FERA

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

May 7, 1980

Docket No. 50-213

Mr. W. G. Counsil, Vice President
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Dear Mr. Counsil:

Enclosed is the staff's evaluation of the implementation of Category "A" Lessons Learned requirements (excluding 2.1.7a) at the Haddam Neck Plant. This evaluation is based on your submitted documentation and discussions held between our staffs on April 10, 1980.

Based on our evaluation, we conclude that the implementation of the Category "A" requirements at Haddam Neck is acceptable except for one open item, 2.1.4 "Containment Isolation." Certain items, identified in the evaluation, will be verified by the Office of Inspection and Enforcement.

This evaluation does not address the Technical Specifications necessary to ensure the limiting conditions for operation and the long-term operability surveillance requirements for the systems modified during the Category "A" review. You should be considering the proposal of such Technical Specifications. We will be discussing this item with you in the near future.

With respect to the above identified open item, you are requested to respond within 14 days from the receipt of this letter with regard to your intention to meet the requirements, specified in our evaluation, for further action to resolve the open issue.

Sincerely,

Dennis M. Crutchfield

Dennis M. Crutchfield, Chief
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Enclosure:
Evaluation

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# EVALUATION OF CATEGORY "A" LESSONS LEARNED IMPLEMENTATION

# BY THE OFFICE OF NUCLEAR REACTOR REGULATION

# HADDAM NECK

# DOCKET NO. 50-213

## Introduction

By letters dated December 31, 1979, January 31, and April 11, 1980, Connecticut Yankee Atomic Power Company (the licensee) submitted documentation of the actions taken at the Haddam Neck Plant (the plant) to implement the requirements resulting from TMI-2 Lessons Learned.

## Evaluation

Details of the NRC's Category "A" requirements and acceptance criteria are documented in NUREG-0578 and NRC letters dated September 13 and October 30, 1979. The number designation of each item is consistent with the identifications used in NUREG-0578.

### 2.1.1 Emergency Power Supplies

#### Pressurizer Heaters

The pressurizer heater power is supplied by four 480 V buses. Upon a loss of offsite power, all four of these buses are energized by the emergency diesel generators. Backup heater capacity of 305 kw is provided on each train. This capacity is sufficient to maintain natural circulation in hot standby conditions. All the heaters are load shed on a safety injection actuation signal. The time required to connect the pressurizer heaters to the emergency buses is consistent with the timely initiation and maintenance of natural circulation. The circuit breaker controls for the backup heaters are safety related. Procedures for manual reconnection of the pressurizer heaters are in the control room.

#### PORVs and Block Valves

There exists the capability to supply the PORVs and the block valves from either the offsite power supply or the emergency power supply. The PORVs require air pressure to operate. The air compressor has the capability of being powered from either offsite power or the onsite emergency power supply. Safety related devices are used for the motive and control power electrical connections.

Pressurizer Level

There are three principle pressurizer level instrument channels. Each channel has an indicator on the control board. Each channel is powered from a vital bus.

The licensee meets the Category "A" requirements of Item 2.1.1.

2.1.2 Relief and Safety Valve Testing

The licensee has committed to participate with the NSSS Owners Group and the Electric Power Research Institute in the development of a solution to this concern. This satisfies the Category "A" requirements of NUREG-0578.

2.1.3.a Direct Valve Indication

The licensee has installed an acoustic monitoring system supplied by Babcock & Wilcox (B&W). The sensor channel detects the valve flow noise and converts it to an AC voltage signal by providing suitable amplification and filtering. The conditioned signal is then converted to a DC level which drives a visual display meter and actuates the alarm.

The licensee has installed one acoustic monitor for the two PORVs and three safety valves. Each PORV has an existing stem mounted limit switch to indicate valve position in the control room. Using a single acoustic monitor downstream of the PORV and safety valves, it is possible to identify the source of the alarm. When a PORV lifts indication will be provided by both the stem mounted limit switch and the acoustic monitoring system. When a safety valve lifts, only the acoustic monitoring system will annunciate. Nevertheless procedure requires the operator to isolate the PORV relief paths whenever the acoustic monitor alarms and pressure falls below the reseat setpoint.

The acoustic monitoring system will be qualified to IEEE Standards 323, 344 and 383 in a B&W program. Safety grade qualification should be complete by April 1981. The current qualification of the limit switches is under review. The licensee has scheduled to have qualified limit switches in place by January 1, 1981. Power to the valve position indication circuits can be supplied from offsite or the emergency diesel generators.

