TENNESSEE VALLEY AUTHORITY

CHATTANOOGA, TENNESSEE 37401

1630 Chestnut Street Tower II

December 10, 1984

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

Dear Mr. Denton:

In the Matter of the	)	Docket Nos. 50-259
Tennessee Valley Authority	)	50-260
		50-296

In accordance with 10 CFR 50.49(h) we are submitting as an enclosure information regarding the status of environmental qualification of the Browns Ferry Nuclear Plant. The enclosure provides a list of equipment components found to be environmentally unqualified because of a lack of conduit seals. The enclosure also provides justification for continued operation with the unqualified components.

Each of the devices will have the proper conduit seal installed as part of the continuing environmental qualification program. Furthermore, any equipment which is to be replaced with a qualified device requiring conduit seals will have the seals installed as part of the modification.

If you need additional information, please get in touch with us through the Browns Ferry Project Manager.

Very truly yours,

TENNESSEE VALLEY AUTHORITY

J.W. Hufham

J. W. Hufham, Manager
Licensing and Regulations

Subscribed and sworn to before
me this 10th day of Dec. 1984.

Paulette H. White

Notary Public
My Commission Expires 8-24-88

Enclosure
cc: See page 2

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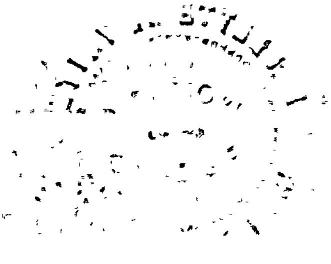

Mr. Harold R. Denton December 10, 1984

cc (Enclosure):
U.S. Nuclear Regulatory Commission
Region II
ATTN: James P. O'Reilly, Regional Administrator
101 Marietta Street, NW, Suite 2900
Atlanta, Georgia 30323

Mr. R. J. Clark
Browns Ferry Project Manager
U.S. Nuclear Regulatory Commission
7920 Norfolk Avenue
Bethesda, Maryland 20814

ENCLOSURE

NOTIFICATION OF ENVIRONMENTAL QUALIFICATION
PURSUANT TO 10 CFR 50.49(h)
BROWNS FERRY NUCLEAR PLANT

<u>ATTACHMENT 1</u> - List of Devices

<u>ATTACHMENT 2</u> - Justification for Continued Operation

ATTACHMENT 1

Devices which are environmentally unqualified due only to the lack of conduit seals:

1-ZS-1-14
1-ZS-1-15
1-ZS-1-26
1-ZS-1-27
1-ZS-1-37
1-ZS-1-38
1-ZS-1-51
1-ZS-1-52
3-ZS-1-14
3-ZS-1-15
3-ZS-1-26
3-ZS-1-27
3-ZS-1-37
3-ZS-1-38
3-ZS-1-51
3-ZS-1-52
1-FSV-43-14
2-FSV-43-14
3-FSV-43-14
1-FSV-64-20
2-FSV-64-20
3-FSV-64-20
1-FSV-64-21
2-FSV-64-21
3-FSV-64-21
1-FSV-64-29
3-FSV-64-29
1-FSV-64-32
3-FSV-64-32
1-FSV-64-141
3-FSV-64-141
1-LT-64-159 A&B
3-LT-64-159 A&B
1-LT-64-160 A&B
3-LT-64-160 A&B
1-TE-64-161 A-H
3-TE-64-161 A-H
1-TE-64-162 A-H
3-TE-64-162 A-H
1-FCV-69-1
1-FS-73-33
2-FS-73-33
3-FS-73-33
3-PS-73-20 A-D
3-PS-73-22 A&B
3-LS-73-56 A&B
3-LS-57-A&B
1-FSV-75-57
3-FSV-75-57
1-FSV-75-58
3-FSV-75-58
1-FSV-76-17
2-FSV-76-17
3-FSV-76-17
1-FSV-76-18
2-FSV-76-18
3-FSV-76-18
1-FSV-76-19
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1-FSV-76-67
3-FSV-76-67
1-FSV-76-68
3-FSV-76-68
1-FM-84-19B
3-FM-84-19B
1-FM-84-20B
3-FM-84-20B
1-FSV-84-8 A-D
3-FSV-84-8 A-D
1-FSV-84-19
3-FSV-84-19
3-FSV-84-20
3-FT-84-19
1-FSV-85-37 A&B
2-FSV-85-37 A&B
3-FSV-85-37 A&B
1-FSV-85-39 A&B
2-FSV-85-39 A&B
3-FSV-85-39 A&B
1-LS-85-45 C-F
3-LS-85-45 C-F
1-RE-90-136
2-RE-90-136
3-RE-90-136
1-RE-90-137
2-RE-90-137
3-RE-90-137
1-RE-90-138
2-RE-90-138
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3-RE-90-139

ATTACHMENT 2

JUSTICIFICATION FOR CONTINUED OPERATION

ADDITIONAL EQUIPMENT NO. EEB-8

TVA ID NO.

1,3-TE-64-161A through -161H, -162A through -162H

MANUFACTURER/MODEL NO.

Weed/Model No. 1B1-25D/612D-1A-C-6-C-17.25-0-0

STATUS IV

Justification for Continued Operation

1. Temperature elements 1,3-TE-64-161A through -161H, -162A through -162H are located in the Pressure Suppression Chamber (Torus Rooms), Room 6, Elevation 519 of the Reactor Building. They are required to operate for 100 days following a LOCA, HELB inside containment, HELB outside containment, or control rod drop accident.

2. Weed Instrument Company, Incorporated, Qualification Test Report No. 548-8854-2 Rev. 1 dated March 1, 1982, denotes that the conduit leading to the assembly heads was integrity-sealed during LOCA simulation tests to prevent moisture intrusion. Conduit seals have not been provided for installed unit 1, 3 installed temperature elements.

3. Temperature elements 1,3-TE-64-161A through -161H, -162A through -162H provide input to the new torus water temperature monitoring system. If the elements do not have their terminals sealed, moisture can cause erratic readings or a loss of indication. The operator uses this monitoring system to decide upon initiation of torus cooling and when to depressurize using the SRV's. The technical specification contain required actions which require torus temperature monitoring. Thus failure of the temperature elements can deprive an operator of important information. This can be counteracted by the operator immediately taking the actions for which he used the torus water temperature indication as decision information. Upon receiving erratic information or loss of indication following a HELB outside containment, the operator could immediately initiate torus cooling and rapidly depressurize the reactor pressure vessel and maintain it at a low pressure. By doing this he counteracts the loss of torus water temperature indication and precludes torus temperature from reaching levels which would cause concern.

4. The above information shows justification for continued use of the temperature elements; however, to maintain environmental qualification conduit seals will be installed during the next scheduled outage as a result of NCR BFNEEB8407.

054296.04

JCO NO. EEB-5

TVA ID NO. 1, 3-FSV-76-53, -54, -64, 3-FSV-76-59, -61, and *3-FSV-76-56

MANUFACTURER/MODEL NO. Valcor/Model No. V526-529-2
*Target Rock/Model No. 77DD-001

STATUS IV

Justification for Continued Operation

1. Solenoid valves 1, 3-FSV-76-53, -54, -64, and 3-FSV-76-56 are locted in the Reactor Building, room 6. Solenoid valves 3-FSV-76-59 and 3-FSV-76-61 are located in the Reactor Building, room 8. The above valves are required to operate for 100 days following a HELB inside containment or a LOCA. These accidents will not cause condensate accumulation in the above solenoid valves at their respective locations.

2. Valcor has indicated that the conduit entry on generically qualified model V526 solenoid valves must be sealed in order to prevent moisture intrusion and possible loss of function during an accident. The environmental qualification testing performed on the model 77CC-001 valve by Target Rock utilized a method of preventing moisture entry into the conduit. The model 77DD-001 valve was qualified by similarity to the model 77CC-001 valve. Without proper conduit sealing, moisture intrusion into the valve could cause a loss of function during an accident.

3. The above information shows that these solenoid valves are not required to operate in an accident environment which will cause condensate to form in the conduit or enter the valve housing. However, to maintain environmental qualification, conduit seals will be installed during the next scheduled outage as a result of NCR BFNEEB8407.

