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RECIP.NAME RECIPIENT AFFILIATION
DENTON,H.R. Office of Nuclear Reactor Regulation, Director
STOLZ,J.F. Operating Reactors Branch 4

SUBJECT: Forwards final resolution of SER & technical evaluation rept items re environ qualfication of safety-related electrical equipment.Response to IE Bulletin 71-01B will be done on unit by unit basis as environ qualification mods completed.

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October 26, 1984

Mr. Harold R. Denton, Director
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Attention: Mr. John F. Stolz, Chief
Operating Reactors Branch No. 4

Subject: Oconee Nuclear Station
Docket Nos. 50-269, -270, -287

Dear Sir:

The Safety Evaluation Report (SER) along with the Technical Evaluation Report (TER) for the Environmental Qualification of Safety-related Electrical Equipment at Oconee Nuclear Station was transmitted by a NRC letter dated April 11, 1983. Within this letter Duke was requested to submit to the NRC additional information/clarafications.

By my letters dated May 19, 1983 and May 20, 1983, Duke provided the requested information/clarafications. As a followup to these correspondences, a meeting with the Staff to discuss and resolve all outstanding equipment environmental qualification issues for Oconee was held on January 31, 1984. During the meeting, Duke agreed to submit additional information and documentation of the January 31, 1984 meeting.

Attached, please find a final report which provides the documentation of the January 31, 1984 meeting and the final resolution of all TER items for the environmental qualification of safety-related electrical equipment at Oconee Nuclear Station.

Finally, Duke will provide to the NRC a revised submittal of our response to IE Bulletin 71-01B to reflect the changes made since the September 18, 1981 submittal. This will be done on a unit by unit basis as environmental qualification mandated modifications are completed.

Very truly yours,

Hal B. Tucker

Hal B. Tucker

PFG:slb

Attachment

A043
1/1

cc: Mr. James P. O'Reilly, Regional Administrator
U. S. Nuclear Regulatory Commission
Region II
101 Marietta Street, NW, Suite 2900
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Mr. J. C. Bryant
NRC Resident Inspector
Oconee Nuclear Station

Ms. Helen Nicolaras
Office of Nuclear Reactor Regulation
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OCONEE NUCLEAR STATION

RESOLUTION OF SAFETY EVALUATION REPORTS

FOR ENVIRONMENTAL QUALIFICATION OF

SAFETY-RELATED ELECTRICAL EQUIPMENT

On April 3, 1983, Duke Power Company received the Technical Evaluation Reports (TER) regarding the Environmental Qualification (EQ) of Safety-Related Electrical Equipment at Oconee Nuclear Station (ONS), Units 1, 2, and 3. The TER's, written by Franklin Research Center under contract to the NRC, noted certain outstanding items concerning environmental qualification of safety-related equipment at ONS. On May 19, 1983, Duke Power Company provided resolutions to the TER's cited items. On January 31, 1984, a meeting was held with members of the NRC Staff to discuss Duke Power Company's resolutions to the TER's items. The final resolutions, as discussed with the staff, for each of the environmental qualification outstanding items listed in the TER's is summarized in Attachment 1. Also, included in Attachment 1 is revised Table 1 (replacement schedule) and Table 2 (TMI items) of the original May 19, 1983 TER response. Discussions also took place at the meeting regarding Duke Power Company's general methodology for compliance with 10CFR50.49(b), which became effective February 22, 1983. The purpose of this document is to provide documentation of the discussions held at the January 31, meeting.

Duke Power Company will provide the NRC with a revised IEB79-01B submittal which will reflect the resolution as provided in Attachment 1. This will be done on a unit by unit basis as EQ mandated modifications are completed.

The NRC requested confirmation that all design-basis events on ONS which could result in a potentially harsh environment, including flooding outside containment, were addressed in identifying safety-related electrical equipment at ONS which was to be environmentally qualified. The environmental effects resulting from postulated design-basis accidents as documented in Chapter 14 of the ONS Final Analysis Report (FSAR), including the Loss-of-Coolant Accident (LOCA) and the Steam-Line Break Accident (SLBA) inside containment were considered in the identification of safety-related electrical equipment which was to be environmentally qualified. Safety-related electrical equipment which could become submerged inside containment following a LOCA was analyzed in response to NRC questions (Ref. Duke letter to Rusche dated October 31, 1975 - Response to Question). The flooding and environmental effects resulting from High Energy Line Breaks (HELB) outside containment, as documented in MDS Report No. OS-73.2 (dated April 25, 1973) and Supplement (dated June 22, 1973), were also considered in the identification of this equipment. Therefore, all design-basis events including accidents at ONS were considered in the identification of safety-related electrical equipment to be environmentally qualified which is in compliance with 10CFR50.49(b)(1).

