

REGULATORY INFORMATION DISTRIBUTION SYSTEM (RIDS)

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 EISENHUT, D.G. Division of Licensing

SUBJECT: Forwards status of NUREG-0737 post TMI requirements requiring action up to & including 820101.

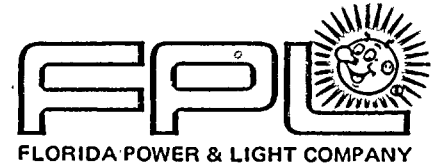
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 TITLE: Response to NUREG-0737/NUREG-0660 TMI Action Plan Rgmts (OL's)

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January 7, 1982
L-82-5

Office of Nuclear Reactor Regulation
Attention: Mr. Darrell G. Eisenhut, Director
Division of Licensing
U.S. Nuclear Regulatory Commission
Washington, D. C. 20555

Dear Mr. Eisenhut:

Re: Turkey Point Units 3 & 4
DOCKET NOS. 50-250 & 50-251
Post-TMI Requirements

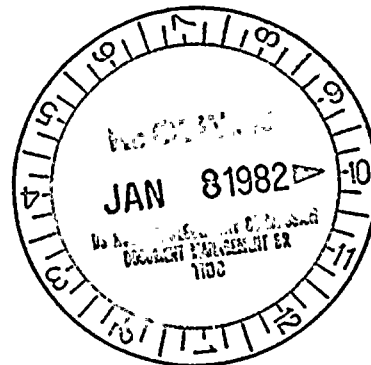
This letter transmits to you the status of those NUREG-0737 items requiring action up to and including January 1, 1982. We are working towards meeting all of the remainder of the requirements and will advise you should problems arise in meeting any of the long-term dates.

Very truly yours,

Robert E. Uhrig
Vice President
Advanced Systems & Technology

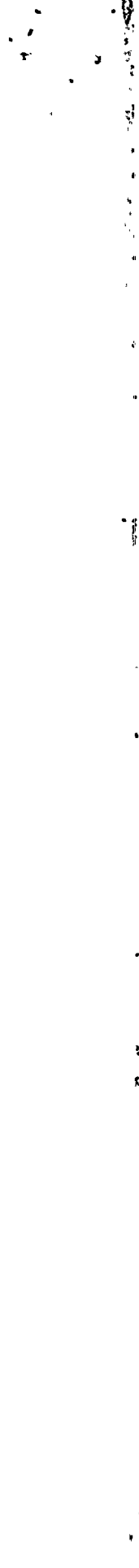
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cc: Mr. J. P. O'Reilly, Region II
Harold F. Reis, Esquire



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1. PLANT SAFETY PARAMETER DISPLAY CONSOLE (I.D.2)

The Safety Assessment System of which the Plant Safety Parameter Display Console is a part is expected to be fully operational following the first refueling outage of each unit after January 1, 1983 based on current scheduled equipment delivery dates.

2. REACTOR COOLANT SYSTEM VENTS (II.B.1)

Our letter L-81-298, dated July 16, 1981 submitted to you the design description of our RCS Vent System. At that time we stated that we would submit operating procedures following the NRC approval of the design. In an effort to expedite your review, it is now our intent to submit a set of operating procedures to you by March 1, 1982.

It is intended that the RCS vent system for Unit 3 will be installed prior to startup from the unit's steam generator repair outage. The Unit 4 system has been only partially installed due to the late delivery of equipment. Since a unit shutdown is required to install the system, it is our intent to install it during the Unit 4 steam generator repair outage.

3. PLANT SHIELDING (II.B.2)

A. In letter L-80-16 dated January 11, 1980 we identified potential problem areas in the plant that could require plant modifications to lower postulated post-LOCA radiation exposures to plant personnel. During the detailed design and engineering effort conducted during the two years since our initial submittal, we have gained additional insight into the specific shielding problems. As a result, certain solutions differing somewhat from those identified in our earlier letters have been implemented or are scheduled to be implemented. However, all of the problem areas identified in our previous submittal have been addressed and resolved.



It is our intent to have all remaining modifications complete by March 31, 1982 with the following exception: The containment isolation valves CV 2819 and CV 2826 (air bleed valves used in the "pump back system") are to be replaced with valves that are qualified to operate in a post-LOCA environment. The manufacturer's shipping date for these valves is February 20, 1982. The installation of the valves requires a unit outage. It is our intent to install these new valves in Unit 3 during the next refueling outage following receipt and to install them in Unit 4 during its upcoming steam generator replacement outage.

B. Radiation qualification of safety related equipment is being addressed through our program to address the NRC concerns expressed in I&E Bulletin 79-01B.

4. POST ACCIDENT SAMPLING CAPABILITY (II.B.3)

The Post Accident Sampling System will not be installed and operational on January 1, 1982 as required by NUREG-0737 due to problems with equipment availability. The online chemistry sampling analyzer is not scheduled to arrive on site until January 31, 1982. It is our intent that the system will be installed and operational prior to the Turkey Point Unit 3 startup from the steam generator repair outage. At that time, the system will be fully operational and environmentally qualified with the exception of the Unit 4 containment isolation valves. The new valves which are qualified to operate in a post-LOCA environment did not arrive on site until after Unit 4 completed its recent refueling and maintenance shutdown. Since the replacement of these valves requires a reactor shutdown, it is our intent to replace the valves during the Unit 4 steam generator repair outage.

Proposed technical specifications will be submitted to you following installation of the system.



5. TRAINING FOR MITIGATING CORE DAMAGE (II.B.4)

The training program we discussed in L-81-183, dated of April 28, 1981 was completed as required prior to October 1, 1981.

6. SAFETY/RELIEF VALVE TESTING (II.D.1)

It is Florida Power & Light Company's intent subject to the schedular constraints of the EPRI Safety and Relief Valve Test Program, to comply with the revised implementation schedule set forth in Mr. Eisenhut's letter of September 29, 1981 (Generic Letter No. 81-36).

7. VALVE POSITION INDICATION (II.D.3)

Our vendor has completed the environmental qualification tests of the equipment. The test reports will be available for inspection at the Turkey Point site as are the test reports that are required to conform to I & E Bulletin 79-01B.

The results of the environmental qualification tests determined that the charge converter must have an additional enclosure added. The new enclosure assemblies are now on site. It is our intent to add the enclosures on Turkey Point Unit 3 prior to startup from the steam generator repair outage and to add the enclosure assemblies on Unit 4 during its steam generator repair outage.

