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RS-10-208

10 CFR 50.90

December 13, 2010

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555-001

Dresden Nuclear Power Station, Units 2 and 3
Renewed Facility Operating License Nos. DPR-19 and DPR-25
NRC Docket Nos. 50-237 and 50-249

Subject Additional Information Supporting the Request for License Amendment Regarding Shutdown Cooling System Isolation Instrumentation

References:

1. Letter from Mr. Jeffrey L. Hansen (Exelon Generation Company, LLC) to U. S. NRC, "Request for License Amendment Regarding Shutdown Cooling System Isolation Instrumentation," dated February 4, 2010

2. Letter from U. S. NRC to Mr. Michael J. Pacilio (Exelon Nuclear), "Dresden Nuclear Power Station, Units 2 and 3 – Request for Additional Information Related to a Modification That Replaces the Temperature-Based Isolation Instrumentation with Reactor Pressure-Based Isolation Instrumentation (TAC Nos. ME3354 and ME3355)," dated September 3, 2010

3. Letter from Mr. Jeffrey L. Hansen (Exelon Generation Company, LLC) to U. S. NRC, "Additional Information Supporting the Request for License Amendment Regarding Shutdown Cooling System Isolation Instrumentation," dated September 15, 2010

4. Letter from Mr. Jeffrey L. Hansen (Exelon Generation Company, LLC) to U. S. NRC, "Follow-up Information Supporting the Request for License Amendment Regarding Shutdown Cooling System Isolation Instrumentation," dated October 6, 2010

5. Letter from U. S. NRC to Mr. Michael J. Pacilio (Exelon Nuclear), "Dresden Nuclear Power Station, Units 2 and 3 – Request for Additional Information Related to a Modification that Replaces the Temperature-Based Isolation Instrumentation (TAC Nos. ME3354 and ME3355)," dated November 23, 2010

In Reference 1, Exelon Generation Company, LLC (EGC) requested an amendment to Renewed Facility Operating License Nos. DPR-19 and DPR-25 for Dresden Nuclear Power Station (DNPS), Units 2 and 3, respectively. Specifically, the proposed amendment revises Technical Specification (TS) 3.3.6.1, "Primary Containment Isolation Instrumentation," Table 3.3.6.1-1, "Primary Containment Isolation Instrumentation," Function 6.a, "Shutdown Cooling System Isolation, Recirculation Line Water Temperature - High," to enable implementation of a modification that replaces the temperature-based isolation instrumentation with reactor pressure-based isolation instrumentation. The proposed modification will address instrumentation reliability problems that have led to interruptions of Shutdown Cooling (SDC) System operation. The proposed change to Primary Containment Isolation System (PCIS) instrumentation Function 6.a is needed to ensure reliable heat removal capability, avert plant transients and challenges to equipment, and minimize unnecessary operator actions during plant shutdowns.

In Reference 2, the NRC forwarded requests for additional information (RAIs) concerning the Reference 1 license amendment request. EGC provided the information requested by the NRC in Reference 3. During a conference call between the NRC and EGC following submittal of the responses to the NRC RAIs, additional follow-up questions were asked by the NRC reviewer to provide clarification of a number of the EGC responses. EGC agreed to provide this follow-up information and the requested information was provided in Reference 4.

Subsequent to Reference 4, the NRC provided EGC with an additional set of RAIs in Reference 5. The response to this request is provided in Attachment 1 to this letter. Additional supporting documentation is provided in Attachments 2 through 5 to this letter.

EGC has reviewed the information supporting a finding of no significant hazards consideration that was provided to the NRC in Reference 1. The additional information provided in this submittal does not affect the bases for concluding that the proposed license amendment does not involve a significant hazards consideration. No new regulatory commitments are established by this submittal.

If you have any questions concerning this letter, please contact Mr. Timothy A. Byam at (630) 657-2804.

I declare under penalty of perjury that the foregoing is true and correct. Executed on the 13th day of December 2010.

Respectfully,

Jeffrey L. Hansen
Manager – Licensing
Exelon Generation Company, LLC

Attachments:

1. Additional Information Supporting the Request for License Amendment Regarding Shutdown Cooling System Isolation Instrumentation
2. Procedure DOA 1000-01, "Residual Heat Removal Alternatives"
3. Procedure DOP 1000-07, "Alternate Shutdown Cooling"
4. Procedure OU-DR-104, "Shutdown Safety Management Program"
5. Procedure OP-DR-104-1001, "Shutdown Risk Management Contingency Plans"

ATTACHMENT 1

Additional Information Supporting the Request for License Amendment Regarding Shutdown Cooling System Isolation Instrumentation

ATTACHMENT 1
Additional Information Supporting the Request for License Amendment Regarding Shutdown Cooling System Isolation Instrumentation

The NRC provided Exelon Generation Company, LLC (EGC) with the following Request for Additional Information (RAI) associated with the Dresden Nuclear Power Station (DNPS), Units 2 and 3, amendment request for revision of Technical Specifications (TS) 3.3.6.1, "Primary Containment Isolation Instrumentation," Table 3.3.6.1-1, "Primary Containment Isolation Instrumentation," Function 6.a, "Shutdown Cooling System Isolation, Recirculation Line Water Temperature – High," to enable implementation of a modification that replaces the temperature-based isolation instrumentation with reactor pressure-based isolation instrumentation. The RAI and the requested information are provided below.

NRC Request:

In reviewing the Exelon Generation Company's (Exelon's) submittal dated October 6, 2010, related to a modification that replaces the temperature-based isolation instrumentation, for the Dresden Nuclear Power Station, Units 2 and 3, the Nuclear Regulatory Commission staff has determined that the following information is needed in order to complete its review:

1. In a letter dated October 6, 2010, Clarification 4 indicates that during a total loss of shutdown cooling (SDC), various alternate core cooling (ACC) methods are available for decay heat removal (DHR) and reactor coolant system (RCS) inventory control. The methods indicated included the condensate/feed and main steam (MS) system, the reactor water cleanup system, control rod drive system and the emergency core cooling systems [including the isolation condenser, high pressure coolant injection, MS relief valves with the suppression pool cooling mode of the low pressure coolant injection system].

Discuss use of the above ACC methods for DHR and RCS inventory control during a total loss of SDC under the following plant conditions:

(1) The reactor pressure vessel (RPV) head is tensioned;
(2) The RPV is detensioned; and
(3) The RV head is removed and the MS line plugs are put in place.

The discussion should address the availability and adequacy of operating procedures to provide clear guidance to the operator for applying the methods.

EGC Response:

NUREG-1449, "Shutdown and Low-Power Operation at Commercial Nuclear Power Plants in the United States," contains the results of the NRC Staff's evaluation of shutdown and low-power operations at commercial nuclear power plants in the United States. The report describes studies conducted by the NRC as well as evaluations of a number of technical issues associated with shutdown and low-power operations. One area addressed is the issue of loss of Residual Heat Removal (RHR) capability (see Section 6.6 of Reference 1). This section states that if RHR is lost in a BWR, "operators can usually significantly extend the time available for recovery of the system by adding water to the core from several sources, including condensate system, low-pressure coolant injection (LPCI) system, core spray (CS) system, and control rod drive (CRD) system." This section of Reference 1 goes on to state that "[i]n the event that RHR cannot be recovered in the short term, alternate RHR methods covered by procedures are normally available. The particular method selected will depend on the plant configuration and the decay heat load." The three plant configurations evaluated in NUREG-1449 include when the reactor vessel head is tensioned, the reactor vessel head is detensioned, and the reactor vessel head is removed and the main steamline plugs are in place.

DNPS abnormal operating procedure DOA 1000-01, "Residual Heat Removal Alternatives," (Reference 2) provides the alternatives available to shutdown a DNPS unit and maintain the reactor in cold or hot shutdown condition based on the availability of specific systems and reactor temperature and pressure. In addition, DNPS system operating procedure DOP 1000-07, "Alternate Shutdown Cooling," (Reference 3) provides an alternate means to remove decay heat from the Reactor when the SDC or support systems are unavailable and other means of maintaining reactor coolant temperature below 212°F are inadequate.

Dresden Procedure DOA 1000-01 is an abnormal operating procedure which is intended to describe the actions to be taken during a system transient that required operator actions to protect personnel, equipment, or to avoid a plant transient that could violate a Technical Specification limit. Entry into this procedure is based on specific abnormal symptoms. The procedure describes the automatic actions that take place for those symptoms and then describes the subsequent operator actions to be taken. Since the abnormal operating procedure identifies automatic as well as subsequent operator actions for a given symptom, the operators are trained in the use of these procedures. This training includes use in simulator exercises. The operators are required to demonstrate the ability to respond appropriately to any given symptom and associated transient.

NUREG-1449 states in Section 6.6 that "[i]f the RV head is tensioned, the reactor pressure vessel (RPV) is first allowed to pressurize and then steam is dumped to the suppression pool via a safety-relief valve (SRV), and makeup water is provided by one of the water sources listed above." The direction to take when there is a loss of SDC is contained in Reference 2, whether the reactor vessel head is tensioned or detensioned. With the head tensioned, Step D.5.b addresses the use of the Main Steam Turbine Bypass valves to remove heat by releasing steam to the condenser and maintaining reactor water level using Feedwater and Condensate Systems. This procedure also provides guidance (i.e., steps D.5 and D.6 of Reference 2) on the use of Reactor Water Cleanup (RWCU) System, CRD System, Main Steamline Drain Valves, and use of Unit House Loads in addition to use of other systems such as the Isolation Condenser System, High Pressure Coolant Injection (HPCI) System and the Electromagnetic Relief Valves (ERVs). The ERVs at DNPS serve the same function as the SRVs, as described in NUREG-1449, when it comes to RPV depressurization (i.e., they dump steam from the RPV to the torus). The operator is trained to use one or more of the specified decay heat removal alternatives, as directed by the Unit Supervisor, to control the reactor water temperature and pressure. If these methods do not decrease and maintain coolant temperature below 212°F then the operator is directed to perform procedure DOP 1000-07 (i.e., Reference 3).

Reference 1 also describes the alternate RHR methods available for the condition where "the RPV head is removed and main steamline plugs are put in place." If the reactor is in Mode 5 with the steamline plugs in place and the Reactor Cavity flooded, Reference 2 directs the operator to go to Step D.7. Reference 2, Step D.7 states that "in Mode 5 with the Reactor Cavity flooded, then use one or more of the following Decay Heat removal alternatives as directed by the Unit Supervisor to control reactor water temperature." These alternatives include the RWCU system, cross cooling from the Fuel Pool Cooling or SDC in the Fuel Pool Cooling Mode, and realign SDC in Fuel Pool Cooling Mode back to the Reactor Cavity.

As described above, DNPS has various alternate core cooling methods available for decay heat removal and reactor coolant inventory control in the event of a loss of SDC. The operators are provided direction for alternate core cooling methods by DNPS procedures that provide clear guidance in the use of these methods for the conditions existing with the RPV head tensioned, the RPV detensioned and the RPV head removed with the main steamline plugs in place.

NRC Request:

2. *Provide a discussion of plant administration controls, programs or procedures which ensure that the equipment (pumps, valves, and instrumentation) needed for the ACC methods is operable.*

EGC Response:

EGC utilizes several procedures to support the decay heat removal function at DNPS Units 2 and 3. In addition to References 2 and 3 described above, DNPS procedure OU-DR-104, "Shutdown Safety Management Program," (Reference 4) defines the key safety functions for DNPS and applies to the planning, scheduling, and execution of work on a unit already in or expected to be in a shutdown mode of operation. This is the site specific procedure that implements the corporate shutdown safety management program. DNPS Procedure OP-DR-104-1001, "Shutdown Risk Management Contingency Plans," (Reference 5) provides the operators with heightened awareness of plant status during outages and ensures that proper contingency plans are in place to reduce shutdown risk.

One of the safety functions addressed in Reference 4 is Decay Heat Removal. Section 4.5.1 provides guidelines for maintaining the reactor decay heat removal key safety function operable. This section specifically states that "[c]ontingency plans should be in place if activities that potentially impact decay heat removal systems must be scheduled during periods of Short Time to boil or reduced inventory." This procedure identifies the primary and alternate sources of shutdown cooling for DNPS. These sources are consistent with the guidance provided in References 2 and 3. Reference 4 drives the station to ensure that if the primary source of SDC is not available for a given plant condition, then an alternate source is maintained available to ensure shutdown safety. Finally, Section 4.5.1.6 of Reference 4 requires that at the beginning of each shift, when applicable, operators are designated and briefed to restore decay heat removal equipment. The briefing includes the procedures and recovery actions, current conditions (i.e., time to boil, core uncovery time, and available equipment), prioritizing the available alternate cooling methods to be employed for the current conditions, and actions needed to restore secondary containment if breached.

Reference 5 Section 4.1 describes the contingency plans to address a loss of decay heat removal. This procedure directs the use of DOA 1000-01 (Reference 2) first and then provides additional guidance to minimize the consequences of a loss of decay heat removal. The procedure provides guidance for controlling containment integrity, use of fuel pool cooling for decay heat removal, actions to take when using the main steam drain lines, and the use of feed and bleed of reactor coolant in Modes 4 and 5.

DNPS TS 3.4.7, "Shutdown Cooling (SDC) System – Hot Shutdown," requires two SDC subsystems to be operable and with no recirculation pump in operation, at least one SDC subsystem shall be in operation in Mode 3 (i.e., Hot Shutdown). In the event that one or two required SDC subsystems are inoperable in Mode 3, TS 3.4.7 requires the operator to initiate actions to restore the required SDC subsystem to operable status and to verify an alternate method of decay heat removal is available for each inoperable required SDC subsystem within one hour. TS Surveillance Requirement (SR) 3.4.7.1 requires verification that one SDC subsystem or recirculation pump is operating every 12 hours when reactor vessel coolant temperature is less than the SDC cut-in permissive temperature. TS 3.4.8, "Shutdown Cooling (SDC) System – Cold Shutdown," contains similar requirements and actions for SDC system operability in Mode 4 (i.e., Cold Shutdown). TS 3.9.8, "Shutdown Cooling (SDC) – High Water Level," and TS 3.9.9, "Shutdown Cooling (SDC) – Low Water Level," address the requirements for SDC subsystem operability in Mode 5 with irradiated fuel in the reactor pressure vessel and water level either greater than or equal to 23 feet above the RPV flange or less than 23 feet above the RPV flange, respectively. These TSs require verifying an alternate method of decay heat removal is available within one hour of the determination that the required number of SDC subsystems are inoperable.

DNPS TS 3.4.7 (Mode 3), 3.4.8 (Mode 4), and 3.9.8 and 3.9.9 (Mode 5) in conjunction with procedures OU-DR-104 and OP-DR-104-1001 ensure that in the event SDC is lost alternate core cooling equipment is available and operable to support the decay heat removal function. Reference 4 specifically states that the "planned removal of Shutdown Cooling Loops from service should not be scheduled during Modes 3, 4, and 5 unless absolutely necessary, to ensure maximum redundancy of the Decay Heat Removal System."

The above guidance is currently provided in plant procedures and the operators are trained in the use of these procedures. These procedures have been in place for an extended period of time and they have been used as necessary during shutdown of DNPS Units 2 and 3.

In summary, DNPS maintains procedural controls during outages to maintain a minimum set of decay heat removal components with alternate methods covered by procedures and the operators are trained to use those alternate sources to cool the core including condensate system, LPCI system, CS system, and CRD system.

References:

1. NUREG-1449, "Shutdown and Low-Power Operation at Commercial Nuclear Power Plants in the United States," dated September 1993

2. Procedure DOA 1000-01, "Residual Heat removal Alternatives," Revision 28

3. Procedure DOP 1000-07, "Alternative Shutdown Cooling," Revision 1

4. Procedure OU-DR-104, "Shutdown Safety Management Program," Revision 12

5. Procedure OP-DR-104-1001, "Shutdown Risk Management Contingency Plans," Revision 6

ATTACHMENT 2

Procedure DOA 1000-01, "Residual Heat Removal Alternatives"

CATEGORY 1

RESIDUAL HEAT REMOVAL ALTERNATIVES

REQUIREMENTS:

NONE.

INDEPENDENT TECHNICAL REVIEW

Disciplines Required:	NPPT	RO	RE/QNE	CH	RS	I&C	M&ES
	[X]	[X]	[ ]	[ ]	[ ]	[ ]	[X]

Unit 1 Review Required: [ ] YES [X] NO

Special Reviews: DEOP Coordinator.

PLANT OPERATIONS REVIEW COMMITTEE (PORC):

PORC REQUIRED: [X] YES* [ ] NO

* PORC required for changes to actions impacting jumper/bypass installation or removal

APPROVAL AUTHORITY:

Station Manager (SM), or designee (PORC Required)
Shift Operations Superintendent (SOS), or designee (PORC NOT required)

POST PERFORMANCE REVIEWS:

NONE.

RESIDUAL HEAT REMOVAL ALTERNATIVES

A. SYMPTOMS:

1. Shutdown Cooling (SDC) Heat Exchanger inlet high temperature alarm (350°F) on Panel 902(3)-4.

2. Reactor Water Cleanup (RWCU) System area temperature alarm on the Area Leak Detection System.

3. SDC Pump Trip alarm on Panel 902(3)-4.

4. Undesirable change in reactor water temperature cooldown rate as shown on the recirculation water temperature recorder or process computer.

5. Unexpected rise in reactor pressure.

6. Unexpected rise in reactor vessel metal temperature.

7. Normal reactor cooldown methods are NOT available.

B. AUTOMATIC ACTIONS:

1. SDC Pump(s) will trip under one of the following conditions:

   a. SDC Pump suction temperature ≥ 350°F (setpoint 339°F rising).

   b. SDC Pump suction pressure ≤ 4 psig for 7.5 seconds ± 2.5 seconds.

2. IF Reactor Recirculation loop temperature rises above 350°F (Analytical Limit) setpoint 339°F OR IF Reactor Water level drops below zero (0) inches, (setpoint + 6.02 inches), THEN the following SDC System valves will automatically close:

   a. MO 2(3)-1001-1A, INLET ISOL VLV.

   b. MO 2(3)-1001-1B, INLET ISOL VLV.

   c. MO 2(3)-1001-2A, 2(3)A PP SUCT VLV.

   d. MO 2(3)-1001-2B, 2(3)B PP SUCT VLV.

   e. MO 2(3)-1001-2C, 2(3)C PP SUCT VLV.

   f. MO 2(3)-1001-4A, 2(3)A PP DISCH VLV.

   g. MO 2(3)-1001-4B, 2(3)B PP DISCH VLV.

   h. MO 2(3)-1001-4C, 2(3)C PP DISCH VLV.

   i. MO 2(3)-1001-5A, OUTLET ISOL VLV.

   j. MO 2(3)-1001-5B, OUTLET ISOL VLV.

C. IMMEDIATE OPERATOR ACTIONS:

NONE.

D. SUBSEQUENT OPERATOR ACTIONS:

NOTE

1. IF SDC is lost (partially or completely) AND operation is required (Mode 3, 4 or 5), THEN evaluate the appropriate Tech Spec LCO AND perform the REQUIRED ACTIONS:
   - Section 3.4.7 (Mode 3)
   - Section 3.4.8 (Mode 4)
   - Section 3.6.1.1 (if loss of SDC causes the Unit to enter Mode 3)
   - Section 3.9.8 (Mode 5)
   - Section 3.9.9 (Mode 5)
2. The Tech Spec requirements (listed above) are applicable throughout performance of this procedure
3. This procedure can be used to satisfy REQUIRED ACTIONS of LCO for Tech Specs 3.4.7, 3.4.8, 3.9.8 or 3.9.9. WHEN choosing the heat removal alternative, THEN consideration should be given to the alternative's heat removal capability (i.e. one loop of SDC will remove approximately 8 MWth, while RWCU will remove approximately 10 MWth).

1. Verify Pressure/Temperature monitoring requirements per Tech Spec 3.4.9 have been initiated.
2. IF Mode 4 can NOT be established/maintained, OR an uncontrolled RCS temperature increase approaching 212°F occurs, THEN review EP-AA-111, Emergency Classification and Protective Action Recommendations.
   - IF conditions of an Emergency Action Level are met, THEN declare Emergency Classification Level AND implement required notifications per EP-AA-114, Notifications.
3. Initiate actions to secure any temporary openings in secondary containment per DAP 07-44. (W-12)

D. 4. IF in Mode 5 with the Reactor Cavity flooded, THEN go to Step D.7, OTHERWISE continue with Step D.5.

5. Use one or more of the following Decay Heat Removal alternatives as directed by the Unit Supervisor to control reactor water temperature/pressure:

a. SDC System if available. (Each SDC loop will remove 8 MWth.)

(1) Place an additional SDC loop in service.

(2) Raise RBCCW flow rate by throttling open MO 2(3)-3704, RBCCW OUTLET VLV.

(3) Raise SDC flow rate by throttling open MO 2(3)-1001-4A(B), 2(3)A(B) PP DISCH VLV.

(4) Start additional SDC Pump to raise circulation in the vessel.

NOTE

1. IF Recirc loop temperature is above the isolation setpoint, THEN the red "TRIP" light will be on.

2. IF Recirc loop temperature is below the isolation reset (≤ 333°F), THEN the red "trip" light on signifies that the temperature element has failed.

(5) IF SDC isolated due to two OR more failed Reactor Recirculation loop temperature element as indicated on any of the following, THEN perform Attachment A, Install/Remove 350°F Recirc Temperature Isolation Bypass.

- 2(3)-260-13A in back of Panel 902(3)-18.
- 2(3)-260-13B in back of Panel 902(3)-18.
- 2(3)-260-8E at Panel 902(3)-21.
- 2(3)-260-8F at Panel 902(3)-21.

NOTE

1. The Turbine Bypass valves will automatically close when the Main Condenser vacuum drops below seven (7) inHg.

2. IF Steam Jet Air Ejector System is in service, THEN Turbine Bypass valves should be closed WHEN reactor water temperature is at approximately 350°F.

3. IF Mechanical Vacuum Pump is in service, THEN Turbine Bypass valves should be closed WHEN reactor water temperature is approximately 300°F.

4. At 300°F, reactor pressure will NOT be adequate to maintain Turbine Seal Steam System operation.

D. 5. b. Main Steam Turbine Bypass Valves. (Each Main Steam Turbine Bypass Valve will remove up to 112 MWth.)

(1) Maintain a maximum cooldown rate of 100°F in any one hour period using the Bypass Valve opening Jack per DGP 02-01 (Tech Spec Section 3.4.9).

(2) Maintain reactor water level using Feedwater and Condensate Systems (DGP 02-01).

(3) IF Steam Jet Air Ejector System is in service, THEN close the turbine bypass valve(s) WHEN reactor water temperature is approximately 350°F (approximately 125 psig).

(4) IF Steam Jet Air Ejector System is NOT in service, THEN close the turbine bypass valve(s) when reactor water temperature is approximately 300°F.

(5) WHEN reactor water temperature drops to 300°F, THEN open the condenser vacuum breaker using MO 2(3)-4901, TURB VACUUM BKR.

D. 5. c. RWCU System (approximately 10 MWth):

(1) Verify RWCU System is in service, OR place RWCU System in service per DOP 1200-01 OR DOP 1200-03, as applicable.

(2) Raise RWCU System flow rate to maximize the heat removal rate.

(3) IF additional heat removal from the vessel is necessary, THEN use RWCU System in blowdown mode per DOP 1200-02, while maintaining reactor water level with Feedwater, Condensate, OR CRD System.

d. Control Rod Drive System.

(1) Raise CRD cooling water flow rate to reduce reactor water temperature.

e. Main Steam Line Drain Valves.

(1) Open the following MSL Drain valves as necessary to reduce reactor pressure and temperature:

- MO 2(3)-220-1, U2(3) MN STM LINES INBD DRN VLV.
- MO 2(3)-220-2, U2(3) MN STM LINES OTBD DRN VLV.
- MO 2(3)-220-4, U2(3) MN STM LINES DRN TO CDSR SV.

f. Unit House Loads. (Unit house loads will remove approximately 126 MWth.)

(1) Maximize operation of the following systems as applicable:

- Steam Jet Air Ejector System.
- Gland Seal Steam System.
- Max Recycle Concentrator Reboiler.

D. 6. IF the above alternatives are NOT sufficient/available to control reactor water temperature/pressure, THEN use one or more of the following ECCS alternatives as directed by the Unit Supervisor to control reactor water temperature/pressure:

a. Isolation Condenser System. (Isolation Condenser System will remove up to 74 MWth.)

(1) Place Isolation Condenser System in service (DOP 1300-03).

b. High Pressure Coolant Injection (HPCI) System. (HPCI will remove up to 37 MWth.).

(1) IF reactor pressure is above 90 psig, THEN initiate HPCI System in pressure control mode (DOP 2300-03).

NOTE

The Suppression Pool water level should be above six (6) feet to ensure exhausted steam from the Electromatic Relief Valves is condensed in the Suppression Pool water.

c. Electromatic Relief Valves. (Each Electromatic Relief Valve will remove approximately 140 MWth.)

(1) Verify Suppression Pool water level is > 6 feet.

(2) Place LPCI in Suppression Pool cooling (DOP 1500-02).

(3) Open one or more Electromatic Relief Valve(s) as necessary to reduce reactor pressure/temperature, while maintaining the cooldown rate below 100°F/hr.

(4) Alternate opening of Electromatic Relief Valve(s) at five (5) minute intervals, in the following sequence to minimize local torus water heating when possible: A, C, E, D, B.

D. 6. c. (5) Monitor Suppression Pool water temperature using one or more of the following:

- Recorders TIRS 2(3)-1640-200A, SUPPRESSION POOL TEMP MONITOR, AND TIRS 2(3)-1640-200B, SUPPRESSION POOL TEMP MONITOR, on back Panel 902(3)-36 (will indicate the torus local and bulk water temperature).

- recorder TR 2(3)-1641-9, TORUS BULK TEMP, on PANEL 902(3)-3.

- Computer points T257/T258 (T357/T358) Suppression Pool water temperature.

(6) IF Suppression Pool water temperature exceeds 95°F, THEN enter the following procedures:

- DEOP 0200-01, Primary Containment Control.

- DOS 1600-20, Suppression Pool Temperature Monitoring.

d. IF all other attempts to maintain coolant temperature < 212°F have failed, THEN as directed by Unit Supervisor, perform DOP 1000-07, Alternate Shutdown Cooling.

D. 7. IF in Mode 5 with the Reactor Cavity flooded, THEN use one or more of the following Decay Heat removal alternatives as directed by the Unit Supervisor to control reactor water temperature:

a. IF required, THEN enter DOA 1900-01, Loss Of Fuel Pool Cooling, concurrently.

b. SDC System, if available.

(1) Place an additional SDC loop in service.

(2) Raise RBCCW flow rate by throttling open MO 2(3)-3704, RBCCW OUTLET VLV.

(3) Raise SDC flow rate by throttling open MO 2(3)-1001-4A(B)(C), 2(3)A(B)(C) PP DISCH VLV.

NOTE

1. IF Recirc loop temperature is above the isolation setpoint, THEN the red "TRIP" light will be on.

2. IF Recirc loop temperature is below the isolation reset (≤ 333°F), THEN the red "trip" light on signifies that the temperature element has failed.

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(4) IF SDC isolated due to two OR more failed Reactor Recirculation loop temperature element as indicated on any of the following, THEN perform Attachment A, Install/Remove 350°F Recirc Temperature Isolation Bypass.

- 2(3)-260-13A in back of Panel 902(3)-18.
- 2(3)-260-13B in back of Panel 902(3)-18.
- 2(3)-260-8E at Panel 902(3)-21.
- 2(3)-260-8F at Panel 902(3)-21.

D. 7. c. Reactor Water Cleanup (RWCU) System.

(1) Verify RWCU System is in service, OR place RWCU System in service (DOP 1200-01).

(a) Raise RWCU System flow rate to maximize the heat removal rate.

(2) Initiate an Action Request to bypass the Regenerative Heat Exchangers.

(3) Provide additional mixing from the Reactor Recirc system if available.

(4) IF additional heat removal from the vessel is necessary, THEN use RWCU System in blowdown mode per DOP 1200-02, while maintaining reactor water level with Condensate, CRD System, Condensate Transfer and Clean Demin water via hoses OR Control Cavity, Dryer/Separator Storage Pit and Fuel Pool Level per DOP 1900-03.

d. Cross Cooling from Fuel Pool Cooling or SDC in the Fuel Pool Cooling Mode.