With regard to 10CFR50.49(b)(2), we have reviewed the appropriate ONS design documents (e.g., elementary diagrams, connection diagrams, P&IDs, cable lists) and have determined that the Oconee design incorporates the following features:

Electrical isolation is provided for all signals leaving the Reactor Protective System (RPS). This isolation is provided by use of isolation amplifiers or by relay contacts. The effect of this isolation is to prevent faults occuring to signal lines outside the RPS cabinets from being reflected into more than one protective channel. The isolation thus provided also assures that two or more protective channels cannot interact through the cross-coupling or faulting of related signal lines. Additionally, faults such as short, open, or grounded circuits of analog output signals from two or more channels have no effect on the protective channels or their functions (Ref. Oconee FSAR Chapter 7).

Electrical isolation for analog signals is provided in the Engineered Safeguards Protective System (ESPS). Also, the use of individual output relays from the ESPS to each controlled device preserves the isolation of each device and of the elements of one protective channel from another. Faults in control wiring between the output relays and its associated control relay in the controller of a protective device will not affect any other device or protective channel action (Ref. Oconee FSAR Chapter 7).

Emergency power system design incorporates many redundant protective features in order to maintain the integrity of the system. These features include phase overcurrent, ground fault, and differential protective relaying. Additionally, each load bus feeder breaker is provided with redundant trip action (Ref. Oconee FSAR Chapter 8).

It should also be noted that in addition to the above design features, Duke Power Company performed an analysis of control systems at Oconee in response to an NRC50.54(f) letter dated September 17, 1979, and IE Information Notice 79-22. The purpose of the analysis was to determine what, if any, design changes or operator actions would be necessary to assure that environments caused by high energy line breaks would not cause a non-safety-related control system to fail in such a manner as to complicate the event beyond the assumptions of the accident analysis. The systems considered in this analysis were identified by B&W and Duke and reviewed by Duke for the interaction described above. The functions(and associated systems) reviewed were reactor power control and shutdown, reactor pressure control, main steam system isolation and pressure control, and feedwater system isolation and pressure control. The results of this analysis were provided to the NRC by letter dated October 5, 1979. Only the steam line break inside containment pressurizer PORV control combination was identified as potentially involving a variation from the safety analysis basis. To preclude this potentially unacceptable interaction, the unqualified component associated with the PORV control systems was relocated to a mild environment.

Based on the design features described above and our previous efforts concerning IE Information Notice 79-22, we conclude that there is reasonable assurance that non-safety-related equipment would not preclude the accomplishment of essential safety functions.

With regard to 10CFR50.49(b)(3), Duke Power Company has provided an integrated plan and schedule for addressing Regulatory Guide 1.97 in Duke Power Company's Response to Supplement 1 of NUREG-0737 for Oconee Nuclear Station. This response was originally submitted by letter from H. B. Tucker to H. R. Denton on April 14, 1983. Upon completion of the review of the requirements of Regulatory Guide 1.97 to the design of Oconee Nuclear Station, equipment identified as requiring environmental qualification if not already listed in the IEB79-01B submittal will be added and qualification documented. Final response to Regulatory Guide 1.97, Duke Power Company's Response to Supplement 1 of NUREG-0737 for ONS Post-Accident Monitoring, is scheduled for submittal in September 1984.

The Staff also requested information concerning equipment qualification mandated maintenance/replacement activity. Equipment qualification mandated maintenance requirements are provided to the station through company procedures. These requirements are identified through review of test reports and manufacturer's documentation. The requirements are integrated into station procedures as supplements to the manufacturer's maintenance recommendations. Duke Power Company procedures also identify specific equipment replacment intervals based on test reports or manufacturer's documentation. The replacement intervals are then incorporated into the station Periodic Maintenance program. Therefore, these company procedures ensure that electrical equipment requiring qualification will be maintained in a qualified state throughout the operating life of the station. The environmental qualification documentation maintained in the ONS Equipment Qualification File complies with the requirements of 10CFR50.49. Environmental qualification documentation is available for NRC audit.

