10 CFR 50.73

BOSTON EDISON

Pilgrim Nuclear Power Station
Rocky Hill Road
Plymouth, Massachusetts 02360

March 29, 1993
BECo Ltr. 93-041

E. T. Boulette, PhD
Senior Vice President — Nuclear

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, D.C. 20555

Docket No. 50-293
License No. DPR-35

Dear Sir:

The enclosed supplemental Licensee Event Report (LER) 92-013-01, "Automatic Closing of Group 1 Containment Isolation Valves Due to False Reactor Vessel High Water Level Signal", is submitted in accordance with 10 CFR Part 50.73.

Please do not hesitate to contact me if there are any questions regarding this report.

E T Boulette
E. T. Boulette

RLC/bal

Enclosure: LER 92-013-01

cc: Mr. Thomas T. Martin
Regional Administrator, Region I
U.S. Nuclear Regulatory Commission
475 Allendale Rd.
King of Prussia, PA 19406

Mr. R. B. Eaton
Div. of Reactor Projects I/II
Office of NRR - USNRC
One White Flint North - Mail Stop 14D1
11555 Rockville Pike
Rockville, MD 20852

Sr. NRC Resident Inspector - Pilgrim Station

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NRC FORM 366 (5-92) U.S. NUCLEAR REGULATORY COMMISSION

APPROVED BY OMB NO. 3150-0104 EXPIRES 5/31/95

ESTIMATED BURDEN PER RESPONSE TO COMPLY WITH THIS INFORMATION COLLECTION REQUEST: 50.0 HRS. FORWARD COMMENTS REGARDING BURDEN ESTIMATE TO THE INFORMATION AND RECORDS MANAGEMENT BRANCH (MNBB 7714), U.S. NUCLEAR REGULATORY COMMISSION, WASHINGTON, DC 20555-0001, AND TO THE PAPERWORK REDUCTION PROJECT (3150-0104), OFFICE OF MANAGEMENT AND BUDGET, WASHINGTON, DC 20503.

LICENSEE EVENT REPORT (LER)

(See reverse for required number of digits/characters for each block)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
Pilgrim Nuclear Power Station	05000 293	1 OF 10

TITLE (4) Automatic Closing of Group I Containment Isolation Valves due to False Reactor Vessel High Water Level Signal

EVENT DATE (5)			LER NUMBER (6)			REPORT NUMBER (7)			OTHER FACILITIES INVOLVED (8)	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	MONTH	DAY	YEAR	FACILITY NAME	DOCKET NUMBER
									N/A	05000
10	24	92	92	013	01	03	29	93	N/A	05000

OPERATING MODE (9)	N
POWER LEVEL (10)	0

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more) (11)

	20.402(b)		20.405(c)	x	50.73(a)(2)(iv)		73.71(b)
	20.405(a)(1)(i)		50.36(c)(1)		50.73(a)(2)(v)		73.71(c)
	20.405(a)(1)(ii)		50.36(c)(2)		50.73(a)(2)(vii)		OTHER (Specify in Abstract below and in Text, NRC Form 366A)
	20.405(a)(1)(iii)		50.73(a)(2)(i)		50.73(a)(2)(viii)(A)		
	20.405(a)(1)(iv)		50.73(a)(2)(ii)		50.73(a)(2)(viii)(B)		
	20.405(a)(1)(v)		50.73(a)(2)(iii)		50.73(a)(2)(x)		

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER (Include Area Code)
Robert L. Cannon - Senior Compliance Engineer	(508) 747-8321

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS

SUPPLEMENTAL REPORT EXPECTED (14)		EXPECTED SUBMISSION DATE (15)	MONTH	DAY	YEAR
YES (If yes, complete EXPECTED SUBMISSION DATE)	X NO				

ABSTRACT (Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines) (16)

On October 24, 1992, at 2228 hours, an automatic actuation of the Main Steam System/Group 1 portion of the Primary Containment Isolation Control System occurred due to a false high Reactor Vessel (RV) water level signal. The actuation resulted in the automatic closing of the related Primary Containment System isolation valves that were in the open position. The isolations occurred when RV water level instrumentation fed from the 'B' Reference Leg Condensing Chamber 12B experienced level perturbations (spiking) due to dissolved noncondensible gases in the reference legs coming out of solution as the RV was being depressurized following a planned plant shutdown. Corrective action included correcting observable external leakage from instrument fittings on 'B' Reference Leg Instrument Racks, and testing for leakage past the instrument bypass valves on 'A' Reference Leg and 'B' Reference Leg Instrument Racks.