(1) Using natural circulation with the Fuel Pool Gates removed.

(2) IF necessary, THEN remove Fuel Pool Gates (DFP 0800-06).

(3) To aid the natural circulation, forced flow between the Fuel Pool and Reactor Cavity can be added (DFP 0800-48).

CAUTION

Before realigning the following systems to the Reactor, consideration must be given to the Fuel Pool Decay heat Load and Temperature.

e. Realign SDC in Fuel Pool Cooling Mode back to the Reactor Cavity:

(1) Secure SDC to Fuel Pool Cooling.

(2) Close valves:

(a) 2(3)-1901-20, U2 FUEL POOL SKIMMER SURGE TK TO S/D CLG OUTLET VLV (U3 FUEL POOL SKIMMER SURGE TK TO SHUTDN CLG OUTLET SV).

(b) 2(3)-1901-64, U2(3) SHUTDN CLG SYS RETURN TO FUEL POOL SYS ISOL VLV.

D. 7. e. (3) Rack-in the breaker on the loop to be started:

(a) Loop 2A:

- 250 VDC MCC #2A, Cubicle E2, MO 2-1001-2A, U2 SDC PUMP 2A SUCTION MOV.

<u>AND</u>

- 250 VDC MCC #2A, Cubicle I1, MO 2-1001-4A, U2 SDC HX 2A OUTLET MOV.

(b) Loop 2B:

- 250 VDC MCC #2A, Cubicle F1, MO 2-1001-2B, U2 SDC PUMP 2B SUCTION MOV.

<u>AND</u>

- 250 VDC MCC #2A, Cubicle I2, MO 2-1001-4B, U2 SDC HX 2B OUTLET MOV.

(c) Loop 2C:

- 250 VDC MCC #2A, Cubicle F2, MO 2-1001-2C, U2 SDC PUMP 2C SUCTION MOV.

<u>AND</u>

- 250 VDC MCC #2A, Cubicle J2, MO 2-1001-4C, U2 SDC HX 2C OUTLET MOV.

(d) Loop 3A:

- 250 VDC MCC #3A, Cubicle E2, MO 3-1001-2A, U3 SDC PUMP 3A SUCTION MOV.

<u>AND</u>

- 250 VDC MCC #3A, Cubicle I1, MO 3-1001-4A, U3 SDC HX 3A OUTLET MOV.

(e) Loop 3B:

- 250 VDC MCC #3A, Cubicle F1, MO 3-1001-2B, U3 SDC PUMP 3B SUCTION MOV.

<u>AND</u>

- 250 VDC MCC #3A, Cubicle I2, MO 3-1001-4B, U3 SDC HX 3B OUTLET MOV.

(f) Loop 3C:

- 250 VDC MCC #3A, Cubicle F2, MO 3-1001-2C, U3 SDC PUMP 3C SUCTION HEADER MOV.

<u>AND</u>

- 250 VDC MCC #3A, Cubicle J2, MO 3-1001-4C, U3 SDC HX 3C OUTLET MOV.

D. 7. e. (4) Open MO 2(3)-1001-2A(B)(C), 2(3)A(B)(C) PP SUCT VLV.

(5) Start 2(3)A(B)(C) SDC Pump.

(6) Throttle open valve MO 2(3)-1001-4A(B)(C), 2A(B)(C) PP DISCH VLV, to the desired flow.

(7) Adjust RBCCW flow to the SDC Heat Exchanger as necessary by throttling MO 2(3)-3704, RBCCW OUTLET VLV.

(8) Place an Out-of-Service on valves 2(3)-1901-20, U2 FUEL POOL SKIMMER SURGE TK TO S/D CLG OUTLET VLV (U3 FUEL POOL SKIMMER SURGE TK TO SHUTDN CLG OUTLET SV), <u>AND</u> 2(3)-1901-64, U2(3) SHUTDN CLG SYS RETURN TO FUEL POOL SYS ISOL VLV, <u>AND</u> hang an Information Card on the Reactor Mode Switch to correct the valve alignment <u>PRIOR</u> to RPV Hydro, Mode 2, or Mode 3 operation.

f. Align Fuel Pool Cooling system to the Reactor Cavity using DOP 1900-01 Step G.8.

E. <u>USER REFERENCES</u>:

1. Technical Specifications:

a. Section 3.4.9, RCS Pressure and Temperature (P/T) Limits.

b. Section 3.4.7, Shutdown Cooling (SDC) System - Hot Shutdown.

c. Section 3.4.8, Shutdown Cooling (SDC) System - Cold Shutdown.

d. Section 3.6.1.1, Primary Containment.

e. Section 3.6.1.3, Primary Containment Isolation Valves (PCIVs).

f. Section 3.9.8, Shutdown Cooling (SDC) - High Water Level.

g. Section 3.9.9, Shutdown Cooling (SDC) - Low Water Level.

h. Section 3.3.6.1, Primary Containment Isolation Instrumentation.

2. Revised Updated Final Safety Analysis Report (RUFSAR):

a. RUFSAR Section 5.4.7, Reactor Shutdown Cooling System.

E. 3. Procedures:

a. CC-AA-112, Temporary Configuration Changes.

b. DAP 07-44, Control Of Temporary Openings In Secondary Containment During Performance Of Work Packages, Surveillances, Or Other Procedures.

c. DEOP 0200-01, Primary Containment Control.

d. DFP 0800-06, Spent Fuel Pool To Reactor Gate Removal And Installation.

e. DFP 0800-48, Spent Fuel Pool Cooling/Reactor Cavity Mixing To Support Decay Heat Removal.

f. DGP 02-01, Unit Shutdown.

g. DOA 1900-01, Loss Of Fuel Pool Cooling.

h. DOP 1000-03, Shutdown Cooling Mode of Operation.

i. DOP 1000-07, Alternate Shutdown Cooling.

j. DOP 1200-01, RWCU Operation During Startup and Shutdown.

k. DOP 1200-02, RWCU System Blowdown.

l. DOP 1200-03, RWCU System Operation with Reactor Plant at Pressure.

m. DOP 1300-03, Manual Operation of the Isolation Condenser.

n. DOP 1500-02, Torus Water Cooling Mode of Low Pressure Coolant Injection System.

o. DOP 1900-03, Reactor Cavity, Dryer Separator Storage Pit And Fuel Pool Level Control.

p. DOP 2300-03, High Pressure Coolant Injection System Manual Startup and Operation.

q. DOS 1600-20, Suppression Pool Temperature Monitoring.

r. EP-AA-111, Emergency Classification and Protective Action Recommendations.

s. EP-AA-114, Notifications.

t. HU-AA-101, Human Performance Tools and Verification Practices.

E. 4. Prints:

a. M-12 (M-345), Main Steam Piping.

b. M-20 (M-353), Reactor Building Cooling Water Piping.

c. M-26 (M-357), Nuclear Boiler & Reactor Recirculation Piping.

d. M-30 (M-361), Reactor Water Cleanup Piping, Sheet 1.

e. M-32 (M-363), Shutdown Reactor Cooling Piping.

f. M-39 (M-369), Reactor Building Equipment Drains.

g. M-48 (M-371), Reactor Water Cleanup Piping, Sheet 2.

h. 12E-2516 (12E-3516), Relaying & Metering & Schematic Diagram, Reactor Shutdown Cooling System, 4160 V Pumps 1002A, 1002B & 1002C.

i. 12E-2517 (12E-3517), Schematic Control Diagram, Reactor Shutdown Cooling System Pumps Control Circuits.

j. 12E-2508 (12E-3508), Schematic Diagram, Primary Containment Isolation System, Shutdown Cooling Isolation Logic, Sheet 8.

k. 12E-2491 (12E-3491), Schematic Diagram, Recirculation System Process Instrumentation Pt.2.

l. 12E-2502A (12E-3502A), Schematic Diagram, Primary Containment Isolation System, Reset Circuit.

F. DISCUSSION:

1. The heat generated in the reactor following reactor scram or reactor shutdown is composed of sensible heat and decay heat. The sensible heat is the energy associated with the elevated temperature of the reactor vessel and the internal components. The decay heat is released as the fission products decay.

During normal operating conditions, heat is lost from the reactor vessel through ambient heat losses, unit house loads and via the RWCU System Non-Regenerative Heat Exchangers.

F. 2. This procedure provides the alternatives available to shutdown the unit and maintain the reactor in cold or hot shutdown condition. Based on the availability of the following systems and reactor pressure/temperature, one or more of these alternatives may be utilized at the discretion of the Unit Supervisor.

a. Main Turbine Bypass Valves to release the steam from the reactor to the condenser and therefore direct the waste heat to the river via the Circulating Water System.

b. SDC System to provide heat removal mechanisms to bring the reactor to a cold (or Hot) shutdown condition. SDC system efficiently directs this waste heat to the river via RBCCW and Service Water Systems.

c. In the event that additional residual heat removal mechanisms are required, the Unit Supervisor should prioritize the methods to be used based on their impact on the environment.

d. The use of the Emergency Core Cooling Systems (ECCS) should always be a last resort. Therefore, priority should be given to raising the heat removal rate of the processes that are already in service. This would include:

(1) Raising the unit house load demands.

(2) Raising the SDC System flow rate.

(3) Raising Reactor Building Closed Cooling Water flow rate to the SDC System Heat Exchanger(s).

(4) Using RWCU System for a feed and bleed process.

(5) Raising CRD System cooling water flow rate.

(6) Placing Isolation Condenser System in service.

(7) Placing HPCI System in the pressure control mode operation.

(8) Opening the Electromatic Relief Valve(s).

F. 3. When the reactor cavity is flooded and the Fuel Pool Gates removed there the other systems become available for dealing with a loss of Decay Heat removal systems. Fuel Pool Cooling can be aligned to the Reactor Cavity as addressed in the UFSAR. Forced circulation can be used to help in using Fuel Pool Cooling or SDC in the Fuel Pool Cooling mode. Cross Cooling has been utilized to cool the Fuel Pool under SP 95-07-84 during D2R14 using forced circulation. Use of an Alternate Decay Heat Removal system per corporate engineering calculation and Calculation DRE 98-016, were we can cool the Reactor Cavity via natural circulation utilizing one Fuel Pool Cooling train and one SDC Train in the Fuel Pool Cooling mode.

W. WRITER'S REFERENCES:

1. INPO SOER 85-4, Loss or Degradation of Residual Heat Removal Capability in PWRs.

2. Response to INPO SOER 85-4, E.D. Eenigenburg to J. Leider, dated September 22, 1987.

3. Vendor Manual, GEK 786, Chapter 17, Shutdown Reactor Cooling System and Reactor Head Cooling System.

4. SP 88-5-66, SDC Pump Flow Verification and Check.

5. GE SIL No. 406, In-Core Instrumentation Protection.

6. INPO SOER 87-2, Inadvertent Draining of Reactor Water to Suppression Pool at BWRs.

7. W.B. Fancher letter to E.D. Eenigenburg dated June 2, 1988 and record of correspondence for letter from R. Magrow.

8. INPO SOER 82-02, Inadvertent Reactor Pressure Vessel Pressurization.

9. NRC Information Notice No. 87-50, Potential LOCA at High-Pressure and at Low-Pressure Interfaces from Fire Damage.

10. SP 95-07-84 Unit 2 Spent Fuel Pool Cooling Contingency Plan.

11. Calculation DRE 98-016, Determination of Spent Fuel Pool and Shutdown Cooling Heat Exchanger Inlet and Outlet Temperatures as Function of Decay Heat Load for Dresden Refuel Outage D3R15.

12. IGAP 959926-35-11, Revise DOA 1000-01, Residual Heat Removal Alternatives, to provide direction to secure secondary containment upon loss of decay heat removal per INPO SOER 09-1, Shutdown Safety.

ATTACHMENT A

INSTALL/REMOVE 350°F RECIRC TEMPERATURE ISOLATION BYPASS

A. Obtain Operations Shift Manager permission to install Jumper:

_______________/_______________________ __________
Print Name / Signature Date

CAUTION

Recirculation Line Water Temperature High trip channels for SDC System Isolation are required to be operable in Modes 1, 2, and 3 per Tech Spec Table 3.3.6.1-1 Function 6.a.

B. IF Tech Spec Table 3.3.6.1-1 Function 6.a, Recirculation Line Water Temperature - High, trip is required per Tech Spec 3.3.6.1, THEN enter the appropriate Tech Spec action Statement(s).

C. Bypass Reactor Recirculation loop temperature elements as follows:

NOTE

Jumper installation is performed using Concurrent Verification (CV) in accordance with HU-AA-101, Human Performance Tools and Verification Practices.

1. Bypass all sensors to the SDC High Temperature Isolation Logic by installing a jumper from 902(3)-4 LL-8 to 902(3)-4 LL-10

   Installed By: ______________________

   Verified By: ______________________

2. Reset the Group 3 isolation using the GROUP 2&3 ISOL RESET switch at Panel 903-5. ________

3. Restart SDC (DOP 1000-03). ________

4. Initiate an Issue Report to repair the failed temperature element(s). ________

ATTACHMENT A
INSTALL/REMOVE 350°F REACTOR RECIRC TEMPERATURE ISOLATION BYPASS

C. 5. Place a TCCP Tag on each of the jumpers.

6. Place the TCCP in the Unit Turnover.

7. Record this Procedure Number in the Control Room TCCP Log as the TCCP Number and complete TCCP Log entries. ________

8. Using Figure 1(2), 12E-2508 (12E-3508) Update, draw in the location of the installed jumpers in Unit 2(3). Record TCCP No on sketch. ________

9. Staple Figure 1(2) to Drawing 12E-2508 (12E-3508) in the Critical Control Room Drawing file. ________

10. Indicate the time frame during which the temporary change may remain installed under the authority of this procedure and actions required if extension is required (CC-AA-112 TCCP Extended Installation Justification Attachment) (for example, jumpers will be installed until ready to exit Mode 4)

____
____
____ ________

11. <u>IF</u> the TCCP must stay in effect after this procedure is completed, <u>THEN</u> transfer control of this TCCP to CC-AA-112. ________

ATTACHMENT A
INSTALL/REMOVE 350°F REACTOR RECIRC TEMPERATURE ISOLATION BYPASS

D. To remove the 350°F Reactor Recirc temperature isolation bypass, perform the following:

1. Obtain Operations Shift Manager permission to remove Jumper:

_______________/_______________________ __________
Print Name / Signature Date

> NOTE
>
> Jumper removal is performed using Concurrent Verification (CV) in accordance with HU-AA-101.

2. Remove jumper and TCCP Tag from 902(3)-4 LL-8 to 902(3)-4 LL-10

   Removed By: ________________________

   Verified By: ________________________

3. Remove the TCCP from the Unit Turnover. ________

4. Remove Figure 1(2) from Drawing 12E-2508 (12E-3508) in the Critical Control Room Drawing file. ________

5. Update the Control Room TCCP Log and Unit Turnover to indicate TCCP Removal. ________

6. Attach this completed Attachment to Appendix A for retention ________

CATEGORY 1

FIGURE 1
12E-2508 UPDATE

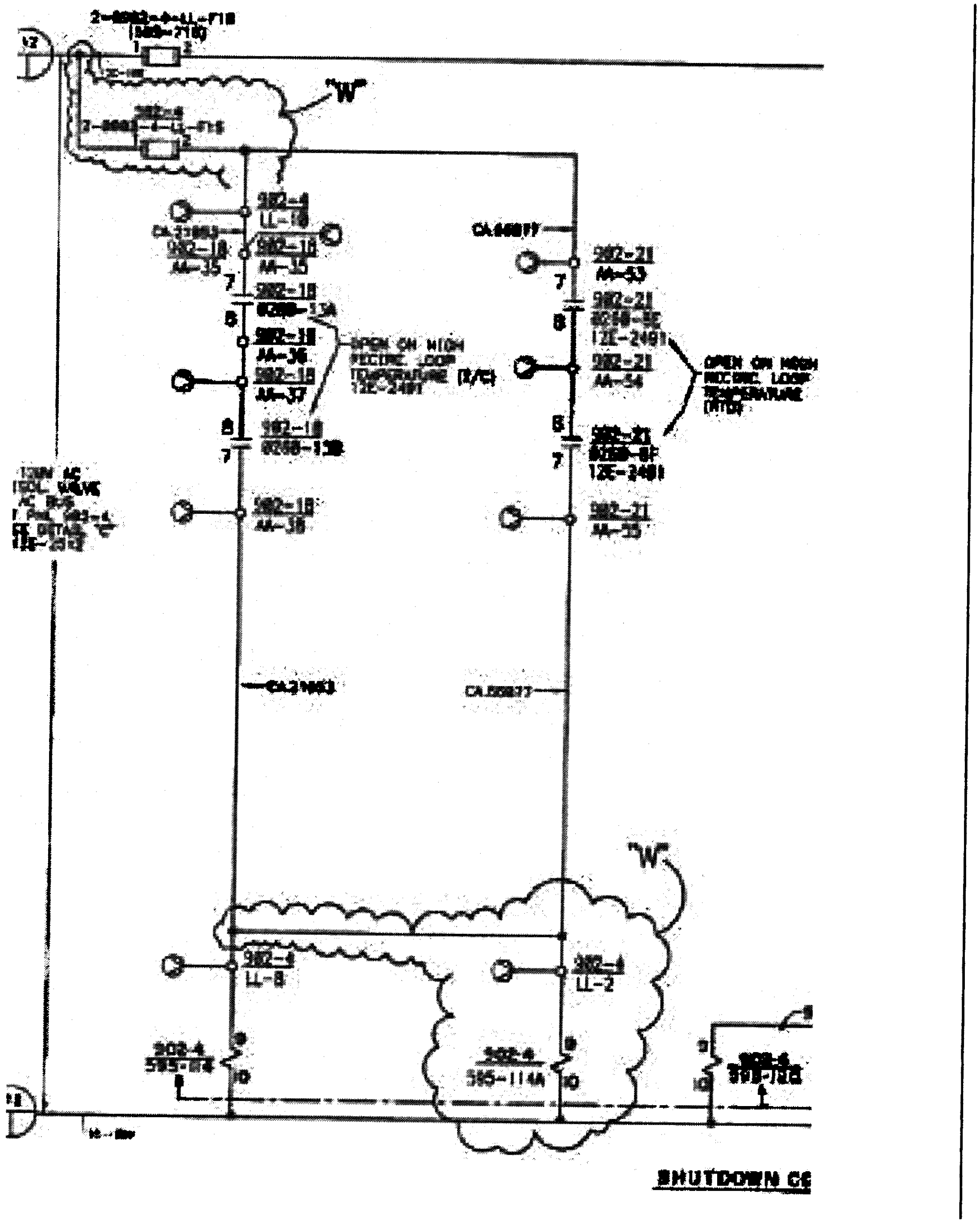
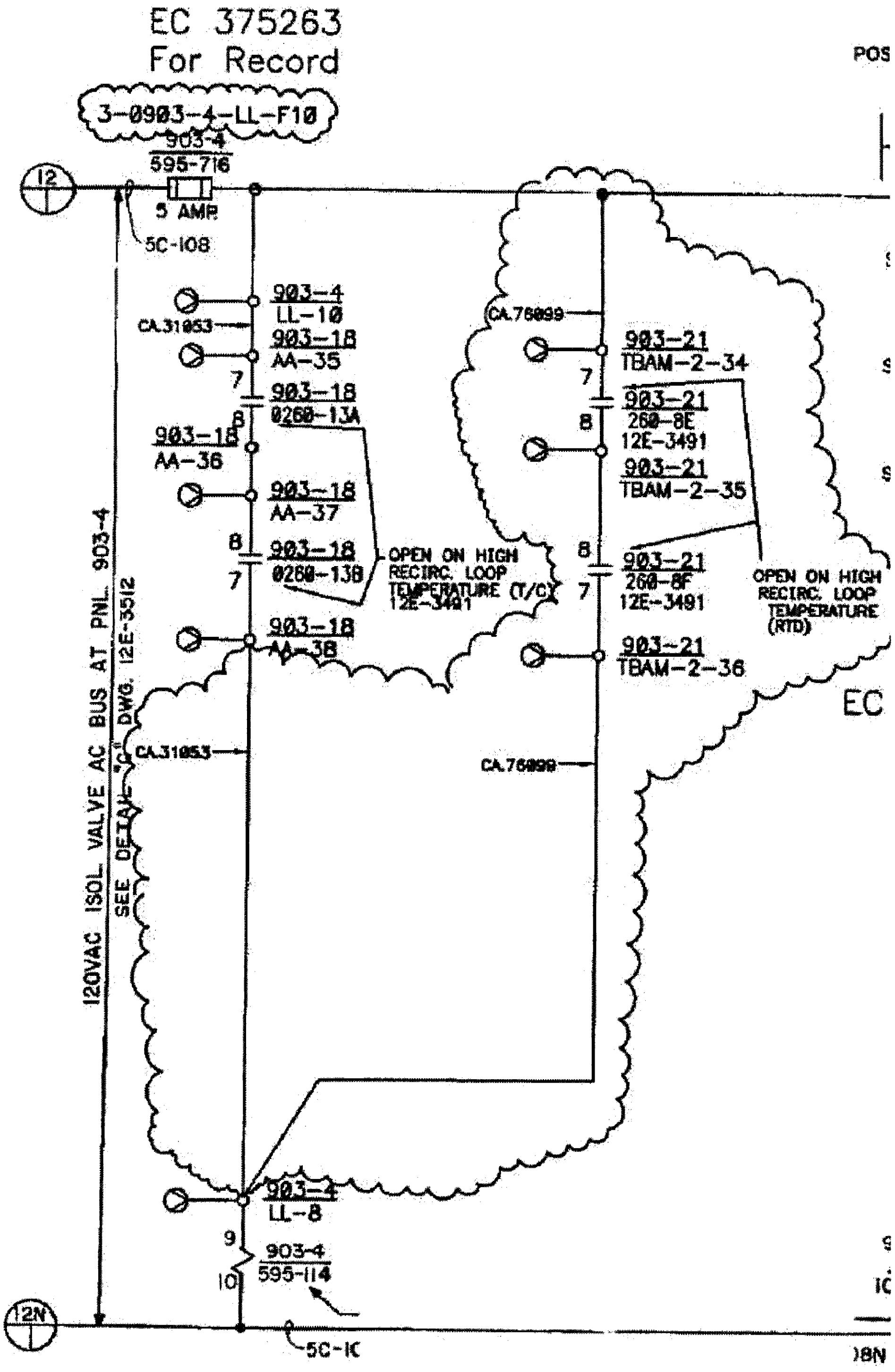


FIGURE 2
12E-3508 UPDATE



ATTACHMENT 3

Procedure DOP 1000-07, "Alternate Shutdown Cooling"

CATEGORY 1

UNIT 2(3)
DOP 1000-07
REVISION 01

ALTERNATE SHUTDOWN COOLING

REQUIREMENTS:

Technical Specifications.

INDEPENDENT TECHNICAL REVIEW:

Disciplines Required:	NPPT	RO	RE/QNE	CH	RS	I&C	M&ES
	[X]	[X]	[ ]	[ ]	[ ]	[ ]	[X]

Unit 1 Review Required: [ ] YES [X] NO

Special Reviews: NONE.

PLANT OPERATIONAL REVIEW COMMITTEE (PORC):

PORC REQUIRED [ ] YES [X] NO

APPROVAL AUTHORITY:

Shift Operations Superintendent (SOS), or designee

POST PERFORMANCE REVIEWS:

NONE.

ALTERNATE SHUTDOWN COOLING

A. PURPOSE:

This procedure provides an alternate means, Alternate Shutdown Cooling, to remove decay heat from the Reactor when the Shutdown Cooling (SDC) or support systems are unavailable AND other means of maintaining reactor coolant temperature below 212°F are inadequate.

B. USER REFERENCES:

1. Technical Specifications:

   a. Section 3.3.6.1, Primary Containment Isolation Instrumentation.

   b. Section 3.3.6.2, Secondary Containment Isolation Instrumentation.

   c. Section 3.4.7, SDC System - Hot Shutdown.

   d. Section 3.4.8, SDC System - Cold Shutdown.

   e. Section 3.4.9, RCS Pressure and Temperature (P/T) Limits.

   f. Section 3.5.2, ECCS - Shutdown.

   g. Section 3.9.8, SDC - High Water Level.

   h. Section 3.9.9, SDC - Low Water Level.

2. Technical Specification Surveillance Requirements:

   a. SR 3.5.2.1.a, ECCS - Shutdown.

3. Procedures:

   a. DOA 0201-04, Loss of Vessel Flange, Shell or Recirculation Loop Temperature Recorders During Heatup or Cooldown.

   b. DOA 1000-01, Residual Heat Removal Alternatives.

   c. DOP 1500-02, Torus Water Cooling Mode Of Low Pressure Coolant Injection System.

   d. DOP 3300-05, Condenser Level Control.

C. SUPPLEMENTS:

1. Attachment A, Reactor Cooldown Monitoring.

D. PREREQUISITES:

1. DOA 1000-01 has been executed AND is incapable of maintaining reactor coolant temperature < 212°F.

2. Torus intact AND at Panel 902-3, LI 2-1602-3, TORUS LVL, has been verified ≥ 10 feet 4 inches (Tech Spec 3.5.2, SR 3.5.2.1.a).

3. At least one (1) ADS valve is available for use.

4. Reactor pressure < 150 psig.

5. Reactor level as designated per Unit Supervisor for current plant conditions.

   SRO Peer Check (Initial/Date) __________/___________

6. At least one (1) LPCI AND two (2) CCSW pumps are available for reactor injection AND torus cooling.

7. SDC System is NOT in operation.

8. Personnel evacuated from the Primary Containment.

9. Unit is in Mode 3 OR 4.

E. PRECAUTIONS:

NONE.

F. LIMITATIONS AND ACTIONS:

1. Moderator temperature change shall NOT exceed 100°F averaged over one (1) hour.

2. Maintain reactor vessel metal temperatures to the right of the limit lines of Tech Spec Figure 3.4.9-2.

3. Prior to AND during the fill of the main steam lines, reactor pressure is required to be maintained < 45 psig.

4. Prior to raising reactor water level above the reactor flange, the differential temperature between the reactor coolant and the reactor head flange must be < 230°F.

5. Reactor coolant temperature must be > 166°F when water is first applied to the flange area. After the flange is covered then moderator temperature is allowed to drop below 166°F. This temperature band is only applicable during forced cooldown of the reactor flange.

6. Maintain temperature of reactor vessel flange and head flange ≥ 83°F when vessel head bolting studs are under tension.

G. PROCEDURE:

1. Monitor reactor vessel moderator AND metal temperatures. Log on Attachment A to ensure Tech Spec limits are met.

NOTE

Steps G.2 and G.3 may be performed concurrently if condensate is being used to fill the reactor vessel.

CAUTION

Do NOT raise level above the main steam lines (conservatively 100 inches indicated) UNTIL reactor pressure is < 45 psig.

2. IF available, THEN utilize condensate system to fill the vessel to below the main steam lines (conservatively 100 inches).

   - Monitor main condenser hotwell level AND makeup as necessary (DOP 3300-05).

3. Start maximum available torus cooling (DOP 1500-02) (approximately 10 kgpm LPCI flow through each test loop).

G. 4. Close the following valves:

- AO 2(3)-203-1A, MSIV.
- AO 2(3)-203-1B, MSIV.
- AO 2(3)-203-1C, MSIV.
- AO 2(3)-203-1D, MSIV.
- AO 2(3)-203-2A, MSIV.
- AO 2(3)-203-2B, MSIV.
- AO 2(3)-203-2C, MSIV.
- AO 2(3)-203-2D, MSIV.
- MO 2(3)-220-1, MSL DRN ISOL VLV.
- MO 2(3)-220-2, MSL DRN ISOL VLV.
- AO 2(3)-220-46, HEAD VENT VALVE.
- AO 2(3)-220-47, HEAD VENT VLV.
- MO 2(2)-1301-1, RX OUTLET ISOL.
- MO 2(3)-1301-2, RX OUTLET ISOL.
- MO 2(3)-2301-4, STEAM ISO VLV.
- MO 2(3)-2301-5, STEAM ISO VLV.

