



UNITED STATES  
NUCLEAR REGULATORY COMMISSION  
WASHINGTON, D. C. 20555

NOV 21 1978

Docket No. 50-382

Mr. D. L. Aswell  
Vice President, Power Production  
Louisiana Power & Light Company  
142 Delaronde Street  
New Orleans, Louisiana 70174

Dear Mr. Aswell:

SUBJECT: ACCEPTANCE REVIEW OF WATERFORD UNIT NO. 3 APPLICATION FOR AN  
OPERATING LICENSE

On September 28, 1978 you tendered an application for an operating license for Waterford Steam Electric Station Unit 3. Your application included the Final Safety Analysis Report and the Environmental Report.

We have completed our review of the application and have concluded that it is acceptable for docketing.

We are developing a schedule for the detailed safety and environmental reviews and will advise you of the key milestones.

During the course of our preliminary review of your FSAR, Enclosure 1 "Request for Additional Information" was generated. The requests are of the type that require an early response for our mutual benefit during the ensuing detailed technical review. Please advise the licensing Project Manager, as soon as possible, of your schedule for docketing the FSAR and for responding to these requests for additional information. Once these dates have been established, we will be able to prepare a schedule for the review of the FSAR.

We consider the environmental report to be sufficiently complete to permit us to accept it for docketing. However, we have identified several areas that require additional information. A request for this information, which will serve as the basis of the agenda for an environmental site visit, will be sent to you prior to the site visit. The site visit will be scheduled at a future date.

We note that we have not received the proposed environmental technical specifications (ETS) as specified in Regulatory Guide 4.2 (Section 6.2). Under separate cover we are sending you a copy of the ETS for the William B. McGuire Nuclear Station, which may be used as a guide in the preparation of the proposed Waterford Unit 3 ETS. The ETS, when approved, will form Appendix B to the Waterford Unit 3 Operating License.

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ADJOINING STATES

Not applicable

OTHERS

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Mr. D. L. Aswell

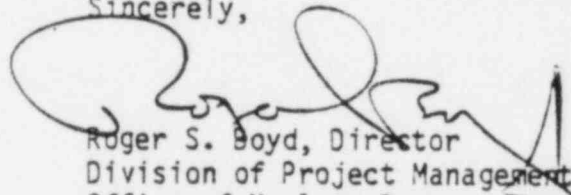
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Your filing of the application and any amendments thereof should include three originals signed under oath or affirmation by a duly authorized officer of your organization. In addition, your filing should include fifteen copies of that portion of the application containing the general information and forty copies of the safety analysis report. As required by 10 CFR 50.30, you should retain an additional ten copies of the general information and thirty copies of the safety analysis report for direct distribution in accordance with the enclosed Distribution List (Enclosure 2) and further instructions that might be provided later. Within ten days after docketing, you must provide an affidavit that distribution in accordance with the Distribution List has been completed. These requirements also apply to all subsequent amendments to your application.

If, during the course of our review, you believe there is a need to appeal a staff position because of disagreement, this need should be brought to the staff's attention as early as possible so that the appropriate meeting can be arranged on a timely basis. A written request is not necessary and all such requests should be initiated through our staff project manager assigned to the review of your application. This procedure is an informal one, designed to allow opportunity for applicants to discuss, with management, areas of disagreement in the case review.

Sincerely,



Roger S. Boyd, Director  
Division of Project Management  
Office of Nuclear Reactor Regulation

1. Request for Additional Information
2. Distribution List

ccs w/enclosures:  
See next page

Mr. D. L. Aswell

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cc: W. Malcolm Stevenson, Esq.  
Monroe & Lemann  
1424 Whitney Building  
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Mr. E. Blake  
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Mr. D. B. Lester  
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ENCLOSURE 1

REQUEST FOR ADDITIONAL INFORMATION

PRELIMINARY REVIEW

WATERFORD UNIT 3

DOCKET NO. 50-382

AUXILIARY SYSTEMS BRANCH  
ACCEPTANCE REVIEW  
WATERFORD STATION, UNIT NO. 3  
DOCKET NO. 50-382

010.1  
(3.4.1)

Expand Section 3.4.1 to include the following:

1. Identify the safety-related systems and components necessary for safe shutdown that should be protected against floods, and show the relationship between the system elevation and the design basis flood levels and conditions established for the site.
2. Describe the structures that house safety-related equipment, and identify the location of exterior or access openings and penetrations that are below the design flood levels.
3. Describe the flood protection provided (e.g., pumping systems, stoplogs, water tight doors, and drainage systems) for safety-related equipment because of its location and the potential of inleakage from such occurrences as cracks in structure walls, leaking water stops, and effects of wind wave action.

010.2  
(3.5)

Provide a tabulation of all safety-related components which are located outdoors and describe the protection to be afforded to these components to prevent their being damaged by tornado generated missiles. Include in this tabulation all HVAC system air intakes and exhausts. Identify the locations of these components, air intakes, and exhausts on the plant arrangement drawings.

010.3  
(3.6)

With respect to your high and moderate energy line analysis provide the following information:

(a) Layout drawings of the safety-related areas outside containment showing the high and moderate energy piping systems and their relation to the safety-related equipment. Indicate the method of protection provided against a high energy piping system failure for each system listed, as well as the method of protection for moderate energy piping systems. Provide results of analyses of the effects on the safety-related systems for all high and moderate energy piping system failure in accordance with our Branch Technical Position ASB 3-1.

(b) Identify areas of system piping where no breaks are postulated (including lengths of pipe such as those located in the main steam and feedwater lines).

010.4  
(3.6)  
(RSP)

We require that the compartments which house the main steam lines and feedwater lines and the isolation valves for those lines, be designed to consider the environmental effects (pressure, temperature, humidity) and potential flooding consequences from an assumed crack, equivalent to the flow area of a single ended pipe rupture in these lines. We require that essential equipment located within the compartment, including the main steam isolation and feedwater valves and their operators be capable of operating in the environment resulting from the above crack.

We also will require that if this assumed crack could cause the structural failure of this compartment, then the failure should not jeopardize the safe shutdown of the plant. In addition, we require that the remaining portion of the enclosed pipe between the containment and the turbine building meet the guidelines of Branch Technical Position APCSB 3-1.

We require that you submit a subcompartment pressure analysis to confirm that the design of the pipe enclosure conforms to our position as outlined above.

We request that you evaluate the design against this staff position, and advise us as to the outcome of your review, including any design changes which may be required. The evaluation should include a verification that the methods used to calculate the pressure build-up in the subcompartments outside of the containment for postulated breaks are the same as those used for subcompartments inside the containment. Also, the allowance for structural design margins (pressure) should be the same. If different methods are used, justify that your method provides adequate design margins and identify the margins that are available. When you submit the results of your evaluation, identify the computer codes used, the assumptions used for mass and energy release rates, and sufficient design data so that we may perform independent calculations.



The peak pressures and temperatures resulting from the postulated break of a high energy pipe located in compartments or buildings is dependent on the mass and energy flows during the time of the break.

Provide the information necessary to determine what terminates the blowdown or to determine the length of time blowdown exists. For each pipe break or leakage crack analyzed, provide the total blowdown time and the mechanism used to terminate or limit the blowdown time of flow so that the environmental effects will not affect safe shutdown of the facility.

010.5 (9.0) Provide a tabulation of all valves in the reactor pressure boundary and in other seismic Category I systems (per Regulatory Guide 1.29) whose operation is relied upon either to assure safe plant shutdown or to mitigate the consequences of a transient or accident. The tabulation should identify the system in which it is installed, the type and size of valve, the actuation type(s), and the environmental conditions to which the valves are qualified.

010.5 (9.1) Provide a more complete description of the containment polar crane and the spent fuel pool cask crane and indicate whether it is in accordance with guidelines of Branch Technical Position ASB 9-1.

010.7 (9.1.2) Provide a layout drawing indicating the path of travel of the spent fuel cask. Show that the spent fuel cask does not travel over spent fuel or safety-related equipment.

010.8  
(9.2.5)

In order to permit an assessment of the ultimate heat sink, provide the results of an analysis of the thirty-day period following a design basis accident.

In submitting the results of the analysis requested, include the following information in both tabular and graphical presentations:

- (1) The total integrated decay heat.
- (2) The heat rejection rate and integrated heat rejected by the station auxiliary systems, including all operating pumps, ventilation equipment, diesels, spent fuel pool makeup, and other heat sources for both units.
- (3) The heat rejection rate and integrated heat rejected due to the sensible heat removed from containment and the primary system.
- (4) The total integrated heat rejected due to the above.
- (5) The maximum allowable inlet water temperature from the ultimate heat sink taking into account the rate at which the heat energy must be removed, cooling water flow rate, and the capabilities of the respective heat exchangers.
- (6) The required and available normal pump suction head (NPSH) to the ultimate heat sink pumps at the minimum ultimate heat sink water level.

The above analysis, including pertinent backup information, is to demonstrate the capability to provide adequate water inventory and provide sufficient heat dissipation to limit essential cooling water operating temperatures within the design ranges of system components.

Use the methods set forth in Branch Technical Position ASB 9-2, to establish the heat produced due to fission product decay and heavy element decay. Assume an initial cooling water temperature based on the most adverse conditions.

010.9  
(9.3.3)

To adequately evaluate Section 9.3.3, "Equipment and Floor Drainage Systems," provide additional information and detail explaining what is provided in each safety-related compartment or area to assure the plant can be safely shut down after a postulated pipe break or crack in any system passing through or terminating in the compartment or area. Describe the protection provided, i.e., equipment or isolable compartment structure or area. In the case where equipment is provided for protection of the safety-related components or systems, describe what protective equipment is provided, where it is installed, and what function(s) does it perform to assure protection from flooding of the safety-related equipment in the compartment or area. Indicate what operator action, if any, and within what time interval it is required to prevent flooding of safety-related equipment. Also, provide the results of an analysis that demonstrates compartment and/or area drains serving safety-related components or systems have been sized for

maximum flow conditions.

010.10  
(10.4.5)

Expand Section 10.4.5 to include a discussion of the means provided to prevent potential flooding of safety-related equipment due to the failure of a system component such as a circulating water pipe line expansion joint.

010.11  
(10.4.7)  
(RSP)

The steam generators are of the top feed type with a feedwater sparger. Provide the necessary information to show that you will comply with the guidelines of ASB 10-2 "Design Guidelines for Water Hammers in Steam Generators with Top Feeding Designs."

Acceptance Review  
Containment Systems Branch  
Waterford Steam Electric Station, Unit 3

022.1 Include the following in the discussions of the main steam line

(6.2) break analyses:

- (a) Provide a single active failure analysis which specifically identifies those safety grade systems and components relied upon to limit the mass and energy release and containment pressure/temperature response. The single failure analysis should include, but not necessarily be limited to auxiliary feedwater and connected systems isolation; feedwater, condensate, and auxiliary feedwater pump trip; the loss of offsite power; diesel failure when loss of offsite power is evaluated; and partial loss of containment cooling systems. Provide justification for the reliance on any equipment which is nonsafety grade in whole or in part.
- (b) Specify and justify the temperature used in the calculation of condensing heat transfer to the passive heat sinks; i.e., specify whether the saturation temperature corresponding to the partial pressure of the vapor, or the atmosphere temperature which may be superheated was used.
- (c) Discuss and justify the analytical model including the thermodynamic equations used to account for the removal of the condensed mass from the containment atmosphere due to condensing heat transfer to the passive heat sinks;

- (d) For the case which results in the maximum containment atmosphere temperature, graphically show the primary shield wall temperature as a function of time;
- (e) Specify and justify the design temperature of the containment primary shield wall, the design temperature of the internal concrete structures, and the temperature used to qualify the safety-related instrumentation located within the containment.

022.2 Provide the following information to supplement the subcompartment  
(c ) analyses presented in Section 6.2.1.6:

- (a) For each pipe break assumed, specify whether the pipe break was postulated for the evaluation of the compartment structural design, component supports design or both.
- (b) For each compartment, provide a table of blowdown mass flow rate and energy release rate as a function of time for the break which results in the maximum structural load, and for the break which was used for the component supports evaluation.
- (c) Provide a schematic drawing showing the compartment nodalization for the determination of maximum structural loads, and for the component supports evaluation. Provide sufficiently detailed plan and section drawings for several views, including principal dimensions, showing the arrangement of the compartment structure,

major components, piping, and other major obstructions and vent areas to permit verification of the subcompartment nodalization and vent locations. Figures 6.2-13e through j lack sufficient detail to permit verification of the subcompartment nodalization.

- (d) Describe the nodalization sensitivity study performed to determine the minimum number of volume nodes required to conservatively predict the maximum pressure load acting on the compartment structure. The nodalization sensitivity study should include consideration of spatial pressure variation circumferentially, axially and radially within the compartment. Describe and justify the nodalization sensitivity study performed for the major component supports evaluation, where transient forces and moments acting on the components are of concern.
- (e) For the compartment structural design pressure evaluation, provide the peak calculated differential pressure and time of peak pressure for each node. Discuss whether the design differential pressure is uniformly applied to the compartment structure or whether it is spatially varied. If the design differential pressure varies depending on the proximity of the pipe break location, discuss how the vent areas and flow coefficients were determined to assure that regions removed from the break location are conservatively designed.

(f) Provide the peak and transient loading on the major components used to establish the adequacy of the supports design. This should include the load forcing functions (e.g.,  $f_x(t)$ ,  $f_y(t)$ ,  $f_z(t)$ ) and transient moments (e.g.,  $M_x(t)$ ,  $M_y(t)$ ,  $M_z(t)$ ) as resolved about a specific, identified coordinate system. Provide the projected area used to calculate these loads and identify the location of the area projections on plan and section drawings in the selected coordinate system. This information should be presented in such a manner that confirmatory evaluations of the loads and moments can be made.

022.3 The Engineered Safety Features Actuation System (ESFAS) provides the  
(6.2) signal for containment isolation, which only occurs on high containment pressure. We require diversity in the parameters sensed for the initiation of containment isolation to provide greater assurance that lines which may be open to the environs, such as the purge system supply and exhaust lines, are isolated over a complete spectrum of postulated pipe breaks. Therefore, discuss your plans for including other signals (e.g., safety injection and high radiation signals) to initiate containment isolation.



022.4 Identify any leakage paths which could bypass the volumes treated  
(6.2) by the Shield Building Ventilation System following a design basis loss-of-coolant accident. Consider isolation valve leakage and leakage through guard pipe welds. Indicate where lines which could be open to containment atmosphere following a LOCA terminate. List the specific leakage paths identified and the Technical Specification commitment you are able to meet for each path. Provide the total leakage specification for leakage to untreated areas. This Technical Specification must be met assuming a single active failure. Additional guidance may be found in Branch Technical Position CSB 6-3, "Determination of Bypass Leakage Paths in Dual Containment Paths."

022.5 Identify the containment isolation arrangements which do not comply  
(6.2) with the explicit requirements of General Design Criteria 55, 56 and 57, and discuss the rationale for concluding that the isolation arrangements are acceptable on some other defined basis.

022.6 The containment sump design does not comply with the recommendations  
(6.2) of Regulatory Guide 1.82. Provide justification for deviating from the recommendations of Regulatory Guide 1.82.

022.7 Following an inadvertent actuation of the containment spray systems,  
(6.2) vacuum breakers will draw on the annulus volume to relieve any external differential pressure buildup on the primary shield building. However, this will create an external pressure buildup on the shield building. Therefore, state the external design pressure of the shield building. Discuss the maximum external pressure that would act on the shield building.

022.8 It is our position that the heating & ventilating containment purge  
(6.2) system (48" lines) should meet the recommendations of Branch Technical Position CSB 6-4, "Containment Purging During Normal Plant Operation." Therefore, propose a purge system that complies with BTP 6-4, and provide the information and analyses identified in the Branch Technical Position.

022.9 Provide the following information regarding the containment leak  
(6.2) testing program:

- a) Identify those fluid lines penetrating the containment which will be vented and drained to ensure exposure of the system containment isolation valves to the containment atmosphere and the full differential pressure during the containment integrated leakage rate (Type A) test. Those systems that will remain fluid filled for the Type A test should be identified and justification given.

- b) Appendix J requires that containment piping penetrations fitted with expansion bellows be tested at Pa. Identify any penetration fitted with expansion bellows that does not have the design capability for Type B testing and provide justification. Where more than one bellows is utilized on a penetration, provide assurance that each bellows will be subjected to Type B testing.
- c) Provide plan and elevation drawings of the personnel air lock, and identify all mechanical and electrical penetrations. Discuss and schematically show the design provisions that will permit the personnel air-lock door seals and the entire air lock to be tested.

Discuss the design capability of the door seals to be leak tested at a pressure of Pa; i.e., the calculated peak containment internal pressure. If it will be necessary to exert a force on the doors to prevent them from being unseated during leak testing, describe the provisions for doing this.

- d) It is our position that all isolation valves provided to satisfy General Design Criteria 54 through 57 (containment isolation valves) should be pneumatically (Type C) leak tested. Alternatively, a containment isolation valve may be exempted from the Type C test requirements if it can be shown that the valve does not constitute a potential containment atmosphere leak path following a loss of coolant accident.

Table 6.2-43 identifies that containment isolation valves that will not be Type C tested. Therefore, justify that they do not constitute potential containment atmosphere leak paths following a LOCA. In this regard, a water seal may be shown to exist that will preclude containment atmosphere leakage. If this approach is taken, discuss how a water seal can be established and maintained using safety grade pipes and components, and considering single failure of active components. System drawings showing the routing and elevation of piping may be used to show the existence of a water seal.

