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VICE PRESIDENT
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November 13, 1990

U. S. Nuclear Regulatory Commission
Washington, DC 20555

ATTENTION: Document Control Desk

SUBJECT: Calvert Cliffs Nuclear Power Plant
Unit Nos. 1 & 2; Docket Nos. 50-317 & 50-318
Response to Station Blackout Safety Evaluation
(TAC Nos. 68525 and 68526)

REFERENCE: Letter from Mr. D. G. McDonald, Jr. (NRC) to Mr. G. C. Creel (BG&E), dated October 10, 1990, Response to Station Blackout Rule

Gentlemen:

Attached is our response to recommendations made by the NRC Staff in a recent Safety Evaluation (SE) (Reference) regarding implementation of 10 CFR 50.63 (station blackout rule).

The primary issue raised in the SE concerned the proposed coping duration. The Staff had disagreed with our assessment of grid-related loss of offsite power events. There was a loss of offsite power in 1987 at the Calvert Cliffs site which the Staff believes was grid-related. However, an analysis of the event led us to the conclusion that it was caused by maintenance and hardware problems. Corrective actions have been taken to resolve the maintenance problems. The hardware problem (transistor failure) was determined to be a random equipment failure and the faulty equipment was replaced. The cause of this event and the conclusions drawn from it were discussed with the NRC Staff during a meeting held November 1, 1990. Questions remained concerning the nature of the transistor failure experienced during the event. Therefore, additional information will be provided concerning the cause of the transistor failure by December 14, 1990. This event can then be characterized as a plant-centered event and does not reflect on the reliability of the electrical grid. This conclusion will support a coping duration of four hours, as originally proposed.

We have evaluated the recommendations contained in the SE. Attachment (1) to this letter describes how we plan to implement those recommendations. The proposed modifications (except for the additional EDGs) will be completed during the next Unit 1 refueling outage (tenth refueling outage) and the next Unit 2 refueling outage

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(ninth refueling outage). The installation of two additional Class 1E diesel generators should be complete by February 1995, however this is subject to change as the modification progresses. A preliminary schedule is attached (Attachment 3). This schedule will be discussed in greater detail with the NRC Staff during the regulatory review process for the diesel generators.

Should you have any further questions regarding this matter, we will be pleased to discuss them with you.

Very truly yours,

for G. C. Creel
Vice President - Nuclear Energy

GCC/PSF/dlm

Attachments
(1) Station Blackout Safety Evaluation Report Responses
(2) Heatup Calculations Methodology and Assumptions
(3) Preliminary Diesel Generator Project Schedule

cc:
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ATTACHMENT (1)

Station Blackout Safety Evaluation Report Responses

2.1 Station Blackout Duration

Recommendation:

The licensee should reevaluate the plant's ability to cope with a station blackout (SBO) based on an eight-hour coping duration and include these analyses with the other documentation supporting the SBO submittal. If the licensee desires reconsideration of the coping category, sufficient justification with appropriate analysis should be provided for staff review which demonstrates the rationale for the July 23, 1987 loss of offsite power (LOOP) not being considered as symptomatic of a grid-related LOOP.

Response:

The offsite power design characteristic group is critical to determining the required coping duration for Calvert Cliffs. The SBO analysis uses a P2 group resulting in a four-hour coping requirement with diesel modifications. A P3 group as proposed by the NRC results in an eight-hour coping requirement. The issue is dependent on how the July 23, 1987 LOOP event is categorized. If it is a "grid-related event," P3 is correct. If it is a "plant-centered event," then P2 is correct.

The July 23, 1987 LOOP was initiated because of a ground fault that occurred on one of the two 500kv lines connecting Calvert Cliffs to the BG&E grid. This fault caused opening of breakers at both ends of the line. The breakers at the Calvert Cliffs end of the second line also tripped due to a defective transistor on a logic circuit card. Thus, the plant was completely isolated from the BG&E grid. It should be noted, however, that one of the 500kv lines was continuously live into the site switchyard and an alternate source from Southern Maryland Electric Cooperative was available at their on-site substation. The July 1987 event did not cause, nor was it caused by, a loss of the BG&E grid. In fact, the BG&E grid remained intact even though both Calvert Cliffs Units went off line. It only involved the loss of one line and the opening of one end of the other.