The event occurred during plant cooldown and depressurization with the Reactor Mode Selector Switch in the SHUTDOWN position. The reactor power level was zero percent. The RV pressure was approximately zero psig with the RV water temperature at approximately 160 degrees Fahrenheit. This report is submitted in accordance with 10CFR50.73(a)(2)(iv). This event posed no threat to the health and safety of the public. Previous similar events were reported in LERs 50-293/90-003-00, 91-008-01, and 92-004-00.

REQUIRED NUMBER OF DIGITS/CHARACTERS
FOR EACH BLOCK

BLOCK NUMBER	NUMBER OF DIGITS/CHARACTERS	TITLE
1	UP TO 46	FACILITY NAME
2	8 TOTAL 3 IN ADDITION TO 05000	DOCKET NUMBER
3	VARIES	PAGE NUMBER
4	UP TO 76	TITLE
5	6 TOTAL 2 PER BLOCK	EVENT DATE
6	7 TOTAL 2 FOR YEAR 3 FOR SEQUENTIAL NUMBER 2 FOR REVISION NUMBER	LER NUMBER
7	6 TOTAL 2 PER BLOCK	REPORT DATE
8	UP TO 18 – FACILITY NAME 8 TOTAL – DOCKET NUMBER 3 IN ADDITION TO 05000	OTHER FACILITIES INVOLVED
9	1	OPERATING MODE
10	3	POWER LEVEL
11	1 CHECK BOX THAT APPLIES	REQUIREMENTS OF 10 CFR
12	UP TO 50 FOR NAME 14 FOR TELEPHONE	LICENSEE CONTACT
13	CAUSE VARIES 2 FOR SYSTEM 4 FOR COMPONENT 4 FOR MANUFACTURER NPRDS VARIES	EACH COMPONENT FAILURE
14	1 CHECK BOX THAT APPLIES	SUPPLEMENTAL REPORT EXPECTED
15	6 TOTAL 2 PER BLOCK	EXPECTED SUBMISSION DATE

REASON FOR SUPPLEMENT

This supplement updates our corrective action as described in our initial report.

BACKGROUND

As discussed in the Similarity to Previous Events section, false high Reactor Vessel (RV) water level spiking has been noted previously during plant shutdowns. LER 50-293/90-003-00 documents an automatic Primary Containment Isolation Control System (PCIS) Group 1 main steam line isolation due to false high reactor water level spiking that occurred during a plant shutdown on March 11, 1990. The cause was believed to be trapped air in the 'B' side RV water level instrumentation reference leg. Corrective action taken was to backfill the sensing lines with demineralized water using an approved station procedure.

LER 50-293/91-008-01 documented three automatic PCIS Group 1 main steam line isolations during a plant shutdown on April 30, 1991. A Multi-Disciplined Analysis Team conducted an in-depth review of the false high RV water level spiking. Root cause determination identified the 'B' Reference Leg head equalizing line connecting the condensing chamber to the reactor vessel as being undersized. Plant modification PDC 91-40, completed during Refueling Outage (RFO) 8, increased the size of this line from 1 inch to 2 inch. Post modification testing was satisfactorily completed by pressurizing the reactor to approximately 380 psig and then depressurizing to 32 psig. This testing did not identify similar false high reactor water level spiking. The plant restarted from RFO 8 on August 11, 1991, and operated until October 30, 1991. RV water level perturbations were noted on the 'B' reactor water level instrumentation during the plant shutdown on October 30, 1991. However, these perturbations were of less magnitude and did not result in a Group 1 isolation. A walkdown of the 'A' and 'B' head equalizing lines identified a difference in the insulation installed. The 'A' head equalizing line had the required 2-1/2 inch insulation but the 'B' head equalizing line had only 1 inch insulation. The required insulation on the 'B' head equalizing line was installed.

