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

February 25, 1980

Docket No. 50-336

Mr. W. G. Counsil, Vice President
Nuclear Engineering & Operations
Northeast Nuclear Energy Company
P. O. Box 270
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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 Millstone, Unit No. 2. This evaluation is based on your submitted documentation and the discussions between our staffs at a site visit on January 14 and 15, 1980.

Based on our evaluation, we conclude that the implementation of the "Category A" requirements at Millstone, Unit No. 2, is acceptable. 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 in communication with you on this item in the near future.

Sincerely,

Robert W. Reid, Chief
Operating Reactors Branch #4
Division of Operating Reactors

Enclosure: As stated

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MILLSTONE UNIT 2

EVALUATION OF CATEGORY "A" LESSONS LEARNED

IMPLEMENTATION

Introduction

By letters dated December 31, 79 and January 31, 1980, Northeast Nuclear Energy Company (NNECO or the .icensee) submitted documentation of the actions taken at Millstone, Unit No. 2 (the plant) to implement the requirements resulting from TMI-2 Lessons Learned. To facilitate our review of the licensee's actions, members of the staff visited the plant on January 14 and 15, 1980.

Evaluation

Our evaluation of NNECO's method of implementing each of the "Category A" requirements and acceptance criteria is 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 supply is provided from redundant Class IE busses. A bank of proportional heaters is connected to both 480V safety related trains. Each bank of heaters has a rating of 160 KW. The adequacy of 160 KW of pressurizer heater capacity for natural circulation was supported by the relevant portion of Combustion Engineering Report CEN-125.

During normal operation, these busses and their heater banks are powered by the normal or reserve station service transformers. During a loss of offsite power the proportional heaters are shed from their respective busses. Once these loads have been shed, they can only be reenergized by manually closing the circuit breaker control switches on the control board after other emergency loads have been reenergized. During a LOCA, with coincident loss of offsite power, the heater loads are shed. However, the emergency diesel generators have sufficient capacity to support the heater loads in addition to the LOCA loads. The diesels have a continuous rating of 2750 KW and the maximum LOCA loading is 2247 KW. Therefore, sufficient margin exists to accommodate the 160 KW heater load.

The control system for the pressurizer heater banks is powered from Class IE busses. We conclude that the pressurizer heater power supply is in conformance with the position in NUREG-0578.

Pressurizer Pilot Operated Relief Valves (PORVs) and Block Valves

The licensee has stated that both the PORVs and their associated block valves have the capability of being powered from either offsite power or emergency power if offsite power is not available. The PORVs have 125 VDC operated solenoid valves and are powered via redundant safety related 125 VDC busses which receive power from Class IE busses. The redundant emergency busses are powered by the station service transformers or by the emergency generators during loss of offsite power. The control power for the PORVs is taken from the same 125 VDC power source that supplies the solenoids.

The PORV block valves are motor operated and are powered from redundant 480 VAC safety related busses. Both the PORVs and the isolation valves are connected to the emergency power sources in accordance with safety grade requirements. Each PORV and its associated block valve are powered from the same emergency power supply.

Based on the above, we conclude that the requirements of this item have been met.

Pressurizer Level Instrumentation

The pressurizer level instrumentation consists of two redundant and physically separated pressurizer level indicators (LIC-110X and LIC-110Y). The indicators are powered by redundant 120 VAC busses which are normally supplied with offsite power or onsite emergency power when offsite power is unavailable. This satisfies the requirements for this item.

2.1.2 Relief and Safety Valve (SV) Testing

The licensee has committed to participate with the NSSS Owner's Group and the Electric Power Research Institute program in the development of a solution to this concern. This satisfies the "Category A" requirements of this item.

2.1.3a Direct Indication of PORV and Safety Valve Positions

The licensee has installed an acoustic monitoring system supplied by Babcock & Wilcox (B&W). The acoustic monitoring system has four channels, one for each of the PORVs and SVs. Each channel consists of an accelerometer, a preamplifier and a monitoring unit in the control room. The accelerometers are mounted 6" downstream of each valve except for one safety valve in which the accelerometer is mounted 12" downstream. The valve discharge induced vibration will excite the accelerometer producing an alarm light on the Reactor Protective System (RPS) panel in the control room. An annunciator alarm at the main control board is also generated.

The acoustic monitoring system will be qualified to IEEE 323, 344 and 383. The licensee has agreed to have the qualification completed by April 1, 1980.

All channels of the acoustic monitoring system are powered from a single Class 1E vital instrument bus. Isolation of the acoustic monitor from the Class 1E bus is provided by the use of fuses as an interim method. The use of fuses in isolating nonsafety grade equipment from a Class 1E power supply is not in accordance with Regulatory Guide 1.75. However, since the acoustic monitors are only connected to one of the redundant Class 1E power supplies and there is a high degree of confidence that the equipment is qualifiable, the interim use of fuses as isolation devices is acceptable.

The licensee has backup procedures for the use of temperature elements and quench tank pressure, temperature and level to aid operator action and diagnosis of valve position.

