Duquesne Light

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October 28, 1982

Director of Nuclear Reactor Regulation
United States Nuclear Regulatory Commission
Attn: Mr. Steven A. Varga, Chief
Operating Reactors Branch No. 1
Division of Licensing
Washington, DC 20555

Reference: Beaver Valley Power Station
Docket No. 50-334, License No. DPR-66
Supplemental Information to Fire Protection - Appendix R Review Report: Allowable time to achieve cold shutdown.

Gentlemen:

Provided in the enclosure is a description of the alternate shutdown procedures which will be revised based on our discussions with your review Staff on October 14, 1982. The procedures will be completed following your final review of our Appendix R submittal report and receipt of the Safety Evaluation Report. The original draft procedures, which are documented in Chapter 7 of our Appendix R submittal report, were written to provide alternate methods of accomplishing station shutdown in a safe manner but not within a 72 hour time period.

Based on the final rule requiring all reactor plants licensed to operate prior to January 1, 1979 to comply with Section III.G - "Fire Protection of Safe Shutdown Capability", Section III.J - "Emergency Lighting", and Section III.O - "Oil Collection Systems for RCP's", it is our opinion that Section III.L - "Alternative and Dedicated Shutdown Capability" does not apply to us. If III.L is not applicable to our facility and since BVPS - Unit No. 1 was designed as a hot shutdown plant and since there are no codified requirements that specify cold shutdown must be achieved within 72 hours, we respectfully disagree that we are required to demonstrate the design capability for achieving cold shutdown within 72 hours for our facility.

The rule provides an exemption procedure in accordance with 10 CFR 50.48 (c) (6), and also 10 CFR 50.12(a). Therefore, if the Staff does not concur with our above stated position, we hereby request an exemption, pursuant to 10 CFR 50.12(a), from the provisions of Section III.L of Appendix R and the requirement to demonstrate the design capability to achieve cold shutdown within 72 hours as specified by your Staff and

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an NRC memorandum from Roger J. Mattson to Richard H. Vollmer, dated July 2, 1982, titled "Position Statement on Allowable Repairs for Alternative Shutdown and on the Appendix R Requirement for Time Required to Achieve Cold Shutdown".

During the meeting held in Bethesda, Maryland on October 14, 1982 between Duquesne Light and members of the NRC review Staff, it was indicated that among the instrumentation they considered required for hot shutdown conditions was a source range flux monitor. Since this instrument is not energized until approximately 20 minutes after a reactor trip and all rods are fully inserted with a stable coolant system temperature, shutdown margin is assured and the inability to monitor neutron flux is not a safety concern. The existing standard Westinghouse Technical Specification (NUREG 0452, Revision 4, 3.3.3.5) for pressurized water reactors recognizes this fact, in that a remote source range flux monitor is permitted to be out of service for 7 days or hot shutdown conditions must be achieved within 12 hours. This capability is consistent with General Design Criteria 19 of 10 CFR 50 and is stated in the basis for the specification. GDC 19 includes consideration of "potential capability for subsequent cold shutdown of the reactor through the use of suitable procedures". With the assurance that power is disconnected from the rod drive motors and cold shutdown boron concentration has been established in the coolant system, shutdown margin is assured as potential sources of dilution water will be disabled prior to cooldown of the Unit from the remote panel. If the Staff does not concur with this evaluation, then an exemption pursuant to 10 CFR 50.12(a) is hereby requested for the requirement of a source range detector which the staff has interpreted as a position relative to Section III.L.2, Appendix R 10 CFR 50.

As previously identified in our Appendix R submittal report, we have considerably upgraded our fire protection program by the implementation of modifications and also administratively. The estimated cost of the modifications which were required as a result of Appendix A fire protection review and subsequent issuance of the Safety Evaluation Report (summarized in Table 2-1) was approximately $21.5 million. Presently, we have two (2) full-time fire protection engineers on-site. The manhours spent on operational surveillance testing for the fire protection systems and equipment is approximately 3,000 manhours/year. This does not include the daily inspection tours by our operations personnel. In addition, an average of 188 manhours/year is spent on the preventive maintenance programs by our mechanical and electric maintenance personnel and 164 manhours by instrumentation and control maintenace personnel. This manpower allocation on the fire protection systems and equipment far exceeds inspection efforts of any single safety system at our facility.

Duquesne Light is fully committed to allocate the necessary resources to address credible fire protection concerns to achieve a timely resolution of all open items identified during the review effort. The review of our facility using the criteria of 50.48 and Appendix R Section III.G., III.J. and III.O, applicable to Beaver Valley Unit 1 has revealed a number of conditions that are not in strict compliance to the codified criteria.

