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

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Docket No. 50-293

JUL 8 1982

MEMORANDUM FOR: H. R. Denton, Director, ONRR

FROM: R. J. Mattson, Director, DSI/ONRR

SUBJECT: GENERIC IMPLICATIONS OF THE RELEASE OF SPENT DEMINERALIZER RESINS FROM PILGRIM, UNIT NO. 1

Reference: PNO-I-82-42/42A

The release of radioactive spent resins from the Pilgrim Power Station, reported in PNO-I-82-42, June 11, 1982, has been reviewed for generic implications in accordance with your request. Based on information in the PN and its update of June 14, 1982, on information in the docket file, and on information obtained in telephone discussions with Region I representatives, a licensee representative, and the Operating Project Manager (DL), it is our conclusion that there are several related factors in this incident which have both generic and licensee - specific implications. These are discussed in items (1) through (5) below.

(1) It is probable that the resins observed and reported in the PN originally escaped from operations involved in a resin cleaning operation for condensate demineralizer resins. Resins were apparently forced up a vent pipe into a ventilation exhaust duct, from which the resins were transported by ventilation air flow. Vent pipes are designed to maintain tank pressure close to atmospheric as tank levels fluctuate and gases evolve from tank contents. Such a design provides a controlled exhaust system rather than a discharge into the building atmosphere; many such vents are present in plant designs. While it is considered good design practice to install screens or filters in such vent lines, there were apparently no such devices in the Pilgrim vents. The Standard Review Plans 11.2 (Liquid Waste Management Systems) and 11.3 (Gaseous Waste Management Systems) and Regulatory Guide 1.143 (Radwaste System Design Guidance) do not specifically address such a design criterion.

(2) It is probable that water entered the ventilation exhaust ducts along with the resins noted in (1), above. While it is not known if this water was significantly radioactive, the presence of the water may have been a factor in the deterioration of filters and filter frames (see (3), below). Vent lines serving liquid systems should be designed to incorporate a device or mechanism, such as a water trap, to prevent the flow of liquids into vent pipes discharging to ventilation exhaust ducts. Neither the applicable Standard Review Plans nor the applicable Regulatory Guide address such a design feature.

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(3) The licensee considers the most probable source of the discharge of radioactively contaminated resins to the roof and ground areas of the plant to be the reactor building ventilation exhaust duct. Based on the dispersal pattern of the resins, we arrived at the same conclusion. As noted in (1) and (2), above, resins are presumed to have entered tank vent pipes leading to ventilation ducts, probably in the form of a slurry. The continuous flow of warm dry air would cause the resin to dry out, leaving a residue of small beads or particles of low density, which can be carried along the duct by the ventilation exhaust air current. In the filtration plenum, air from the ventilation exhaust ducts is passed first through a fiberglass prefilter media and then through a HEPA (High Efficiency Particulate Air) filter. Air flow through the filters is horizontal and there is about a four-foot space (measured horizontally) between the prefilter banks and HEPA filter banks. Linear face flow velocity (design) of the prefilters is about 250 linear feet per minute, or about 3 mph. Each HEPA filter module has a dimensional cross-section of about 4 ft^2 and has a rated capacity, when new, of 1,000 cfm at a 1" (water) pressure drop; the face velocity for a HEPA filter is also about 250 linear feet per minute or about 3 mph.

An IE Health Physics appraisal team visited Pilgrim in January and February, 1980. The team's report, dated July, 1980, noted that the prefilters were "disintegrating in place" (Section 4.2.3.2, page 55) but that no damage to the HEPA filters could be observed by visual inspection. This situation was apparently not corrected until the refueling outage which began in September, 1981. In fairness to the licensee, though, it should be noted that the prefilter disintegration was not included as a "significant finding" by the NRC in the appraisal. While there may be extenuating circumstances which are not apparent from the IE appraisal, there appear to be no reasons why these non ESF systems could not have been taken out of service for replacement or repair in a more expeditious manner.

While we have not been able to determine the exact condition of the HEPA filters at the time of their replacement in September, 1981, licensee representatives did state many of the HEPA filters were found to be damaged. It should be pointed out that no release of resins had been identified at that time and no tests were performed to determine the nature or extent of leakage or damage. The staff considers that the Pilgrim occurrence has no direct implications as to the integrity of adequately tested and maintained HEPA filters in ESF filter systems but, rather, emphasizes the need for regular testing and surveillance where a specified level of performance is to be achieved and maintained. The occurence is, however, a clear demonstration that plant operators cannot neglect HEPA filter systems indefinitely and then expect them to perform as designed.