We find that the use of acoustic monitors, as implemented by the licensee, meet the requirements of this item.

2.1.3b Instrumentation for Inadequate Core Cooling

The "Category A" requirements of this item are that the licensee develop and implement procedures to check the degree of reactor coolant subcooling using existing instrumentation, install a subcooling meter, and submit the proposed design of any additional instrumentation (e.g., reactor vessel water level) to enhance the indication of inadequate core cooling.

Review of the plant operating procedures revealed instructions to the operators to both monitor the degree of subcooling, using the subcooling meter plant computer and steam tables as backup, and maintain the reactor coolant in a subcooled condition. Curves are provided which indicate saturated conditions as well as various margins of subcooling.

The licensee has installed a Subcooling Margin Monitor (SMM), designed by Combustion Engineering, with continuous digital indication in the control room. This SMM uses one T-hot instrument input from each loop. To provide redundant T-hot input to subcooled margin monitoring and display, the licensee relies on the plant computer. The plant computer has continuous indication of subcooled margin and uses as input different T-hot instrument inputs, one from each loop.

The licensee is currently reviewing the qualifications of the temperature inputs and will make modifications if existing instruments are not qualified.

Present subcooling margin is calculated in the range of 515-665 F. As a backup the licensee can use incore thermocouples with a range of 70-2500 F. The licensee also intends to increase the range of inputs to the SMM in conjunction with modifications made as a result of the qualification program.

The licensee has met the "Category A" requirements of Item 2.1.3b. Additional work in this area may be necessary as a result of analyses performed for Item 2.1.9 of NUREG-0578.

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 re-opening of containment isolation valves.

The licensee's submittal included a table of the essential and non-essential systems and the bases for the essential system 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 prevents the reopening of the isolation valves while resetting the isolation signal. Furthermore, the reopening of the isolation valves requires deliberate operator action. To prevent inadvertent reopening, the system design includes a three position spring-return-to-neutral switch. In addition, each valve has "seal-in" relays which maintain the valve in the closed position following containment isolation reset. Therefore, the operator must deliberately turn each individual hand switch to the "open" position after the isolation signal is reset. Our conclusion is that the licensee's containment isolation design meets the NUREG-0578 Section 2.1.4 containment isolation requirements and is therefore acceptable.

2.1.5a 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. This requirement does not apply to the licensee since recombiners located wholly within the containment are used.

2.1.5c Recombiner Procedures

The NRC's position is that the procedures for use of the recombiners be reviewed considering shielding requirements and personnel exposure limitations.

The plant utilizes recombiners located inside the containment. Controls for operating the recombiners are located inside the control room. During the site visit we discussed the licensee's review of the recombiner operating procedures and agreed that no modifications are required.

We have concluded that the licensee has met the NUREG-0578 requirements for review of the recombiner procedures, Section 2.1.5.c.

2.1.6a Systems Integrity

The licensee has provided a list of those systems which he has determined may contain radioactivity following an accident. These systems are the safety injection, containment spray, shutdown cooling, containment sump recirculation, and reactor coolant sampling systems. 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 also measured and reported the final system leak rates to the NRC.

The licensee has established a preventive maintenance program for the systems which may contain radioactivity following an accident. This program includes pressure testing once per refueling cycle, review of system design and construction to ensure the potential for release is minimized, and a review of procedures to assure that leak tightness is emphasized.

The licensee has also stated that should their review of transient and accident analysis (item 2.1.9) indicate that other systems could be used following an accident, they will incorporated these systems into their leak reduction program.

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 action as appropriate.

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

2.1.6b Plant Shielding Review

The licensee's December 31, 1979 submittal included a design review of plant shielding and environmental qualification of equipment. The licensee has performed the design review assuming the systems identified in Item 2.1.6a contain radioactivity. As specified in the October 30 letter, the licensee has used the source term for his review. The licensee has determined high radiation areas and identified components which may be affected. The licensee has stated that affected components will be fully qualified by Janaury 1, 1981, if equipment is available. 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. A detailed evaluation of the submittal will be performed at a later date. We conclude that the licensee has met the "Category A" requirements for this item.

2.1.7b Auxiliary Feedwater Flow Indication

Currently one auxiliary feedwater (AFW) flow channel exists for each steam generator. The associated indications in the control room indicate from 0-300 gpm. To satisfy the single failure requirements, the licensee relies on the existing steam generator level indication system. The licensee states that the AFW flow instruments are testable, powered from vital buses and have an accuracy within $\pm$ 10%.

We have reviewed the licensee's submittal of December 18, 1979 on this subject and have confirmed during the site visit that there is AFW flow and steam generator level indication in the control room.

We find that the licensee meets the intent of this Lessons Learned item.

2.1.8a Post-Accident Sampling

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

NNECO's January 31, 1980 submittal indluded a copy of the Millstone, Unit No. 2 procedures for atmosphere and gaseous effluent samples following a postulated accident. The licensee has also provided procedures for performing the required analysis on the samples.

