NRC Form 366
(9-83)

U.S. NUCLEAR REGULATORY COMMISSION
APPROVED OMB NO. 3150-0104
EXPIRES: 8/31/85

LICENSEE EVENT REPORT (LER)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
Catawba Nuclear Station, Unit 1	0 5 0 0 0 4 1 3	1 OF 1 2

TITLE (4) Wide Range Temperature Monitoring Instrumentation Technically Inoperable During Certain Accident Conditions Due To Installation Adn Design Deficiencies

EVENT DATE (5)			LER NUMBER (6)			REPORT DATE (7)			OTHER FACILITIES INVOLVED (8)	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	MONTH	DAY	YEAR	FACILITY NAMES	DOCKET NUMBER(S)
0 1	1 6	8 8	8 8	— 0 0 3 —	0 0	0 2	1 5	8 8	Catawba, Unit 2	0 5 0 0 0 4 1 4
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OPERATING MODE (9): 1

POWER LEVEL (10): 1 0 0

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

	20.402(b)		20.405(c)		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)	X	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	
	AREA CODE	
Julio G. Torre, Associate Engineer - Licensing	7 0 4	3 7 3 - 8 0 2 9

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

CAUSE	SYSTEM	COMPONENT	MANUFAC-TURER	REPORTABLE TO NPRDS		CAUSE	SYSTEM	COMPONENT	MANUFAC-TURER	REPORTABLE TO NPRDS

SUPPLEMENTAL REPORT EXPECTED (14)

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

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

On January 15, 1988 at 1030 hours, with Unit 2 in Mode 6, refueling, 6 of the 8 Unit 2 Reactor Coolant (NC) System Wide Range (W/R) hot and cold leg Resistance Temperature Detector (RTD) cables were discovered to have been improperly installed. The RTD cables were originally supplied by Westinghouse with a sealed stainless steel bellows hose to maintain environmental qualification. However, the affected Unit 2 RTD cables and bellows hoses had been cut during initial installation. As a result, the RTD's environmental qualification was invalidated. Since the corresponding Unit 1 RTD's were suspected to be similarly installed, they were declared inoperable at 1400 hours, and Unit 1 shutdown was commenced at 2250 hours. Unit 1 entered Mode 3, Hot Standby, at 0541 hours, and Mode 4, Hot Shutdown, at 2230 hours, on January 16, 1988. Subsequently, the corresponding Unit 1 W/R hot and cold leg RTD cables were also found to be improperly installed. Unit 1 was in Mode 1, Power Operation, at 100 percent power at the time of discovery. Both Units have operated in all modes with the affected RTD's technically inoperable.

This incident is attributed to an installation deficiency prior to startup of each Unit due to a misinterpretation of notes on the connection diagrams. A design deficiency also contributed to this event because environmentally qualified (sealed) junction boxes should had been specified on the connection diagram. The affected Unit 1 RTD's/cables have been replaced. The affected Unit 2 RTD's/cables will be replaced prior to Unit heatup following the current refueling outage. The connection diagram has been revised as appropriate. The health and safety of the public were unaffected by this event.

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BACKGROUND:

The Reactor (EIIS:RCT) Coolant (EIIS:AB) (NC) System Wide Range (W/R) Resistance Temperature Detectors (EIIS:DET) (RTD's) provide continuous hot and cold leg temperature indication to Control Room Operators (CROs) under normal and accident conditions. These measured temperatures are available to CROs via meters, chart recorders and the Inadequate Core Cooling System (ICCS) monitors. The W/R temperature indications are utilized to monitor the NC System during Unit heatup and cooldown operations when the Narrow Range (N/R) RTD's are off scale low. They do not provide any automatic control functions. There is one W/R RTD installed in a well in each of the four NC loop's hot and cold legs.