The root cause of the ground fault was the lack of resources needed to clear the right of way in a timely manner. The tree which caused the ground fault had been identified during previous surveys as one which needed to be cut. However, due to a lack of resources assigned to the forestry group, it was not cut down immediately. Corrective actions have included increasing expenditures for right of way clearing, cutting down all identified problem trees and increasing patrols to detect early tree growth. We believe these actions will prevent a recurrence of this type of fault.

A failed transistor on a logic circuit card in the static protective relay panel caused the loss of the second 500kv offsite power line. The logic card was part of the timing circuit for the permissive signal to the breakers on line 5051. The failed transistor caused a permissive signal to be incorrectly received by the breakers and they opened. The cause of the failure was determined to be a random equipment failure. The cards used in the protective relay circuits are tested every 18-24 months using a method which would have detected a failed transistor. Based on testing results, transistor failures in these cards have not occurred anywhere else at BG&E generating stations. These protective relays are used throughout BG&E's 500/230kv system and have provided reliable service since they were first installed in 1968. To ensure a clear understanding of the nature of the transistor failure, we will provide additional information concerning our evaluation of the failure by December 14, 1990.

Two system enhancements are planned which would prevent this type of loss of offsite power in the future. A third 500kv line is planned between our switchyard and PEPCO's Chalk Point Generating Station. Another 500kv line is planned between our Waugh Chapel substation and PEPCO's Brighton substation. These lines will provide a direct connection between Calvert Cliffs and the regional 500kv grid. The third line to the Chalk Point Generating Station would have prevented the loss of offsite power experienced in 1987.

Therefore, based on the root causes described above, we concluded that the 1987 LOOP was a plant-centered event. Calvert Cliffs meets the criteria for a P2 classification and, as a result, we remain in a four-hour coping category. The remaining responses assume a four hour coping duration.

2.2.2 Proposed AAC Power Source

Recommendations:

1. The licensee should submit separately for staff review the overall design information on the proposed EDG modifications and installation of the additional EDGs. This information should include the modifications to the EDGs' buses, cables and associated systems. The licensee should also include information on the EDG (spare) when used as an AAC source and when substituted for a dedicated EDG when it is out for maintenance and repair. LCO and TS changes on the dedicated EDGs and the proposed AAC source should also be provided. In addition, this information should also be included in the documentation supporting the SBO submittal maintained by the licensee.

2. The licensee should demonstrate that the AAC power source is available for supplying the SBO loads within one hour of the onset of an SBO event by conducting the appropriate testing in accordance with the guidance of NUMARC 87-00, Appendix B, Item B.12.

Response:

Information concerning the overall design of the additional diesel generators will be submitted to the NRC as part of the licensing process for the diesels. We expect to submit detailed design documents once vendors and a final design are selected. Technical Specifications will be submitted to include the new diesel arrangement in the Limiting Conditions of Operation (LCO) and Surveillance Requirements. Use of the spare (alternate AC source/Class 1E backup) diesel will be described. A description of the diesel generator modifications and installation will also be maintained with other SBO documentation.

As part of the diesel generator modification program, the spare diesel will be tested on a one-time basis to ensure that it can perform its AAC function. This test will be conducted during an appropriate Unit outage and will be performed in accordance with the guidance of NUMARC 87-00, Appendix B, Item B.12.

2.3.1 Condensate Inventory for Decay Heat Removal

Recommendation:

The licensee should confirm that there is sufficient inventory to remove the decay heat from both units and also provide for cooldown in the Non-Blacked-Out (NBO) unit and include this confirmation in the documentation supporting the SBO submittal maintained by the licensee.

