Entergy

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Pete Dietrich
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JAFP-10-0072
June 17, 2010

United States Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, D.C. 20555

SUBJECT: Response to Request for Additional Information Re: James A. FitzPatrick Nuclear Power Plant Application for Amendment to Modify the Technical Specifications Requirements for Testing of the Shutdown Cooling System Isolation, Reactor Pressure – High Function (TAC No. ME1819)
James A. FitzPatrick Nuclear Power Plant
Docket No. 50-333
License No. DPR-59

References:
1. Entergy Letter, JAFP-09-0086, Application for Amendment to Modify the Technical Specifications Requirements for Testing of the Shutdown Cooling System Isolation, Reactor Pressure – High Function (TAC No. ME1819), dated July 31, 2009

2. NRC Request For Additional Information Regarding James A. FitzPatrick Nuclear Power Plant Application for Amendment to Modify the Technical Specifications Requirements for Testing of the Shutdown Cooling System Isolation, Reactor Pressure – High Function (TAC No. ME1819), dated February 2, 2010

3. Entergy Letter, JAFP-10-0033, Response to Request for Additional Information Re: James A. FitzPatrick Nuclear Power Plant Application for Amendment to Modify the Technical Specifications Requirements for Testing of the Shutdown Cooling System Isolation, Reactor Pressure – High Function (TAC No. ME1819), dated March 5, 2010

4. NRC Request For Additional Information Regarding James A. FitzPatrick Nuclear Power Plant Application for Amendment to Modify the Technical Specifications Requirements for Testing of the Shutdown Cooling System Isolation, Reactor Pressure – High Function (TAC No. ME1819), dated April 13, 2010

Dear Sir or Madam:

On July 31, 2009, Entergy Nuclear Operations, Inc. (ENO) submitted an application for amendment to the Technical Specifications (TS) for the James A. FitzPatrick Nuclear Power

Plant (JAF), to revise the surveillance testing requirements for the Shutdown Cooling System Isolation, Reactor High Pressure Function (Reference 1). On February 2, 2010, JAF received a request for additional information from the Nuclear Regulatory Commission (NRC) staff (Reference 2). The request was subsequently clarified in a conference call with the staff on February 17, 2010. Based on the clarifying discussions with the staff, ENO provided the responses documented in Reference 3. After reviewing Reference 3, the NRC staff advised ENO of additional questions. The additional questions were discussed in a teleconference on April 5, 2010, with written questions being provided on April 13, 2010 (Reference 4).

Based on the discussions with the staff, ENO is providing this response to the request for additional information. Attachment 1 provides a response to each RAI question.

The attached response does not affect the No Significant Hazards Determination submitted with the amendment application, dated July 31, 2010.

There are no new commitments made in this letter.

Questions concerning this report may be addressed to Mr. Joseph Pechacek, Licensing Manager, at (315) 349-6766.

I declare under penalty of perjury that the foregoing is true and correct.

Executed on the 17th day of June 2010.

Sincerely,

Pete Dietrich
Site Vice President - JAF

PD/JP/ed

Attachments: 1. Responses to Request for Additional Information Questions
2. Source Document Excerpts

cc: next page

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Document Component(s):
001 Transmittal letter JAFP-10-0072 with Attachments

JAFP-10-0072

Attachment 1

Responses to Request for Additional Information Questions

(7 Pages)

Question 1:

"In Attachment 1, on page 3, of the licensee's RAI response letter dated March 5, 2010 (JAFP-10-0033), it states in the first paragraph:

> "Following performance...to review the completed procedure to ensure acceptance criteria was satisfied and that "As-Found" and "As-left" values are within their required tolerance."
>
> And in the second paragraph, it states:
>
> "If a Condition Report documents that an instrument failed to meet Level 1 acceptance criteria and could not be restored to within tolerance, the instrument would be declared INOPERABLE and the appropriate Condition associated with the applicable Limiting Condition for Operation (LCO) would be entered."

a. Clarify the use of the terms and provide the values of the "acceptance criteria", the "required tolerances", and "Level 1 acceptance criteria."

b. Provide the detailed calculation and/or determination of "As-Found" and "As-Left" setting tolerances of the entire instrument loop."

Response:

1.a In accordance with our Surveillance Testing Program procedure AP-19.01, acceptance criteria is defined as "The measure against which Surveillance Test Procedure (STP) performance results are evaluated to determine if SRs [Surveillance Requirements] are met." In paragraph one of the RAI response letter dated March 5, 2010 (JAFP-10-0033), the use of the term "acceptance criteria" is in reference to the Technical Specification Allowable Values (TSAVs). The surveillance review, discussed in the RAI response, compares the calibration results to the TSAVs. The acceptance criteria that would be used for this comparison would be the TSAV for the Shutdown Cooling System High Reactor pressure isolation function which is ≤ 74 psig.