G. 5. IF injection will be routed through B LPCI loop, THEN:

a. Close MO 2(3)-1501-38B, TORUS CLG/TEST.

CAUTION
IF MO 2(3)-1501-38A is throttled open > 36 seconds, THEN the LPCI system SHALL be declared inoperable.

b. Throttle open MO 2(3)-1501-38A, TORUS CLG/TEST, UNTIL desired flow rate is obtained (approximately 5000 gpm per pump).

c. Close MO 2(3)-1501-21B, LPCI VLV.

d. Verify ONE of the following conditions:

(1) Reactor Recirculation pumps are in operation

OR

(2) The following valves are closed:

(a) MO 2(3)-0202-5A, 2(3)A PP DISCH VLV

(b) MO 2(3)-0202-5B, 2(3)B PP DISCH VLV

CAUTION
Do NOT exceed 100°F/hour cooldown rate.

e. Place ONE of the ERV keylock switches in MAN:

- 2(3)-0203-3B, 2(3)B ERV.
- 2(3)-0203-3C, 2(3)C ERV.
- 2(3)-0203-3D, 2(3)D ERV.
- 2(3)-0203-3E, 2(3)E ERV.

f. Open MO 2(3)-1501-22B, INJ VLV.

CAUTION

1. Do NOT exceed 5000 gpm per LPCI pump as indicated on FR 2(3)-1540-7, LPCI SYS TOTAL FLOW.
2. Do NOT exceed reactor pressure of 150 psig.
3. Do NOT cycle ADS valves when flooded.
4. Do NOT raise level above the main steam lines (conservatively 100 inches indicated) UNTIL reactor pressure is < 45 psig.

G. 5. g. Throttle open MO 2(3)-1501-21B, LPCI VLV, to achieve a slow increasing reactor level AND cooldown rate ≤ 100°F/hr.

h. Throttle closed MO 2(3)-1501-38A, TORUS CLG/TEST, to maintain LPCI pump flow < 5000 gpm per pump or to increase LPCI injection.

i. Control reactor cooldown rate ≤ 100°F/hr by controlling LPCI injection flow by:

- Throttling MO 2(3)-1501-21B, LPCI VLV.

AND/OR

- Throttling MO 2(3)-1501-38A, TORUS CLG/TEST.

j. Control Reactor pressure < 150 psig, but high enough to keep the ERV open by:

- Throttling MO 2(3)-1501-21B, LPCI VLV.

AND/OR

- Throttling MO 2(3)-1501-38A, TORUS CLG/TEST.

k. Control reactor level AND pressure when moderator temperature can be maintained between 180 and 200°F.

CAUTION

When raising level it is possible, because of off normal calibration conditions, to quench the reactor vessel shell without a corresponding decrease in vessel flange temperature.

l. IF reactor level approaches 185 inches indicated reactor level, THEN monitor reactor vessel shell temperature for signs of a sharp increase in shell cooldown rate.

G. 5. m. IF indicated reactor level reaches 190 to 210 inches, THEN stop level increase.

n. IF additional ERV flow is required, THEN open an additional ERV by placing ONE of the ERV keylock switches in MAN:

- 2(3)-0203-3B, 2(3)B ERV.
- 2(3)-0203-3C, 2(3)C ERV.
- 2(3)-0203-3D, 2(3)D ERV.
- 2(3)-0203-3E, 2(3)E ERV.

o. Control reactor cooldown rate ≤ 100°F/hr by controlling LPCI injection flow by:

- Throttling MO 2(3)-1501-21B, LPCI VLV.

AND/OR

- Throttling MO 2(3)-1501-38A, TORUS CLG/TEST.

p. Control Reactor pressure < 150 psig, but high enough to keep the ERVs open by:

- Throttling MO 2(3)-1501-21B, LPCI VLV.

AND/OR

- Throttling MO 2(3)-1501-38A, TORUS CLG/TEST.

q. Control reactor level AND pressure when moderator temperature can be maintained between 180 and 200°F.

r. WHEN desired to terminate Alternate Shutdown Cooling, THEN:

(1) Close MO 2(3)-1501-22B, INJ VLV.

(2) Place all ADS valve Keylock switches to AUTO.

(3) Terminate torus cooling (DOP 1500-02).

(4) Exit this procedure.

G. 6. IF injection will be routed through A LPCI loop, THEN:

a. Close MO 2(3)-1501-38A, TORUS CLG/TEST.

CAUTION

IF MO 2(3)-1501-38B is throttled open > 36 seconds, THEN the LPCI system SHALL be declared inoperable.

b. Throttle open MO 2(3)-1501-38B, TORUS CLG/TEST, until desired flow rate is obtained (approximately 5000 gpm per pump).

c. Close MO 2(3)-1501-21A, LPCI VLV.

d. Verify ONE of the following conditions:

(1) Reactor Recirculation pumps are in operation.

OR

(2) The following valves are closed:

(a) MO 2(3)-0202-5A, 2(3)A PP DISCH VLV.

(b) MO 2(3)-0202-5B, 2(3)B PP DISCH VLV.

CAUTION

Do NOT exceed 100°F/hour cooldown rate.

e. Place ONE of the ERV key locks in OPEN:

- 2(3)-0203-3B, 2(3)B ERV.
- 2(3)-0203-3C, 2(3)C ERV.
- 2(3)-0203-3D, 2(3)D ERV.
- 2(3)-0203-3E, 2(3)E ERV.

f. Open MO 2(3)-1501-22A, INJ VLV.

CAUTION

1. Do NOT exceed 5000 gpm per LPCI pump as indicated on FR 2(3)-1540-7, LPCI SYS TOTAL FLOW.

2. Do NOT exceed reactor pressure of 150 psig.

3. Do NOT cycle ADS valves when flooded.

4. Do NOT raise level above the main steam lines (conservatively 100 inches indicated) UNTIL reactor pressure is < 45 psig.

G. 6. g. Throttle open MO 2(3)-1501-21A, LPCI VLV, to achieve a slow increasing reactor level AND cooldown rate ≤ 100°F/hr.

h. Throttle closed MO 2(3)-1501-38B, TORUS CLG/TEST, to maintain LPCI pump flow < 5000 gpm per pump or to increase LPCI injection.

i. Control reactor cooldown rate ≤ 100°F/hr by controlling LPCI injection flow by:

- Throttling MO 2(3)-1501-21A, LPCI VLV.

AND/OR

- Throttling MO 2(3)-1501-38B, TORUS CLG/TEST.

j. Control Reactor pressure < 150 psig, but high enough to keep the ERV open by:

- Throttling MO 2(3)-1501-21A, LPCI VLV.

AND/OR

- Throttling MO 2(3)-1501-38B, TORUS CLG/TEST.

k. Control reactor level AND pressure when moderator temperature can be maintained between 180 and 200°F.

CAUTION

When raising level it is possible, because of off normal calibration conditions, to quench the reactor vessel shell without a corresponding decrease in vessel flange temperature.

l. IF reactor level approaches 185 inches indicated reactor level, THEN monitor reactor vessel shell temperature for signs of a sharp increase in shell cooldown rate.

G. 6. m. IF indicated reactor level reaches 190 to 210 inches, THEN stop level increase.

n. IF additional ERV flow is required, THEN open an additional ERV:

- 2(3)-0203-3B, 2(3)B ERV.
- 2(3)-0203-3C, 2(3)C ERV.
- 2(3)-0203-3D, 2(3)D ERV.
- 2(3)-0203-3E, 2(3)E ERV.

o. Control reactor cooldown rate ≤ 100°F/hr by controlling LPCI injection flow by:

- Throttling MO 2(3)-1501-21A, LPCI VLV.

AND/OR

- Throttling MO 2(3)-1501-38B, TORUS CLG/TEST.

p. Control Reactor pressure < 150 psig, but high enough to keep the ERVs open by:

- Throttling MO 2(3)-1501-21A, LPCI VLV.

AND/OR

- Throttling MO 2(3)-1501-38B, TORUS CLG/TEST.

q. Control reactor level AND pressure when moderator temperature can be maintained between 180 and 200°F.

r. WHEN desired to terminate Alternate Shutdown Cooling, THEN:

(1) Close MO 2(3)-1501-22A, INJ VLV.

(2) Place ALL ADS valve keylock switches to AUTO.

(3) Terminate torus cooling (DOP 1500-02).

(4) Exit this procedure.

H. DISCUSSION:

1. This procedure establishes Alternate Shutdown Cooling from the reactor when Shutdown Cooling (SDC) is unavailable. LPCI is used to provide flow from the torus to the reactor and the Relief valves provide a pathway for water to flow from the reactor to the torus. This removes heat from the reactor and deposits it in the torus. Heat in the torus is removed by use of the LPCI system in torus cooling mode.

2. Entry into this procedure is made when the Shutdown Cooling has failed, attempts to restore SDC have failed, and back-up methods of heat removal have been attempted per the abnormal procedure for Loss of Shutdown Cooling.

3. For implementation of this procedure, the ADS valves require pressurizing of the reactor to approximately 50 psig in order for the Target Rock to open. ERVs have been found to open with lower pressure. It is anticipated that the vessel will pressurize but is limited to < 150 psig in the Reactor vessel to stay within the assumptions made in the analysis and testing performed (reference generic report NEDE-24988-P) and the Tech Spec mode applicability for HPCI and Isolation Condenser.

4. Engineering analysis of a water hammer event in 1985 demonstrated that the main steam lines can be filled with water provided the differential pressure between the reactor steam dome and the saturation pressure of the reactor coolant is controlled and maintained < 50 psig.

W. WRITER'S REFERENCES:

1. Dresden Station NUREG 0737 Item II.D.1 "Performance Testing of BWR Safety /Relief Valves", response to NRC request for additional information; E.D. Swartz letter to D.G. Eisenhut dated October 15, 1982.

2. LS 05-83-08-035, August 25, 1983 D.L. Crutchfield letter to D.L. Farrar- Subject: NUREG 0737 Item II.D.1, Safety and Relief Valve Testing.

3. NEDE-24988-P, Analysis of Generic BWR Safety /Relief Valve Operability Test Results

4. IR 663242, IDNS Comments Regarding Raw Water Piping Issue Affects.

ATTACHMENT A

REACTOR COOLDOWN MONITORING

The following data must be verified to satisfy Tech Spec requirements.
The PRIMARY indication should be used when available.
The ALTERNATE indication should only be used if the primary is NOT available.

CHECK 1		
During cooldown, at least once per 30 minutes, rate of change of primary system coolant AND bottom head temperatures shall be determined to be ≤ 100°F per hour. (Tech Spec SR 3.4.9.1)		
PRIMARY	< 212°F	Panel 902(3)-4, TR 2(3)-260-11, RECIRC LOOP TEMPS
	≥ 212°F	Panel 902(3)-5, UR 2(3)-640-27, WR RX PRESS (Saturation temperature corresponding to pressure.)
ALTERNATE	< 212°F	None; refer to DOA 0201-04, Loss of Vessel Flange, Shell or Recirculation Loop Temperature Recorders During Heatup or Cooldown.
	≥ 212°F	Panel 902(3)-4, TR 2(3)-260-11, RECIRC LOOP TEMPS

CHECK 2	
During cooldown, at least once per 30 minutes, determine that the vessel metal temperature and pressure are in the Acceptable Region of Tech Spec Figure 3.4.9-2. (Tech Spec SR 3.4.9.1)	
Upper Vessel and Beltline Region	
PRIMARY	Panel 902(3)-4, TR 2(3)-260-11, RECIRC LOOP TEMPS (Must have flow through loop. This is representative of the most restrictive beltline region metal temperatures.)
ALTERNATE	Panel 902(3)-5, UR 2(3)-640-27, WR RX PRESS (Saturation temperature corresponding to pressure) (Alternate may be used without forced Recirc flow.)
Bottom Head	
PRIMARY	Panel 902(3)-21, TR 2(3)-263-104, Point 6, BOTTOM HEAD DRAIN (Must have flow through the line.)
ALTERNATE	Panel 902(3)-21, TR 2(3)-263-104, Point 9, VESSEL BOTTOM HEAD (This must be corrected for inside surface metal temperature by subtracting 75°F from the indicated temperature.)

ATTACHMENT A (Continued)
REACTOR COOLDOWN MONITORING

Unit ________ Date __________________

Time	Reactor Coolant Temperature (°F)	Reactor Pressure (psig)	Reactor Upper Vessel Temperature (°F)	Reactor Bottom Head Temperature (°F)	Check 1 (✓)	Check 2 (✓)	Initial

ATTACHMENT 4

Procedure OU-DR-104, "Shutdown Safety Management Program"

SHUTDOWN SAFETY MANAGEMENT PROGRAM

1. PURPOSE

1.1. This procedure defines the key safety functions and the safety level colors for Dresden Station.

1.2. This procedure provides guidance for the manual determination of Shutdown Safety Colors.

1.3. This procedure applies to the planning, scheduling, and execution of work on a unit already in or expected to be in a shutdown mode of operation. It does not apply to Unit 1, which is permanently shutdown.

1.4. This procedure is the site specific procedure that implements corporate shutdown safety management program procedure OU-AA-103. Implementation of both procedures is required to ensure full compliance with the shutdown safety program.

2. TERMS AND DEFINITIONS

2.1. **Key Safety Functions**:

1. AC Power – Section 4.4
2. Decay Heat Removal – Section 4.5
3. Fuel Pool Cooling – Section 4.6
4. Inventory Control – Section 4.7
5. Vital Support Systems – Section 4.8
6. Reactivity Control – Section 4.9
7. Containment – Section 4.10

2.2. **Safety Level Colors**:

2.2.1. Dresden uses the N system as described below:

1. N = Given Plant conditions, the minimum number of pathways required to safely protect a key Safety Function.

2. Safety Level Colors are assigned as follows:

TOTAL NUMBER OF PATHWAYS AVAILABLE	NO HIGH RISK ACTIVITIES ARE INPROGRESS WHICH AFFECT THE KEY SAFETY FUNCTION BEING EVALUATED	HIGH RISK ACTIVITIY IN PROGRESS WHICH AFFECTS THE KEY SAFETY FUNCTION BEING EVALUATED
N+2	Green	Green
N+1	Green	Yellow
N	Yellow	Orange
<N	Red	Red

2.3. **Available**: For the purposes of this procedure, a system, structure or component (SSC) along with its necessary auxiliary systems, controls, instrumentation and power supplies is capable of performing its intended function and can be placed in service by immediate manual (simple operator actions) or automatic means. **(CM-3)**

2.4. **Containment closure**: The action to secure secondary containment and its associated structure, systems, and components as a functional barrier to fission products release under existing plant conditions (i.e., Time to Boil).

2.5. **Contingency Plan:** A plan of actions to:

1. Provide response actions for postulated events that would present a challenge to Key Safety Functions.
2. Maintain Defense-in-Depth by alternate means when pre-outage planning reveals that specified systems, structures, or components will be unavailable.
3. Restore Defense-in-Depth when system availability drops below the planned Defense-in-Depth during the outage.
4. Minimize the likelihood of a loss of Key Safety Functions during higher-risk evolutions.

2.6. **Decay Heat Removal (DHR) Capability:** The ability to maintain reactor coolant system and spent fuel pool temperature and/or pressure below specified limits following a shutdown.

1. Mode 4 and 5 – ability to maintain < 212° F
2. Mode 3 – ability to reach < 212° F within reasonable time unless plans are to remain in mode 3

2.7. **Defense-in-depth**: For the purpose of managing risk during shutdown, Defense-in-Depth is the concept of providing systems, structures, and components to ensure backup of Key Safety Functions using redundant, alternate, or diverse methods.

2.8. **Elevated Risk**: Any ORANGE or RED shutdown safety status.

2.9. **First Time Evolutions**: Those activities (affecting Shutdown Safety) that have never been conducted on the equipment.

2.10. **Forced Outages**: For the purpose of managing risk during shutdown any outage that requires unit shutdown and entry into modes of operation for which the SSMP is applicable, and were not identified and planned at least one month in advance of the outage.

2.11. **High Risk Activity:** Activities, plant configurations, or conditions during shutdown where the plant is more susceptible to an event causing the loss or challenge to a Key Safety Function.

2.12. **Inventory Control:** Measures established to ensure that irradiated fuel assemblies remain adequately covered to maintain heat transfer and shielding capabilities.

2.13. **Limiting Condition for Operation (LCO) 3.0.4.b**: LCO 3.0.4.b allows entry into a MODE or other specified condition in the Applicability with inoperable equipment required by TS, provided that a risk assessment demonstrates the acceptability. OU-AA-103 attachment 1 must be completed if LCO 3.0.4.b is implemented. Additional guidance and restrictions are provided in OU-AA-103.

2.14. **Lowered Inventory:** Level at or below the flange, fuel in the vessel and RPV head de-tensioned.

2.15. **Procedural and Paragon model changes:** content (philosophical) changes to the site-specific procedure and PARAGON outage models must be approved by SSRB.

2.16. **Protected Equipment:** Equipment (or systems) whose availability has been physically identified as essential to ensure either defense-in-depth of a Key Safety Function is maintained or overall risk levels are maintained. **(CM-3)**

2.17. **Reactivity Control:** Measures established to preclude inadvertent criticality, power excursions or losses of shutdown margin, and to predict and monitor core behavior.

2.18. **Time to Boil**: Given the plant configuration, decay heat load, and location of the fuel from the previous operating cycle, the time it would take to reach bulk coolant saturation temperature with **no** Decay Heat Removal systems in operation. Consider the reactor and spent fuel pool separately or as one body depending on plant conditions.

2.18.1. **Short Time to Boil periods**: The periods from when the Reactor is shut down until the fuel pool gates are removed, and from fuel pool gates installation until Rx startup are considered to be Short Time to Boil periods.

2.18.2. **Long Time to Boil period**: Period between fuel pool gates removal and installation.

2.19. **Time to Uncover the Core:** Given the plant configuration, decay heat load, and location of the fuel from the previous operating cycle, the time it would take to reduce the reactor vessel inventory to the top of the active fuel by boiling.

2.20. **Schedule Changes**: A schedule change as it relates to the SSMP is an alteration in the sequencing for removal / restoration of equipment or an alteration in the sequencing of plant configuration changes for those activities that support Key Safety Functions and thus alters their relationship from the previously approved schedule. Shifting of equipment removal / restoration or plant configuration changes forward or backward in time does **not** constitute a schedule change as long as their relationship to the previously approved sequence in the outage network remains intact.

2.21. **<u>Switchyard Work Affecting Shutdown Safety</u>**: Work in the Switchyard(s) that significantly increases the potential for initiating a Loss of Off-Site Power event, or loss of power to a component that may affect shutdown safety. **(CM-3)**

3. <u>GENERAL GUIDELINES AND POLICIES</u>

3.1. OU-AA-103, SHUTDOWN SAFETY MANAGEMENT PROGRAM provides additional guidance. Both procedures should be reviewed and implemented when addressing shutdown safety conditions.

3.2. ENSURE to review Dresden's response to SOER 09-1 prior to revising this procedure to verify no commitments will be impacted by the change. If any of the SOER will be affected then follow the process for commitment changes.

3.3. **<u>Equipment Availability</u>**: The following guidelines will be used to determine availability of equipment:

3.3.1. Procedures, standing orders, work instructions or briefed contingency plans (reviewed and approved) exist for using the equipment to meet its intended function. **(CM-3)**.

3.3.2. A system does **not** need to be operable as defined in the Technical Specifications to be considered available.

3.3.3. Credit may be taken for reasonable actions either in the control room or in-plant.

3.3.4. When determining "Reasonable Actions" the time required to place the equipment in service to maintain the Key Safety Function should be considered. An example would be Time to Boil compared to the time required to place a Decay Heat Removal (DHR) system in service.

3.3.5. The time it takes to restore the equipment shall not exceed half the time equipment is required to be placed in operation (time to boil and core uncovery time) unless otherwise specified in the procedure (such as, time to secure secondary containment shall not exceed time to boil).

3.3.6. Motor operated valves (such as LPCI injection, Core Spray injection, SDC) with de-energized power source (480V AC feed breaker or MCC) may be considered available if all of the following conditions are met: **(CM-3)**

1. Meet the availability definition above
2. Can be manually operated
3. Not being worked on

4. They are not the only train supporting the key safety function. Example is when MCCs 28-7/29-7 are de-energized, then one of the core spray or condensate pumps must be available.

5. No high risk evolutions impacting the associated key safety function in progress.

3.3.7. A system may be considered available with a portion of the system out of service as long as the system can still perform its intended function. **(CM-3)**

1. A LPCI pump may be considered available with the minimum flow or test valve out of service as long as the pump functionality and injection path are not impacted.

3.3.8. A system cannot be considered available if its functionality is removed (e.g. clearance applied, drained, breached, etc.). **(CM-3)**

3.3.9. Credit may also be taken for temporary alterations (e.g., power supplies), contingency plans, and line-ups, provided procedural guidance or work instructions are available, reviewed and approved. Credited temporary power or temporary back-up equipment must be installed and tested to consider a component available. **(CM-3)**

3.3.10. Time to secure secondary containment shall not exceed the time to boil for the current plant conditions.

3.3.11. Since time to boil will be zero while in mode 3, the following criteria will be used to define the availability of decay heat removal equipment in the event SDC and/or RWCU trip or have to be temporarily removed from service.

1. Both secondary and primary containment are maintained.

2. If tripped, the cause of the trip is quickly identified and isolated.

3. There is reasonable assurance that the equipment can be restarted and the unit will reach cold shutdown condition within reasonable time.

4. Actions to restore the system are simple and use approved procedures or approved written instructions.

3.3.12. A system/pump may be considered available during divisional Bus Under-voltage and ECCS Integrated Functional (UV) Test even with their breaker racked in test or manual injection valves shut under the following guidelines:

1. The UV can be stopped at any time.
2. Actions can be immediately initiated to rack in breaker and/or open the valves.
3. Approved instructions are available to re-establish the injection path.
4. Operators are briefed prior to the start of the UV test.
5. At least one other injection source capable of injection to vessel (other than systems associated with UV test) must also be available.
6. At least one other Decay Heat Removal system or loop (other than systems associated with UV test) must be available.

3.3.13. A pump control switch may be in pull-to-lock (PTL) and still be considered 'available' as long as there are no Clearance Order cards preventing the use of the pump.

3.4. **Lowered Inventory**:

3.4.1. Efforts shall be made to minimize periods of lowered inventory conditions. **(CM-3)**

3.4.2. Reactor Cavity Draindown SHALL be considered as infrequently performed activity (IPA).

3.4.3. Lowered inventory conditions, other than the normal cavity flood up and draindown, SHALL be clearly identified in the outage schedule. **(CM-3)**

3.5. **Operations SHALL notify the Shutdown Risk Manager:**

3.5.1. Prior to making shutdown safety related equipment unavailable unless previously planned.

3.5.2. Immediately any time shutdown safety related equipment is found or made unavailable due to a failure or emergent work.

3.6. **Shutdown Safety Management Plan (SSMP):**

3.6.1. All Dresden Station Refuel Outages and Planned Outages containing significant work on systems that support Key Safety Functions shall have the Shutdown Risk Plans reviewed and approved by the PORC Committee.

3.6.2. Following approval of the outage specific Shutdown Safety Management Plan (SSMP), additional changes that impact Key Safety Functions will be reviewed and approved by the SSRB. **(CM-3)**

3.6.3. If the SSMP has been reviewed and approved by PORC, the SSRB should consider whether these changes should be presented to PORC. Things to address when considering if a second PORC will be required:

1. Impact on the overall unit color.
2. Impact on the individual KSF color.
3. Changes to contingency plans required for ORANGE or RED conditions.
4. Major changes to the SSMP.

3.6.4. A copy of the SSMP SHALL be maintained in the OCC and Main Control Room. The shutdown risk manager will update these copies as changes are made and as deemed necessary. **(CM-3)**

3.6.5. The plan shall include, at a minimum, the following: **(CM-3)**

1. Overall profile for the shutdown unit. This can be in the form of a color printout of PARAGON risk level analysis (preferred) or similar profile such as shutdown risk schedule.
2. Overall profile for the opposite unit. This can be in the form of a color printout of PARAGON risk level analysis (preferred) or similar profile such as online risk schedule.
3. Shutdown safety review. This includes a summary of the overall unit status and a brief description for each of the KSF.
4. Contingency plans.
5. Other pertinent information such as high risk evolutions.

3.6.6. The Shutdown Safety Manager SHALL provide shutdown risk information to OCC and Operations shift personnel via formal briefings each shift and as risk conditions change. In addition, the Shutdown Safety Manager SHALL provide look-ahead analysis of proposed schedule changes and prepares Shutdown Safety Review Board review packages.

3.6.7. At least once per day (or as emergent conditions dictate), the Shutdown Risk Manager, or designee, shall analyze the updated schedule using either PARAGON or Attachment 1 or 2 (Equipment Availability) and provide a SSMP analysis look-ahead to the site. At a minimum, the SSMP analysis will include the following:

1. A color print out of the PARAGON Risk Level Analysis or a copy of the shutdown risk schedule (hammocks) if PARAGON is not available for at least the next 24 hours.
2. A summary of available KSF systems/equipment/trains
3. A list of Protected Equipment
4. Time to boil and core uncovery time
5. Any additional pertinent information as deemed necessary (high risk evolutions, minimum equipment required to prevent color change, major upcoming evolutions that may impact shutdown safety, moderator and fuel pool temperature).

3.7. Any deviations from defense-in-depth attributes contained in INPO 06-008, Guidelines for the Conduct of Outages at Nuclear Power Plants, must be thoroughly understood and approved by senior managers. **(CM-3)**

3.8. First time evolutions to be evaluated for risk impact and, if appropriate, conducted during Long Time to Boil periods and not in lowered inventory condition. **(CM-3)**

3.9. **Contingency Plans**:

3.9.1. Contingency plans should be prepared prior to the pre-outage shutdown safety risk assessment and other independent assessments. **(CM-3)**

3.9.2. Contingency plans will be generated:

1. As required by OU-AA-103, Shutdown Safety Management Program.
2. For ORANGE and RED conditions **(CM-3).**
3. When shutdown risk is YELLOW and defense-in-depth for the particular key safety function is reduced to one normal method or equipment.
4. Additional contingency plans may be established as deemed necessary by the SSRB for YELLOW or GREEN conditions **(CM-3).**

3.9.3. Contingency plans shall address actions to restore equipment needed for key safety functions and/or the use of alternate and backup equipment **(CM-3).**

3.9.4. OP-DR-104-1001, SHUTDOWN RISK MANAGEMENT CONTINGENCY PLANS, outlines contingency plans for the various key safety functions.

3.10. **Defense-in-depth**:

3.10.1. The ultimate goal is to maintain a full compliment of equipment and functions required for all of the key safety functions or, at a minimum, for all key safety functions to remain GREEN. All efforts shall be made to restore unavailable equipment and/or functions in an expeditious manner and, as practical, to maintain all key safety functions GREEN. **(CM-3)**

3.10.2. During outage planning, the minimum requirement to avoid risk color change shall be identified (attachment 5) and included in the SSMP.

3.10.3. During outage executions, compliance with defense-in-depth shall be verified once per shift or before major safety system availability drops below the planned defense-in-depth. This may be performed by running PARAGON or use of the equipment availability checklist. This requirement applies for all refuel outages and, when deemed necessary by the Shutdown Safety Review Board (SSRB), during maintenance and forced outages.

3.11. **High Risk Activities**:

3.11.1. When determining if a "High Risk Activity" exists, consider any work or condition that has a reasonable potential to reduce the number of systems being taken credit for to support a Key Safety Function. An example would be the performance of work that has a reasonable potential to cause the loss of a Decay Heat Removal system that is being taken credit for, and reasonable actions to restore the system cannot be maintained.

3.11.2. High risk activity review SHALL be conducted in accordance with OU-AA-103, Shutdown Safety Management Program.

3.11.3. Concurrent high risk activities affecting the same key safety function should be avoided if possible. **(CM-3)**

3.11.4. If an activity/evolution is deemed high risk to shutdown safety, then it should be input into PARAGON via the scheduling tools and results should be evaluated.