Response:

We have reviewed our condensate inventory calculations and have determined that adequate inventory exists to support both a blacked-out unit and a non-blacked-out unit in a safe shutdown condition for four hours. Both units will be maintained in HOT STANDBY (**MODE** 3) during the four-hour event. The condensate requirements are the same for the blacked-out and non-blacked-out Units. This calculation is included with our other SBO documentation.

2.3.3 Compressed Air

Recommendation:

The licensee should establish procedures and simulate appropriate action and provide operator training to assure that decay heat removal can be adequately maintained during the first hour of an SBO event.

Response:

An Emergency Operating Procedure (EOP-7) has been established which describes operator actions during a SBO event. Operator training, as well as simulator training, has taken place to ensure that the operators can adequately respond to a SBO event. The procedure gives guidance for the establishment of an adequate heat sink. It addresses the communications necessary between the operators stationed at the atmospheric dump valves, at the auxiliary feedwater flow control valves, and in the control room.

It should be noted that the Technical Evaluation Report attached to the NRC SE incorrectly identifies the valves used to remove heat from the reactor core. Atmospheric dump valves, not PORVs, are used to remove heat from the steam generators. The atmospheric dump valves are air operated and will be manually cont olled after a station blackout begins. Emergency Operating Procedure (EOP-7) gives guidance concerning their operation. We have no plans to modify the atmospheric dump valves.

The PORVs are the power operated relief valves on the pressurizer. We will provide DC power to these valves and their associated block valves. This allows the PORVs to lift if the reactor coolant system should become overpressurized. This potential leakage path can be closed off, by the PORV block valves, which are also being modified to operate from DC power.

2.3.4 Effects of Loss of Ventilation

Recommendation:

The licensee should reanalyze the the heatup analyses for the areas of concern based on a one-hour duration, including information to demonstrate the acceptability of the methodology, assumptions and initial conditions used in the calculations. This assumes HVAC for dominant areas of concern will be be powered by the proposed AAC power source after one hour. Also, the licensee should document additional justification as to why it is not necessary to open cabinet doors in the control room within one-half hour after the onset of an SBO event. The licensee should include the above analyses and results in the documentation supporting the SBO submittal maintained by the licensee.

Response:

Calculations using NUMARC 87-00 Section 7.2.4 and other methods were performed to determine the SBO temperature after four hours. The results are listed below. These rooms are dominant areas of concern even though NUMARC 87-00 recommends only the AFW Pump Rooms be treated as such.

ROOM TEMPERATURES

Room No.	Room Name	Doors Open	4 hr SBO Temp (°F)
A306(302)	Cable Spreading Room	none	103
A315(309)	Main Steam Piping Pen Rm	none	207
A316(310)	East Piping Pen Room	none	154
A317(311)	27' Switchgear Room	none	129
A405	Control Room	none	103
A431(406)	DAS Room	none	130
A430(407)	45' Switchgear Room	none	127
C230(229)	Containment	none	185
T603(605)	AFW Pump Room	yes	137

Note that the values for the control and DAS rooms are based on a modified ceiling configuration opened to allow better air circulation. The rooms where temperatures were not calculated do not contain equipment credited during a blackout.

Only three of the rooms listed above have fire protection systems that actuate at high temperatures. They are rooms A315(309) at 286°F, A316 (310) at 212°F minimum, and T603(605) at 212°F. Since the expected SBO temperatures are well below these values, no fire protection systems will be improperly actuated during an SBO.

This four-hour case bounds a room heatup for one hour, with a subsequent use of HVAC. A description of the heatup calculations is attached (Attachment 2). It describes the methodology and assumptions used in the calculations. The calculations were performed assuming that the proposed modifications are complete. The cabinet doors for the control room panels are assumed to be opened within 30 minutes. The emergency operating procedure will be revised to ensure that the operators open the front of the control room panels within 30 minutes. It should be noted that the panels have no backs. They are completely open to the control room atmosphere. The calculations described in the attachment are included with the SBO documentation.