TER ITEM NO. 100

TVA ID NO. 1, 3-FSV-76-50, and -52

MANUFACTURER/MODEL NO. Valcor/Model No. V526-529-2

STATUS IV

Justification for Continued Operation

1. Solenoid valves 1, 3-FSV-76-50 and 1, 3-FSV-76-52 are located in the Reactor Building, room 8. They are required to operate for 100 days following a HELB inside containment or a LOCA. These accidents will not cause condensate accumulation in the above solenoid valves at their location.

2. Valcor has indicated that the conduit entry on generically qualified model V526 solenoid valves must be sealed in order to prevent moisture intrusion and possible loss of function during an accident.

3. The above information shows that these solenoid valves are not required to operate in an accident environment which will cause condensate to form in the conduit or enter the valve housing. However, to maintain environmental qualification, conduit seals will be installed during the next scheduled outage as a result of NCR BFNEEB8407.

TER ITEM NOS. 58, 61

TVA ITEM NO. 3-PS-73-20A, -20B, -20C, -20D, -22A, -22B

MANUFACTURER/MODEL NO. ASCO/Model SB21AMR/TE20A32R

STATUS IV

Justification for Continued Operation

1. Pressure switches 3-PS-73-20A, -20B, -20C, -20D, -22A, -22B are located in the southwest pump room, room 2, elevation 519 of the Reactor Building. They are required to operate for 1 day following a LOCA, HELB inside containment (except a HPCI, High Pressure Coolant Injection line break) or HELB outside containment (except a HPCI line break). Only a HELB outside containment could result in moisture intrusion of pressure switch.

2. Automatic Switch Company's (ASCO) Qualification Test Report No. AQR-101083/revision 0 dated October 3, 1983, denotes use of liquatite conduit to connect the electrical chamber of the pressure switch to outside the test chamber during environmental DBE simulation tests. Conduit seals have not been provided for installed unit 3 pressure switches.

3. Pressure switches provide a trip signal to the HPCI system due to "High Rupture Disk Pressure" (PS-73-20A, -20B, -20C, -20D) or "High Exhaust Pressure" (PS-73-22A, -22B). Their locations are such that a HPCI HELB outside containment affects them short term and that some non-HPCI HELBs outside containment affect them long term (· 1 hour). Switches are not required to be qualified for a HPCI HELB outside containment. Their failure mode is to generate a trip signal. Thus the HPCI system could be lost after 1 hour due to a continuous trip signal being generated from switch failure.

   If the HPCI is being used to provide Reactor Pressure Vessel (RPV) inventory makeup following a non-HPCI HELB outside primary containment and it fails, the operator can rapidly depressurize the RPV and use either Residual Heat Removal (RHR) or Core Spray (CS) for low pressure makeup. RPV pressure could be maintained through remote manual operation of safety relief valves. RHR torus cooling may be required depending upon torus water temperature. This shutdown method is an accepted safety grade mode but it is undesirable in that the 100°F cooldown rate on the RPV would be exceeded. This is an operational concern with RPV fatigue usage rather than a nuclear safety concern. Hence, operation without sealed switches as acceptable.

4. The above information shows justification for continuous use of the pressure switches; however, to maintain environmental qualification, conduit seals will be installed during the next scheduled outage as a result of NCR BFNEEB8407.

TER ITEM NO. 012

TVA ID NO. 1, 3-FSV-76-49, -51, -55, -57, -58, -60, -62, -63, -65, -66, -67, -68, 1-FSV-76-56, -59, and -61

MANUFACTURER/MODEL NO. Valcor/Model No. V526-529-2

STATUS IV

Justification for Continued Operation

1. Solenoid valves 1, 3-FSV-76-49 and 1, 3-FSV-76-51 are located in the Reactor Building, room 8. Solenoid valves 1, 3-FSV-76-55, -57, -58, -63, -65, -66, -67, -68, and 1-FSV-76-56 are located in the Reactor Building, room 6. Solenoid valves 1-FSV-76-59, -61, and 1, 3-FSV-76-60, -62 are located in the Reactor Building, room 8.

   The above valves are required to operate for 100 days following a HELB inside containment or a LOCA. These accidents will not cause condensate accumulation in the above solenoid valves at their respective locations.

2. Valcor has indicated that the conduit entry on generically qualified model V526 solenoid valves must be sealed in order to prevent moisture intrusion and possible loss of function during an accident.

3. The above information shows that these solenoid valves are not required to operate in an accident environment which will cause condensate to form in the conduit or enter the valve housing. However, to maintain environmental qualification, conduit seals will be installed during the next scheduled outage as a result of NCR BFNEEB8407.

TER ITEM NO. 123 ADDITIONAL EQUIPMENT NO. EEB-2

TVA ID NO. 1, 3-FSV-75-57, -58

MANUFACTURER/MODEL NO. ASCO/Model 206-380-3F

STATUS IV

Justification for Continued Use

1. Solenoid valves 1, 3-FSV-75-57, -58 are located in the northwest pump room, room 3, elevation 519 of the Reactor Building. They are required to operate for 1 day following a LOCA, HELB inside containment, HELB outside containment, or control rod drop accident. Only a HELB outside containment could result in moisture intrusion into solenoid enclosure.

2. Automatic Switch Company (ASCO) Qualification Test Report, ASQ 21678/TR revision A, dated July 1979, denotes use of liquatite flexible conduit to connect solenoid enclosure to outside of test chamber. Conduit seals have not been provided for installed unit 1, 3 solenoid valves.

3. During LOCA simulation test, the flexible conduit's plastic liquid tight covering broke down allowing spray solution to enter the solenoid enclosure and degrade the coil insulation, resulting in current leakage to ground. However, test valves demonstrated operability for a minimum of 4 days into test.

4. It has been determined that solenoid valves 1, 3-FSV-75-27, -28 have an acceptable failure mode, i.e., that solenoid failure would result in required safety function being achieved.

5. The above information shows justification for continued use of solenoid valves; however, to maintain environmental qualification, conduit seals will be installed during the next scheduled outage as a result of NCR BFNEEB8407.

TER NO. 42

TVA ID NO.

1,3-ZS-1-14,-26,-37,-51

MANUFACTURER/MODEL NO.

NAMCO/EA740-50100

STATUS IV

Justification for Continued Operation

1. Limit switches 1, 3-ZS-1-14, -26, -37, -51 are located in the drywell, room 0, el. 571 feet 9 inches. They are required to operate for 1 minute following an HELB outside containment. An HELB outside containment will not cause a harsh environment inside the drywell.

2. NAMCO Controls EA740 Qualification Test Report dated February 22, 1979, states that it is the user's responsibility to seal the conduit entry into the device to prevent moisture intrusion and possible loss of function during an accident.

3. Date codes of limit switches 1, 3-ZS-1-14, -26, -37, -51 denote dates of manufacture prior to November 1980; therefore, these switches were assembled with Accobest gaskets. NAMCO Controls maintenance instruction, EA749 20010, states that during schedule maintenance (first 1 to 1-1/2 years) that the Accobest gaskets are to be replaced with silicone gaskets. This has not been accomplished.

4. The above information shows that limit switches are not required to operate in an accident environment which will cause condensate to form in the conduit or enter switch housing. However to maintain environmental qualification, conduit seals will be installed during next scheduled outage as a result of NCR BRNEEB8407, and Accobest gaskets replaced during next scheduled maintenance.

TER ITEM NO. 122

TVA ID NO. 1,3-FSV-64-29, -32

MANUFACTURER/MODEL NO. ASCO/Model No. NP831664E

STATUS IV

Justification for Continued Operation

1. Solenoid valves 1, 3-FSV-64-29 and 1, 3-FSV-64-32 are located in the Reactor Building, rooms 12 and 8, respectively. They are required to operate for 100 days following a HELB inside containment, a LOCA, or RDA. The above accidents will not cause condensate accumulation in the above solenoid valves at their locations.

2. ASCO Qualification Test Report No. AQS21678/TR, revision A (dated July 1979) states that it is the user's responsibility to seal the conduit entry into the device to prevent moisture intrusion and possible loss of function during an accident.

3. The above information shows that these solenoid valves are not required to operate in an accident environment which will cause condensate to form in the conduit or enter the valve housing. However, to maintain environmental qualification, conduit seals will be installed during the next scheduled outage as a result of NCR BFNEEB8407.

TER NO. 43

TVA ID NO.

1, 3-ZS-1-15, -27, -38, -52

MANUFACTURER/MODEL NO.