LER 50-293/92-004-00 documented three automatic PCIS Group 1 isolations that occurred during plant shutdown and depressurization on March 26 and 27, 1992. The first and third Group 1 isolations were due to a false high reactor vessel water level signal. The second isolation was due to RV water level expansion (swell) when a Main Steam Isolation Valve was reopened following the first isolation. The false high RV water level signals were attributed to improper thermal performance of the 'B' Reference Leg condensate chamber and head equalizing line connecting the condensate chamber to the RV. Corrective actions included removal of insulation from the 'B' head equalizing line to improve thermal performance of the condensing chamber. These actions have not been successful in precluding the RV water level perturbations (spiking) and subsequent false RV high water level signals.

TEXT (If more space is required, use additional copies of NRC Form 366A) (17)

EVENT DESCRIPTION

On October 24, 1992, at 2228 hours, an automatic actuation of the Main Steam System/Group 1 portion of the Primary Containment Isolation Control System (PCIS) occurred due to Reactor Vessel (RV) water level perturbations resulting in a false high RV water level signal.

The actuation resulted in the following designed responses. The inboard and outboard Primary Containment System (PCS) Main Steam Isolation Valves (MSIVs) were in the closed position and remained closed. The inboard and outboard PCS/Sample System Valves (AO-220-44 and -45) in the open position, closed automatically. Inboard PCS/Main Steam drain valve MO-220-1 in the open position, closed automatically. Outboard drain valve MO-220-2 remained in the open position because it is controlled by circuitry not associated with the affected sensors.

The event occurred during the final stages of a planned shutdown while completing Attachment D ("Maneuvering to Cold Shutdown with MSIVs Open") of procedure 2.1.5 (Rev. 41) "Controlled Shutdown from Power". At the time of the event, a level spike of approximately 29 inches (limited by instrument range) was experienced by the level instruments on Instrument Rack C-2206 that are connected to the sensing line for RV water level Reference Leg 'B' and Condensing Chamber 12B.

Problem Report 92.9203 was written to document the event. The NRC Operations Center was notified in accordance with 10 CFR 50.72 on October 24, 1992, at 2304 hours.

This event occurred with the Reactor Mode Selector Switch in the SHUTDOWN position. The reactor power level was zero percent with the control rods in the fully inserted position. The RV pressure was approximately zero psig with the RV water temperature at approximately 160 degrees Fahrenheit.

CAUSE

The primary cause of this event is attributed to a false RV high water level signal resulting from level perturbations (spiking) in RV water level Reference Leg 'B' due to non-condensable gases coming out of solution as the reactor was being depressurized.

It is believed the following three contributing factors are required for spiking to occur:

- Buildup of non-condensable gas concentrations in a reference leg condensing chamber.

- Small leaks in the reference leg piping that result in the transport of dissolved gases in the direction toward the location of the leak.

- Horizontal or near horizontal sections of reference leg piping that provide locations where the accumulation of gases coming out of solution can displace Reference Leg water.

The water level spiking phenomenon has occurred on at least five separate occasions at Pilgrim Station. The phenomenon has been random in frequency but is generally repeatable in magnitude and duration at various low reactor pressures. This phenomenon has been attributed to volumes of dissolved gas coming out of solution and traveling at predictable velocity along a known geometry of reference leg piping until escaping into the condensing chamber. These traveling pockets of gas were confirmed during the level perturbations of the 'B' Reference Leg by the use of ultrasonic sensors installed on the vertical piping section of Reference Legs 'A' and 'B'. Spiking has not been observed above approximately 425 psig and has occurred only during depressurizations during shutdown operations. The instruments connected to the 'B' Reference Leg have experienced spiking beginning in the range of 350 to 425 psig (approximately 5 inch spikes), whereas the instruments connected to the 'A' Reference Leg begin to exhibit spiking in the range of 60 - 70 psig (approximately 2 inch spikes). 'A' Train instrument responses have generally been bounded by 'B' Train instrument responses in amplitude, duration, and reactor pressure when spiking began to occur. Typically, spike amplitudes have increased as reactor pressure decreased with the largest observed spikes, of approximately 29 inches (limited by instrument range), occurring at approximately zero psig. Above approximately 100 psig, the largest observed spikes have been approximately 6 inches in the 'B' Train instruments and no spiking has occurred in 'A' Train instruments. Durations are typically 20 to 30 seconds but frequency of spiking increases at very low pressures (e.g., <10 psi) and has affected indicated level for several minutes. Spikes have always indicated higher than actual water level.