- e) Identify all containment isolation valves for which the applied test pressure is not in the same direction as the pressure existing when the valve is required to perform its safety function, and provide evidence to show acceptability of testing the valve with pressure applied in the reverse direction.
- f) For each fluid line that penetrates the containment schematically, show the isolation valve arrangement and the design provisions that will permit the isolation valves to be leak tested. Indicate the direction in which the valves will be leak tested.

Show the test, vent and drain (TVD) connections that will be installed to facilitate the performance of the Type A and Type C leak tests required by Appendix J to 10 CFR Part 50. Discuss the

isolation provisions for the TVD connections, including the administrative controls that will be exercised to assure proper isolation.

- 022.10 The following information is required describing the component  
(6.2) thermal analyses performed as part of the environmental qualification. Each component that must function during an MSLB should be addressed explicitly.
- a. Provide external and sectional diagrams of each component analyzed showing principle dimensions, materials of construction, and cross sections modeled for analysis.
  - b. Provide a detailed description of each thermal model indicating basic assumptions and showing the model mock up with principle dimensions, materials, and material thermal properties.
  - c. Perform the analysis using the correlation provided in the attached CSB Interim Evaluation Model.
  - d. Identify the specific point on the component which was analyzed and justify that this location is the most critical or conservative with regard to potential component failure.

CSB Interim Evaluation Model  
Environmental Qualification for Main  
Steam Line Break Inside Containment  
(Operating License Applicants Only)

Analyses of main steam line break (MSLB) accidents inside PWR dry-type containments have predicted temperature transients which exceed the qualification temperature of some safety related equipment. As a result there is a concern regarding the capability of this equipment to survive such an event to assure safe plant shutdown. This concern is related to Issue 25 of NUREG-0153 dated September, 1976.

The NRC has identified this matter as a Category A Technical Safety Activity and is currently pursuing a program to resolve this concern. In the meantime it is required that you perform an evaluation of the containment environmental conditions associated with a MSLB accident as well as a LOCA and justify that the essential equipment needed to mitigate these accidents have been adequately qualified.

Since the NRC generic effort on this concern is still in progress, we are providing the analytical assumptions which are acceptable for the interim period. These models and assumptions are acceptable for the spectrum of MSLB accidents.

1. Containment Environmental Response

a. Heat transfer coefficient to heat sinks.

The Uchida heat transfer correlation (data) should be used while in the condensing mode. A natural convection heat transfer

coefficient should be used at all other times. The application of these correlations should be as follows:

(1) Condensing heat transfer

$$q/A = h_u \cdot (T_s - T_w)$$

where  $q/A$  = the surface heat flux

$h_u$  = the Uchida heat transfer coefficient

$T_s$  = the steam saturation (dew point) temperature

$T_w$  = surface temperature of the heat sink

(2) Convective heat transfer

$$q/A = h_c \cdot (T_v - T_w)$$

where  $h_c$  = convective heat transfer coefficient

$T_v$  = the bulk vapor temperature.

All other parameters are the same as for the condensing mode.

b. Heat sink condensate treatment

When the containment atmosphere is at or below the saturation temperature, all condensate formed on the heat sinks should be transferred directly to the sump. When the atmosphere is superheated a maximum of 8% of the condensate may be transferred

to the vapor region. The revaporization should be calculated as follows:

$$M_r = X \cdot q / (h_v \cdot h_L)$$

where  $M_r$  = revaporization rate

$X$  = revaporization fraction (0.08)

$q$  = surface heat transfer rate

$h_v$  = enthalpy of the superheated steam

$h_L$  = enthalpy of the liquid condensate entering the sump region (i.e., average enthalpy of the heat sink condensate boundary layer)

c. Heat sink surface area

The surface area of the heat sinks should correspond to that used for the containment design pressure evaluation.

d. Single active failure evaluation

Single active failures should be evaluated for those containment safety systems and components relied upon to limit the containment temperature/pressure response to a MSLB accident. This evaluation



should include, but not necessarily be limited to, the loss or availability of offsite power (whichever is worse), diesel generator failure when loss of offsite power is evaluated, and loss of containment heat removal systems (either partial or total).

e. Containment heat removal system actuation

The time determined at which active containment heat removal systems become effective should include consideration of actuation sensors and setpoints, activation delay time, and system delay time (i.e., time required to come into operation).

f. Identification of most severe environment

The worst case for environmental qualification should be selected considering time duration at elevated temperatures as well as the maximum temperature. In particular, consider the spectrum of break sizes analyzed and single failures evaluated.

2. Safety Related Component Thermal Analysis

Component thermal analyses may be performed to justify environmental qualification test conditions less than those calculated during the containment environmental response calculation. The thermal analysis should be performed for the potential points of component failure such as thin cross sections and temperature sensitive parts where thermal stressing, temperature-related degradation, steam or chemical interaction at elevated temperatures, or other thermal effects could

result in failure of the compartment electrically or mechanically.  
The heat transfer rate to components should be calculated as follows:

a. Condensing heat transfer rate

$$q/A = h_{cd} \cdot (T_s - T_w)$$

where  $q/A$  = component surface heat flux

$h_{cd}$  = condensing heat transfer coefficient  
= the larger of 4x Tagami Correlation or 4x  
Uchida Correlation

$T_s$  = saturation temperature (dew point)

$T_w$  = component surface temperature

b. Convective heat transfer

A convective heat transfer coefficient should be used when the condensing heat flux is calculated to be less than the convective heat flux. During the blowdown period, a forced convection heat transfer correlation should be used. For example:

$$NU = C (Re)^n$$

where  $Nu$  = Nusselt No.

$Re$  = Reynolds No.

$C, n$  = empirical constants dependent on geometry  
and Reynolds No.

The velocity used in the evaluation of Reynolds number may be determined as follows:

$$V = 25 \frac{M_{BD}}{V_{CONT}}$$

where V = velocity in ft/sec

$M_{BD}$  = the blowdown rate in lbm/hr

$V_{CONT}$  = containment volume in ft<sup>3</sup>

After the blowdown has ceased or reduced to a negligibly low value, a natural convection heat transfer correlation is acceptable. However, use of a natural convection heat transfer coefficient must be fully justified whenever used.

### 3. Evaluation of Environmental Qualification

The component peak surface temperature(s) ( $T_{CS}$ ) should be computed using items 1 and 2 above. The component qualification temperature ( $T_{CQ}$ ) should be determined from the actual environment test conditions. Where components have been "bathed" in a saturated steam or steam/air environment for extended periods (e.g., 10 minutes), the qualification temperature is the test chamber temperature. For components subjected to test conditions substantially removed from the steam saturation point or for short durations (e.g., less than 10 minutes), the qualification temperature must be justified by experimental thermocouple readings on the component surface or analyses which minimizes the heat flux to the component.

If the component surface temperature,  $T_{CS}$ , is less than or equal to the component qualification temperature,  $T_{CQ}$ , the component may be considered qualified for an MSLB environment during the interim period.

If the component surface temperature is greater than the qualification temperature, then (a) provide additional justification that the component can operate in environments equal to or greater than that \_\_\_\_\_ which would result in the calculated peak surface temperature, or (b) provide a requalification package for the component, or (c) provide appropriate protection to assure that the component will not experience a surface temperature in excess of the qualification temperature,  $T_{CQ}$ .

32.1 Environmental Qualification of Class 1E Equipment  
(3.11)

In order to ensure that your environmental qualification program conforms with General Design Criteria 1, 2, 4 and 23 of Appendix A and Sections III and XI of Appendix B to 10 CFR Part 50, and to the national standards referenced in Part II "Acceptance Criteria" (which includes IEEE Std 318) contained in Standard Review Plan Section 3.11, the following information on the qualification program is required for all Class 1E equipment.

- (1) Identify all Class 1E Equipment, and provide the following:
  - a. Type (functional designation)
  - b. Manufacturer
  - c. Manufacturer's type number and model number
  - d. The equipment should include the following, as applicable:
    - 1) Switchgear
    - 2) Motor control centers
    - 3) Valve operators
    - 4) Motors
    - 5) Logic equipment
    - 6) Cable
    - 7) Diesel generator control equipment
    - 8) Sensors (pressure, pressure differential, temperature and neutron)
    - 9) Limit Switches
    - 10) Heaters
    - 11) Fans
    - 12) Control Boards
    - 13) Instrument racks and panels
    - 14) Connectors
    - 15) Electrical penetrations
    - 16) Splices
    - 17) Terminal blocks

(c) Categorize the equipment identified in (1) above into one of the following categories:

- a. Equipment that will experience the environmental conditions of design basis accidents for which it must function to mitigate said accidents, and that will be qualified to demonstrate operability in the accident environment for the time required for accident mitigation with safety margin to failure.
- b. Equipment that will experience environmental conditions of design basis accidents through which it need not function for mitigation of said accidents, but through which it must not fail in a manner detrimental to plant safety or accident mitigation, and that will be qualified to demonstrate the capability to withstand any accident environment for the time during which it must not fail with safety margin to failure.
- c. Equipment that will experience environmental conditions of design basis accidents through which it need not function for mitigation of said accidents, and whose failure (in any mode) is deemed not detrimental to plant safety or accident mitigation, and need not be qualified for any accident environment, but will be qualified for its non-accident service environment.
- d. Equipment that will not experience environmental conditions of design basis accidents and that will be qualified to

demonstrate operability under its normal or abnormal service environment. This equipment would normally be located outside the reactor containment.

- (3) For each type of equipment in the categories of equipment listed in (2) above provide separately the equipment design specification requirements, including:
  - a. The system safety function requirements.
  - b. An environmental envelope as a function of time which includes all extreme parameters, both maximum and minimum values, expected to occur during plant shutdown, normal operation, abnormal operation, and any design basis event (including LOCA and MSLB), including post event conditions.
  - c. Time required to fulfill its safety function when subjected to any of the extremes of the environmental envelope specified above.
  - d. Technical bases should be provided to justify the placement of each type equipment in the categories 2.b and 2.c listed above.
- (4) Provide the qualification test plan, test set-up, test procedures, and acceptance criteria for at least one of each group of equipment

of (1.d) as appropriate to the category identified in (2) above. If any method other than type testing was used for qualification (operating experience, analysis, combined qualification, or on-going qualification), describe the method in sufficient detail to permit evaluation of its adequacy.

- (5) For each category of equipment identified in (2) above, state the actual qualification envelope simulated during testing (defining the duration of the hostile environment and the margin in excess of the design requirements). If any method other than type testing was used for qualification, identify the method and define the equivalent "qualification envelope" so derived.
- (6) A summary of test results that demonstrates the adequacy of the qualification program. If analysis is used for qualification, justification of all analysis assumptions must be provided.
- (7) Identification of the qualification documents which contain detailed supporting information, including test data, for items 4, 5 and 6.

In addition, in accordance with the requirements of Appendix B of 10 CFR 50, the staff requires a statement verifying: 1) that all Class 1E equipment has been qualified to the program described above, and 2) that the detailed qualification information and test results are available for an NRC audit.



32.2  
(3.11)

Table 3.11-1 "Environmental Design Categories" lists the temperature profile inside containment under a LOCA accident condition as 368.7°F. There is no information on the temperature profile inside containment under a Main Steam Line Break (MSLB) accident condition. Provide the Maximum temperature profile under a MSLB accident. We also require that the environmental qualification program be modified to ensure that Class 1E equipment located within containment and needed to mitigate the effects of a MSLB accident are qualified to the more extreme (temperature) environment produced by that accident. Otherwise, justify why your qualification program is adequate.

32.3  
(3.10)

Describe the conformance to the guidance of Regulatory Guide 1.100 "Seismic qualification of Electric Equipment for Nuclear Power Plants" as it related to Class 1E equipment described in FSAR Section 3.10. Justify any exception taken.

32.4  
(3.11)

Describe the qualification program for meeting the aging requirements of IEEE Std. 323-1974, for Class 1E equipment used in the plant.

32.5  
(3.11)

Table 3.11-1, "Environmental Design Categories", lists that the reactor auxiliary building normal temperature is 80-104°F, while under accident conditions, the temperature will be 80-120°F. In Table 3.11-2, "Environment Design Parameters for Equipment Required to Function During and Subsequent to Any Design Basis Accident," some of the safety equipment are stated to be qualified to 104°F only. Modify your environmental qualification program to meet the higher temperature requirement, or justify why your qualification profile is adequate.

32.6  
71  
71  
In general, the information provided in the FSAR concerning the the applicability of safety system design criteria is inadequate. Provide the information required by sections 7.2.2, 7.3.2, 7.4.2, and 7.6.2 of Regulatory guide 1.70, "Standard format and Content of Safety Analysis Report for Nuclear Power Plants." In addition, revise FSAR Section 7.1.2.7, "Comparison of Design with NRC Regulatory Guides," to provide the information required in Section 7.1.2 of R.G. 1.70. Please include the Regulatory Guides issued after 1.89 that are applicable, such as R.G. 105 "Instrument Setpoints", R.G. 118 "Periodic Testing of Electric Power and Protection System", R.G. 100 "Seismic Qualification of Electric Equipment." These are the Category II Regulatory Guide for which conformance needs to be evaluated for older plants and plants presently under review.

32.7  
(7.2)  
The computer based Core Protection Calculator System (CPCS) as a part of the plant protection system has been reviewed under the ANO-2 docket. You have stated that the number of Control Element Assemblies (CEAS) is different for Waterford Unit 3 in comparison with ANO-2. Define and document all details of design, qualification, and criteria differences for CPCS for this plant in comparison with ANO-2.

32.8  
(6.3)  
(7.3)  
Describe the automatic switchover features with adequate logic diagrams which are provided to assure the proper operation of the LPSI pump minimum flow line isolation valves and the SIS sump isolation valves under the "short term recirculation" mode as stated in FSAR Section 6.3.2.9.4.

32.9 (10.3) (7.3) The FSAR has not provided adequate information to assure that the design of the initiation, actuation and control portions of the Main Steam Isolation System will perform their functions assuming any single failure in the instrumentation and control system following a steam line Break Accident (MSLB). For instance, no information is available in the FSAR to define the signal that would assure the closure of turbine stop valves or any other steam valves for an MSLB, should the MSIV fail to close. Provide additional information to address how the instrumentation and control portions of the main steam isolation system conform to the requirements in IEEE Std. 279-1971.

32.10 (10.4) (7.3) The FSAR has not provided adequate information how the ESFAS signal is implemented on the Main Feedwater Isolation System. Provide additional information to address how the instrumentation and control portions of the main Feedwater isolation system conform to the requirements in IEEE Std. 279-1971.

32.11 (7.3) Provide the results of an analysis which show that no adverse effects will occur as a result of loss of offsite power to the engineered safety features actuation system at any time following the onset of a LOCA or other accident condition in the Waterford Unit 3.

32.12 (7.3) (10.4) There are several errors on Fig. 10.4-6, "Emergency Feedwater System":  
(a) The middle pump is not a motor driven pump.  
(b) The power supplies for turbine trip valve, turbine governor control system, and overspeed device are 125 Vdc.  
Please correct these errors.

32.13  
(7.3)

It is stated in FSAR Section 7.3.1.2.1 "Design Basis Information for BAF System Equipment" Item 9(a) that system response times are given in Table 7.3-2. However, there is no system response time information in Table 7.3-2. Please provide this information.

32.14  
(7.4)

It is stated in FSAR Section 7.4.1.6, "Emergency Shutdown from outside the Control Room", that the transfer switches are provided for transferring control functions from the control room to the auxiliary control panel (LCP -43). Since this panel contains some of redundant safety related circuitries, the concern is that a postulated exposure fire at auxiliary panel might damage the control function in the main control room. Provide the detailed arrangement of this transfer scheme and address the fire protection aspects of the auxiliary control panel. Please use Regulatory Guide 1.120 "Fire Protection Guidelines for Nuclear Power Plants" as a guide.

32.15  
(7.5)

It is stated in FSAR Section 7.5.1.8 "Bypass and Inoperable Status Indication" that the bypass and inoperable status system is actuated through the plant computer. However, in Section 7.7.1.6 you also stated that none of the plant computer functions are required to ensure plant safety or permit safe plant operation. Justify the use of plant computer to actuate the bypass and inoperable status system and address how this system is in conformance with Regulatory Guide 1.47.

- 32.15  
(7.5) Verify that Bypassed and Inoperable Status Indication panel (Table 7.5-2) should cover all systems you identified in Section 7.1.1 "Systems Required for safe shutdown and ESF support Systems." Justify not including safety systems such as: Containment Cooling System; Combustible Gas Control System; Atmospheric Steam Dump System; Diesel Fuel Oil Storage & Transfer System; and HVAC systems for Safety related equipment areas in Table 7.5-2.
- 32.17  
(7.5) Figs. 7.5-2, 7.5-3, 7.5-4, and 7.5-5 indicate that these instrument cabinets are located inside containment. Provide the cross reference of your environmental qualification document to assure these instrument cabinets are qualified to appropriate worst case environments inside containment.
- 32.18  
(7.5)  
(6.3) Fig. 6.3-9 "Long Term Cooling Plan After LOCA" presents the operator procedures to be used after a LOCA. Correlate this figure with your Post Accident Monitoring Instrumentation (PAMI). Verify that the PAMI provides adequate information to support Fig. 6.3-9's requirement. Also provide cross reference with your environmental qualification document to assure that the PAMI System is qualified to appropriate worst case environments inside containment.
- 32.19  
(7.6) You have not addressed the instrumentation and control of the Reactor Coolant System overpressurization protection during startup and shutdown for Waterford Unit 3. Provide your design information in accordance with Standard Review Plan Section 5.2.2. (NUREG - 75/087)

040.00 Power Systems Branch

040.01 (8.1) Provide a listing of the following for the containment electrical penetrations by voltage class:  $I^2t$  ratings, maximum predicted fault currents, identification of maximizing faults, protective equipment setpoints, and expected clearing times.