NAMCO/EA740-50100

STATUS IV

Justification for Continued Operation

1. Limit switches 1, 3-ZS-1-15, -27, -38, -52 are located in the main steam valve vault, room 7, el. 565 of the Reactor Building. They are required to operate for 1 minute following an HELB outside containment.

2. NAMCO Controls EA740 Qualification Test Report dated February 22, 1979, states that it is the user's responsibility to seal the conduit entry into the device to prevent moisture intrusion and possible loss of function during an accident.

3. Date codes of limit switches 1, 3-ZS-1-15, -22, -38, -52 denote dates of manufacturer prior to November 1980; therefore, these switches were assembled with Accobest gaskets. NAMCO Controls maintenance instructions, EA749 20010, states that during schedule maintenance (first 1 to 1-1/2 years) that the Accobest gaskets are to be replaced with silicone gaskets. This has not been acomplished.

4. Outboard limit switches 1, 3-ZS-1-15, -27, -38, -52, upon closure of the main steam isolation valves (MSIV's), initiate a reactor scram signal to the Reactor Protection System (RPS). A HELB inside the main steam valve vault will lead to rapid closure of the MSIV's via input from the main steam isolation temperature switches. These temperature switches will initiate MSIV closure whenever a HELB exists which could affect the limit switches. Thus the limit switches will provide their function before sufficient moisture intrusion occurs which causes failure.

5. The above information shows justification for continued use of the limit switches; however, to maintain environmental qualification, conduit seals will be installed during next scheduled refueling outage as a result of NCR BFNEEB8407 and Accobest gaskets replaced during next scheduled maintenance.

TER ITEM NO. 076 ADDITIONAL EQUIPMENT NO. EEB-7

TVA I.D. NO.

1,3-LS-85-45C,-45D,-45E,-45F

MANUFACTURER/MODEL NO.

MAGNETROL/MODEL 402

STATUS IV

JUSTIFICATION FOR CONTINUED USE

1. Level switches 1, 3-LS-85-45C,-45D,-45E,-45F are located in the general floor area, room 8, elevation 565 of the reactor building. They are required to operate for 1 hour following a HELB inside containment, HELB outside containment or control rod drop accident. Only a HELB outside containment could result in moisture instrusion into level switch housing.

2. Acton Environmental Testing Corporation Test Report No. 17344-82N-A dated January 24, 1983, states that it is the user's responsibility to seal the conduit entry into the device to prevent moisture intrusion and possible loss of function during an accident. Conduit seals have not been provided for installed unit 1, 3 level switches.

3. Level switches 1,3-LS-85-45C,-45D,-45E,-45F monitor Scram Discharge Instrument Volume (SDIV) and provide a scram initiate signal to Reactor Protective System (RPS) when high water level is detected (approximate 50 gal accumulation). Redundant level switches, 1,3-LS-85-45A,-45B, -45G,-45H, provide the same input, but are from a different manufacturer and do not require conduit seals.

   Failure of the switches could prevent a scram signal being generated from these devices which would reduce the NRC required redundancy.

   The switches are located where a main steam HELB outside containment could affect them short term and a Reactor Core Isolation Cooling (RCIC) HELB outside containment could affect them long term (>1 hour). In the event of a main steam HELB outside containment, a scram will occur very rapidly. The SDIV level switches are not required once a scram has been generated. Thus, the switches do not need to perform their safety function by the time a main steam HELB outside containment would affect them. Also they have a 1 hour operability requirement. Hence, the RCIC HELB outside containment is of no concern.

4. The above information shows justification for continued use of level switches. However, to maintain environmental qualification, conduit seals will be installed during the next scheduled outage as a result of NCR BFNEEB8407.

TER ITEM NO. 119

TVA ID NO. 1,3-FSV-76-24

MANUFACTURER/MODEL NO. ASCO/Model No. NP831664E

STATUS IV

Justification for Continued Operation

1. Solenoid valves 1, 3-FSV-76-24 are located in the Reactor Building, room 8. They are required to operate for 1 day following a HELB inside containment, a LOCA, or RDA. The above accidents will not cause condensate accumulation in the above solenoid valves at their location.

2. ASCO Qualification Test Report No. AQS21678/TR, revision A (dated July 1979) states that it is the user's responsibility to seal the conduit entry into the device to prevent moisture intrusion and possible loss of function during an accident.

3. The above information shows that these solenoid valves are not required to operate in an accident environment which will cause condensate to form in the conduit or enter the valve housing. However, to maintain environmental qualification, conduit seals will be installed during the next scheduled outage as a result of NCR BFNEEB8407.

024292.05

TER NO. 074

TVA I.D. NO. 1,3-FSV-64-141

MANUFACTURER/MODEL NO. ASCO/206-832-2RF

STATUS IV

JUSTIFICATION FOR CONTINUED OPERATION

1. Solenoid valves 1,3-FSV-64-141 are located in the general floor area of the reactor building, room 8. They are required to operate for 100 days following the start of a LOCA or HELB inside primary containment. These accidents will not create a moisture condensing environment at these device locations.

2. ASCO states in their qualification test report, AQS21678/TR Rev. A dated July 1979, that it is the user's reponsibility to seal the conduit entry into the device in such a manner as to prevent moisture intrusion and possible loss of function during an accident.

3. The above information shows that these solenoid valves are not required to operate in an accident environment which will cause condensate to form or enter the conduit. However, to ensure environmental qualification, conduit seals will be installed during the next scheduled outage as a result NCR BFNEEB8407.

054293.03

TER NO. 118

TVA I.D. NO. 3-FSV-84-19 and 1,3-FSV-84-20

MANUFACTURER/MODEL NO. ASCO/NP831665E

STATUS IV

JUSTIFICATION FOR CONTINUED OPERATION

1. Solenoid valves 3-FSV-84-19 and 1,3-FSV-84-20 are located in the reactor building, room 12. They are required to operate for 100 days following the start of a LOCA and HELB inside containment. These accidents will not create a moisture condensing environment at these device locations.

2. ASCO states in their qualification test report, AQS21678/TR Rev. A dated July 1979 that it is the user's responsibility to seal the conduit entry into the device in such a manner as to prevent moisture intrusion and possible loss of function during an accident.

3. The above information shows that these solenoid valves are not required to operate in an accident environment which will cause condensate to form or enter the conduit. However, to ensure environmental qualification, conduit seals will be installed during the next scheduled outage as a result of NCR BFNEEB8407.

054293.03

TER NO. 15

TVA ID NO.

1, 3-FM-84-19B, -20B

MANUFACTURER/MODEL NO.

Fisher Controls/Type 546

STATUS IV

Justification for Continued Operation

1. Electropneumatic transducers 1, 3-FM-84-19B, -20B are located in the Reactor Building, room 12. They are required to operate for 100 days following a LOCA and a HELB inside primary containment. Neither a LOCA nor a HELB inside containment will cause a harsh environment in the area that these devices are located.

2. As documented in Wyle Laboratories' report No. 17504-1, revision A, dated July 15, 1982, these transducers were tested with the conduit routed outside the accident test chamber and sealed. Therefore, in order to maintain the qualification of these transducers, the conduit entry into the devices must be sealed.

3. The above information shows that these transducers are not required to operate in an accident environment which will cause condensate to form in the conduit or enter the transducer housing. However, to maintain environmental qualification, conduit seals will be installed during the next scheduled refueling outage as determined by the resolution of NCR BFNEEB8407.

TER NO. 10

TVA ID NO.

3-FT-84-19

MANUFACTURER/MODEL NO.

Rosemount/1153DB3

STATUS IV

Justification for Continued Operation

1. Transmitter 3-FT-84-19 is located in the general floor area of the Reactor Building, elevation 621.25, room 12. It is required to operate for 100 days following a LOCA or HELB inside primary containment. Neither a LOCA nor a HELB inside primary containment will cause a harsh environment in the area that this transmitter is located.

2. Rosemount report No. 108025, revision B, dated February 22, 1983, states that all conduit connections, as well as the pipe plug used to seal off the unused conduit hub, must be sealed with a qualified thread sealant to prevent moisture entry to the terminal cavity of the transmitter in the event of a LOCA.

3. The above information shows that this transmitter is not required to operate in an accident environment which will cause condensate to form in the conduit or enter the terminal cavity of the transmitter. However to maintain environmental qualification, conduit seals will be installed during the next scheduled outage as determined by the resolution of NCR BFNEEB8407.