We note, however, that in the present regulatory climate, licensees, in general, have no compelling motivation to perform surveillance which is not formally required of them, especially when inoperability of a system will not lead to noncompliance. The fact that deteriorating prefilters were observed during the Pilgrim Health Physics appraisal and that radioactive resins were found to be present in the ventilation exhaust ducts was not evidence that Technical Specification release limits or Appendix I criteria were being exceeded and, therefore, there was no violation of regulatory requirements to initiate corrective action. The periodic testing, or replacement of non-ESF filtration system components represents an expenditure of money and manpower with little tangible benefit when only routine normal operation is considered; in an era of tight money and budgetary restraints, plant managers may be hard-pressed to justify to upper levels of utility management the expenditure of even a few thousands of dollars at a very high cost-benefit ratio.

(4) Technical Specifications require periodic testing of ESF filter systems at nearly all plants, as well as surveillance of parameters such as pressure drop, which are indicative of system condition and performance. Normal ventilation exhaust air filter systems are not ESF systems and, therefore, are not subject to Technical Specification requirements for testing and surveillance. Non-ESF ventilation exhaust filter systems are installed in nuclear power plant buildings to reduce releases of airborne material to levels that satisfy the criteria of Appendix I to 10 CFR Part 50; Pilgrim, Unit 1, is only one of many plants which do not regularly inspect, check, or test their non-ESF filter systems.

While the failure or procrastination on the part of operating plants to regularly test and assure the proper functioning of these systems may be interpreted by some parties as failing to provide maximum protection to the environment, making such testing a firm commitment would necessitate a substantial revision in the basic NRC philosophy of plant safety and environmental protection. Commitments made by applicants in their FSAR to Regulatory Guide 1.140, "Design, Testing, and Maintenance Criteria for Normal Ventilation Exhaust System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants," is the method currently used by NRR to implement design guidance and testing programs for non-ESF filter systems. Such criteria had not been established by the NRC when Pilgrim 1 was licensed in 1972, so it is likely that no commitment was ever made by Boston Edison to provide surveillance testing of the non-ESF filters at Pilgrim 1.

(5) The Licensee and IE (reference IE Health Physics Appraisal Report for Pilgrim, dated June 22, 1980, page 54) have been aware for over two years that radioactive resin beads and fines were present in Pilgrim ventilation exhaust ducts. The same appraisal report, page 55 notes serious deficiencies in the condition of ventilation exhaust prefilters and the presence of approximately six inches of spilled radioactive (2R/hr) resins on the floor of a room in the Radwaste Building (p. 48), as well as loose contamination up to 90 mrads/hr on the floor immediately outside that room. In view of the unique and highly visible nature of resin beads, the rather high radioactive contamination levels associated with the resin, and the knowledge that resins had been a problem in several areas of the plant for over two years, the Licensee's statement (PN Update June 14, 1982) that the resins had probably been released prior to September 1981 seems to indicate, at best, an absence of recognition of potential problems on the part of plant management. To admit that external plant contamination of this order of magnitude had gone unnoticed and undetected for over eight months would seem to admit to the existence of inadequacies in the Health Physics program.

IE COORDINATION

Our review has been coordinated with IE personnel at Bethesda, Region I, and the Resident Inspectors' office. The Radiological Safety Branch (IE) is currently reviewing completed Health Physics appraisal reports for other plants to identify any similar circumstances to confirm the generic nature of the Pilgrim incident and support the need for issuance of

guidance to licensees; this review has not been completed but will be made available at a later date.

SUMMARY

As the result of our review of the Pilgrim, Unit 1, PNO of June 11, 1982 (PNO-I-82-42), the staff suggests the following:

(1) As a short-term action, recommend to IE that an information notice be issued to all operating reactors which (a) describes the Pilgrim 1 resin dispersal event, (b) requests plants to voluntarily institute a surveillance program for existing non-ESF filtration systems if one does not exist and (c) requests that tank vent designs be reviewed and that, if appropriate and feasible, modifications be made to prevent inadvertent release of resins or liquids to the ventilation system. NRR staff is available to provide assistance to IE in the preparation of such a circular.