2.3.6 Reactor Coolant Inventory

Recommendation:

The licensee should perform the necessary analyses to show that a reactor coolant inventory loss of 112 gpm does not result in core uncovery during an eight-hour SBO event. The licensee should include these analyses and results in the documentation supporting the SBO submittal maintained by the licensee.

Response:

Analyses were performed to determine the effect of reactor coolant inventory loss during a four-hour SBO event. Reactor coolant leakage was determined as follows: 25 gpm per reactor coolant pump, 10 gpm identified leakage and 10 gpm miscellaneous leakage. Additionally, we assume a letdown flow of 128 gpm for the first 30 minutes of the event. The analysis shows that no core uncovery occurs during a four-hour SBO event. The assumptions and results of this analysis are maintained as part of our SBO documentation.

2.5 Proposed Modifications

Recommendation:

The licensee should provide a full description including the nature and objectives of the required modifications identified above in the documentation supporting the SBO submittal that is to be maintained by the licensee. It should be noted that the modifications relating to the reconfiguration of the existing EDGs and the addition of two others have not been reviewed under the SBO review and should be submitted separately for staff review as indicated in the recommendations in Section 2.2.2.

Response:

A full description of each proposed modification (except the additional EDGs) will be included in the SBO documentation. This description will include the objectives of the modification. These modifications will be completed during the next Unit 1 refueling outage (tenth refueling outage) and the next Unit 2 refueling outage (ninth refueling outage). The description of the additional EDGs will be provided to the NRC for review as a separate package (see 2.2.2 response). The EDGs should become operational in February 1995.

2.6 Quality Assurance and Technical Specifications

Recommendation:

The licensee should implement a quality assurance program that meets, as a minimum, the guidance of RG 1.155, Appendix A, for any equipment not presently covered by an equivalent QA program.

Response

A QA program that meets the guidance of Regulatory Guide 1.155, Appendix A is being developed to cover equipment needed for SBO that is not presently covered by our existing QA program. We expect this program to be implemented by the end of 1991.

2.7 EDG Reliability Program

Recommendation:

The licensee should verify that a program that meets the guidance of RG 1.155, Section 1.2, is in place and include this verification in the documentation supporting the SBO submittal that is to be maintained by the licensee.

Response:

We have committed to a target reliability of 0.975 and that reliability is tracked and maintained under our existing program. We are enhancing that program based upon the guidance given in NUMARC 87-00, Appendix D and working copies of Regulatory Guide 1.9, Revision 3, as appropriate. This program will meet the guidance of RG 1.155, Section 1.2. The EDG reliability program will be fully implemented by September 30, 1991.

ATTACHMENT (2)

Heatup Calculations
Methodology and Assumptions

METHODOLOGY

The departures from the methodology outlined in NUMARC 87-00 involve the temperature calculations for the cable spreading rooms and the containment as outlined below.

Cable Spreading Rooms

In the NUMARC 87-00 methodology, a constant heat load is applied to the room in which the temperature is being calculated. However, because the heat loads in the cable spreading rooms change with time, an alternate methodology is employed.

The temperature rise of the cable spreading room is calculated using Bechtel's microcomputer application program PCFLUD (MAP-120). PCFLUD can calculate the time-dependent temperature rise of a room, given heat sinks and a heat load. PCFLUD is utilized because it is capable of modelling time-dependent heat loads. The use of PCFLUD has been accepted by the NRC in numerous equipment qualification and room temperature applications.

The methodology described in NUMARC 87-00 is a 4 hour approximation, while PCFLUD provides a transient, a more realistic, model of room heat-up.

The heat load in the cable spreading room changes with time. The heat load is 19000 W for the first 30 minutes. Then after 30 minutes when the computer is turned off, the 25 kVA inverters are de-energized. When energized, the inverters add to the room a heat load of 5526 W. Since the energized inverter temperature ($126^{0}C$) is warmer than the room, the inverter continues to supply heat to the room, even when de-energized. After 30 minutes the cooldown of the inverters is modelled using standard heat transfer techniques, such that the heatup of the cable spreading room is maximized.