TER NO. 14

TVA I.D. NO. 1,3-FSV-84-8A, 8B, 8C, 8D

MANUFACTURER/MODEL NO. TARGET ROCK/73FF

STATUS IV

JUSTIFICATION FOR CONTINUED OPERATION

1. Solenoid valves 1,3-FSV-84-8A, 8B, 8C, 8D are located in the general floor area of the reactor building, room 8. They are required to operate for 10 days after the start of a LOCA or HELB inside primary containment. These accidents will not create a moisture condensing environment at these device locations.

2. The environmental qualification testing performed on this valve by Target Rock utilized a method of preventing moisture entry into the conduit. Without proper conduit sealing, moisture intrusion into the valve could cause a loss of function during an accident.

3. The above inforamtion shows that these solenoid valves are not required to operate in an accident environment which will cause condensate to form or enter the conduit. However, to ensure environmental qualification, conduit seals will be installed during the next scheduled outage as a result of NCR BFNEEB8407.

054293.03

The following JCO applies to FSV-64-20, -21.

Justification for Continued Operation

The specified environmental conditions and current qualification levels are as follows:

Parameter	Specification	Qualification
Operating time	100 days	3.75 days
Temperature	126°F	340°F
Pressure	Atmospheric	79.7 lb/in^2a
Relative humidity	100%	100%
Chemical spray	N/A	N/A
Radiation	5.1×10^6 rads, gamma	3×10^7 rads, gamma
Aging	N/A	
Submergence	N/A	N/A

1. Valves FSV-64-20 and FSV-64-21 are used to provide vacuum relief to the torus.

These valves are normally closed. However, if open, the valve must be able to close in the event of an accident. It has been determined that these solenoid valves will receive their accident closure signal within 10 seconds. Once the valves close, they are not required to reopen for accident mitigation. Analysis of the actual physical configuration of the valves indicates that all credible postulated solenoid electrical failures would result in closure of the valves, thus the valves will fail safe. These valves are required to mitigate an LOCA or HELB inside primary containment and an RDA.

WPHTX8300B45F is a water proof, high temperature, ASCO 3-way solenoid valve. The "X" identifies a monel core tube and a standard Class H coil is used in this valve. The temperature characteristics of ASCO coils are shown on page 8 of the ASCO catalog No. 30A. The temperature limitations of the coil are based on a combination of temperature rise from power input and outside temperatures. The ambient normal range to 77°F is based on long-term continuous operation and as indicated on page 8 of ASCO catalog No. 30A, the valves can be used where the ambient temperature occasionally reaches 104°F. Thus, these valves for their 10 second period of operation do not exceed the design temperature limits. Additionally, the temperature rise from power input during this 10 second period will be substantially less than that experienced for long-term continuous duty.

The seats and gaskets for these valves are contructed with Buna-N. A materials analysis of these solenoid valves reveals that Buna-N material has the greatest potential for failure due to radiation exposure. ASCO claims (ASCO Switch Co. letters to EDS Nuclear dated January 3, 1980, and August 11, 1980) that Buna-N is typically good after exposure to 7×10^6 rads and that Class H coil insulation is still serviceable after exposure to 1×10^8 rads.

According to several studies including the guidelines furnished in Bulletin 79-01B, a more conservative valve for Buna-N is 1×10^6 rads. EPRI Radiation Threshold Test Report NP-2129 dated November 1981 states that Buna-N material is still good after exposure to 2×10^6 rads. It is further noted that similar ASCO solenoid valves have been successfully tested after radiation exposures up to 3×10^7 rads (Rockwell Test Report 2792-03-02, revision 1, dated May 17, 1979). Thus, no significant degradation from the effects of radiation exposure experienced at Browns Ferry Nuclear Plant is anticipated.

Rockwell Report 2792-03-01, Rev. 1 - This report covered an 8300 series valve (test specimen model HTX8320). All valves of a particular series number are of the same basic design. Differences in materials, etc., are noted by differences in prefix or suffix letters.

The ASCO 8320 valves are miniature size, 3-way solenoid valves having valve bodies that are substantially different from the 8300 series valves. However, the same types of electrical coils are used for both the 8300 and

8320 series valves. Likewise, the same types of material for seats and gaskets are employed in both the 8300 and 8320 series valves. Therefore, it is TVA's engineering judgement that the similarities that do exist and are pertinent to the qualification effort for FSV-64-20 and FSV-64-21.

Based on similarities between this valve and the actual valve tested, this valve has been judged to surpass the basic environmental values listed. In addition, even should the valve fail, it has been determined that it would fail safe. That is, should the diaphragms fail due to radiation, the valve would close and perform the isolation function required of it.

3. TVA has identified no materials in the device known to be susceptible to significant degradation from the effects of thermal or radiation aging for the 40-year normal environment. The limiting material in the valve is the Buna-N diaphragm which can tolerate the given 40-year normal environment. Also, there are no known materials within these valve which are subject to significant degradation from the effects of thermal aging transients of the magnitude and durations experienced under Browns Ferry Nuclear Plant accident conditions.

   For thermal aging, Arrhenics techniques were applied and a 40-year life at 104°F was established for Buna-N. The normal average days peak temperature for the specified environment is 90°F.

Based on the test data, analysis, and vendor information presented in the above notes, TVA considers FSV-64-20 and FSV-64-21 fully qualified for their 40-year specified environment following installation of a conduit seal. However, because these items are not required to be qualified for an accident which would cause condensation at the valve location, interim operation without a conduit seal is acceptable (reference Failure Evaluation dated September 20, 1984, NEB 840920 260).

The following JCO applies to FSV-76-17, -18, -19.

Justification for Continued Operation

The specified environmental conditions and current qualification levels are as follows:

Parameter	Specification	Qualification
Operating time	1 day	3.75 days
Temperature	117°F	340°F
Pressure	Atmospheric	79.7 lb/in^2a
Relative humidity	100%	100%
Chemical spray	N/A	N/A
Radiation	1.1×10^6 rads, gamma	3×10^7 rads, gamma
Aging	N/A	
Submergence	N/A	N/A

1. These valves are used for primary containment isolation in the containment inerting system. These valves are normally closed. However, if open, the valves must be able to close in the event of an accident. It has been determined that these solenoid valves will receive their accident closure signal within 10 seconds. Once the valves close, they are not required to reopen for accident mitigation. Analysis of the actual physical configuration of the valve indicates that all creditable postulated solenoid electrical failures would result in closure of the valves, thus the valves would fail safe. These valves are only required to mitigate an LOCA or HELB inside primary containment and a Rod Drop Accident. The peak temperature experienced is 117°F and at 10 seconds the temperature is below 102°F. Also, reevaluation of radiation exposure during an LOCA or HELB accident has shown the accident exposure to be 1.1×10^6 rads. The RDA produces no harsh postaccident environment.

   WPHTX8300B68U is a water proof, high temperature, ASCO 3-way solenoid valve. The "X" identifies a monel core tube and a standard Class H coil is used in this valve. The temperature characteristics of ASCO coils are shown on page 8 of ASCO catalog No. 30A. The temperature limitations of the coil are based on a combination of temperature rise from power input and outside temperatures. The ambient normal range of 77°F is based on long-term continuous operation and as indicated on page 8 of ASCO catalog No. 30A, the valves can be used where the ambient temperature occasionally reaches 104°F. Thus, for their 10 second period of operation, these valves do not exceed their design temperature limits. Additionally, the temperature rise from power input during this 10 second period will be substantially less than that experienced for long-term continuous duty.

   The seats and gaskets for these valves are constructed with Buna-N. A materials analysis of these solenoid valves reveals that Buna-N material has the greatest potential for failure due to radiation exposure. ASCO claims (ASCO Switch Co. letters to EDS Nuclear dated January 3, 1980, and August 11, 1980) that Buna-N is typically good after exposure to 7×10^6 rads and that Class H coil insulation is still serviceable after exposure to 1×10^8 rads.

   According to several studies including the guidelines furnished in Bulletin 79-01B, a more conservative value for Buna-N is 1×10^6 rads. EPRI (Radiation Threshold Test Report NP-2129 dated November 1981) states that Buna-N material is still good after exposure to 2×10^6 rads. It is further noted that similar ASCO solenoid valves have been successfully tested (Rockwell Test Report 2792-03-02, revision 1, dated May 17, 1979) after radiation exposure in excess of 4×10^6 rads. Nearly all of these radiation values are above the 1.1×10^6 rad exposure that these valves will experience. Thus, no significant degradation from the effects of radiation exposure experienced at Browns Ferry Nuclear Plant is anticipated.