(2) As a longer term action, revise Regulatory Guide 1.143, "Design Guidance for Radioactive Waste Management Systems, Structures, and Components installed in Light-Water-Cooled Nuclear Power Plants," and Standard Review Plan 11.2, "Liquid Waste Management Systems," to include design guidance and acceptance criteria which address (a) the incorporation of filters or screens in the design of vents from tanks which may contain resins, and (b) the incorporation of provisions into the vent design such as filters traps or check valves to prevent or minimize the flow of liquids through vent lines while permitting pressure equalization within the tank.

Roger J. Mattson

R. J. Mattson, Director
Division of Systems Integration
Office of Nuclear Reactor Regulation

JUL 8 1982

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Proposed
Change 85-01
BECo 85-022

February 1, 1985

Mr. Domenic B. Vassallo, Chief
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Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D. C. 20555

License DPR-35
Docket 50-293

Proposed Technical Specification Change
Tables 3.1.1 and 3.2.A

Dear Sir:

Pursuant to 10 CFR 50.90, Boston Edison Company hereby proposes the attached modification to Appendix A of Operating License No. DPR-35. This modification adds a note to Tables 3.1.1 and 3.2.A concerning the Main Steam Line High Radiation instrument setpoint. It is proposed to address the expected increase in main steam line radiation associated with injecting hydrogen as a mitigator of intergranular stress corrosion cracking of stainless steel piping.

Should you wish further information concerning this proposal, please contact us.

Very truly yours,

WD Harrington

Attachment

Three signed originals and 37 copies

cc: See next page

Commonwealth of Massachusetts)
County of Suffolk)

Then personally appeared before me W. D. Harrington, who, being duly sworn, did state that he is Senior Vice President - Nuclear of the Boston Edison Company, the applicant herein, and that he is duly authorized to execute and file the submittal contained herein in the name and on behalf of the Boston Edison Company and that the statements in said submittal are true to the best of his knowledge and belief.

My Commission expires: October 21, 1988

Peter M Kahler
Notary Public

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cc: Mr. Robert Hallisey, Director
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Proposed Change

Reference is made to Pilgrim Operating License No. DPR-35, Appendix A, pages 27, 29, 45 and 46. Page 27 contains Table 3.1.1, "Reactor Protection System (SCRAM) Instrumentation Requirement," and page 29 provides notes associated with that table. Page 45 contains Table 3.2.A, "Instrumentation That Initiates Primary Containment Isolation", and Page 29 provides notes associated with that table.

Currently, Table 3.1.1 states that the main steam line high radiation trip level setting is "$\leq$ 7x Normal Full Power Background." This applies to Refuel, Startup, Hot Standby and Run modes.

The desired amendment would add a note reference, (18), immediately following the trip level setting and next to the "x" in the "Run" column of the table. Note (18) is a new note added to Table 3.1.1 notes found on page 29, and shall state:

> Within 24 hours prior to the planned start of hydrogen injection with the reactor power at greater than 20% rated power, the normal full power radiation background level and associated trip setpoints may be changed based on a calculated value of the radiation level expected during the injection of hydrogen. The background radiation level and associated trip setpoints may be adjusted based on either calculations or measurements of actual radiation levels resulting from hydrogen injection. The background radiation level shall be determined and associated trip setpoints shall be set within 24 hours of re-establishing normal radiation levels after completion of hydrogen injection, or prior to withdrawing control rods at reactor power levels below 20% rated power.

Table 3.2.A includes the main steam line high radiation instrumentation. The proposed change adds the identical wording of the note described above to Page 46 as note (9), and places a (9) on Page 45 next to High Radiation Main Steam Line Tunnel.

Reason for Change

This amendment is proposed to permit changes in the normal full power background trip level setting for the Main Steam Line High Radiation scram and isolation setpoints to allow hydrogen injection as an IGSCC mitigating activity.

The purpose of hydrogen injection is to allow Boston Edison to evaluate the feasibility and efficacy of hydrogen water chemistry as a mitigator of intergranular stress corrosion cracking (IGSCC) of stainless steel BWR piping, and employ it should the evaluation demonstrate its usefulness as a mitigator. This technique consists of adding hydrogen to the primary coolant to lower the free oxygen concentration by suppressing radiolysis of water. By reducing free oxygen, one of the three necessary causative agents of IGSCC is eliminated. Testing at Pilgrim will be predicated upon experience gained from the hydrogen water chemistry programs developed and conducted by General Electric at their San Jose facilities, and at the Dresden-2 Unit of Commonwealth Edison Company. Permanent implementation will be based on plant specific data.