Containment

For the containment heat-up calculation, in addition to heat loads, energy is assumed to be added to the containment via steam leakage from the reactor coolant system (predominantly from the reactor coolant pump seals).

Because the methodology described in NUMARC 87-00 does not adequately address the heat load from leaking steam, a transient analysis is performed using the Bechtel Standard Computer Program COPATTA. The COPATTA program is a transient analysis code used for determining pressure and temperature histories in single compartments such as containments. The code is capable of modelling multi-region heat sinks, mass and energy release and miscellaneous energy release directly to the containment atmosphere, and calculating the containment temperature as a function of time. The program is described in detail in the Bechtel Topical Report BN-TOP-3 [Reference (1)].

ATTACHMENT (2)

Heatup Calculations
Methodology and Assumptions

The input for this calculation was based on a loss of coolant accident analysis previously performed for Calvert Cliffs. The mass and energy release from the reactor coolant pump seals and other sources, was substituted for the break mass and energy release in the previous calculation and the active engineered safety features were prohibited from actuating for four hours. The only cooling mechanism modelled was heat transfer through the heat sinks.

In each instance described above, where the method used to calculate the room heat-up was different than that described in NUMARC 87-00, standard heat transfer techniques have been applied. The techniques are commonly used in other applications where room and/or equipment temperatures are required.

ASSUMPTIONS

Concrete Walls Greater Than 8" Thick

NUMARC 87-00 states that typically "wall temperature does not change appreciably throughout the transient as shown in Appendix E (2.5°C or 4.5°F)." The calculation in Appendix E which supports this conclusion uses a wall thickness of 8". For the room heat-up calculations, where the initial temperature of the room of interest (T room) is less than that of the surrounding rooms (Tadj rooms), an average wall temperature, calculated by

$$\text{Tavg} = \frac{\text{T room} + \text{Tadj rooms}}{2}$$

is used as a constant wall temperature to determine heat transfer from the room to the wall over a 4 hour period However, for concrete walls that are substantially thicker than 8", this technique yields an unrealistically high average temperature to be used in the heat sink calculation. Therefore, in the Calvert Cliffs SBO room heat-up calculations, for wall $\geq$ 21" thick, heat sink average temperatures are based on the temperature of the concrete 8" closest to the room of interest. A calculation has been performed to demonstrate that the average temperature of the 8" of concrete wall closest to the room of interest does not vary substantially over a 4 hour period (less than 4.5°F per Appendix E of NUMARC 87-00).

Concrete Masonry Unit Walls

For concrete masonry unit (CMU) walls, justification is provided for using either a fraction of the total surface area and the full thickness of the blocks, or the total surface area and a fraction of the thickness as a heat sink, modelling the wall as solid concrete. Typically one third of the voids of the CMU block are filled with grout and cross members also make the block more solid. Detailed heat transfer analyses were performed (Reference 2) in which typical 8" and 12" CMU blocks were modelled in two dimensions and exposed to various initial temperature gradients and heat fluxes. The heat transmission through the block was then compared with the heat transmission through a solid concrete wall. Based on this comparison, an equivalent fractional area or thickness of the wall was determined. For the control room and the cable spreading rooms heat-up calculations, credit is taken for the CMU walls as heat sinks by using fractional equivalent solid walls.

Treatment of Insulated Heat Generation Sources

The NUMARC 87-00 methodology does not provide any guidelines for determining a surface temperature of an insulated heat generating source. Therefore, general conduction, radiation and convection relationships for cylinders and/or flat plates [Reference (3)] are used to determine a surface temperature for insulated hot pipes and the insulated AFW pump turbines. To calculate a room heat load from these insulated surfaces, the Equation on page 7-19 and Equation E-9 of NUMARC 87-00 are coupled with the heat conduction equations.