040.02 Provide a more detailed description of the separation afforded the third-of-a-kind ESF loads. Include the power, control, and instrumentation circuits and the mechanisms by which the transfer are accomplished.

040.03 (8.3) (RSP) In addition to the undervoltage scheme currently provided to detect loss of offsite power at the safety busses, we require that a second level of voltage protection be provided with a time delay in order to protect the onsite power system from any adverse effects that could result from a sustained degraded voltage condition on the off-site power system. This second level of voltage protection shall satisfy the following criteria:

- a) The selection of voltage and time set points shall be determined from an analysis of the voltage requirements of the safety-related loads at all onsite system distribution levels;
- b) The voltage protection shall include coincidence logic to preclude spurious trips of the offsite power source;

- c) The time delay selected shall be based on the following conditions:
  - (1) The allowable time delay, including margin, shall not exceed the maximum time delay that is assumed in the FSAR accident analyses;
  - (2) The time delay shall minimize the effect of short duration disturbances from reducing the availability of the offsite power source(s); and
  - (3) The allowable time duration of a degraded voltage condition at all distribution system levels shall not result in failure of safety systems or components;
- d) The voltage sensors shall automatically initiate the disconnection of offsite power sources whenever the voltage set point and time delay limits have been exceeded;
- e) The voltage sensors shall be designed to satisfy the applicable requirements of IEEE Std. 279-1971, "Criteria for Protection Systems for Nuclear Power Generating Stations"; and
- f) The Technical Specifications shall include limiting condition for operation, surveillance requirements, trip set points with minimum and maximum limits, and allowable values for the second level voltage protection sensors and associated time delay devices.

NOTE:

The following is a definition of what is meant by the term "applicable requirements" found in item 040.03(e). This relates the criteria of IEEE Std 279-1971 to the degraded grid voltage sensors.

- 1) Class 1E equipment shall be utilized and shall be physically located at and electrically connected to the emergency switch-gear.
- 2) An independent scheme shall be provided for each division of emergency power.
- 3) Capability for test and calibration during power operation shall be provided.
- 4) Annunciation must be provided in the control room for any by-passes incorporated into the design.



Provide the details of your design that meets the above position.

040.04  
8.3  
(RSP)

We require that the diesel generator bus load shedding design automatically prevent load shedding of the emergency bus once the diesel generator is supplying power to the emergency bus. The design shall also include the capability of the load shedding feature to be automatically reinstated if the diesel generator supply breaker is tripped. The automatic bypass and reinstatement feature shall be verified during the periodic testing required by Item 040.05. State your intent to comply with this position and provide justification for any exceptions taken.

In the event an adequate basis can be provided for retaining the load shed feature when loads are energized by the onsite power system, we will require that the setpoint value in Technical Specifications, which is currently specified as "...equal to or greater than..." be amended to specify a value having maximum and minimum limits. The bases for the setpoints and limits selected must also be documented.

040.05  
8.3  
(RSP)

We will require that the Technical Specifications include a test requirement to demonstrate the full functional operability and independence of the onsite power sources at least once per 18 months during shutdown. The Technical Specifications shall include a requirement for tests: (1) simulating loss of offsite power; (2) simulating loss of offsite power in conjunction with a safety feature actuation signal; and (3) simulating interruption and subsequent reconnection of onsite power sources to their respective buses. Proper operation shall be determined by:

- a) Verifying that on loss of offsite power the emergency buses have been de-energized and that the loads have been shed from the emergency buses in accordance with design requirements.
- b) Verifying that on loss of offsite power the diesel generators start on the autostart signal, the emergency buses are energized with permanently connected loads, the auto-connected shutdown loads are energized through the load sequencer, and the system operates for five minutes while the generators are loaded with the shutdown loads.
- c) Verifying that on safety features actuation signal (without loss of offsite power) the diesel generators start on the autostart signal and operate on standby for five minutes.
- d) Verifying that on loss of offsite power in conjunction with a safety features actuation signal the diesel generators start on the autostart signal, the emergency buses are energized with permanently connected loads, the auto-connected emergency (accident) loads are energized through the load sequencer, and the system operates for five minutes while the generators are loaded with the emergency loads.
- e) Verifying that on interruption of the onsite sources the loads are shed from the emergency buses in accordance with design requirements and that subsequent loading of the onsite sources is through the load sequencer.

040.06  
8.3  
(RSP)

The voltage levels at the safety-related buses should be optimized for the full load and minimum load conditions that are expected throughout the anticipated range of voltage variations of the offsite power source by appropriate adjustment of the voltage tap settings of the intervening transformers. We require that the adequacy of the design in this regard be verified by actual measurement, and by correlation of measured values with analytic results. Provide a description of the method for making this verification; before initial reactor power operation, provide the documentation required to establish that this verification has been accomplished.

040.07  
8.3  
(RSP)

It appears that the 6.9 Kv circuit containment penetrations are vulnerable to a single failure of DC control power which would prevent the clearing of an electrical fault by both the primary and back-up protection schemes. If this is the case, it is the staff's position that either the penetrations be demonstrated to be able to sustain the maximum fault current for the time necessary to clear the fault by some other means or the single failure identified above must be eliminated from the design.

040.08  
8.3

Provide a description of the physical arrangement utilized in your design to connect the field cables inside containment to the containment penetrations, e.g. connectors, splices, or terminal blocks. Provide supportive documentation that these physical interfaces are qualified to withstand a LOCA or steam line break environment.

040.09  
8.3 Provide a listing of all motor operated valves within your design that require power lock out in order to meet the single failure criterion and provide the details of your design that accomplish this requirement. Your response should also address your conformance to Branch Technical Position ICSB 18 (PSB) in Appendix 8A of the Standard Review Plan.

040.10  
8.3 Provide the details of your design of the DC power system that assures equipment will be protected from damaging overvoltages from the battery chargers that may occur due to faulty regulation or operator error.

040.11  
8.2  
8.3 Provide the results of a review of your operating, maintenance, and testing procedures to determine the extent of usage of jumpers or other temporary forms of bypassing functions for operating, testing, or maintaining of safety related systems. Identify and justify any cases where the use of the above methods cannot be avoided. Provide the criteria for any use of jumpers for testing.

040.12  
8.2  
8.3 We request that you perform a review of the electrical control circuits for all safety related equipment, so as to assure that disabling of one component does not, through incorporation in other inter locking or sequencing controls, render other components inoperable. All modes of test, operation, and failure should be considered. Describe and state the results of your review.

040.13 State the nominal value and the maximum and minimum spread values of voltage and frequency of the offsite power source that assure satisfactory operability of all electrical equipment of the station during all modes of operation.

Provide a discussion as to how the above envelopes of frequency and voltage compare with the expected envelopes of the offsite power source.

040.14 Your discussion of diesel generator prototype qualification is limited to a reference to the Cooper qualification program. Sufficient information is required to permit an independent evaluation of the appropriateness of the Cooper reference. Provide a comparison of the Cooper and Waterford diesel generators and include such items as power rating, voltage, and  $WR^2$ .

040.15 What is your criteria for determining which 120 volt ac control circuits require back-up circuit protection for assuring containment integrity? Are there any such circuits in your design? If so, identify same.

040.16 The frequency bounds of  $\pm 10\%$  for the diesel generators for applying and dropping load exceed the recommendations of Regulatory Guide 1.9. Further justification is required to support this aspect of your design.

040.17  
(8.3)  
(RSP)

In the discussion of conformance with Regulatory Guide 1.75 (Section 8.3.1.2.13), the Waterford Unit 3 criteria for cable and raceway identification is provided. One criterion requires that only the ends of a cable be identified and with requirement for identifying intermediate points. This criterion is not justified by the supporting bases and is unacceptable. This is also the case for raceway identification.

We require that the Waterford Unit 3 cable and raceway identification criteria be modified to include the requirement for suitable identification at fixed intervals over the entire length of the cable or raceway. The basis for this requirement is to facilitate initial verification that the installation is in conformance with the separation criteria. These markings may be applied during installation if equipment has been purchased without same.

Regulatory Guide 1.75 provides an acceptable method of meeting our requirements for identification. Other equivalent identification schemes are also acceptable. Provide modified cable and raceway identification criteria in conformance with this position.

040.18  
(9.5.2)

The information regarding the onsite communications system (Section 9.5.2) does not adequately cover the system capabilities during transients and accidents. Provide the following information:

- 7.
- (a) Identify all working stations on the plant site where it may be necessary for plant personnel to communicate with the control room or the emergency shutdown panel during and/or following transients and/or accidents (including fires) in order to mitigate the consequences of the event and to attain a safe cold plant shutdown.
  - (b) Indicate the maximum sound levels that could exist at each of the above identified working stations for all transients and accident conditions.
  - (c) Indicate the types of communication systems available at each of the above identified working stations.
  - (d) Indicate the maximum background noise level that could exist at each working station and yet reliably expect effective communication with the control room using:
    - 1. the page party communications systems, and
    - 2. any other additional communication system provided that working station.

- (e) Describe the performance requirements and tests that the above onsite working stations communication systems will be required to pass in order to be assured that effective communication with the control room or emergency shutdown panel is possible under all conditions.
- (f) Identify and describe the power source(s) provided for each of the communications systems.
- (g) Discuss the protective measures taken to assure a functionally operable onsite communication system. The discussion should include the considerations given to component failures, loss of power, and the severing of a communication line or trunk as a result of an accident or fire.

040.19  
(9.5.3)

Identify the vital areas and hazardous areas where emergency lighting is needed for safe shutdown of the reactor and the evacuation of personnel in the event of an accident (including fire). Tabulate the lighting systems provided in your design to accommodate those areas so identified.

040.20  
(9.5.4)

Section 9.5.4.1 emergency diesel engine fuel oil storage and transfer system (EDEFSS) does not reference ANSI Standard N195 "Fuel Oil Systems for Standby Diesel Generators". Indicate if you intend to comply with this standard in your design of the EDEFSS; otherwise provide justification for non-compliance. (SRP 9.5.4, Rev. 1, Part II, Item 12).



040.21  
(9.5.4)

In Section 9.5.4.2 you state diesel fuel oil delivery to the site in by truck. In the event of an accident it may be necessary to operate a diesel generator continuously for a period of 30 days or more. Under this condition discuss your plans to provide fuel oil to maintain required onsite inventory. In your discussion include sources where diesel quality fuel oil will be available and the distances required to be travelled from the source to the plant. Also discuss how fuel oil will be delivered onsite under extremely unfavorable environmental conditions.

040.22  
(9.5.4)

Figure 9.5-3 shows diesel fuel oil feed tank vent lines terminating outdoor with vent screen covers. Provide your justification why these lines are not provided with flame arrestors

040.23  
(9.5.4)

Figure 9.5-3 shows vent and fill lines for the diesel oil storage tanks. Indicate where these lines are located. Discuss how these lines are protected from tornado missiles, and also from entrance of water into the storage tank during adverse environmental conditions. Also justify your use of non service Class 3 materials.

040.24  
(9.5.4)

Figure 9.5-3 shows two items on the fuel oil storage tanks labeled 7EG3-19 and 7EG3-20. Identify and describe the purpose of these items and justify your use of non service Class 3 materials.

040.25  
(9.5.4)

Figure 9.5-3 shows non service class 3 piping from the outlet of valves 3EG-V611A and 3EG-V630A. It is assumed that this non service class 3 materials does not apply to line 3EG1-13A including valve 3EG-V626. Clarify this diagram to indicate service class 3 materials for this line.

040.26  
(9.5.4)

Assume an unlikely event has occurred requiring operation of a diesel generator for a prolonged period that would require replenishment of fuel oil without interrupting operation of the diesel generator. What provision has been made in the design of the fuel oil storage fill system to minimize the creation of turbulence of the sediment in the bottom of the storage tank. Stirring of this sediment during addition of new fuel has the potential of causing the overall quality of the fuel to become unacceptable and could potentially lead to the degradation or failure of the diesel generator.

040.27  
(9.5.4)

Discuss the precautionary measures that will be taken to assure the quality of and reliability of the fuel oil supply for emergency diesel generator operation. Include the fuel oil impurity and quality limitations as well as diesel index number or its equivalent, entrained moisture, sulfur, particulates and other deliterious substances, periodic inspection, and periodic testing (including interval between tests) of fuel oil. In your discussion include reference to industry (or other) standards which will be followed to assure a reliable fuel oil supply to the emergency generator. (SRP 9.5.4, Part III, Items 3 and 4.)

- 040.28  
(9.5.4) Discuss what precautions have been taken in locating the fuel oil day tank and connecting fuel oil piping with regard to possible exposure to ignition sources such as open flames and hot surfaces. (SRP 9.5.4, Part III, Item 6.)
- 040.29  
(9.5.5) Provide a tabulation showing the individual and total heat removal rates for each major component and subsystem of the diesel generator cooling water system. Discuss the design margin (excess heat removal capability) included in the design of major components and subsystems. (SRP 9.5.5, Part III, Item 1.)
- 040.30  
(9.5.5) Describe the provisions made in the design of the diesel engine cooling water system to assure that all components and piping are filled with water. (SRP 9.5.5, Part III, Item 2.)
- 040.31  
(9.5.5) Indicate the measures to preclude long-term corrosion and organic fouling in the diesel engine cooling water system that would degrade system cooling performance, and the compatibility of any corrosion inhibitors or antifreeze compounds used with the materials of the system. Indicate if the water chemistry is in conformance with the engine manufacturers recommendations. (SRP 9.5.5, Part III, Item 1c.)
- 040.32  
(9.5.5) The diesel engine generator sets should be capable of operation at less than full load for extended periods without degradation of performance or reliability. Provide a discussion of your diesel engine operating parameters, including minimum load requirements, and relate this to anticipated minimum loads under accident recovery conditions and during accident standby operation when offsite power is available. (SRP 9.5.5, Part III, Item 7.)

040.33  
(9.5.6)

Provide a discussion of the measures taken in the design of the standby diesel generator air starting system to preclude the fouling of the starting air valve (or other control valves) or filter with contaminants such as oil carry over and rust. (SRP 9.5.6, Part III, Item 1).

040.34  
(9.5.7)

For the diesel engine lubrication system in Section 9.5.7 provide the following information: 1) define the temperature differentials, flow rate, and heat removal rate of the interface cooling system external to the engine and verify that these are in accordance with recommendations of the engine manufacturer; 2) discuss the measures that will be taken to maintain the required quality of the oil, including the inspection and replacement when oil quality is degraded; 3) describe the protective features (such as blowout panels) provided to prevent unacceptable crankcase explosion and to mitigate the consequences of such an event; and 4) describe the capability for detection and control of system leakage. (SRP 9.5.7, Part II, Items 8a, 8b, 8c, Part III, Item 1.)

040.35  
(9.5.7)

What measures have been taken to prevent entry of deleterious materials into the engine lubrication oil system due to operator error during recharging of lubricating oil or normal operation. (SRP 9.5.7, Part III, Item 1c.)

040.36  
(9.5.8)

Describe the instrumentation, controls, sensors and alarms provided in the design of the diesel engine combustion air intake and exhaust system to warn the operators when design parameters are exceeded. (SRP 9.5.8, Part III, Items 1 and 4).

040.37  
(10.1)

Provide the criteria and bases for the various steam and condensate instrumentation systems. The FSAR should differentiate between operating and required safety instrumentation.

040.38  
(10.2)

Expand your discussion of the turbine speed control and overspeed protection system. Provide additional explanation of the turbine and generator electrical load following capability for the turbine speed control system with the aid of system schematics (including turbine control and extraction steam valves to the heaters). Tabulate the individual speed control protection devices (normal, emergency and backup), the design speed (or range of speed) at which each device begins operation to performs its protective function (in terms of percent of normal turbine operating speed). In order to evaluate the adequacy of the control and overspeed protection system provide schematics and include identifying numbers to valves and mechanisms (mechanical and electrical) on the schematics. Describe in detail, with references to the identifying numbers, the sequence of events in a turbine trip including response times, and show that the turbine stabilizes. Provide the results of a failure mode and

effects analysis for the overspeed protection systems. Show that a single steam valve failure cannot disable the turbine overspeed trip from functioning. (SRP 10.2, Part III, items 1, 2, 3 and 4).

040.39  
(10.2)

Describe with the aid of drawings, the bulk hydrogen storage facility including its location and distribution system. Include the protective measures considered in the design to prevent fires and explosions during operations such as filling and purging the generator, as well as during normal operations.

040.40  
(10.2)

Discuss the effects of a high and moderate energy piping failure or failure of the connection from the low pressure turbine to condenser on nearby safety related equipment or systems. Discuss what protection will be provided the turbine overspeed control system equipment, electrical wiring and hydraulic lines from the effects of a high or moderate energy pipe failure so that the turbine overspeed protection system will not be damaged to preclude its safety function. (SRP 10.2, Part III, Item 8, SRP 10.4.1, Part III, Item 3a.)