8. AUXILIARY FEEDWATER SYSTEM EVALUATION (II.E.1.1)

A. AFW SYSTEM FLOW RATE DESIGN BASES AND CRITERIA

In letter L-80-419 dated December 26, 1980, we stated that the analyses required to document the design bases system flow requirements for the AFW system were underway and would be supplied



upon completion. Attachment 1 contains Florida Power & Light Company's final response on the subject. The evaluation addresses Enclosure 2 of the NRC letter of October 16, 1979 as well as position (3) of NUREG-0737 item II.E.1.1.

The loss of main feedwater transient serves as the design basis for the minimum flow required for the smallest capacity single auxiliary feedwater pump for Turkey Point Units 3 & 4.

B. SHORT TERM ITEMS

Our short term modifications to the AFW system have been completed for both units except for those specific items listed below. The redundant Condensate Storage Tank level indicator has not been installed in Unit 3 but will be operational prior to startup from the steam generator repair outage. The modification of two out of three steam supply valves from A.C. to D.C. MOV's has been delayed because of equipment availability problems. The required equipment has a shipping date of March 8, 1982. The modification to provide Lube Oil Cooling from the discharge of the AFW pumps has been delayed due to equipment availability problems. All equipment is on site with the exception of a relief valve that has a shipping date of June 1983. It is intended that the lube oil modification be made within one month of receipt of the valve. The automatic flow control modifications for Unit 3 will be completed prior to startup from the steam generator repair outage.

C. LONG TERM ITEMS

The steam and feedwater piping modifications to insure redundancy in what are now common sections of piping are planned for installation and operability sometime in 1983. A more precise date cannot be given at this time. The new auxiliary feedwater control valves are expected to be shipped by July 2, 1982. It is our intent to perform the majority of the modifications for both units during the Unit 4 steam generator repair effort. It should also be noted that some of



the modifications can only be performed while both plants are shutdown. It is also our intent to phase in the new equipment and not replace it all at one time.

Removal of non-seismic piping from the suction lines for Unit 3 is planned to be completed prior to startup from the steam generator repair outage. The modification on Unit 4 is planned to be done during its steam generator repair outage.

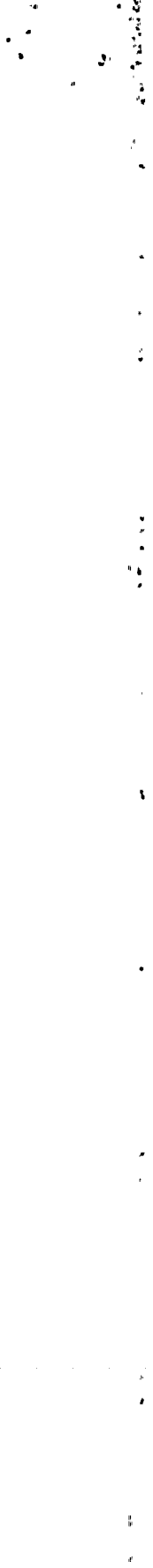
9. AFW INITIATION AND FLOW (II.E.1.2.(2.C))

The modification for safety grade, redundant flow indication has been completed for Unit 4 and will be completed on Unit 3 prior to startup from steam generator repair outage with the following exception. Due to an oversight, the power supplies for the flow indication and flow control are not environmentally and seismically qualified. Qualified power supplies will be installed as soon as possible.

10. CONTAINMENT ISOLATION DEPENDABILITY (II.E.4.2)

A. In our letter L-80-419, dated December 26, 1980 we stated that some of our shorter term modifications originally planned for completion by January 1, 1981 have been rescheduled due to longer than expected lead times for the delivery of safety related valves. These modifications (described in L-80-88, dated March 19, 1980) have been completed for Turkey Point Unit 4. The modifications will be completed on Unit 3 prior to startup from the steam generator repair outage.

B. A Safety Evaluation Report enclosed in a letter from S. A. Varga to R. E. Uhrig dated August 31, 1981 concluded that the requirements of Item II.E.4.2(5) of NUREG-0737, with the additional guidelines



developed by the staff, have been met for the Turkey Point Units. We therefore consider Item II.E.4.2.(5) complete with no modifications necessary.

11. ADDITIONAL ACCIDENT-MONITORING INSTRUMENTATION (II.F.1)

NOBLE GAS MONITORS/IODINE, PARTICULATE SAMPLING (II.F.1(1) AND II.F.2(2))

A. All of the plant effluent monitors have been installed as of January 1, 1982. The plant vent effluent monitor may require additional modifications to provide isokinetic sampling. Our engineering department is currently evaluating whether a modification is necessary. If the modification is determined to be necessary, we will inform you when it is complete and implemented. Although the effluent monitors have been installed, the system must yet be tested and calibrated, procedures written and our technicians trained by the manufacturer in its proper use. We intend to have the system operational (except for isokinetic sampling in the plant vent) by March 1, 1982. We will submit our proposed technical specifications to you prior to that date.

B. CONTAINMENT HIGH RANGE RADIATION MONITOR (II.F.1(3))

The containment high range radiation monitors have been installed in Turkey Point Unit 4 and will be installed in Unit 3 prior to the startup from the current steam generator repair outage. Our proposed technical specifications will be submitted to you prior to March 1, 1982.

C. CONTAINMENT PRESSURE MONITOR (II.F.1(4))

The wide range portion (0-180 psig) of the containment pressure monitors has been installed in Turkey Point Unit 4 and will be installed in Unit 3 prior to startup from the steam generator repair outage. Delivery of the transmitters for the vacuum portion



(0 to -5 psia) of the system is currently scheduled for May 22, 1982. The installation of the transmitters is not outage related and will be done promptly following receipt of the equipment. Our proposed technical specifications for the monitors will be submitted to you once the system is installed.

D. CONTAINMENT WATER LEVEL MONITOR (II.F.1(5))

The containment water level monitors have been installed and are operational in Turkey Point Unit 4 and will be installed in Unit 3 prior to the startup from the steam generator repair outage. Our proposed technical specifications for the monitors will be submitted prior to March 1, 1982.

E. CONTAINMENT HYDROGEN MONITORS (II.F.1(6))

The containment hydrogen monitors have been partially installed in both Turkey Point Units 3 and 4. The delay in installation has been caused by equipment delivery problems. It is our intent to have the system installed and operational by March 1, 1982 with the exception of heat tracing required on the inlet sample lines. The heat tracing is not expected to be shipped by the manufacturer until April 24, 1982. The installation of the heat tracing is not outage related and will be installed promptly following receipt.

Technical specifications for the monitor will be submitted to you once the system is completely installed.