CORRECTIVE ACTION

The minor external leakage (i.e., fittings, etc.) on the 'A' and 'B' Reference Legs was quantified prior to the MCO-9 shutdown as approximately 0.14 milliliters/day and 97 milliliters/day, respectively. To minimize the amount of disolved gases being drawn down into the reference legs, identified leakage was corrected (i.e., fittings tightened, valve packing adjusted, etc.) for instruments located on Instrument Racks that are part of the 'A' and 'B' Reference Legs.

Following plant restart from MCO-9, the external leakage was again quantified and found to have been significantly reduced. The external leakage for the 'B' Instrument Racks was reduced from 97 milliliters/day to 0.5 milliliters/day. These measurements are not exact due to the inherent difficulty in collecting the leakage.

TEXT (If more space is required, use additional copies of NRC Form 366A) (17)

To address possible internal leakage paths, instrument bypass valve leakage was checked in accordance with Temporary Procedure (TP) 92-47, "Reactor Vessel Instrument Reference Lines, Internal and External Leakage Investigation Racks 2205, 2206, 2251, 2252, 2275 and 2276". The results of the testing indicated little or no leakage could be identified. On the 'A' side, leakage could not be quantified due to leakage past the rack isolation valve.

Resistance temperature devices (RTDs) were previously installed via Temporary Modification 91-44 to monitor the thermal performance of both the 'A' and 'B' head equalizing lines and condensing chambers (Reference LER 50-293/92-004-00). The RTDs were checked prior to plant restart from MCO-9 to confirm their accuracy in preparation for continued monitoring of the condensing chambers. The results of the accuracy checks were satisfactory.

The insulation on 'B' head equalizing line was reinstalled. The 'B' head equalizing line thermal insulation was previously removed via Temporary Modification 92-13 (Reference LER 50-293/92-004-00) to improve the head equalizing line condensation. The modification was intended to allow the head equalizing line temperature to more closely follow the moderator temperature during reactor depressurization thus reducing vaporization in the head equalizing line. Removal of the insulation was also expected to increase the removal of non-condensable gases from the condensing chamber due to increased condensing chamber condensate return flow to the reactor vessel. A review of the relative volume of dissolved gas that evolved in the reference legs during the October 24, 1992, shutdown as compared to the March 26, 1992, shutdown concluded no appreciable improvement resulted from the removal of the insulation.

On August 19, 1992, the NRC issued Generic Letter (GL) 92-04 regarding resolution of issues related to reactor vessel water level instrumentation in Boiling Water Reactors (BWRs). Prior to issuance of GL 92-04, Boston Edison Company (BECo) was investigating RV water level inconsistencies and spiking issues. BECo's response to GL 92-04 included the following additional actions:

- BECo commissioned a consultant (S. Levy, Incorporated) to analyze Pilgrim's design and operational data to determine the root cause, assess the significance, and recommend possible corrective actions.

- Operations personnel continue to monitor for indicated water level mismatches between water level instruments by comparing readings from different instruments on a routine basis.

- Appropriate operations crew response to various water level scenarios that could result from this phenomenon was demonstrated on the Pilgrim simulator and was observed by NRC personnel.

- BECo is participating in the BWR Owner's Group (BWROG) effort on water level and endorses the plan provided in the BWROG letter to the NRC dated August 12, 1992. We also support the BWROG plan provided in its September 24, 1992 letter to the NRC. Pilgrim is scheduled to start its next refueling outage, RFO #9, in April 1993.

- If our assessment of the BWROG program indicates modifications are necessary to ensure level instrumentation is of high functional reliability, such modifications will be scheduled for future implementation. We will keep the NRC advised of our progress on this topic in our semiannual submittals of the Long Term Plan.