In paragraph two of the RAI response letter dated March 5, 2010 (JAFP-10-0033), the use of the term "Level 1 acceptance criteria" is in reference to the Allowable Values (AV) as provided in our Technical Specifications (TS). This term is defined in our Surveillance Testing Program procedure AP-19.01 as "The measure that defines characteristics of a system or component that, if <u>not</u> met, result in a violation of TS [Technical Specifications], TRM [Technical Requirements Manual], ODCM [Offsite Dose Calculation Manual], or plant safety design bases as stated in the UFSAR [Updated Final Safety Analysis Report] or EN-DC-167." [Detail Added] The TSAV for the Shutdown Cooling System High Reactor pressure isolation function is ≤ 74 psig. Therefore, this value is considered a Level 1 acceptance criterion.

The use of the term "required tolerances" is in reference to the criteria provided in the procedures for the calibration of the equipment. The required tolerances are referred to as "As-Found Zones" and "As-Left Tolerances". The 4-20mA transmitter output represents a 1200 psig range. The instrument calibration criteria or required tolerances for the Slave Trip Units (STUs), [02-3STU-251A and 02-3STU-251D] are ± 3.39 psig (±0.04 mA, As-Found Zone) and ± 2.4 psig (± 0.03 mA, As-Left Tolerance).

1.b The detailed calculation was previously provided with the RAI response letter dated March 5, 2010 [Entergy calculation number JAF-CALC-NBS-02052]. The shutdown cooling function is not required to mitigate the consequences of an accident and thus the instrument errors due to harsh environmental effects including process measurement and Insulation Resistance Effect (IRE) do not need to be considered. Per the setpoint calculation methodology presented in ENN-IC-G-003, the As-Found setting tolerance (AFZ) is set the same as the Allowable Value to Trip Setpoint Margin (AVTSM) and the values are scaled as necessary for limitations of test equipment (refer to Section 7.1 in JAF-CALC-NBS-02052 for scaling). The AVTSM is specified for each individual instrument in sections 6.5.1, 6.5.2, and 6.5.3 of JAF-CALC-NBS-02052 as follows: Transmitter AVTSM (AFZ) = ±13.83 psig, Master Trip Unit (MTU) AVTSM (AFZ) = ±3.53 psig and STU AVTSM (AFZ) = ±3.39 psig.

The combination of these individual instrument As-Found setting tolerances equate to the As-Found setting tolerance for the entire instrument loop. This value is presented in section 6.5.4 and is specified as ±14.67 psig ($AVTSM_{Channel}$).

The As-left Tolerance (ALT) for each individual instrument is provided in JAF-CALC-NBS-02052 sections 6.2.1.10, 6.2.2.10, and 6.2.3.10 as follows: Transmitter ALT = ±3.00 psig, MTU ALT = ±2.25 psig, and STU ALT = ±2.4 psig.

Due to the manner in which the instruments are calibrated, the ALT for the entire instrument loop is not necessary and thus is not computed in the setpoint calculation JAF-CALC-NBS-02052. If it was necessary, it would be calculated by combining the ALT of each individual instrument using a Square Root Sum of the Squares method since each of the ALT errors are random in nature. For your reviewing convenience, the ALT for the entire instrument loop is presented below:

$$ALT_{Loop} = \pm \sqrt{PT_{ALT}{}^2 + MTU_{ALT}{}^2 + STU_{ALT}{}^2}$$

$$ALT_{Loop} = \pm \sqrt{3.00^2 + 2.25^2 + 2.4^2} \text{ (all units are psig)}$$

$$ALT_{Loop} = \pm 4.45 \text{ psig}$$

Question 2:

"In attachment 2, section 8.2.3, of the licensee's RAI response, it states:

> "Use of the reactor steam dome pressure measurement (ATTS) to provide the SDC isolation function will result in the utilization of an instrument loop that has a larger instrument uncertainty than originally provided by the pressure switches."

The licensee calculated the allowable value (Section 6.5, Attachment 2 of the RAI response) by applying the "method 3" of ISA-RP 67.04.02-2000. The licensee also calculated the limiting trip setpoint LTS = EAL – CU (section 6.4) with a round up margin of 0.72 psig (section 6.3.1). However, the round up of $AVTSM_{Channel}$ from ±14.67 psig to ±15 psig increases the allowance for instrument channel deviation (section 6.3.1); thus is less conservative and cannot be counted as an addition margin.

Explain how you can ensure that the field trip setpoint is conservative with respect to the analytical value (AL) with "method 3" and with such a small margin in the LTS calculation."

Response:

As defined in our instrument loop accuracy and setpoint methodology procedure ENN-IC-G-003, the "Allowable Value to Trip Setpoint Margin" (AVTSM) is the difference in the parameter of interest between the Allowable Value (AV) and the Trip Setpoint. At a minimum, the AVTSM consists of all channel As-Found Tolerance (AFT) uncertainties. The AVTSM can be all uncertainties representing normal operation conditions that will be experienced between successive calibrations."