12. INSTRUMENTATION FOR DETECTION OF INADEQUATE CORE COOLING (II.F.2)

INSTALLATION OF LEVEL INSTRUMENTATION

Purchase Orders for the C-E designed heated junction thermocouple system have been issued to the appropriate vendors. It is our intent to have the system installed and operational following the first refueling



outage for each unit after January 1, 1983 dependent upon equipment availability.

13. THERMAL MECHANICAL REPORT--EFFECT OF HIGH PRESSURE INJECTION ON VESSEL INTEGRITY FOR SMALL-BREAK LOSS-OF-COOLANT ACCIDENT WITH NO AUXILIARY FEEDWATER (II.K.2.13)

This item requires a detailed analysis of the thermal-mechanical conditions in the reactor vessel during recovery from small breaks with an extended loss of all feedwater. Westinghouse (in support of the Westinghouse Owners Group) is performing an analysis for generic Westinghouse plant groupings to address this issue which is scheduled to be submitted to the NRC by the end of 1981. This generic study will be applicable to Turkey Point Units 3 & 4 and will be referenced as necessary to completely address NRC concerns.

14. POTENTIAL FOR VOIDING IN THE RCS DURING TRANSIENTS (II.K.2.17)

Westinghouse (in support of the Westinghouse Owners Group) has performed a study which addresses the potential for void formation in Westinghouse designed nuclear steam supply systems during natural circulation cooldown/depressurization transients. This study has been submitted to the NRC by the Westinghouse Owners Group (Letter OG-57, dated April 20, 1981 from R. W. Jurgensen to P.S. Check).

In addition, the Westinghouse Owners Group has developed a natural circulation cooldown guideline that takes the results of the study into account so as to preclude void formation in the upper head region during natural circulation cooldown/depressurization transients, and specifies those conditions under which upper head voiding may occur. These Westinghouse Owners Group generic guidelines have been submitted to the NRC (Letter OG-42 dated November 30, 1981 from R. W. Jurgensen to D. G. Eisenhut). The generic guidance developed by the Westinghouse Owners Group (augmented as appropriate with plant specific consideration) has



been utilized in the preparation of the Turkey Point plant specific operating procedures as described in L-81-513, dated December 4, 1981.

15. SEQUENTIAL AUXILIARY FEEDWATER FLOW ANALYSIS (II.K.2.19)

Subsequent to the issuance of NUREG-0737 and as documented in a letter dated July 1, 1981 from S. A. Varga to R. E. Uhrig, the NRC has completed a generic review on the this subject and concluded that the concerns expressed in Item II.K.2.19 are not applicable to NSSSs with inverted U-tube steam generators such as those designed by Westinghouse. Therefore, this item is not applicable to Turkey Point and no futher action is necessary.

16. AUTOMATIC TRIP OF REACTOR COOLANT PUMPS DURING LOSS-OF-COOLANT ACCIDENT (II.K.3.5)

Westinghouse (in support of the Westinghouse Owners Group) has performed an analysis of delayed reactor coolant pump trip during small-break LOCAs. This analysis is documented in Reference 1. In addition, Westinghouse (again in support of the Westinghouse Owners Group) has performed test predictions of LOFT Experiments L3-1 and L3-6. The results of these predictions are documented in References 2,3, and 4.

Based on: 1) the Westinghouse analysis, 2) the excellent prediction of the LOFT Experiment L3-6 results using the Westinghouse analytical model, and 3) Westinghouse simulator data related to operator response time, the Westinghouse and Florida Power & Light Company position is that automatic reactor coolant pump trip is not necessary since sufficient time is available for manual tripping of the pumps.

Our understanding of the schedule for rinal resolution of this issue is:

- A. Once the NRC formally approves the Westinghouse model, a 3-month study period will ensue during which the Westinghouse Owners Group



will attempt to demonstrate compliance with some NRC acceptance criteria for manual RCP trip. The NRC acceptance criteria will accompany their formal approval of the Westinghouse models.

- B. If, at the end of the 3-month period, the Westinghouse Owners Group cannot show compliance with the acceptance criteria, the NRC will formally notify utilities that they must submit an automatic RCP trip design.

This letter expands our position transmitted to you on August 6, 1981 (Letter L-81-343).

References:

- (1) "Analysis of Delayed Reactor Coolant Pump Trip During Small Loss of Coolant Accidents for Westinghouse Nuclear Steam Supply Systems," WCAP-9584 (proprietary) and WCAP-9585 (non-proprietary), August 1979.
- (2) Letter OG-49, dated March 3, 1981, R.W. Jurgensen (Chairman, Westinghouse Owners Group) to D. F. Ross, Jr. (NRC).
- (3) Letter OG-50, dated March 23, 1981, R. W. Jurgensen (Chairman, Westinghouse Owners Group) to D. F. Ross, Jr. (NRC).
- (4) Letter OG-60, dated June 15, 1981, R. W. Jurgensen (Chairman Westinghouse Owners Group) to P.S. Check (NRC).

17. EFFECT OF LOSS OF AC POWER ON PUMP SEALS (II.K.3.25)

This item requires that the consequences of a loss of RCP seal cooling due to a loss of AC power (defined as loss of offsite power) for at least 2 hours be demonstrated.



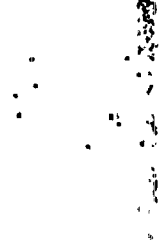
During normal operation, seal injection flow from the chemical and volume control system is provided to cool the RCP seals and the component cooling water system provides flow to the thermal barrier heat exchanger to limit the heat transfer from the reactor coolant to the RCP internals. In the event of loss of offsite power the RCP motor is deenergized and both of these cooling supplies are terminated; however, the diesel generators are automatically started and either seal injection flow or component cooling water to the thermal barrier heat exchanger is automatically restored within seconds. Either of these cooling supplies is adequate to provide seal cooling and prevent seal failure due to loss of seal cooling during a loss of offsite power for at least 2 hours.

18. REVISED SMALL-BREAK LOCA METHODS TO SHOW COMPLIANCE WITH 10CFR50, APPENDIX K (11.K.3.30)

This item requires that the analysis methods used by NSSS vendors and/or fuel suppliers for small-break LOCA analysis for compliance with Appendix K to 10 CFR Part 50 be revised, documented, and submitted for NRC approval.