In addition to the W/R loop temperature indication (0 - 700 degrees F), each loop contains N/R hot and cold leg RTD's (530 - 630 degrees F) installed in bypass piping. The N/R RTD's are immersion type which provide rapid response to NC System temperature changes. Since they are not installed directly in each NC loop, the N/R RTD's rely on NC pump (EIIS:P) operation to provide flow through the bypass piping for proper loop temperature indication. The N/R RTD's provide CROs with the most accurate NC loop temperature indication when at normal operating temperature and are used to provide various automatic control functions.

The In-Core Instrumentation (ENA) System provides 65 thermocouples (EIIS:THC) (100 - 700 degrees F normal range) installed on the Reactor upper internals to measure core outlet temperature. This data is available to CROs on the ICCS monitors.

The ICCS monitors provide CROs with a graphic display of actual measured NC temperature and pressure super-imposed upon a background outlining safe temperature and pressure limits. There are two trains of ICCS. Train A monitors loops C and D while train B monitors loops A and B. Train B is the designated Post Accident Monitor (PAM). Also displayed on the ICCS monitors are: 1)degrees of subcooling based upon the 5 highest reading ENA thermocouples, 2)degrees of subcooling based upon each loop's W/R RTD reading, and 3)W/R NC pressure.

There are two loops of W/R hot and cold leg RTD's designated as PAM instrumentation (loops A and B). Technical Specification 3.3.3.6 requires that if both channels are inoperable, the inoperable channel(s) must be restored to operable status within 48 hours or be in at least Hot Standby within the next 6 hours and in Hot Shutdown within the following 6 hours. The operability of the PAM instrumentation ensures that sufficient information is available on selected plant parameters to monitor and assess these variables following an accident.

DESCRIPTION OF INCIDENT:

On November 11, 1978, the original NC System W/R hot and cold leg RTD installation drawings approved by Westinghouse Electric Corporation were supplied to Duke Power Company's Design Engineering (D/E) Department. The RTD drawings showed the cables protected by a spiral or braided steel jacket and specified in a note that the cables should enter the junction box from the side to avoid contamination by condensation or drippage. The junction boxes are a Duke Power supplied item.

During environmental qualification testing performed by Westinghouse, the RTD's/cables were immersed in water. This resulted in unacceptable low readings for some of the RTD's due to moisture migration through the cable insulation to the RTD lead wires. To correct this problem, the cables were covered with stainless steel bellows hose with a stainless steel overbraid and a sealed junction box was specified. A sealed fitting on the junction box end of the bellows hose was added, which when connected to a sealed junction box, would totally protect the cable insulation from water/moisture in the installed environment.

On March 6, 1980, the original Unit 1 W/R hot and cold leg RTD connection diagram was completed and approved by D/E. The diagram was apparently based upon the information provided on the original Westinghouse RTD installation drawings. Note 6 on the connection diagram stated that "if the leads supplied by the vendor are covered with braided armor, field is to add flexible conduit for adequate protection" after installation. Additionally, note 12 stated that the "field is to cut the leads if too long for proper installation, conductors must be the same length". The connection diagram also showed a standard splash proof junction box was to be utilized to splice the RTD lead wires to the field wiring.

On October 8, 1981, Westinghouse revised the affected RTD installation drawing to show the added bellows hose with stainless steel overbraid and the required vapor tight junction box. On November 13, 1981, D/E received the revised Westinghouse drawing which was then routed to appropriate D/E personnel for review. The Design Engineer responsible for the RTD's environmental qualifications reviewed the drawings without taking actions to change the connection diagram for Unit 1.

On September 16, 1982, the original Unit 2 W/R hot and cold leg RTD connection diagram was completed and approved by D/E. The Unit 2 W/R RTD connection diagram contained the same notes as the Unit 1 connection diagram and also specified that a standard splash proof junction box was to be utilized to splice the RTD lead wires to the field wiring.

On May 17, 1983, Construction Electricians spliced the NC loop 1A and 1B W/R cold leg RTD lead wires to the field wiring utilizing Raychem splices inside splash proof junction boxes. On May 20, 1983, a Duke Power Quality Assurance Inspector originated a Discrepancy Report because the RTD cables were covered with braided stainless steel and had not been covered with flexible conduit for protection as she thought was required by note 6 on the connection diagram. The Discrepancy Report was given to a Construction Electrical Technical Support (ETS) Technician for resolution.