Evaluations of the significance of these deviations has lead to the conclusion that conformance is not required where alternate methods, or "equivalent protection" (ie., the ability to quantitatively demonstrate that existing fire protection modifications provide a level of safey protection which is equivalent to that required by the strict prescriptions of Appendix R) would provide reasonable assurance that safe shutdown capability would be maintained, thereby assuring the protection of the public health and safety.

Please contact my staff if additional information or clarification is necessary.

Very truly yours,

Robert F. Balcerek for

J. J. Carey
Vice President, Nuclear

Enclosures

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Appendix "R" - Alternate Shutdown Procedures

The procedures necessary for maintaining hot shutdown conditions will be structured such that the operators will not have to lift leads, b) install jumpers or c) remove inaccessible fuses. These internal review criteria were forwarded to us October 14, 1982 in a "MEMORANDUM for R. J. Vollmer from R. J. Mattson dated July 2, 1982", which represent new requirements beyond that which were originally issued in 10 CFR 50.48 and Appendix R Items IIIG, J and O applicable to Beaver Valley Unit I.

The intent of the original procedures was to maintain hot shutdown conditions (Tavg 350F-547F) for 72 hours utilizing equipment identified in Chapter 4, "Systems Required for Safe Shutdown", of our Appendix R submittal and emergency power systems only, after which an approach to cold shutdown would begin using offsite power systems.

There is no technical basis for a safety concern associated with maintaining the reactor coolant system at hot shutdown conditions for extended periods providing reactivity, pressure, coolant inventory, temperature and heat sink are maintained within appropriate limits. Beaver Valley Unit 1 was designed as a hot shutdown plant and there are no codified requirements in 10 CFR applicable to Beaver Valley 1 that specify cold shutdown must be achieved within 72 hours. For this reason, no specific exemptions were requested from Appendix R 10 CFR 50 (III.L) since it is not applicable to our facility.

The following are the major steps that would be taken to achieve and maintain hot shutdown conditions:

1. Subcriticality would be achieved by rod insertion from the Control Room or Rod Drive MG set area. The Station Emergency Plan would be initiated, and the NSS Administrative Assistant would obtain additional support personnel to augment the on-shift personnel. The protection systems would be defeated after removing sources of fire induced LOCAs.

2. The Backup Indicating Panel (BIP) would be placed into service to monitor pressurizer pressure/level, selected core exit thermocouples, cold leg temperatures and steam generator levels. Steam pressure would be monitored locally as necessary.

3. The NSS (Nuclear Shift Supervisor) would monitor the panel indications and direct two (2) operators. One operator would be located in the 722' elevation of the Auxiliary Building and would maintain boration and charging flow paths for reactor coolant system inventory and establish charging system boron concentrations based on instructions from the NSS.

The boration for plant cooldown would be accomplished in one of the following manners:

- Emergency manual boration of a 7000 ppm boron solution if a Boric Acid Transfer Pump is available
- Controlled manual injection, utilizing a high head charging pump, of 20,000 ppm boron solution using the Boron Injection Tank, or
- Controlled manual injection of 2000 ppm boron solution using the RWST Valves (MOV-CH-115B or D)

No load operating temperatures (approximately 547F) would be maintained until the requisite cooldown boron concentration was verified. The boration rate would be dependent on the status of available letdown flow paths from the reactor coolant system to avoid solid water conditions in the pressurizer at operating temperatures.

Letdown from the reactor coolant system would be provided by one of the following flow paths dependent on the availability of equipment and the fire affected zone:

- Normal letdown flow path to the Charging System,
- Loop drain flow path to Primary Drains Tank #1 using valves (MOV-RC-557A, B or C),
- Letdown Relief Valve (RV-CH-203) using valves (LCV-CH-460A and B, TV-CH-200A, B or C),
- Reactor Coolant Pump Seal Leakoff Header Relief (RV-CH-382), or
- Reactor Vessel Head Vent System

Seal injection flow, charging flow and letdown flow would be balanced to maintain reactor coolant system inventory during the boration and subsequent cooldown.

The second operator would be located in the Main Steam Valve Room/ Auxiliary Feed Pump Area. This operator would manually regulate feed flow and bleed steam through the Decay Heat Removal Valve or Steam Generator Power Operated Relief Valves. Communications that could be used include local telephone (PAX Lines), page party, radio, or sound powered jacks and headsets.

4. A gradual cooldown depressurization would begin following stable control of reactor coolant system parameters and verification of reactor coolant system boron concentration. The rate of the cooldown/depressurization would depend on the availability or use of non-IE equipment i.e., reactor coolant pumps, natural circulation cooldown, spray valve availability, shroud cooling, boration confirmation, containment cooling, pressurizer heater capacity, etc.) In any case, the plant could be maintained at hot shutdown conditions and a cooldown initiated once the availability of this equipment had been assessed. This assessment could be made by the Shift Technical Advisor while stable primary plant parameters were

being maintained. We have attached a graph showing the various cooldown/depressurization envelopes based on present Appendix G limits.