Westinghouse feels very strongly and Florida Power & Light Company agrees that the small-break LOCA analysis model currently approved by the NRC for use on Turkey Point is conservative and in conformance with Appendix K to 10 CFR Part 50. However, (as documented in Reference 1) Westinghouse believes that improvement in the realism of small-break calculations is a worthwhile effort and has committed to revise its small-break LOCA analysis model to address NRC concerns (e.g., NUREG-0611, NUREG-0623, etc.). This revised Westinghouse model is currently scheduled for submittal to the NRC by April 1, 1982 as documented in Reference 2.

- (1) Letter NS-TMA-2318, dated September 26, 1980, T. M. Anderson (Westinghouse) to D. G. Eisenhut (NRC).
- (2) Letter NS-EPR-2524, dated November 25, 1981, E. P. Rane (Westinghouse) to D. G. Eisenhut (NRC).



ENCLOSURE 1

AWF SYSTEM FLOW RATE DESIGN BASES AND CRITERIA

In the question and answer format that follows, the questions are taken from enclosure 2 of the NRC letter of October 16, 1979.



Question 1

- a. Identify the plant transient and accident conditions considered in establishing AFWS flow requirements, including the following events:
- 1) Loss of Main Feed (LMFW)
 - 2) LMFW w/loss of offsite AC power
 - 3) LMFW w/loss of onsite and offsite AC power
 - 4) Plant cooldown
 - 5) Turbine trip with and without bypass
 - 6) Main steam isolation valve closure
 - 7) Main feed line break
 - 8) Main steam line break
 - 9) Small break LOCA
 - 10) Other transient or accident conditions not listed above.
- b. Describe the plant protection acceptance criteria and corresponding technical bases used for each initiating event identified above. The acceptance criteria should address plant limits such as:
- 1) Maximum RCS pressure (PORV or safety valve actuation)
 - 2) Fuel temperature or damage limits (DNB, PCT, maximum fuel central temperature)
 - 3) RCS cooling rate limit to avoid excessive coolant shrinkage
 - 4) Minimum steam generator level to assure sufficient steam generator heat transfer surface to remove decay heat and/or cool down the primary system.

Response to 1.a

The Auxiliary Feedwater System serves as a backup system for supplying feedwater to the secondary side of the steam generators at times when the feedwater system is not available, thereby maintaining the heat sink capabilities of the steam generator. As an Engineered Safeguards System, the Auxiliary Feedwater System is directly relied upon to prevent core damage and system overpressurization in the event of transients such as a loss of normal feedwater or a secondary system pipe rupture, and to provide a means for plant cooldown following any plant transient.

Following a reactor trip, decay heat is dissipated by evaporating water in the steam generators and venting the generated steam either to the condensers through the steam dump or to the atmosphere through the steam generator safety valves or the power-operated relief valves. Steam generator water inventory must be maintained at a level sufficient to ensure adequate heat transfer and continuation of the decay heat removal process. The water level is maintained under these circumstances by the Auxiliary Feedwater System which delivers an emergency water supply to the steam generators. The Auxiliary Feedwater System must be capable of functioning for extended periods, allowing time either to restore normal feedwater flow or to proceed with an orderly cooldown of the plant to the reactor coolant temperature where the Residual Heat Removal System can assume the burden of decay heat removal. The Auxiliary Feedwater



System flow and the emergency water supply capacity must be sufficient to remove core decay heat, reactor coolant pump heat, and sensible heat during the plant cooldown. The Auxiliary Feedwater System can also be used to maintain the steam generator water levels above the tubes following a LOCA. In the latter function, the water head in the steam generators serves as a barrier to prevent leakage of fission products from the Reactor Coolant System into the secondary plant.

DESIGN CONDITIONS

The reactor plant conditions which impose safety-related performance requirements on the design of the Auxiliary Feedwater System are as follows for Turkey Point Units 3 and 4

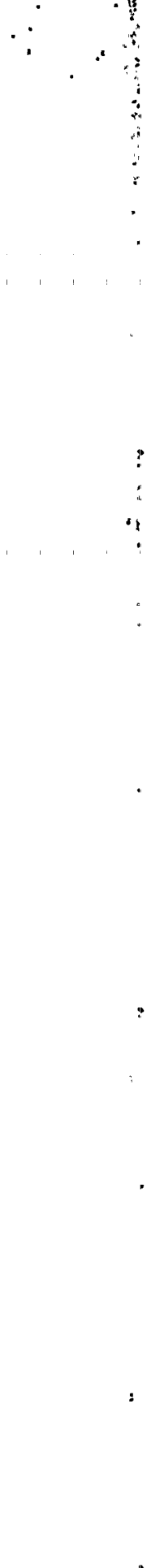
- Loss of Main Feedwater Transient
 - Loss of main feedwater with offsite power available
 - Station blackout (i.e., loss of main feedwater without offsite power available)
- Rupture of a Main Steam Line
- Loss of all AC Power
- Loss of Coolant Accident (LOCA)
- Cooldown

Loss of Main Feedwater Transients

The design loss of main feedwater transients are those caused by:

- Interruptions of the Main Feedwater System flow due to a malfunction in the feedwater or condensate system
- Loss of offsite power or blackout with the consequential shutdown of the system pumps, auxiliaries, and controls

Loss of main feedwater transients are characterized by a rapid reduction in steam generator water levels which results in a reactor trip, a turbine trip, and auxiliary feedwater actuation by the protection system logic. Following reactor trip from high power, the power quickly falls to decay heat levels. The water levels continue to decrease, progressively uncovering the steam generator tubes as decay heat is transferred and discharged in the form of steam either through the steam dump valves to the condenser or through the steam generator safety or power-operated relief valves to the atmosphere. The reactor coolant temperature increases as the residual heat in excess of that dissipated through the steam generators is absorbed. With increased temperature, the volume of reactor coolant expands and begins filling the pressurizer. Without the addition of sufficient auxiliary feedwater, further expansion will result in water being discharged through the pressurizer safety and relief valves. If the temperature rise and the resulting volumetric



expansion of the primary coolant are permitted to continue, then (1) pressurizer safety valve capacities may be exceeded causing overpressurization of the Reactor Coolant System and/or (2) the continuing loss of fluid from the primary coolant system may result in bulk boiling in the Reactor Coolant System and eventually in core uncovering, loss of natural circulation, and core damage. If such a situation were ever to occur, the Emergency Core Cooling System would be ineffectual because the primary coolant system pressure exceeds the shutoff head of the safety injection pumps, the nitrogen over-pressure in the accumulator tanks, and the design pressure of the Residual Heat Removal Loop. Hence, the timely introduction of sufficient auxiliary feedwater is necessary to arrest the decrease in the steam generator water levels, to reverse the rise in reactor coolant temperature, to prevent the pressurizer from filling to a water solid condition, and eventually to establish stable hot standby conditions. Subsequently, a decision may be made to proceed with plant cooldown if the problem cannot be satisfactorily corrected.