Even though increasing the $AVTSM_{Channel}$ from ±14.67 psig to ±15 psig increases the allowance for instrument channel deviation, the direction of conservatism is based on perspective. By increasing this allowed channel deviation, it is less likely that the TSAV would be exceeded. This is because the resultant the field trip setpoint would be set to a lower value. This is considered conservative since the SDC isolation function occurs on an increasing pressure and a lower setpoint would actuate earlier. The potential for exceeding the TSAV is further minimized by the application of the Surveillance Testing Program, Corrective Action Program, and instrument calibration procedures. The details of these processes were discussed in the RAI response letter dated March 5, 2010 (JAFP-10-0033). These processes are in place to monitor and identify trends of instrument behavior relative to the TSAVs. Therefore, the combination of procedures, the TSAV, and the additional allowance for instrument channel deviation between the field trip setpoint and the AV will adequately protect the Analytical Limit (AL).

Increasing the AVTSM is only acceptable as long as the AL is adequately protected. As requested, the margin in the field trip setpoint calculation and assurance of a conservative field trip setpoint with respect to the AL using the ISA-RP 67.04 "method 3" are evaluated below.

Calculation JAF-CALC-NBS-02052 determines both a total loop uncertainty (CU_{STU} in Section 6.3.1) and the Allowable Value to Trip Setpoint Margin ($AVTSM_{Channel}$ in Section 6.5.4). Per our setpoint calculation methodology, one main difference between the total loop uncertainty and the AVTSM is the instrument errors associated with harsh environments or abnormal conditions. The difference between the calculated total loop uncertainty of ±15.28 psig and the calculated $AVTSM_{Channel}$ of ±14.67 psig is 0.58 psig. Considering this and the difference of 1 psig between the AL of ≤75 psig and the TSAV of ≤74 psig, there is additional margin of 1.0 psig – 0.58 psig = 0.42 psig between the AL and the AV. Though this may not be considered a large amount of margin, it is still margin. The amount of applied margin is not a specified value; thus, any amount of margin adds conservatism. From an operational perspective, application of too much margin can have negative impacts such as pushing the setpoint into the normal operating region of the process.

Based on the preceding information, the allowed instrument channel deviation has been increased to help ensure the TSAV is not exceeded. In addition, Entergy has procedures in place to monitor and identify trends of instrument behavior relative to the TSAVs. Also, a margin of 0.42 psig is provided between the Shutdown Cooling System High Reactor pressure isolation function AV and the associated AL. Furthermore, the Rosemount Analog Transmitter Trip System (ATTS) that will be used for the Shutdown Cooling System High Reactor pressure isolation function has a proven history of reliability and accuracy at JAF. Therefore, the nominal field trip setpoint of 59 psig is considered conservative and will adequately protect the AL.

Question 3:

In Attachment 2, Table 1, of the licensee's RAI response, it states that the humidity, radiation, and temperature effects for the total loop uncertainty calculation of pressure transmitters and ATTS under LOCA and HELB conditions are same as those under normal condition. This statement is based on assumption 1.2.6. Assumption 1.2.6 states that "DBD-010 (Ref 4.2.11) identifies the SDC Isolation function as not having any accident mitigating functions." However, according to TS Table 3.3.6.1-1 (page 5 of 6), Function 6a "SDC Isolation, Reactor Pressure – Hi" is used to initiate a primary containment isolation function. Assumption 1.2.6 seems to contradict with this TS requirement. Please provide the following:

a. Where the pressure transmitter 02-3PT-55A/D is located (inside or outside the primary containment)?

b. What are the worst-case environmental conditions at the locations for the transmitters and ATTS?

c. What are the worst-case environmental conditions under which the SDC isolation valves are required to be actuated?

d. Provide a clarifying description regarding the conditions under which the function of this instrument channel is required when using the SDC line that resolves the apparent contradiction between assumption 1.2.6 and the TS requirement of Function 6a in the TS Table 3.3.6.1-1 (page 5 of 6).

e. Provide copies of the source documents that justify the values for seismic, radiation, humidity, and temperature values used in the calculation, JAF-CALC-NBS-02052.

Response:

3.a The pressure transmitters, 02-3PT-55A and 02-3PT-55D, are located outside of primary containment on the 300 foot elevation of the reactor building.

3.b The worst case environmental conditions for the transmitters 02-3PT-55A, B (RB EL. EQ Node RB300Switch) are 100% relative humidity, 3.19 x 10^5 Gamma and 3.04 X 10^5 Beta and 215°F.

The worst case environmental conditions for the transmitters [02-3PT-55C and 02-3PT-55D] (RB EL. EQ Node RB300North) are 100% relative humidity, 1.07 x 10^5 Gamma and 3.04 X 10^5 Beta and 158.38°F.

The worst case environmental conditions (Ref.DBD-70) for the ATTS (RR EL. 284) are 50% relative humidity, 1.75 x 10^2 R TID (40 years) and 104°F.

As discussed in section 1.2.6 of JAF-CALC-NBS-02052, the Shutdown Cooling System High Reactor pressure isolation function is not credited in any accident analysis and is not required to mitigate the consequences of an accident. Therefore, the worst case environmental conditions that the transmitters 02-3PT-55A and 02-3PT-55D will be exposed to when the instruments are required to operate is the same as normal environmental conditions. For further information, see response to 3.c and 3.d below.