The ETS Technician remembers contacting someone in D/E to assist in resolution of the problem. However, he cannot recall to whom he spoke. He believes that he received verbal authorization to cut the leads to allow for installation of the flexible conduit from someone in D/E. The ETS Technician's personal notes and files were discarded by his successor after his transfer to another work location. Therefore, no records of his conversations which might help justify his decisions are available.

On May 24, 1983, the ETS Technician documented the resolution for the Discrepancy Report, which was for the Construction Electricians to install flexible conduit over the vendor leads and to cut the leads if necessary per notes 6 and 12 respectively on the connection diagram

In order to install the flexible conduit over the RTD cable bellows hoses and stainless steel overbraid, Construction Electricians had to cut the fittings off the ends of the bellows hoses which connected to the junction boxes. The 20 foot cables were then cut to the lengths necessary to span the distance from the RTD's on each loop to their respective splash proof junction boxes. All of the Unit 1 and Unit 2 NC System W/R hot and cold leg RTD's were subsequently installed in this manner (invalidating their environmental qualifications). The Unit 1 and Unit 2 loop A and B W/R cold leg RTD's were later replaced with dual element RTD's with environmentally qualified cables under a Nuclear Station Modification (NSM) to provide indication to the Standby Shutdown Facility (SSF).

On November 22, 1984, Unit 1 entered Mode 3, following initial fuel loading. On April 23, 1986, Unit 2 entered Mode 3, following initial fuel loading.

On November 5, 1987, an Electrical Design Engineer was performing an inspection of Raychem splices on safety related instrumentation in Unit 1 Containment. He discovered that the W/R hot and cold leg RTD junction boxes were not sealed as he thought was required. He did not notice that the RTD cables had been improperly installed. D/E subsequently verified that the junction boxes were required to be sealed and originated a Problem Investigation Report (PIR) on November 13, 1987. D/E notified Catawba's Compliance section of the Unit 1 RTD junction box problem and the possible operability concerns. The Unit 2 W/R RTD's were declared inoperable per Technical Specification 3.3.3.6 at 1530 hours. Unit 1 was in a mode in which the Technical Specification did not apply. At approximately 1800 hours, D/E issued a Statement of Operability which justified continued operation through the use of compensatory measures which could be administratively implemented in the event of a high energy line break (HELB) inside Unit 2 Containment. This was based upon a maximum 60 degree F negative error being induced by moisture migration through the small amount of cable insulation (inside the unsealed junction boxes) exposed to Containment atmosphere following a HELB. The oncoming Control Room Operators were trained to implement the compensatory measures in the event of a HELB inside Unit 2 Containment.

On November 14, 1987, at 0800 hours, the Unit 2 W/R RTD's were declared operable and power operation of the Unit was continued. On November 23, 1987, the affected Unit 1 W/R RTD junction boxes were sealed by filling them with epoxy. The environmental qualifications of the Unit 1 W/R RTD's were thought to be restored at this time. However, the RTD cables were still susceptible to moisture migration because the cables (outside the epoxied junction boxes) were still not sealed from the Containment atmosphere.

On November 30, 1987, Station management initiated a Duke Power Incident Investigation Report (IIR) to report the improper W/R RTD junction box installation (see LER 413/87-43).

On January 15, 1988, at 1030 hours, a Projects Engineer discovered the Unit 2 W/R RTD cables and their bellows hoses had been improperly installed. The Engineer was inspecting the RTD junction boxes which were going to be replaced during the refueling outage with environmentally qualified models. Due to his familiarity with the Westinghouse RTD installation drawings, he was expecting to find excess cable coiled up between the RTD's and their junction boxes. However, this was not the case and it prompted him to closely examine the cabling outside the junction boxes. The examination revealed that the sealed fittings connecting the bellows hoses to the junction boxes had been removed and replaced with standard attachment hardware and that the cable bellows hoses had been cut and flexible conduit installed over the cables. After exiting Containment, the Engineer originated a PIR, notified the Compliance section, and wrote an inoperability statement.