The blackout transient differs from a simple loss of main feedwater in that emergency power sources must be relied upon to operate vital equipment. The loss of power to the electric driven condenser circulating water pumps results in a loss of condenser vacuum and condenser dump valves. Hence, steam formed by decay heat is relieved through the steam generator safety valves or the power-operated relief valves. The calculated transient is similar for both the loss of main feedwater and the blackout, except that reactor coolant pump heat input is not a consideration in the blackout transient following loss of power to the reactor coolant pump bus.

The Loss of Main Feedwater transient serves as the basis for the minimum flow required for the smallest capacity single auxiliary feedwater pump for Turkey Point Units 3 and 4. The pump is sized so that any single pump will provide sufficient flow against the steam generator safety valve set pressure (with 3% accumulation) to prevent water relief from the pressurizer. The same criterion is met for the Station Blackout transient, where A/C power is assumed to be unavailable.

Rupture of a Main Steam Line

Because the rupture of a main steam line may result in the complete blowdown of one steam generator, a partial loss of the plant heat sink is a concern. The main steamline rupture accident conditions are characterized initially by plant cooldown, and hence, auxiliary feedwater flow is not needed during the early stage of the transient to remove decay heat from the Reactor Coolant System. Provisions must be made in the design of the auxiliary feedwater system to allow termination of flow to the faulted loop and to provide flow to the intact steam generators during the controlled cooldown following the steamline break accident.



Loss of All AC Power

The loss of all AC power is postulated as resulting from accident conditions wherein not only onsite and offsite AC power is lost but also AC emergency power is lost as an assumed common mode failure. Although this accident scenario is not a design basis for the Turkey Point 3 and 4 units, features are incorporated into the design to provide auxiliary feedwater independent of this sequence. Steam is provided from each of the three steam lines to power three turbine driven auxiliary feedwater pumps, each of which can deliver flow to all steam generators through a common header and maintain the plant at hot shutdown until AC power is restored.

Loss-of-Coolant Accident (LOCA)

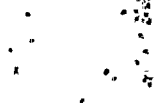
The loss of coolant accidents do not impose on the auxiliary feedwater system any flow requirements in addition to those required by the other accidents addressed in this response. The following description of the small LOCA is provided here for the sake of completeness to explain the role of the auxiliary feedwater system in this transient.

Small LOCA's are characterized by relatively slow rates of decrease in reactor coolant system pressure and liquid volume. The principal contribution from the Auxiliary Feedwater System following such small LOCAs is basically the same as the system's function during hot shutdown or following a spurious safety injection signal which trips the reactor. Maintaining a water level inventory in the secondary side of the steam generators provides a heat sink for removing decay heat and establishes the capability for providing a buoyancy head for natural circulation. The auxiliary feedwater system may be utilized to assist in a system cooldown and depressurization following a small LOCA while bringing the reactor to a cold shutdown condition.

Cooldown

The cooldown function performed by the Auxiliary Feedwater System is a partial one since the reactor coolant system is reduced from normal zero load temperatures to a hot leg temperature of approximately 350°F. The latter is the maximum temperature recommended for placing the Residual Heat Removal System (RHRS) into service. The RHRS completes the cooldown to cold shutdown conditions.

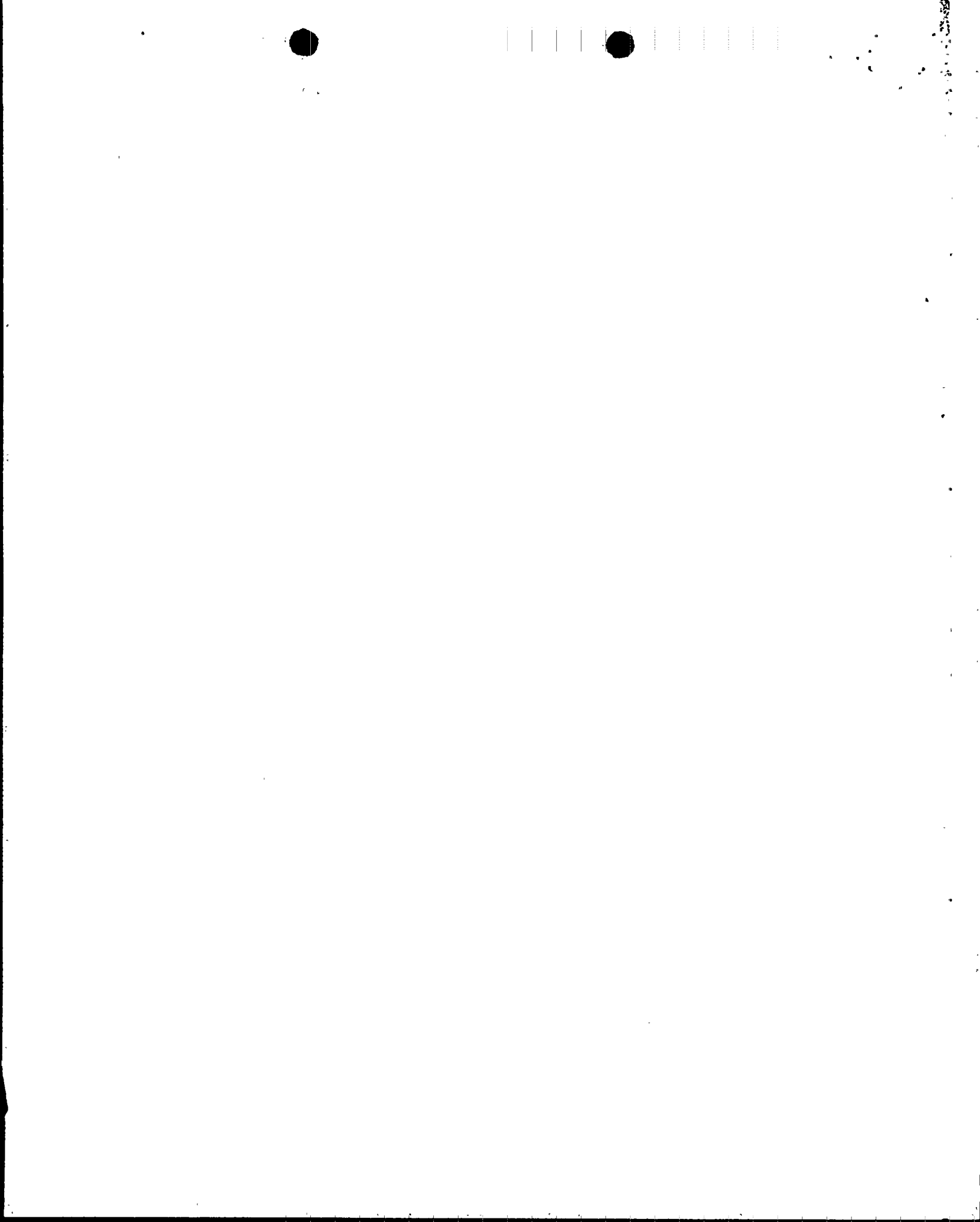
Cooldown may be required following expected transients, following an accident such as a main feedline break, or it may be a normal cooldown prior to refueling or performing reactor plant maintenance. If the reactor is tripped following extended operation at rated power level, the AFWS is capable of delivering sufficient AFW to remove decay heat and reactor coolant pump (RCP) heat following reactor trip while maintaining the steam generator (SG) water level. Following transients or accidents, the recommended cooldown rate is consistent with expected needs and at the same time does not impose additional requirements on the capacities of the auxiliary feedwater pumps, considering a single failure. In any event, the process consists of being able to dissipate plant sensible heat in addition to the decay heat produced by the reactor core.