3.11.5. All high risk evolutions shall be identified (attachment 4) during outage planning and included in the SSMP.

3.12. **Heavy Loads**:

3.12.1. If a heavy lift is scheduled and the drop zone could affect equipment that is monitored by decay heat removal key safety function (KSF), then identify a minimum set of safe shutdown equipment that will remain available to provide continued decay heat removal for the shutdown unit.

3.12.2. If a drop could damage a containment boundary and containment is required, then a High Risk Activity shall be considered for the containment KSF.

3.12.3. If a drop could damage an un-isolable reactor vessel or fuel pool boundary, then a Potential to Drain Activity shall be considered for the affected KSF.

3.12.4. Engineering controls such as additional barriers to prevent damage from a drop may be used. These controls may eliminate the need to consider equipment unavailable, schedule a High Risk Activity, or schedule a Potential to Drain Activity.

3.12.5. For heavy lifts performed in support of an outage that could affect equipment on the operating unit, notify the on line risk manager to perform the necessary risk assessment for the impending activities.

3.13. Events may occur that could place the shutdown unit in a condition outside the bounds of shutdown risk management procedures.

Example – defense in depth requirement is met, however, due to unforeseen condition (equipment failure, human performance or others), the associated Key Safety Function requirement is no longer met (A rod out block is enforced, but rods were withdrawn due to a mechanical failure or human performance). In this event, the following actions SHALL be implemented:

1. Shutdown Safety Review Board (SSRB) will immediately convene and evaluate the condition

    A. Following the decision tree in attachment 3, and based on the definition and intent of the Key Safety Function, determine the applicable actions.

    B. Engineering assistance (such as Nuclear Engineer for reactivity KSF) may be required to evaluate the condition and determine appropriate actions.

2. The event SHALL be discussed with senior management and concurrence of senior management must be obtained.

3. Complete attachment 1 of OU-AA-103 to document the condition and actions taken.

3.14. **Protected Equipment**:

3.14.1. Equipment shall be designated as protected pathway and posted as governed by OP-AA-108-117, PROTECTED EQUIPMENT PROGRAM, and the following guidelines:

1. Orange and Red conditions **(CM-3)**
2. At a minimum, one in-service decay heat removal train must always be protected. **(CM-3)**
3. At a minimum, one reactor inventory make-up train and required support systems during lowered RCS inventory condition
4. At a minimum, one spent fuel pool cooling train once core offload starts until the time to boil in the spent fuel pool is greater than 24 hours
5. A loss of running or in service equipment (SDC pump when on, 4KV bus when required energized) will cause a color change to ORANGE or RED. **(CM-3)**
6. Available off site power source when off site power is down to a single source (line or breaker).
7. As deemed necessary by SSRB. **(CM-3)**

3.14.2. Work on or near (within 2 feet) protected equipment will generally not be allowed unless otherwise allowed by OP-AA-108-117.

3.15. **Switchyard Work Affecting Shutdown Safety**:

3.15.1. All switchyard work will be controlled per OP-AA-108-107-1002, INTERFACE AGREEMENT BETWEEN EXELON ENERGY DELIVERY AND EXELON GENERATION FOR SWITCHYARD OPERATIONS.

3.15.2. Efforts shall be made to schedule switchyard work affecting Shutdown Safety (e.g AC Source) during periods of Long Time to Boil and when not in lowered inventory conditions. **(CM-3)**

3.15.3. Efforts shall be made NOT to schedule high risk switchyard work with other AC power high risk evolutions such as undervoltage testing. An evaluation of concurrent AC and switchyard high risk evolutions shall be completed prior to execution. **(CM-3)**

3.15.4. Switchyard high risk evolutions SHALL be avoided when either DIV 1 or DIV 2 AC power is not available. Station Manager's approval MUST be obtained if this condition cannot be met.

3.16. **On line Unit Interface**

3.16.1. Prior to the start of the outage, an interface agreement between the shutdown unit and the on line unit shall be completed between the Cycle Manager, Station Risk Engineer and Shutdown Safety Manager. The agreement will ensure integration of the on line and shutdown risk assessment models.

3.16.2. Opposite unit impact:

1. The Shutdown Safety Manager SHALL inform the on line Cycle Manager of all outage activities that may impact on line risk.

2. Prior to the outage, the SSM shall review all on line activities during the outage to ensure the KSF are not impacted by the on line work.

3. All shutdown safety activities impacting the opposite unit risk shall be coded and identified in the outage schedule.

4. **MAIN BODY**

4.1. All Dresden Station Refuel Outages and Planned Outages containing significant work on systems that support Key Safety Functions shall have the Shutdown Risk Plans reviewed and approved by the PORC Committee.

4.2. **Manually Determining Shutdown Safety Colors**

4.2.1. To manually determine the Key Safety Function Shutdown Safety Colors, go to each of the 7 Key Safety Functions and perform the following:

1. Utilize the Schedule and Attachment 1 or 2, as necessary to determine equipment availability.

2. Select the Applicable Mode and Plant Condition that matches the existing plant conditions.

3. Determine how many of the pieces of the listed equipment are available and determine the total point value for that Key Safety Function.

4. Determine if a High Risk Activity, which affects the Key Safety Function, is in progress.

5. Using the Key Safety Functions table select the appropriate column and point value.

6. Go to the last column on the right side where the risk level color is listed.

4.3. Unit Risk Level Determination

4.3.1. If AC power key safety function is Yellow, then the overall Unit Shutdown Risk level is Yellow.

4.3.2. For all safety functions other than AC, if there are less than 2 Yellow Safety Functions and No Orange OR Red: The Unit Shutdown Risk Level is Green

4.3.3. If there are 2 OR more Yellow Safety Functions and NO Orange OR Red: The Unit Shutdown Risk Level is Yellow

4.3.4. If there is an Orange Safety Function and NO Red Safety Function: The Unit Shutdown Risk Level is Orange

4.3.5. If there is a Red Safety Function: The Unit Shutdown Risk Level is Red.

4.3.6. Unplanned color changes:

1. Notify outage manager or his designee.
2. Notify the Plant Manager for entry into an ORANGE or RED condition.
3. Initiate an IR.
4. Review OP-AA-106-101-1001, Event Response Guidelines, to determine if a PROMPT INVESTIGATION is required.

4.4. AC Power Key Safety Function

4.4.1. Guidelines

1. There will always be at least two independent power supplies to at least one 4kv Vital Bus.

2. In Modes 4 and 5 the Reserve Aux. Transformer, or Unit Aux. Transformer (back feed mode), or Vital Bus Crosstie Breakers from the other unit will be available.

3. Diesel Generators: (Unit SBO DG, 2/3 EDG, Unit EDG)

   A. There will be at least one EDG or SBO Diesel Generator available at all times.

   B. With only one DG available, AC function shall be no better than YELLOW

   C. With no DG available the AC Key Safety Function shall be no better than ORANGE.

   D. During Short Time to Boil or Lowered inventory periods with no DG available the AC Key Safety Function shall be no better than RED.

   E. DGs do not have to be capable of Auto closing on the bus to be considered available. Credit is taken for manual breaker alignment. Manual breaker alignment is allowed for the DGs just as it is allowed for Bus and Unit cross-tie capability. DGs are typically considered available during performance of Diesel surveillances, including UV test.

4. Every attempt should be made to schedule work on Vital Buses and their associated Emergency Diesel Generators concurrently. Bus relay work and testing should be scheduled in the work window for the component affected.

5. Work on major electrical equipment should be avoided during lowered inventory periods or when time to boil is short.

6. Off-site power:

   A. Ring bus work should be coordinated with availability of off-site power feeds to ensure adequate availability at all times

   B. Station Transformers can only be considered available when there is an off-site supply of power to that transformer. It is important to be aware of off-site power supplies when determining transformer availability.

   C. Work to be performed in the switchyard should be coordinated with the rest of the outage schedule so that High Risk Activities are not performed during periods of high risk in the plant or when redundant power trains are out of service. Ensure adequate emergency power during periods of high risk in the switchyard. **(CM-3)**

   D. Because the off-site transmission lines feeding power to the station are maintained and under the control of the load dispatcher, and not the station Shift Manager, it is imperative that there is sufficient coordination between onsite and off-site personnel to prevent reductions in available off-site power sources below determined safe shutdown levels. **(CM-3)**

   E. When coordinating work on transmission lines and work on station equipment, it is important to ensure that station configurations are maintained in a status to permit off-site power feed to the station. **(CM-3)**

   F. Because the weather has the potential to adversely affect high voltage transmission lines, it is important to regularly monitor weather forecasts to ensure adequate sources of off-site power are maintained in periods of inclement weather.

   G. All efforts shall be made to maintain two CB's available at all times to provide power to the available transformer (see list below) with at least one in-service line providing power through each CB. In the event that only one of the listed CBs is available for an outage unit's energized transformer, **THEN** this condition shall be considered as a high risk and the available off site power source shall be protected. **(CM-3)**

      – TR 21 – CB 1-2, 1-7 (unit on backfeed)
      – TR 22 – CB 2-3, 3-4
      – TR 31 – CB 9-10, 10-11 (unit on backfeed)
      – TR 32 – CB 4-8, 8-9, 8-15

H. When required to establish protective pathways on Off-Site line(s), Bulk Power must be notified so the TSO will not inadvertently remove the line from service from the other end. Bulk Power shall be notified following removal of protected pathways.

4.4.2. Assessment of AC Power Shutdown Safety Color

1. Primary Power Supplies

A. TR 21 in Back Feed mode or TR 22 in normal mode of operations fed from the 345KV yard.

1. 345kv Line 0302 – will be counted any time it is available
2. 345kv Line 1220 – will be counted any time it is available
3. 345kv Line 1221 - will be counted any time it is available
4. 345kv Line 1222 - will be counted any time it is available
5. 345kv Line 1223 - will be counted any time it is available
6. 345kv Line 8014 - will be counted any time it is available
7. 345kv Line 2311 - will be counted any time it is available

B. TR 22 in normal mode of operations fed from the 138kv Yard.

1. 138kv Line 1210 - will be counted any time it is available
2. 138kv Line 1207 - will be counted any time it is available
3. 138kv Line 1206 - will be counted any time it is available
4. 138kv Line 1205 - will be counted any time it is available
5. 138kv Line 0904 - will be counted any time it is available
6. 138kv Line 0903 - will be counted any time it is available

C. TR 31 in Back Feed mode or TR 32 in normal mode of operations fed from the 345kv Yard.

1. 345kv Line 0302 - will be counted any time it is available
2. 345kv Line 1220 - will be counted any time it is available
3. 345kv Line 1221 - will be counted any time it is available
4. 345kv Line 1222 - will be counted any time it is available
5. 345kv Line 1223 - will be counted any time it is available
6. 345kv Line 8014 - will be counted any time it is available
7. 345kv Line 2311 - will be counted any time it is available.

2. Alternate Power Supplies

A. Emergency Diesel Generator 2/3 will be counted any time it is available.

B. Emergency Diesel Generator 2 will be counted any time it is available.

C. Emergency Diesel Generator 3 will be counted any time it is available.

D. Station Blackout Diesel Generator 2 will be counted any time it is available.

E. Station Blackout Diesel Generator 3 will be counted any time it is available.

F. Bus 23-1 to 33-1 Cross-tie will be counted any time Vital Buses 23-1 and 33-1 and both Division I Cross-tie breakers are available.

G. Bus 24-1 to 34-1 Cross-tie will be counted any time Vital Buses 24-1 and 34-1 and both Division II Cross-tie breakers are available.

3. Applicable Modes: 3, 4, 5, & De-fueled

   A. Determine the total number of AC Power Supplies by adding the following equipment that is available:

      1. Unit Aux. Transformer (Back feed mode) - 2 points
         - If ONLY one CB is available **THEN** this condition shall be considered as high risk.

      2. Reserve Aux. Transformer - 2 points
         - If ONLY one CB is available **THEN** this condition shall be considered high risk.

      3. Unit Emergency Diesel Generator - 1 point

      4. Shared Emergency Diesel Generator – 1 point

      5. Unit Station Blackout Diesel Generator - 1 point

      6. Division I 4KV Unit Cross Tie Breakers - 1 point

      7. Division II 4KV Unit Cross Tie Breakers - 1 point

   B. Determine availability of diverse power sources. If diversity is not available, then determine if time to boil is short or long.

   C. Determine if any High Risk Activities affecting the AC Power Key Safety Function are in progress.

D. Use the table below to determine the Shutdown Safety Color

AC Power Key Safety Function **Applicable Modes: 3, 4, 5, & De-fueled**					
Power Sources **Diversity Available**		**Power Sources** **Diversity NOT Available**			
		Long Time to Boil AND not in Lowered Inventory Condition		**Short Time to Boil OR Lowered Inventory**	
NO HIGH RISK ACTIVITIES IN PROGRESS	**HIGH RISK ACTIVITIES IN PROGRESS**	**NO HIGH RISK ACTIVITIES IN PROGRESS**	**HIGH RISK ACTIVITIES IN PROGRESS**	**NA**	**ASSIGNED SAFETY LEVEL COLOR**
4 points	5 points	NA	NA	NA	GREEN(*)
3 points	4 points	NA	NA	NA	YELLOW(*)
2 points	3 points	≥2 points	≥3 points	NA	ORANGE(*)
≤ 1 point	≤ 2 points	≤ 1 point	≤ 2 points	YES	RED(*)

(*) **NOTES:**

1. Both 4KV AC divisions are required for GREEN condition.
2. A minimum of 2 Diesel Generators must be available to be GREEN.
3. AC key safety function can be no better than ORANGE with NO DG.
4. AC key safety function can be no better than ORANGE with NO off site power available.

5. Diversity rules:

   A. Three diverse groups apply:

      1. **"OSP"** - Offsite power through the Auxiliary Transformer TR 21(22)/TR 31(32) or opposite unit Aux. or Reserve Aux. transformer through the 4 KV cross tie.

      2. **"SBO DG"** – Outage unit SBO DG

      3. **"EDG"** - EDG 2/3 or Outage Unit EDG

   B. Two of the above 3 groups are required to consider diversity is available.

   C. For condition of Short Time to Boil **OR** Lowered Inventory, at least two of the three diversity groups must be present or risk will be RED.

   D. For conditions of Long Time to Boil **AND** not Lowered Inventory, at least two of the three diversity groups must be present or risk will be no better than ORANGE.

4.5. <u>Reactor Decay Heat Removal Key Safety Function</u>

4.5.1. Guidelines

> **CAUTION (CM-3)**
>
> Prior to relying on Fuel Pool Cooling or Shutdown Cooling in the FPC Mode as the only systems for Reactor Decay Heat Removal, a Decay Heat Removal Analysis MUST be performed.

1. The periods from when the Reactor is shutdown until the fuel pool gates are removed and from fuel pool gates installation until Rx startup are considered to be Short Time to Boil periods. Any time other than that defined by Short Time to Boil is considered to be Long Time to Boil period.

2. In Modes 4 and 5, Shutdown Cooling Loops that are lined up to Fuel Pool Cooling are still available to the Reactor (SDC Mode).

3. The planned removal of Shutdown Cooling Loops from service should <u>NOT</u> be scheduled during Modes 3, 4, and 5 unless absolutely necessary, to ensure maximum redundancy of the Decay Heat Removal System. **(CM-3)**

4. During short time to boil or lowered inventory conditions, a diesel generator must be available to power the SDC pump to consider a Shutdown Cooling Loop available.

5. Activities that may impact the decay heat removal systems/components should be scheduled during periods of Long Time to Boil, high coolant inventory, component is secured, or defueled conditions. Contingency plans should be in place if activities that potentially impact decay heat removal systems must be scheduled during periods of Short Time to Boil or reduced inventory.

6. At the beginning of each shift, when decay heat removal equipment is required to be in service, a NSO and NLO shall be designated and briefed to restore decay heat equipment. Brief shall include: **(CM-3)**

   A. Applicable procedure(s) and recovery actions.

   B. Current conditions such as time to boil, core uncovery time, available equipment and functions.

   C. Describe and prioritize the available alternate cooling methods to employ for the current conditions including use of contingency systems and components to provide sufficient defense-in-depth.

   D. Personal safety precautions for the possible plant conditions.

   E. Actions to restore secondary containment, if breached.

4.5.2. <u>Assessment of Decay Heat Removal Shutdown Safety Color</u>

1. Primary Sources

   A. 'A' Shutdown Cooling in SDC Mode will be counted any time it is available.

   B. 'A' Shutdown Cooling in FPC Mode will be counted any time it is available in Modes 4 and 5.

   C. 'B' Shutdown Cooling in SDC Mode will be counted any time it is available.

   D. 'B' Shutdown Cooling in FPC Mode will be counted any time it is available in Modes 4 and 5.

   E. 'C' Shutdown Cooling in SDC Mode will be counted any time it is available.

   F. 'C' Shutdown Cooling in FPC Mode will be counted any time it is available in Modes 4 and 5.

2. Alternate Sources

A. 'A' Shutdown Cooling in FPC Mode will only be counted after the cavity is flooded and fuel pool gates are removed or forced circulation is established.

B. 'B' Shutdown Cooling in FPC Mode will only be counted after the cavity is flooded and fuel pool gates are removed or forced circulation is established.

C. 'C' Shutdown Cooling in FPC Mode will only be counted after the cavity is flooded and fuel pool gates are removed or forced circulation is established.

D. Reactor Water Cleanup System will be counted whenever it is available.

E. 'A' Fuel Pool Cooling will only be counted after the cavity is flooded and fuel pool gates are removed or forced circulation is established.

F. 'B' Fuel Pool Cooling will only be counted after the cavity is flooded and fuel pool gates are removed or forced circulation is established.

G. Main steam electromatic relief valves (ERV) as directed in DOP 1000-07, Alternate Shutdown Cooling, will be counted whenever available in **modes 3 & 4 only** with the following restrictions:

1. 2 ERVs available (B, C, D, E)

2. 1 LPCI pump available for vessel make up and torus cooling.

3. 2 CCSW pumps available for torus cooling.

H. LPCI/CCSW as directed per OP-DR-104-1001 and DOP 1900-03 will be counted whenever available in **modes 4 & 5 only** with the following restrictions:

1. One LPCI loop available to support reactor cavity drain down through SDC

AND

2. A second LPCI loop available for injection through a LPCI heat exchanger with one LPCI pump and one CCSW pump for cooling.

3. Applicable Modes 3, 4, & 5 with Fuel Pool Gates installed (Short Time to Boil):

   A. Determine the availability of:

      1. Shutdown Cooling trains available to the Reactor - 1 point each

      2. Reactor Water Clean Up System – ½ point (provided RWCU is available in the blowdown mode and its heat removal capability as listed in ECR #379206 is equivalent to ½ the current reactor decay heat load.

      3. Main steam electromatic relief valves (ERV) – 1 point (**Modes 3 & 4 ONLY.** See restrictions in the previous section, Alternate Sources).

      4. LPCI/CCSW – 1 point (**Modes 4 & 5 ONLY.** See restrictions in the previous section, Alternate Sources).

   B. Determine if any High Risk Activities affecting the Decay Heat Removal Key Safety Function are in progress.

   C. Use the table below to determine the Shutdown Safety Color.

Reactor Decay Heat Removal Key Safety Function **Applicable Modes 3,4, & 5 with Fuel Pool Gates Installed**		
SHORT TIME TO BOIL **PERIODS**		
NO HIGH RISK ACTIVITIES IN PROGRESS	**HIGH RISK ACTIVITIES IN PROGRESS**	**ASSIGNED SAFETY LEVEL COLOR**
2 1/2 points	3 1/2 points	GREEN
2 points	3 points	YELLOW
1 point	2 points	ORANGE
0 points	1 point	RED

4. Applicable Mode: 5 with Reactor Cavity Flooded and Fuel Pool Gates removed:

   A. Determine the availability of:

      1. Fuel Pool Cooling Trains (1 pump and 1 heat exchanger) - 1/2 point each

      2. Shutdown Cooling Loops aligned to Fuel Pool Cooling - 1/2 point each

      3. Reactor Water Clean Up System – ½ point (provided RWCU is available in the blowdown mode and its heat removal capability as listed in ECR #379206 is equivalent to ½ the current reactor decay heat load.

      4. Shutdown Cooling Loops available to the Reactor -1 point each

   B. Determine if any High Risk Activities affecting the Decay Heat Removal Key Safety Function are in progress.

   C. Use the Table below to determine the Shutdown Safety Color:

Reactor Decay Heat Removal Key Safety Function **Applicable Mode: 5 with Reactor Cavity Flooded and Fuel Pool Gates Removed**		
LONG TIME TO BOIL PERIODS		
NO HIGH RISK ACTIVITIES IN PROGRESS	**HIGH RISK ACTIVITIES IN PROGRESS**	**ASSIGNED SAFETY LEVEL COLOR**
1-1/2 points	2-1/2 points	GREEN (*)
1 point	2 points	YELLOW
1/2 point	1-1/2 points	ORANGE
0 points	≤1 point	RED

(*) Risk level can be no better than YELLOW if a single failure results in going from GREEN to RED (e.g. one SDC cooling pump available and no fuel pool cooling or RWCU).

4.6. <u>Fuel Pool Cooling Key Safety Function</u>

4.6.1. Guidelines

1. During refueling operations, the period from when the first Fuel Bundle is unloaded from the Reactor until the Reactor Core is reloaded is considered to be the High Fuel Pool Decay Heat period. Any time other than that defined by High Fuel Pool Decay Heat is considered to be a Low Fuel Pool Decay Heat period.

2. Prior to the start of fuel offload verify: **(CM-1) (CM-3)**

   A. The ability to align a spare loop of SDC to the Spent Fuel Pool within 8 hours of the loss of the operating Shutdown Cooling loop in FPC Assist mode is maintained.

   <u>**OR**</u>

   B. Engineering evaluation determined that it is acceptable <u>NOT</u> to have a backup SDC loop in the fuel pool assist mode available within eight hours while fuel is offloaded from the RPV for the upcoming outage. This evaluation will be performed prior to the refuel outage and specify the acceptable time limit for alignment of the SDC Loop to Fuel Pool Assist (FPA) Mode, based upon current fuel offload calculations, if required.

3. All planned activities, which impact the functionality of the Fuel Pool Cooling system, will be completed before the start of the outage, with the exception of Electrical Bus Outages. **(CM-3)**

4. The only Fuel Pool Cooling system work permitted during the outage will be that which is to correct emergent problems. This work will be considered high priority.

5. At such time that calculations determine the amount of decay heat in the fuel pool to be low, relaxed Defense in Depth measures may be taken.

6. To consider a Shutdown Cooling Loop available during fuel pool high decay heat periods either the Unit SBO Diesel Generator or the associated Emergency Diesel Generator can supply power to it.

4.6.2. Assessment of Fuel Pool Cooling Shutdown Safety Color

1. Primary Fuel Pool Cooling Supply

A. 'A' Fuel Pool Cooling will be counted any time it is available.

B. 'B' Fuel Pool Cooling will be counted any time it is available.

C. 'A' Shutdown Cooling in FPC Mode will be counted any time the loop is available and the SDC to FPC spectacle flange is rotated to open.

D. 'B' Shutdown Cooling in FPC Mode will be counted any time the loop is available and the SDC to FPC spectacle flange is rotated to open.

E. 'C' Shutdown Cooling in FPC Mode will be counted any time the loop is available and the SDC to FPC spectacle flange is rotated to open.

2. Alternate Sources

A. 'A' Shutdown Cooling will be counted any time the loop is available, the SDC to FPC spectacle flange is rotated to blind or the 1901-20 and 1901-64 valves are closed, the cavity is flooded and fuel pool gates are removed or forced circulation is established.

B. 'B' Shutdown Cooling will be counted any time the loop is available, the SDC to FPC spectacle flange is rotated to blind or the 1901-20 and 1901-64 valves are closed, the cavity is flooded and fuel pool gates are removed or forced circulation is established.

C. 'C' Shutdown Cooling will be counted any time the loop is available, the SDC to FPC spectacle flange is rotated to blind or the 1901-20 and 1901-64 valves are closed, the cavity is flooded and fuel pool gates are removed or forced circulation is established.

D. Reactor Water Cleanup System will be counted any time it is available, the cavity is flooded and fuel pool gates are removed or forced circulation is established.

3. Applicable Modes: 3, 4, and 5 with the Reactor Cavity NOT Flooded

   A. Determine the number of Fuel Pool Cooling systems which are available:

      1. Fuel Pool Cooling Trains - 1 point each

      2. Shutdown Cooling aligned to Fuel Pool Cooling - 1 point each

   B. Determine if any High Risk Activities affecting the Fuel Pool Cooling Key Safety Function are in progress.

   C. Use the table below to determine the Shutdown Safety Color:

Fuel Pool Cooling Key Safety Function **Applicable Modes: 3, 4, and 5 with the Reactor Cavity NOT Flooded**		
NO HIGH RISK ACTIVITIES IN PROGRESS	**HIGH RISK ACTIVITIES IN PROGRESS**	**ASSIGNED SAFETY LEVEL COLOR**
2 points	3 points	GREEN
1 point	2 points	YELLOW
0 points	1 point	ORANGE
N/A	0 points	RED

4. Applicable Modes: 5 and De-fueled with the Reactor Cavity Flooded.

   A. Determine the number of Fuel Pool Cooling systems which are available.

      1. Shutdown Cooling in the Fuel Pool Cooling Mode – 1 point each

      2. Fuel Pool Cooling Trains - ½ point each

      3. Shutdown Cooling Trains Available with Fuel Pool Gates removed or forced circulation establish - ½ point each

      4. Reactor Water Clean Up System – ½ point (provided RWCU is available in the blowdown mode and its heat removal capability as listed in ECR #379206 is equivalent to ½ the current reactor decay heat load.

   B. Determine if the Fuel Pool Decay Heat is High or Low

   C. Determine if any High Risk Activities affecting the Fuel Pool Cooling Key Safety Function are in progress.

D. Use the table below to determine the Shutdown Safety Color:

Fuel Pool Cooling Key Safety Function **Applicable Modes: 5 and De-fueled with the Reactor Cavity Flooded**				
LOW DECAY HEAT PERIODS		**HIGH DECAY HEAT PERIODS**		
NO HIGH RISK ACTIVITIES IN PROGRESS	**HIGH RISK ACTIVITIES IN PROGRESS**	**NO HIGH RISK ACTIVITIES IN PROGRESS**	**HIGH RISK ACTIVITIES IN PROGRESS**	**ASSIGNED SAFETY LEVEL COLOR**
1 point	1-1/2 points	2-1/2 points	3 points	GREEN (*)
1/2 point	1 point	2 points	2-1/2 points	YELLOW
0	≤1/2 point	1-1/2 points	2 points	ORANGE
N/A	N/A	1 point	1-1/2 points	RED

(*) Risk level can be no better than YELLOW if a single failure results in going from GREEN to RED (e.g. high decay heat condition, 2 FPC pumps available with one SDC loop available to both fuel pool and cavity decay heat removal).

4.7. Inventory Control Key Safety Function

4.7.1. Guidelines

1. Work and testing on systems connected to the Reactor Coolant System will be performed such that no water movement will occur except as intended.

   A. As much work and testing, as practicable, will be performed isolated from the Reactor Coolant System.

   B. Systems will be verified filled and vented prior to stroking the boundary valves.

2. Lowered inventory conditions:

   A. Efforts shall be made to minimize periods of lowered inventory conditions. **(CM-3)**

   B. High-risk activities and major work on electrical distribution systems should be deferred to periods other than during a lowered inventory condition, if possible. An evaluation of the risk and impact shall be performed if this condition cannot be met.

3. To consider Core Spray or LPCI pumps available:

   A. The watertight doors must be available to be closed (when there is a large volume of water in the torus that will result in an overflow from the torus basement into the corner rooms due to a torus leak) <u>AND</u>

   B. Torus level above 10'4" or CST contains 140,000 gal. (230,000, if the other unit is running) <u>AND</u>

   C. The discharge lines are maintained full <u>AND</u>

   D. During short time to boil or lowered inventory conditions, a diesel generator must be available to power the LPCI and Core Spray pumps to consider them available.

4. CRD, Condensate Transfer, Clean Demin, and the Fire System are considered emergency sources of makeup and are not considered available systems during normal shutdown. These systems may be used as part of contingency plans.

5. If LPCI Injection valves are unavailable in one loop, LPCI Loop Select logic will be forced to select the available loop as the injection path for LPCI to be considered available.