- 040.41  
(10.2) In the turbine generator section discuss: 1) the valve closure times and the arrangement for the main steam stop and control and the reheat stop and intercept valves in relation to the effect of a failure of a single valve on the overspeed control functions; 2) the valve closure times and extraction steam valve arrangements in relation to stable turbine operation after a turbine generator system trip; (SRP 10.2, Part III, Items 3,4).
- 040.42  
(10.4.4) Provide additional description (with the aid of drawings) of the turbine by-pass valves and associated controls. In your discussion include the principle of operation, construction and set points, and the malfunctions and/or modes of failure considered in the design of the turbine by-pass system. (SRP 10.4.4, Part III, Item 1).
- 040.43  
(10.4.4) Provide the results of a failure mode and effects analysis to determine the effect of malfunction of the turbine by-pass system on operation of the reactor and turbine generator unit. (SRP 10.4.4, Part III, item 4).
- 040.44  
(10.4.4) Assure that a high energy line failure of the turbine by-pass system (TBS) will not have an adverse effect or preclude operation of turbine speed controls or any safety related components or systems located close to the TBS. (SRP 10.4.4, Part III, Item 4).

## 110.0 MECHANICAL ENGINEERING BRANCH

- 110.1  
(3.9.1) We require that you expand FSAR section 3.9.1.5 to more specifically address the consideration of asymmetric load effects on reactor coolant system components and supports which could result from postulated reactor coolant pipe breaks within cavities located inside containment. Enclosure 1 describes the information that is required.
- 110.2  
(3.9.3)  
(5.4.14) Describe the allowable buckling loads for Class 1, 2 and 3 component supports subjected to normal, upset, emergency, and faulted load combinations.
- 110.3  
(3.9.3) Provide the basis for selecting the location, required load capacity, and structural and mechanical performance parameters of safety related hydraulic snubbers in order to achieve a high level of operability assurance, including:
- (a) A description of the analytical and design methodology utilized to develop the required snubber locations and characteristics.
  - (b) A discussion of design specification requirements to assure that required structural and mechanical performance characteristics and product quality are achieved.
  - (c) Procedures, controls to assure correct installation of snubbers and checking the hot and cold settings during plant start-up tests.
  - (d) Provisions for accessibility for inspection, testing, and repair or replacement of snubbers.



- b. Pump suction and discharge nozzles to piping terminal ends;
- c. Steam generator inlet and outlet nozzles to piping terminal ends.

Note that postulated steam line breaks may control the design of certain steam generator supports, and, therefore must also be considered in the support design.

Provide an assessment of the effects of asymmetric pressure differentials<sup>1/</sup> on these systems/components in combination with all external loadings including safe shutdown earthquake loads. For the combination of dynamic responses within the reactor coolant pressure boundary and its supports, which result from the coincidence of an SSE and LOCA, the square root of the sum of the squares (SRSS) technique is acceptable contingent upon performance of an elastic dynamic analysis to meet the appropriate ASME Code, Section III, Service Limits. In all other cases, dynamic responses shall be combined by absolute summation unless justification acceptable to the staff is provided for any other method of combination.

- a. limited displacement break areas
- b. fluid-structure interaction
- c. actual time-dependent forcing function
- d. reactor support stiffness
- e. break opening times

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<sup>1/</sup> Blowdown jet forces at the location of the rupture (reaction forces), transient differential pressures in the annular region between the vessel and the shield, and transient differential pressures across the core barrel within the reactor vessel.

## ENCLOSURE 1

-Recent analyses have shown that certain reactor system components and their supports may be subjected to previously underestimated asymmetric loads under the conditions that result from the postulation of ruptures of the reactor coolant piping at various locations. It is therefore necessary to reassess the capability of these reactor system components to assure that the calculated dynamic asymmetric loads resulting from these postulated pipe ruptures will be within the bounds necessary to provide high assurance that the reactor can be brought safely to a cold shutdown condition. For the purpose of this request for additional information the reactor system components that require reassessment shall include:

- a. Reactor Pressure Vessel
- b. Fuel Assemblies, including Grid Structures
- c. Control Rod Drives
- d. ECCS Piping that is attached to the Primary Coolant Piping
- e. Primary Coolant Piping
- f. Reactor Vessel, Steam Generator, and Pump Supports
- g. Reactor Internals
- h. Biological Shield Wall
- i. Steam Generator and Pump Compartment Wall if any
- j. Steam Generator

The following information should be included in the FSAR about the effects of postulated asymmetric LOCA loads on the above-mentioned reactor system components and the reactor cavity structure.

1. Provide arrangement drawings of the reactor vessel, steam generator and pump support systems to show the geometry of all principal elements and materials of construction.
2. Consider all postulated breaks in the reactor coolant system, including the following locations:
  - a. Reactor vessel hot and cold leg nozzle to piping terminal ends;

3. If the results of the assessment required by 2. above indicate loads leading to inelastic action in these systems or displacement exceeding previous design limits provide an evaluation of the following:
  - a. Inelastic behavior (including strain hardening) of the material used in the system design and the effect of the load transmitted to the backup structures to which these systems are attached.
4. For all analysis performed, include the method of analysis, the structural and hydraulic computer codes employed, drawings of the models employed and comparisons of the calculated to allowable stresses and strains or deflections with a basis for the allowable values.
5. Demonstrate that active components will perform their safety function when subjected to the postulated loads resulting from a pipe break in the reactor coolant system.
6. Demonstrate the functional capability of any essential piping when using service level C or D limits. Guidance on acceptable methods for proceeding with the demonstration is provided in Enclosure 2.

## ENCLOSURE - 2

INTERIM TECHNICAL POSITION  
FUNCTIONAL CAPABILITY OF PASSIVE PIPING COMPONENTS  
MECHANICAL ENGINEERING BRANCH  
DIVISION OF SYSTEMS SAFETY

### I. Introduction

The functional capability of all piping components in essential ASME Class 1, 2 and 3 piping systems designed to Levels C or D service limits is required to be demonstrated. Applicants may choose to use the criteria in Section II which require no further proof of functional capability. Piping components within Section III require additional analytical or experimental proof that functional capability has been maintained.

The technical content of this position is based upon integrated experimental and analytical studies of piping system components performed at the Oak Ridge National Laboratory for the U.S. Nuclear Regulatory Commission. The program of studies, the analytical and experimental results, discussions and recommendations have been documented in a report, "Evaluation of the Plastic Characteristics of Piping Products in Relation to ASME Code Criteria, ORNL/Sub-2913/8."

### II. Situations in which Functional Capability is Assured without Further Proof

#### A. Class 1 Piping Components:

1. Functional capability may be considered assured without further proof for any Class 1 piping component when the Level "A" or "B" or "C" limit is used with Equation (9) of NB-3650 provided  $D_o/t < 50$ , where  $D_o$  is the outside diameter and  $t$  is the wall thickness of the piping component. The Level "C" limit to be satisfied for the above verification procedure is:
  - 1.5  $S_y$  for austenitic piping components, and
  - 2.25  $S_m$  for ferritic piping components
2. For tees and branch connections, the Level "D" limit may be used with Equation (9) of NB-3650 without additional requirements for functional verification, provided  $D_o/t < 50$ .

The Level "D" limit to be satisfied for the above verification procedure is:

2.0  $S_y$  for austenitic piping components, and

3.0  $S_m$  for ferritic piping components

B. Class 2/3 Piping Components:

1. Functional capability may be considered assured for Class 2/3 piping components for Levels "A" and "B" limits in Equation (9) of NC-3652.1 or ND-3652.1 provided  $D_o/t \leq 50$ .
2. For tees and branch connections, Level "C" limits may be used without additional requirements for functional verification. However, for elbows or bends, the following additional requirements shall be met whenever Level "C" limits are specified:
  - (a) Use  $(0.8 B_2)$  instead of  $(0.75 i)$  but not less than 1.0.
  - (b) Use  $(1.5 S_y)$  or  $(1.8 S_m)$ , whichever is lower for the right-hand side of Equation (9).

In each of the above cases,  $D_o/t$  shall be equal to or less than 50.

3. Class 2/3 piping components may be evaluated as Class 1 piping components for verifying functional capability, provided the rules and limits as specified in item II.A., above, are met.

III. Situations in which Functional Capability Requires Additional Demonstration

A. Class 1 Piping Components:

1. Piping components other than tees and branch connections, such as elbows, pipe bends and straight pipe, using Level "D" limits.
2. Any piping components with  $D_o/T > 50$ .

B. Class 2/3 Piping Components:

1. Straight pipe when Level "C" limits are used.

2. Elbows or pipe bends which cannot meet the requirements specified in item II.B.2, above, when Level "C" limits are specified.
3. All piping components when Level "D" limits are used. (NOTE: The ORNL report recommends against the use of Level "D" limits when functional capability must be maintained.)
4. Any piping components with  $D_o/t > 50$ .

#### IV. Definitions

Functional Capability - Capability of piping components to deliver rated flow and retain dimensional stability when the design and service loads, and their resulting stresses and strains, are at prescribed levels.

Piping Components - These items of a piping system, such as tees, elbows, bends, pipe and tubing and branch connections, constructed in accordance with the rules of Section III of the ASME Code.

Piping System - A group of connected piping components and other associated Code components (i.e., pumps, valves, vessels) performing jointly a specified plant function or, in the case of multifunctional systems, more than one function.

Essential Piping Systems - Piping systems which are necessary (a) for safe shutdown of the plant and to maintain the plant in a safe shutdown condition, or (b) to prevent or mitigate the consequences of an accident which could result in potential offsite exposures exceeding the guidelines of 10 CFR Part 100.

121.0

MATERIALS ENGINEERING BRANCH - MATERIALS INTEGRITY SECTION

121.1  
(5.4.2.2)

Confirm that the preservice and inservice inspection of steam generator tubing will be conducted in accordance with Regulatory Guide 1.83 Revision 1. If any of these examination requirements cannot be met, a complete technical justification to support your conclusions must be provided.

LOUISIANA POWER AND LIGHT

WATERFORD - UNIT 3

Structural Engineering Branch

Docket No. 50-382

Acceptance Review

( 30.01 Discuss how the lateral, overturning, and upward hydrostatic pressures  
(3.4.2) due to the maximum probable flood (including wave action) were considered in the design of the walls and foundation slab of the seismic Category I structures.

130.02 Information required in SRP Section 3.5.3.1 was included in Waterford 3  
(3.5.3) FSAR Sections 3.5.3.1 and 3.5.3.2, while nothing was provided to meet the provisions of SRP Section 3.5.3.2. Address the requirements of SRP Section 3.5.3.2 and, in addition, provide a table summarizing



the wall and roof thickness and the concrete strengths, including the age specified, for each tornado missile barrier.

130.03  
(3.7.1.1) Explain the statement made in the FSAR that "The relationship between the horizontal and vertical design response spectra is in conformance with NRC Regulatory Guide 1.60".

130.04  
(3.7.2.1) Discuss the considerations given to the following topics:

- a. The torsional, rocking, and translational responses of the structures and their foundations.
- b. The maximum relative displacement among supports of seismic Category I structures, systems, and components.

130.05  
(3.7.2.6) It is stated that the maximum response in each element (due to earthquake motion) is obtained by considering each horizontal and vertical earthquake component separately. Clarify this to indicate whether the two responses obtained for each element (N - S plus vertical, and E - W plus vertical) are combined using the absolute sum method. Demonstrate the extent to which your approach results in a design equivalent to that obtained by combining responses in accordance with Regulatory Guide 1.92.

130.06  
(3.8.2.3) The terminology used in regard to the load combinations given on FSAR pg. 3.8-13 is not the same as that used in SRP Section 3.8.2 II (3). Show by direct comparison how the ten load cases given in the FSAR comply with those required in the SRP.

211.0 REACTOR SYSTEMS BRANCH

211.1  
(3.5.1.2) The Waterford 3 FSAR does not provide the information required by section 3.5.1.2 of Regulatory Guide 1.70, Revision 2. Provide a tabulation of the information required by this section to show that the safety-related structures, systems and components inside containment required for safe shutdown of the reactor are protected from missiles generated inside containment.

211.2  
(5.2.2) A description of the design features which will be provided to mitigate the consequences of overpressurization events while operating at low temperatures is not provided in the Waterford 3 FSAR. Provide a description of the features which will be provided on the Waterford 3 unit. Our position regarding overpressurization protection while operating at low temperatures is attached. Your description should address each portion of this position.

211.3  
(5.4.7) Nuclear plants must have the capability to be taken to a cold shutdown condition using only safety-grade equipment, assuming onsite or offsite power is available and considering a single failure. Provide information to show that the Waterford 3 unit has this capability. Our position regarding this capability is attached.

211.4  
(6.3) Section 6.3.3.1 states that due to the similarity which exists for the NSSS of the 34xx MWT reactor plants, a conservative generic LOCA blowdown analysis to be used for all these reactors has been performed. Provide specific justification for reference of this analysis by the Waterford 3 unit.

211.5  
(6.3) The ECCS process flow diagrams in the Waterford FSAR do not provide the various piping flow rates for all operational modes of interest. Provide the following:

- (1) Process flow diagrams of high pressure and low pressure systems including piping flow rates with all pumps running; and,
- (2) The total pumped ECCS flow rate distribution to the intact loops as well as the broken loop with the worst single failure in the ECCS.

211.6  
(6.3) Figure 6.3-1 in the Waterford FSAR does not allow the staff to make an adequate evaluation of the Waterford ECCS. Provide ECCS piping and identification diagrams with legible symbols and grid coordinates.

211.7  
(6.3)

The information in the Waterford FSAR regarding post-LOCA passive failures is not complete. It is the Reactor Systems Branch position that detection and alarms be provided to alert the operator to passive ECCS failures during long-term cooling which allow sufficient time to identify and isolate the faulted ECCS line. The leak detection system should meet the following requirements:

- (1) Identification and justification of maximum leak rate should be provided.
- (2) Maximum allowable time for operator action should be provided and justified.
- (3) Demonstration should be provided that the leak detection system will be sensitive enough to initiate (by alarm) operator action, permit identification of the faulted line, and isolation of the line prior to the leak creating undesirable consequences such as flooding of redundant equipment or excessive radioactive fluid. The minimum time to be considered is 30 minutes.
- (4) It should be shown that the leak detection system can identify the faulted ECCS train and that the leak is isolable.
- (5) The leak detection system must meet the following standards:
  - a) Control Room Alarm
  - b) IEEE 279-1971, except single failure requirements

211.8  
(6.3)  
(15.1)

List all valves which might have their motors or controls flooded following a LOCA or steam line break. If any are flooded, evaluate the potential consequences of this flooding both for short- and long-term ECCS functions and containment isolation. List all control room instrumentation lost following these accidents.

211.9  
(15.4.1)

The staff has specific time criteria for acceptable operator action during a boron dilution event, namely:

- (1) 30 minutes during refueling, and
- (2) 15 minutes at all other times.

The reference point for "starting the clock" is when there is an identifiable alarm in the control room alerting the operator to the situation.

- 211.9  
(15.4.1) For each of the cases evaluated in the SAR, identify the alarm that alerts the operator, provide the time interval from this alarm to when the core would go critical, and identify Limiting Conditions of Operation for the Technical Specifications related to the sensors, alarms, and equipment necessary to mitigate all of these events.
- 211.10  
(14.2.7) Section 14.2.7 of the Waterford 3 FSAR states that demonstration of adequate NPSH and vortex control as required by Regulatory Guide 1.79 will be conducted by calculations. We require that you perform or reference tests which verify vortex control, available net positive suction head and acceptable pressure drops across screening, suction lines and valves, during the recirculation mode of ECCS operation. Temporary holding facilities and/or scaled testing may be appropriate if suitably justified.
- 211.11  
(none) Discuss the loss of instrument air for Waterford 3 showing that it meets the appropriate acceptance criteria for a moderate frequency event. Provide a detailed failure modes and effects discussion consistent with the next question. Causes and potential systems interactions should be particularly addressed and the loss of instrument air should be considered during all phases of reactor operation. Also, present your plans and capability for preoperational or startup tests to substantiate the analyses.
- 211.12  
(none) Operational analyses or failure mode and effects analysis of the various plant responses to the Chapter 15 events are required. To complement the SAR discussions in this regard, provide a summary of a systematic functional analysis of components required for each event analyzed in Chapter 15.0. The summary should be shown in the form of simple block diagrams beginning with the event, branching out to the various possible protection sequences for each safety action required to mitigate the consequences of the event (e.g., core cooling, containment isolation, pressure relief, scram, operator action, etc.), and ending with an identification of the specific safety actions being provided.
- When complete, each protection sequence diagram should clearly identify (for each event) the safety systems required to function to provide the safety actions necessary to mitigate the consequences of the transient or accident (during any plant operating state). An example of such a systematic functional analysis is contained in "Transactions of the American Nuclear Society 1973 Winter Meeting", November 11-15, pages 339-340.

211.13  
(15.0)

Discuss how the analyses in Chapter 15 provide a basis for partial loop operation (3 reactor coolant pumps) of the Waterford 3 unit.