The MTUs and STUs that make up the ATTS are physically located in the Relay Room. This room is below the Control Room and has an HVAC system that controls the environmental conditions. Therefore, the worst case environmental conditions that the MTUs and STUs will be exposed to when the instruments are required to operate are the same as normal environmental conditions. The normal environmental conditions are presented in Section 3.2 Table 1 of JAF-CALC-NBS-02052.

3.c To satisfy the TS requirement Function 6a in TS Table 3.3.6.1-1, the SDC isolation valves, 10MOV-17 and 10MOV-18, must close when a reactor high pressure condition exists. When this condition occurs, if the valves are not already closed, they will automatically close with a trip of the SDC System High Reactor pressure isolation logic. Since the plant is normally operating and a reactor high pressure condition exists, these valves are normally closed.

When plant power/pressure is reduced, the SDC isolation valves, 10MOV-17 and 10MOV-18, can only be opened when: 1) the Shutdown Cooling System High Reactor pressure isolation logic is reset; and 2) the Plant Operator manually initiates SDC by opening the valves. The logic of the circuit is designed such that a low reactor pressure condition (plant must be shutdown) is required in order for the isolation logic to be reset. Therefore, the plant condition that would exist when the SDC isolation valves are required to actuate, per Function 6a of TS Table 3.3.6.1-1 (from open to closed), is plant heatup from a shutdown condition. If the plant is in a state of heatup, normal environmental conditions exist. Thus the worst case environmental conditions under which the SDC isolation valves are required to actuate (from an open to a closed state) are normal environmental conditions.

3.d The conditions under which the function of this instrument channel is required will be discussed in reference to reactor low and high pressure conditions:

Low Reactor pressure – Plant shutdown:
When reactor pressure is equal to or less than the SDC logic reset value, a reactor low pressure logic condition is generated which interfaces with two separate logic circuits via relay contacts. This will allow two functions to occur: 1) the manual opening of the SDC isolation valves, 10MOV-17 and 10MOV-18, will be permitted to support the operation of the SDC mode of RHR; and 2) a portion of the RHR injection valve (10MOV-25A, B) control circuit will be satisfied. This separate RHR injection valve control circuit will automatically close the RHR (LPCI) injection valves 10MOV-25A and 10MOV-25B if 10MOV-17 and 10MOV-18 are both open and a reactor low level or a drywell high pressure signal is received. The purpose of this second portion of the logic is to isolate a possible reactor drain path which could have been created during the SDC mode of operation.

High Reactor pressure – Plant operating:
Under this condition, an isolation of the SDC mode of RHR is required per TS Table 3.3.6.1-1 Function 6a. This requirement is satisfied as follows: When reactor pressure is greater than the SDC logic trip value, the isolation logic will trip generating a reactor high-pressure logic condition to automatically close the SDC isolation valves 10MOV-17 and 10MOV-18. The purpose of closing 10MOV-17 and 10MOV-18 is to prevent the

over pressurization of the RHR pump suction piping. In addition, the logic associated with the RHR injection valve control circuit (10MOV-25A and 10MOV-25B) that is in place only during SDC mode of RHR will be tripped (disabled) since 10MOV-25A and 10 MOV-25B are closed during normal plant operation.

The discussion above supports the association between Section 1.2.6 of JAF-CALC-NBS-02052 and the TS requirement of Function 6a in TS Table 3.3.6.1-1.

3.e Due to the size of the source documents, only the pertinent excerpts that support the environmental conditions as presented in Section 3.2 of JAF-CAL-NBS-02052, Table 1 are attached. As discussed in Section 1.2.5 of JAF-CALC-NBS-02052, seismic uncertainty effects are not included, because the instrument loop's function is not required during and following a seismic event. Should a seismic event occur, operability of the instruments will be evaluated as part of an abnormal operating procedure.

JAFP-10-0072

Attachment 2

Source Documents Excerpts
(12 Pages)