The staff additionally requested Duke Power Company to submit all applicable JCO's that are currently being relied upon. Attachment II addresses those JCO's.

Based on the information summarized in this document and detailed in the Oconee IEB79-01B submittal, Oconee can continue to operate without undue risk to the public health and safety.

ATTACHMENT I

RESOLUTIONS TO SPECIFIC EQUIPMENT EQ OUTSTANDING ITEMS

TER ITEM NO. (UNIT CROSS REF.) UNIT 1	UNIT 2	UNIT 3	DESCRIPTION	NRC CATORGY	DEFICIENCY	RESOLUTION
1	-	-	Rotork VMTR LPSW 565, 566	II.A	Aging, Radiation	Documentation from Rotork verifies that all Rotork NA1 operators are qualified by Rotork Test Report TR-116 (TR-116 was accepte[d] by FRC - See TER Item 11).
2,3,4 5,6	3,4,9,10 16, 71	4,8,11,13 35,61	Limitorque VMTR PR-1,6,7,9 GWD-12 CC-7 CS-5 RC-5,6 FDW-103,104,105, 107 LPSW-6,15 LP-1,2 LWD-1	II.C	Aging	Additional documentation from Limitorque (Report B0058) establishes a 40 year qualified life for the Oconee specific conditions.
7	11	12	Limitorque VMTR LP-21,22 HP-24,25	III.B	Equipment not in scope of review	FRC classified this equipment in Category III.B. However, Duke believes these valves should be in the scope of the IEB79-01B review. As such, this equipment should fall into Category I.B, "Equipment Qualification Pending Modification". These VMTR's are to be replaced with qualified Limitorque VMTR's. See Table 1 for replacement schedule and qualification reference.
8	5,7	6	Limitorque VMTR LP-17,18	I.B	Radiation, Aging	These VMTR's are to be replaced with qualified Limitorque VMTR's (See Table 1 for replacement schedule and qualification reference).
9,10	8	7	Limitorque VMTR LP-3 LPSW-16,19,22	II.C	Aging	Additional documentation from Limitorque (Report B0058) establishes a 40 year qualified life for the Oconee specific conditions.

ATTACHMENT I

RESOLUTIONS TO SPECIFIC EQUIPMENT EQ OUTSTANDING ITEMS

TER ITEM NO. (UNIT CROSS REF.) UNIT 1	UNIT 2	UNIT 3	DESCRIPTION	NRC CATORGY	DEFICIENCY	RESOLUTION
11	-	-	Rotork VMTR LP-105	I.A	N/A	Valve motor operators are qualified (Rotork Report TR-116)
12,13, 14	13	9,14	Limitorque VMTR PR-15,19 LP-5,8,12,14,15, 16,19,20 LPSW-4,5	II.C	Aging	Additional documentation from Limitorque (Report B0058)establishes a 40 year qualified life for the Oconee specific conditions
15,16	20,22	15,16	Target Rock SLND RC-162,163,164 165	II.A	Qualification review in progress (TMI)	Qualification is established under Target Rock Test Report 2375. However, because of specific system design requirement these valves must be located below the post-LOCA water level. Therefore, RC-162, 163 are being upgraded to assure submerged operation capability (JCO previously provided - See Table 2). NOTE: RC-164, 165 on Unit 1 are not electrically operated valves (manual operation only) and therefore, they do not fall within the scope of 10CFR50.49.
17,18,19 70,71,73	17,18,19, 66,67,69	18,19,20, 64, 66,20	SLND-Various Manufacturer SV-3,4,5,16,31,32, 33,34,37,38,75, 76,90,95	I.B	Documentation	SLND's have been replaced with qualified ASCO SLND. See Table 1 for replacement schedule and qualification reference.
20	24	21	Skinner SLND LP-121 LWD-1026	II.A	Qualification review in progress (TMI)	LP-121 will be replaced with a qualified ASCO SLND (See Table 2). LWD-1026 - See Note A