## ATTACHMENT 1

### Overpressurization Protection of Pressurized Water Reactors While Operating at Low Temperature

1. A system should be designed and installed which will prevent exceeding the applicable Technical Specifications and Appendix G limits for the reactor coolant system while operating at low temperatures. The system should be capable of relieving pressure during all anticipated overpressurization events at a rate sufficient to satisfy the Technical Specification limits, particularly while the reactor coolant system is in a water-solid condition.
2. The system must be able to perform its function assuming any single active component failure. Analyses using appropriate calculational techniques must be provided which demonstrate that the system will provide the required pressure relief capacity assuming the most limiting single active failure. The cause for initiation of the event, e.g., operator error, component malfunction, will not be considered as the single active failure. The analysis should assume the most limiting allowable operating conditions and systems configuration at the time of the postulated cause of the overpressure event. All potential overpressurization events must be considered when establishing the worst case event. Some events may be prevented by protective interlocks or by locking out power. Those events should be reviewed on an individual basis. If the interlock/power lockout is acceptable, it can be excluded from the analyses provided the controls to prevent the event are in the plant Technical Specifications.
3. The system must meet the design requirements of IEEE 279 (see Implementation). The system may be manually enabled, however, the electrical instrumentation and control system must provide alarms to alert the operator to:
  - a. properly enable the system at the correct plant condition during cooldown,
  - b. indicate if a pressure transient is occurring.
4. To assure operational readiness, the overpressure protection system must be tested in the following manner:
  - a. A test must be performed to assure operability of the system electronics prior to each shutdown.
  - b. A test for valve operability must, as a minimum be conducted as specified in the ASME Code Section XI.
  - c. Subsequent to system, valve, or electronics maintenance, a test on that portion(s) of the system must be performed prior to declaring the system operational.

5. The system must meet the requirements of Regulatory Guide 1.26, "Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants" and Section III of the ASME Code.
6. The overpressure protection system must be designed to function during an Operating Basis Earthquake. It must not compromise the design criteria of any other safety-grade system with which it would interface, such that the requirements of Regulatory Guide 1.29, "Seismic Design Classification" are met.
7. The overpressure protection system must not depend on the availability of offsite power to perform its function.
8. Overpressure protection systems which take credit for an active component(s) to mitigate the consequences of an overpressurization event must include additional analyses considering inadvertent system initiation/actuation or provide justification to show that existing analyses bound such an event.

## ATTACHMENT 2

### Cold Shutdown Capability

- (1) Provide the capability to cool down to cold shutdown assuming the most limiting single failure in approximately 36 hours or show that manual actions inside or outside containment or return to hot standby until the manual actions or maintenance can be performed provides an acceptable alternative.
- (2) Provide the capability to depressurize the reactor coolant system with only safety-grade systems assuming a single failure, or show that manual actions inside or outside containment or remaining at hot standby until manual actions or repairs are complete provides an acceptable alternative.
- (3) Discuss the capability for boration with only safety-grade systems assuming a single failure or show that manual actions inside or outside containment or remaining at hot standby until manual action or repairs are completed provides an acceptable alternative. The most reactive rod must be assumed stuck out of the core.
- (4) Discuss the capability for the collection and containment of DHR system pressure relief valve discharge.
- (5) Conduct or reference applicable tests to study the mixing of the added borated water and the cooldown under natural circulation conditions with and without a single failure of a steam generator atmospheric dump valve.
- (6) Commit to providing specific procedures for cooling down using natural circulation and submit a summary of these procedures.
- (7) Provide a Seismic Category 1 AFW supply for at least four hours at hot shutdown plus cooldown to the RHR system cut-in based on the longest time (for only onsite or offsite power and assuming the worst single failure), or show that an adequate alternate Seismic Category 1 source is available.



221.0

REACTOR ANALYSIS SECTION, ANALYSIS BRANCH

221.1  
(4.4.6)

With regard to the Vibration and Loose Parts Monitoring System to be provided for Waterford Unit 3, additional description should include a discussion of the capability of the components inside containment to remain operational following the seismic events up to and including the Operating Basis Earthquake. A discussion should also be provided of any analysis and/or tests to demonstrate that the system will be adequate for the normal operating radiation, vibration, temperature and humidity environment of the reactor system. The staff requires a minimum of two sensors at each natural collection region.

221.2  
(15.0)

Our review of the CESEC code is awaiting responses by CE to outstanding questions. The available experimental data for verification of the above code is limited. We require that some startup tests be performed for demonstration of the transient characteristics of the plant and for verification of the analytical methods used to predict limiting plant transients. Accordingly, provide the details of your proposed startup test program to obtain the verification data discussed above. This program should reflect the results of tests performed at similar facilities which are applicable for verification of the Waterford 3 analyses.

ACCEPTANCE REVIEW FOR WATERFORD UNIT 3

231.0 Reactor Fuels Section, Core Performance Branch

231.1 An unexpected degradation of guide tubes that are under Control Element Assemblies (CEAs) was recently observed in irradiated fuel assemblies taken from operating Combustion Engineering reactors. Apparently, coolant turbulence is responsible for inducing vibratory motions in the normally fully withdrawn control rods. When these vibrating rods are in contact with the inner surface of the guide tubes, a wearing of the guide tube wall has taken place. Significant wear has been found to be confined to the relatively soft Zircaloy-4 guide tube because the Inconel-625 cladding on the control rods is a relatively hard wear surface. The extent of the observed wear has appeared to be plant dependent, but has in some cases extended completely through the guide tube wall. Combustion Engineering is actively searching for a solution to the wear problem. The applicant should, at the earliest possible date, provide the staff with an analysis of the method by which he will deal with this problem, which will ultimately require resolution before the issuance of the operating license.

- 231.2 The WSES-3 FSAR should provide the results of analysis that show that the Waterford Unit 3 fuel assemblies can withstand the combined seismic and LOCA mechanical loads.
- 231.3 A requirement for routine surveillance has been established and is discussed in Revision 1 of Section 4.2 of the Standard Review Plan. The WSES-3 FSAR does not provide for such a program. Accordingly, the applicant should propose a surveillance program that includes a description of (a) the on-line fuel rod failure detection method, (b) CEA integrity assurance, and (c) a post-irradiation fuel surveillance plan.

313.0

ACCIDENT ANALYSIS BRANCH313.1  
(2.1.2.1)

As stated in the FSAR, section 2.1.2.1, "LP & L owns, in title, all surface rights within the exclusion area boundary, and there is no intention to allow exploration for subsurface minerals from points on the surface of the exclusion area". Please provide information by what authority LP & L will be able to legally control the mineral rights and exploration, and clarify how undermining of the exclusion area (reactor) will be prevented if exploration is attempted from points outside of the exclusion area.

313.2  
(2.1.3.5)

The PSAR listed Kenner, La. as the nearest population center. Kenner with a 1970 population of 30,000 is 13 miles SE of the plant and borders on the west boundary of New Orleans. The FSAR refers to New Orleans as the population center now with the western boundary of 11.6 miles from the plant and a 1970 population of over 1,000,000. Unless Kenner has been encompassed or incorporated into the city of New Orleans, explain why the population center has changed.

313.3  
(3.5.1.6)

Section 3.5.1.6 states that a discussion of aircraft hazards is contained in section 2.2. Section 2.2.3.7 Aircraft Operations Evaluation, does not contain an aircraft hazard analysis of all airports located within five miles of the site. Provide an aircraft hazard analysis for the Triche airstrip, 2.2 miles east of the site.

313.4  
(3.5.1.3)

It appears that information has not been provided concerning turbine valve testing, and turbine characteristics. Provide the information requested as detailed in the Standard Format document, section 3.5.1.3, items 5 and 6.

313.5  
(6.1.2)

In Section 6.1.2.1 you indicate that certain equipment and coatings inside containment purchased prior to March 1975 do not satisfy Regulatory Guide 1.54, but that you have reviewed these exceptions on an individual basis and have judged the equipment and coatings to be acceptable. List the equipment and coatings in this category and provide the basis for judging each exception to be acceptable.

313.6  
(6.4)

Provide a schematic and a piping and instrumentation diagram (P & ID) of the control room ventilation system for both normal and emergency modes of operation. Indicate air flows (cfm) and the location of all equipment (fans, filters, dampers, instrumentation).

313.7  
(6.4.2)

Explain the modes of operation indicated in Section 6.4.2.2 for the control room ventilation system after a toxic gas or radiological emergency. Include information indicating what actions are automatic, assumed ventilation flow rates, and a definition of your term "partial filtration" as used in Section 6.4.2.2.

- 313.8  
(2.2, 6.4) For the toxic gas protection analysis presented in Section 2.2.3 (and referenced in Section 6.4.4.2), complete Table 2.2-3 and include this information in Table 2.2-4 for Hooker Chemical Co. (0.8 mi SSE), Union Carbide (1.2 mi ESE), Shell Chemical Co. (2.5 mi E), and Shell Oil Co. (3.5 mi E).
- 313.9  
(2.2, 6.4) Table 2.2-4 indicates that there are a number of occurrences that can lead to toxic gas concentrations higher than the applicable toxicity limits in the control room. This table states that, for these toxic gases, the protection features assumed to adequately protect the control room operator are either (1) administrative procedures to alert the control room operator, or (2) the buildup of control room concentration is sufficiently slow to allow isolation/donning of self-contained breathing apparatus. Document and discuss these administrative procedures in detail for each toxic gas and how they will be implemented, and verify that the buildup of control room concentration for each toxic gas will be sufficiently slow to allow isolation/donning of self-contained breathing apparatus.
- 313.10  
(6.4.4) Section 6.4.4.2 indicates that "human detection" will be relied on for detection of "some... of the postulated toxic gas accidents." Indicate which toxic gases these are, what type of human detection you propose (e.g., odor, eye irritation), levels at which human detection will occur, and the bases for your statement "there will be no chronic effects from exposure," for each toxic gas listed.
- 313.11  
(6.4.4) Section 6.4.4.2 indicates that chlorine and ammonia detectors will be provided at the normal outside air intake to the control room. Provide the design criteria for these monitors (redundancy, type of activation, setpoint, resulting action, seismic criteria, power supply).

321.0 EFFLUENT TREATMENT SYSTEMS BRANCH

- 321.1 Provide the quality group classification for the main condenser  
(10.4.2) evacuation system.
- 321.2 Provide the quality group classification for the turbine gland  
(10.4.3) sealing system.
- 321.3 Provide the quality group classification for the steam genera-  
(10.4.8) tor blowdown system downstream of the containment isolation  
valves.
- 321.4 Provide a comparison of the design, testing, and maintenance  
(9.4, criteria for the air filtration and adsorption units installed  
11.3) in the normal ventilation exhaust systems with the criteria  
in Regulatory Guide 1.140, "Design, Testing, and Maintenance  
Criteria for Normal Ventilation Exhaust System Air Filtration  
and Adsorption Units of Light-Water-Cooled Nuclear Power Plants.
- 321.5 Describe the storage area and provide the storage capacity  
(11.4) for compacted (dry) waste.
- 321.6 Provide a description of the solid waste process control program,  
(11.4) including a set of process parameters (pH, ratio of waste to  
solidification agent, temperature, etc.) which will provide  
operating boundary conditions within which reasonable assurance  
can be given that solidification of wet wastes will be complete.  
The criteria for assurance of waste solidification is contained  
in Branch Technical Position - ETSB 11-3 (Rev. 1), "Design  
Guidance for Solid Radioactive Waste Management Systems Installed  
in Light-Water Cooled Nuclear Power Reactor Plants."
- 321.7 Provide the sensitivity of the radiation monitors listed in  
(11.5) Table 11.5-1.

331.0

RADIOLOGICAL ASSESSMENT331.1  
(12.1.1)

Your description of your compliance with the guidance of Regulatory Guide 8.8 (Revision 3) states that these considerations were not implemented:

1. C.2.b(4), provisions for the shielded components to be removed readily from the cubicle for repair or replacement where such work is expected or anticipated.
2. C.2.i(7), use of radiation-damage-resistant seals and gaskets, and by using valve back seats...
3. C.2.i(6), leakage of contaminated coolant from the primary system can be reduced by using line-loaded valve packings.
4. C.2.h(1b), use of larger diameter piping (to minimize plugging).

Discuss your plan to implement alternative actions to assure that exposures will be ALARA.

331.2  
(12.1.1)

Describe the features that you have incorporated into your design to maintain occupational radiation exposure ALARA by minimizing and controlling the buildup, transport and deposition of activated corrosion products in reactor coolant and auxiliary systems. Include information on the following steps taken to minimize Co-58 and Co-60, including:

331.2  
(12.1.1)

cont'd

- 1) The use of reduced nickel content in systems in contact with reactor coolant.
- 2) Low cobalt impurity specification in system in contact with reactor coolant.
- 3) The minimization of high cobalt, hard facing wear materials in the systems in contact with reactor coolant.
- 4) The use of high flow rate/high temperature filtrations for systems in contact with reactor coolant.
- 5) The selection of valves and packings materials to minimize crud buildup and maintenance.
- 6) Provisions for decontamination of components and systems contaminated with activated corrosion products.
- 7) The types of cleanup systems for removal of crud from primary coolant during operation.

331.3  
(1.2)

Discuss the design for the solid waste handling system for remote operation with a minimum of contamination. (pg. 12.1-15)

331.4  
(12.2.2)

Provide the radioactive gaseous effluent releases due to removal of reactor vessel head, movement of spent fuel or relief valve venting. (pg. 12.2-4)

331.5  
(12.3.1)

Describe the radiation protection aspects of decommissioning that you have included in your design to ensure that occupational doses will be ALARA.



331.6  
(12.3-13)

Provide the criteria established for the changeout of air filters and adsorbers in the air cleaning system.

331.7  
(12.3.1.6)

Your discussion of the radiation fields (neutron and gamma) resulting from streaming through the annulus around the reactor vessel is inadequate. Provide justification that the proposed shielding will give expected radiation fields in the reactor building that are sufficiently small to allow for expected occupancy during plant operation.

.8  
(12.4.2)

Provide a breakdown of the activities which are included in the total 291 man-rem/unit for maintenance. Regulatory Guide 8.19, "Occupational Radiation Dose Assessment in Light Water Reactor Power Plants--Design Stage Man-Rem Estimates", (attached) which has been published for comment will provide guidance.

ACCEPTANCE REVIEW QUESTIONS  
WATERFORD STEAM ELECTRIC PLANT, UNIT 1  
DOCKET NO. 50-382

371.01  
(2.4.2.3)

Discuss the effects on your analyses of the Roof Top Drainage System being partially blocked. It is reasonable to assume that at least a part of the system would not pass the design rate of 6 inches/hour due to debris accumulation, etc. in the drains. We suggest that the design loading for the roofs be compared to the loading that could occur due to ponding caused by blocked drains. If there is sufficient margin in the design, the Roof Top Drainage System may not be critical.

371.02  
(2.4.2.3)  
RSP

Although you state in Section 2.4.2.2 that you used a Standard Project Storm without pumping to evaluate local flooding, Section 2.4.2.3 states that a 20-year storm was used. Your use of a 20-year storm coincident with an OBE is not considered conservative. Provide analyses of the site flooding potential based on an OBE (which fails the sump pumps) coincident with a Standard Project Storm. References for this position are Regulatory Guide 1.59, Revision 2, "Design Basis Floods for Nuclear Power Plants" and ANSI N170-1976, "Standards for Determining Design Basis Flooding at Power Reactor Sites".

371.03  
(2.4.3.7)

You have assumed that the mode of levee crevasse is similar to the TVA Flood Study of its Watts Bar Nuclear Plant. The study you are referring to is under active review at this time, and a final staff

decision on its acceptability has not been reached. Therefore, our positions on the Watts Bar studies may directly affect our review of your analyses.

371.04  
(2.4.3.7)

Your assumption that the levee erodes at a uniform rate may not be conservative. Our experience and studies by others, such as TVA, indicate that erosion failure rates are uniform for only a very short period, followed by a sudden, almost instantaneous, failure down to the toe of the embankment. These sudden collapses can be caused by piping or toe erosion which undermine the embankment. Accordingly, provide substantiation that your assumptions result in conservative estimates, or assume more conservative failure modes and re-evaluate the effects on the NPIS.

ACCEPTANCE REVIEW - FSAR  
WATERFORD - UNIT 3  
DOCKET NO. 50-382

372.0 Meteorology  
(2.3)

372.01 "During the period 1871-1963, 47 tropical storms or hurricanes passed  
(2.3.1) within 100 nautical miles of the Waterford site." Provide the number of tropical storms or hurricanes that have passed within 100 nautical miles of the Waterford site since 1963. Give the maximum wind speed and gusts of any of these storms (except Hurricane "Betsy", September 1965, which has already been described) that have exceeded wind velocities of 50 mph.

372.02 Give the reference for the following statement about lightning on  
(2.3.1) page 2.3-3. "There are about 2.5 cloud to ground strikes per square mile per year."

372.03 Provide the probability of a lightning strike to safety-related struc-  
(2.3.1) tures utilizing the estimates of lightning flashes to ground per unit area and considering the "attractive area" of the structures. (See for example, "Electrical Protection Guide for Land-Based Radio Facilities" by D. Bodle, 1971, (JES-159-3-3M 3/76), Joslyn Electronic Systems, or, "Lightning Protection" by J. L. Marshall, 1973, John Wiley & Sons, Inc.)

372.04 Give the average path length and width used to calculate your probability  
(2.3.1) of a tornado striking the site. If the data period used for calculating

the probability of a tornado strike does not extend to the present time, examine any tornado occurrences since the end of the original data period and compare their path lengths and widths with those from the original data period.