Since the Unit 1 RTD's were suspected to be similarly installed, they were declared inoperable at 1400 hours. Unit 1 was in Mode 1, Power Operation, at 100 percent power and Unit 2 was in Mode 6, Refueling, at the time of this discovery.

D/E was unable to determine the degree of inaccuracy which could be expected with the entire RTD cable (outside the junction box) exposed to Containment atmosphere following a HELB. Therefore, Unit 1 shutdown was commenced at 2250 hours. The Unit entered Mode 3 at approximately 0541 hours, on January 16, 1988. Upon entry into Containment following Unit 1 shutdown, the RTD's were found to have been similarly installed. Unit 1 cooldown proceeded and Mode 4, Hot Shutdown, was entered at 2230 hours.

The Unit 1 RTD's were replaced and their junction boxes refilled with epoxy. Unit 1 returned to Mode 1 on January 23, 1988, at 0755 hours.

CONCLUSION:

This incident is attributed to an installation deficiency. The affected Unit 1 and Unit 2 W/R RTD's/cables were incorrectly installed during plant construction prior to initial startup of each Unit. The improper installation is attributed to the misinterpretation of the notes on the connection diagrams which directed the installers to cut the leads if too long for proper installation and to cover the leads with flexible conduit if supplied by the vendor with braided armor. According to D/E, the notes were intended to apply only to the individual leads inside each cable, not to allow cutting of the 20-foot RTD cable/bellows hose or covering the cable/bellows hose with flexible conduit. The RTD's/cables were installed using the notes on the connection diagrams without formal D/E approval through the normal Variation Notice process. Verbal D/E approval may have been granted to allow Construction to cut the individual RTD leads. Communications problems could have been present and the terms cable and leads interchanged/confused resulting in the RTD cables being cut. If D/E approval through the Variation Notice process had been requested at the time, this incident may have been avoided.

A design deficiency also contributed to this event. As identified in LER 413/87-43, the W/R hot and cold leg RTD junction boxes were not environmentally qualified due to a Design Engineer's oversight when reviewing the Westinghouse RTD installation drawing revisions. The responsible Engineer (deceased) apparently considered the use of Raychem splices and splash proof junction boxes to be an acceptable substitute for the vapor tight junction box which was specified on the Westinghouse drawings. As a result, he did not take action to change the connection diagrams for Catawba's W/R RTD's, and standard splash proof junction boxes were installed. If environmentally qualified junction boxes had been specified on the connection diagrams and installed for these RTD's, it is possible that Construction personnel would have realized the need to install the RTD's/cables as supplied by the vendor. This also would have reduced the possibility of misinterpretation of the notes, because the sealed fitting on the junction box end of the bellows hose had to be cut off to install the flexible conduit over the cable. That note was probably intended to provide extra physical protection to the leads if they were supplied by the vendor with a braided armor covering.

The original Westinghouse installation drawing stated the RTD's/cables could be supplied with a braided armor covering. The revised Westinghouse drawing clearly showed the RTD's to be supplied with the improved stainless steel bellows hose fittings and stainless steel overbraid covering the cables (containing the leads). This design provided sufficient physical protection of the lead wires and eliminated the need to install flexible conduit over the cable. The note directing the installation of the flexible conduit over the leads could have been deleted by the Design Engineer responsible for review of the revised Westinghouse drawings. The revised Westinghouse drawings and the Environmental Qualification Report did not explain the importance of or describe what constituted a sealed junction box; this may have contributed to the Engineer's oversight.

The Design Engineer who originally discovered the incorrect junction boxes on November 5, 1987, was not performing an inspection of cables outside the junction boxes. He was inspecting the Raychem splices inside the junction boxes when he made his discovery. At the time, he had no reason to believe another discrepancy existed. Additionally, D/E did not have any documentation (Variation Notice) which indicated the improper installation of the cables/bellows hoses had been performed. Therefore, the initial D/E investigation of the incident seemed to be adequate at the time.