ATTACHMENT I

RESOLUTIONS TO SPECIFIC EQUIPMENT EQ OUTSTANDING ITEMS

TER ITEM NO. (UNIT CROSS REF.) UNIT 1	UNIT 2	UNIT 3	DESCRIPTION	NRC CATORGY	DEFICIENCY	RESOLUTION
21	21	17	Target Rock SLND RC-155 thru 160	II.A	Qualification review in progress (TMI)	Qualification is established under Target Rock Test Report 2375. (See Table 2).
22	25	23	Valcor SLND SV-210 thru 219	II.A	Qualification review in progress (TMI)	Qualification is established under Valcor Test Report QR70900-21-1 (See Table 2)
23	27	24	DeLaval LTRM LT-90,91	II.A	Qualification review in progress (TMI)	Qualification is established under DeLaval Report 45700-1. (See Table 2).
24	29	26	Rosemount PTRM Model 1153 PT-230,231	II.A	Qualification review in progress (TMI)	Qualification is established under Rosemount Test Report 3788 (See Table 2).
25	28	25	Rosemount PTRM Model 1152GP PT-17,18,19,20	II.C	Aging	Qualified life is established under B&W Report 58-0261-00&58-0220-00 and supplemental calculations.
26	30	27	Mercoid PRSW Model APW-7041 PS-65,66,67,68	II.A	Documentation	See Note B
27	31	28	Rosemount RTD's 177GY RD84A,B; 85A,B	II.A	Similarity, Aging, Time, Profile, Envelope, Chemical Spray	Qualification is established under B&W Qualification Reports for 177GY&177HW RTD's and supplemental documentation.
28	37	29	Louis Allis FMTR Pent. Room Cooling Fan Motors 1A,1B	II.A	Similarity, Aging, Radiation	Additional documentation from Louis Allis establishes similarity, aging, and radiation qualification.

ATTACHMENT I

RESOLUTIONS TO SPECIFIC EQUIPMENT EQ OUTSTANDING ITEMS

TER ITEM NO. (UNIT CROSS REF.) UNIT 1	UNIT 2	UNIT 3	DESCRIPTION	NRC CATORGY	DEFICIENCY	RESOLUTION
29	38	54	Joy/Reliance FMTR RBCU 1A, 1B, 1C	II.A	Similarity, Chemical Spray	Joy Manufacturing Test Report X-604 and supplemental documentation from Joy establishes similarity and chemical spray qualification.
30,31, 32,33	35,36,33, 34	55,30,31	Westinghouse PMTR RB Spray 1A,1B HPI 1A,1B,1C LPI 1A,1B,1C	II.A	Similarity	Additional documentation from Westinghouse establishes similarity between installed motors and motors qualified under WCAP's 7829 and 8754.
34	43	34	Viking,Penetrations	I.A	N/A	Penetrations are qualified.
35	52	37	Okonite Cable	II.A	Similarity	Additional documentation from Okonite establishes similarity between installed cable and cable tested in the referenced test report.
36	2	1	Rotork VMTR NA1 PR-59,60	II.A	Qualification Review in progress	Qualification is established under Rotork Test Report TR-116.
37	6	5	Limitorque VMTR LPSW-18,21,24 BS-1,2	II.C	Aging	Additional documentation from Limitorque (Report B0058) establishes a 40 year qualified life for the Oconee specific conditions.
38	54	39	Okonite Cable	II.C	Aging	Okonite Test Reports 110E and 141 and supplemental documentation establish the qualified life for this cable.
39	39	33	States Terminal Blocks	I.A	N/A	Terminal Blocks are qualified.
40	53	38	Okonite Cable	II.A	Similarity	Additional documentation from Okonite establishes similarity between installed cable and cable tested in the referenced test report.