Response to 1.b

Table 1B-1 summarizes the criteria which are the general design bases for each event, discussed in the response to Question 1.a, above. Specific assumptions used in the analyses to verify that the design bases are met are discussed in response to Question 2.

The primary function of the Auxiliary Feedwater System is to provide sufficient heat removal capability for heatup accidents following reactor trip to remove the decay heat generated by the core and prevent system overpressurization. Other plant protection systems are designed to meet short term or pre-trip fuel failure criteria. The effects of excessive coolant shrinkage are bounded by the analysis of the rupture of a main steam pipe transient. The maximum flow requirements determined by other bases are incorporated into this analysis, resulting in no additional flow requirements.



Question 2

Describe the analyses and assumptions and corresponding technical justification used with plant condition considered in 1.a above including:

- a. Maximum reactor power (including instrument error allowance) at the time of the initiating transient or accident.
- b. Time delay from initiating event to reactor trip.
- c. Plant parameter(s) which initiates AFWS flow and time delay between initiating event and introduction of AFWS flow into steam generator(s).
- d. Minimum steam generator water level when initiating event occurs.
- e. Initial steam generator water inventory and depletion rate before and after AFWS flow commences -- identify reactor decay heat rate used.
- f. Maximum pressure at which steam is released from steam generator(s) and against which the AFW pump must develop sufficient head.
- g. Minimum number of steam generators that must receive AFW flow; e.g., 1 out of 2? 2 out of 4?
- h. RC flow condition -- continued operation of RC pumps or natural circulation.
- i. Maximum AFW inlet temperature.
- j. Following a postulated steam or feed line break, time delay assumed to isolate break and direct AFW flow to intact steam generator(s). AFW pump flow capacity allowance to accommodate the time delay and maintain minimum steam generator water level. Also identify credit taken for primary system heat removal due to blowdown.
- k. Volume and maximum temperature of water in main feed lines between steam generator(s) and AFWS connection to main feed line.
- l. Operating condition of steam generator normal blowdown following initiating event.
- m. Primary and secondary system water and metal sensible heat used for cooldown and AFW flow sizing.
- n. Time at hot standby and time to cooldown RCS to RHR system cut in temperature to size AFW water source inventory.



Analyses have been performed for the Loss of Main Feedwater and the loss of offsite AC power to the Station, the transients which define the AFWS performance requirements. These analyses have been provided for review and have been approved in the Applicant's FSAR.

In addition to the above analyses, calculations have been performed specifically for Turkey Point Units 3 and 4 to determine the plant cooldown flow (storage capacity) requirements. The LOCA analysis, as discussed in response 1.b, incorporates the system flow requirements as defined by other transients, and therefore is not performed for the purpose of specifying AFWS flow requirements. Each of the analyses listed above are explained in further detail in the following sections of this response. /

Loss of Main Feedwater (Blackout)

A loss of main feedwater analysis was performed in FSAR Section 14.1.11 for the purpose of showing that a single auxiliary feedwater pump delivering flow to two steam generators does not result in filling the pressurizer. Furthermore, the peak RCS pressure remains below the criterion for Condition II transients and no fuel failures occur (refer to Table 1B-1). Table 2-1 summarizes the assumptions used in this analysis. The transient analysis begins at the time of reactor trip. This can be done because the trip occurs on a steam generator level signal, hence the core power, temperatures and steam generator level at time of reactor trip do not depend on the event sequence prior to trip. Although the time from the loss of feedwater until the reactor trip occurs cannot be determined from this analysis, this delay is expected to be 50-60 seconds. The analysis assumes that the plant is initially operating at 102% (calorimetric error) of 2300 Mwt. A very conservative assumption is made in defining decay heat and stored energy in the RCS. The reactor is assumed to be tripped on low-low steam generator level, allowing for level uncertainty. The FSAR shows that there is a considerable margin with respect to filling the pressurizer. A Station Blackout transient with the assumption that the smallest auxiliary feedwater pump operates results in even more margin.

This analysis establishes the capacity of the smallest single pump and also establishes train association of equipment so that this analysis remains valid assuming the most limiting single failure.

Plant Cooldown

Minimum flow requirements from the previously discussed transients meet the flow requirements of plant cooldown. This operation, however, defines the basis for tankage size, based on the required cooldown duration, maximum decay heat input and maximum stored heat in the



system. As previously discussed in response 1A, the auxiliary feedwater system partially cools the system to the point where the RHRS may complete the cooldown, i.e., 350°F in the RCS. Table 2-1 shows the assumptions used to determine the cooldown heat capacity of the auxiliary feedwater system.

The cooldown is assumed to commence at 2345 MWt power, and maximum trip delays and decay heat source terms are assumed when the reactor is tripped. Primary metal, primary water, secondary system metal and secondary system water are all included in the stored heat to be removed by the AFWS. See Table 2-2 for the items constituting the sensible heat stored in the NSSS.

This operation is analyzed to establish minimum tank size requirements for auxiliary feedwater fluid source which are normally aligned.



QUESTION #3

Verify that the AFW pumps in your plant will supply the necessary flow to the steam generator(s) as determined by items 1 and 2 above considering a single failure. Identify the margin in sizing the pump flow to allow for pump recirculation flow, seal leakage and pump wear.