In sum, the purpose of employing hydrogen water chemistry is to effect the following benefits:

1) The elimination or reduction of IGSCC concerns.

2) The elimination of the costly replacement or repair of IGSCC, as well as the lost plant availability associated with such activities.

3) The reduction of radiation exposure to personnel engaged in pipe crack repairs and non-destructive examinations which stem from IGSCC.

The following data is to be provided by initial implementation:

1. The relationship of hydrogen level to oxygen level in the primary coolant system.

2. The identification of changes to plant chemistry, ion transport, conductivity, and reduction potential.

3. The determination of general in-plant and site boundary radiation increases due to increased N-16 activity.

4. The determination of specific locations where additional shielding may be required to support continued use of hydrogen injection.

5. The assessment of offgas system performance with hydrogen injection.

6. The adequacy of injection locations.

7. The adequacy of sampling equipment and procedural requirements.

8. The effectiveness of the hydrogen addition system to control free oxygen levels.

9. The evaluation of a permanent hydrogen injection installation.

When hydrogen is injected for oxygen suppression, nitrogen (N-16) carry-over increases in the main steam, which increases radiation in areas where main steam is found. The increased carry-over and radiation is caused by a conversion of N-16 from a soluble to a volatile form in the reactor.

The requested revision of Tables 3.1.1 and 3.2.A, and the addition of notes (18) and (9) permit an increase in the Main Steam Line High Radiation scram and isolation setpoints to allow operation with the expected higher radiation levels resulting from hydrogen injection. The main steam high radiation setpoint will remain at "$\leq$7 Normal Full Power Background"; however, because of increased N-16 in the steam, the background radiation level used to determine the high radiation setpoint will be increased prior to injection in accordance with a calculated background level value. The license amendment would permit the full load background radiation level to be adjusted during early implementation to correct for uncertainties in the initial calculated value. Pre-injection setpoints will be restored following the conclusion of injection, or when power is decreased to below 20% power. Hydrogen injection will not be performed with the reactor less than 20% power.

Boston Edison, with the aid and cooperation of General Electric, is planning to conduct hydrogen injection at Pilgrim during a 3 to 4 day period in March, 1985. Should this initial injection demonstrate the efficacy of further injection, Boston Edison will resume injection consistent with the data gleaned from the initial injection. We cannot commence prior to the approval of this proposed amendment by NRC.

Safety Considerations

At the maximum planned hydrogen injection rate, initially to be approximately 18 SCFM, experience indicates an expected increase of approximately 3 to 8 times the normal main steam line background radiation level. The only event which takes credit for the main steam line high radiation (MSLRM) trip is the design basis control rod drop accident (CRDA). As stated in Section 14.7.1.2 of the Pilgrim FSAR, a CRDA is only of concern below 10% of rated power. Since the Main Steam Line Radiation Monitor (MSLRM) setpoint will be adjusted at power levels above 20% power, the FSAR analysis and the design function of the MSLRM trip will remain valid. An increase in the MSLRM setpoint will not impact any other FSAR Chapter 14 accident or transient analysis since no credit is taken for MSLRM trips. Therefore, this proposed technical specification change will not reduce plant safety margins. In addition, the effect of hydrogen injection on the gaseous effluent release rate is expected to be insignificant because of the short decay time for N-16.

Boston Edison will maintain radiation protection/ALARA practices and procedures during injection. Initially, injection is being conducted, in part, to determine area radiation levels, which will, in turn, be used to determine shielding or procedural adjustments to minimize personnel exposure when hydrogen injection is employed.

This proposed change has been reviewed and approved by the Operations Review Committee (ORC), and reviewed by the Boston Edison Nuclear Safety Review and Audit Committee (NSRAC).