Primary plant pressure reductions would be established by any of the following means, dependent on the availability of equipment and the location of the fire:

- Pressurizer Heaters, Reactor Coolant Pumps and Spray Valves
- Pressurizer vapor space venting through the Reactor Vessel Head Vent or Sample System
- Charging through the Auxiliary Spray Valve (MOV-CH-311), when pressurizer temperature was within 320°F of the charging system suction source temperature
- Alternate filling and draining operations within the pressurizer indicating range to affect cooldown of the pressurizer
- Matching plant cooldown rates within Appendix G limits to pressurizer pressure decreases due to ambient heat losses
- Placing a nitrogen blanket in the pressurizer, or floating a solid pressurizer on the letdown system or low head safety injection/accumulator subsystems during low pressure conditions.

5. Bleeding steam would be the primary method of heat removal and this method would be utilized until it became essentially ineffective for heat removal, specifically when core decay heat production equaled steam heat removal capability, Tave would stabilize. This temperature and the time required to attain it would vary with core power history and the cooldown rate of the primary system. Under a natural circulation cooldown condition (loss of offsite power, component cooling water or instrument air), a maximum cooldown rate of 25°F/hr. is used to preclude reactor vessel head void formation and pressure is kept at approximately 2000 psig until the hot leg temperatures are 450°F. A subcooling margin of 175° must be maintained if the CRDM Shroud Cooling Fans and Reactor Coolant Pumps are not in service. The SIS accumulators would be isolated at approximately 1000 psig in the reactor coolant system. At 350°F, 800-900 psig the system would be placed in a "soak condition" for 20 hours to permit cooldown of the vessel head region and to reduce the thermal stresses within the reactor vessel for Appendix G considerations. Cooldown and depressurization would then continue until maximum steam bleed rate and core decay heat levels stabilize Tave below 240°F and approximately 25 psig steam pressure.

6. At this point, each steam generator would sequentially be filled rapidly to approximately 75% level to maximize the ΔT and energy transfer from the primary to the secondary, producing a relatively sharp drop in system temperature.

7. After stabilizing temperature and pressure following the fill of the steam generators, the main steam lines would have cradles installed to support the static loading of water to permit the secondary side of the steam generators to be filled water solid.

If condenser vacuum was available, the steam generators would have a vacuum placed on them for heat removal instead of the water to water heat exchanger mode of operation.

8. The steam generators would then be filled using auxiliary feedwater. The discharge valve of the AFW Pump would be throttled when steam generator levels exceeded 90%, at which time the 2" bypass valves around the Main Steam Isolation Valves (MSIVs) would be opened to preclude a secondary side pressure surge when the steam generator went water solid.

9. After all 3 steam generators were water solid, the (3) 2" bypass valves around the MSIVs would be throttled in conjunction with feedwater flow to stabilize flow to each steam generator. This water would be diverted to the main condenser through the condenser steam dump system which would then be drained to several locations.

10. This mode of heat removal would continue until the coolant system average temperature was below 200°F. A safety related backup source of makeup water is available for this function to continue indefinitely in the existing Auxiliary Feedwater System.

The steps that will be taken within the procedure for the alternate shutdown methods to preclude spurious valve operation are not considered necessary for maintenance of hot shutdown conditions unless that event would:

- Redirect or drain fluid inventories
- Block vital pump cooling or lubrication flow paths
- Inhibit flow within a critical flow path
- Cause flooding of a compartment containing equipment necessary for maintaining hot shutdown.

These valves will be positioned and de-energized in the required state necessary for hot shutdown condition on a priority basis. Motor operated valves within the flow paths previously identified are three phase 480 volt motors.

For three-phase power cables, in order for a piece of equipment to be activated in forward or reverse, it would be necessary for two cables to become severed and then for those two cables to become reconnected, to an energized power source. It would also require that one of the cables be connected to a power source and the other to a piece of equipment. This type of failure is not considered to be credible, therefore, prompt positioning and de-energization of 3 phase 480 volt motor operated valves is not considered necessary for maintaining hot shutdown conditions.