6. Make up pump control switches may be in pull-to-lock (PTL) and still be considered 'available' as long as there are no Clearance Order cards preventing the use of the pump.

7. ECCS system may be considered available as an injection system even with its full flow test valve and/or minimum flow valve OOS, provided the ECCS system is otherwise available.

   A. Due to the limited flow through the minimum flow valve, it may be OOS open or closed to consider the pump available.

   B. Due to the high flow through the test valve, it can only be taken OOS in the closed position to be able to consider the pump available.

4.7.2. Assessment of Inventory Control Shutdown Safety Color

1. Primary Sources

   A. ‘A’ Core Spray will be counted any time it is available.

   B. ‘B’ Core Spray will be counted any time it is available.

   C. ‘A’ LPCI will be counted any time it is available.

   D. ‘B’ LPCI will be counted any time it is available.

   E. ‘C’ LPCI will be counted any time it is available.

   F. ‘D’ LPCI will be counted any time it is available.

   G. The Condensate System will be counted as one source any time one pump is running or the system is vented and pressurized and a flow path to the reactor, including a source of water, are available.

2. Applicable Modes: 3

   A. Determine the availability of the following systems:

      1. Core Spray Systems - 1 Point each

      2. LPCI Sub Systems - 1 Point each
         – A Sub System = at least 1 pump and an injection path.

      3. Condensate System - 1 Point total

   B. Determine if any Potential to Drain activities or High Risk Activities affecting the Inventory Control Key Safety Function are in progress.

   C. Use the table below to determine the Shutdown Safety Color:

Inventory Control Key Safety Function **Applicable Modes: 3**		
NO POTENTIAL TO DRAIN OR HIGH RISK ACTIVITIES IN PROGRESS	**POTENTIAL TO DRAIN OR HIGH RISK ACTIVITIES IN PROGRESS**	**ASSIGNED SAFETY LEVEL COLOR**
4 points	5 points	GREEN
3 points	4 points	YELLOW
2 points	3 points	ORANGE
1 point	2 points	RED

3. Applicable Modes: 4 and 5

   A. Determine the availability of the following pumps and flow paths:

      1. Core Spray Pumps and flow paths - 1 point each

      2. LPCI Pumps with flow paths - 1 point each

      3. The Condensate System - 1 point ONLY

   B. Determine if the Fuel Pool Gates are in or out.

   C. Determine if any Potential to Drain activities or High Risk Activities affecting the Inventory Control Key Safety Function are in progress.

   D. Use the table below to determine the Shutdown Safety Color:

Inventory Control Key Safety Function Applicable Modes: 4 and 5				
FUEL POOL GATES IN (CM-3)		**FUEL POOL GATES OUT**		
NO POTENTIAL TO DRAIN OR HIGH RISK ACTIVITIES IN PROGRESS	**POTENTIAL TO DRAIN OR HIGH RISK ACTIVITIES IN PROGRESS**	**NO POTENTIAL TO DRAIN OR HIGH RISK ACTIVITIES IN PROGRESS**	**POTENTIAL TO DRAIN OR HIGH RISK ACTIVITIES IN PROGRESS**	**ASSIGNED SAFETY LEVEL COLOR**
3 points	4 points	2 points	3 points	GREEN
2 points	3 points	1 point	2 points	YELLOW
1 point	2 points	0 points	1 point	ORANGE
0 points	1 point	N/A	0 points	RED

4. Applicable Modes: De-fueled

   A. Determine the availability of the following pumps and flow paths:

      1. Core Spray Pumps and flow paths - 1 point each

      2. LPCI, pumps with a flow path - 1 point each

      3. The Condensate System - 1 point ONLY

   B. Determine if any Potential to Drain activities or High Risk Activities affecting the Inventory Control Key Safety Function are in progress.

   C. Use the table below to determine the Shutdown Safety Color:

Inventory Control Key Safety Function Applicable Modes: De-fueled		
NO POTENTIAL TO DRAIN OR HIGH RISK ACTIVITIES IN PROGRESS	**POTENTIAL TO DRAIN OR HIGH RISK ACTIVITIES IN PROGRESS**	**ASSIGNED SAFETY LEVEL COLOR**
1 point	$\geq$2 points	GREEN
0 points	1 point	YELLOW
N/A	0 points	ORANGE

4.8. **Vital Support Key Safety Function**

4.8.1. Guidelines

1. 2/3 RBCCW pump will be considered available when either 4KV feed from bus 24-1 or 34-1 is available.

2. 2/3 service water pump will be considered available when either 4KV feed from bus 24 or 34 is available.

4.8.2. Assessment of Vital Support Shutdown Safety Color

1. Primary Sources

   A. Unit 2(3)A RBCCW Pump will be counted whenever it is available.

   B. Unit 2(3)B RBCCW Pump will be counted whenever it is available.

   C. The 2/3 RBCCW Pump will be counted whenever it is available.

   D. 2A Service Water Pump will be counted whenever it is available.

   E. 2B Service Water Pump will be counted whenever it is available.

   F. 3A Service Water Pump will be counted whenever it is available.

   G. 3B Service Water Pump will be counted whenever it is available.

   H. 2/3 Service Water Pump will be counted whenever it is available.

2. Applicable Modes: 3, 4, 5, & De-fueled

   A. Determine the availability of the following cooling trains:

      1. Unit 2, 2/3, and 3 Service Water Pumps. - 1 Point each

      2. Outage Unit and 2/3 RBCCW Pumps. - 1 Point each

   B. Determine if any High Risk Activities affecting the Vital Support Key Safety Function are in progress.

   C. Use the table below to determine the Shutdown Safety Color:

Vital Support Key Safety Function **Applicable Modes: 3, 4, 5, & De-fueled**				
RBCCW PUMPS		**SERVICE WATER PUMPS**		
NO HIGH RISK ACTIVITIES IN PROGRESS	**HIGH RISK ACTIVITIES IN PROGRESS**	**NO HIGH RISK ACTIVITIES IN PROGRESS**	**HIGH RISK ACTIVITIES IN PROGRESS**	**ASSIGNED SAFETY LEVEL COLOR**
3 points	N/A	4 points	5 points	GREEN
2 points	3 points	3 points	4 points	YELLOW
1 point	2 points	2 points	3 points	ORANGE
0 points	≤1 point	≤1 point	≤2 points	RED

4.9. Reactivity Control Key Safety Function

4.9.1. Guidelines

1. The Reactivity Control Key Safety Function identifies specific equipment, which is or is not available, to determine the risk level. Because specific equipment is identified, the point system is NOT used.

2. All reactivity control actions are planned and well controlled with procedures and the Unit Supervisor in complete command and control. Any manipulations, which affect any parameter of reactivity, are monitored to ensure reactivity is added from a single source only.

3. All transfer of special nuclear material and reactivity control shall be in accordance with approved Move Sheets per NF-AA-310.

4. Work or testing, which does not impact the 'all rods in' condition or indication, may be done at any time.

5. Prior to control rod withdrawal from an empty cell, that cell shall be verified empty of its 4 fuel bundles (T.S. 3.10.5). All rods must be inserted to reload fuel (other than spiral reload following a full core offload).

6. Prior to control rod withdrawal from a cell containing fuel assemblies in Mode 5, core verification will be performed per NF-AA-330 AND either analytical Shutdown Margin (SDM) of 0.38 ΔK/K will be verified OR a Single Rod Sub-Critical Demonstration will be performed, along with the remaining actions required by T.S. 3.10.4.

    A. The Core Verification from the previous cycle remains valid until any fuel assembly has been added to the core OR any fuel assembly has been shuffled in the core.

    B. Analytical SDM is assumed to be met at all times during fuel moves based on the evaluation performed prior to the start of the fuel moves. If a bundle were to be mispositioned or unable to be placed in the desired core location, the shuffle would stop until an evaluation is performed by NF.

7. SRM's will only be counted if two (2) or more are available.

8. Fuel Moves are defined as any movement of irradiated fuel bundles over irradiated fuel in the reactor vessel or fuel pool which have the potential to damage fuel.

9. Core Alterations are defined as per Technical Specification 1.1. The movement of control rods in an empty cell is not considered a core alteration.

10. All Rods In is defined as all rods fully inserted, regardless of whether the cell contains fuel assemblies or is empty.

4.9.2. Assessment of Reactivity Control Shutdown Safety Color

1. Primary Sources

    A. Source Range Monitors

    B. All Rods Inserted

    C. Neutron Monitoring Shorting Links

    D. Shutdown Margin Demonstration

    E. Single Rod Sub-criticality Demonstration

    F. Grapple Refuel Interlocks

    G. One Rod Out Interlock

    H. Rod Block Inserted

2. Applicable Modes: 3 & 4

    A. Determine the availability of the following:

        1. All Rods Inserted (Window Green regardless of the status of One Rod Out Permit or SRMs Available.)

        2. Rod Block Inserted (Window Green regardless of the status of One Rod Out Permit or SRMs Available.)

        3. One Rod Out Permit

        4. SRMs Available

    B. Determine if any High Risk Activities affecting the Reactivity Control Key Safety Function are in progress.

C. Use the table below to determine the Shutdown Safety Color:

Reactivity Control Key Safety Function Applicable Modes: 3 & 4				
NO HIGH RISK ACTIVITIES IN PROGRESS		**HIGH RISK ACTIVITIES IN PROGRESS**		**ASSIGNED SAFETY LEVEL COLOR**
All rods inserted OR Rod out Block inserted	**N/A**	**All rods inserted OR Rod out Block inserted**	**N/A**	**GREEN**
One Rod Out Permit Available	**≥ 3 SRMS**	**One Rod Out Permit Available**	**≥ 3 SRMS**	**GREEN**
One Rod Out Permit Available	**2 SRMS**	**N/A**	**N/A**	**YELLOW**
One Rod Out Permit Available	**<2 SRMS**	**N/A**	**N/A**	**ORANGE**
One Rod Out Permit Unavailable	**N/A**	**One Rod Out Permit Available**	**≤ 2 SRMS**	**RED**
N/A	**N/A**	**One Rod Out Permit Unavailable**	**N/A**	**RED**

3. Applicable Mode: 5 with NO Fuel Moves in the RPV OR Core Alterations

   A. Determine the availability of the following:

      1. All Rods Inserted

      2. Rod Block Inserted

      3. One Rod Out Permit

   B. Use the table below to determine the Shutdown Safety Color:

Reactivity Control Key Safety Function **Applicable Mode: 5 with NO Fuel Moves in the RPV OR Core Alterations**	
CONTROL ROD INTERLOCKS	**ASSIGNED SAFETY LEVEL COLOR**
All Rods Inserted OR Rod Block Inserted OR One Rod Out Permit Available	**GREEN**
All Rods NOT Inserted AND Rod Block NOT Inserted AND One Rod Out Permit Unavailable	**YELLOW**

4. Applicable Modes 5 with Fuel Moves in the RPV and/or Core Alteration:

   A. Determine the availability of the following equipment/condition:

      1. All Rods Inserted

      2. Rod Block Inserted

      3. One Rod Out Permit

      4. SRMs Available

      5. Refueling Interlocks

      6. Shutdown Margin

         – During and in between fuel moves (fuel shuffle #1 & #2) – Analytical SDM of .38 ΔK/K shown for every core configuration change that places a fuel assembly into a new core location (Refer to step 4.9.1.6.B).

         – After all fuel moves (completion of fuel shuffle #2) OR prior to pulling a control rod in a fueled cell –

           – Core Audit Completed AND Analytical SDM of .38 ΔK/K

             OR

           – Single Rod Sub-Critical Demonstrated.

   B. Determine if any High Risk Activities affecting the Reactivity Control Key Safety Function are in progress. **Any High Risk Activities will result in a RED Window.**

C. Use the table below to determine the Shutdown Safety Color:

Reactivity Control Key Safety Function **Applicable Mode: 5 with Fuel Moves in the RPV AND/OR Core Alterations**		
Fuel moves and Control Rod Movement <u>OR</u> during CRD Removal with fuel in the cell	**When Cycling Control Rods <u>OR</u> when Fuel is moved with NO rod movement**	**ASSIGNED SAFETY LEVEL COLOR**
N/A	**7.5**	**GREEN**
7	**7**	**YELLOW**
6	**6**	**ORANGE**
5	**5**	**RED**
Any High Risk Activity		**RED**

<u>Value</u> <u>Safety Function Paths/Systems/Method</u>

1.0 At least 2 SRM indications, alarms, and rod blocks functional. During Core Alterations, 1 SRM must be functional in the Core Alteration quadrant and another SRM in an adjacent quadrant.

0.5 An additional ½ point may be credited if a minimum of 3 SRM indications, alarms, and rod blocks are functional supporting Core Alterations.

1.0 "One Rod Out" interlock functional (may be bypassed per T.S. 3.10.5 for associated rods), or Refueling interlocks functional as required by Technical Specification, or Mode Switch controlled in Shutdown as required by refueling procedures.

5.0 "Control Rods Fully Inserted" (NOTE: Can take credit for this with a single control rod removed from a fueled cell per Tech Spec, or multiple rods in defueled cells removed per Tech Spec, or with single control rod cycling in progress).

-OR-

4.0 All Control Rods <u>NOT</u> fully inserted <u>AND</u> Shutdown Margin <u>IS</u> met.

NOTES:

1. Reactivity Control level can be no better than Yellow when performing either of the following (maintain high level of sensitivity and awareness during multiple evolutions that directly impact or could impact reactivity in an event of an error):

   - Fuel moves (in RPV or fuel pool) and Control Rod movement.

   - OR -

   - Control Rod Drive removal in a loaded cell as allowed by Tech Specs

2. This Reactivity Control assessment does not apply for:

   - Control Rod Blade Removal in unloaded fuel cells
   - Control Rod Drive Mechanism Removal in unloaded fuel cells
   - Cycling drives in unloaded fuel cells

4.10. **Containment Key Safety Function**

4.10.1. Guidelines

1. The Containment Key Safety Function identifies specific equipment, which is or is NOT available to determine the risk level. Because specific equipment is identified, the point system is NOT used.

2. DAP 07-44, Control of Temporary Openings in Secondary Containment During Performance of Work Packages, Surveillances, or Other Procedures, shall control all openings in Secondary Containment.

3. The following guidelines must be followed when breaching secondary containment: **(CM-3)**

   A. Breach of secondary containment should be avoided during short time to boil periods.

   B. The time to secure secondary containment shall not exceed the time to boil.

   C. Approved written instructions ready for re-establishing secondary containment.

   D. Operations and work group are briefed.

4.10.2. Assessment of Containment Shutdown Safety Color

1. Primary Sources

   A. Primary Containment will be counted available whenever the physical condition is in compliance with Technical Specifications.

   B. Secondary Containment will be counted available when one of the following conditions is met.

      1. Whenever the physical condition is in compliance with Technical Specifications **(CM-2)**

      2. During events of short duration where Secondary Containment dp is less than .25 inch of vacuum water gauge. This condition will NOT be considered a High Risk Activity.

      3. Whenever the physical condition is NOT in compliance with Technical Specifications for less than 4 hours. This condition will be considered a High Risk Activity.

   C. SBGT 'A' will be counted whenever it is available.

   D. SBGT 'B' will be counted whenever it is available.

2. Applicable Modes: 3

   A. Determine the availability of:

      1. Primary Containment

      2. Secondary Containment

      3. Standby Gas Treatment Trains

   B. Determine if any Potential to Drain activities or High Risk Activities affecting the Containment Key Safety Function are in progress.

C. Use the table below to determine the Shutdown Safety Color:

Containment Key Safety Function Applicable Modes: 3			
NO POTENTIAL TO DRAIN OR HIGH RISK ACTIVITIES IN PROGRESS	**POTENTIAL TO DRAIN OR HIGH RISK ACTIVITIES IN PROGRESS**	**STANDBY GAS TREATMENT TRAINS AVAILABLE**	**ASSIGNED SAFETY LEVEL COLOR**
Primary Containment AND Secondary Containment Available	N/A	2	GREEN
Primary Containment AND Secondary Containment Available	N/A	1	YELLOW
N/A	Primary Containment AND Secondary Containment Available	2	YELLOW
N/A	Primary Containment AND Secondary Containment Available	1	ORANGE
Primary Containment AND Secondary Containment Available	Primary Containment AND Secondary Containment Available	0	RED
Primary Containment OR Secondary Containment Unavailable	Primary Containment OR Secondary Containment Unavailable	N/A	RED

3. Applicable Modes: 4, 5 and De-fueled with NO Potential to Drain Activities in progress.

   A. Determine the availability of:

      1. Secondary Containment

      2. Standby Gas Treatment Trains

   B. Use the table below to determine the Shutdown Safety Color:

Containment Key Safety Function **Applicable Modes: 4, 5 and De-fueled with NO Potential to Drain Activities in progress.**		
SECONDARY CONTAINMENT AVAILABLE	**STANDBY GAS TREATMENT TRAINS AVAILABLE**	**ASSIGNED SAFETY LEVEL COLOR**
YES	≥1	GREEN
YES	0	YELLOW
NO	N/A	YELLOW

4. Applicable Modes: 4, 5, with Potential to Drain Activities in progress

   A. Determine the availability of:

      1. Secondary Containment

      2. Standby Gas Treatment Trains

   B. Determine if any High Risk Activities affecting the Containment Key Safety Function are in progress.

   C. Use the table below to determine the Shutdown Safety Color:

Containment Key Safety Function **Applicable Modes: 4, 5, and De-fueled with Potential to Drain Activities in progress**			
NO HIGH RISK ACTIVITIES IN PROGRESS	**HIGH RISK ACTIVITIES IN PROGRESS**		
SECONDARY CONTAINMENT AVAILABLE	**SECONDARY CONTAINMENT AVAILABLE**	**STANDBY GAS TREATMENT TRAINS AVAILABLE**	**ASSIGNED SAFETY LEVEL COLOR**
YES	N/A	2	GREEN
YES	N/A	1	YELLOW
N/A	YES	2	YELLOW
N/A	YES	1	ORANGE
YES	YES	0	RED
NO	NO	N/A	RED

5. **DISCUSSION:**

5.1.1. Equipment Availability Through Simple Operator Actions **(CM-3)**

1. Examples of what may be considered as simple operator actions:

    A. Manually opening and closing a MOV

    B. Valving in a pump (open suctions and/or discharge valve) as long as the pump was not OOS for maintenance or drained

    C. Installation and removal of electrical jumper to bypass interlocks

    D. Turning a 480V MCC breaker on or off

    E. Energizing temporary power source that has been installed and verified functional

2. Examples of what may **NOT** be considered as simple operator actions:

    A. Fill and vent of a system/equipment after draining for maintenance or testing.

    B. Clear a tag out and return equipment to service.

    C. Hooking up temporary power or TMOD

    D. Equipment trip or system isolation which requires troubleshooting

    E. Going on backfeed or coming off backfeed

5.1.2. Safety Level Colors

1. **GREEN:** Based on the combination of available pathways and activity types, a failure or error could be easily mitigated without presenting a significant challenge in that Key Safety Function. This represents optimal defense-in-depth with all or nearly all mitigation equipment available. Generally this means that there are at least N+2 pathways.

2. **YELLOW:** Based on the combination of available pathways and activity types, a failure or error can still be mitigated but might present a challenge in that Key Safety Function. This represents minimal defense-in-depth with more than the minimum of "N" pathways available. There is generally some redundancy – at least N+1 pathways.

3. **ORANGE**: Based on the combination of available pathways and activity types, a failure or error would potentially lead to the loss of the Key Safety Function. This represents no defense-in-depth, i.e., generally only N (minimum pathways) are available to provide the safety function.

4. **RED**: Based on the combination of available pathways and activity types, the Key Safety Function is potentially not maintained. This represents a condition in which the safety function is not supported relative to its success criteria, i.e., generally fewer than N pathways available.

5.1.3. AC Power

1. Due the importance of the AC power KSF and its impact on other KSF, the risk level in certain situations was reduced to the next level as a conservative approach and to raise level of awareness. Examples:

   A. The point system allows the color to be yellow with NO DG or NO off site power available. The decision was made to default to no better than orange in this condition.

   B. The point system allows the AC KSF to remain GREEN with one of the 4KV divisions not available. The decision was made to be no better than yellow in this condition.

2. The opposite unit SBO DG may be considered as a diverse source in a contingency plan but will not be credited as an AC source. The following restrictions apply:

   A. **Unit 2** - to allow diversity group credit for SBO DG3, both EDG3 and the Div. 1 cross-tie must also be available.

   B. **Unit 3** - to allow diversity group credit for SBO DG2, both EDG2 and the Div. 1 cross-tie must also be available

3. The SBO diesel generators are considered as a diverse power supply as compared to the emergency diesel generators:

   A. The SBO DGs are in different locations.

   B. The SBO DGs are air cooled versus water-cooled and have a different engine configuration.

   C. The SBO DGs controls are different from the EDG (electronic versus electrical).

   D. The SBO DGs governing system is different from the EDG (electronic versus mechanical).

5.1.4. Decay Heat Removal

1. Long time to boil is limited to the period when the reactor cavity is flooded and fuel pool gates are removed. Once the gates are removed, reactor cavity time to boil increases to approximately 10 hours. This time is sufficient to allow actions to restore decay heat removal capability. As such, the time to boil is considered long.

2. RWCU availability:

    A. EC #379206 was performed by Engineering to determine RWCU heat removal capability. The results indicate a wide range of decay heat removal capabilities dependant on the reactor water temperature, RWCU mode of operation (normal or blowdown) and RBCCW temperature.

        1. Normal mode of operation (recirculation back to the reactor) – the heat removal capability is low such that no credit will be taken for system operation in this mode.

        2. Blow down mode – EC #379206 provides tables and graphs for the various conditions to be considered to determine the RWCU heat removal capability. Restrictions were included in the calculations to avoid system high temperature isolation (150° F) and prevent exceeding heat exchanger design temperature. The calculations were based on 70° F RBCCW temperature.

    B. In addition to EC #379206, current reactor (and fuel pool, if the fuel pool gates are removed) decay heat values will also be needed to determine if RWCU can be credited as a viable decay heat removal system. These values are normally provided by Nuclear Engineering prior to each refuel outage as part of ADHR calculations. For all other outages, new calculations will have to be provided by Nuclear Engineering.

    C. The system will be given a maximum of ½ point if its heat removal capability as listed in EC #379206 is ≥ ½ the current reactor (and fuel pool, if the fuel pool gates are removed).

3. Use of ERVs as a decay heat removal system is controlled under DOP 1000-07, Alternate Shutdown Cooling. It is limited to ONLY modes 3 & 4 and requires the availability of at least 1 LPCI pump and 2 CCSW pumps for RPV injection and torus cooling. A minimum of 2 ERVs required ensures redundancy and decay heat removal capability.

4. Use of LPCI as a decay heat removal system was evaluated under EC evaluation # 381208. The EC is based on the use of torus water for make up to the reactor without torus cooling. Additional restriction is added in the procedure to require CCSW for torus cooling to ensure long term availability of LPCI in the drain down mode as a decay heat removal system. One LPCI loop will be used to drain down the reactor through SDC while the other LPCI loop will be used for vessel make up and torus cooling. The system will be placed in operation using applicable operating procedures. Use of LPCI/CCSW is directed under OP-DR-104-1001 and DOP 1900-03 andis restricted to mode 4 & 5 by operations due to the risk in mode 3.

5.1.5. **Attributes of Excellent Defense-in-Depth Programs:** (as listed in INPO 06-008, Guidelines for the Conduct of Outages at Nuclear Power Plants)

1. Station management is involved in developing, monitoring, and validating the outage defense-in-depth plan.
2. The defense-in-depth program is proceduralized and establishes system and support system requirements for each safety function, contingency systems, mitigation strategies, and training for station personnel, especially operations and supplemental personnel.
3. Shutdown safety is integrated into the outage schedule to ensure sufficient defense-in-depth. Independent reviews verify that the defense-in-depth plan is appropriate.
4. Compliance with the defense-in-depth plan is verified at least once per shift and before major safety systems or components are removed from service.
5. The shutdown safety program is designed such that as much defense-in-depth as is achievable is established and maintained.
6. Contingency plans and temporary measures are used to improve defense-in-depth when required safety system availability drops below the planned defense-in-depth level.
7. Emergent work, expanded scope, and major schedule changes are reviewed prior to schedule issuance to ensure defense-in-depth levels are maintained.
8. Clear ownership for shutdown safety is established within the line organization.
9. Defense-in-depth and outage risks are clearly communicated and understood at appropriate levels of the organization.

6. COMMITMENTS

6.1. CM-1, Letter Dated November 18, 1996 from John B. Hosmer to US NRC. Subject “Response to NRC Final Report on Spent Fuel Storage Pool Safety Issues” (Step 3.7.1.2). Commitment Change Tracking #09-19.