ATTACHMENT I

RESOLUTIONS TO SPECIFIC EQUIPMENT EQ OUTSTANDING ITEMS

TER ITEM NO. (UNIT CROSS REF.) UNIT 1	UNIT 2	UNIT 3	DESCRIPTION	NRC CATORGY	DEFICIENCY	RESOLUTION
41	44	-	Brand-Rex Cable	IV	Documentation not made available	Qualification is established under Duke Power Company Test Report TR-032 (OM-360-19).
42	45	-	Raychem Cable	II.C	Aging	Qualification is established under Duke Power Company Test Report TR-032 (OM-360-19) and supplemental calculation.
43	46	45,46	Raychem Cable	II.A	Aging	Qualification is established under Duke Power Company Test Report TR-032 (OM-360-19) and supplemental calculation.
44	47	40	B.I.W. Cable	I.B	Documentation	Qualification is established under B.I.W. Test Report 82E047 (OM-360-40).
45,46	51,42	47	Anaconda Cable	II.A	Similarity	Additional documentation from Anaconda establishes similarity between installed cable and cable tested in the referenced test report.
47	48	41	B.I.W. Cable	I.B	Documentation	Qualification is established under B.I.W. Test Report 82E047(OM-360-40).
48,49	49,50	43,44	Samual Moore Cable (Eaton)	II.A	Similarity	Additional documentation from Samual Moore (Eaton) established similarity between installed cable and cable tested in the referenced test report.
50	-	-	B.I.W.-hook-up wire	II.C	Aging	Additional documentation establishes qualified life of this cable.
51	41	48	Anaconda Cable	II.C	Aging	Additional documentation establishes qualified life of this cable.

ATTACHMENT I

RESOLUTIONS TO SPECIFIC EQUIPMENT EQ OUTSTANDING ITEMS

TER ITEM NO. (UNIT CROSS REF.) UNIT 1	UNIT 2	UNIT 3	DESCRIPTION	NRC CATORGY	DEFICIENCY	RESOLUTION
52,55	60	58	Motorola PTRM Model 56PM PT-4,5	I.B	Documentation	PTRM's have been replaced with qualified Barton Model 764 transmitters. See Table 1 for replacement schedule and qualification reference.
53	59	57	Motorola PTRM Model 56PH PT-21P,22P,23P	II.A	Documentation, Aging Chemical Spray, Radiation	B&W Report 58-0093-00 and supplemental documentation establishes qualification of those transmitters.
54	58	59	Motorola PTRM Model 56PM PT-7	I.B	Documentation	See Note C
56,57, 58	64,63,64	3,63,62	Limitorque VMTR HP-3,4,20 HP-409,410 HP26,27	II.C	Aging	Additional documentation from Limitorque (Report B0058) establishes a 40 year qualified life for the Oconee specific conditions.
59	14	10	Limitorque VMTR LWD-1028 DW-278	II.A	Qualification Review in Progress	See Note D
60	15	36	Limitorque VMTR BS-3,4	II.C	Aging	Additional documentation from Limitorque (Report B0058) establishes a 40 year qualified life for the Oconee specific conditions.
61	1	2	Rotork VMTR Model NA1 LP-103,104	II.A	Aging, Radiation	Documentation from Rotork verifies that all Rotork NA1 operators are qualified by Rotork Test Report TR-116 (TR-116 was accepted by FRC-See TER Item 11)
62	61	56	Rosemount LTRM Model 1152DP LT-80,81,82,83	I.B	Aging, Submergence	B&W Reports 58-0261-00&58-0220-00 and supplemental calculation establishes a qualified life for these transmitters. Submergence is addressed in Note E.

ATTACHMENT I

RESOLUTIONS TO SPECIFIC EQUIPMENT EQ OUTSTANDING ITEMS

TER ITEM NO. (UNIT CROSS REF.) UNIT 1	UNIT 2	UNIT 3	DESCRIPTION	NRC CATORGY	DEFICIENCY	RESOLUTION
63	62	60	Mercoid PRSW Model APW7041153 PS-18 thru 23	III.B	Equipment not in scope of review	See Note F
64	55	50	Motorola PTRM Model 56 PT-24,25,26,27	I.B	Documentation	See Note G
65	57	49	Meletron PRSW PS-60,61	II.A	Documentation	See Note H
66	56	51	Rosemount PTRM Model 1152GP PT-166P	II.C	Aging	See Note I
67	32	32	General Semiconductor Transzorb	II.A	Radiation, Temperature, Qualification time, Pressure, Humidity, Aging	Additional documentation from General Semiconductor (Temperature Test) establishes qualification.
68	26	52	Bailey Meter LTRM LT4P3,4P2,4P1	II.A	Aging	See Note J
69	23	22	Lawrence SLND DW-280	I.B	Documentation (TMI)	See Note K
72	68,70	65	Super Splice SLND SV-36	I.B	Documentation	SLND has been replaced with qualified ASCO SLND See Table 1 for replacement schedule and qualification reference.