The licensee meets the Category "A" requirements for Item 2.1.3.a.

2.1.3.b Instrumentation for Detection of Inadequate Core Cooling

By participation in the Westinghouse Owners Group, the licensee has reviewed events which have the potential for causing inadequate core cooling. These results have been documented in response to IE Bulletin 79-06C. The Westinghouse Group has determined that there is sufficient information from existing instrumentation to detect inadequate core cooling. Also additional procedural guidance has been developed by the Owners Group. The licensee has updated emergency procedures and operator training based on these guidelines.

The licensee has evaluated the Owners Group solutions to a reactor vessel water level indication system and states that he has no plans to install a reactor vessel level meter at this time. Adequacy of the licensee's procedures and his position on additional instrumentation will be reviewed by the NRC staff.

Subcooling Meter

The licensee has installed a CE subcoo' margin monitor to provide continous digital display of temperat' or pressure margin to saturation. Five in core thermocouples with a range of 100-700°F provide temperature input. The highest temperature of the five is used for the calculation. Two reactor coolant system pressurr inputs with a range of 0-3000 psig are used as pressure inputs.

The subcooled margin monitor is designed to IEEE Standards 344-1975 and 323-1974. The pressure transmitters used are not qualified as safety grade equipment. The licensee has on order, two new pressure transmitters and intends to install these during the 1980 refueling outage, subject to receipt of equipment. Saturation curves and procedures for their use are available in the control room.

The licensee meets the Category "A" requirements for Item 2.1.3.b.

2.1.4 Containment Isolation

The NRC requirements are that the licensee is to: (a) carefully reconsider their determination of which system should be considered essential or non-essential for safety; (b) modify systems as necessary to isolate all non-essential systems by automatic, diverse, safety grade isolation signals; and (c) modify systems as necessary to assure that the resetting of the containment signals does not cause the inadvertent reopening of containment isolation valves.

The licensee's December 31, 1979, January 31, and April 11, 1980 submittals identified the essential and non-essential systems and the bases for the essential systems classification. Non-essential systems are isolated on diverse signals consisting of a safety injection signal and a high containment pressure signal.

The design of the control system for automatic containment isolation valves has been modified to prevent the reopening of the isolation valves while resetting the isolation signal.

The isolation control circuit has been modified so that all but 16 automatic isolation valves are reopened on an individual basis. The reset features for these remaining 16 valves are such that they are reset in groups from five Pilot Solenoid Valves. These valves are to be modified so that reopening is on a valve-by-valve basis. This modification is to be made during the next refueling outage. The refueling outage is currently scheduled to begin on May 3, 1980.

The modification will result in the requirement that 14 control switches be placed in the closed (safe) position to allow reset of the containment isolation signal. With this design the inability to close any one of the 14 control switches would render all of the associated isolation valves inoperable. The licensee stated in their April 11, 1980 submittal th' one group of the

affected systems may be beneficial during post-accident conditions to bring the plant to stable conditions. This is the group of systems that could affect Reactor Coolant Pump (RCP) operations (i.e., RCP seal water, RCP cooling water to the bearing oil coolers and RCP Thermal Barriers cooling water). The licensee stated in their April 11, 1980 submittal for Item 2.1.6.a that Connecticut Yankee Atomic Power Company has "...concluded that post-accident operation of the RCPs could indeed be beneficial...". They also stated that the postulated control failure could be circumvented by jumpering.

The staff informed the licensee, during our April 10, 1980 discussions, that the use of jumpers in Class 1E equipment to correct the postulated failure is unacceptable. The licensee is evaluating several alternative solutions to the above problem. They have committed to advise the staff of the results of their investigation before restart from the upcoming refueling outage. This outage is scheduled to begin May 3, 1980.

The staff has concluded that modifications made to implement the requirements of Item 2.1.4 should not, subject to a failure, preclude the use of systems deemed "potentially beneficial" in mitigating an accident. Therefore, this problem should be resolved prior to restart from the upcoming refueling outage. In all other respects the licensee's actions satisfy the Category "A" requirements of Item 2.1.4.

2.1.5 Dedicated Penetrations for External Recombiners or Post-Accident External Purge System

The NRC's position is that dedicated containment isolation systems should be used for the external recombiners or purge systems that meet redundancy and single failure requirements.

There are two (2) systems that the licensee could use to reduce combustible gas concentrations within the containment. The primary means of purging is the containment air particulate monitoring system and the backup system is through the bypass line on the containment purge system. The use of these systems for combustible gas control is described in the licensee's December 31, 1979 submittal.