The Electricians who subsequently filled the Unit 1 RTD junction boxes with epoxy could not be expected to have discovered the cable/bellows hose discrepancy without being very familiar with the Westinghouse installation drawing. There was no reason for them to review that drawing for the work they were performing. The Duke RTD connection diagram does not show any special cable covering other than conduit. The connection diagram references the Westinghouse installation drawing.

The affected Unit 1 W/R RTD's have been replaced and their associated junction boxes refilled with epoxy. D/E revised the RTD connection diagram to eliminate the note requiring installation of flexible conduit over the leads if supplied by the vendor with braided armor. The affected Unit 2 W/R RTD's are in the process of being replaced and are being tracked by the Projects and Integrated Scheduling sections to ensure completion prior to heatup to Mode 3, following the current refueling outage.

In addition to LER 413/87-43, there have been eight other previous events involving Technical Specification violations due to Design Deficiencies at Catawba (see LERs 413/85-23, 413/85-68, 413/86-15, 414/86-45, 414/87-08, 413/87-12, 413/87-05, and 413/87-36). These earlier incidents did not involve environmental qualifications of equipment. Therefore, the corrective actions identified could not have prevented or shortened the duration of this event. There have been three previous events involving Technical Specification violations due to Installation Deficiencies at Catawba (see LERs 413/84-18, 413/86-15, and 413/87-02). These incidents did not involve environmental qualifications of equipment. Therefore, the corrective actions taken could not have prevented or shortened the duration of this event. This type of incident is considered to be recurring.

In addition to LER 413/87-43, there has been one other previous incident at Catawba involving non-environmentally qualified equipment (see Duke Power IIR C86-112-1). The incident resulted when wiring which could not be verified as environmentally qualified was installed in valve motor operators by the manufacturer.

This incident is not NPRDS reportable since no equipment malfunctions were involved.

<u>CORRECTIVE ACTION</u>:

SUBSEQUENT

(1) Affected Unit 1 W/R RTD's were declared inoperable.

(2) Unit 1 was shutdown in accordance with Technical Specifications.

(3) Affected Unit 1 W/R RTD's were replaced and their junction boxes refilled with epoxy.

(4) Note 6 on the RTD connection diagram was deleted.

(5) The affected Unit 2 W/R RTD's are currently being replaced.

<u>SAFETY ANALYSIS</u>:

The core exist thermocouples and the hot and cold leg wide range RTDs are used by the operator in addition to or instead of the narrow range hot and cold leg RTDs to determine temperature in the NC system. During mitigation of a high energy line break inside containment, the operator bases decisions and actions on these temperature indications. For design basis events which cause an adverse environment inside containment, temperature indications are used for three purposes:

(A) To determine the coldest temperature in the NC system
(B) To determine the hottest temperature in the NC system
(C) To determine the subcooling in the NC system.

(A) DETERMINATION OF COLDEST TEMPERATURE IN NC SYSTEM

This determination is typically made by monitoring the lowest cold leg RTD indication that is believed to be valid. The coldest NC system temperature is used for:

(1) Calculation of Cooldown Rate

Since the effect on the cold leg RTDs is a downward bias, a larger than actual cooldown rate might have been calculated. This might have caused implementation of emergency procedures dealing with mitigation of pressurized thermal shock conditions. These procedures might have mandated "soak" times, i.e., periods during which NC system cooldown is suspended to allow equalization of thermal stresses in the reactor vessel. These suspensions or the implementation of these additional procedures would have caused delays in cooldown and depressurization to cold shutdown.

(2) Determination of Unit Status with respect to Pressure-Temperature Operating Limits During Cooldown

Because of the unquantifiable nature of the bias introduced by submergence it is possible that the operator might have spent time performing unnecessary depressurizations to bring the unit below the maximum allowable pressure for an erroneously low temperature indication. Depending on the relative magnitude of the errors in the hot vs. cold temperature indications, the operator might even have been faced with conflicting indications of allowable minimum and maximum pressure. These situations would have led to operator confusion and attendant delay.

