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

April 29, 1980

Docket No. 50-309

Mr. Robert H. Groce
Senior Engineer - Licensing
Maine Yankee Atomic Power Company
25 Research Drive
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Dear Mr. Groce:

Enclosed is the staff's evaluation of the implementation of Category "A" Lessons Learned requirements (excluding 2.1.7.a) at Maine Yankee Atomic Power Station. This evaluation is based on your submitted documentation and the discussions between our staffs at a site visit on February 27, 1980.

Based on our evaluation, we conclude that the implementation of the Category "A" requirements at Maine Yankee 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,

Morton B. Fairtile for

Robert W. Reid, Chief
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Enclosure:
Category "A" Evaluation

cc w/enclosure:
See next page

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MAINE YANKEE

EVALUATION OF CATEGORY "A" LESSONS LEARNED

IMPLEMENTATION

Introduction

By letters dated December 28, 1979 and January 7, 10 and 22, February 22, and March 5, 1980 Maine Yankee Atomic Power Company (the licensee) submitted documentation of the actions taken at Maine Yankee (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 February 27, 1980.

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 supply design provides the capability to supply power from either the offsite power source or an emergency power source when the offsite power source is not available. There is sufficient heater capacity, 150 kw on each train of emergency power, to maintain natural circulation in the hot standby condition. Each bank of heaters is connected to an independent power supply by a Class 1E circuit breaker. The pressurizer heaters are automatically shed from the emergency power source upon initiation of a safety injection signal. Procedures are in the control room for the manual reconnection of the pressurizer heaters to the emergency buses.

Pressurizer PORVs and Block Valves

There are two PORVs in parallel, each with an associated motor operated block valve. The two 480V AC motor operated block valves are powered from 480V AC emergency motor control centers - one from each emergency power train. Each PORV is operated by 480V AC solenoids connected to the same emergency motor control center as its associated block valve.

Both the PORV and block valves are connected to the emergency bus in accordance with safety grade requirements.

Pressurizer Level Instrumentation

The pressurizer level instruments are powered by an emergency power source and therefore, capable of being supplied from either the offsite power source or the diesels when offsite power is not available.

Based on the above, we conclude that the requirements of 2.1.1 have been met.

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 Indication of PORV/Safety Valve Position

The licensee has installed an acoustic monitoring system supplied by Babcock and Wilcox (B&W). The acoustic monitoring system has one channel for each PORV and one channel for the three safety valves which have a common discharge header. Each channel consists of an accelerometer, a preamplifier and a monitoring unit in the control room. The accelerometer is mounted downstream of the valve. The valve discharge induced vibration will excite the accelerometer producing an alarm light on the monitor unit in the control room. In addition, an audio alarm is initiated on a monitor unit in the control room.

The acoustic monitoring system will be qualified to safety grade criteria. Completion of the qualification program is expected by fall of 1981.

We find that the use of the acoustic monitoring system for direct valve indication, as implemented by the licensee, meets the requirements of 2.1.3.a.

2.1.3.b Instrumentation for Inadequate Core Cooling

The licensee has installed a subcooled margin monitor designed by Combustion Engineering (CE). The subcooled margin monitor is a micro computer based instrument which continuously displays the margin to saturation. It is designed for use as a post-accident monitoring instrument. The monitor is qualified as safety grade.

The monitor provides the operator with continuous digital display of either the pressure or temperature margin to saturation conditions. Two core exit thermocouples, in different core quadrants, provide temperature input to the monitor. The range of the thermocouples is 0-750°F and the highest temperature is used for the margin calculation. The licensee has committed to install three safety grade RTD's, one in each hot leg, to be used as input to the monitor. These will replace the core exit thermocouples and will be installed by January 1, 1981. Three safety grade pressure inputs, covering the range of 0-3250 psig, are used by the monitor. A subcooled margin alarm is provided.

In addition to the above, the licensees' plant computer continuously displays margin to saturation using temperature inputs different from those used by the subcooled margin monitor. Steam tables and procedures covering their use are also available to the operator.

The licensee is pursuing various reactor vessel level measurement alternatives through the CE Owners Group. A bouyant type device and a differential pressure measurement are being studied. The licensee is following the CE submittal of the heated junction thermocouple concept. The licensee will evaluate the various concepts and will elect one or more for level measurement. The elected concepts will be submitted to the NRC with the intent of installing the equipment by January 1, 1981.

Based on the above, we conclude that the licensee's implementation of this Category "A" item is acceptable.