OCONEE NUCLEAR STATION

NOTES TO RESOLUTIONS TO SPECIFIC

EQUIPMENT EQ OUTSTANDING ITEMS

NOTE A: Skinner Solenoid Valve (Unit 1 TER item 20; Unit 2 TER item 24; Unit 3 TER item 21)

This valve is located in the Auxiliary Building and exposed to the post-LOCA recirculation radiation environment. However, it has been determined that this valve is non-safety-related and not required to function in a harsh environment. Therefore, the valve is not within the scope of 10CFR50.49.

NOTE B: Mercoid Pressure Switches (Unit 1 TER item 26; Unit 2 TER item 30; Unit 3 TER item 27)

These pressure switches provide a trip signal to the Reactor Protection System on high Reactor Building (RB) pressure. The safety function of these switches is accomplished within approximately 2 minutes of the event which caused the RB high pressure.

These switches are located in the penetration room and therefore, will not experience the accident environment inside the RB. The only harsh environment which these switches experience is radiation due to recirculation piping located in the penetration room. As recirculation begins approximately 30 minutes following a LOCA, the switches will have performed their safety function well before experiencing the harsh radiation environment. Additionally, failure of the switches after performing their safety function is of no consequence as the trip signal is locked in by the Reactor Protection System. It should also be noted, that no indication is provided by these switches.

NOTE C: Motorola 56PM Transmitter (Unit 1 TER item 54; Unit 2 TER item 58; Unit 3 TER item 59)

This transmitter has been determined not to be required to perform a safety function in a harsh environment; therefore, the transmitter is not within the scope of 10CFR50.49.

NOTE D: Limitorque Motor Operated Valves (Unit 1 TER item 59; Unit 2 TER item 14; Unit 3 TER item 10)

These valves are non-safety-related and not required to perform a safety function in a harsh environment; therefore, these valves are not within the scope of 10CFR50.49.

NOTE E: Rosemount 1152DP Transmitters (Unit 1 TER item 62; Unit 2 TER item 61; Unit 3 TER item 56)

These transmitters are located inside the Reactor Building and function to provide level control for the operation of the Emergency Feedwater System. As noted in Duke's response to the NRC's previous equipment qualification SER, these transmitters were to be relocated above the maximum post-LOCA water level. Due to the sloping requirements of the instrument lines to the transmitters in conjunction with available shield wall instrument line penetration locations, it has been determined that these transmitters cannot be relocated above the maximum post-LOCA water level. Therefore, Duke has further evaluated the application of these transmitters and the potential effects of submergence and have determined that these transmitters do not require relocation. Our determination is based on the following:

a) The function of these transmitters is required following certain high energy line breaks (e.g., main steamline, main feedwater line break, small-break LOCA) excluding the design basis LOCA. The present location of these transmitters would accomodate the water level expected for a small-break LOCA or other HELB event, assuming a reasonable cooldown of the Reactor Coolant System to establish long-term cooling via the Low Pressure Injection system.

b) In the unlikely event that these transmitters become submerged following a HELB for which they are required to function, favorable experience from TMI-2 demonstrates that Rosemount 1152 transmitters are capable of operating submerged for approximately 2 months which significantly exceeds the required operating time for Oconee. Additionally, preliminary submergence testing by Duke on a Rosemount 1152 have proved very favorable.

c) As a further precaution concerning submergence, the cable entrances into these transmitters have been sealed with Scotchcast 9 and Oconee maintenance procedures require replacing the appropriate O-ring seal any time the transmitter cover, flange, or seal between the sensor or electronics is removed.

NOTE F: Mercoid Pressure Switches (Unit 1 TER item 64: Unit 2 TER item 55; Unit 3 TER item 50)

These pressure switches initiate Engineered Safeguards on high Reactor Building pressure. These switches were classified by FRC as Category III.B because they perform their safety function prior to exposure to a harsh environment. Duke Power Company believes that these switches should be placed in Category II.A as are the Reactor Protection System trip switches (TER item 26, Unit 1).