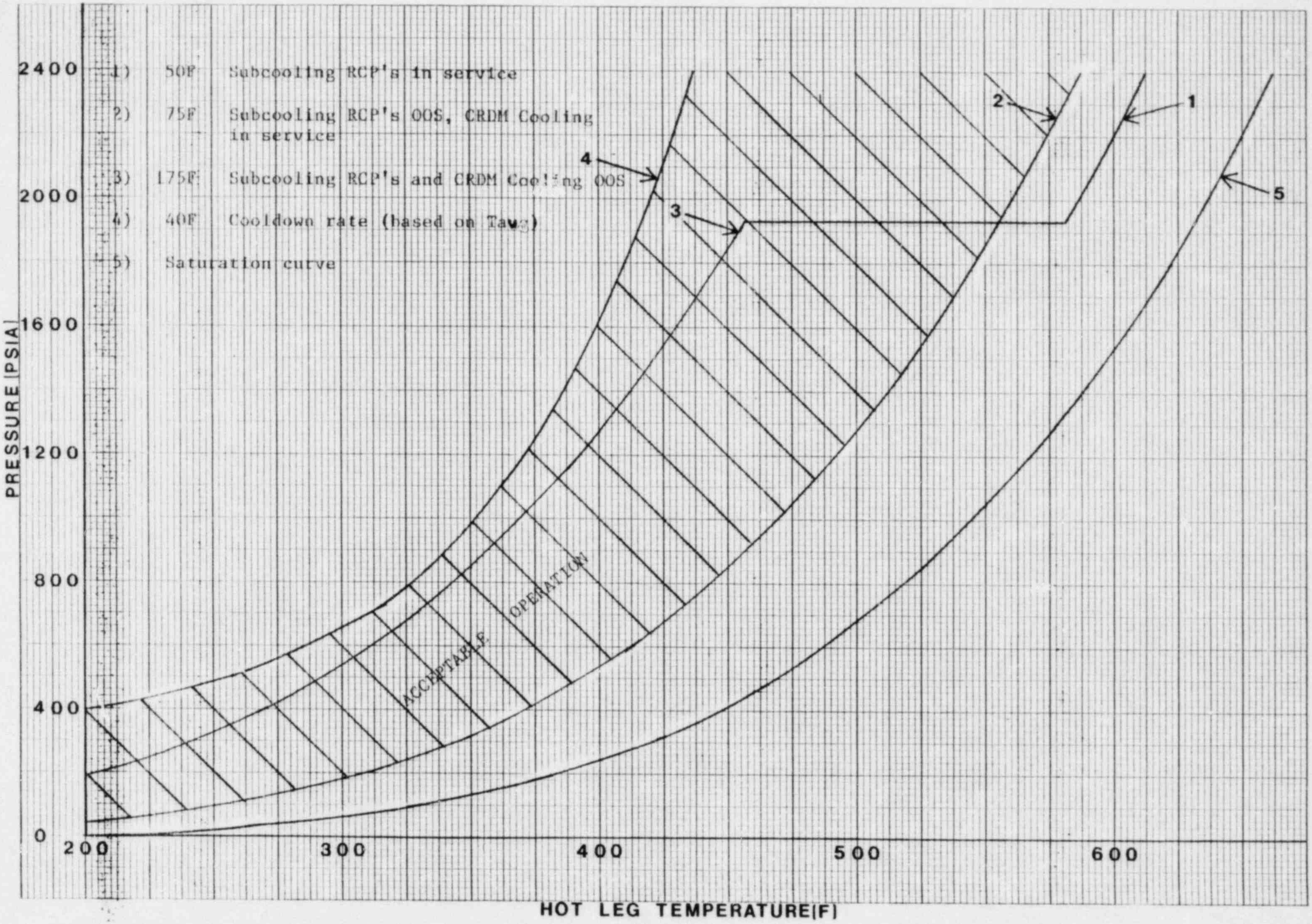
The second type of failure considered is two-wire power circuit (direct current). For the failure of such cables to activate equipment, either of the following would have to occur:

1. The cables would have to be severed and then two wires of one connected to two wires of the other. It would also require that one of the cables be connected to a power source and the other to a piece of equipment, or

2. The cable to the equipment in question would have to fail such that one wire goes to ground and the other hot shorts tc another DC wire of the same polarity. If a ground exists on the battery of the opposite polarity (the grounded leg of the equipment), then the equipment would be activated.

While these sequences are not as unlikely as in the case of three-phase wires, they still are considered improbable and not necessary for evaluating the ability to maintain hot shutdown conditions. Therefore, steps to be taken to assure that long term flow paths are not interrupted by any postulated failure, however incredible, will be taken after stable hot shutdown parameters are established.

Procedure steps will be taken to preclude failures on control circuits from changing the status of operating equipment that is necessary for shutdown, ie, control circuit power removal.


K+E 10 X 10 TO ½ INCH 7 X 10 INCHES
KEUFFEL & ESSER CO. MADE IN U.S.A.
1) 50F Subcooling RCP's in service
2) 75F Subcooling RCP's OOS, CRDM Cooling in service
3) 175F Subcooling RCP's and CRDM Cooling OOS
4) 40F Cooldown rate (based on Tavg)
5) Saturation curve
ACCEPTABLE OPERATION
1
2
3
4
5
PRESSURE [PSIA]
2400
2000
1600
1200
800
400
0
200
300
400
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600
HOT LEG TEMPERATURE(F)