Significant Hazards Considerations

The Commission has provided guidance concerning the application of the standards for determining whether license amendments involve no significant hazards considerations by providing certain examples (48 FR 14870). Example (vi) of actions involving no significant hazards consideration is a change which may reduce in some way a safety margin, but where the results of the change are clearly within all acceptable criteria. The change proposed by this applications fits this example because it would permit the normal full power background level, associated with the Main Steam High Radiation scram and isolation setpoints, to be increased only so as to compensate for the anticipated increase in the main steam radiation levels during hydrogen injection. The capability to monitor for fuel failures, which is the mission of the MSLR trip setpoint, is maintained by: (1) the continued operability of the main steam radiation monitors, which provide signals to the reactor protection system and primary containment isolation system; (2) routine radiation surveys; (3) the performance of primary coolant water analyses; and (4) the continued operability of the Steam Jet-Air Ejector Off-Gas Radiation Monitor.

Although the potential for error exists whenever instrument setpoints are adjusted, the resulting increase in the probability or consequences of accidents previously evaluated is considered insignificant because of Boston Edison's existing quality assurance program and operating procedures as applied to instrument adjustments.

If, due to a recirculation pump trip or other unanticipated power reduction event, the reactor drops below 20% rated power without setpoint readjustment, control rod withdrawal is prohibited until the necessary setpoint readjustment is made. This ensures that fuel failures of the type concerning the MSLRM are unlikely.

Radiation protection practices will be performed during initial hydrogen injection based upon a pre-injection radiation (ALARA) review. During initial injection, special radiation level surveys will be performed and protective actions will be taken, as appropriate, to control all onsite personnel exposure. As data is gathered and assessed, steps will be taken to make permanent those changes to plant design and procedures deemed appropriate to minimize personnel exposure during the injection of hydrogen. Changes in gaseous effluent release rates for hydrogen injection are expected to be negligible due to the short decay times for N-16.

Based on the diverse means for maintaining the ability to detect fuel failures, on the protection of primary coolant system piping promised by implementing hydrogen water chemistry, on the efficacy of existing programs and procedures to assure accurate instrument setpoint adjustment, on both routine and exceptional ALARA actions to be taken prior to and during injection, on the ability of existing technical specifications to ensure that inimical control rod movement cannot occur below 20% power, and on the insignificant effect of increased N-16 activity on gaseous effluent release rates, Boston Edison concludes that the proposed amendment will not significantly increases the probability or consequences of accidents previously considered, will not create the possibility of a new or different accident from any previously evaluated, and will not significantly reduce a safety margin. Therefore, Boston Edison proposes to the NRC that it should make a determination that the proposed amendment does not involve significant hazards considerations.

Schedule of Change

This change will become effective upon Boston Edison's receipt of approval by the NRC. It is our intention to begin injection in March, 1985. We request that NRC act expeditiously on this change to allow the fulfillment of that schedule, which is determined, in part, by the availability of General Electric personnel.

Fee Determination

Pursuant to 10CFR 170.12 (c), an application fee of $150.00 is included with this proposed amendment.

TABLE 3.1.1

REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENTATION REQUIREMENT

Minimum Number Operable Inst. Channels per Trip (1) System	Trip Function	Trip Level Setting	Modes in Which Function Must Be Operable			Action(1)
			Refuel (7)	Startup/Hot Standby	Run	
1	Mode Switch in Shutdown		X	X	X	A
1	Manual Scram		X	X	X	A
	IRM					
3	High Flux	≤120/125 of full scale	X	X	(5)	A
3	Inoperative		X	X	(5)	A
	APRM					
2	High Flux	* (14) (15)	(17)	(17)	X	A or B
2	Inoperative		X	X(9)	X	A or B
2	Downscale	≥2.5 Indicated on Scale	(11)	(11)	X(12)	A or B
2	High Flux (15%)	≤15% of Design Power	X	X	(16)	A or B
2	High Reactor Pressure	≤1085 psig	X(10)	X	X	A
2	High Drywell Pressure	≤2.5 psig	X(8)	X(8)	X	A
2	Reactor Low Water Level	≥9 In. Indicated Level	X	X	X	A
2	High Water Level in Scram Discharge Tank	≤39 Gallons	X(2)	X	X	A
2	Turbine Condenser Low Vacuum	≥23 In. Hg Vacuum	X(3)	X(3)	X	A or C
2	Main Steam Line High Radiation	≤7X Normal Full Power Background (18)	X	X	X(18)	A or C
4	Main Steam Line Isolation Valve Closure	≤10% Valve Closure	X(3)(6)	X(3)(6)	X(6)	A or C
2	Turb. Cont. Valve Fast Closure	≥150 psig Control Oil Pressure at Acceleration Relay	X(4)	X(4)	X(4)	A or D
4	Turbine Stop Valve Closure	≤10% Valve Closure	X(4)	X(4)	X(4)	A or D

