedwater system. The initial design change incorporated a control grade system which would later be upgraded to safety grade. The majority of the modifications centered around SOV and actuation logic circuitry installation.

- o During the following refueling outage, the automatic initiation system was upgraded to safety-grade. This involved upgrades to the steam generator level instrumentation and the SOV's. With regard to the SOV's, complications arose in wiring the replacement SOV's because their electrical pigtails were shorter than those on the original SOV's. A Design Change Notice (DCN) was initiated to allow the interim use of an intermediate section of spliced-in wiring. Subsequent to the SOV change-out, including wiring, a QC inspection revealed that identification tags were not in place on the SOV's or their wiring. We believe that a tagging error, as opposed to a wiring error, occurred at this time. Thus at this time the system was installed in full compliance with the design, however with SOV's (4 out of 6) mislabeled.

- o During the 1983 refueling outage, final disposition with respect to the spliced-in sections of wiring was accomplished by a rewiring of the SOV's. It is our postulation that the wiring error occurred at this time as a result of the complete unwiring, and then, rewiring of the SOV's to facilitate splice elimination. At this time, as at present, the SOV ID tags were in the improper position in 4 of 6 instances. Wiring was done to the improperly tagged valves in accordance with design drawings resulting in the cross-connections being incorporated.

Based on the above, we believe the design was correctly incorporated initially. A review of the design drawings show they accurately reflect the intended design. A surveillance test performed on the auxiliary feedwater system did not detect the error because <u>overall system operability was not and is not affected</u>. No divisional separation was violated by the wiring error since it occurred in the wiring between main control board terminal blocks and the adjacently mounted SOV's. Rewiring and relabeling of the solenoids will be corrected during the upcoming refueling outage to be consistent with the original design intent.

Additional efforts are in progress to identify any appropriate corrective actions necessary to prevent recurrence. We will be providing additional details on these efforts directly to the Resident Inspector as they become available.

We note that this error was not identified by the Plant Design Change Task Group (PDCTG) because the task group did not perform or cause to have performed a systematic field verification of circuitry. The PDCTG did include a detailed review of this design change and its related drawings, which as described above, were correct. Further, preoperational and surveillance testing were reviewed and found to be adequate. The error has no effect on system operability. Hence, it was reasonable for the PDCTG to not pursue field verification of the wiring and tubing.

Conclusion

While the issues discussed above have apparently introduced some confusion and questions from the Staff, the bottom line is that the auxiliary feedwater system and its automatic initiation scheme is fully operable and in conformance with all design basis requirements. On this basis, no specific reports were provided to the NRC, except for LER #85-005 and the August, 1985 informational letter. Follow-up actions remain in progress regarding the design basis reanalysis effort, resolution of the PDCTG concerns and correction of the labeling and wiring problem. We will continue to adhere to our commitments and reporting obligations as these efforts continue.

The unique characteristics of the defect in question here, specifically that the wiring problem did not impact system operability, serve to shield identification of the problem by tests designed to reveal the presence of problems affecting system operability. Hence, this problem does not void the credibility of these independent investigations. While it cannot be guaranteed that no system internal errors exist, the fact that they are not revealed by system tests is generally indicative of the absence of significant errors.

Recent significant strengthening of the controls governing design changes and preoperational testing requirements provide increased assurance that safety significant defects will not occur. Importantly, periodic surveillance tests and the inherent defense-in-depth concept exist to protect against potential unknown problems.

On the other hand, the discovery of this "latent" wiring problem and the thoroughness of the subsequent investigation provides additional insurance as to the integrity of our personnel and the soundness of our current testing programs.

We are hopeful that this letter will resolve all recent Staff questions regarding the auxiliary feedwater system, and we remain available to provide any further clarification that may be required.