These switches are located in the penetration room and therefore, will not experience the accident environment inside the RB. The only harsh environment which these switches experience is radiation due to recirculation piping located in the penetration room. The safety function of these switches is accomplished within approximately 2 minutes of the event which caused the RB high pressure. Since recirculation does not begin until approximately 30 minutes following a LOCA, the switches will have performed their safety function well before experiencing the harsh radiation environment. Additionally, failure of these switches after performing their safety function is of no consequence as the initiating signal is locked in by the Engineered Safeguard. It should also be noted that no indication is provided by these switches.

NOTE G: Motorola 56 PM Transmitters (Unit 1 TER Item 64; Unit 2 TER Item 55; Unit 3 TER Item 50)

These transmitters are located inside the Reactor Building and function to provide steam generator pressure indication. However, because alternate steam generator pressure indication is available through instruments located in mild environment, these Motorola 56PM transmitters are not required to perform a safety function in a harsh environment. Therefore, these transmitters are not within the scope of 10CFR50.49.

NOTE H: Meletron Pressure Switches (Unit 1 TER Item 65; Unit 2 TER Item 57; Unit 3 TER Item 49)

These pressure switches are non-safety-related and are provided for equipment protection purposes for non-safety-related radiation monitors. Therefore, these switches are not within the scope of 10CFR50.49.

NOTE I: Rosemount 1152 GP Transmitter (Unit 1 TER Item 66; Unit 2 TER Item 56; Unit 3 TER Item 51)

This transmitter is non-safety-related and is provided as an input to the Integrated Control System for controlling pressurizer heaters. Pressurizer heater control is not required for Design Basis Accident conditions that create a harsh environment; therefore, this transmitter is not required to function. Additionally, this transmitter does not provide control board indication for operation action. Therefore, this transmitter is not within the scope of 10CFR50.49

NOTE J: Bailey BY Transmitters (Unit 1 TER Item 68; Unit 2 TER Item 26; Unit 3 TER Item 52)

These transmitters are located in the Reactor Building and provide indication of pressurizer level. However, for the design basis of the Oconee Nuclear Station, the pressurizer level transmitters are not required to function for accident mitigation nor are they used as a primary means of monitoring safety system performance. Therefore, these transmitters are not within the scope of 10CFR50.49.

NOTE K: Lawrence Solenoid Valve (Unit 1 TER Item 69)

This valve is located in the Auxiliary Building and exposed to the post-LOCA recirculation radiation environment. However, this valve is non-safety-related and not required to function in the harsh environment; therefore, this valve is not within the scope of 10CFR50.49.

TABLE 1

RESOLUTION OF NRC/FRC TECHNICAL EVALUATION REPORT ITEMS

TER CATEGORY I.B ITEMS - REPLACEMENT SCHEDULE*

TER			PLANT ID	SCHEDULED OUTAGE DATE FOR REPLACEMENT		
UNIT 1	UNIT 2	UNIT 3		UNIT 1	UNIT 2	UNIT 3
8	5, 7	6	LP-17, 18	10/84	complete	complete
17	17, 18, 69	18, 19, 20, 64	SV-75, 76, 31, 4, 34, 3, 16	complete	complete	complete
18	-	-	SV-33	complete	complete	complete
19	19	20	SV-32	complete	complete	complete
52, , 55	60	58	PT-4, 5, 6	complete	complete	complete
70	66	77	SV-90, 95	complete	complete	complete
71	67	20	SV-5	complete	complete	complete
72, 73	68, 70	65	SV-36, 37, 38	complete	complete	complete

* Justifications for Continued Operation (JCO's) have previously been submitted for these items. Refer to Duke Power Company letter, W.O. Parker to H.R. Denton dated February 25, 1982.

1. LP-17, 18: Limitorque Valve Motor Operators; Limitorque Report 600456
2. All SV Designations: ASCO Solenoid Valve Operators; ASCO Report AQS21678/TR
3. PT-4, 5, 6: Barton Pressure Transmitters (764); Barton Report R3-764-9