- 372.05 (2.3.1) Provide estimates (with references) of the maximum wind speeds that were observed from tornados that have occurred in the vicinity of the Waterford site.
- 372.06 (2.3.1) Justify the deviation from Regulatory Guide 1.76, "Design Basis Tornado for Nuclear Power Plants," (April 1974), paying particular attention to the pressure drop of three psi in three seconds.
- 372.07 (2.3.1) Provide the thickness of any ice and the duration for which it persisted as a result of the one glaze storm "reported in the region by the U.S. Weather Bureau in the 28-year period of record 1925-1953." Give similar information for glaze storms (if any) that have occurred since 1953 in the region of the Waterford site.
- 372.08 (2.3.2) Provide offsite wind speed and wind direction data from New Orleans International Airport for July 1972 through June 1975 and February 1977 to February 1978. Compare these data to the 1951 through 1960 period of wind data from the New Orleans International Airport that have already been provided.

- 372.09  
(2.3.2) Snowfalls of 2 inches or more were reported for December 1963, January 1881, and February 1899 and 1895. Provide the depths of these snowfalls.
- 372.10  
(2.3.2) Provide a map of the Waterford Unit 3 site showing the actual site boundary.
- 372.11  
(2.3.3) Studies have shown that wind sensors should be mounted on booms such that the sensors are at least one tower width away from an open-latticed tower and at least two stack or tower widths away from a stack or closed tower. For temperature sensors, mounting booms need not be as long as those for wind sensors but must be unaffected by thermal radiation from the tower itself. No temperature sensors may be mounted directly on stacks or closed towers. Mounting booms for all sensors should be oriented normal to the prevailing wind at the site. Provide information on how the wind and temperature sensors are mounted on the Waterford meteorological tower.
- 372.12  
(2.3.3) Are the data values obtained by the data logger system time averaged or instantaneous? Discuss any effects that may have occurred as a result of changing the scanning time of the data logger from five to one minute.

- 372.13  
(2.3.3) Regulatory Guide 1.23 identifies recommended accuracies of the entire meteorological data collection and reduction system. The sensor specifications identified in Section 2.3.3.2 are for the meteorological sensors independent of the data recorders and data analysis procedures. Give overall system accuracies considering the sensors, data recorders and data analysis procedures together. Do these accuracies vary when using digitized analog strip chart data as a replacement for missing data? If so, provide these accuracies.
- 372.14  
(2.3.3) Provide the dates and times of significant instrument outage, the causes of the outage, and the corrective action taken.
- 372.15  
(2.3.3) Discuss how hourly values were determined by using data obtained from the data logger system. Also identify the criteria used to determine if sufficient data were available to constitute a "valid" hour for data collection.
- 372.16  
(2.3.3) Discuss how missing data are handled during data processing and give the fraction of the meteorological data acquired through the data logger system that was initially considered missing until replaced by data from another source.
- 372.17  
(2.3.3) As discussed in Regulatory Guide 1.70, onsite meteorological data should be available on magnetic tape. Having access to onsite meteorological data on magnetic tape would facilitate the review of atmospheric dispersion characteristics. If available, provide onsite meteorological data for the period July 1972 through June 1975 and February 1977 to

February 1978 in the form of hour-by-hour averages on magnetic tape using the enclosed format. (This question is the same as 372.03 of the Environmental Acceptance Review. Only one magnetic tape need be provided.)

372.18 Provide a detailed description of the proposed display for monitoring  
(2.3.3) meteorological parameters in the control room.

372.19 Provide the building dimensions that were used to estimate a  
(2.3.4) value of  $2468\text{m}^2$  minimum building cross sectional area (page 2.3-13).

2.20 Provide the basis for assigning calm wind speeds a value of 0.18 m/s.  
(2.3.4)

372.21 The atmospheric dispersion model and procedures used to evaluate  
(2.3.4) dispersion conditions to be used in an assessment of the consequences of design basis accidents described in Section 2.3.4 are based on Regulatory Guide 1.4 and Section 2.3.4 of the Standard Review Plan. After review of the results of recent atmospheric dispersion field experiments, we have developed a modified procedure for calculating short-term relative concentration (X/Q) values which considers the following:

- (1) lateral plume meander;
- (2) atmospheric dispersion conditions as a function of direction;
- (3) wind direction frequencies; and
- (4) exclusion area boundary distances as a function of direction.

Enclosed is a copy of DRAFT Regulatory Guide 1.XXX, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear



Power Plants" (9/23/77), which describes the new procedure in detail. We believe that this model will provide an improved characterization of atmospheric dispersion conditions around the Waterford site. Also enclosed is the interim branch technical position concerning use of these two models. During our review, we will examine X/Q values for appropriate time periods for design basis accident evaluations using the modified model described in the enclosed DRAFT Regulatory Guide, and compare them with X/Q values calculated using the model described in Section 2.3.4 of the Standard Review Plan. Therefore, provide exclusion area boundary distances as a function of direction using the procedure described in the DRAFT Regulatory Guide.

It is our position that either the draft Regulatory Guide 1.XXX, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants" (dated September 23, 1977), or the procedures described in Standard Review Plan Section 2.3.4 may be used to evaluate atmospheric transport conditions for analysis of accidents with the following amendments to the draft regulatory guide model: (a) a limiting sector X/Q value at the 0.5% probability level be used\*, (b) the accumulated frequency of the limiting sector X/Q or higher value in all sectors may not exceed 5% for the site, and; (c) normalization of individual sector probability distributions is not used.

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\*Amendment based on Memorandum from H. R. Denton to D. R. Muller,  
Subject: Proposed New Meteorological Model, dated August 2, 1978.

SEP 23 1977

DRAFT

REGULATORY GUIDE 1.XXX  
ATMOSPHERIC DISPERSION MODELS FOR POTENTIAL ACCIDENT  
CONSEQUENCE ASSESSMENTS AT NUCLEAR POWER PLANTS

A. INTRODUCTION

Section 50.34 of 10 CFR Part 50 requires that each applicant for a construction permit or operating license provide an analysis and evaluation of the design and performance of structures, systems and components of the facility with the objective of assessing the risk to public health and safety resulting from the operation of the facility. Section 50.34 of 10 CFR Part 50 further states that the site evaluation factors identified in 10 CFR Part 100 shall be included in the analysis and evaluation described above. Section 100.10 of 10 CFR Part 50 states that meteorological conditions at the site and surrounding area are to be included in the factors to be considered in assessing the consequences of potential reactor accidents.

This guide provides acceptable procedures and assumptions that may be used to determine appropriate atmospheric dispersion conditions for assessing the consequences of potential nuclear power plant reactor accidents which are made as required by Section 100.11 of 10 CFR Part 50.

The Regulatory Position presented in this guide represents a substantial change in procedures used to determine atmospheric dispersion conditions appropriate for use in assessing the potential offsite radiological

consequences resulting from a range of postulated accidental releases of radiological material to the atmosphere.

This guide provides an acceptable methodology for determining site specific relative concentrations ( $\chi/Q$ ) and replaces portions of Regulatory Guide 1.3, Revision 2, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Boiling Water Reactors," Regulatory Guide 1.4, Revision 2, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Pressurized Water Reactors," Regulatory Guide 1.5, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Steam Line Break Accident for Boiling Water Reactors," Regulatory Guide 1.24, "Assumptions Used for Evaluating the Potential Consequences of a Pressurized Water Reactor Radioactive Gas Storage Tank Failure," Regulatory Guide 1.25, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors," Regulatory Guide 1.77, "Assumptions Used for Evaluating a Control Rod Ejection Accident for Pressurized Water Reactors," and Regulatory Guide 1.98, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Radioactive Offgas System Failure in a Boiling Water Reactor."

## B. DISCUSSION

The procedural changes contained in this guide are based on a review of recent experimental data on diffusion from ground-level releases without buildings present and from releases at various locations on reactor facility

buildings during stable atmospheric conditions with light wind speeds (Refs. 1-6), and a recognition that meteorological evaluation procedures should provide estimates of the variations in atmospheric dispersion that occur as a function of wind direction and distance from the source to receptor.

The procedures described in this guide incorporate the results of the atmospheric tests referred to above which verify the existence of effluent plume "meander" under stable (E, F and G) atmospheric conditions, as defined by the  $\Delta T$  criteria in Regulatory Guide 1.23 (Ref. 7), when wind speeds are light. Effluent concentrations measured over a period of one hour under such conditions have been shown to be substantially lower than would be predicted using the traditional curves (Ref. 8) of lateral and vertical plume spread, based upon current atmospheric stability criteria. The procedures in this guide also recognize that atmospheric dispersion conditions are frequently directionally dependent; that is, certain air flow directions can exhibit substantially more or less favorable diffusion conditions than others, and the wind can transport effluents in certain directions more frequently than in others.

### C. REGULATORY POSITION

This section identifies the atmospheric transport and diffusion models, methods of evaluating boundary distances for the exclusion area and the outer boundary of the low population zone for purposes of estimating dispersion values, and the methods of establishing  $\chi/Q$  value distributions and selecting  $\chi/Q$  values to be used in consequence assessments that are acceptable to the NRC staff.

## 1. Calculation of Relative Atmospheric Concentration $\chi/Q$ Values

$\chi/Q$  values should be calculated at appropriate distances (see C.2 below) for each wind direction (16 compass points; 22-1/2 degree sectors centered on true north, etc.) based on wind speed and atmospheric stability class indicated by vertical temperature gradient ( $\Delta T$ ), as defined in Regulatory Guide 1.23 for distances to 80 km (50 mi) from the site. Either hourly averaged data or joint frequency distributions of hourly data may be used. When joint frequency distributions are used, the wind speed for  $\chi/Q$  calculations should be the maximum value in the wind speed class interval so that the individual  $\chi/Q$  values are calculated to represent the minimum value in the cumulative frequency class interval. The distribution is then enveloped by the maximal  $\chi/Q$  values. Thus, when the cumulative probability distributions of  $\chi/Q$  are assessed, each  $\chi/Q$  value represents that which is equaled or exceeded within the class interval (Ref. 9). When hourly data are used, the wind speed for  $\chi/Q$  calculation should be the "hourly averaged" wind speed as defined in Regulatory Guide 1.23. Calms should be defined as hourly average wind speeds below the starting speed of the anemometer, and should be assigned a wind speed equal to that of the anemometer or vane starting speed, whichever is higher. When joint frequency distributions are used, wind directions during calm conditions should be assigned in proportion to the directional distribution of the lowest non-calm wind speed class. When hourly data are used, wind directions during calm conditions should be assigned in proportion to the directional distribution of non-calm conditions with a wind speed less than 0.7 meters per second (m/s) (the wind speed class limit, i.e., 1.5 mph).

Formulae and parameters presented in this section should be used in the absence of site specific diffusion data unless unusual siting, meteorological or terrain conditions dictate the use of other models or considerations. For example, quality controlled, site-specific atmospheric diffusion tests may be used as a basis for modifying the formulae and parameters.

a. Short-term ( $\leq 2$  hours) release period calculations

Acceptable mathematical models for calculating  $\chi/Q$  values appropriate for short time period atmospheric dispersion calculations are presented below. Meteorological data and calculations for the one hour time period are assumed to apply over the entire two hour release period. This assumption has been confirmed as reasonably conservative, considering the variation with time of postulated accidental releases. If releases associated with a given postulated event are estimated to occur in a period substantially less than one hour (i.e., less than 20 minutes), the applicability of the models should be evaluated on a case-by-case basis.

(1) Releases through vents or other building penetrations

This class of release modes includes all release points or areas which are lower than two and one half times the height of adjacent solid structures (Ref. 10). The formulae and assumptions are:

(a) During conditions of neutral (D) and stable (E, F and G) stability when the speed at the 10 meter level is less than 6 m/s, credit for horizontal plume meander can be considered such that

$$\frac{\chi}{Q} = \frac{1}{\bar{u}_{10} \pi \Sigma_y \sigma_z} \quad (1)$$

whenever the  $\chi/Q$  value, calculated using Equation 1, is less than the greater value calculated from either

$$\frac{\chi}{Q} = \frac{1}{\bar{u}_{10} (\pi \sigma_y \sigma_z + A/2)} \quad (2)$$

or

$$\frac{\chi}{Q} = \frac{1}{\bar{u}_{10} (3\pi \sigma_y \sigma_z)} \quad (3)$$

where

- $\chi/Q$  is the relative concentration ( $\text{sec}/\text{m}^3$ ) at ground level,  
 $\pi$  is 3.14159,  
 $\bar{u}_{10}$  is the wind speed (m/s) at 10 meters above grade,  
 $\Sigma_y$  is the lateral plume spread (m), a function of atmospheric stability, wind speed  $\bar{u}_{10}$  and downwind distance from release. For distances to 800 meters,  $\Sigma_y = M\sigma_y$ ;  $M$  being a function of atmospheric stability and wind speed (see Figure 3). For distances greater than 800 meters,  $\Sigma_y = (M-1)\sigma_{y800m} + \sigma_y$ ,  
 $\sigma_y$  is the lateral plume spread (m), a function of atmospheric stability and distance (Figure 1),



$\sigma_z$  is the vertical plume spread (m), a function of atmospheric stability and distance (Figure 2), and

A is the smallest vertical plane, cross-sectional area ( $m^2$ ) of the building from which the effluent is released.

Otherwise  $x/Q$  is the greater value calculated from either Equation 2 or 3.

In other words, calculate  $x/Q$  values based on Equations 1, 2, and 3. Compare the values computed from Equations 2 and 3, and select the higher value. Compare this higher value with the value calculated through use of Equation 1, and select the lower of these two values to represent the  $x/Q$  value for postulated release and atmospheric conditions. Examples and a detailed explanation of the rationale are given in Appendix A.

(b) During all other atmospheric stability and/or wind speed conditions,  $x/Q$  is the greater value calculated from either Equation 2 or 3.

## (2) Stack Releases

A stack release is assumed when the effluent is exhausted from a release point that is higher than two and one half times the height of adjacent solid structures (Ref. 10). The general formula and assumptions are:

$$\frac{x}{Q} = \frac{1}{\pi \bar{u}_h \sigma_y \sigma_z} \exp \left[ \frac{-h_e^2}{2\sigma_z^2} \right] \quad (4)$$

where

$\bar{u}_h$  is the wind speed (m/s) which represents conditions at the release height,

$h_e$  is the effective height (m) determined from

$$h_e = h_s - h_t,$$

$h_s$  is the height of the release point above plant grade, and

$h_t$  is the maximum terrain height above plant grade between the release point and the point for which the calculation is made, but should not be allowed to exceed  $h_s$ .

The other parameters in Equation 4 have been defined previously.

Atmospheric stability for determination of  $\sigma_y$  and  $\sigma_z$  is obtained from the vertical temperature differences ( $\Delta T$ ) between the release height and the 10-meter level as described in Regulatory Guide 1.23.

For those cases where fumigation conditions are to be evaluated for elevated releases, the formula and assumptions are:

$$\frac{X}{Q} = \frac{1}{(2\pi)^{1/2} \bar{u} \sigma_y h_e} \quad (5)$$

where

$\bar{u}$  is wind speed (m/s) representative of the layer,  $h_e$ , for which a value of 2 m/s is a reasonably conservative assumption in most cases,

$\sigma_y$  is the lateral plume spread (m) at a given distance which is usually assumed for a moderately stable (F) atmospheric stability condition which normally precedes the onset of fumigation, and

$h_e$  is as defined above for elevated releases.

The  $\chi/Q$  value calculated by Equation 5 should not exceed  $\frac{1}{\pi \bar{u} \sigma_y \sigma_z}$

b. Release periods greater than 2 hours

The average  $\chi/Q$  values should be calculated for appropriate time periods during the course of the postulated accident as described below. The time periods for averaging should represent intra-diurnal, diurnal and synoptic meteorological regimes (e.g., 8 and 16 hours and 3 and 26 days as presented in Section 2.3.4 of Regulatory Guide 1.70) (Ref. 11). The  $\chi/Q$  value for each appropriate time period at the distance of interest in each direction sector should be obtained by a logarithmic interpolation between the calculated value that is selected using the procedure described in Section C.3.a below, assumed as a "2 hour" value, and the annual average (8760 hour) value at the distance of interest in that direction sector (Ref. 9).

The annual average  $\chi/Q$  value should be calculated using the method described in Regulatory Guide 1.111, Section C.1.c. (Ref. 12), but with  $h_e$  determined as described in Section C.1.a.(2) above.

2. Determination of Distances for  $\chi/Q$  Calculations

In order to take into consideration the possibility of airflow trajectory deviations, plume segmentation (particularly in light wind, stable conditions), and the potential for wind speed and direction frequency shifts from year to year, the following procedure should be used to determine the distance from which the calculations of relative concentrations ( $\chi/Q$ ) are made.

For each wind direction sector, the minimum distance (exclusion area or LPZ) to be assumed for the sector of interest should be defined as the minimum distance within that sector and one-half of the width of the direction sector on either side of the sector of interest. Effectively, this distance is the minimum distance of either the exclusion area or LPZ within a 45 degree direction sector, centered on the direction sector of interest. However, should there not be a well defined exclusion boundary in a sector (e.g., a sector extending seaward at a coastal site) then the distance for that sector should be taken as that distance over which the applicant or licensee intends to have control.

### 3. Determination of $\chi/Q$ Values by Sector

#### a. Assessment of $\chi/Q$ 's at the exclusion distance

Acceptable procedures for selecting the  $\chi/Q$  values to be used in the consequence assessment analyses for both the "conservative" and "realistic" accident conditions (see Section 2.3.4 of Ref. 11) are described below. For the realistic assessment, fumigation conditions may be ignored.