(3) Determination of Procedural Mitigation Path

Depending on whether the coldest NC system temperature is above or below 400 degrees F, the operator is directed along differing paths to recover the plant from a high energy line break inside containment. If NC system temperature is above 400 degrees F at the time of safety injection (SI) termination, recovery is continued in EP/1C1, SI Termination Following High Energy Line Break Inside Containment. If temperature is below 400 degrees F at this time, recovery is continued in EP/1D1, SI Termination Following Steam Line Break. While each procedure contains instructions for SI termination, the method in EP/1D1 is more rapid, as befits a condition where the concern is repressurization from cold conditions. In addition, EP/1D1 includes extra steps dealing with depressurization to within the previously discussed pressure-temperature limits. Having to reinitiate SI after a too rapid termination or to perform an unnecessary depressurization would have delayed the operator.

(4) Boration to Maintain Required NC System Boron Concentration

Having indicated temperatures colder than actual would have caused the operator to delay a post-accident cooldown while the NC system was borated to achieve the required boron concentration to maintain the Technical Specification shutdown margin.

(5) Determination of Required Component Status per Technical Specifications

During a cooldown the low setpoint on the pressurizer PORVs is enabled per Technical Specification 3.4.9.3.a. Erroneously low temperature indications might have caused an unnecessary depressurization to bring the NC system pressure below this setpoint. Also during a cooldown certain SI pumps are disabled per Technical Specification 3.5.3.a. Erroneously low temperature indications might have caused this disabling to be performed too early, resulting in a delay if the pumps were subsequently required.

(B) DETERMINATION OF HOTTEST TEMPERATURE IN NC SYSTEM

This determination is typically made by monitoring the highest hot leg RTD or core exit thermocouple indication that is believed to be valid. The hottest NC system temperature is directly used to determine unit status with respect to pressure-temperature operating limits during cooldown. It is also indirectly used in the subcooling calculation as discussed below. Because of the nature of the degraded measurement accuracy, two situations must be considered:

(1) The actual hottest temperature in the NC system is higher than the indicated hottest temperature

(2) The actual hottest temperature in the NC system is lower than the indicated hottest temperature

The first situation might lead to NC system voiding including NC pump cavitation since the operator might unknowingly violate the Net Positive Suction Head (NPSH) limit. The second situation might cause an unnecessary cooldown. However, since the upper limit of the discrepancy in this case is 45 degrees F, such a cooldown would not be great enough to cause a safety problem.

(C) DETERMINATION OF NC SYSTEM SUBCOOLING

The NC system subcooling is in general available to the operator as six numerical values divided into three numbers on each of two trains. One train gives the subcooling in two hot legs (based on the wide range RTDs) and in the upper plenum (based on twenty core exit thermocouples). The other train gives the subcooling in the other hot legs (based on the wide range RTDs) and an alternate indication of the upper planum subcooling (based on twenty different core exit thermocouples). All indications are internally adjusted for instrument error before being displayed. The indication on which decisions are based is typically the lowest one believed to be valid. For design basis events which cause an adverse environment inside contianment, subcooling is used for the following purposes:

(1) SI Termination

NC system subcooling is one of three criteria that would probably be sequentially satisfied for a high energy line break inside containment --NC system pressure stable or increasing, subcooling greater than a limiting value, and pressurizer level on scale. Indicated subcooling lower than actual might delay termination. Indicated subcooling higher than actual would probably have little effect since the pressurizer level criterion must also be satisfied for termination. The Reactor Vessel Level Indication System could be used to verify that NC system is actually water solid when the pressurizer level comes on scale.

(2) NC Pump Trip

The NC pumps are tripped in the emergency procedures for a high energy line break inside containment if indicated subcooling is less than 0 degrees F and one Safety Injection System (NI) or Chemcial and Volume Control (EIIS:CB) System (NV) pump is supplying SI flow to the NC system. This trip is to reduce inventory depletion during a small LOCA. Indicated subcooling being higher than actual would delay the pump trip. The delay would be reduced since the instrument error adjustment used to calculate the indicated subcooling would offset the thermocouple for RTD error under consideration. Indicated subcooling being lower than actual might cause a trip when none was required. This would complicate recovery from non-LOCA events.