RESPONSE TO #3

Figure 3-1 schematically shows the major features and components of the Auxiliary Feedwater System for Turkey Point Units 3 and 4. Flow rates for the design transients described in Response 2 are tabulated in Table 3-1 considering the following single failures.

- A. A/C Train Failure
- B. Pump Failure
- C. AFWS Flow Control Valves Failure (failure to assume proper preset position)

Modifications being made to the system, which include automatic flow control and redundant flow paths, will automatically provide a minimum of 200 gpm to each steam generator. NOTE: Figure 3-1 does not reflect the proposed modifications.

The Turkey Point Units 3 & 4 auxiliary feedwater pumps were procured to supply a net flow of 600 gpm at 1191 psia. The minimum recirculation flow and seal leakage are added to this flow to obtain the design point for the pump.



TABLE 1B-1

Criteria for Auxiliary Feedwater System Design Basis Conditions

<u>Condition or Transient</u>	<u>Classification*</u>	<u>Criteria*</u>	<u>Additional Design Criteria</u>
Loss of Main Feedwater	Condition II	Peak RCS pressure not to exceed design pressure. No consequential fuel failures	
Station Blackout	Condition II	(same as LMFW)	Pressurizer does not fill with 1. single aux. feed pump feeding 2 SGs.
Loss of all A/C Power	N/A	Note 1	
Loss of Coolant	Condition III	10 CFR 100 dose limits 10 CFR 50 PCT limits	
	Condition IV	10 CFR 100 dose limits 10 CFR 50 PCT limits	
Cooldown	N/A		100°F/hr 547°F to 350°F

*Ref: ANSI N18.2 (This information provided for those transients performed in the FSAR).

Note 1 Although this transient establishes the basis for AFW pump powered by a diverse power source, this is not evaluated relative to typical criteria since multiple failures must be assumed to postulate this transient.



TABLE 2-1

Summary of Assumptions Used in AFW Design Verification Analyses

<u>Transient</u>	<u>Loss of Feedwater (station blackout)*</u>	<u>Cooldown</u>
a. Max reactor power	102% of 2300 MWt	2345 MWt
b. Time delay from trip signal to rod motion	2 sec	2 sec
c. AFW actuation signal/time delay for AFW flow	10-10 SG level/ 3 minutes	NA
d. SG water level at time of reactor trip	10-10 SG level (10 SG level)	NA
e. Initial SG inventory	37,314 lbm/SG at trip (52,000 lbm/SG at trip)	37,200 lbm/SG @ 516°F
Rate of change before & after AFW actuation	See FSAR Section 14.1.11 and 14.1.12	N/A
decay heat	ANS + 20%	N/A
f. AFW pump design	1133 psia	1133 psia
g. Minimum # of SGs which must receive AFW flow	2 of 3	N/A
h. RC pump status	Tripped @ reactor trip	Tripped
i. Maximum AFW temperature	120°F	100°F
j. Operator action	N/A	N/A
k. MFW purge volume/temp.	182 ft ³ /440°F	450 ft ³ / 440°F
l. Normal blowdown	none assumed	none assumed
m. Sensible heat	see cooldown	Table 2-2
n. Time at standby/time to cooldown to RHR	2 hr/4 hr	2 hr/4 hr
o. AFW flow rate	600 GPM - constant (min. requirement)	variable

* Value shown only if different from Loss of Feedwater

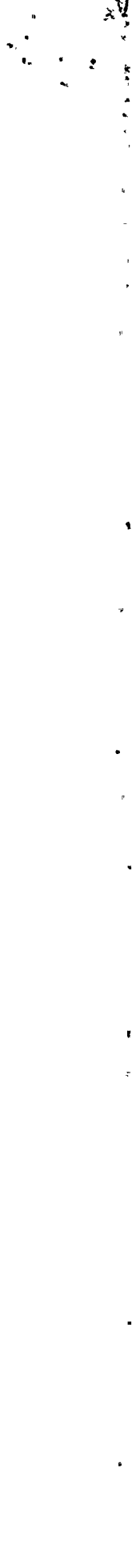


TABLE 2-2

Summary of Sensible Heat Sources

Primary Water Sources (initially at 2345 Mwt power temperature and inventory)

- RCS fluid
- Pressurizer fluid (liquid and vapor)

Primary Metal Sources (initially at 2345 Mwt power temperature)

- Reactor coolant piping, pumps and reactor vessel
- Pressurizer
- Steam generator tube metal and tube sheet
- Steam generator metal below tube sheet
- Reactor vessel internals

Secondary Water Sources (initially at 2345 Mwt power temperature and inventory)

- Steam generator fluid (liquid and vapor)
- Main feedwater purge fluid between steam generator and AFWS piping.

Secondary Metal Sources (initially at 2345 Mwt power temperature)

- All steam generator metal above tube sheet, excluding tubes.



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TABLE 3-1

AUXILIARY FEEDWATER FLOW (1) TO STEAM GENERATORS
FOLLOWING AND ACCIDENT/TRANSIENT WITH SELECTED SINGLE FAILURE - GPM

<u>Accident/Transient</u>	Single Failure		
	<u>Elec. A.C. Train Failure</u>	<u>Pump Failure</u>	<u>CV(2) Failure</u>
	A	B	C
1. Loss of Main FW	600 (3)	600 (3)	600
2. Blackout	600 (3)	600 (3)	600
3. Cooldown	600	600	600

NOTES:

1. Items 1 thru 3 are minimum expected flows to intact loops.
2. Including only those CVs in the AFWS. "Failure" is defined as failure of the valve to assume its proper preset controlled position.
3. Flow is automatically initiated and controlled to 200 gpm to each steam generator.