The licensee has contacted two vendors who will supply sampling system proposals by March 1, 1980. The licensee will review the vendor proposals and select the optimum program. The licensee has stated that he plans to submit the final design to the NRC by April 1, 1980. This schedule will allow for NRC review prior to the final implementation date of January 1, 1981. We find this schedule acceptable.

2.1.8b High Range Radiation Monitors

NNECO has implemented interim procedures and installed equipment for the quantification of noble gas effluents released form the Unit 1 and Unit 2 stacks as a result of an accident at the Unit 2 plant.

In our review of this item we found the licensee had not committed to monitor noble gas releases from the atmosphere steam dump valves. Following discussions with the staff, the licensee has committed to provide for monitoring of noble gas releases up to 10E3 /cc from the steam dump valves. He has further committed to have the portable equipment and associated procedures available by February 29, 1980. The Office of Inspection and Enforcement will verify that the equpiment and procedures are available for use.

The licensee has provided a description of his interim system/method to be used to determine radioiodine and particulate effluents and therefore meets the requirements of item 2.1.8.b for radioiodine and particulate monitoring.

2.1.8.c Improved Iodine Instrumentation

The licensee has designated portable air samplers to fulfill this function. The sampling media is a silver impregnated silica gel cartridge. This system meets the requirements of NUREG-0578. The licensee has also provided assurance that all areas occupied by essential personnel (both control rooms, technical support center and operational support center) will be monitored. Therefore, we conclude that the licensee meets the requirements of NUREG-0578, Item 2.1.8.c.

2.2.1.a Shift Supervisor Responsibilities

The NRC requirement for this item is 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's submittal is responsive to our requirements. In addition, during the staff's site visit we verified that the licensee's management directives and administrative procedures adequately addressed 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 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 NNECO and Northeast Utilities Service Company (NUSCO) to provide the required operating experience assessment function. The NNECO and NUSCO 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 submittal describing their STA programs. In addition, during the site visit we discussed the program with the licensee and determined that a satisfactory STA program is in operation. We find that their STA program is in agreement with the staff's requirements described in Section 2.2.1.c 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's submittal indicated that checklists and logs have been provided which satisfy our acceptance criteria. Further, he has established a system to evaluate the effectiveness of the shift turnover procedure.

During the site visit our check of the revised shift turnover procedure checklists and logs confirmed that the licensee has addressed this position.

We conclude that the licensee has satisfied the requirements of Item 2.2.1.c related to shift turnover procedures. Adequacy of the checklists and logs will be performed by the Office of Inspection and Enforcement and will be documented by appropriate Inspection Reports.

2.2.2a Control Access

The authority and responsibility of the person in charge of the control room to limit access is established in Administrative Control Procedure (ACP) 6.01 "Control Room Procedure". Specifically, in addition to persons authorized to enter the control room, access is limited to those who have a need to enter the control room. The authority to limit access is given to the Shift Supervisor or, in his absence, the Supervising Control Operator.

Operating Procedure 2501 established lines of authority and communication both in the control room and between the control room and other designated support areas outside the control room to be used during an accident. ACP QA 6.01 also establishes the presence and authority of the Shift Supervisor as the person in charge of the control room during an accident. This procedure also specifies by title which personnel with Senior Reactor Operator licenses may relieve the Shift Supervisor.

Based on our review of these procedures, we find that the licensee has satisfactorily implemented this item.

2.2.2b Onsite Technical Support Center (TSC)

The licensee has established a TSC directly adjacent to the Unit 2 control room in the computer room. NNECO has developed plans and procedures for engineering/management support and staffing of the TSC. Installed in the TSC are dedicated communications between the TSC, control room, and near site emergency operations center. The TSC also has an extension to the dedicated communication "hot-line" to the NRC.

The TSC is served by the same ventilation system that serves the control room. The capability to monitor direct radiation is provided by portable monitors which are presently available onsite. The licensee has developed procedures that assure monitoring capability for direct and airborne radiation.

The TSC has the capability to directly monitor any plant parameters available from the plant computer on a CTR display. A high resolution closed circuit TV, remotely controlled from the TSC, gives the TSC personnel the capability to directly monitor the control boards. In the TSC is a microfiche file of plant drawings. A microfiche reader is available in the TSC. A long range plan for upgrading the TSC has been submitted for our review.

We have determined that the licensee has implemented the "Category A" requirements of this item.

2.2.2c Onsite Operational Support Center (OSC)

The licensee has designated an area directly adjacent to the control room as an OSC. The emergency plan has been revised to reflect the existence of the OSC and establish methods and lines of communication and management. We find the OSC to be acceptable.

NRP Reactor Coolant System Venting

The licensee has proposed a design for venting of the reactor vessel head and the pressurizer in fulfillment of the Short-Term Lessons Learned requirement.

Conclusion

Based on the above, subject to our Office of Inspection and Enforcement verification as noted, we find that implementation of the "Category A" Lessons Learned requirements at Millstone, Unit No. 2, is acceptable.

Dated: February 25, 1980