REFERENCES

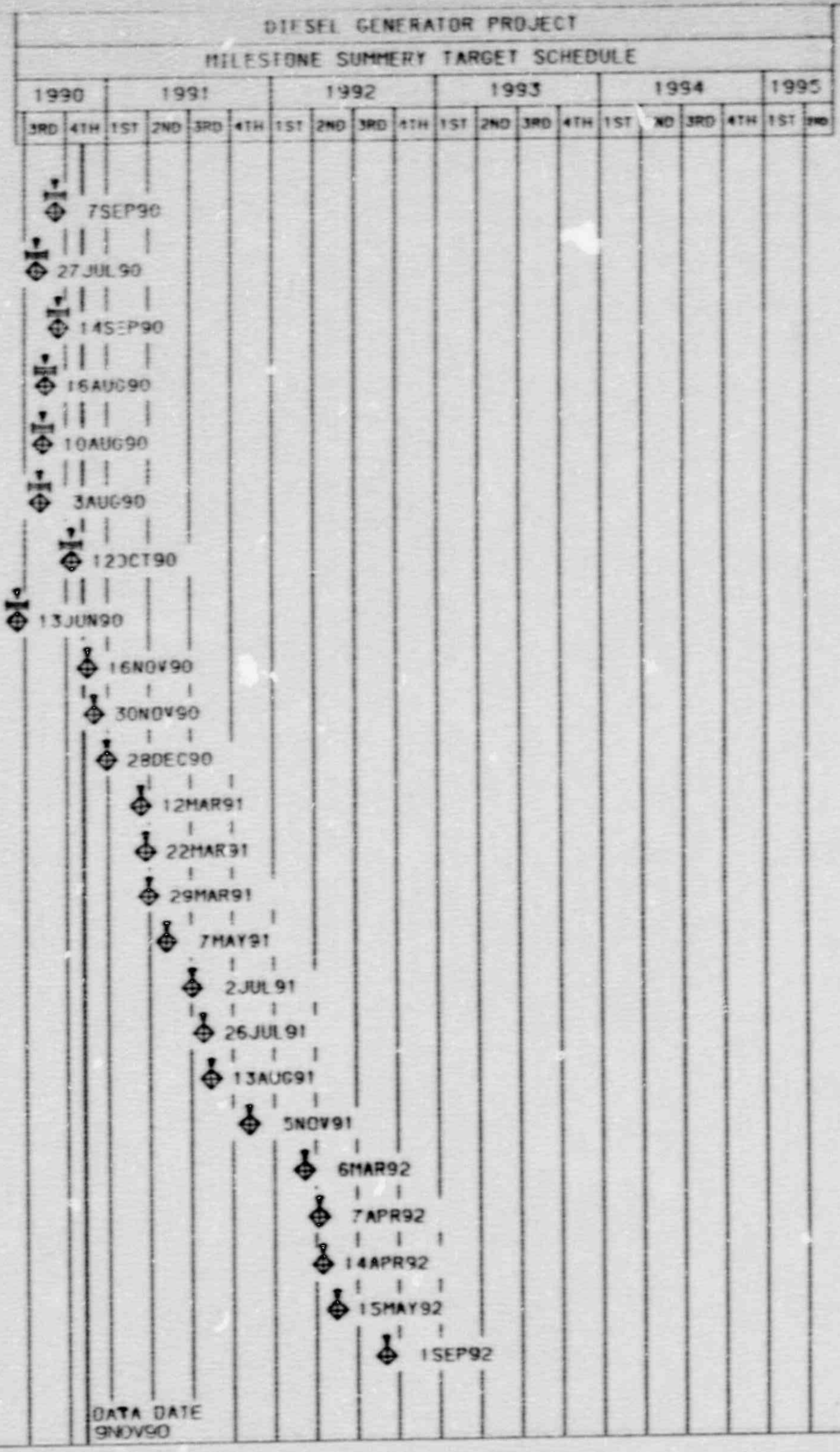
1. Bechtel Topical Report BN-TOP-3, "Performance and Sizing of Dry Pressure Containments," Rev. 4, March 1983.