2. Rockwell Report No. 2792-03-02, Rev. 1 - This report covered an 8300 series valve (test specimen model HTX 8320). All valves of a particular series number are of the same basic design. Differences in materials, etc., are noted by differences in prefix or suffix letters. The ASCO 8320 valves are miniature size, 3-way solenoid valves having valve bodies that are

substantially different from the 8300 series valves. However, the same types of electrical coils are used for both the 8300 and 8320 series valves. Likewise, the same types of material for seats and gaskets are employed in both the 8300 and 8320 series valves. Therefore, it is TVA's engineering judgement that the similarities that do exist are pertinent to the qualification effort for FSV-76-17, -18, and -19.

3. Based on similarities between this valve and the actual valve tested, this valve has been judged to surpass the basic environmental values listed. In addition, even should the valve fail, it has been determined that it would fail safe. That is, should the diaphragms fail due to radiation, the valve would close and perform the isolation function required of it.

   TVA has identified no materials in the device susceptible to significant degradation from the effects of thermal or radiation aging for the 40-year normal environment. The limiting material in the valve is Buna-N diaphragms which can tolerate the given normal environment for 40 years. Also, there are no known materials within this valve which is subject to significant degradation from the effects of thermal aging transients of the magnitude and durations experienced under Browns Ferry Nuclear Plant accident conditions.

   For thermal aging, Arrhenics techniques were applied and a 40-year life at 104°F was established for Buna-N. The normal average day peak temperature for the specified environment is 90°F.

Based on the test data, analysis, and vendor information presented above, TVA considers FSV-76-17, FSV-76-18, and FSV-76-19 fully qualified for their 40-year specified environment following installation of a conduit seal. However, because these items are not required to be qualified for an accident which would cause condensation at the device location, interim operation without a conduit seal is acceptable (reference Failure Evaluation dated September 20, 1984, NEB 840920 260).

The following JCO applies to RE-90-136, -137, -138, -139.

Justification for Continued Operation

The specified environmental conditions and current qualification levels are as follows:

Parameter	Specification	Qualification
Operating Time	1 minute	
Temperature	140°F	392°F
Pressure	atmospheric	250 lb/in^2g
Relative Humidity	50%	98%
Chemical Spray	N/A	N/A
Radiation	2×10^6 Rads, gamma	
Aging		
Submergence	N/A	N/A

Gamma Sensitive Ion Chamber Physical Characteristics

Diameter	3.06 inches
Overall Length	13.47 inches
Weight	2 lbs-10 oz.
Case and Electrodes Fabricated From:	Aluminum
Insulators Fabricated From:	Alumina and Mica
Filling Gas	Argon
Connectors	Type HN (Aluminum)

Maximum Operating Environment

Gamma Sensitivity	3.7×10^{-10} A/R/hr $\pm$ 20%
Temperature	200°C (392°F)
Pressure	250 lb/in^{2}g
Vibration	Designed to Meet MIL-STD-167
Shock	Designed to Meet MIL-S-901C
Humidity	98%

Accident conditions do not expose this equipment to harsh temperature, pressure, relative humidity, or radiation. The specified environmental parameters represent the "worst case" normal environment. A maximum abnormal temperature of 160°F and relative humidity of 100% could occur for up to 1 percent of the plant life and would exist for up to eight hours per excursion.

These instruments monitor the main steam lines from the nuclear boiler to the turbine for gamma radiation to detect major fission product release from the core. These detectors are located immediately downstream of the outer isolation valve inside the steam tunnel. Detected high radiation levels initiate (1) Nuclear Boiler Scram, (2) closure of the main steam line isolation valves, and (3) turn off mechanical vacuum pump and closure of mechanical vacuum pump line valve. Power failure to any component operates the trip circuits.

The only component on the detector known to be susceptable to radiation damage is the mica insulation. According to NASA report No. CR-1787, "Radiation Effects Design Handbook," dated July 1971, mica insulating material exhibited no significant changed in physical characteristics following a dose of 1×10^8 rads, gamma at 200°C or electrical chacteristics following a dose of 1×10^{10} rads, gamma.

Moisture may collect inside the connector housings when it is exposed to high humidity environments at temperatures under 100°C. General Electric Company recommends that a sealing material be applied to the threads at the junction of the collector housing and ion chamber body. The sealing material selected should have a higher melting temperature than the ambient temperature the chamber will be subjected to.

If installation of a conduit seal does not preclude moisture intrusion through the threaded connection at the junction of the collector housing, then a sealing material will also be installed there per GE's recommendation.

These devices have been determined to be fully qualified following installation of a conduit seal.

Because this equipment is not required to mitigate an accident that would result in a condensing humidity environment, interim operation is deemed acceptable.

Sources of Information

GE specification 22A 1363

BFN Operation and Maintenance Instructions GEK-13920C

NASA Report No. CR-1787 dated July 1971.

The following JCO applies to FSV-85-37 A,B and FSV-85-39 A,B.

Justification for Continued Operation

The specified environmental conditions and current qualification levesl are as follows:

Parameter	Specification	Qualification
Operating time	1 hour	6 hours
Temperature	163°F	212°F
Pressure	Atmospheric	16.6 psia
Relative Humidity	100%	100%
Chemical Spray	N/A	
Radiation	2×10^5 Rads, gamma	1×10^6 Rads, gamma
Aging		
Submergence	N/A	

These items are fully qualified following installation of a conduit seal. Interim operation without a conduit seal is considered acceptable because these devices woule "fail safe" in the unlikely event of a failure.

1. Test data, vendor information and miscellaneous documents which were used to support operating time, temperature, humidity, radiation and aging qualification are listed below. References are made to these documents throughout the text of this appendix.

   (1) BWR Equipment Qualification Summary QSR-097-A-01 dated 10/14/80
   (2) BWR Equipment Qualification Summary QSR-097-A-02 dated 10/13/80
   (3) ASCO HCU Environmental Evaluation Report #383HA820 dated 2/23/73
   (4) Wyle Laboratory HCU Test Report #384HA183 dated 7/16/73
   (5) Asco Test Report and Certification Statement in Response to Philadelphia Electric Company's Peach Bottom Atomic Power Station query dated 6/5/80
   (6) Isomedix Testing Division - Asco Solenoid Valve Qualification Testing - Test Report #AQS21678/TR dated July 1979
   (7) Rockwell Test Report 2792-03-02 Rev.1 dated 5/17/79
   (8) Memorandum: Automatic Switch Company to EDS Nuclear, Inc. (subject: Asco Solenoid Valve Coil Radiation Exposure) dated 8/11/80
   (9) Memorandum: Automatic Switch Company to EDS Nuclear, Inc. (subject: Indian Point Nuclear Plant, Asco Valve Radiation Capability) dated 1/3/80
   (10) TVA - Browns Ferry - Mechanical Maintenance Instruction #28 - Control Rod Drive Hydraulic Unit dated 4/8/81
   (11) EPRI Radiation Threshold Test Report NP-2129 dated November 1981

2. Asco 3-way solenoid pilot valves HVA-90-405-2A were all manufactured in accordance with a single Asco drawing and all valves with the HVA prefix were specifically made for General Electric design applications. The suffix denotes the valve size and coil type; Model HVA-90-405-2A is a 2" 60-cycle AC valve. The pilot head subassembly for these valves, HVA-90-441-1A, contains a class f coil which is Asco's standard high temperature code (FT).

3. Calculated Browns Ferry Nuclear Plant accident temperatures peak at 163°F within 30 seconds, continue to decrease to 140°F within 2 minutes, and stabilize to 100°F within 24 hours. The operating time requirement for these valves is 1 hour.

   Tests (reference notes 1.1, 1.2, 1.3, and 1.4 of this appendix) with HVA-90-405 valves were conducted which exposed the valves to 212°F and 100 percent relative humidity for six consecutive hours. The valves operated satisfactorily after these tests. Asco (reference note 1.5 certifies that the HVA-90-405 valves will satisfactorily perform for the designed 40-year life after being exposed to an accident temperature that peaks at 233°F at 100 percent relative humidity for approximately one minute and remains at temperatures in excess of 140°F for several days and above 120°F for at least 2 weeks. Asco bases their certification statement on a combination of engineering analysis (supported by test data) and operating experience and material manufacturers published data (reference note 1.5). There are no known materials within these valves which are subject to significant degradation from the effects of thermal aging transients of the magnitudes and durations experienced under Browns Ferry Nuclear Plant accident conditions.