ADDITIONAL INFORMATION

During plant depressurization following a reactor scram (ref. LER 93-004-00) on March 13, 1993, the reactor water level instrumentation was monitored for possible spiking. The plant had operated for 80 days prior to the scram. The monitoring indicated no signs of spiking on the 'A' side level instrumentation (e.g., two independent Condensing Chambers, 12A and 13A and their associated eight level transmitters).

Data indicates spiking for 'B' side level instrumentation downstream of Condensing Chamber 12B did not begin until reactor pressure decreased to approximately 180 psi. The spikes observed during depressurization were typically 2 inches in amplitude and in one instance, reached a maximum of 4 inches for instruments downstream of Condensing Chamber 12B. The specific shape and duration for each of these spikes indicates non-condensable gases did not migrate below the 69' Reactor Building elevation. Data indicates a significant reduction in amplitude, frequency and total number of spikes during the shut down as compared with data from other shutdowns with comparable operating time. The observed instrument response was as predicted following the maintenance in MCO-9 to minimize leakage from fittings at the instrument racks. It demonstrates that leakage reduction significantly reduces the amount of non-condensable gases in the related reference leg.

SAFETY CONSEQUENCES

This event posed no threat to the public health and safety.

The level spiking condition does not increase the probability of occurrence of an accident previously evaluated in the safety analysis report because the spiking only has the potential to affect mitigating systems and not accident initiators. The level spiking condition does not create the possibility of an accident of a different type than any previously evaluated in the safety analysis report. Although it is postulated that hydrogen gas may accumulate in the condensing pots, there is no ignition source in the water/steam environment that could lead to an explosion or burn.

The level spiking condition does not increase the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report. All safety functions are performed such that consequences of transient and accident analyses are unaffected. The level spiking condition does not increase the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the safety analysis report. The only active safety-related components that can potentially be affected by level spiking are the level and pressure transmitters connected to the instrument legs. Spiking of 29 inches at low reactor pressures is well within the design limits of these transmitters and would have no adverse effects. Although spiking may cause level inaccuracy and safety system initiation delays, the equipment can perform its specified safety functions. Similarly, the level spiking condition does not create the possibility of a different type of malfunction of equipment important to safety than any previously evaluated in the safety analysis report. The Final Safety Analysis Report (FSAR) acknowledges some level errors occur during design basis events. In selecting reactor water level as an input parameter to initiating safety functions, it is understood that some inaccuracy is inherent in the design. Deviations between actual and sensed level are expected due to pressure and thermal density effects, reference leg flashing, coolant voiding, and flow across instrument nozzles. In selecting level trip settings, FSAR 7.2.4 states that the "trip point selected is not the only value of the trip point results in acceptable results relative to the fuel or nuclear system process barrier. Trip setting selection is based on operating experience and is constrained by the safety design basis". Trip functions at levels different than assumed in accident analyses due to level spiking do not lead to unacceptable results.

The level spiking condition does not reduce the margin of safety as defined in the basis for any Technical Specification. Above 785 psig reactor pressure, a MCPR safety limit is prescribed in Technical Specifications to provide margin to the point of departure from nucleate boiling for the fuel. Although it is recognized that a departure from nucleate boiling would not result in fuel damage, a Minimum Core Power Ratio (MCPR) of greater than 1.04 is selected to account for operating state and calculational uncertainties and to provide a margin to the conditions which would produce onset of transition boiling (i.e., MCPR of 1.0). Above 785 psig, water level spiking is not expected and no challenge to the MCPR margins of safety exists. Below 785 psig, the fuel integrity safety limit is 25% of rated thermal power. Any event that initially occurs with reactor pressure below 785 psig would occur with reactor power well below 25% (i.e., less than 5%) and there is no challenge to margins of safety.

Safety margins are generally defined for analysis under limiting conditions (e.g., 100% power). Water level spiking has no effect on analyses of events at 100% power and safety margins are, therefore, not affected. As discussed above for accident and transient analyses, the effects of water level spiking are bounded by existing analyses and do not result in exceeding defined acceptance criteria (e.g., offsite dose, peak clad temperature, Technical Specification safety limits). As specific examples, core uncovery does not occur before closure of containment isolation valves and delays in response of the 2/3 core coverage interlock do not lead to core uncovery exceeding that analyzed in limiting Loss of Coolant Accident analyses.