*APRM high flux scram setpoint ≤(.65W + 55) $\frac{FRP}{MFLPD}$ Two recirc. pump operation

10. Not required to be operable when the reactor pressure vessel head is not bolted to the vessel.

11. The APRM downscale trip function is only active when the reactor mode switch is in run.

12. The APRM downscale trip is automatically bypassed when the IRM instrumentation is operable and not high.

13. An APRM will be considered inoperable if there are less than 2 LPRM inputs per level or there is less than 50% of the normal complement of LPRM's to an APRM.

14. W is percent of drive flow required to produce a rated core flow of 69 Mlb/hr. Trip level setting in percent of design power (1998 MWt).

15. See Section 2.1.A.1.

16. The APRM (15%) high flux scram is bypassed when in the run mode.

17. The APRM flow biased high flux scram is bypassed when in the refuel or startup/hot standby modes.

18. Within 24 hours prior to the planned start of hydrogen injection with the reactor power at greater than 20% rated power, the normal full power radiation background level and associated trip setpoints may be changed based on a calculated value of the radiation level expected during the injection of hydrogen. The background radiation level and associated trip setpoints may be adjusted based on either calculations or measurements of actual radiation levels resulting from hydrogen injection. The background radiation level shall be determined and associated trip setpoints shall be set within 24 hours of re-establishing normal radiation levels after completion of hydrogen injection or prior to withdrawing control rods at reactor power levels below 20% rated power.

TABLE 3.2.A

INSTRUMENTATION THAT INITIATES PRIMARY CONTAINMENT ISOLATION

Minimum # of Operable Instrument Channels Per Trip System (1)	Instrument	Trip Level Setting	Action (2)
2(7)	Reactor Low Water Level	≥9" indicated level (3)	A and D
1	Reactor High Pressure	≤110 psig	D
2	Reactor Low-Low Water Level	at or above -49 in. indicated level (4)	A
2	Reactor High Water Level	≤48" indicated level (5)	B
2(7)	High Drywell Pressure	≤2.5 psig	A
2	High Radiation Main Steam Line Tunnel (9)	≤7 times normal rated full power background	B
2	Low Pressure Main Steam Line	≥880 psig (8)	B
2(6)	High Flow Main Steam Line	≤140% of rated steam flow	B
2	Main Steam Line Tunnel Exhaust Duct High Temperature	≤170°F	B
2	Turbine Basement Exhaust Duct High Temperature	≤150°F	B
1	Reactor Cleanup System High Flow	≤300% of rated flow	C
2	Reactor Cleanup System High Temperature	≤150°F	C

NOTES FOR TABLE 3.2.A

Whenever Primary Containment integrity is required by Section 3.7, there shall be two operable or tripped trip systems for each function.

2. Action

If the first column cannot be met for one of the trip systems, that trip system shall be tripped. If the first column cannot be met for both trip systems, the appropriate action listed below shall be taken.

A. Initiate an orderly shutdown and have the reactor in Cold Shutdown Condition in 24 hours.

B. Initiate an orderly load reduction and have Main Steam Lines isolated within eight hours.

C. Isolate Reactor Water Cleanup System.

D. Isolate Shutdown Cooling.

3. Instrument set point corresponds to 129.5" above top of active fuel.

4. Instrument set point corresponds to 78.5" above top of active fuel.

5. Not required in Run Mode (bypassed by Mode Switch).

6. Two required for each steam line.

7. These signals also start SBGTS and initiate secondary containment isolation.

8. Only required in Run Mode (interlocked with Mode Switch).

9. Within 24 hours prior to the planned start of hydrogen injection with the reactor power at greater than 20% rated power, the normal full power radiation background level and associated trip setpoints may be changed based on a calculated value of the radiation level expected during the injection of hydrogen. The background radiation level and associated trip setpoints may be adjusted based on either calculations or measurements of actual radiation levels resulting from hydrogen injection. The background radiation level shall be determined and associated trip setpoints shall be set within 24 hours of re-establishing normal radiation levels after completion of hydrogen injection or prior to withdrawing control rods at reactor power levels below 20% rated power.