6.2. CM-2, Letter RS-05-114 dated August 22, 2005 (Step 3.11.2.1.B.1)

6.3. CM-3, SOER 09-1, Shutdown Safety

7. REFERENCES

7.1. OU-AA-103, Shutdown Safety Management Program

7.2. OU-AA-103-1001, Shutdown Safety Plan Independent Reviews

7.3. OP-DR-104-1001, Shutdown Risk Management Contingency Plans

7.4. OP-AA-106-101-1001, Event Response Guidelines

7.5. OP-AA-108-117, Protected Equipment Program

7.6. NF-AA-330, Special Nuclear Material Physical Inventories

7.7. ER-AA-600-1023, ORAM-SENTINEL and PARAGON Model Capability

7.8. ER-AA-600-1043, Shutdown Risk Management Interface

7.9. OP-AA-108-107-1002, Agreement Between Exelon Energy Delivery and Exelon Generation for Switchyard Operations

7.10. CRM Update Form DR-CRM-009

7.11. FASA 89764-04, Self-Assessment Final Report

7.12. ATI #117038-11, Resolve Deficiencies Identified in Design Basis Maint. FASA

7.13. INPO SER 2-08, Reduced Shutdown Safety Margins

7.14. INPO SOER 09-1, Shutdown Safety

7.15. NUMARC 91-06, Guidelines for the Management of Planned Outages at Nuclear Power Plants

7.16. INPO 06-008, Guidelines for the Management of Planned Outages at Nuclear Power Plants

7.17. Dresden UFSAR Section 9.1.2.3.1 and 9.1.3.1

7.18. DOP 1000-07, Alternate Shutdown Cooling.

7.19 EC evaluation # 381208.

8. ATTACHMENTS

8.1. Attachment 1 – Unit 2 Equipment Availability

8.2. Attachment 2 – Unit 3 Equipment Availability

8.3. Attachment 3 – Decision Tree for Conditions Outside the Bounds of Shutdown Risk Management Procedures

8.4. Attachment 4 – High Risk Activities

8.5. Attachment 5 - Minimum Requirement to Prevent Risk Color Change

ATTACHMENT 1
Unit 2 Equipment Availability
Page 1 of 2

	Available / Yes	Unavailable / No		Available / Yes	Unavailable / No
Electrical Distribution			125vdc Bus 2A-2		
Unit 2 Res Aux Xfmr (TR22)			125vdc Bus 2B-1		
Unit Aux Xfmr Backfeed (TR21)			125vdc Bus 2B-2		
Unit 3 Res Aux Xfmr (TR32)			125vdc Rx Bldg Dist Pnl		
Unit 3 Aux Xfmr Backfeed (TR31)					
4kv Bus 23			125vdc Battery Chgr (1)		
4kv Bus 24			24/48vdc Battery		
4kv Bus 23-1			24/48vdc Bus 2A		
4kv Bus 24-1			24/48vdc Bus 2B		
4kv Bus 33			24/48vdc Battery Chgrs (4)		
4kv Bus 34			**Off Site Power**		
4kv Bus 33-1			345kv Line 0302		
4kv Bus 34-1			345kv Line 1220		
4kv Bus 23-1 to 33-1 Xtie			345kv Line 1221		
4kv Bus 24-1 to 34-1 Xtie			345kv Line 1222		
4kv Bus 61			345kv Line 1223		
4kv Bus 40			345kv Line 8014		
EDG 2			345kv Line 2311		
EDG 2/3 to unit 2			345kv Ring Bus Bkr 1-2		
SBO DG Unit 2			345kv Ring Bus Bkr 1-7		
EDG 3					
EDG 2/3 to unit 3					
SBO DG Unit 3					
480v Bus 28			345kv Ring Bus Bkr 2-3		
480v Bus 29			345kv Ring Bus Bkr 3-4		
480v Bus 28 to 29 Xtie			345kv Ring Bus Bkr 4-5		
480v MCC 28-1			345kv Ring Bus Bkr 5-6		
480v MCC 28-2			345kv Ring Bus Bkr 6-7		
480v MCC 28-3			345kv Ring Bus Bkr 4-8		
480v MCC 28-7			345kv Ring Bus Bkr 8-9		
480v MCC 29-1			345kv Ring Bus Bkr 8-15		
480v MCC 29-2			345kv Ring Bus Bkr 11-14		
480v MCC 29-3			345kv Ring Bus Bkr 10-11		
480v MCC 29-4			345kv Ring Bus Bkr 9-10		
480v MCC 29-7			345kv Ring Bus Disc 14-15		
480v MCC 29-8			138kv Line 0904		
480v MCC 29-9			138kv Line 1205		
Unit 2 Essential Service Bus			138kv Line 1210		
Unit 2 Instrument Bus			138kv Line 0903		
250vdc Battery			138kv Line 1206		
250vdc Bus 2			138kv Line 1207		
250vdc Bus 2A					
250vdc Bus 2B			138kv Bkr 1-2		
250vdc Battery Chgr (1)			138kv Bkr 2-3		
125vdc Main or Alt Battery			138kv Bkr 3-4		
125vdc Bus 2A-1					

ATTACHMENT 1
Unit 2 Equipment Availability
Page 2 of 2

	Available / Yes	Unavailable / No		Available / Yes	Unavailable / No
***Decay Heat Removal / Fuel Pool Cooling**					
SDC Inlet from Rx A loop			2A FPC Pump and Hx		
SDC Inlet from Rx B loop			2B FPC Pump and Hx		
SDC Return to Rx A loop			2A SDC to FPC		
SDC Return to Rx B loop			2B SDC to FPC		
SDC Pump 2A to Reactor			2C SDC to FPC		
SDC Pump 2B to Reactor			2 ERVs (circle)	B C D E	
SDC Pump 2C to Reactor			1 LPCI pump (circle)[1]	A B C D	
RWCU			CCSW pumps** (circle)[1]	A B C D	
Fuel Pool Gates Out			LPCI injection loops[1]	A B	
Containment					
Primary Containment					
Secondary Containment			**Reactivity Control**		
SBGT Train A			All Rods Fully Inserted		
SBGT Train B			SRM 21		
Inventory Control			SRM 22		
LPCI Pump 2A			SRM 23		
LPCI Pump 2B			SRM 24		
LPCI Pump 2C					
LPCI Pump 2D			Rod Block Manually Inserted		
Div 1 LPCI Injection			One Rod Out Permit		
Div 2 LPCI Injection			Refuel Interlocks		
Div 1 LPCI Crosstie					
Div 2 LPCI Crosstie			**Vital Support**		
LPCI Loop Select forced to Div 1			2A RBCCW Pump		
LPCI Loop Select forced to Div 2			2B RBCCW Pump		
Core Spray 2A			2/3 RBCCWPump		
Core Spray 2B			2A Serv Wtr Pump		
Torus level >10'4" **OR** CST >= 140,000 gals (230,000 gals if U3 is running)			2B Serv Wtr Pump		
Condensate System			2/3 Serv Wtr Pump		
2A Rx Feed Header			3A Serv Wtr Pump		
2B Rx Feed Header			3B Serv Wtr Pump		
Plant Status					
Mode					
Moderator Temperature					
Fuel Pool Temperature					
Time To Boil					
Time to Core Uncovery					
Protected Pathways					

* NLO and NSO briefed at the beginning of the shift for restoration of SDC if it were lost.

** One LPCI pump & two CCSW pumps (same division) required to credit ERV as a decay heat removal system

[1]Both LPCI injection loops, one LPCI and one CCSW pump (same division) required to credit LPCI (in drain down mode) as a decay heat removal system.

Completed By: ________________________ Date:____________ Time:___________

ATTACHMENT 2
Unit 3 Equipment Availability
Page 1 of 2

	Available / Yes	Unavailable / No		Available / Yes	Unavailable / No
Electrical Distribution			125vdc Bus 3B-1		
Unit 3 Res Aux Xfmr (TR32)			125vdc Bus 3B-2		
Unit Aux Xfmr Backfeed (TR31)			125vdc Rx Bldg Dist Pnl		
Unit 2 Res Aux Xfmr (TR22)			125vdc Battery Chgr (1)		
Unit 2 Aux Xfmr Backfeed (TR21)					
4kv Bus 23			24/48vdc Battery		
4kv Bus 24			24/48vdc Bus 3A		
4kv Bus 23-1			24/48vdc Bus 3B		
4kv Bus 24-1			24/48vdc Battery Chgrs (4)		
4kv Bus 33			**Off Site Power**		
4kv Bus 34			345kv Line 0302		
4kv Bus 33-1			345kv Line 1220		
4kv Bus 34-1			345kv Line 1221		
4kv Bus 23-1 to 33-1 Xtie			345kv Line 1222		
4kv Bus 24-1 to 34-1 Xtie			345kv Line 1223		
4kv Bus 71			345kv Line 8014		
4kv Bus 40			345kv Line 2311		
EDG 3			345kv Ring Bus Bkr 1-2		
EDG 2/3 to unit 3			345kv Ring Bus Bkr 1-7		
SBO DG Unit 3			345kv Ring Bus Bkr 2-3		
EDG 2					
EDG 2/3 to unit 2					
SBO DG Unit 2					
480v Bus 38			345kv Ring Bus Bkr 3-4		
480v Bus 39			345kv Ring Bus Bkr 4-5		
480v Bus 38 to 39 Xtie			345kv Ring Bus Bkr 5-6		
480v MCC 38-1			345kv Ring Bus Bkr 6-7		
480v MCC 38-2			345kv Ring Bus Bkr 4-8		
480v MCC 38-3			345kv Ring Bus Bkr 8-9		
480v MCC 38-4			345kv Ring Bus Bkr 8-15		
480v MCC 38-7			345kv Ring Bus Bkr 11-14		
480v MCC 39-1			345kv Ring Bus Bkr 10-11		
480v MCC 39-2			345kv Ring Bus Bkr 9-10		
480v MCC 39-3			345kv Ring Bus Disc 14-15		
480v MCC 39-7			138kv Line 0904		
Unit 3 Essential Service Bus			138kv Line 1205		
Unit 3 Instrument Bus			138kv Line 1210		
250vdc Battery			138kv Line 0903		
250vdc Bus 3			138kv Line 1206		
250vdc Bus 3A			138kv Line 1207		
250vdc Bus 3B					
250vdc Battery Chgr (1)			138kv Bkr 1-2		
125vdc Main or Alt Battery			138kv Bkr 2-3		
125vdc Bus 3A-1			138kv Bkr 3-4		
125vdc Bus 3A-2					

ATTACHMENT 2
Unit 3 Equipment Availability
Page 2 of 2

	Available / Yes	Unavailable / No		Available / Yes	Unavailable / No
***Decay Heat Removal / Fuel Pool Cooling**					
SDC Inlet from Rx A loop			3A FPC Pump and Hx		
SDC Inlet from Rx B loop			3B FPC Pump and Hx		
SDC Return to Rx A loop			3A SDC to FPC		
SDC Return to Rx B loop			3B SDC to FPC		
SDC Pump 3A to Reactor			3C SDC to FPC		
SDC Pump 3B to Reactor			2 ERVs (circle)	B C	D E
SDC Pump 3C to Reactor			1 LPCI pump (circle)[1]	A B	C D
RWCU			CCSW pumps** (circle)[1]	A B	C D
Fuel Pool Gates Out			LPCI injection loops[1]	A B	
Containment					
Primary Containment					
Secondary Containment			**Reactivity Control**		
SBGT Train A			All Rods Fully Inserted		
SBGT Train B			SRM 21		
Inventory Control			SRM 22		
LPCI Pump 3A			SRM 23		
LPCI Pump 3B			SRM 24		
LPCI Pump 3C					
LPCI Pump 3D			Rod Block Manually Inserted		
Div 1 LPCI Injection			One Rod Out Permit		
Div 2 LPCI Injection			Refuel Interlocks		
Div 1 LPCI Crosstie					
Div 2 LPCI Crosstie			**Vital Support**		
LPCI Loop Select forced to Div 1			3A RBCCW Pump		
LPCI Loop Select forced to Div 2			3B RBCCW Pump		
Core Spray 3A			2/3 RBCCW Pump		
Core Spray 3B			2A Serv Wtr Pump		
Torus level >10'4" **<u>OR</u>** CST >= 140,000 gals (230,000 gals if U2 is running)			2B Serv Wtr Pump		
Condensate System			2/3 Serv Wtr Pump		
3A Rx Feed Header			3A Serv Wtr Pump		
3B Rx Feed Header			3B Serv Wtr Pump		
Plant Status					
Mode					
Moderator Temperature					
Fuel Pool Temperature					
Time To Boil					
Time to Core Uncovery					
Protected Pathways					

* NLO and NSO briefed at the beginning of the shift for restoration of SDC if it were lost.

** One LPCI pump & two CCSW pumps (same division) required to credit ERV as a decay heat removal system

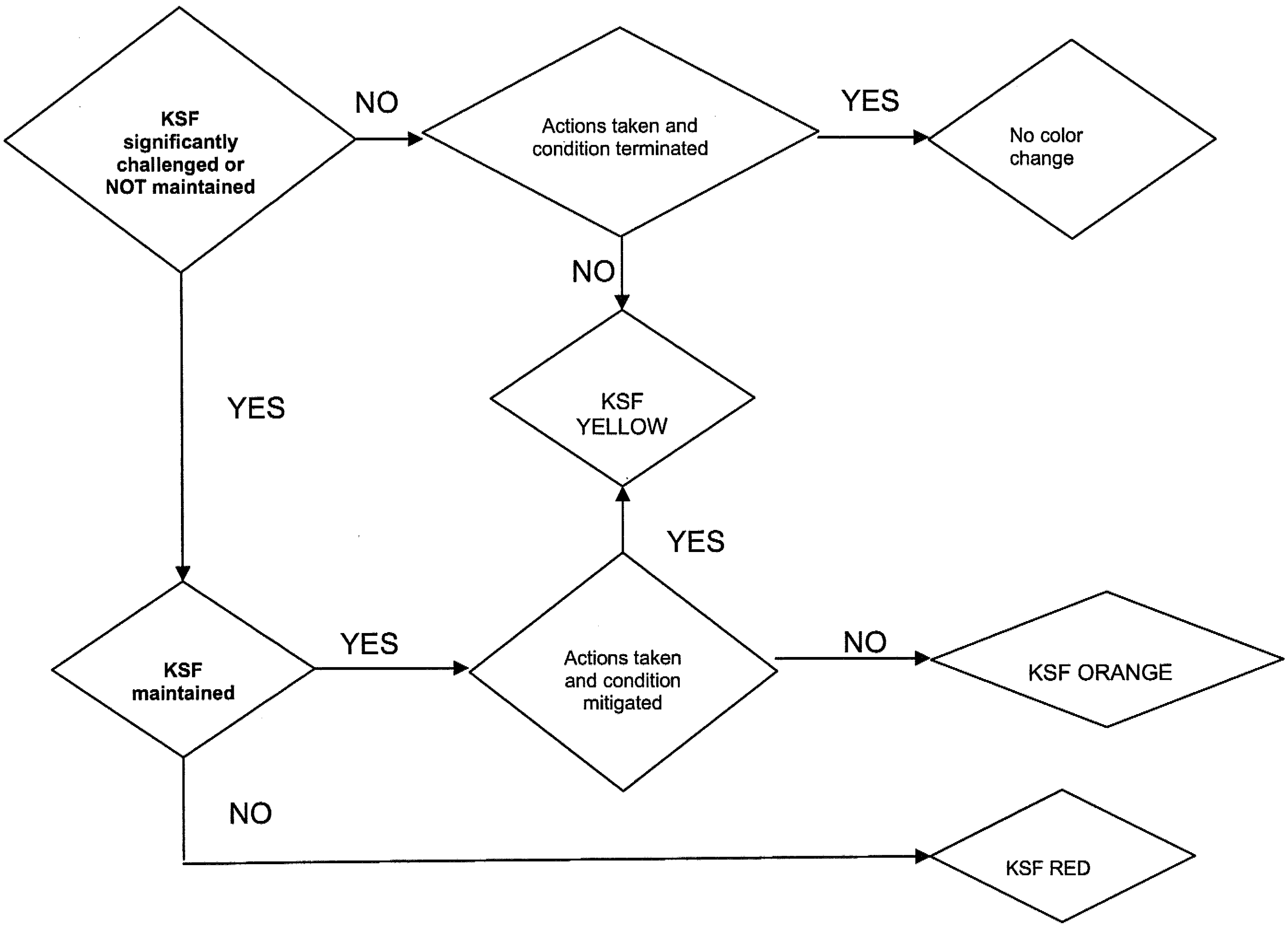
[1]Both LPCI injection loops, one LPCI and one CCSW pump (same division) required to credit LPCI (in drain down mode) as a decay heat removal system.

Completed By: ________________________ Date: ____________Time:___________

ATTACHMENT 3

Decision tree for conditions **outside the bounds**

of shutdown risk management procedures



NOTES

- Unlike normal shutdown risk assessment, this attachment addresses conditions after the fact and where the required equipment/condition is available but did not prevent the event.
- If the KSF color is already higher than what is determined in this evaluation, then the KSF SHALL remain at the higher status color.
- If a KSF color change results, then it will be in effect for the duration of the event. Once the event is terminated the KSF will return to the previous designation or as determined by the SSRB.
- Actions taken may be automatic or manual and must be completed within reasonable time.

ATTACHMENT 4

High Risk Activities

Activity Name*	Description	Comments**

* Activity name as listed in the schedule (ORAM hammock)

** Comments to address condition and applicable compensatory/contingency plans.

ATTACHMENT 5

Minimum Requirement to Prevent Risk Color Change

Key Safety Function	Conditions	Minimum Required	Comments
AC Power			
Shutdown Cooling			
Fuel Pool Cooling			
Inventory Control			
Vital Support			
Containment			

ATTACHMENT 5

Procedure OP-DR-104-1001, "Shutdown Risk Management Contingency Plans"

Exelon Nuclear

SHUTDOWN RISK MANAGEMENT CONTINGENCY PLANS

1. PURPOSE

1.1. To provide heightened awareness of plant status during outages.

1.2. To ensure that proper contingency plans are in place.

2. TERMS AND DEFINITIONS

2.1. Terms and definitions are as specified in OU-DR-104, Shutdown Safety Management Program.

3. POLICY

3.1. Industry experience has shown that plants can be susceptible to a variety of events that can challenge safety during shutdown conditions.

3.2. This T&RM is **not** intended for use as a procedure, but rather as a guideline for consideration in response to these types of events. Operations Management should be consulted during any use of these contingency plans.

3.3. The following events are of particular importance and establishing contingency plans to address them during an outage is the purpose of this document:

4.1 LOSS OF DECAY HEAT REMOVAL

4.2 LOSS OF FUEL POOL COOLING

4.3 LOSS OF REACTOR COOLANT INVENTORY (Fuel in vessel, Gates installed)

4.4 LOSS OF REACTOR COOLANT INVENTORY (Fuel in vessel, Gates open)

4.5 LOSS OF ONSITE/OFFSITE AC POWER

4.6 LOSS OF DC POWER

4.7 REACTIVITY CONTROL/SHUTDOWN MARGIN

4.8 CONTAINMENT INTEGRITY CONTROL

3.4. In addition to the contingency plans described, "Time To Core Uncovery" and "Time-To-Boil" curves for the vessel and Fuel Pool are provided as attachments. These curves provide a conservative estimate of the time frame to restore needed core inventory systems, shutdown cooling or fuel pool cooling equipment/systems to place the plant in a stable condition. Interpolation of the time to boil curves is allowed on the y-axis only, due to the linear relationship of the temperature curves.

- The "Time-To-Boil" estimate should be considered when determining which approach to take to address each event.
- The curves use conservative assumptions and will show boiling to occur sooner than it would actually happen.
- There are no specific curves showing a partial offload/reload. The Reactor Engineer shall review fuel pool curves (Attachments M and N) prior to a refueling outage to verify their applicability.
- The Reactor Engineer should always be consulted for additional guidance when the curves are used in response to a loss of cooling event.

4. MAIN BODY

4.1. LOSS OF DECAY HEAT REMOVAL:

4.1.1. **PERFORM** appropriate actions of DOA 1000-01, Residual Heat Removal Alternatives.

4.1.2. **If** there are any openings in secondary containment, **then** secure any breach **prior** to reaching 212°F.

4.1.3. Additional guidance may be necessary to minimize consequences of this event. As appropriate, **CONSIDER** the following:

1. **If** the Drywell Head has **not** been removed, **then** **CONSIDER** re-closing the equipment **and** personnel hatches (**REFER** to 4.8, Containment Integrity Control).
2. **If** the Fuel Pool is connected to the reactor cavity, **then** **CONSIDER** using Fuel Pool Cooling (DOP 1900-03, Reactor Cavity, Dryer/Separator Storage Pit And Fuel Pool Level Control).
   A. **If** a second Fuel Pool Cooling loop is to be placed in service, **then** **ENSURE** the Fuel Pool Filter/Demin Bypass Valve is opened to handle the increased flow.
3. **If** main stream drain lines are to be used as a drain path, **then** **ENSURE** MSIVs are closed to **PREVENT** flooding main steam lines.

NOTE: In Modes 4 or 5, a feed and bleed of reactor coolant can be utilized for decay heat removal. This method utilizes the drain down path from the reactor, Shutdown Cooling and LPCI piping to the Torus with makeup via the opposite LPCI Loop and the LPCI Heat Exchanger. The LPCI loops must be split to utilize this method. The LPCI loop with a heat exchanger must be selected for the injection loop. (W-6.3.15)

4. **If** a feed and bleed of reactor coolant can be utilized for decay heat removal, **then PERFORM** Feed and Bleed of the Cavity for Decay Heat Removal per DOP 1900-03.

4.2. LOSS OF FUEL POOL COOLING:

4.2.1. Response to this event will be in accordance with DOA 1900-01, Loss of Fuel Pool Cooling.

4.2.2. Additional guidance may be necessary to minimize the consequences of this event.

4.2.3. Worst-case temperature rise should be approximately 6.5°F per hour.

4.3. LOSS OF REACTOR COOLANT INVENTORY (Fuel in vessel, Gates installed):

4.3.1. Response to this event will be in accordance with the appropriate annunciator procedures, depending on plant conditions.

1. A loss of level in the reactor vessel/reactor cavity will be observed using various means and dealt with as appropriate.
2. Additional guidance may be necessary to minimize the consequences of this event.

4.3.2. As appropriate, **CONSIDER** the following:

1. Possible make-up water sources available to the Control Room Operator:
   A. Four (4) LPCI Pumps from the Suppression Pool.
   B. Two (2) Core Spray Pumps from the Suppression Pool.
   C. Four (4) Condensate Booster Pumps from the Main Condenser.
   D. Raising CRD Cooling Water flow (smaller source of water).
2. **If** plant conditions require alternate sources of make-up water, **then REFER** to DEOP 0500-03, Alternate Water Injection Systems.
3. Initiate actions to secure any temporary openings in secondary containment per DAP 07-44. (W-6.2.2)

4. **<u>If</u>** plant conditions allow, **<u>then</u> USE** various hose connections on the refuel floor as smaller make-up sources:

   A. Contaminated Demin – connections along south wall.

   B. Clean Demin – connections in floor boxes.

   NOTE: Service water (fire protection) is used as a last resort due to poor water quality.

   C. Service Water (Fire protection) – various connections.

4.3.3. Fuel Pool Gates will remain installed during this event to provide a dedicated source of water for the fuel in the Fuel Pool. This action removes the Fuel Pool Cooling Pumps (via the Condensate Transfer System) as a source of make-up to the Reactor Cavity.

4.4. <u>LOSS OF REACTOR COOLANT INVENTORY</u> (Fuel in vessel, Gates open):

NOTE: For Fuel Pool gates to be open, Main Steam Line plugs must be installed. This will minimize the consequences of any leaks from work on Main Steam Lines.

NOTE: Actions performed on the Refuel Floor (Elev 613) by the fuel handling supervisor concerning a loss of level in the Fuel Pool/reactor Cavity Shall be governed by DFP 0850-01, Slow Or Rapid Water Level Loss in Fuel Pool/reactor Cavity.

4.4.1. Response to this event will be in accordance with the appropriate annunciator procedures, depending on plant conditions.

1. A loss of level in the reactor vessel/reactor cavity will be observed using various means and dealt with as appropriate.

2. Additional guidance may be necessary to minimize the consequences of this event.

4.4.2. As appropriate, **CONSIDER** the following:

1. Possible make-up water sources available to the Control Room Operator:

   A. Four (4) LPCI Pumps from the Suppression Pool.

   B. Two (2) Core Spray Pumps from the Suppression Pool.

   C. Four (4) Condensate Booster Pumps from the Main Condenser.

   D. Two (2) Fuel Pool Cooling Pumps from the skimmer surge tank/Condensate Storage Tank (via the Condensate Transfer System).

      - Once Fuel Pool gates are installed (DFP 0850-01), Fuel Pool Cooling Pumps will no longer serve as a water source for the reactor cavity.

E. Raising CRD Cooling Water flow (smaller source of water).

2. **If** plant conditions require alternate sources of make-up water, **then** **REFER** to DEOP 0500-03, Alternate Water Injection Systems.

3. Initiate actions to secure any temporary openings in secondary containment per DAP 07-44. (W-6.2.2)

4.4.2 4. **If** plant conditions allow, **then** **USE** various hose connections on the refuel floor as smaller make-up sources:

A. Contaminated Demin – connections along south wall.

B. Clean Demin – connections in floor boxes.

NOTE: Service water (fire protection) is used as a last resort due to poor water quality.

C. Service water (Fire protection) – various connections.

5. **If** additional sources of water are needed for RPV makeup **or** Fuel Pool makeup, **then** CONSIDER using strategies listed in TSG-3, Operational Contingency Action Guidelines.

4.5. LOSS OF ONSITE/OFFSITE AC POWER:

4.5.1. **PERFORM** appropriate actions of DGA-12, Partial Or Complete Loss of AC Power.

4.5.2. Additional guidance may be necessary to minimize the consequences of this event. As appropriate, **CONSIDER** the following:

- **If** onsite **or** offsite AC Power is designated any condition other than "green" as determined by OU-DR-104, Shutdown Safety Management Program, **then** the Shift Manager (**or** designee) **PERFORM** a walkdown of appropriate essential AC components (such as switchgear, DG Rooms, and switchyards) during periods of increased plant risk.

4.6. LOSS OF DC POWER:

4.6.1. **PERFORM** appropriate actions of DOA 6900 series procedures.

4.6.2. Additional guidance may be necessary to minimize the consequences of this event.

4.7. REACTIVITY CONTROL/SHUTDOWN MARGIN:

NOTE: ALL fuel handling operations will be conducted carefully in accordance with appropriate Fuel Handling **and** Engineering procedures.

4.7.1. **If** an event concerning an abnormal reactivity control/shutdown margin occurs, **then CONSIDER** the following as conditions dictate:

1. **NOTIFY** the Shift Manager of ALL abnormal occurrences.

2. **If** it is believed that a criticality accident has occurred, **then INITIATE** appropriate actions of DOA 0010-06, Criticality Accident in a Special Nuclear Material Area.

4.7.2. Any time a fuel movement deviates from the Fuel Move Sheet, the Fuel Handling Supervisor shall **ENSURE** the following are completed:

1. **STOP** all fuel movements.

NOTE: An inadvertently miss-loaded fuel assembly should **not** be removed once it has been completely lowered **until** its removal from the core has been evaluated.

2. **VERIFY** fuel assemblies are in a safe, conservative location.

3. **NOTIFY** a QNE prior to resuming normal fuel movement.

4. **OBTAIN** a revised Fuel Move Sheet that reflects existing conditions.

4.8. <u>CONTAINMENT INTEGRITY CONTROL</u>:

NOTE: Primary **and** secondary containment are controlled per the applicable Technical Specifications.

4.8.1. **If** either of the following occur:

- A challenge to Primary **or** Secondary containment,
- Re-establishing Primary/Secondary Containment is required,

Then CONSIDER the following, as appropriate:

1. **PERFORM** appropriate actions of DOA 7500-01, Standby Gas Treatment System Fan Trip.
2. **PERFORM** appropriate actions of DOA 1600-10, Secondary Containment Verification / Restoration.
3. **CONTACT** Mechanical Maintenance to close the Drywell Equipment Hatch under an A priority.
   - Closure of only the inner equipment hatch skin is required.
   - The personnel hatch can be closed without assistance from maintenance.
4. **<u>If</u>** Secondary Containment can **<u>not</u>** be maintained, **<u>then</u>** immediately SUSPEND movement of recently irradiated fuel.
   - **<u>If</u>** the grapple is loaded **<u>when</u>** the Fuel Handling Supervisor is notified to suspend fuel movement, **<u>then</u> MOVE** the loaded grapple to its intended location **<u>or</u>** its original location, whichever presents the shortest time over irradiated fuel.

4.8.2. **MAINTAIN** control of temporary openings in Secondary Containment per DAP 07-44, Control Of Temporary Openings In Secondary Containment During Performance Of Work Packages, Surveillances, Or Other Procedures. Upon notification from appropriate shift personnel, **SEAL** any such openings immediately.

4.8.3. To access the reactor building via Unit 2 Trackway **or** Unit 3 Material interlock doors, **MAINTAIN** control per the applicable procedure:

- DAP 13-03, Unit 2 Reactor Building Trackway Interlock Door Access.
- DAP 13-14, Unit 3 Reactor Building Material Interlock Door Access Control.

1. Immediately **CLOSE <u>and</u> LATCH** any interlock door upon notification from appropriate shift personnel.

5. **DOCUMENTATION**

5.1. NONE.

6. **REFERENCES**

6.1. Commitments

NONE.

6.2. Other References:

6.2.1. EC 371913, Time to Boil Curves.

6.2.2. IGAP 959926-35-14, Revise OP-DR-104-1001, Shutdown Risk Management Contingency Plans, to provide direction to secure secondary containment upon loss of decay heat removal per INPO SOER 09-1, Shutdown Safety.

6.3. User References

6.3.1. DAP 07-44, Control Of Temporary Openings In Secondary Containment During Performance Of Work Packages, Surveillances, Or Other Procedures.

6.3.2. DAP 13-03, Unit 2 Reactor Building Trackway Interlock Door Access.

6.3.3. DAP 13-14, Unit 3 Reactor Building Material Interlock Door Access Control.

6.3.4. DEOP 0500-03, Alternate Water Injection Systems.

6.3.5. DFP 0850-01, Slow Or Rapid Water Level Loss in Fuel Pool/reactor Cavity.

6.3.6. DGA-12, Partial Or Complete Loss of AC Power.

6.3.7. DOA 0010-06, Criticality Accident In A Special Nuclear Material Area.

6.3.8. DOA 1000-01, Residual Heat Removal Alternatives.

6.3.9. DOA 1600-10, Secondary Containment Verification / Restoration.

6.3.10. DOA 1900-01, Loss of Fuel Pool Cooling.

6.3.11. DOA 6900 series procedures.

6.3.12. DOA 7500-01, Standby Gas Treatment System Fan Trip.

6.3.13. DOP 1900-03, Reactor Cavity, Dryer/Separator Storage Pit And Fuel Pool Level Control.

6.3.14. OU-DR-104, Shutdown Safety Management Program.

6.3.15. EC 381208, Alternative Decay Heat Removal Using Draindown Through SDC/LPCI.

7. ATTACHMENTS

7.1. Attachment A, Time-to-Boil with Head On.

7.2. Attachment B, Time-to-Core Uncovery with Head On.

7.3. Attachment C, Time-to-Boil while Flooded to Flange Dryer/Separator In.

7.4. Attachment D, Time-to-Core Uncovery while Flooded to Flange Dryer/Separator In.

7.5. Attachment E, Time-to-Boil while Flooded to Flange Dryer/Separator Removed.

7.6. Attachment F, Time-to-Core Uncovery while Flooded to Flange Dryer/Separator Removed.

7.7. Attachment G, Time-to-Boil while Flooded Up, Gates In; Dryer/Separator In (Dryer/Separator Pit not included).

7.8. Attachment H, Time-to-Core Uncovery while Flooded Up, Gates In; Dryer/Separator In (Dryer/Separator Pit not included).