4. Calculated accident pressure increases from 14.4 psia to 14.7 psia and returns to 14.5 psia within 20 seconds.

   Test (see notes 1.3 and 1.4) descriptions make no mention of pressure conditions during the tests, however, normal atmospheric pressure is assumed. Asco (see note 1.5) states that HVA-90-405 valves will satisfactorily operate after being exposed to an accident pressure that peaks at 16.6 psia and remains near this level for a time period greater than 20 seconds. Furthermore, similar Asco 3-way solenoid valves have been tested (see notes 1.6 and 1.7) at much greater pressures with satisfactory test results.

5. Calculated accident radiation exposure is $6x10^4$ rads. The combined accident and integrated 40-year life dosage is $1.6x10^5$ rads.

   Asco (reference note 1.5) certifies that HVA-90-405 valves will satisfactorily perform for the designed 40-year life with a total exposure of $6.49x10^4$ rads. This certification was in response to a specific use situation at Peach Bottom Atomic Power Station and doesn't infer that the valves will fail because of exposure to higher radiation levels. Asco has also subjected class f coils (reference note 1.8) to $1x10^8$ rads and the coils were still serviceable. A materials analysis of the solenoid valve reveals that the diaphragm which is made with Buna-N material has the greatest potential for failure due to radiation exposure. Asco claims (reference note 1.9 of this appendix) that Buna-N is typically good after exposure to $7x10^6$ rads. According to several studies including the guidelines furnished in bulletin 79-01B, a more conservative value for Buna-N is $1x10^6$ rads. An EPRI study (reference note 1.11) states that Buna-N material is still good after exposure to $2x10^6$ rads. It is further noted that similar Asco solenoid valves have been successfully tested (reference notes 1.6 and 1.7) after radiation exposures in excess of $4x10^6$ rads. Thus, there are no significant degradation from the effects of radiation exposure experienced under Browns Ferry Nuclear Plant accident conditions.

6. Based on the combinations of test data, analysis, vendor experience, and information presented in the above paragraphs, these solenoid pilot valves are fully qualified for their accident environment.

   It should also be noted that these valves are required to open for accident mitigation purposes. No credit is taken for closing the valves. Analysis of the actual physical configuration of the valves indicates that all postulated solenoid failures would result in opening of the valves, thus, the valves will fail safe. Ultimate failure of the diaphragm due to radiation damage could not result in valve closure. TVA also has in place a normal maintenance procedure (reference note 1.10) which requires periodic inspection and replacement of parts causing sluggish valve operation or causing excessive leakage. This procedure divides the CRD modules into 5 series and at each refueling outage diaphragm and body gaskets are replaced on one of these series.

The following JCO applies to FSV-43-14.

Justification for Continued Operation

The specified environmental conditions and current qualification levesl are as follows:

Parameter	Specification	Qualification
Operating time	1 hour	3.75 days
Temperature	$132^{o}F$	$340^{o}F$
Pressure	Atmospheric	79.7 psia
Relative Humidity	100%	100%
Chemical Spray	N/A	
Radiation	$6.1x10^{4}$ Rads, gamma	$3x10^{7}$ Rads, gamma
Aging		
Submergence	N/A	

1. This valve is needed to provide ioslation of a recirculation pump sampling line. The valve is normally closed. However, if open, the valve must be able to close in the event of an accident. It has been determined that the solenoid valve will receive its accident closure signal within ten seconds. Once the valve closes, it is not required to reopen for accident mitigation. Analysis of the actual physical configuration of the valves indicates that all credible postulated solenoid electrical failures would result in closure of the valve, thus the valves would fail safe. This valve is only required to mitigate an LOCA or HELB inside primary containment and an RDA. The peak accident temperature experienced is 145°F and at ten seconds the temperature is below 102°F.

   X8300-B61F is a general service ASCO 3-way solenoid valve. The "X" identifies a monel core tube and a standard Class H coil is used in this valve. The temperature characteristics of ASCO coils are shown on page 8 of ASCO catalog No. 30A. The temperature limitations of the coil are based on a combination of temperature rise from power input and outside temperatures. The ambient normal range to 77°F is based on long-term, continuous operation as indicated on page 8 of ASCO catalog No. 30A, the valves can be used occasionally where the ambient temperature reaches 104°F. Thus, FSV-43-14 for its ten-second period of operation does not exceed the design temperature limits. Additionally, the temperature rise from power input during this ten-second period will be substantially less than that experienced for long-term, continuous duty.

   The seats and gaskets for FSV-43-14 are constructed with Buna-N. A materials analysis of the solenoid valve reveals that Buna-N material has the greatest potential for failure due to radiation exposure. ASCO claims (ASCO Switch Co. letters to EDS Nuclear dated January 3, 1980, and August 11, 1980) that Buna-N is typically good after exposure to 7×10^6 rads, and that Class H coil insulation is still serviceable after exposure to 1×10^8 rads.

   According to several studies including the guidelines furnished in Bulletin 79-01B, a more conservative value for Buna-N is 1×10^6 rads. EPRI Radiation Threshold Test Report NP-2129 dated November 1981 states that Buna-N material is still good after exposure to 2×10^6 rads. It is further noted that similar ASCO solenoid valves have been successfully tested (Rockwell Test Report 2792-03-02, revision 1, dated May 17, 1979) after radiation exposures in excess of 4×10^6 rads. All of these radiation values are well above the 5×10^3 rad exposure that FSV-43-14 will experience. Thus, there is no significant degradation from the effects of radiation exposure experienced under Browns Ferry Nuclear Plant accident conditions.

2. Rockwell Report No. 2792-03-02, Rev. 1 - This report covered an 8300 series valve (test specimen model HTX8320). All valves of a particular series number are of the same basic design. Differences in materials, etc., are noted by differences in prefix or suffix letters. The ASCO 8320 valves are miniature size 3-way solenoid valves having bodies that are substantially different from the 8300 series valves. However, the same types of electrical coils are used for both the 8300 and 8320 series valves.

Likewise, the same types of material for seats and gaskets are employed in both the 8300 and 8320 sereis valves. Therefore, it is TVA's engineering judgement that the similarities that do exist are pertinent to the qualification effort for FSV-43-14.

3. TVA has identified no materials in the device susceptible to significant degradation from the effects of thermal or radiation aging for the 40-year normal environment. The limiting material in the valve is Buna-N diaphragms which can tolerate the given normal environment for 40 years. Also, there are no known materials within these valves which are subject to significant degradation from the effects of thermal aging transients of the magnitude and durations experienced under Browns Ferry Nuclear Plant accident conditions.

   For thermal aging, Arrhenius techniques were applied and a 40-year life at 104°F was established for Buna-N. The normal average day peak temperature for the specified environment is 90°F.

The following JCO applies to FS-73-33 (Unit 1 only).

Justification for Continued Operation

The specified environmental conditions and current qualification levels are as follows:

Parameter	Specification	Qualification
Operating time	1 day	6 hours
Temperature	124°F	200°F
Pressure	Atmospheric	15 lb/in^2a
Relative humidity	100%	100%
Chemical spray	N/A	N/A
Radiation	$3.2x10^5$	$3x10^6$
Aging	N/A	
Submergence	N/A	N/A

1. Barton Test Report No. R3-288A-1, GE Purchase Part Drawing 158B7015, GE Instrument Data Sheet 234A9300, NUTECH Communications Record dated September 23, 1980, File No. 101.2401.00300, and Wyle Summary Report QSR-027-A-02.

2. The operating time of 1 day is longer than the test duration of 6 hours for relative humidity, in TVA's engineering judgment, the device should adequately meet the operating time requirements.

3. A material analysis reveals no materials subject to significant degradation from the effects of thermal aging due to the normal environment.

4. A material analysis indicates that turbine oil is generally used as the bellows fill. Most oils are acceptable to 10^6 rads. (Radiation effect on electrical/ electronic materials.) Barton report No. R3-288A-1 indicates that their hydrocarbon oil has been tested and is acceptable to $3x10^6$ rads. Although the radiation test was not performed concurrent or before the balance of the qualification tests detailed in the Wyle Summary Report, QSR-027-A-02, TVA feels the device will function properly for the relatively low required radiation dose. The specific dose of $3.2x10^5$ rads is based on a full one-year accident exposure added to the forty-year normal integrated dose.