Based on our review of the licensee's combustible gas control systems, we have concluded that their system meets the NUREG-0578 Section 2.1.5 requirements for dedicated penetrations and are, therefore, acceptable.

2.1.6.a Systems Integrity

The licensee has provided a list of those systems which he has determined may contain radioactivity following an accident. These systems include the high pressure safety injection and charging systems in the recirculation mode, residual heat removal, sampling and instrumentation systems. He has committed to include the hydrogen purge system in the program. He has also provided a description of the immediate leak reduction program which included walk down inspections to identify leakage, cleanup and repair of these systems.

The licensee has completed measuring final system leak rates and reported the results.

The licensee has established a preventative maintenance program for systems which may contain activity following an accident which includes testing once per refueling cycle to ensure that essential systems are maintained leaktight.

Our October 30, 1979 clarification letter requested the licensee to include a review of potential release paths due to design and operator deficiencies as discussed in the October 17, 1979 letter regarding North Anna. The licensee has analyzed their plant with regard to the North Anna Incident and scheduled corrective actions as deemed necessary.

Based on the above information, we conclude that the licensee has met the Category "A" requirements for this item.

2.1.6.b Plant Shielding Review

The licensee's December 31, 1979 submittal includes a design review of plant shielding. The licensee has performed the design review assuming the systems identified in Item 2.1.6.a contain radioactivity. The licensee has used the source term as specified in the October 30 letter for his review. The licensee has determined high radiation areas and identified components which may be affected. They have also identified areas where access may be required. For these areas, corrective actions will be taken to assure that the necessary functions can be performed. The licensee has proposed to defer the corrective actions until the staff has reviewed the evaluation and assessed the overlap of the effort with the continuing SEP review. The staff will perform a detailed evaluation of the submittal at a later date and will judge at that time the merits of deferring corrective actions to SEP.

The licensee has not included an evaluation of the environmental qualification of equipment as this is scheduled as an SEP topic. The currently scheduled date for the SEP environmental qualification submittal is June 2, 1980. Since the licensee is already undertaking an equipment qualification review, we will not require a further review and will evaluate the licensee's SEP submittal with regard to Lessons Learned.

We conclude that the licensee has satisfied the Category "A" requirements for this item.

2.1.7.b Auxiliary Feedwater (AFW) Flow Indication

The licensee has installed control grade AFW flow instruments for each steam generator. In order to satisfy the single failure criteria, the licensee will use the existing Steam Generator Level Indication System as a backup. The AFW flow instrument channels are capable of being powered from normal offsite or the emergency diesel generators. Accuracy of the AFW flow is within the requirement of $\pm 10\%$.

The licensee meets the Category "A" requirements of Item 2.1.7.b.

2.1.8.a Post-Accident Sampling

The licensee's December 31, 1979 submittal contains a design review of the plant sampling capability for primary cooland and containment air samples assuming a source as specified in NUREG-0578.

The licensee has implemented interim procedures for obtaining and analyzing reactor coolant and containment atmosphere samples with the existing equipment. The procedures include provisions for keeping occupational exposures as low as reasonably achievable. A copy of the procedure was included in the January 31, 1980 submittal.

The licensee's interim procedures did not include provisions for performing a hydrogen or gross gas analysis of the reactor coolant sample or for performing an isotopic noble gas or hydrogen analysis of the containment atmosphere sample. The licensee has committed to incorporate provisions for the above analyses in the existing procedures by April 30, 1980. IE will assure that these modifications have been made.

The licensee has not provided conceptual designs for reactor coolant and containment atmosphere monitoring in order to meet the Category "B" requirements for this item. However, the licensee has committed to provide the design on or about April 29, 1980. We find this schedule acceptable.

Based on the above information, we conclude that the licensee has satisfied the Category "A" requirements for this item.

2.1.8.b High Range Radiation Monitors

The licensee has implemented interim procedures and installed equipment for the quantification of noble gas effluents released from the plant stack and the main steam safety relief and atmospheric dump valves if the existing instrumentation goes offscale. The licensee has identified these as the final release points which include all other individual sources.

The noble gas release rates will be determined using detectors located at preselected monitoring locations. Detector readings can be converted to exhaust concentrations as specified using the existing procedures. IE will assure that the procedures for monitoring of the steam dump valves are in effect.

The licensee has also implemented and provided a description of the interim system/method to be used to determine radioiodine and particulate effluents.