7.9. Attachment I, Time-to-Boil while Flooded Up, Reactor and Dryer/Separator Pit.

7.10. Attachment J, Time-to-Core Uncovery while Flooded Up, Reactor and Dryer/Separator Pit.

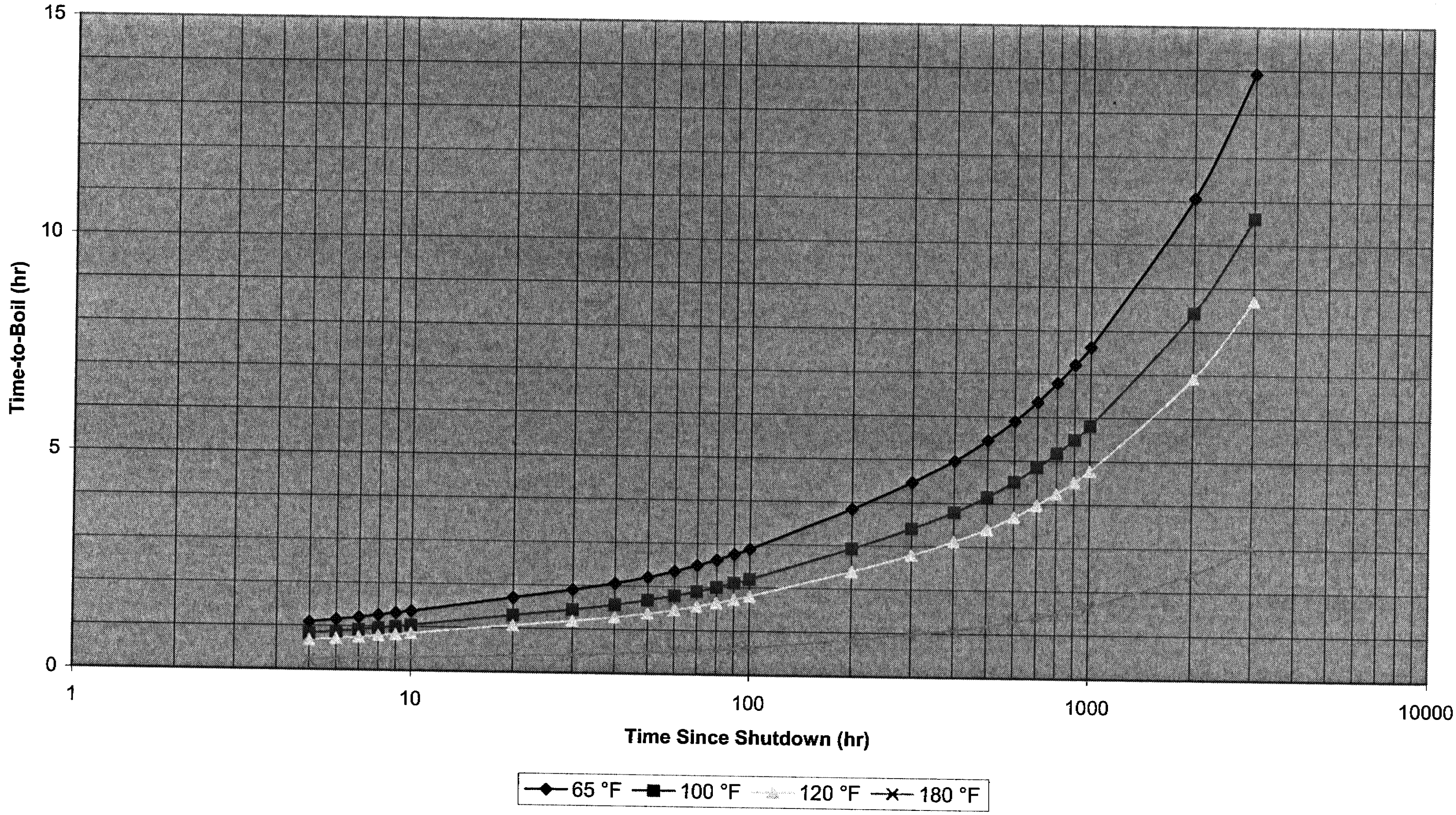
7.11. Attachment K, Time-to-Core Uncovery while Flooded Up, Reactor and Dryer/Separator Pit.

7.12. Attachment L, Time-to-Core Uncovery with Gates Out; Cavity, Dryer/Separator Pit, & Spent Fuel Pool.

7.13. Attachment M, Time-to-Boil for Spent Fuel Pool (Isolated).

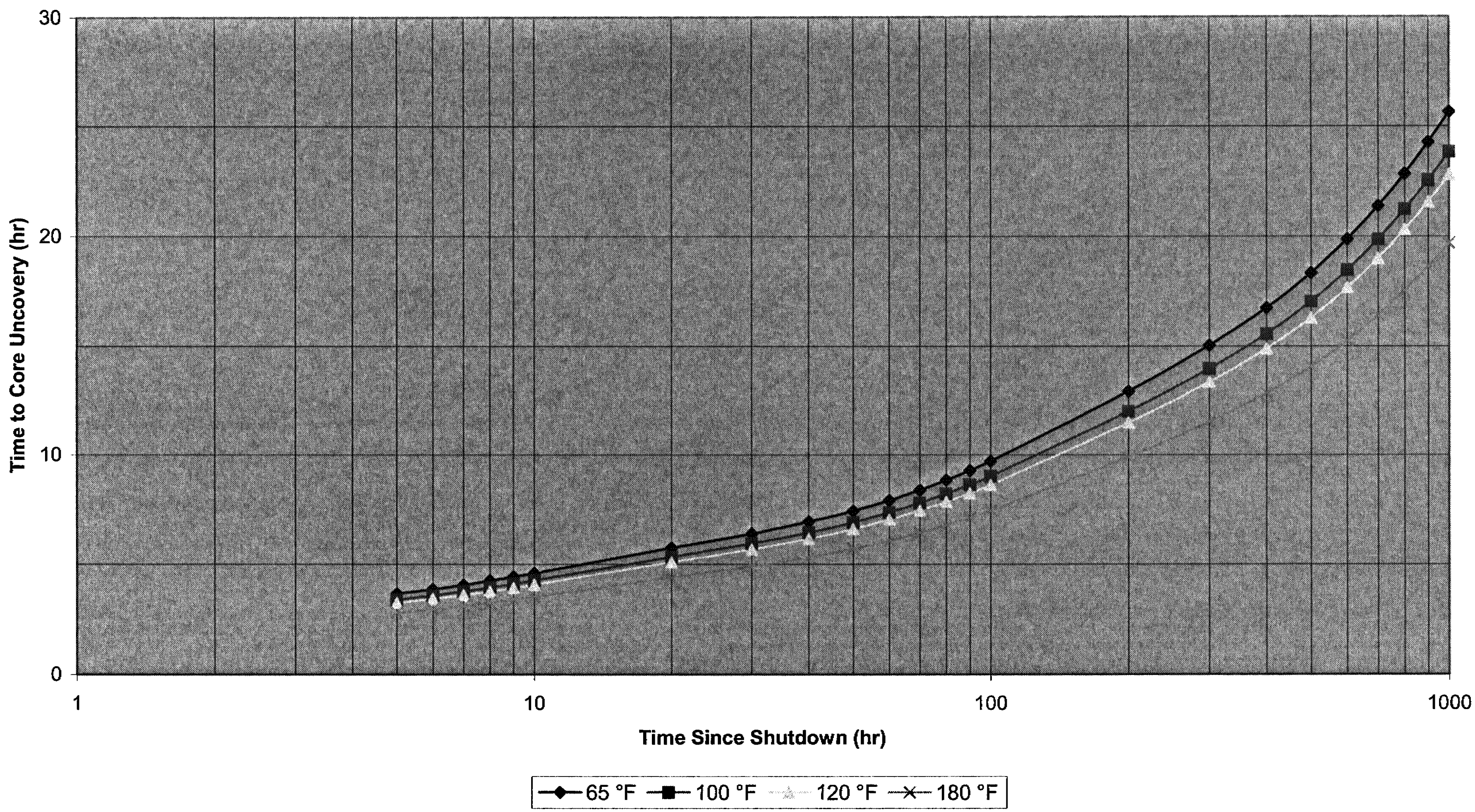
7.14. Attachment N, Time-to-Spent Fuel Pool Pump Cavitation Limit.

Attachment A
Time-to-Boil with Head On



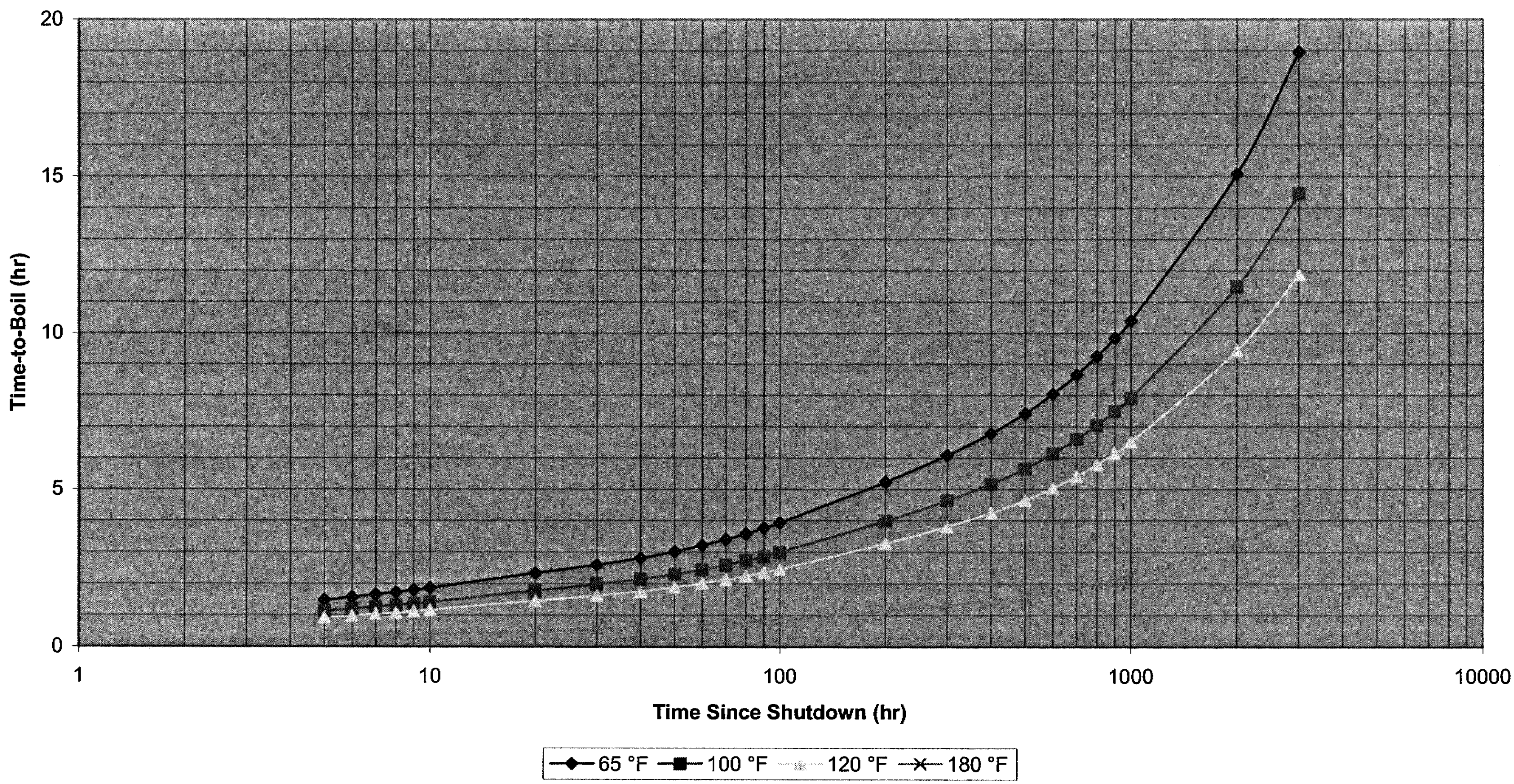
CHOOSE CURVE BASED ON INITIAL WATER TEMPERATURE

Attachment B
Time-to-Core Uncovery with Head On



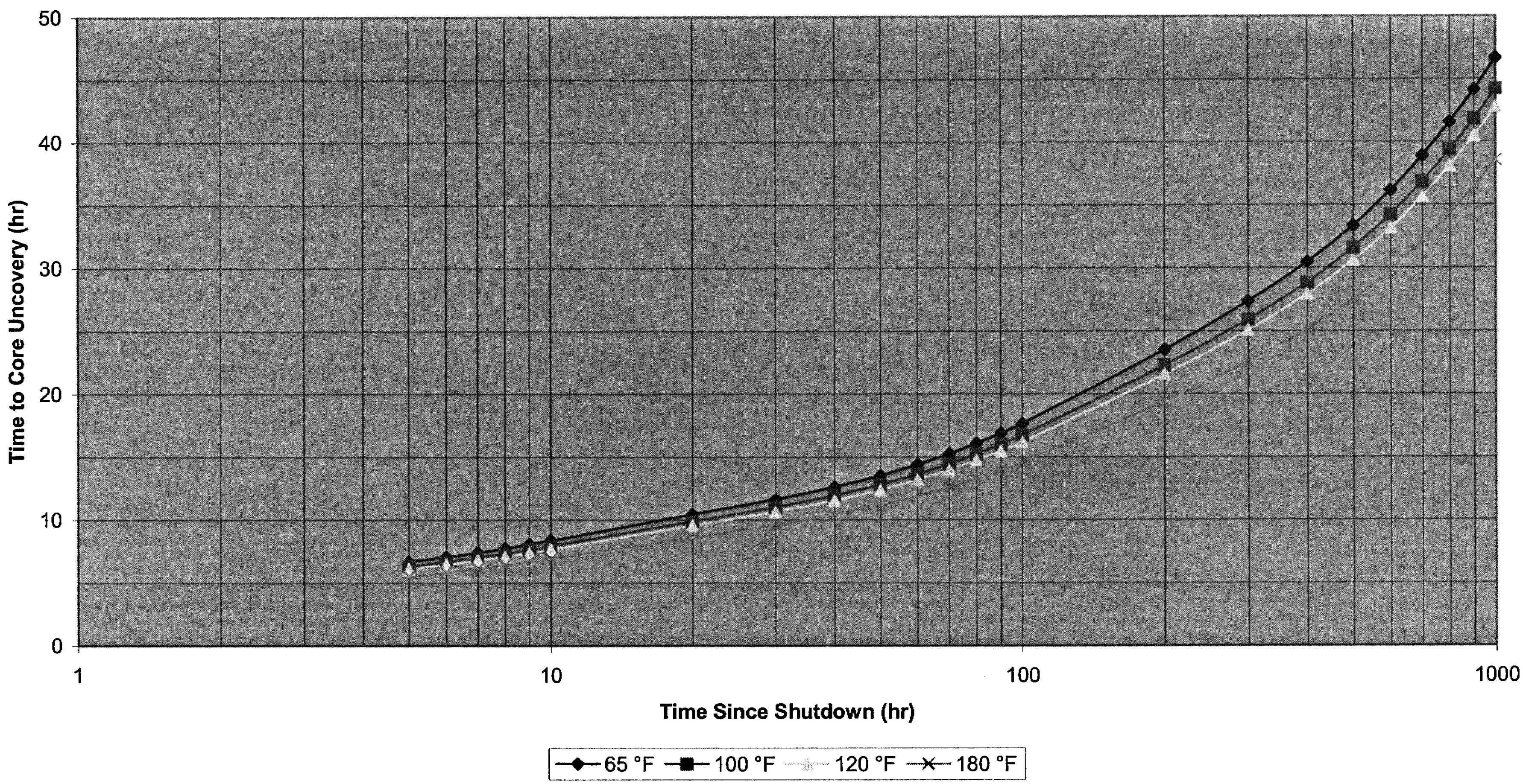
CHOOSE CURVE BASED ON INITIAL WATER TEMPERATURE

Attachment C
Time-to-Boil while Flooded to Flange
Dryer/Separator In



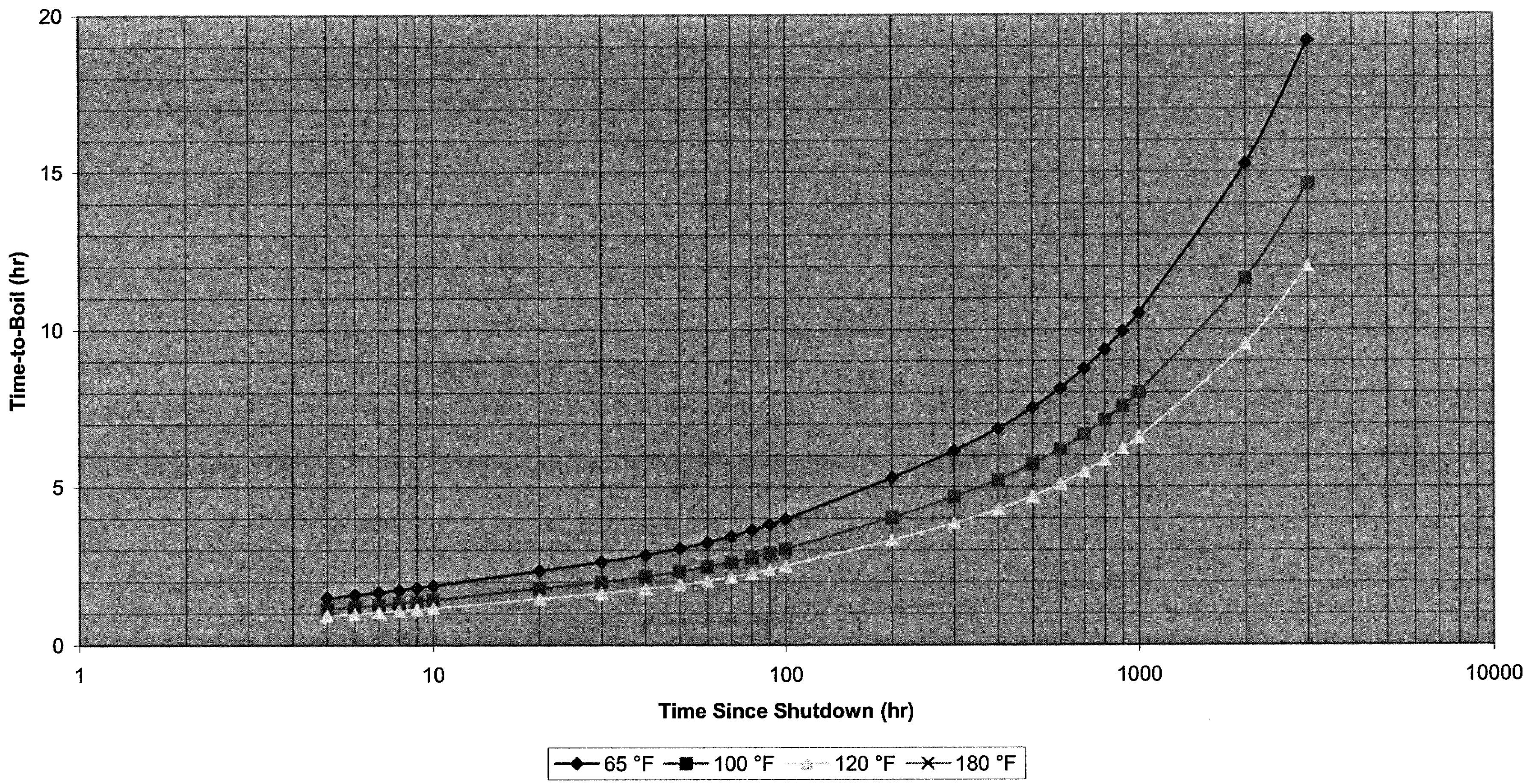
CHOOSE CURVE BASED ON INITIAL WATER TEMPERATURE

Attachment D
Time-to-Core Uncovery while Flooded to Flange
Dryer/Separator In



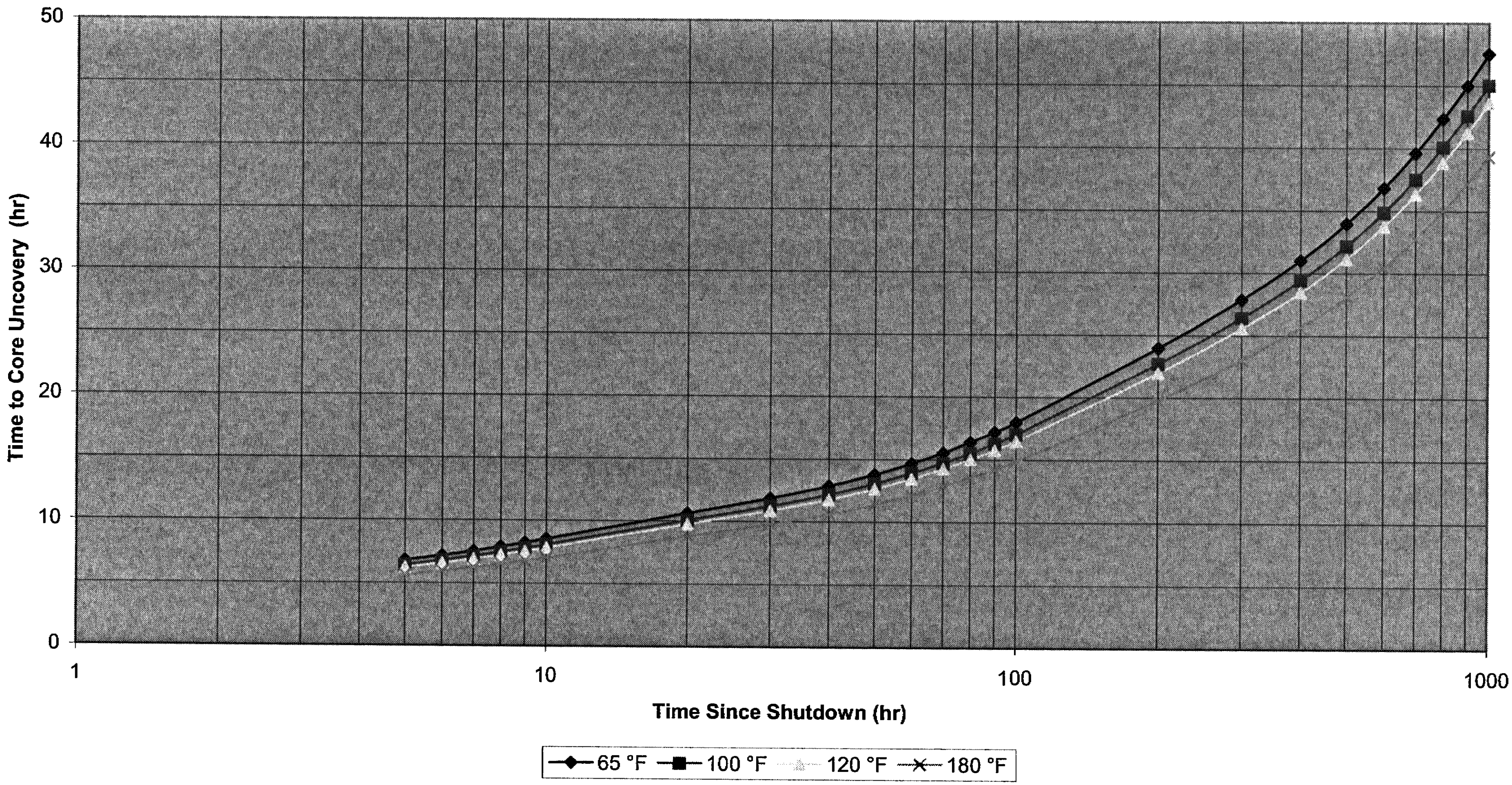
CHOOSE CURVE BASED ON INITIAL WATER TEMPERATURE

Attachment E
Time-to-Boil while Flooded to Flange
Dryer/Separator Removed



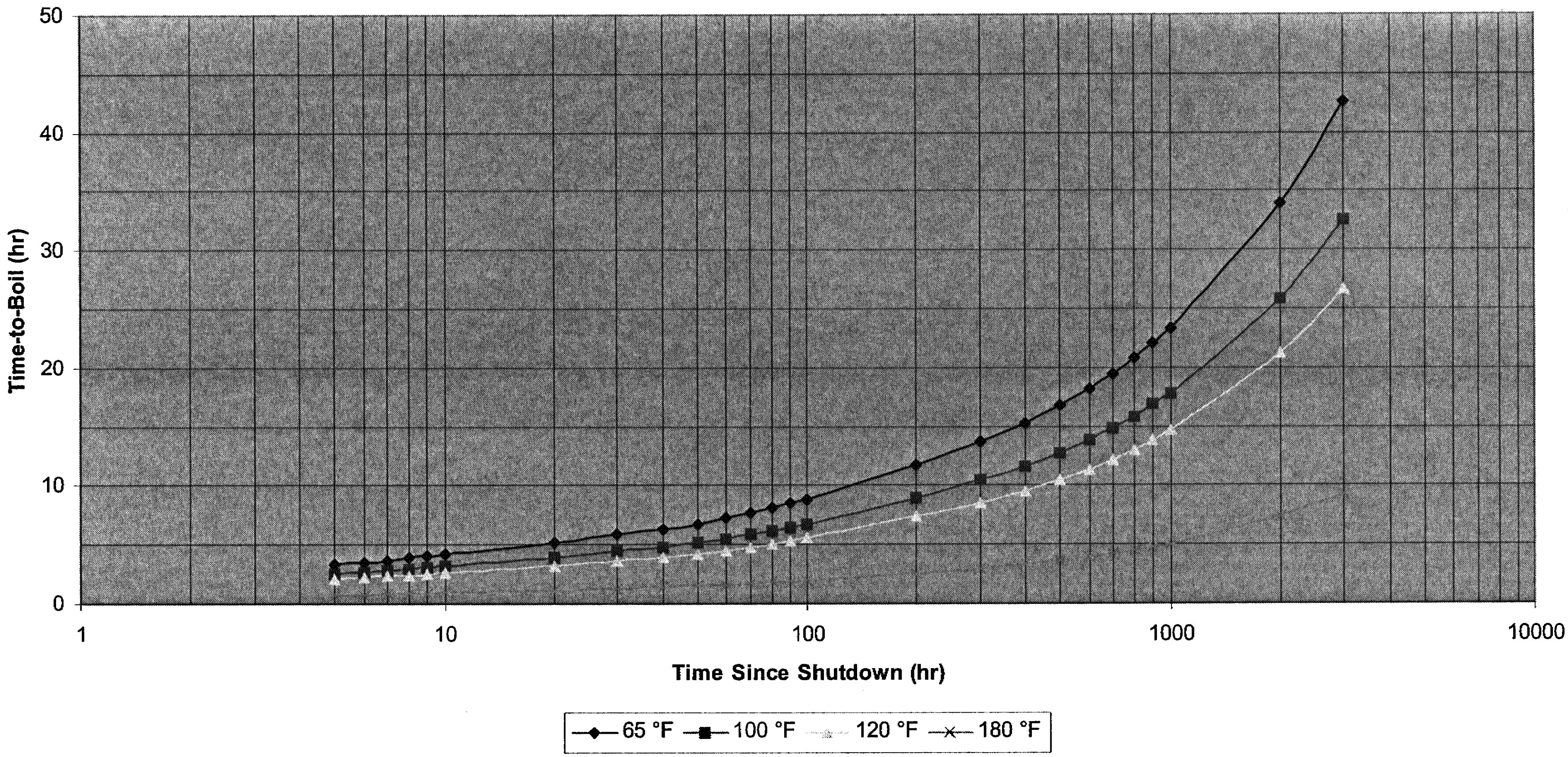
CHOOSE CURVE BASED ON INITIAL WATER TEMPERATURE

Attachment F
Time-to-Core Uncovery while Flooded to Flange
Dryer/Separator Removed