Limiting safety system settings are settings on instrumentation that initiate the automatic protective action at a level such that the safety limits will not be exceeded. The region between the safety limit and these settings represents margin with normal operation lying below these settings. The margin has been established so that, with proper operation of the instrumentation, the safety limits will never be exceeded. These settings are limiting at full power conditions. As has been demonstrated, level spiking at low power, low pressure conditions represents no challenge to safety limits. Analytical conservatisms that may be reduced by level spiking trip delays are compensated by the fact that level spiking only affects safety system responses at low powers and pressures.

This report is submitted in accordance with 10CFR50.73(a)(2)(iv) because the Group 1 portion of the PCIS logic circuitry actuated.

SIMILARITY TO PREVIOUS EVENTS

A review was conducted of Pilgrim Station Licensee Event Reports (LERs) submitted since January 1984. The review focused on LERs submitted in accordance with 10 CFR 50.73(a)(2)(i)(B) and (a)(2)(iv) that involved PCIS Group 1 actuations due to false high RV water level signals or due to the opening of the MSIVs with the reactor pressurized. Five similar events were identified in LERs 50-293/84-019-00, 50-293/89-007-00, 50-293/90-003-00, 50-293/91-08-01 and 50-293/92-004-00.

LER 84-019-00 documents a Group 1 isolation signal that occurred when reactor water level indication from the 'A' level instrumentation trended up to +45 inches. The cause was excess cooling in the area of the 'A' Reference Leg. Corrective action, related to the response of Generic Letter 84-23, included installing new reference legs outside the Drywell, minimizing the vertical piping drop inside the drywell and replacing the former reactor water level instrumentation with transmitters and electronic switching devices. The installations were completed during Refueling Outage 7.

LER 89-007-00 documents a Group 1 isolation signal due to high reactor water level during MSIV testing. The high level occurred after opening an MSIV causing an expansion (swell) of the reactor vessel water. The root cause was an inadequacy in the development and review of the approved test procedure and a relatively fast opening time for the MSIV. The procedure did not identify the affect on reactor water level when opening a MSIV with a 150 psig differential pressure. The procedure concern was dispositioned and the MSIV maintenance procedure was revised relative to the MSIV opening time range.

LER 90-003-00 documents a Group 1 isolation signal that occurred when reactor water indication from the 'B' level instrumentation rapidly increased from +25 inches to +50 inches for approximately thirty seconds. The cause was believed to be trapped air in instrumentation sensing lines. A station procedure was developed that provided the necessary instructions for backfilling the sensing lines. Additionally, certain surveillance procedures were revised to minimize the possibility of introducing air into the system during the surveillance.

LER 91-008-01 documents three Group 1 isolations that occurred on April 30, 1991, during a plant shutdown due to false high reactor water level indications. The cause was believed to be an undersized 'B' Reference Leg condensing chamber head equalizing line connecting the condensate chamber to the reactor vessel. Corrective actions taken included increasing the size of the head equalizing line from 1 inch to 2 inches. This modification was completed during Refueling Outage 8.

LER 92-004-00 documents three Group 1 isolations that occurred on March 26 and 27, 1992, during a plant shutdown due to false high reactor water level signals. The false signals were attributed primarily to improper thermal performance of the 'B' Reference Leg condensate chamber and head equalizing line connecting the condensate chamber to the RV. Corrective action included removing the insulation from the "B" head equalizing line.

LICENSEE EVENT REPORT (LER)
TEXT CONTINUATION

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	PAGE (3)
Pilgrim Nuclear Power Station	05000 293	92	- 013 -	01	10 OF 10

TEXT (If more space is required, use additional copies of NRC Form 366A) (17)

ENERGY INDUSTRY IDENTIFICATION SYSTEM (EIIS) CODES

The EIIS codes for this report are as follows:

COMPONENTS	CODES
Valve, Isolation	ISV
Recorder, Level	LR
Transmitter, Level	LT
Switch, Level	LIS
SYSTEMS	
Containment Isolation Control System (PCIS)	JM
Engineered Safety Feature Actuation System (PCIS)	JE
Incore/Excore Monitoring System (RV Water Level)	IG