5. The worst-case accident temperature profile for compartment No. 2 is based on an HPCI HELB. In this case, the HPCI system would be isolated and inoperable; therefore, the switch would not be required. The next worst accident temperature profile to affect compartment No. 2 would be an MSLB in the steam vault. The maximum temperature from this break would be 124°F; therefore, the switch is qualified to the required temperature parameter. GE purchase part drawing 158B7015 and Instrument Data sheet 234A9300 show that the accident temperature vendors specified values for this component.

6. NUTECH communications record dated September 23, 1980, file No. 101.2401.00300, documents vendors concurrence that model 289 and 289A are identical for environmental qualification purposes.

This device is fully qualified following installation of a conduit seal.

Analysis shows that if the device fails during a non-HPCI HELB that RPV pressure can be maintained low through remote manual operation of the safety relief valves (SRV). (See Failure Evaluation dated September 20, 1984 NEB 840920 260.) Therefore, TVA considers interim operation of this device to be acceptable until a conduit seal can be installed.

The following JCO applies to FS-73-33 (Units 2, 3)

Justification for Continued Operation

The specified environmental conditions and current qualification levels are as follows:

Paramater	Specification	Qualification
Operating time	1 day	6 hours
Temperature	108°F	212°F
Pressure	Atmospheric	15 lb/in^2a
Relative humidity	100%	100%
Chemical spray	N/A	N/A
Radiation	$3.2x10^5$	$1x10^6$
Aging	N/A	
Submergence	N/A	N/A

1. Barton Test Report No. R3-288A-1, GE Purchase Part Drawing 158B7015, GE Instrument Data Sheet 234A9300, NUTECH Communications Record dated September 23, 1980, File No. 101.2401.00300, and Wyle Summary Report QSR-027-A-02.

2. The operating time of 1 day is longer than the test duration of 6 hours for relative humidity, in TVA's engineering judgment, the device should adequately meet the operating time requirements.

3. A material analysis reveals no materials subject to significant degradation from the effects of thermal aging due to the normal environment.

4. A material analysis indicates that turbine oil is generally used as the bellows fill. Most oils are acceptable to 10^6 rads. (Radiation effects on electrical/ electronic materials.) Barton report No. R3-288A-1 indicates that their hydrocarbon oil has been tested and is acceptable to 3×10^6 rads. Although the radiation test was not performed concurrent or before the balance of the qualification tests detailed in the Wyle Summary Report, QSR-027-A-02, TVA feels the device will function properly for the relatively low required radiation dose. The specific dose of 3.2×10^5 rads is based on a full one-year accident exposure added to the forty-year normal integrated dose.

5. The worst-case accident temperature profile for compartment No. 2 is based on an HPCI HELB. In this case, the HPCI system would be isolated and inoperable; therefore, the switch would not be required. The next worst accident temperature profile to affect compartment No. 2 would be an MSLB in the steam vault. The maximum temperature from this break would be 124°F; therefore, the switch is qualified to the required temperature parameter. GE purchase part drawing 158B7015 and Instrument Data sheet 234A9300 show that the accident temperature vendors specified values for this component.

6. NUTECH communications record dated September 23, 1980, file No. 101.2401.00300, documents vendors concurrence that model 289 and 289A are identical for environmental qualification purposes.

This device is fully qualified following installation of a conduit seal.

Analysis shows that if the device fails during a non-HPCI HELB that RPV pressure can be maintained low through remote manual opertion of the safety relief valves (SRV). (See Failure Evaluation dated September 20, 1984 NEB 840920 260.) Therefore, TVA considers interim operation of this device to be acceptable until a conduit seal can be installed.

The following JCO applies to FCV-69-1 (Unit 1 only)

Justification for Continued Operation

The specified environmental conditions and current qualification levels are as follows:

Paramater	Specification	Qualification
Operating time	1 hour	30 days
Temperature	325°F	385°F
Pressure	69.4 psia	90 psia
Relative humidity	100%	100%
Chemical spray	Demineralized water	Boric Acid Solution
Radiation	1.06×10^8	2.04×10^8
Aging	N/A	
Submergence	N/A	N/A

1. Wyle test report No. 43979-1 reports results from LOCA testing on two Rotork motor actuators. The LOCA simulated temperature/ pressure profile initiates at 385°F, 75 lb/in^{2}g and continues with step decreases in temperature and pressure for 30 days. Both test specimens had radiation exposures (LOCA plus 40-year normal) - total integrated dose of 2.038 x 10^8 rads, gamma. Both specimens underwent environmental (temperature, pressure, steam) and mechanical aging to place the specimen in an end-of-life condition prior to functional testing.

2. Wyle test report No. 43979-1 reports test results on 2 Rotork actuators subjected to a high temperature steam environment simulating post high energy pipe break conditions. These actuators performed successfully under load at 492°F, +20°F, -0°F, 10 lb/in^{2}g, for 10 minutes.

3. The available qualification test reports for this actuator are on devices which were tested with sealed conduit equivalents. It has been indeterminant if moisture intrusion will cause failure. The assumed failure mode is for the valve to fail "as is." The safety function of the valve is to close for either isolation of a RWCU HELB-OPC or to prevent recirculation of radioactive fluid through nonqualified piping following an LOCA or HELB-IPC. As the valve is located inside primary containment, it is exposed to either LOCA or HELB-IPC conditions. An LOCA would lead to rapid closure of the valve as the auto close signal parameters would be reached within 1 minute of accident initiation. It is reasonable to conclude that the valve will close prior to failure of the operator as closure will be complete within 1-1/2 minutes of LOCA initiation (30 second closure time).

   An HELB-IPC could introduce large quantities of steam and moisture into the drywell prior to auto close being initiated for the valve. Thus, the valve can be postulated to fail open. However, there are two valves, FCV-69-2 and FCV-69-12, located outside the primary containment (both of which are qualified) which receive closure signals based on the same logic as for FCV-69-1. Even considering a single failure, recirculation of water from/to the RPV would be precluded. If FCV-69-2 failed open as the single failure, then environmentally nonqualified portions of the RWCU system would be open to the RPV. If either of two normally closed valves, FCV-69-16 or -17, were to open in conjunction with PCV-69-15 opening, then RPV water could leave the primary and secondary containments (assuming other blockages are not created). While these latter valves are not included in the environmental qualification program, they are similar to valves being qualified and their control and power cables are identical to those being qualified. Thus, spurious opening of these devices is extremely unlikely.

   Based upon the above operation with 1-FCV-69-1 without conduit sealing is judged to be acceptable.

TER NO. EEB-9

TVA ID NO. 3-LS-73-56A, -56B

MANUFACTURER/MODEL NO. Magnetrol/Model 291

STATUS IV

Justification for Continued Operation

1. Level switches 3-LS-73-56A, -56B are located in the northeast pump room, room 4, elevation 519 of the Reactor Building. They are required to operate for 1 day following a HELB (except HPCI line break) inside primary containment or a HELB (except HPCI line break) outside primary containment.

2. The environmental qualification testing performed on these level switches by Acton Environmental Testing Corporation (Magnetrol Test Report 3170-254, Revision 2) utilized a method of preventing moisture intrusion into the switch housing during HELB environment simulation tests. Without proper conduit sealing, moisture intrusion into the switch housing could cause loss of function during an accident.

3. The failure mode resulting from moisture intrusion during an accident was reviewed by NEB (NEB 840917 220) to determine if an adverse impact on nuclear safety was created. It was determined that the failure mode was acceptable--device failure resulted in the required safety function being achieved or not being defeated.

4. The above information shows justification for continued use of level switches; however, to maintain environmental qualification, conduit seals will be installed during the next scheduled outage as a result of NCR BFNNEB8407.

TER NO. 62

TVA ID. NO. 3-LS-73-57A, -57B

MANUFACTURER/MODEL NO. Magnetrol/Model 291

STATUS IV

Justification for Continued Use

1. Level switches 3-LS-73-57A, -57B are located in the pressure suppression chamber (Torus rooms), room 6, elevation 519 of the Reactor Building. They are required to operate for 1 day following a HELB (except for HPCI line break) inside primary containment or a HELB (except HPCI line break) outside primary containment.