* Applicable to JAF-CALC-NBS-02052

ENVIRONMENTAL QUALIFICATION

Table 4.2.1.1.1 Normal Reactor Building Temperatures

EQ Node	Old HELB Node	Temp. (°F)	Percent (%) at Temp.	Reference	Comment
RB300RWCUA	300-2	120	100	[5.4.2.1.1.5] [5.4.2.1.1.2]	Normal ambient temperatures in these rooms have generally increased from the original design values provided in GE Specification No. 22A2928 [5.4.13.1]. Therefore, the EQ normal ambient temperature has been revised upwards periodically.
RB300SWITCH	300-3	85 100	90 10	[5.4.2.1.1.2]	
RB300MGSET	300-4	85 100	90 10	[5.4.2.1.1.2]	
RB300NORTH	300-5	85 100	90 10	[5.4.2.1.1.2]	
RB300STAIR	300-6	85 100	90 10	[5.4.2.1.1.2]	
RB300HTEXCH	300-7	105	100	[5.4.2.1.1.2]	
RB300TANK	300-8	85 100	90 10	[5.4.2.1.1.2]	
RB300SAMPLE	300-9	85 100	90 10	[5.4.2.1.1.2]	
RB300SLUDGE	300-10	104	100	[5.4.13.1]	No temperature data identified. No EQ equipment in enclosed room location. Temperature assumed equivalent to worst-case RB general area normal, from GE Report 22A2928 [5.4.13.1].
RB326	326-1 326-2 326-3 326-4 326-5 326-6 326-7 326-8	85 100	90 10	[5.4.2.1.1.2]	Per Reference [5.4.2.1.1.2, page 9], The Fuel Pool Cooling Heat Exchanger Room equipment does not provide a significant source of heat to the room due to the following: 1. The maximum design temperature of the fuel pool (135°F) is greater than the fuel pool inlet temperature to the heat exchangers (125°F), and 2. The heat exchangers are operated on an intermittent basis to offset the heat load input from the stored spent fuel. Therefore, this room is considered to have an equivalent normal ambient temperature as the RB 326 ft. General Area.

Plant and Equipment Operation

Normal ambient temperatures may fluctuate significantly between times of plant operation and shutdown periods. Note that the average temperature for the Reactor Building over the life of the plant is currently based on plant operational temperatures, which is conservative. Future use of plant capacity factors and temperature monitoring of shutdown periods can provide more realistic normal ambient temperatures. Note that certain rooms, such as the RHR Heat Exchanger Rooms, may be at higher temperatures when the plant is not in operation.

Similar effects can be observed for certain equipment during equipment operational periods, which is closely related to localized area temperatures. For example, the rooms for pumps and heat exchangers experience higher temperatures during periods of equipment operation due to the additional heat input [5.4.2.1.1.2, pg. 2].

Note that this report (JAF-RPT-MISC-04046) does *not* examine internal equipment heating (e.g., electrical MCC and switchgear enclosure, energized Solenoid Operated Valve (SOV), motor, or Motor Operated Valve (MOV) internal temperatures). Rather, this information is provided in the individual EQ Qualification Documentation Reports (QDRs).

4.2.1.2 Pressure

The normal Reactor Building pressure is maintained between (-) 0.10 inches of water gage (abbreviated in w.g.) to (-) 1.0 inches w.g. [5.4.13.1, Section A-I].

4.2.1.3 Relative Humidity

Reactor Building normal relative humidity (RH) varies between 20% and 90% with a nominal value of 40% RH, per GE Document No. 22A2928 [5.4.13.1, Section A-I]. The power uprate effect on these values is negligible, in that slightly warmer fluid in the steam lines may increase the overall Reactor Building temperatures. This slight increase in temperature increases the capability of the air to hold moisture and result in a slight decrease in the Reactor Building normal relative humidity. DBD-066 "Inside Design Conditions" Table confirms the above normal RH values [5.4.2.1.1.4].

4.2.1.4 Radiation

Gamma radiation is the primary contributor to normal service doses. There is no neutron dose outside the Drywell and beta dose from fission or corrosion products are contained within the reactor coolant and auxiliary systems.

Appendix B provides the normal radiation dose and dose rate for each Reactor Building EQ Node based on the values presented in Table 4.2.1.4.

Design normal radiation values in the Reactor Building were provided in GE Document No. 22A2928 [5.4.13.1, Section B]. Subsequently, actual radiation measurements made during power operation have been incorporated in to the radiation service conditions. Table 4.2.1.4 combines the data from the various radiation monitoring surveys/references for the EQ normal service conditions.

The effects of power uprate are included in Table 4.2.1.4. Since primary coolant activities are controlled (for example, RWCU flow maximization, zinc injection and piping decontamination), power uprate conditions were not expected to significantly affect Reactor Building normal service conditions [5.4.12.6, Section 4.2.1]. Subsequent reviews of normal radiation values after power uprate identified that Elev. 326' experienced increased radiation values [5.4.1.1.4.2]. A radiological survey was also performed in 1999 of all Reactor Building areas. An assessment of the data in Memorandum JRES-99-200 [5.4.2.1.4.4] identified elevated radiation doses at Elev. 326' and 344'. These effects have been incorporated into the normal service dose rates in Table 4.2.1.4.

Table 4.2.1.4 General Area Reactor Building Normal Radiation

Reactor Building Floor Elevation (1)	Dose Rate (mr/Hr)	Dose (Rad)	Reference
227' (2)	200	7.0E4	[5.4.2.1.4.1] [5.4.2.1.4.2]
272' (3)	50	1.8E4	[5.4.2.1.4.1] [5.4.2.1.4.2]
300' (5)	50	1.8E4	[5.4.2.1.4.1] [5.4.2.1.4.2]
326'	200	7.0E4	[5.4.1.1.4.2]
344'	15	5.26E3	[5.4.1.1.4.2] [5.4.2.1.4.4] [5.4.2.1.4.3]
369' (4)	15	5.26E3	n/a

Notes:

1. For specific EQ Node doses, refer to the Appendices of this report.
2. The Reactor Building at Elev. 227' is representative of the Torus Room, as confirmed by historical field verifications [5.4.2.1.4.1].
3. The Reactor Building at Elev. 272' is considered representative of the SBGT Room, which is in a separate enclosed room adjacent to the Reactor Building general area. SBGT equipment is considered to have minimal internal normal dose.
4. Radiation monitoring has not been performed for Elev. 369'. The dose rate at Elev. 369' is considered the same as that of Elev. 344'.
5. RWCU heat exchanger room dose rate is 15 Rad/hr with a 40-year integrated normal service dose of 5.4E6 Rad. RWCU filter and tank dose rate is 10 Rad/hr with a 40-year normal service dose of 3.6E6 Rad. Values are based on Reference 5.4.13.1.