Revised 7/19/84

TABLE 2

OCONEE NUCLEAR STATION - UNITS 1, 2, AND 3

RESOLUTION OF NRC/FRC TECHNICAL EVALUATION REPORT ITEMS

CATEGORY II.A - TMI ITEMS

TER ITEM NO.			PLANT ID	QUALIFICATION REFERENCE/ RESOLUTION
UNIT 1	UNIT 2	UNIT 3		
15	20	16	RC-162, 163	Target Rock Corp. Report No. 2375 (OM-360-32) Additionally, refer to the text concerning potential submergence of these valves and the attached JCO.
16	22	15	RC-164, 165	Unit 1 valves are manual. Unit 2 and 3 - Target Rock Corp. Report No. 2375
20	24	21	LP-121	This valve is located in a liquid sample path and is subject to the post-LOCA recirculation radiation environment. - ASCO Report AQS21678/TR
21	21	17	RC-155 thru 160	Target Rock Corporation Report No. 2375 (OM-360-32)
22	25	23	SV-210 thru 219	Valcor Test Report QR70900-21-1 (OM-360-33)
23	27 (Listed under I.B)	24	LT-90, 91	DeLaval Report No. 45700-1 (OM-360-38)
24	29	26	PT-230-231	Rosemount Test Report 3788 (OM-360-37)
36	2	1	PR-59, 60	Rotork Test Report TR116

ATTACHMENT II

JUSTIFICATION FOR CONTINUED OPERATION

1. Reference: W.O. Parker, Jr. letter to H. R. Denton dated February 25, 1982, Item 5.

   Equipment: Limitorque Valve Motors

Valve No.	System	Position	Location	Service
HP-24	HPI	N.C.	Aux. Bld.	Borated Water Storage Tank Supply to HPI Pump 1A
HP-25	HPI	N.C.	Aux. Bld.	Borated Water Storage Tank Supply to HPI Pump 1C
LP-17	LPI	N.C.	Pent. Rm.	LPI Lead A Shutoff Valve-RB Isolation
LP-18	LPI	N.C.	Pent. Rm.	LPI Lead B Shutoff Valve-RB Isolation
LP-21	LPI	N.O.	Aux. Bld.	Borated Water Storage Tank Train A Outlet Valve
LP-22	LPI	N.O.	Aux. Bld.	Borated Water Storage Tank Train B Outlet Valve
PR-1	RBV	N.C.	R. B.	RB Purge Exhaust Valve-RB Isolation
PR-6	RBV	N.C.	R. B.	RB Purge Inlet Valve-RB Isolation

   Location: See above

   Problem: No brake qualification data

Justification:

1. PR-1 and PR-6 are located inside the Reactor Building and are normally closed except for limited purging during cold shutdown. Additionally, power to these valves is removed during normal station operation. Therefore, since these valves are normally in their safety position (closed) with power removed and they do not perform an accident mitigating function, they are considered qualified for their application and replacement is not required. It should also be noted, that Reactor Building isolation is further assured by redundant/diverse outside containment isolation values.

2. HP-24, HP-25, LP-21, and LP-22 have been replaced with qualified Limitorque valve motors on Units 1, 2, and 3.

3. LP-17 and LP-18 have been replaced on Units 2 and 3 with qualified Limitorque valve motors. LP-17 and LP-18 are scheduled to be replaced on Unit 1 during the next refueling outage (October, 1984). LP-17 and LP-18 are located outside the Reactor Building and perform their safety function (Reactor Building isolation) prior to being exposed to a harsh environment (Recirculation radiation only).

2. Equipment: Target Rock Solenoid Valves

Valve No.	System	Position	Location	Service
RC-162	RBSM	N.C.	R.B.	Liquid Sample Valve (TMI)
RC-163	RBSM	N.C.	R.B	Liquid Sample Valve (TMI)

Problem: Target Rock solenoid valve have been environmentally qualified but have not been qualified to sustain full submergence. These valve have been located near the RB floor as to allow sampling from a depressurized RCS by gravity feed. By being close to the floor, they may be submerged post-accident and they have not been demonstrated to be able to function in this condition.

Justification:

1. The Unit 3 valves have been modified in accordance with Target Rock modification procedures for submergence operation.

2. Unit 1 and 2 valves will be modified during the next outage of sufficient duration to complete the modification. Unit 1 is scheduled for October 1984 and Unit 2 is scheduled for February 1985. Additionally, an alternate means of sampling is available from the RB sump. This alternate means of sampling is outlined in "Discussion of Compliance to the NRC Confirmatory Order Pertaining to NUREG-0737," dated August 15, 1983 which was transmitted to the NRC under a cover letter from H. A. Tucker to H. R. Denton, dated August 15, 1983.