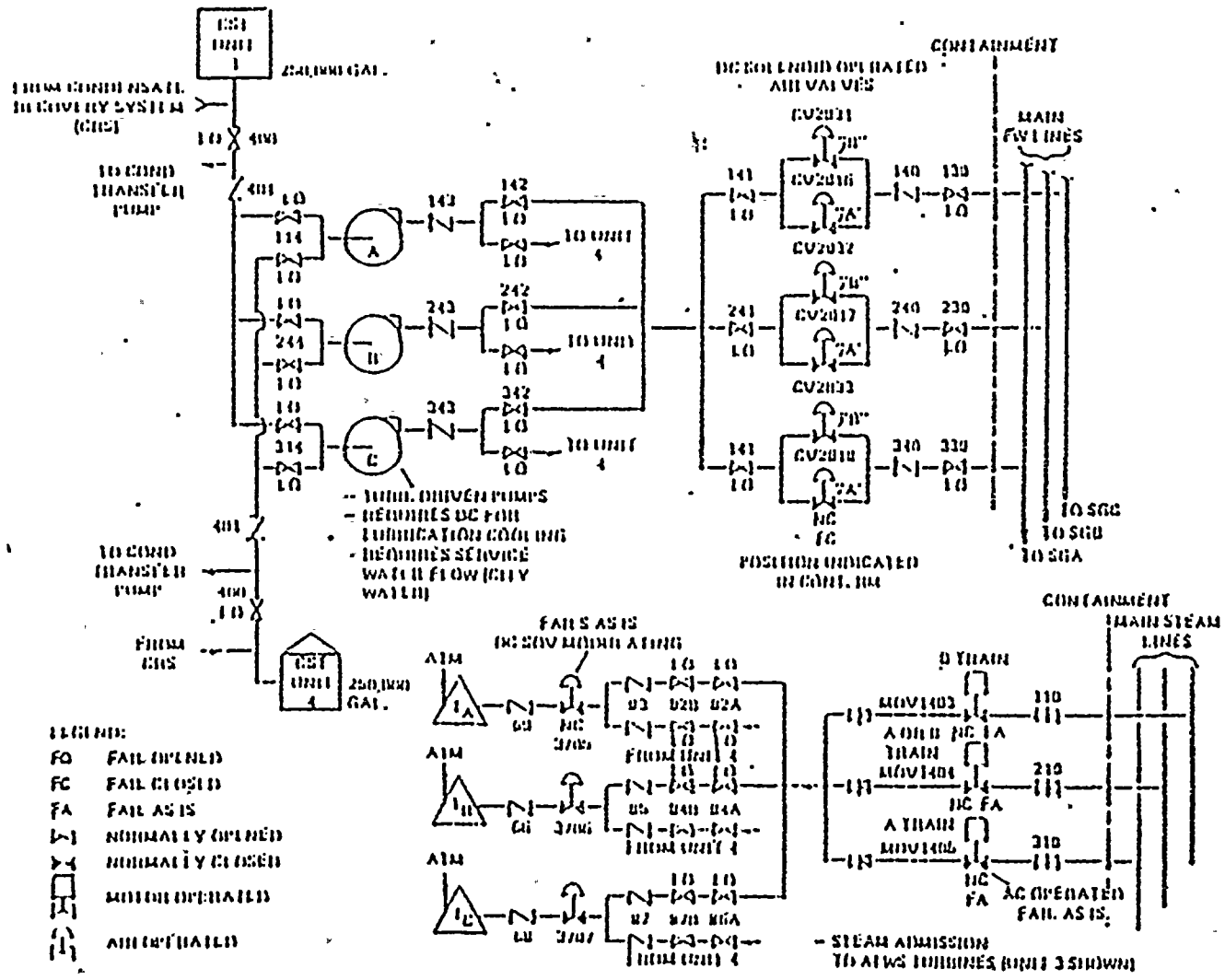


Figure 3-1

Auxiliary Feedwater System

Turkey Point Units 3 and 4



4 11 1

Attachment #1

WESTINGHOUSE NSD CERTIFICATE OF CONFORMANCE

Exceptions to Florida Power and Light Company Purchase Order #93000-81417 and DWA #38608 with regards to the NRC questions 1 through 3 on AFWS flow requirements design basis information.

Westinghouse -

1. Did not specifically address events (5), (6) and (7) in its response to Question 1 Section A.
2. Did not specifically address events (5), (6), (7) and (8) in its response to Question 1 Section B.
3. Did not, in providing plant protection acceptance criteria, specifically address the four "plant limits" cited in Question 1 Section B for each event. (Reference: Table 1B-1)
4. Did not address events (2), (3), (5), (6), (7), (8) and (9) in its response to Question 2.
5. Did not address margin in sizing the pump flow to allow for recirculation flow, etc.

Justification of exceptions taken to Florida Power and Light Company's order for providing the AFWS design basis flow requirements-responses to NRC questions:

1. The transients and conditions resulting from events (5) and (6) which impose safety-related performance requirements on the design of the AFWS are bounded by the transients and conditions resulting from other events which were specifically addressed by Westinghouse in its response to Question 1 Section A.

Although event (7) typically serves as a design basis for plants which must meet today's licensing requirements, it is not part of the design basis for this plant and is not discussed in the response to Question 1 Section A.

2. Same justification as in (1) above. See paragraph 3 below for justification for omitting (8) in the response to Question 1 Section B.
3. The "plant limits" cited in the NRC question which should be addressed by the acceptance criteria were cited as examples of plant limits (Note the use of the phrase "such as") to be addressed and not a requirement. In two cases, the so-called "plant limits" cited in Section B (Items 3 and 4) are not plant limits under any recognized licensing basis and are somewhat undefined as stated - e.g., in our opinion there is not such thing as a plant limit on "RCS cooling rate to avoid excessive coolant shrinkage" and the meaning of "excessive" is subject to debate in this context.

The Westinghouse response to this question provided the plant protection acceptance criteria which were part of the original design and licensing basis for these events. Where these criteria can be related to limits on specific equipment or systems, the response included these limits.

4. a) Discussion of event (2) was unnecessary in the response to Question 2 because event (1) is more limiting with respect to defining minimum AFWS flow requirements than event (2). This was noted in the response to Question 1 Section A. Differences between event (1) and event (2) assumptions are shown in Table 2-1 however.
 - b) As was noted in the response to Question 1 Section A, event (3) does not impact the establishment of AFWS flow requirements. Its impact is only in determining (or indicating) the necessity for power and control for an AFWS pump which is not dependent on AC power and can maintain the plant at hot shutdown until AC power is restored. Since this event has no impact on determining AFWS flow requirements, it was not addressed in the response to Question 2.
 - c) As noted previously in (1) above, events (5) and (6) are bounded by other events and thus do not determine the limiting (max/min) flow requirements for the AFWS and thus are not addressed in the response to Question 2.
 - d) As noted previously in (1) above, events (7) and (8) are not a part of the design basis for this plant and therefore are not addressed in Question 2.
 - e) Event (9), as noted in the response to Question 2, is not used in determining design basis flow requirements for the AFWS.
5. Westinghouse did not address design margin in the response to Question 3 because Westinghouse did not design the AFWS. Westinghouse's role was to specify design criteria which the designer (customer utility or Architect Engineer) had to meet.