##### (1) Non-fumigation conditions

Cumulative probability distributions of the  $\chi/Q$  values, as determined from Section C.1.a above at the distances determined from Section C.2 above, excluding fumigation from elevated releases, should be constructed for each of the 16 cardinal compass point directions (22-1/2 degree direction sectors). Each directional probability distribution should be normalized to 100%. If joint frequency table data are used to calculate the  $\chi/Q$  values, the cumulative probability distribution function should be computed such as to envelope the data points.

The effective probability level ( $P_e$ ) for the selection of the  $\chi/Q$  value in each direction sector should be determined (Ref. 9) by first multiplying the probability level ( $P$ ), selected as 5% for the conservative accident assessment, by the ratio of the total number of hours ( $N$ ) having valid wind and stability data in the meteorological data record (1 year = 8760 hours) to the number of those hours ( $n$ ) in which the wind flow was into the direction sector of interest, and then dividing this product by the total number of sectors ( $S$ ) (16 for sectors of  $22\frac{1}{2}$  degrees). For the realistic accident assessment  $\chi/Q$  determination as described in Section 2.3.4 of Regulatory Guide 1.70 (Ref. 11),  $P$  should be selected as 50%. This procedure, in equation form may be stated as:

$$P_e = \frac{P (N/n)}{S} \quad (6)$$

where the individual terms in the equation are described as above. It should be noted that  $P_e$  can exceed 100% if  $n$  is sufficiently small. In those directions, the selection of a  $\chi/Q$  value may be ignored unless the  $\chi/Q$  values for that sector are very high when compared with  $\chi/Q$  values at  $P_e$  in other direction sectors.

For each assessment, the  $\chi/Q$  values that are selected, as described above, for the 16 directions are compared and the highest value is selected.

(2) Fumigation conditions - conservative assessment

In the absence of information which indicates that fumigation conditions occur substantially less than five percent of the time,  $\chi/Q$  values should be calculated, assuming fumigation conditions, for each of the 16 directions sectors using Equation 5.

## (a) Inland sites

For elevated releases at sites located at distances equal to or greater than 3200 meters from large bodies of water (e.g., oceans or a Great Lake), a fumigation condition at the exclusion distance should be assumed to exist at the time of the accident and continue for one-half hour (Ref. 13). In this case, two  $\chi/Q$  values, one for the 0 to 1/2-hour time period and the other for the 1/2 to 2-hour time period following the accident, should be selected for the accident consequence analysis using the following procedures.

For the 0 to 1/2-hour time period  $\chi/Q$  values should be determined, using Equation 5 for sectors in which the effective height of release ( $h_e$ ) is greater than 0, or using Equation 4 and the selection procedure described in Section C.3.a.(1) above for sectors in which  $h_e = 0$ , for each of the 16 direction sectors.

For the 1/2 to 2-hour time period,  $\chi/Q$  values for each of the 16 direction sectors should be determined using Equation 4 and the selection procedure described in Section C.3.a.(1) above.

## (b) Coastal sites

For elevated releases at sites located less than 3200 meters from large bodies of water, a fumigation condition at the exclusion distance should be assumed to exist at the time of the accident and continue for four hours (Ref. 13) in each of the onshore and along shore airflow directions. The  $\chi/Q$  value to be used in the accident consequence analysis for the 0 to 2 hour period following an accident, in this case, is the maximum of the 16 individual direction sector  $\chi/Q$  values, calculated and selected as described

above for the 0 to 1/2-hour time period. Therefore, two-hour  $\chi/Q$  values for exclusion distances should be based entirely on fumigation conditions.

This conservative assessment does not consider frequency and duration of fumigation conditions as a function of airflow direction. If information can be presented to substantiate the actual directional occurrence and duration of fumigation conditions at a site, the assumptions of fumigation in all appropriate directions and of duration of one-half hour and four hours may be modified. Then fumigation need only be considered for airflow directions in which fumigation has been determined will occur and of a duration determined from the study. For example, examination of site-specific information at a location in a pronounced river valley may indicate that fumigation conditions occur predominately during the down-valley "drainage flow" regime and persist for durations of about one-half hour. Therefore, in this case airflow directions other than the down-valley directions can be excluded from consideration of fumigation conditions, and the duration of fumigation would still be considered as one half hour. On the other hand, sites in open terrain (non-coastal) may show no directional preference for fumigation conditions, but may show durations much less than one half hour. In this case, fumigation should be considered for all directions, but with durations much less than one-half hour.

b. Assessments of  $\chi/Q$ 's at the LPZ

Acceptable procedures for selecting the  $\chi/Q$  values to be used in the consequence assessments are described below.

In most cases, the highest  $\chi/Q$  values for the appropriate time periods will all occur within the same 22-1/2 degree direction sector.

However, for those sites at which the highest  $\chi/Q$  values for the various time periods do not all occur within the same direction sector, an evaluation of the consequences of the potential accident should be made for each sector using the  $\chi/Q$  values in that sector for the course of the accident analysis. The  $\chi/Q$  values, for that sector which produces the greatest potential risk to the health and safety of the public (i.e., the highest dose estimate), should be considered controlling.

(1) Non-fumigation conditions

The 16 sets of  $\chi/Q$  values obtained by using the interpolation procedure described in Section C.1.b above should be compared, and the values for the sector, evaluated as described above, should be considered controlling. This procedure may be used for both the conservative and realistic accident assessments.

(2) Fumigation conditions - conservative assessment

For elevated releases at sites located at distances equal to or greater than 3200 meters from large bodies of water, the  $\chi/Q$  value for each sector, at the LPZ, for the 0 to 1/2 hour and 1/2 to 2 hour time periods following the accident should be determined as described for this case in Section C.3.a.(2) above.

For elevated releases at sites located less than 3200 meters from large bodies of water, the  $\chi/Q$  value for each sector, at the LPZ, for the 0 to 4 hour period following an accident should be evaluated as described for this case in Section C.3.a.(2) above.



D. IMPLEMENTATION

The purpose of this section is to provide information to applicants regarding the NRC staff's plans for using this regulatory guide.

This guide reflects current practice adopted by the Commission. Except in those cases in which the applicant proposes an acceptable alternative method for complying with specified portions of the Commission's regulations, the method described herein will be used in the evaluation of submittals for operating license or construction permit applications docketed after \* \_\_\_\_\_. The method described herein will be considered for licensing actions concerning operating reactors on an individual basis. If an applicant wishes to use this regulatory guide in developing submittals for operating license or construction permit applications docketed on or before \* \_\_\_\_\_, the pertinent portions of the application will be evaluated on the basis of this guide.

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\* Date 4 months after publication for public comment.

## REFERENCES

1. Van der Hoven, I., " A Survey of Field Measurements of Atmospheric Diffusion Under Low-Wind Speed Inversion Conditions," Nuclear Safety, March-April 1976, Vol. 17 No. 4.
2. Start, G. E. et al., "Rancho Seco Building Wake Effects On Atmospheric Diffusion," NOAA Technical Memorandum ERL ARL-XX (in draft), Air Resources Laboratory, Idaho Falls, Idaho, 1977, available from Publication Services, Environmental Research Laboratories, National Oceanic and Atmospheric Administration, Boulder, Colorado 80302.
3. Wilson, R. B., et al., "Diffusion Under Low Windspeed Conditions Near Oak Ridge, Tennessee," NOAA Technical Memorandum ERL ARL-61, Air Resources Laboratory, Idaho Falls, Idaho, 1976, available from Publication Services, Environmental Research Laboratories, National Oceanic and Atmospheric Administration, Boulder, Colorado 80302.
4. Sagendorf, J. F. and C. R. Dickson, "Diffusion Under Low Windspeed, Inversion Conditions," NOAA Technical Memorandum ERL ARL-52, Air Resources Laboratory, Idaho Falls, Idaho, 1974, available from Publication Services, Environmental Research Laboratories, National Oceanic and Atmospheric Administration, Boulder, Colorado 80302.

REFERENCES (Cont'd.)

5. Gulf States Utilities Company, "Dispersion of Tracer Gas at the Proposed River Bend Nuclear Power Station," Preliminary Safety Analysis Report, Amendment 24, Docket Numbers 50-458 and 50-459, 1974
6. Metropolitan Edison Company, "Atmospheric Diffusion Experiments with SF<sub>6</sub> Tracer Gas at Three Mile Island Nuclear Station Under Low Wind Speed Inversion Conditions," Final Safety Analysis Report, Amendment 24, Docket Number 50-289, 1972
7. Regulatory Guide 1.23 (Safety Guide 23), "Onsite Meteorological Programs," U.S. Nuclear Regulatory Commission, Washington, D.C.
8. Gifford, F. A., Jr., "An Outline of Theories of Diffusion in the Lower Layers of the Atmosphere," Chapter 3 in Meteorology and Atomic Energy 1968 (D. H. Slade, Ed), available as TID-24190 from the National Technical Information Service, Springfield, VA 22151.
9. Markee, E. H., Jr. and J. R. Levine, "Probabilistic Evaluations of Atmospheric Diffusion Conditions for Nuclear Facility Design and Siting," Paper in Proceedings of the American Meteorological Society Conference on Probability and Statistics in Atmospheric Sciences, Las Vegas, Nevada, November 1977. (in draft)

REFERENCES (Cont'd.)

10. Synder, W. H. and R. E. Lawson, Jr., "Determination of a Necessary Height for a Stack Close To A Building-A Wind Tunnel Study," Atmospheric Environment, Vol. 10, pp 683-691, Pergamon Press, 1976
11. Regulatory Guide 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants - LWR Edition," U.S. Nuclear Regulatory Commission, Washington, D.C.
12. Regulatory Guide 1.111, "Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water-Cooled Reactors," U.S. Nuclear Regulatory Commission, Washington, D.C.
13. Van der Hoven, I., "Atmospheric Transport and Diffusion at Coastal Sites," Nuclear Safety, Vol. 8, pp 490-499, 1967

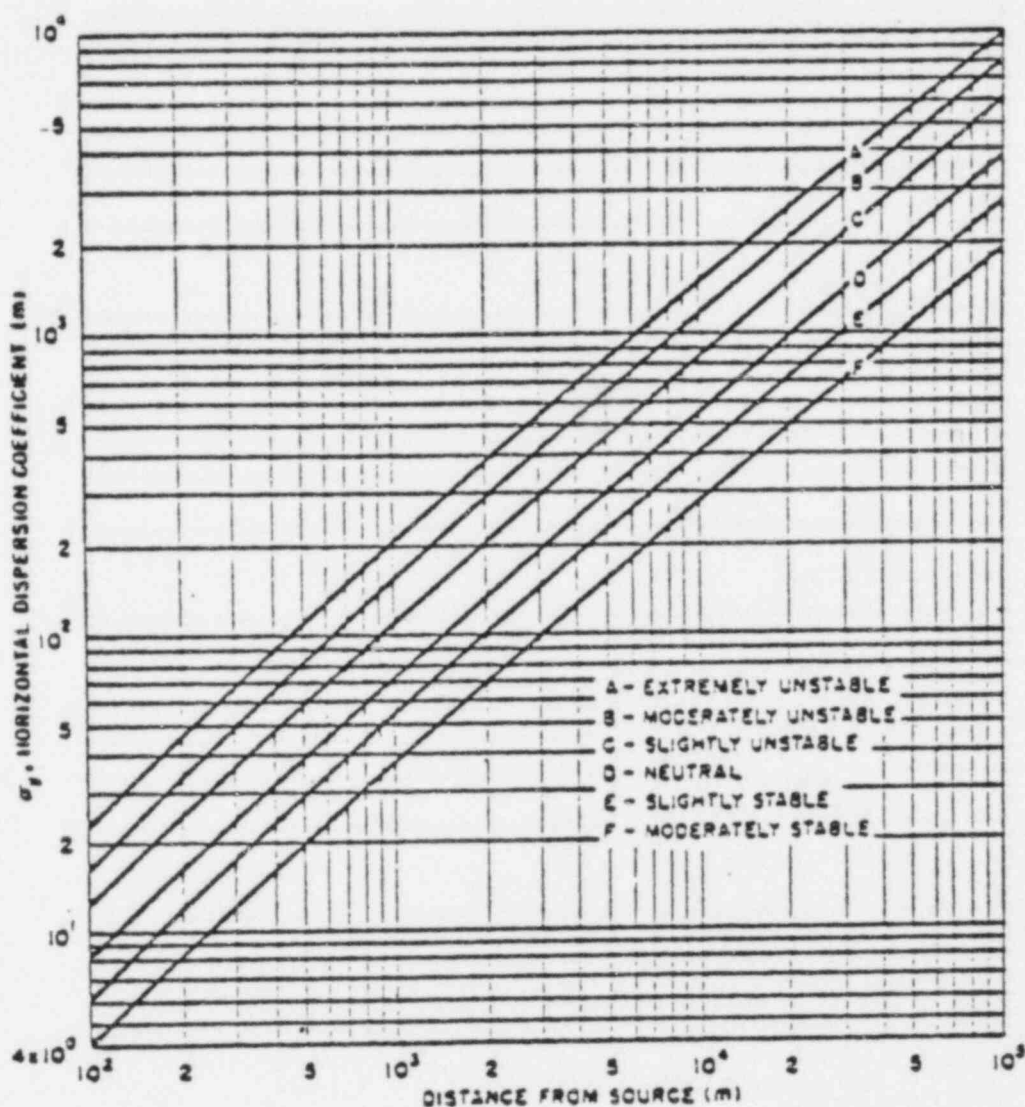


Figure 1. Lateral diffusion,  $\sigma_y$ , vs. downwind distance from source for Pasquill's turbulence types (Ref. 8).

For purposes of estimating  $\sigma_y$  during extremely stable (G) conditions, without plume meander or other lateral enhancement, the following approximation is appropriate:

$$\sigma_y (G) = \frac{2}{3} \sigma_y (F)$$

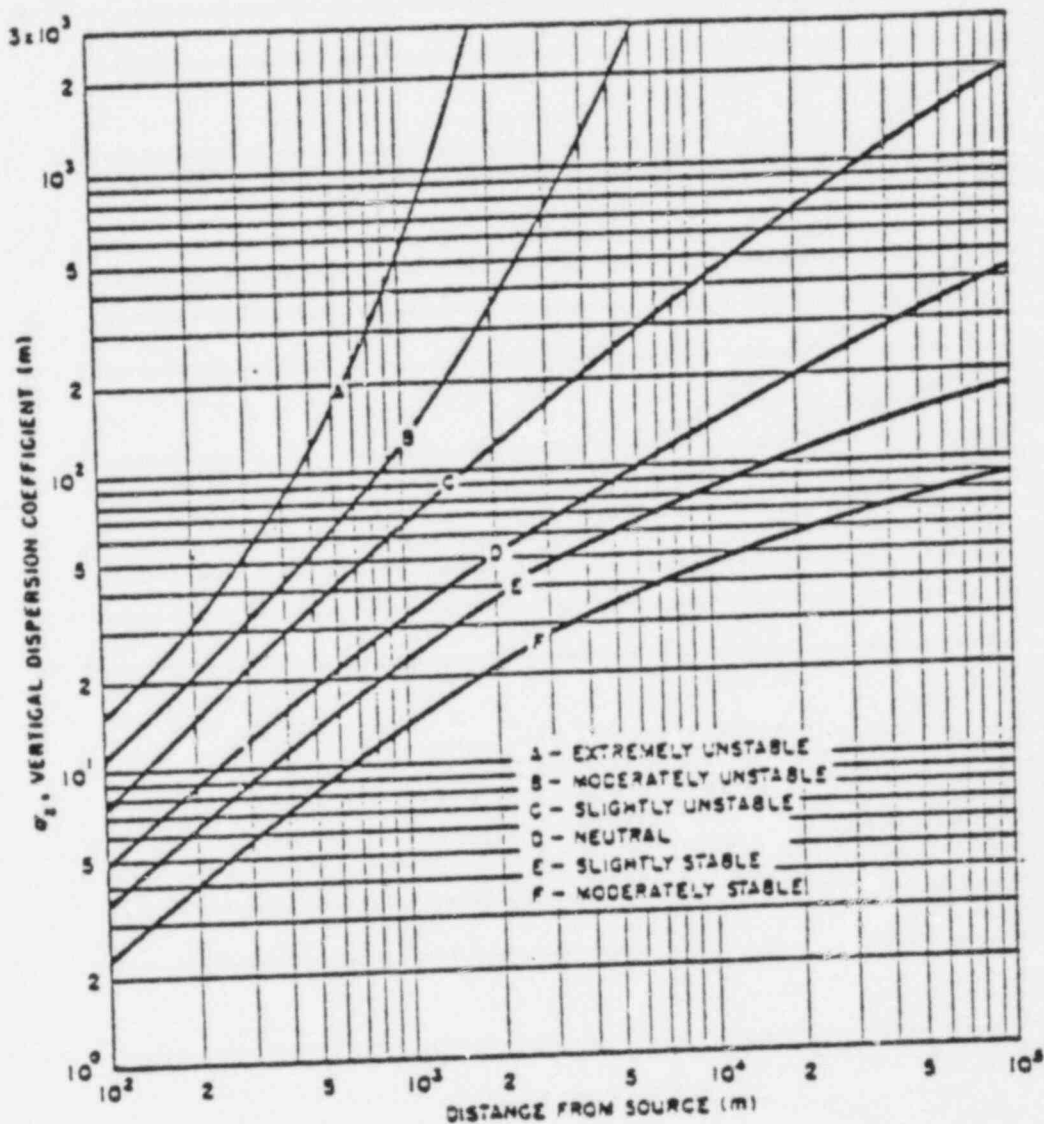


Figure 2. Vertical diffusion,  $\sigma_z$ , vs. downwind distance from source for Pasquill's turbulence types (Ref. 8).