2.1.4 Containment Isolation

The NRC requirements are that the licensee: (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 resetting of the containment isolation signal does not cause the inadvertent reopening of containment isolation valves.

The licensee's submittals of December 28, 1979 and March 5, 1980 identified the essential and non-essential systems and the bases for the essential system classification. Modifications were made so that non-essential systems are isolated on diverse signals consisting of a safety injection signal and a high containment pressure signal.

The design of the isolation valve control system was modified to prevent the reopening of the isolation valves following the resetting of the isolation signal. The new design requires the control room operator to not only reset the isolation signal, but also to reset the circuitry for each specific isolation valve. Furthermore, the circuitry for each valve incorporates its own reset switch and seal-in relay.

We conclude that the licensee's containment isolation design meets the NUREG-0578, Section 2.1.4 containment isolation requirements and is therefore acceptable.

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

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

The licensee utilizes a hydrogen purge system as described in their December 28, 1979 and March 5, 1980 submittals for post accident hydrogen control. During a March 17, 1980 phone conversation between the licensee and the staff, the licensee committed to modify this system to meet the staff's NUREG-0578 Section 2.1.5.a single failure and redundancy requirements. The modifications, will be installed on a schedule consistent with our Category "B" January 1, 1981 requirements.

We have reviewed the hydrogen purge system and the proposed modifications to the system and find that it meets the requirements of NUREG-0578, Section 2.1.5.a. (Section 2.1.5.b is not applicable to Maine Yankee.)

2.1.5.c Recombiner Procedures

The licensee has reviewed its procedures for hydrogen purge and has determined that modifications such as will be done for Item 2.1.6.b (Shielding Review) and/or the requirement for a containment hydrogen concentration indicator are necessary. These modifications will be made by January 1, 1981. The licensee has met the requirements of Item 2.1.5.c.

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 (HPSI), low pressure safety injection (LPSI), containment spray, residual heat removal, letdown, purification, charging, waste liquid, seal water supply and return, primary sampling, post accident purge, containment air particulate and gas monitoring and waste gas systems.

The licensee has provided final system leak rates and is investigating corrective actions for one of the systems to assure that the system leakage will be minimal.

In a phone call on March 18, 1980 the licensee committed to incorporate a long term leakage reduction program which will assure system integrity on a refueling cycle frequency.

The new program must be put into the plant procedures prior to the next scheduled refueling outage. IE will assure that the appropriate procedure revisions have been made.

Our October 30, 1979 clarification letter requests 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 concluded that corrective actions are not necessary.

Since the licensee has committed to implementing a leak reduction program which specifies testing at refueling cycle intervals, and because he has completed the immediate leak reduction program, we conclude that the licensee has met the requirements of this item in an acceptable manner.

2.1.6.b Plant Shielding Review

The licensee's submittal dated March 5, 1980 contains a preliminary design review of plant shielding. The licensee has committed to provide a more detailed shield design review including calculated dose rates and required fixes by June 1980. The design review was performed using the clarification provided in the October 30, 1979 Denton letter as guidance. The analysis assumes that radioactivity is contained in those systems which will not automatically isolate following an accident. Our more detailed review will determine if other systems which may carry radioactivity will have to be included.

The licensee has identified those operations which may involve personnel exposure following an accident and identified corrective actions where necessary.

The licensee has not provided a review of the environmental qualification of safety equipment located outside containment. However, the licensee has indicated that this will be done as required by IE Bulletin 79-01B. The licensee has agreed in a phone call on March 18, 1980 to provide this review with the June 1980 shielding design review.

During our review of Section 2.1.4 it was noted that certain DC operated isolation valves cannot be reopened from the control room. Some of these valves are part of systems whose operability may be beneficial following an accident. The licensee is currently reviewing means to assure that radiation levels will not preclude operation of these valves. This review is scheduled to be completed by April 30, 1980 with modifications for the valves in potentially beneficial systems to be complete by June 30, 1980.

A detailed evaluation of the submittal will be performed at a later date. We conclude that the licensee has implemented the Category "A" requirements for this item in an acceptable manner.

2.1.7.b Auxiliary Feed Flow Indication

Control grade auxiliary feed flow instrumentation has been provided for each steam generator. This instrumentation while not environmentally qualified has been seismically qualified to IEEE 344-1971 and IEEE 344-1975. Although this instrumentation is not fully qualified there is a high degree of confidence that it is qualifiable without modification. These instruments will be powered from a single vital bus and will be connected via a single dedicated breaker. Therefore, any instrument fault causing the breaker to trip will not affect other vital equipment on the bus and any voltage or current spike generated in the equipment and transmitted back through the breaker prior to its tripping could only affect one of the vital busses. As a backup to this flow instrumentation steam generator level instrumentation is available that reads out in the control room.