Document No. DBD-070

ENTERGY NUCLEAR NORTHEAST

JAMES A. FITZPATRICK NUCLEAR POWER PLANT

* Applicable to JAF-CALC-NBS-02052

DESIGN BASIS DOCUMENT

FOR THE

CONTROL ROOM RELAY ROOM VENTILATION AND COOLING SYSTEMS

	Print/Sign/Date		Print/Sign/Date
Prepared by:		ENN DBD Owner:	MATTHEW E. CLARK Matthew E. Clark 5/30/06
Reviewed by:		ENN Prog. Engr.:	John M. Erkan John M. Erkan 5/30/06
Approved by:		ENN Mgr. Design Eng.:	John P. Gerety 6/1/06

When an air handling unit, 70AHU-3A or B, is operating, the corresponding chilled water pump, 70P-9A or B, starts automatically when the outside air temperature is above nominal 50°F as sensed by temperature switch 70TS-106A or B. The respective flow switch, 70FIS-100A or B, sensing the flow of water automatically starts the corresponding chiller, 70RWC-2A or B. (Ref. 6.7.6, 6.7.7, 6.8.I.15, 6.8.1.17, 6.13.17, Subsect. 7.1.42)

The operating chiller, 70RWC-2A or B, supplies chilled water to the air handling unit cooling coils. The relative humidity of the air discharged by the air handling unit is controlled by 70TIC-105 and 70TIC-107 by regulating chilled water flow through control valves 70TCV-121A, B (Subsect. 2.2.3). 70TIC-105 and 70TIC-107 also control electric heaters E-8 and E-7 respectively, which are installed in the ducts, to raise the discharge air temperature and effectively reduce relative humidity as required in response to room thermostats 70TIC-106 and 70TIC-108 (Ref. 6.8.3.2, 6.8.1.1, 6.8.1.18, 6.8.2.43).

When the Control Room temperature rises above 98°F, temperature switch 70TS-109A or B de-energizes the actuators of both return air dampers to open both dampers, and closes the outside air and exhaust air dampers. (Subsect. 7.1.28, 8.1.6) It also starts the redundant standby air handling unit, exhaust fan and chiller, and interrupts the power supply to electric heaters 70E-7 and 70E-8. (Ref. 6.8.1.1, 6.8.1.6, 6.8.1.12, 6.8.1.13, 6.8.1.18, 6.8.2.43)

Water is not directly added to the control room. This lack of humidification control means that lower relative humidities cannot be controlled. Static discharges may occur due to low relative humidities. Static discharges generally can occur when humidity falls below 45%. Electrostatic charges are generated when materials of high electrical resistance move against each other. The accumulation of such charges can create static discharges and can potentially destroy data stored on magnetic disks and tapes or possibly ignite an explosive gas mixture. However, no explosive gases or safety-related equipment relying on magnetic disks or tapes exists in the control room. (Ref. 6.3.8, 6.8.3.1, 6.8.3.2) (ACTS 99-45457, 99-45458, 99-45656, 99-45391, 99-45392)

2.2.1.2 RRHV System Normal Operating Functions

Requirements: The Relay Room shall be supplied with outside filtered air (Ref. 6.7.1).

The Relay Room shall be maintained at a slight positive pressure (Ref. 6.7.1.).

The Relay Room shall be maintained at a temperature between 60°F and 90°F and a relative humidity between 40% and 50% for equipment operability (Ref. 6.10.3).

INSTRUCTION MANUAL 4247-1

OPERATIONS MANUAL

TRIP/CALIBRATION SYSTEM

MODEL 510DU

* Applicable to JAF-CALC-NBS-02052

Rosemount

TABLE 4

COPPER WIRE, DC RESISTANCE

(Per ASTM Specification B1-56)

AWG Size	DC Resistance at 20°C Maximum Ohms per 1,000 Feet
8	0.824
10	1.04
12	1.65
14	2.63
16	4.18
18	6.64
20	10.50

NOMINAL POWER CONSUMPTION (exclusive of output loads):

Master Trip Unit (with or without Analog Output): 105 mA.

Slave Trip Unit: 90 mA.

Calibration Unit: 140 mA.

Readout Assembly: 475 mA.

OUTPUTS:

24 Vdc for each trip output and gross failure output;

12 Vdc for each trip status output;

24 Vdc calibration status signal for remote indication;

Optional 1 to 5 Vdc analog signals proportional to transmitter input.