Based on the above information, we conclude that the licensee has met the Category "A" requirements for this item.

2.1.8.c Improved Iodine Instrumentation

The licensee has designated a single cart mounted monitor to fulfill this function for both the control room and the Technical Support Center (TSC).

Since these areas share the same ventilation system, a single monitor is adequate. The monitor will be equipped with a silver impregnated silica-gel cartridge and will allow for continuous readout and alarm.

The emergency operations center will be monitored using a standard air sampler equipped with a silver impregnated silica-gel cartridge. Following sample collection, the cartridge will be counted using a hand probe.

Procedures for use of the equipment in all areas are in place.

Based on the above information, we conclude that the licensee has met the requirements for this item.

2.2.1.a Shift Supervisor Responsibilities

The NRC requirements for this item are to revise, as necessary, the responsibilities of the Shift Supervisor such that he can provide command oversight of operations and perform management review of ongoing operations that are important to safety.

The licensee has issued management directives and revised their Plant Procedures, APM 1.1-1-c, Rev. 8 in response to the staff's requirements. We have verified that the licensee's management directives and administrative procedures adequately address this position.

We conclude that the licensee has satisfied the requirements of NUREG-0578, Item 2.2.1.a, for delineation of Shift Supervisor responsibilities.

2.2.1.b Shift Technical Advisor

The NRC requirement is for the licensee to provide an on-shift technical advisor (STA) to the Shift Supervisor to serve the two functions of accident assessment and operating experience assessment. As a supplement to the operating staff, the STA must be available to the control room to assist in diagnosing an off-normal event.

The licensee has implemented a program, described in Station Policy CYSP-30-6, to provide an onsite STA to provide the shift operating crew with an independent accident assessment capability. In addition, programs have been established at both Connecticut Yankee Atomic Power Company and Northeast Utilities Service Company to provide the required operating experience assessment function. The Connecticut Yankee Atomic Power Company and Northeast Utilities Service Company programs are coordinated with the STA requirements to ensure close coupling of the STA accident assessment and the operating experience assessment program.

We have reviewed the licensee's December 31, 1979 submittal describing their STA programs. We find that their STA program is in agreement with the staff's requirements described in Section 2.2.1.b of NUREG-0578 and is, therefore, acceptable.

2.2.1.c Shift and Relief Turnover Procedures

The NRC requirement is for the licensee to assure that procedures are adequate to provide guidance for a complete and systematic turnover between the off-going and on-coming shift to assure that critical plant parameters are within limits and that the availability and alignment of safety systems are made known to the on-coming shift.

The licensee has revised their Plant Operating Procedures, ADM 1.1-44, and developed the required shift turnover logs and checklists.

Based on our discussions with the licensee, we have concluded that they have satisfied the requirements of Item 2.2.1.c related to shift turnover procedures. Adequacy of the checklists and logs will be performed by IE and will be documented by appropriate Inspection Reports.

2.2.2.a Control Room Access

Procedures exist that establish the authority and responsibility of the person in charge of the control room to limit access to those who may be requested or required to support operations. In addition, procedures establish a line of authority in the control room and limit those in charge of the control room during an accident to persons possessing a current SOL.

The licensee has satisfied the requirements of this item.

2.2.2.b TSC

A TSC has been established in the operations supervisor's office, adjacent to the control room. A black and white video system capable of scanning the full control board as well as access to the plant computer provide monitoring capability of plant parameters. Plant operating procedures, diagrams and other pertinent documentation is available. Dedicated communications exist which allow simultaneous communications with the control room, NRC and near-site emergency operations center. Procedures exist which define activation and staffing of the TSC. The TSC and control room share a common ventilation system. The licensee has provided additional details regarding the long term TSC.

Based on the above, we find the licensee has met the Category "A" requirements for this item.

2.2.2.c Onsite Operational Support Center (OSC)

An OSC has been established within the viewing gallery outside the control room. The OSC is equipped with a communications system to the control room.

Based on the above, we find the licensee has met the Category "A" requirements for this item.

NRR Reactor Coolant System Venting

The licensee has proposed a design for venting of the reactor coolant system in fulfillment of the Short-Term Lessons Learned Requirement.

Conclusion

Based on the above, subject to IE verification as noted, we find that implementation of the Category "A" Lessons Learned Requirements, except for 2.1.4, is acceptable. Item 2.1.4 requires further licensee action before it can be found acceptable.

Dated: May 7, 1980