2. The environmental qualification testing performed on these level switches by Acton Environmental Testing Corporation (Magentrol Test Report 3170-254, Revision 2) utilized a method of preventing moisture intrusion into the switch housing during HELB environment simulation tests. Without proper conduit sealing, moisture intrusion into the switch housing could cause loss of function during an accident.

3. The failure mode resulting from moisture intrusion during an accident was reviewed by NEB (NEB 840917 220) to determine if an adverse impact on nuclear safety was created. It was determined that the failure mode was acceptable--device failure resulted in the required safety function being achieved or not being defeated.

4. The above information shows justification for continued use of level switches; however, to maintain environmental qualification, conduit seals will be installed during the next scheduled outage as a result of NCR BFNNEB8407.

NMP-10104

B6
Rfd

NIAGARA MOHAWK POWER CORPORATION

NIAGARA MOHAWK

300 ERIE BOULEVARD, WEST
SYRACUSE, N. Y. 13202

December 7, 1984

File Room EW-350

Director
Office of Inspection and Enforcement
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Attn: Document and Control Desk

Re: Docket No. 50-220
DPR-63

Dear Sir:

Submitted herewith is the Report of Operating Statistics and Shutdown for November 1984 for the Nine Mile Point Nuclear Station Unit #1.

Also included is a narrative report of Operating Experience for November 1984.

Very truly yours,

Thomas E. Lempges

Thomas E. Lempges
Vice President
Nuclear Generation

TEL/lo
attachments
cc: Director, Office of I&E (10 copies)

AGARA MOHAWK POWER CORPORATION

NINE MILE POINT NUCLEAR STATION UNIT #1

NARRATIVE OF OPERATING EXPERIENCE

The station operated during the month of November 1984 with a Unit Availability Factor of 92.7% and a Net Design Electrical Capacity Factor of 89.1%. There were 0 challenges to Electromatic Relief Valves. Reductions in Capacity Factor were due to Failure of #13 Feedwater Controller and oil leak in turbine Control Cabinet.

CLASS I WORK - MECHANICAL MAINTENANCE - NOVEMBER 1984

WR#	29805	Repaired CRD Accumulator 26-19
WR#	29832	Checked RBCLC #12 Heat Exchanger
WR#	29945	Replaced fuel pump - Diesel Generator 103
WR#	29953	Replaced stem packing - CRD 10-11 Foot Valve #111

CLASS I WORK - ELECTRICAL MAINTENANCE - NOVEMBER 1984

WR# 30066 — Reactor Building Closed Loop Cooling Pump Motor #13 Brkr-Tripping, The breaker was tested, a chart recorder was installed on the cable leads and monitored for 12 days. No problems were discovered, motor back in service

MO# 1927 — This major order involves updating station equipment for Equipment Qualification. The work performed includes wiring position limit switches, differential pressure transmitters and sealing condulets with Biscoseal. Geared limit switch grease was changed for the Emergency Condenser Isolation Valves that are to be installed for the existing valves. The systems involved are the Reactor Containment N_2 Purge and Fill, Reactor Core Spray, Containment Spray Raw Water, Containment Spray and Post LOCA Containment Vent System.

CLASS I WORK - INSTRUMENTATION & CONTROL - NOVEMBER 1984

WR# 29814 — System 12 O^2 Analyzer Reading down scale on zero to 5% scale. Also reading too high on zero to 25% scale. (recalibrated per procedure ICP201.2 H_2O_2.)

WR# 29239 — Instrument Air System #11 Instrument air loading solenoid leaking through (replaced loading solenoids U1 & U2)

WR# 29951 — Accumulator level switch 42-39 not functioning properly (replaced switch)

WR# 27972 — IPRM Flux Amplifier 44-25B erratic (repaired broken shield wire on input plug)

OPERATING DATA REPORT

DOCKET NO. 50-220
DATE 12/5/84
COMPLETED BY T.W. Roman
TELEPHONE (315) 349-2422

OPERATING STATUS

1. Unit Name: Nine Mile Point Unit #1
2. Reporting Period: November 1984 11/1/84-11/30/84
3. Licensed Thermal Power (MWt): 1850
4. Nameplate Rating (Gross MWe): 640
5. Design Electrical Rating (Net MWe): 630
6. Maximum Dependable Capacity (Gross MWe): 620
7. Maximum Dependable Capacity (Net MWe): 610
8. If Changes Occur in Capacity Ratings (Items Number 3 Through 7) Since Last Report, Give Reasons:

Notes

9. Power Level To Which Restricted, If Any (Net MWe):
10. Reasons For Restrictions, If Any:

	This Month	Yr.-to-Date	Cumulative
11. Hours In Reporting Period	720	8041.0	133321.2.
12. Number Of Hours Reactor Was Critical	688	5722.5	92023.7
13. Reactor Reserve Shutdown Hours	0	0	1204.2
14. Hours Generator On-Line	667.5	5639.0	89127.3
15. Unit Reserve Shutdown Hours	0	0	20.4
16. Gross Thermal Energy Generated (MWH)	1,209,353.0	9,970412.0	148,064,852.0
17. Gross Electrical Energy Generated (MWH)	410076.0	3332016.0	48963797.0
18. Net Electrical Energy Generated (MWH)	397719.0	3230638.0	47425397.0
19. Unit Service Factor	92.7	70.1	66.9
20. Unit Availability Factor	92.7	70.1	66.9
21. Unit Capacity Factor (Using MDC Net)	90.6	65.9	58.3
22. Unit Capacity Factor (Using DER Net)	89.1	64.8	57.4
23. Unit Forced Outage Rate	7.3	0.9	16.5

24. Shutdowns Scheduled Over Next 6 Months (Type, Date, and Duration of Each):

25. If Shut Down At End Of Report Period, Estimated Date of Startup:
26. Units In Test Status (Prior to Commercial Operation):

	Forecast	Achieved
INITIAL CRITICALITY		
INITIAL ELECTRICITY		
COMMERCIAL OPERATION		

UNIT SHUTDOWNS AND POWER REDUCTIONS

DOCKET NO. 50-220
UNIT NAME 9 Mile Pt. Unit #1
DATE 12/5/84
COMPLETED BY T.W. Roman
TELEPHONE 9315)349-2422

REPORT MONTH November 1984

No.	Date	Type[1]	Duration (Hours)	Reason[2]	Method of Shutting Down Reactor[3]	Licensee Event Report #	System Code[4]	Component Code[5]	Cause & Corrective Action to Prevent Recurrence
84-14	11/8/84	F	.10	A					#13 Feedwater Pump Controller replacement caused power reduction to 80% power.
84-15	11/11/84	F	37.5	A	1				Shutdown because of oil leak in Turbine Control Cabinet
84-16	11/14/84	F	15.0	A	3				While starting up, scramed on Rx Lo Level due to mechanical Pressure Regulator malfunction.

1
F: Forced
S: Scheduled

2
Reason:
A-Equipment Failure (Explain)
B-Maintenance or Test
C-Refueling
D-Regulatory Restriction
E-Operator Training & License Examination
F-Administrative
G-Operational Error (Explain)
H-Other (Explain)

3
Method:
1-Manual
2-Manual Scram.
3-Automatic Scram.
4-Other (Explain)

4
Exhibit G - Instructions for Preparation of Data Entry Sheets for Licensee Event Report (LER) File (NUREG-0161)

5
Exhibit I - Same Source

(9/77)

AVERAGE DAILY UNIT POWER LEVEL

DOCKET NO. 50-220
UNIT 9 Mile Pt. #1
DATE 12/5/84
COMPLETED BY TW Roman
TELEPHONE (315) 349-2422

MONTH November 1984

DAY	AVERAGE DAILY POWER LEVEL (MWe-Net)	DAY	AVERAGE DAILY POWER LEVEL (MWe-Net)
1	610	17	613
2	610	18	612
3	607	19	612
4	607	20	611
5	611	21	612
6	610	22	611
7	608	23	611
8	583	24	614
9	604	25	610
10	606	26	609
11	459	27	610
12	0	28	612
13	0	29	610
14	388	30	611
15	540	31	
16	572		

INSTRUCTIONS

On this format, list the average daily unit power level in MWe-Net for each day in the reporting month. Compute to the nearest whole megawatt.

(9/77)