TRIP OUTPUT LOGIC:

Normal trip output logic provides a 24 Vdc output when transmitter current is greater than the trip point. Reversed trip output logic provides a 24 Vdc output when transmitter current is less than the trip point. Trip output logic is indicated by NORM and REV at the trip output logic switch on the printed circuit board (Figure 9).

TRIP STATUS OUTPUT/LED LOGIC:

Normal trip status output logic provides a 12 Vdc output and trip status LED on when trip output is 24 Vdc. Reversed trip status output logic provides a 12 Vdc output and trip status LED on when trip output is 0 Vdc. Trip status output logic is indicated by NORM and REV at the trip status output/LED logic switch on the printed circuit board (Figure 9).

Accompanying Instrumentation

RELAYS:

Relays should be selected and supplied by the customer in accordance with specific environmental requirements.

TRANSMITTERS:

Two-wire, 4 to 20 mA transmitters are required, and should be purchased by the customer according to particular requirements. Rosemount's line of Model 1151 and 1152 Pressure Transmitters, and the Model 535E Temperature Transmitter are qualified for use in nuclear power generating stations.

CABINET:

A standard 19-inch rack for housing the 510DU System, power supplies, and relays should be supplied by the customer. Location of the cabinet at the customer site should be such that the resistance of the wire used to make connections between the 510DU System and the transmitters does not exceed 16 ohms. This assures a minimum turn-on voltage for the transmitters of 15.0 Vdc when the power supply is at a minimum value of 22.0 Vdc (adverse operating conditions).

Environmental Specifications

PLANT ENVIRONMENTAL CONDITIONS: See Table 5.

SEISMIC VIBRATION:

The Card File, Master and Slave Trip Units, and Calibration Unit operate during and after exposure to seismic vibration of 11 g's peak in all axes.

ELECTROMAGNETIC SUSCEPTIBILITY:

The 510DU operates in EMI conditions normally expected in a power plant control room environment, provided that shielded wires are used for transmitter connections and the auxiliary analog output.

TABLE 5

ENVIRONMENTAL CONDITIONS

Plant Operating Condition	Environment	Temperature		Relative Humidity (%)	Radiation Exposure			Power Supply (Vdc)
		°F	°C		Dose Rate (Rads Si/Hour)	Integrated Dose (Rads Si) 6 Months	Integrated Dose (Rads Si) 40 Years	
Normal	Normal	60 to 90	15.56 to 32.22	40 to 50	5.0×10^{-4}	–	1.75×10^{2}	23.5 to 26.5
Adverse	Normal	40 to 120	4.44 to 48.89	10 to 60	1.0×10^{-3}	3.0	–	22.0 to 28.0
Normal	High	40 to 104	4.44 to 40.00	20 to 90	1.5×10^{-2}	–	5.3×10^{3}	23.5 to 26.5
Adverse	High	40 to 156	4.44 to 68.90	20 to 99	6.5×10^{2}	1.7×10^{5}	–	22.0 to 28.0

INSTRUCTION MANUAL 4471-1
Revision A

Instruction Manual

Operations Manual
Trip/Calibration System
Model 710DU

* Applicable to JAF-CALC-NBS-02052

Rosemount®

TABLE 4

COPPER WIRE, DC RESISTANCE
(Per ASTM Specification B1-56)

AWG Size	DC Resistance at 20°C Maximum Ohms per 1,000 Feet
8	0.824
10	1.04
12	1.65
14	2.63
16	4.18
18	6.64
20	10.50

TRIP OUTPUT LOGIC:

Normal trip output logic provides a 24 Vdc output when input signal is greater than the trip point. Reversed trip output logic provides a 24 Vdc output when input signal is less than the trip point. Trip output logic is indicated by NORM and REV at the trip output logic switch on the printed circuit board (Figure 8).

TRIP STATUS LED LOGIC:

Normal trip status logic provides a 12 Vdc output and trip status LED on when trip output is 24 Vdc. Reversed trip status logic provides a 12 Vdc output and trip status LED on when trip output is 0 Vdc. Trip status logic is indicated by NORM and REV at the trip status LED logic switch on the printed circuit board (Figure 8).

Accompanying Instrumentation

RELAYS:

Relays should be selected and supplied by the customer in accordance with specific environmental requirements.

INPUT SENSORS:

Transmitters: Two-wire or 4-wire, 4-20mA transmitters are required, and should be purchased by the customer according to particular requirements. Rosemount's line of Model 1152 and 1153 Pressure Transmitters are qualified for use in Nuclear Power Generating Stations.

Resistance Temperature Detectors (RTD's): Shielded, 3-wire, platinum RTD's with R_0 = 100 ohms are required and should be purchased by the customer according to particular requirements.

CABINET:

A standard 19-inch rack for housing the 710DU System, power supplies, and relays should be supplied by the customer. Location of the cabinet at the customer site should be such that the resistance of the wire used to make connections between the 710DU System and the transmitters does not exceed 16 ohms. This assures a minimum turn-on voltage for the transmitters of 15.0 Vdc when the power supply is at a minimum value of 22.0 Vdc.