Very truly yours,

CONNECTICUT YANKEE ATOMIC POWER COMPANY

J. F. Opeka

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E. J. Mroczka

By: E. J. Mroczka
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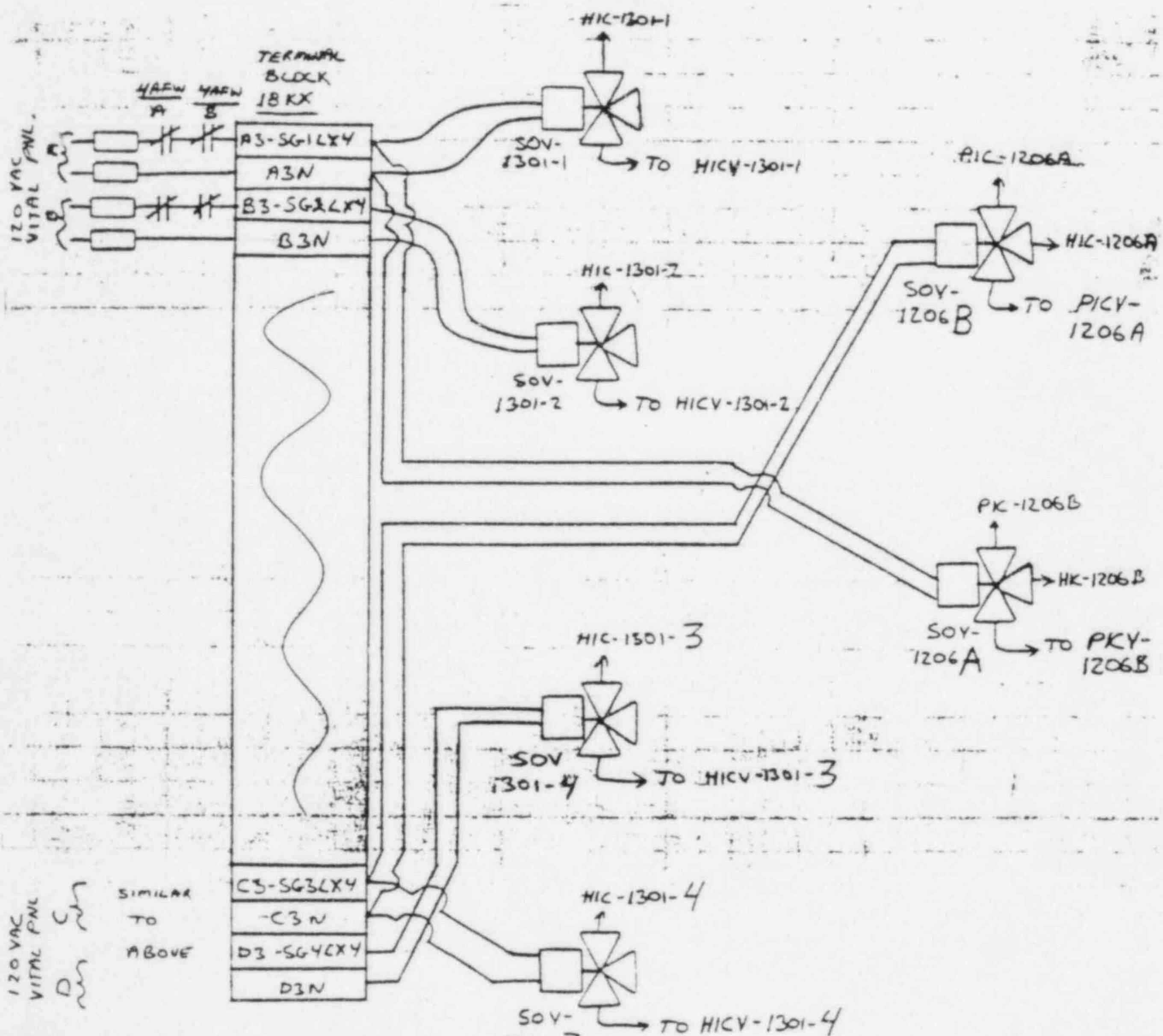
cc: Dr. T. E. Murley

Wiring Diagram
Auxiliary Feedwater Automatic
Initiation Solenoids



Haddam Neck Plant
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