(3) Cold Leg Accumulator (EIIS:ACC) Isolation

The cold leg accumulators are isolated during a controlled cooldown and depressuization when NC system pressure is less than 1000 psig and indicated subcooling is greater than 0 degrees F. Indicated subcooling being higher than actual might cause premature accumulator isolation. However, for a controlled shutdown following a design basis event, accumulator injection should not be necessary if it has not already occurred. Indicated subcooling being lower than actual might delay isolation and result in unnecessary accumulator injection, although this is only barely possible because of the pressure difference between the isolation setpoint and the upper limit of the allowable accumulator pressure bend. Accumulator reisolation setpoint and the upper limit of the allowable accumulator pressure bend. Accumulator reisolation, or compensation for the extra inventory if reisolation could not be performed, would delay mitigation of the accident.

(4) Cooldown to Achieve Adequate Subcooling

Certain scenarios involving high energy line break inside containment procedurally require that the operator initiate a cooldown to achieve 50 degrees F indicated subcooling. This occurs for scenarios which use EP/1C1, SI Termination Following High Energy Line Break Inside Containment and for which one NV pump operating through the normal charging lines can compensate for NC system inventory depletion or shrinkage. Indicated subcooling being higher than actual is not a safety concern since the 50 degrees F is only operational margin. Because stable conditions are already being maintained with minimal inventory might cause an unnecessary cooldown or might result in conflicting guidance if subcooling based on core exit thermocouples is relatively low while cold leg temperature based on wide range RTDs was also relatively low. In either case the operator would be delayed.

As discussed above, the effects of degradation in the ability to measure NC system temperature after a high energy line break inside containment might include the following:

(1) Operator confusion based on conflicting instrument indications or procedural instruments.

(2) Delays in the post-accident recovery caused by operator confusion, performance of unnecessary actions, or additional actions to reverse earlier unnecessary actions based on incorrect temperature or subcooling measurements.

(3) Possible damage to the NC pumps from violation of NSPH limits.

(4) Voiding in the NC system during depressurizations.

(5) Complication of recovery from non-LOCA scenarios due to unnecessary NC pump trip.

(6) Somewhat greater NC system inventory depletion during a small LOCA due to delayed NC pump trip.

As can be seen from the above list, with the exception of item 6, the FSAR Chapter 15 safety analysis are not affected by degradation of NC system temperature measurement accuracy. This is because 1) no protection system actuations are based on the wide range RTDs or the core exit thermocouples and 2) the operator actions assumed in the Chapter 15 analyses are either not based on

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

these instrument indications or contain sufficient time margin to accommodate any delays caused by incorrect indications. It should be noted that no credit is taken for NC pump operation in any Chapter 15 analysis. The FSAR Chapter 15 analysis do not in general cover post-accident recovery. As can be seen from the above list, recovery actions might be delayed or complication by the reduction in NC system temperature measurement accuracy. However, these delays and complications would not prevent the operators from placing the unit in a stable shutdown condition following a high energy line break inside containment.

This incident is reportable pursuant to 10 CFR 50.73, Section (a)(2)(i)(A).

The health and safety of the public were unaffected by this event.

DUKE POWER COMPANY
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VICE PRESIDENT
NUCLEAR PRODUCTION

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February 15, 1988

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

Subject: Catawba Nuclear Station, Unit 1
Docket No. 50-413
LER 413/88-03

Gentlemen:

Pursuant to 10 CFR 50.73 Section (a) (1) and (d), attached is Licensee Event Report 413/88-03 concerning Technical Specification violations because the Wide Range Temperature Monitoring Instrumentation was rendered technically inoperable during certain accident conditions due to installation and design deficiencies. This event was considered to be of no significance with respect to the health and safety of the public.

Very truly yours,

Hal B. Tucker

Hal B. Tucker

JGT/1399/sbn

Attachment

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IE22
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