Environmental Specifications

ENVIRONMENTAL CONDITIONS: See Table 5.

SEISMIC VIBRATION:

The Card File, Master Trip Units, and Slave Trip Units operate during and after exposure to seismic vibration with a ZPA of 1.17 g OBE and 1.75 g SSE. See Rosemount Report D8200037.

ELECTROMAGNETIC SUSCEPTIBILITY:

The 710DU operates in EMI conditions normally expected in a power plant control room environment, provided that shielded wires are used for all signal connections and the auxiliary analog output.

TABLE 5

ENVIRONMENTAL CONDITIONS

Operating Condition		Normal	Transient	Accident (Includes Margin)
Temperature	°F	60 to 90	160 for 24 hours once per year	185° for 6 hours
				150° for 8 hours
	°C	15 to 32	71 for 24 hours once per year	85° for 6 hours
				65.6° for 8 hours
Relative Humidity		40 to 50%	90% for 24 hours once per year	90% for 14 hours
Radiation		$\leq 10^5$ Rad (air) TID over 20 years		2×10^5 Rad (air) TID in 24 hours
Power Supply		22 to 28 Vdc		

★ Applicable to TAF-CALL-NBS-02052 [redacted]

Ref. #44

GENERAL ELECTRIC

NUCLEAR ENERGY DIVISION

Document No. 22A2928 Rev. 2

General Electric Class

TRANSMITTAL

PROJECT(S) STANDARD PLANTS 236X350A,B,AG4,BG4; CHINSHAN, FITZPATRICK, FUKUSHIMA-2, COOPER, SUSQUEHANNA, NEWBOLD 1&2, HATCH-2, LIMERICK

TITLE OF DOCUMENT BWR EQUIPMENT ENVIRONMENTAL INTERFACE DATA

TYPE OF DOCUMENT:
[ ] PURCHASE SPECIFICATION
[X] SYSTEM DESIGN SPECIFICATION
[ ] INSTALLATION SPECIFICATION
[ ] ______

REPLACES DOCUMENT NO.

RECEIVED JUN 1 1971 D. B. SISSON

PIPING OR COOLING SYSTEM INVOLVED

RESPONSIBLE ENGINEER VM DOCHEZ ISSUED BY R. NORTON DATE APR 16 1971

REFERENCES
MASTER PARTS LIST (MPL) NOS. A61-4270, 1-132; A71-4030 (Hatch)
SPECIFICATIONS
DRAWINGS
OTHER

REVISION RECORD
REVISED PER (ECA, ECN, ETC.) ECN NE24900
SHEETS AFFECTED GENERAL REVISION REVISION IDENTIFIED WITH

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AA02		109	1	Spain Pouch		
125	1	135	1	Hughes DE		1
135	1	354	1	Japan Pouch		
354	1	361	2		126	6
361	2	527QCEI	1		377	1
371	1	595LIB	1	Rudock	752	1
510	1	650E	1	MacMillan	Wilm	1
595LIB	1	663	1	(PL only)		
632	1	711	1			
650	1	752	1			
650E	1	753	1			
663	1	760	1			
722D	1	761				
724	1	594	1			
753	1	583	1			
7[illegible]	1	630	5	Mo[illegible]		
7[illegible]						

Cont'd on Page 2

GENERAL ELECTRIC

ATOMIC POWER EQUIPMENT DEPARTMENT

DESIGN SPECIFICATION

SPEC. NO. 22A2928 REV. NO. 2
SH NO. 16 CONT ON SHEET F

RADIATION ENVIRONMENTAL CONDITIONS

Section B. (Con.) IV. Rad-Waste Building

Equipment or Area	Radiation Type	1) Operating Dose Rate Plant Oper	System Oper	Accident Dose Type	Dose Rate	2) Integrated Dose Normal	Accident
Control Room	Gamma		0.001			3.5×10^2	
Valve & Pump Rooms	Gamma		0.020			7.0×10^3	
Storage Tanks (Unprocessed)	Gamma		20.0			7.0×10^6	
Centrifuge	Gamma		100.0			1×10^7	

V. REACTOR CONTROL ROOM

Equipment or Area	Radiation Type	1) Operating Dose Rate Plant Oper	System Oper	Accident Dose Type	Dose Rate	2) Integrated Dose Normal	Accident
Control Room	Gamma	0.0005				1.75×10^2	3×10^0

1) Gamma Dose Rate — Rads (Carbon)/hour
Neutron flux — Neutrons/cm^2-sec
2) Gamma Dose — Rads (Carbon)
Neutron fluence — Neutrons/cm^2 (NVT)
Normal Conditions — Integrated over 40 years - 100 percent load factor at rated power.
Accident Conditions — Integrated over 6 months LOCA Analysis was based on the assumption that 100 percent of the noble gases, 50 percent of the halogens, and 1 percent of the solid fission products were released from the core.

ISSUED: APR 16 1971