2. J. E. Amrhein, Masonry Design Manual, Masonry Institute of America, Los Angeles, CA, 1969.

3. F. P. Incropera and D. P. Dewitt; Fundamentals of Heat Transfer, Wiley & Sons, 1981.

ATTACHMENT (3)

DIESEL-1008A	BECHTEL ISSUE REV. A PROJECT SCOPE DOCUMENT	7SEP90
DIESEL-7012	ISSUE FIRST REV OF PROJECT SCHEDULE (OBJ #09)	27JUL90
DIESEL-7015	PREPARE JUSTIFICATION OF PROJECT (OBJ #08)	14SEP90
DIESEL-8002	"SBO" COMMITMENTS IN CTS (OBJECTIVE # 16)	16AUG90
DIESEL-2015	DEFINE EDG SELECTION CRITERIA (OBJ 20)	10AUG90
DIESEL-4010	ISSUE PRE-QUALIFICATION QUESTICNNAIRE (OBJ #12)	3AUG90
DIESEL-4009	IDENTIFY A/E FIRMS TO BID (OBJ 13)	12OCT90
DIESEL-1010	DRAFT PROJECT SCOPE DOCUMENT COMPLETE (OBJ #15)	13JUN90
DIESEL-1005	DIESEL GEN TECH RQMTS DOCUMENT COMPL (OBJ #10)	16NOV90
DIESEL-7009	REWRITE PROJECT PLAN (OBJECTIVE #07)	30NOV90
DIESEL-2007	RFQ FOR DIESELS ISSUED (OBJ #23)	28DEC90
DIESEL-4007	ISSUE RFQ FOR A/E FIRM (OBJ 25)	12MAR91
DIESEL-2016	START EVALUATIONS OF EDG PROPOSALS (OBJ 24)	22MAR91
DIESEL-7014	BUDGET ESTIMATE PREPARED (OBJECTIVE #14)	29MAR91
DIESEL-4015	A/E BIDS RECIEVED	7MAY91
DIESEL-4016	A/E BID EVALUATION COMPLETE	2JUL91
DIESEL-2012	DIESEL GENERATOR CONTRACT AWARDED	26JUL91
DIESEL-4013	A/E CONTRACT AWARDED	13AUG91
DIESEL-4018	START DETAILED DESIGN ENGINEERING	5NOV91
DIESEL-9010A	UNIT 1 REFUELING OUTAGE - SPRING 1992	6MAR92
DIESEL-1015	A/E FIRM PREPAIR CONSTTION PACKAGE SCOPE	7APR92
DIESEL-3006	ISSUE RFQ FOR CONSTRUCTION FIRMS	14APR92
DIESEL-9011	COMPLETE UNIT 1 AND COMMON "SBO" MODS	15MAY92
DIESEL-3010	CONSTRUCTION CONTRACT AWARDED	1SEP92



DIESEL GENERATOR PROJECT

MILESTONE SUMMERY TARGET SCHEDULE

DIESEL-3012 CONSTRUCTION START

DIESEL-9050A UNIT 2 REFUELING OUTAGE - SPRING 1993

DIESEL-2014 FIRST DIESEL GENERATOR DELIVERED

DIESEL-9051 COMPLETE UNIT 2 "SBO"MODS

DIESEL-9100A START UNIT 1 REFUELING OUTAGE - SPRING 1994

DIESEL-5012 DIESEL GENERATOR CONSTRUCTION COMPLETE

DIESEL-9101 COMPLETE UNIT 1 REFUELING OUTAGE - SPRING 1994

DIESEL-6012 FIRST DIESEL OPERABLE (MILESTONE)

DIESEL-6009 SECOND DIESEL OPERABLE (MILESTCNE)