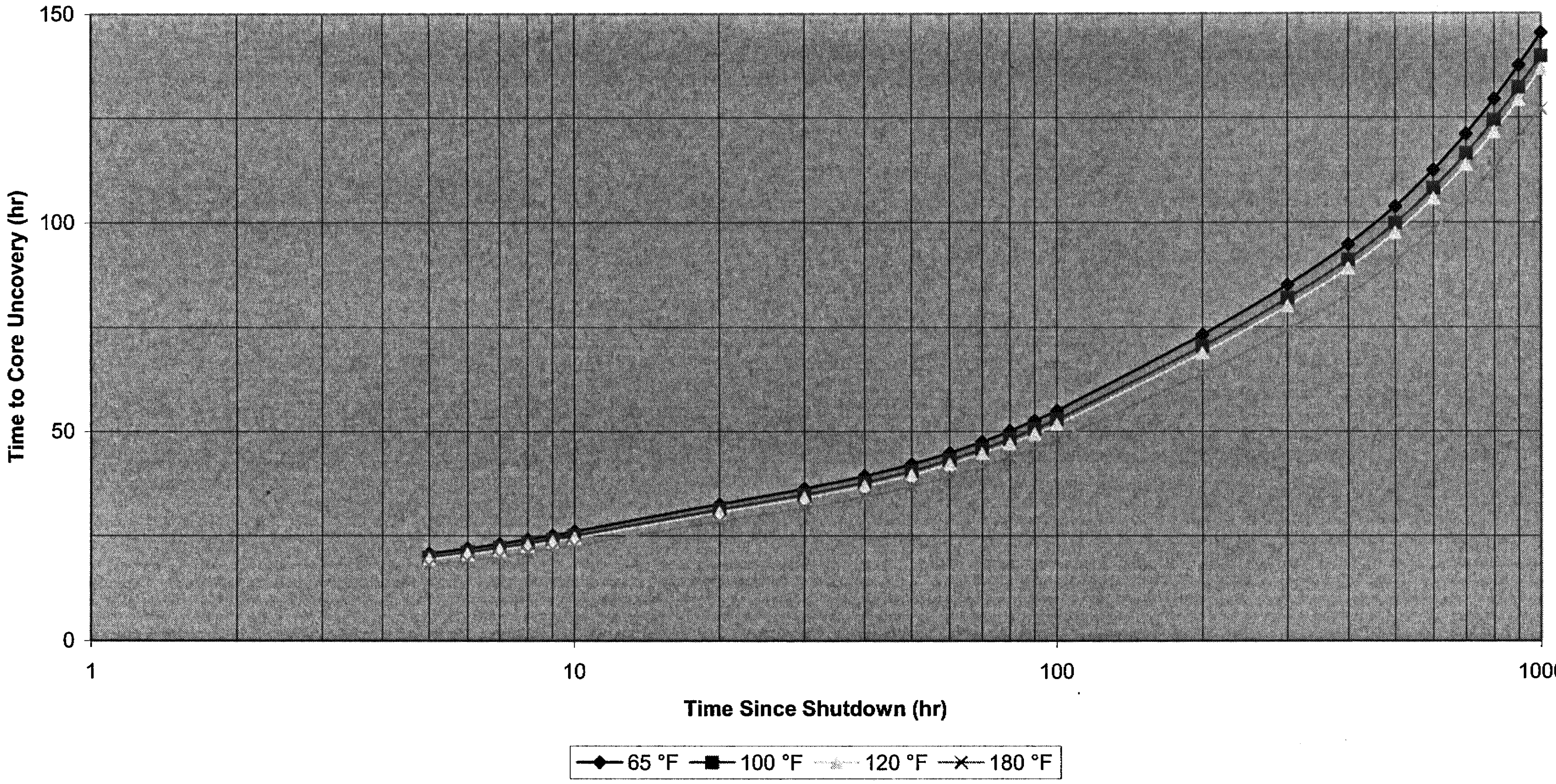
CHOOSE CURVE BASED ON INITIAL WATER TEMPERATURE

Attachment G
Time-to-Boil while Flooded Up
Gates In; Dryer/Separator In
(Dryer/Separator Pit not included)



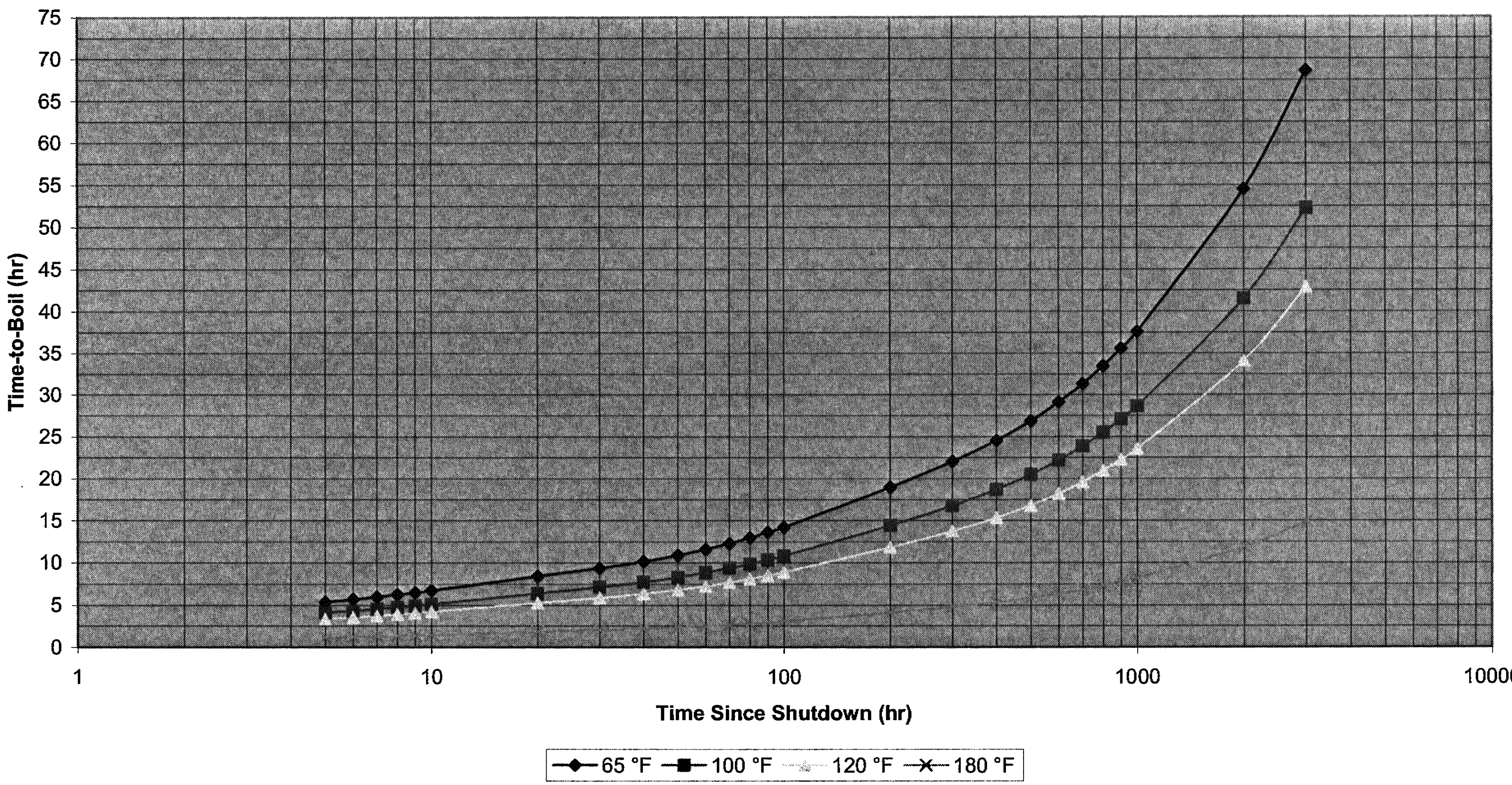
CHOOSE CURVE BASED ON INITIAL WATER TEMPERATURE

Attachment H
Time-to-Core Uncovery while Flooded Up
Gates In; Dryer/Separator In
(Dryer/Separator Pit not included)



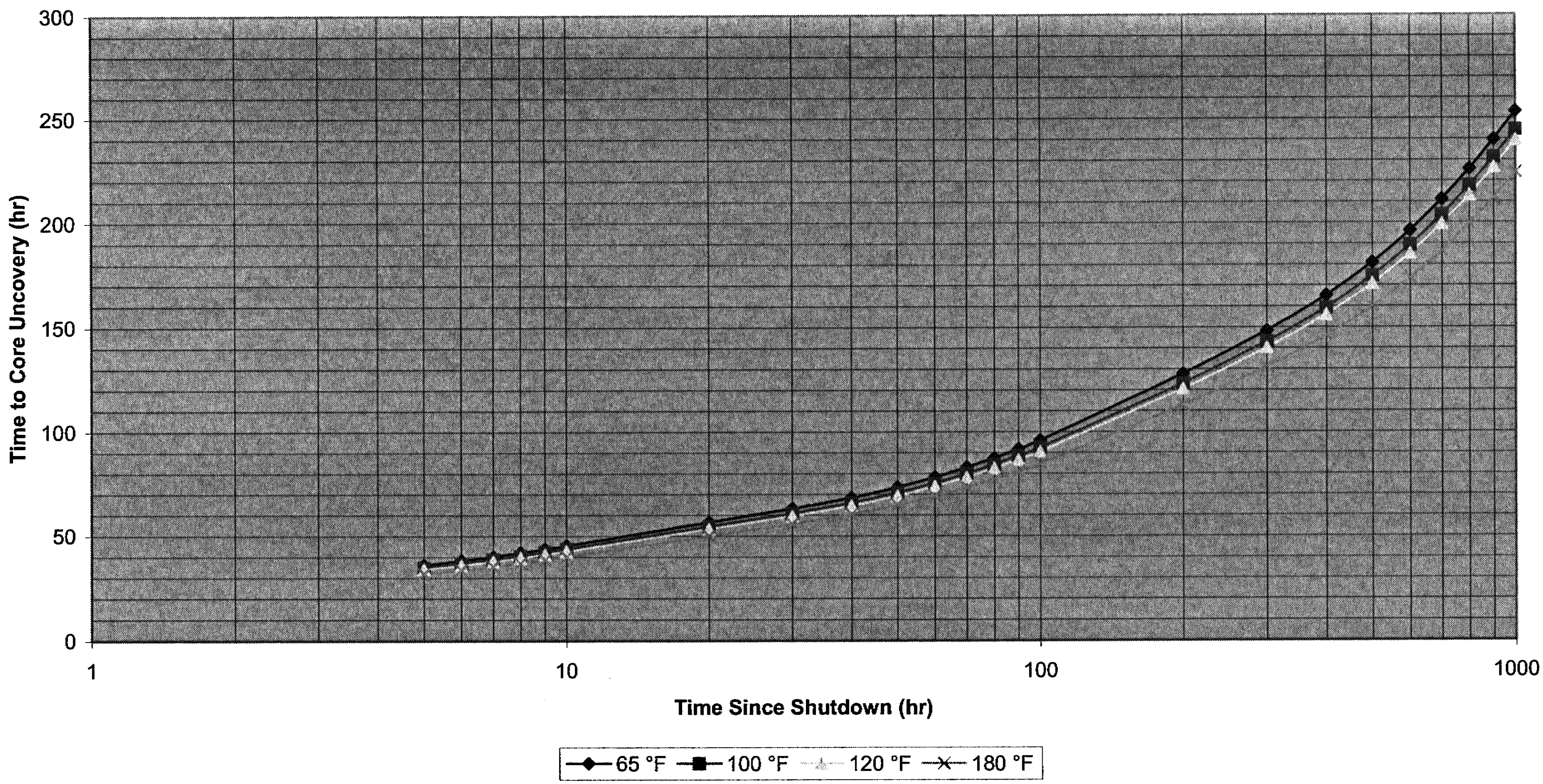
CHOOSE CURVE BASED ON INITIAL WATER TEMPERATURE

Attachment I
Time-to-Boil while Flooded Up
Reactor and Dryer/Separator Pit



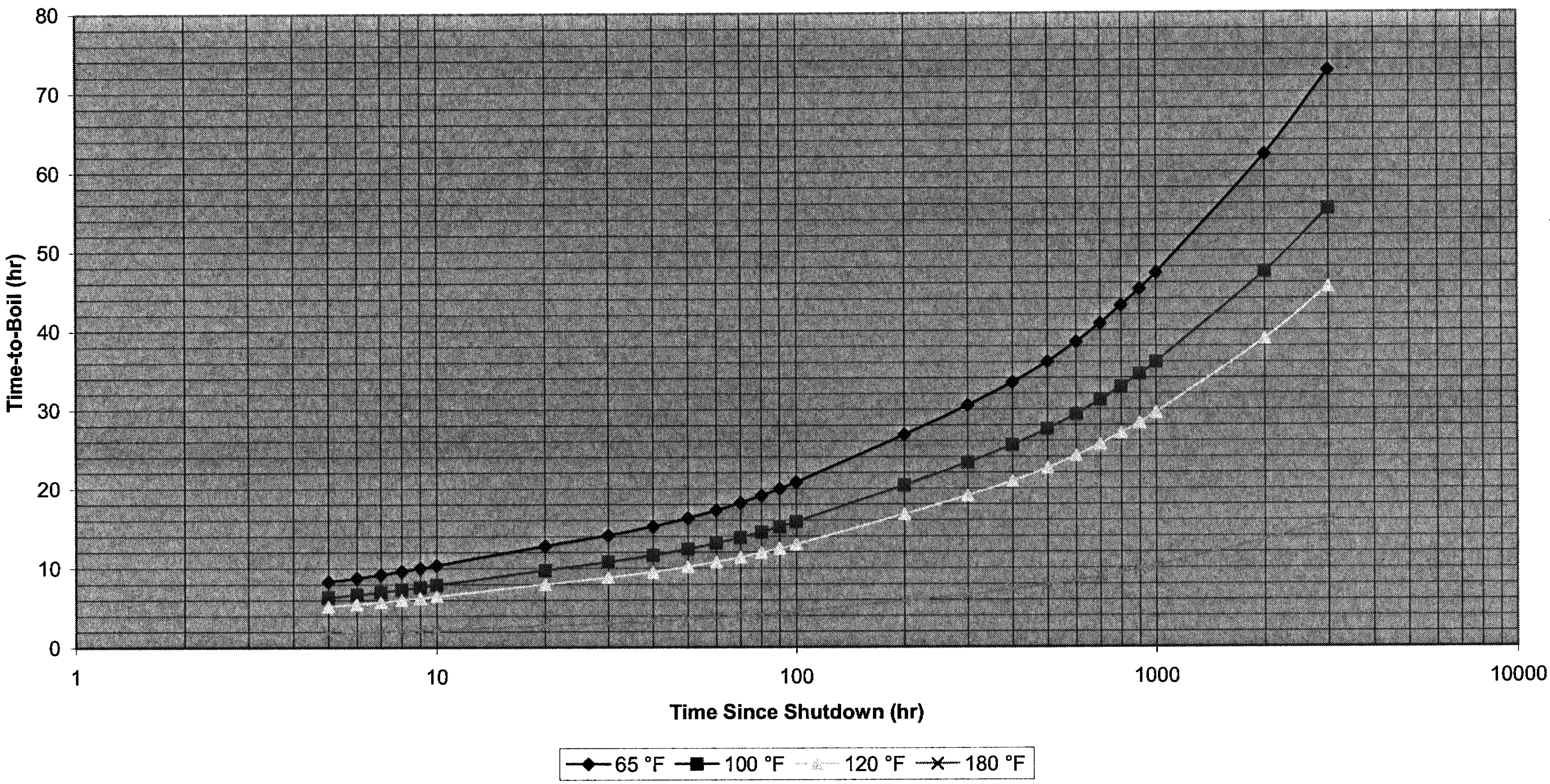
CHOOSE CURVE BASED ON INITIAL WATER TEMPERATURE

Attachment J
Time-to-Core Uncovery while Flooded Up
Reactor and Dryer/Separator Pit



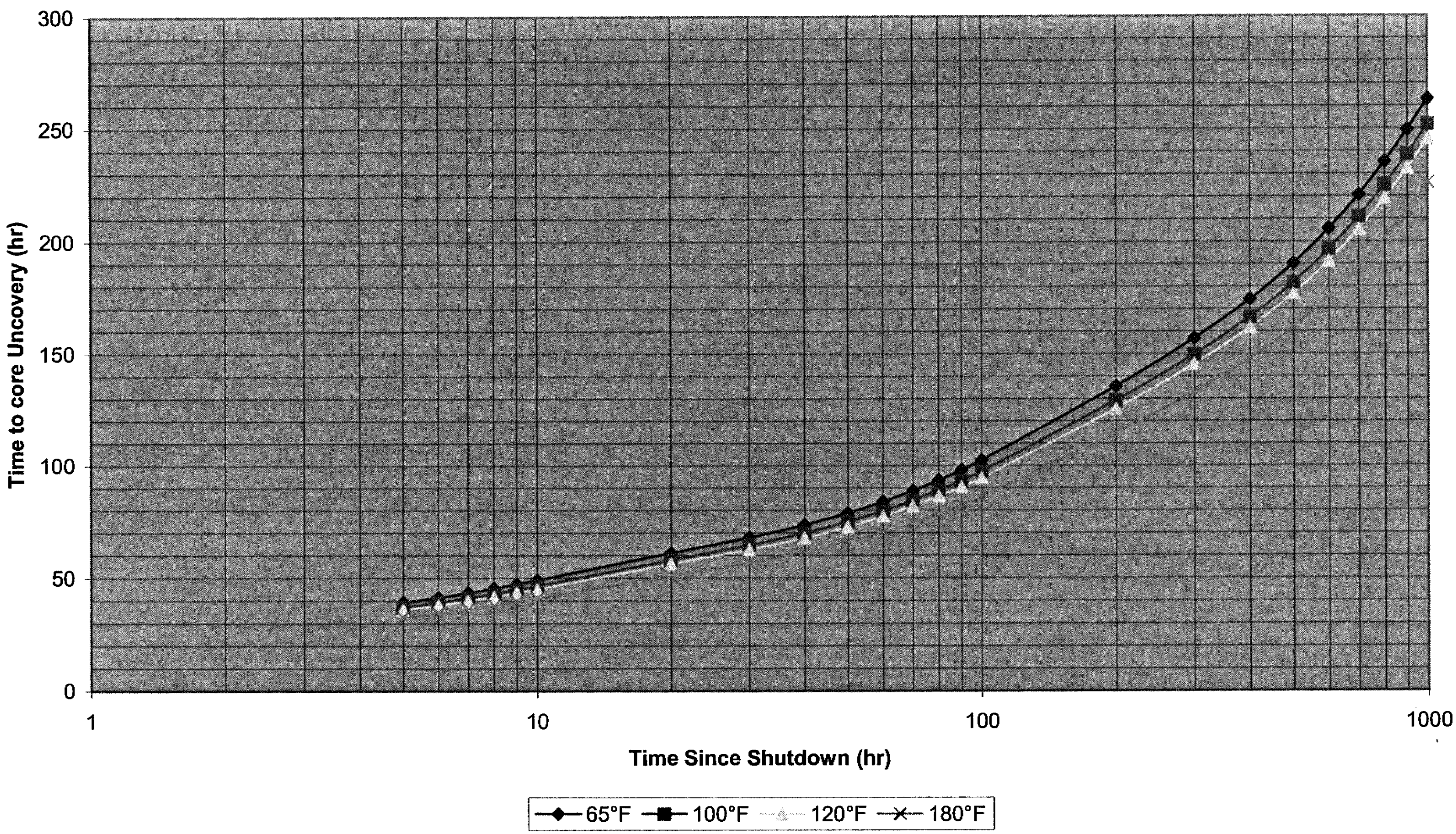
CHOOSE CURVE BASED ON INITIAL WATER TEMPERATURE

Attachment K
Time-to-Boil with Gates Out
Cavity, Dry/Separator Pit, & Spent Fuel Pool



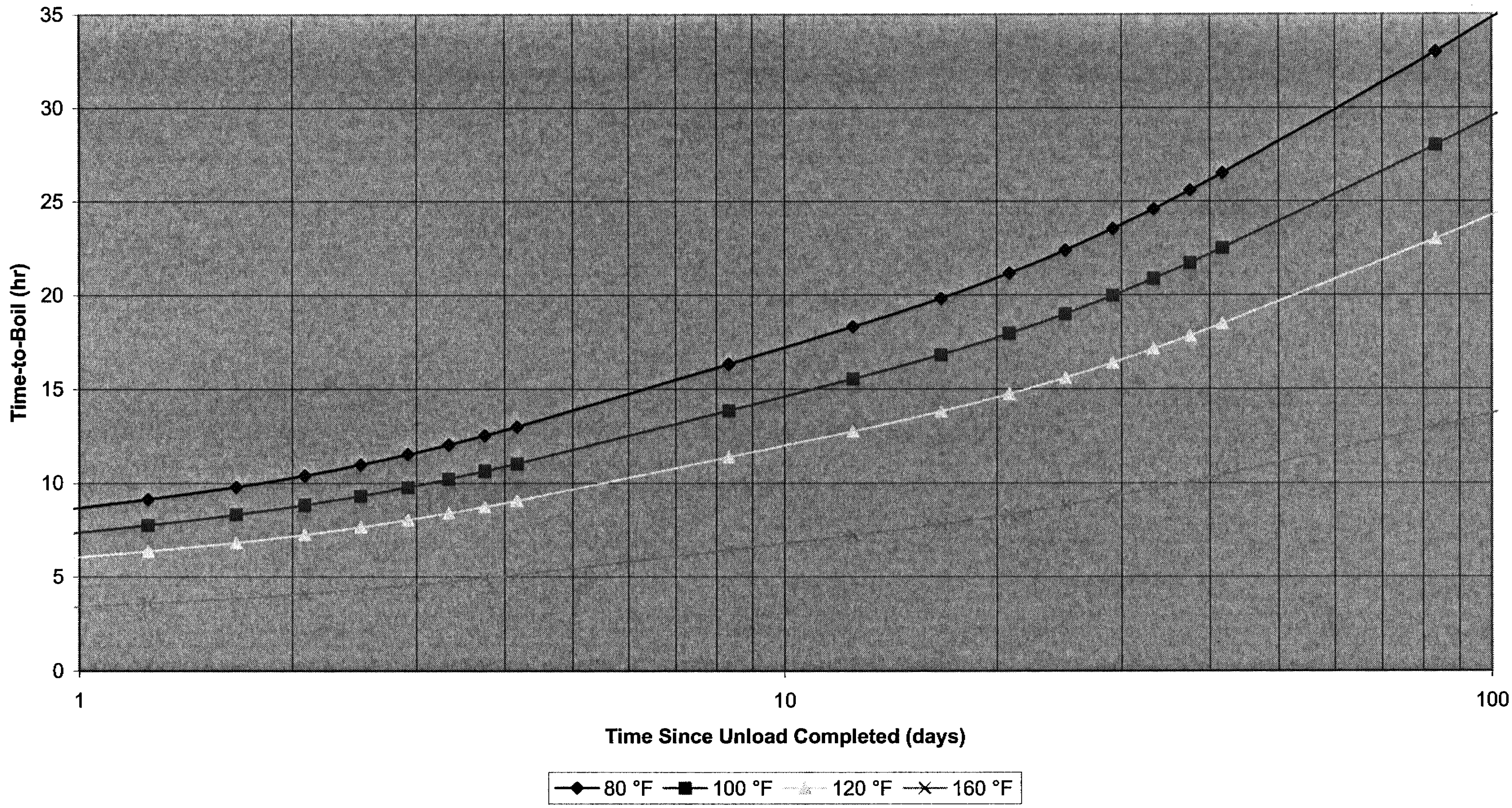
CHOOSE CURVE BASED ON INITIAL WATER TEMPERATURE

Attachment L
Time-to-Core Uncovery with Gates Out
Cavity, Dryer/Separator Pit, & Spent Fuel Pool



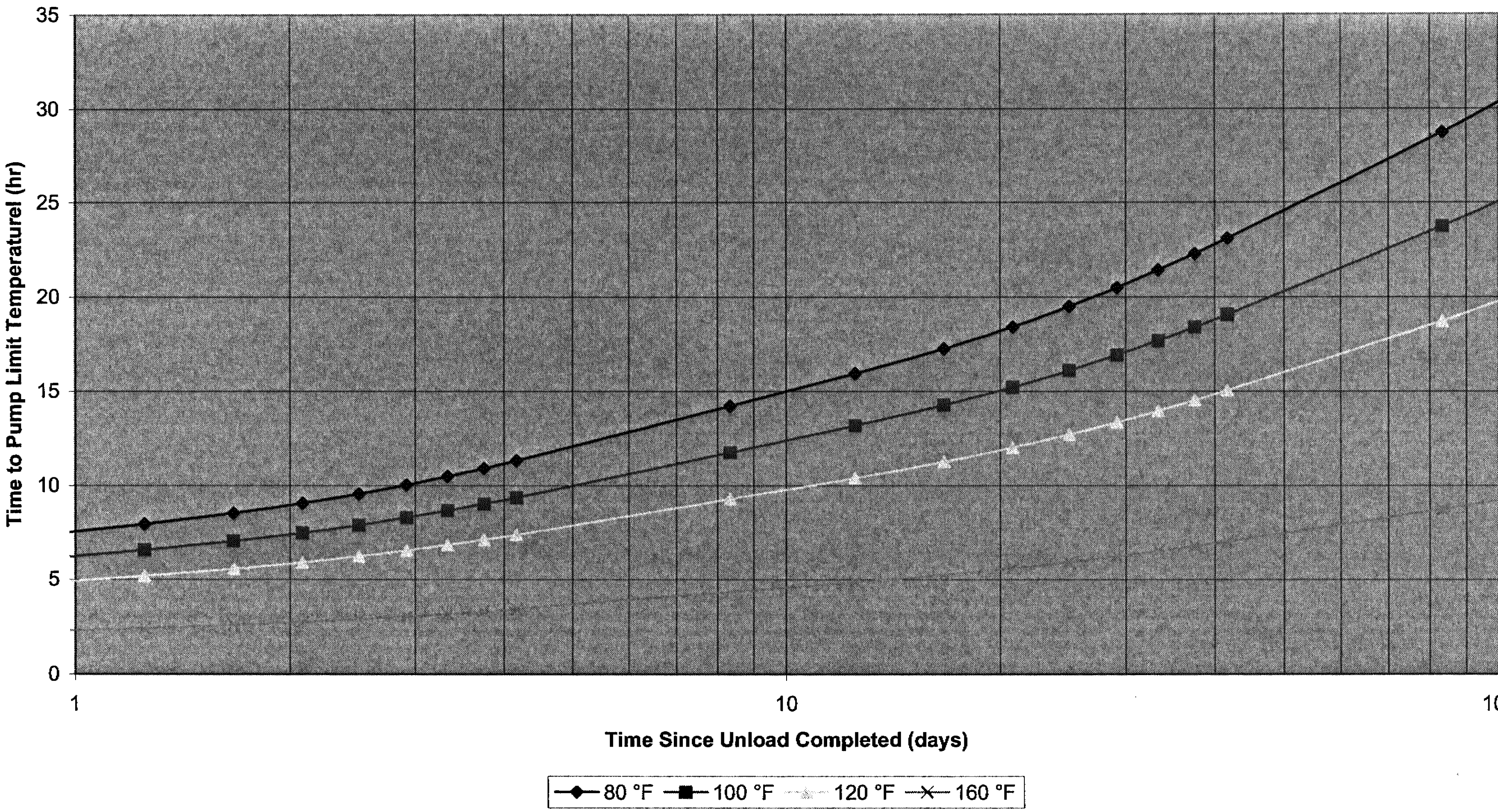
CHOOSE CURVE BASED ON INITIAL WATER TEMPERATURE

Attachment M
Time-to-Boil for Spent Fuel Pool (Isolated)



CHOOSE CURVE BASED ON INITIAL WATER TEMPERATURE

Attachment N
Time-to-Spent Fuel Pool Pump Cavitation Limit



CHOOSE CURVE BASED ON INITIAL WATER TEMPERATURE