For purposes of estimating  $\sigma_z$  during extremely stable (G) conditions, the following approximation is appropriate:

$$\sigma_z(G) = \frac{3}{5} \sigma_z(F)$$

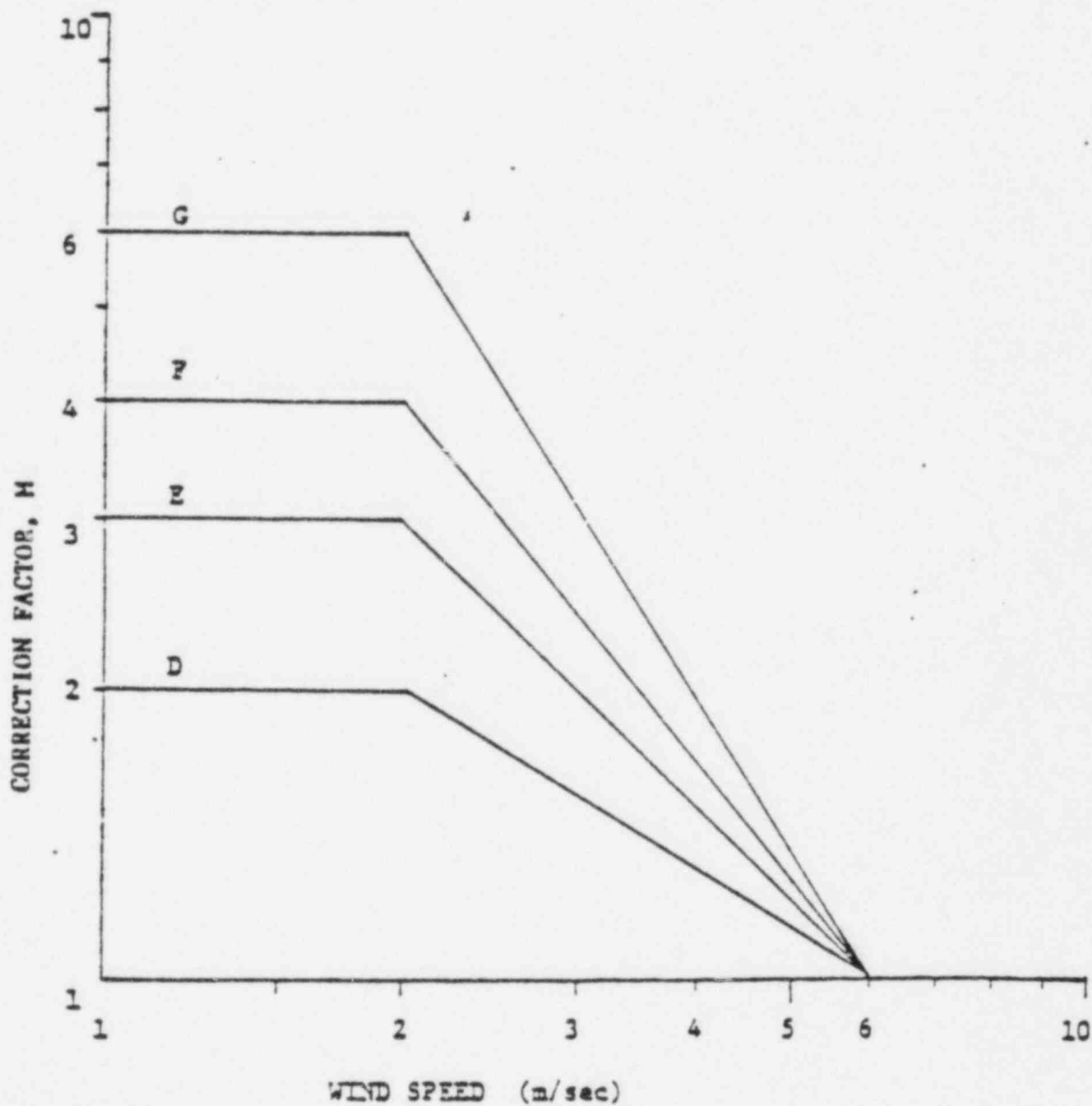


Figure 3. Correction factors for Pasquill-Gifford  $\sigma_y$  values.

(Based on analyses of Ref. 2)

## APPENDIX A

ATMOSPHERIC DIFFUSION MODEL FOR RELEASES THROUGH  
VENTS AND BUILDING PENETRATIONSRationale

The effects of building wake mixing and ambient plume meander on atmospheric dispersion is expressed in this guide in terms of conditional use of Equations 1, 2 and 3. Equation 1 is an empirical formulation based on atmospheric diffusion experiment results (Reference 2) and includes the combined effects of increased plume meander and of building wake in the horizontal crosswind direction over time periods of one hour when the wind speed is light. Although the results could not be quantified, these experiments also indicate that vertical building wake mixing is not as complete during light wind, stable atmospheric conditions as during moderate wind, unstable conditions. Equations 2 and 3 are formulations which have had widespread acceptance within the meteorological community over a period of many years (Ref. 8), but have been recently found to provide estimates which are too conservative at least for the light wind, stable atmospheric conditions (Ref. 1 and 2). Therefore, based on the principles that horizontal plume meander dominates dispersion during light wind, stable conditions and that meander diminishes as the wind speed increases and the atmospheric stability decreases while building wake mixing becomes more effective in dilution of effluents, the conditional use of Equations 1, 2 and 3 is appropriate for providing reasonable  $\chi/Q$  estimates.



Example

Figure A-1 shows plots of  $\chi/Q$  times the wind speed  $\bar{u}_{10}$  versus downwind distance for Equations 1, 2 and 3 for atmospheric stability class G. Equation 1 is plotted for  $M = 2, 3$  and 6. Figure A-2 shows plots of  $\chi/Q$  times  $\bar{u}_{10}$  versus downwind distance based on the conditional use of Equations 1, 2 and 3 as described in the Regulatory Position for wind speed conditions appropriate for  $M = 2, 3$  and 6. Comparison of Figure A-1 to Figure A-2 shows that for  $M = 6$ , Equation 1 is used for all distances since the  $\chi \bar{u}_{10}/Q$  for Equation 1 is less than the values calculated for the greater value produced by either Equation 2 or Equation 3 at all distances. For  $M = 3$ , the values from Equation 1 are used for distances beyond 0.8 km since the greater value produced by either Equation 2 or Equation 3 is greater than the value produced by Equation 1. However, for distances less than 0.8 km, Equation 1 equals Equation 3. Therefore, the appropriate  $\chi/Q$  value is determined from Equation 3 since Equation 1 is not less than Equation 3, and Equation 3 produces the higher value when compared with Equation 2. When  $M = 2$ , Equation 1 will not be used at all since it is never less than the greater value produced by either Equation 2 or Equation 3. Instead, Equation 3 will be used up to 0.8 km and Equation 2 will be used beyond 0.8 km.

DRAFT

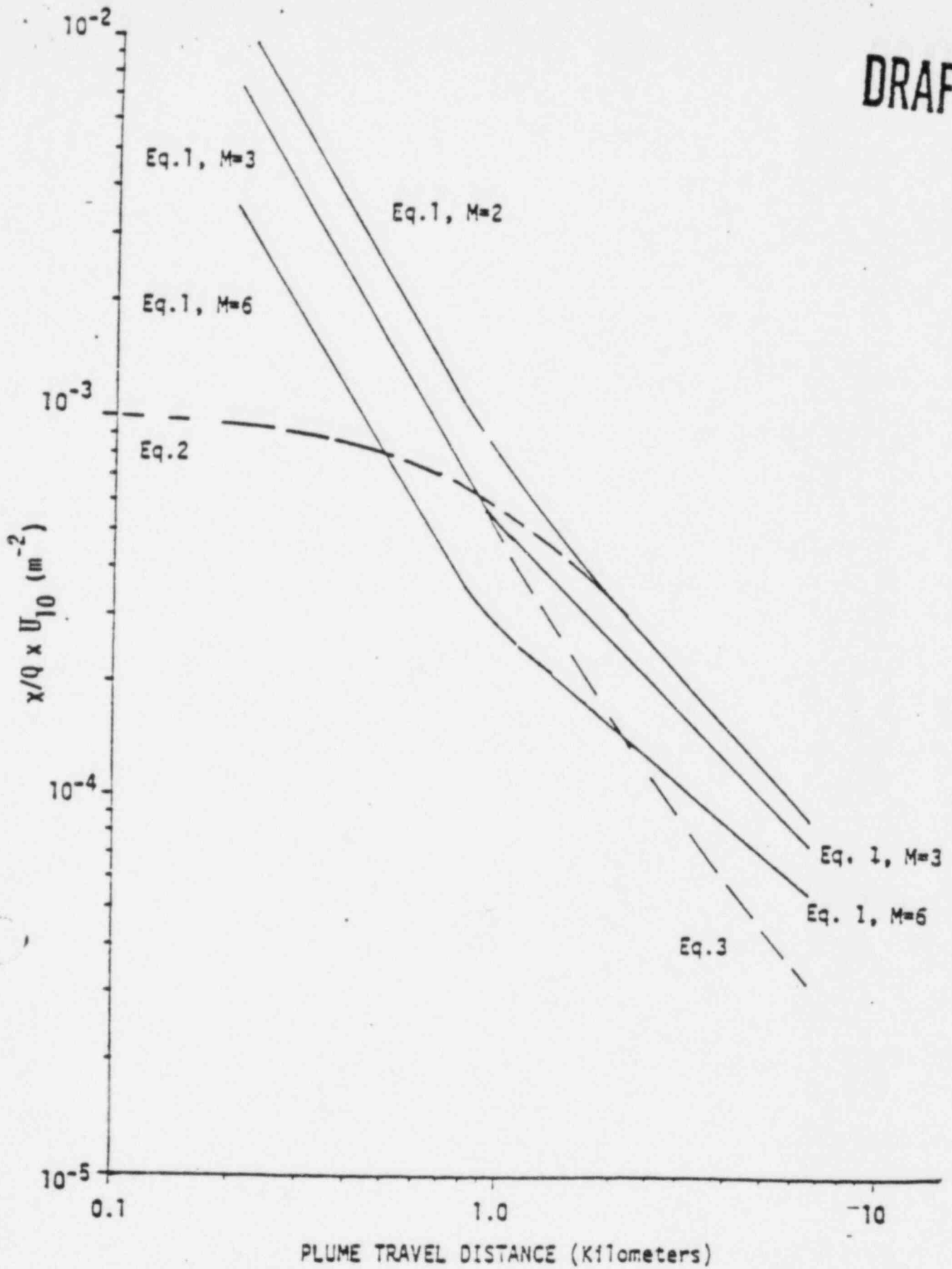


Figure A-1.  $x \bar{U}_{10}/Q$  as a function of plume travel distance for G stability condition using Equations 1, 2 and 3.

DRA

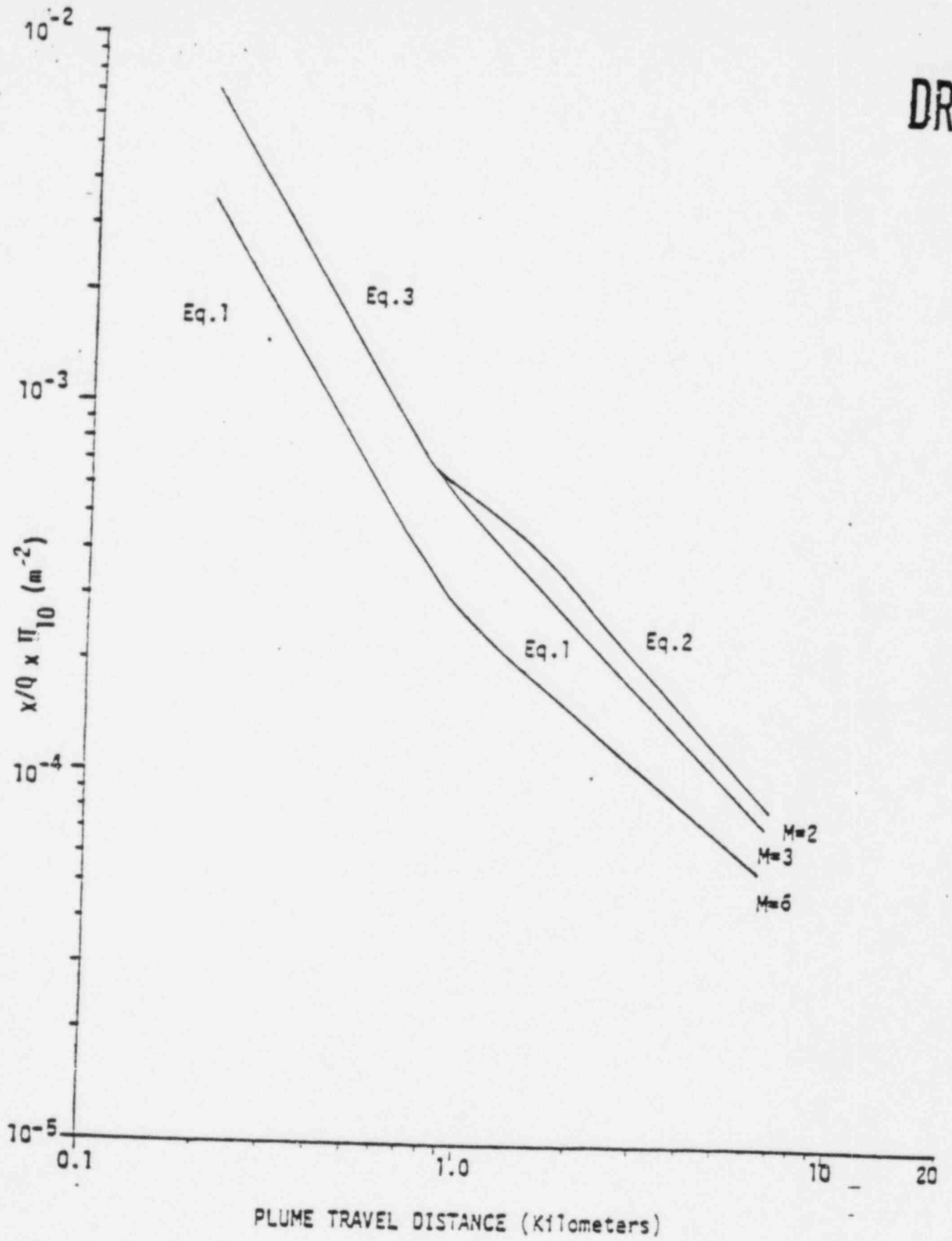


Figure A-2. Regulatory Position on  $x \bar{U}_{10}/Q$  as a function of plume travel distance for G stability condition.

WATERFORD STEAM ELECTRIC STATION UNIT NO. 3

REQUEST FOR INFORMATION

422.0 Conduct of Operations

422.1 Describe your specific provisions for providing offsite  
(13.1.1.2) technical support for the plant staff in the areas of:

- a. Nuclear, mechanical, structural, electrical, thermal-hydraulic, metallurgical and materials, and instrumentation and controls engineering.
- b. Plant chemistry.
- c. Health physics.
- d. Fueling and refueling operations support.
- e. Maintenance support.

As part of this response, provide a further breakdown of the organization units reporting to the Nuclear Projects Manager, the Chief Engineer (G.O. Engineering), and other organizational units that will provide offsite technical support for the operation of WSES Unit 3. Include the number of professional persons assigned to each of these units and the resumes of individuals fulfilling the areas of responsibility noted above. Provide the same type of information if the responsibility will be assigned to Middle South Services.

422.2  
(13.1.3.1)

It is not possible to make a comparison between all of your plant staff positions shown in Figure 13.1-2 and ANSI/ANS-3.1-1978. Therefore, provide a list of all your plant staff personnel from the technician and mechanic level to the Station Superintendent (excluding administrative personnel) and the comparable ANSI/ANS-3.1-1978 position. If you do not consider there to be a comparable ANSI/ANS-3.1-1978 position, provide a description of the qualification requirements for each such position.

CHAPTER 14 - INITIAL TEST PROGRAM

REQUEST FOR INFORMATION

- 423.1  
(14.2.7) Your proposed method for conformance with Regulatory Positions 8, 9, and 10 in Regulatory Guide 1.80, "Preoperational Testing of Instrument Air Systems," is not acceptable. Your application should be modified to propose in-plant testing to simulate loss of air supply to the system loads (valve operators, actuators, instruments, etc.) to assure that their response is in accordance with design analysis for a full or partial loss of air supply pressure.
- 423.2  
(14.2.8) Your description of your program for utilization of reactor operating and testing experiences in development of the test program is not acceptable. Your application should be modified to identify the sources of information used, the position title or organizational unit responsible for review and evaluation of this information, the types of reactors or specific reactors to be included in the program, the conclusions reached from the study, and the effect/changes made to the test program.
- 423.3  
(14.2.12) The test abstracts provided for preoperational and startup tests are not in accordance with the intent of Section 14.2 of the Standard Format document and are in a form that is not suitable for evaluation by the staff. All test abstracts should be modified to identify specific 1) test objectives, 2) test methods, and 3) acceptance criteria for each test.

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Louisiana Power & Light Co.

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