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

2.1.8.a Post-Accident Sampling

The licensee has performed a design review of the plant post-accident sampling capability for primary coolant and containment air.

The licensee has committed to implementing procedures for obtaining and analyzing pressurized and unpressurized reactor coolant and containment air samples utilizing the existing equipment. The new procedures include provisions for maintaining personnel exposures ALARA while obtaining and analyzing the sample. IE will assure that the new procedures are in place and training has been completed.

The licensee has indicated that modifications or design changes will be necessary to meet the January 1, 1981 requirements. They have committed to provide the proposed new sampling design by May 1, 1980.

Based on the above, we conclude that licensee implementation of this requirement is acceptable.

2.1.8.b High Range Radiation Monitors

The licensee has installed instrumentation with remote readout capability to quantify noble gas release rates from the plant stack if the existing instrumentation goes offscale. The steam dump valves will be monitored using portable instrumentation with an individual making in-situ readings and verbally reporting to the control room. The licensee has also indicated that procedures have been developed to determine release rates of up to 10,000 Ci/second. The licensee has stated that procedures include provisions for minimizing personnel exposures, calculational methods, reporting of results and instrument calibration. IE will assure that the specified equipment is readily available and the procedures are in place.

The licensee has stated that procedures for obtaining and analyzing radioiodine and particulate samples of plant effluents following an accident are included in current plant procedures. IE will assure that the procedures are in place.

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

2.1.8.c Improved Iodine Instrumentation

The licensee has proposed to take airborne samples using a standard air sampler which will draw the air through a charcoal cartridge. The cartridge will then be counted using one of the three existing plant GeLi systems. The licensee has stated that the procedure is in effect and includes a requirement for the dedication of one plant GeLi system. The licensee has stated that the control room, and technical support center (TSC) will be adequately sampled. The proximity of the analysis room to the control room and TSC will ensure that samples can be analyzed in a timely manner.

Our Office of IE will assure that the equipment is available and procedures are in effect.

Based on the above, we conclude the licensee has met the requirements of this item.

2.2.1.a Shift Supervisor (SS) Responsibilities

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

During our site visit of February 27, 1980 we reviewed the management directives dealing with SS responsibilities that were issued by the licensee. In addition, the licensee's submittal of March 5, 1980 includes a copy of the appropriate revised administrative procedures. These documents satisfy our requirements.

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

2.2.1.b Shift Technical Advisor (STA)

The NRC requirement is for the licensee to provide an on-Shift Advisor to the SS 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 off-normal events.

The licensee has implemented the program described in their March 5, 1980 submittal which establishes an onsite STA to provide the shift operating crew with an independent accident assessment capability. The STA's will also fulfill the required operating experience assessment function.

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.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 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 requirement.

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 IE and will be documented in appropriate Inspection Reports.

2.2.2.a Control Room Access

The licensee has procedures that establish the authority for the person in charge of the control room to limit access during an off-normal incident. Additionally, procedures also specify the line of succession for personnel in charge of the control room and limit these personnel only to those holding a Senior Reactor Operators License.

2.2.2.b Technical Support Center (TSC)

The TSC has been established and consists of the plant computer center and the second floor of the Technical Support Building. Procedures have been inplemented which provide for engineering/management support and staffing of the TSC. Dedicated communications have been provided to the NRC and to the control room and Emergency Operations Facility. Since the plant computer room is part of the TSC, access to plant parameters is provided. The portion of the TSC that is in the Technical Support Building contains plant drawings and other up-to-date plant design documents. The licensee has implemented procedures to transfer the function of the TSC to the plant computer room or the control room should a portion of the TSC become uninhabitable. The upgrading of the TSC to meet the long term requirements has been addressed.

2.2.2.c Onsite Operational Support Center (OSC)

The OSC has been established and has been provided with communications to the control room. The OSC is located in the security guards' offices. Procedures have been developed that delineate the lines and methods of communication and management.

We conclude that the Category "A" requirements of 2.2.2 are satisfied.

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 our Office of IE verification as noted, we find that implementation of the Category "A" Lessons Learned Requirements at Maine Yankee are acceptable.

Dated: April 29, 1980