

June 12, 1987

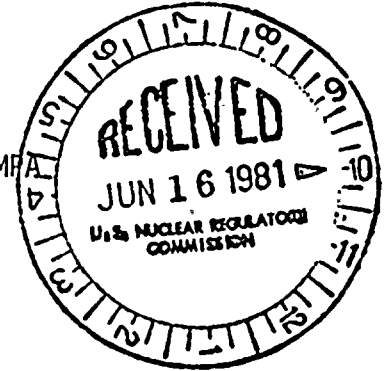
Docket Nos.: 50-389

Dr. Robert E. Uhrig, Vice President
Advanced Systems and Technology
Florida Power & Light Company
P. O. Box 529100
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Dear Dr. Uhrig:

SUBJECT: ST. LUCIE, UNIT 2 DRAFT SER FROM THE MECHANICAL ENGINEERING
BRANCH (MEB)

The MEB has completed the review of the St. Lucie, Unit 2 FSAR through Amendment 1. They have chosen not to develop a round of questions but to proceed directly to a draft SER input. FP&L should prepare an agenda for a meeting in which we can discuss and resolve the open issues in our review. We anticipate this meeting being held over a 3-5 day period at a mutually agreeable site. We propose this meeting be held the week of July 27, 1981. After this meeting and any necessary follow-up, we will update the SER input into a form sufficiently clean for publication. We want to emphasize that we expect this extended meeting to resolve almost all of these open issues. Therefore, you should bring the NSSS, AE, and your people necessary to both discuss technical details and make binding commitments. We recommend the meeting be held at the Ebasco offices in New York.

Please conduct Mr. Nerses (301-492-7468), St. Lucie 2 Project Manager, if you need to discuss this matter further.

Sincerely,

151

Robert L. Tedesco, Assistant Director
for Licensing
Division of Licensing

Enclosure:
As stated

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APP 3

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UNITED STATES
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JUN 12 1981

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A handwritten signature in dark ink, appearing to read "Robert L. Tedesco".

Robert L. Tedesco, Assistant Director
for Licensing
Division of Licensing

Enclosure:
As stated

cc: See next page

JUN 12 1964

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JUN 12 1981

ST. LUCIE PLANT UNIT NO. 2 DRAFT SER

3.6.2 Determination of Break Location and Dynamic Effects Associated with the Postulated Rupture of Piping

The review performed under this section pertains to the applicant's program for protecting safety-related components and structures against the effects of postulated pipe breaks both inside and outside containment. The effect that breaks or cracks in high and moderate energy fluid systems would have on adjacent safety-related components or structures are required to be analyzed with respect to jet impingement, pipe whip, and environmental effects. Several means are normally used to assure the protection of these safety-related items. They include physical separation, enclosure within suitably designed structures, the use of pipe whip restraints, and the use of equipment shields.

Our review under Standard Review Plan Section 3.6.2, "Determination of Break Locations and Dynamic Effects Associated with the Postulated Rupture of Piping," was concerned with the locations chosen by the applicant for postulating piping failures. We also reviewed for the size and orientation of these postulated failures and how the applicant calculated the resultant pipe whip and jet impingement loads which might affect nearby safety-related components.

I. REACTOR COOLANT SYSTEM (RCS) MAIN LOOP PIPING

The applicant has referenced Topical Report CENPD-168A "Design Basis Pipe Break's for the Combustion Engineering Two Loop Reactor Coolant System." While this report provides a generically acceptable basis for the implementation of criteria for postulated pipe breaks in the RCS and provides a level of protection equivalent to that resulting from the application of the criteria of Regulatory Guide 1.46, the staff's position is that each application that references this Topical Report must also include additional information to ensure that the plant under review is within the limits of CENPD-168A,

Revision 1. The additional information required in this regard that is not covered in the FSAR at this time is:

- (1) Details of the actual limited displacement break flow area and the actual break separation time at any circumferential break location are needed for this specific plant.
- (2) It must be demonstrated that St. Lucie plant analysis system parameters fall within the design envelope of CENPD-168, Revision 1.
- (3) Assurance should be provided that the criteria used to predict break location, as referenced to CENPD-168A, Revision 1 is used for reactor coolant system piping only. If this criteria is used for piping other than the RCS, additional justification must be provided.

Additional items not covered by CENPD-168, and which should be provided for the reactor coolant system of the plant are the pipe whip restraint parameters such as stiffness values and gap sizes.

The applicant has made a commitment to provide the following items in a future amendment to the FSAR.

- (1) High Energy pipe rupture analysis inside containment (Appendix 3.6A)
- (2) High Energy pipe rupture analysis outside containment (Appendix 3.6B)
- (3) Pipe whip restraints and break locations (Appendix 3.6C)
- (4) Structural details of the pipe whip restraints (Appendix 3.6D)
- (5) Main Steam and feedwater dynamic analysis (Appendix 3.6E).
- (6) Moderate Energy analysis (Appendix 3.6F)

Items (1) through (6) above will be considered to be open issues until this information has been submitted and reviewed by the staff.

II. SYSTEMS OTHER THAN RCS MAIN LOOP

The criteria for defining break and crack locations and configurations, the analytical methods used to define the forcing functions to verify the integrity and operability of mechanical components, component supports and piping systems are adequate and in compliance with Section 3.6.2 of the Standard Review Plan. However, the following additional information is required:

- (1) The FSAR Section 3.6.2.2 should be clarified to show that the requirement of $.8(S_h + S_A)$ is based on the sum of Equations (9) and (10) of paragraph NC-3652 of the ASME B&PV Code, Section III and not Equation (9) and (10) individually.
- (2) Provide criteria for postulated pipe breaks in both high and moderate energy piping systems in the containment penetration area.
- (3) Provide the basis for the $.8S_A$ criteria for expansion stresses which is stated in Section 3.6.2.2.2 (2) of the FSAR.
- (4) Provide a listing of the high energy systems that are considered for pipe rupture analysis. In addition provide a summary of the results of the analyses of these systems to demonstrate that essential systems, components, and supports will not be impaired as a result of high energy pipe breaks.
- (5) When longitudinal breaks are postulated, assurance must be provided that they are chosen in the location that is likely to cause the maximum damage.

Subject to resolution of the above open issues in Paragraphs 3.6.1.I and 3.6.1.II, our findings are as follows:

The applicant has proposed criteria for determining the location, type and effects of postulated pipe breaks in high energy piping systems and postulated pipe cracks in moderate energy piping systems. The applicant has used the effects resulting from these postulated pipe failures to evaluate the design of systems, components, and structures necessary to safely shut the plant down and to mitigate the effects of these postulated piping failures. The applicant has stated that pipe whip restraints, jet impingement barriers, and other such devices will be used to mitigate the effects of these postulated piping failures. We have reviewed these criteria and have concluded that they provide for a spectrum of postulated pipe breaks and pipe cracks which includes the most likely locations for piping failures, and that the types of breaks and their effects are conservatively assumed. We find that the methods used to design the pipe whip restraints provide adequate assurance that they will function properly in the event of a postulated piping failure. We further conclude that the use of the applicant's proposed pipe failure criteria in designing the systems, components, and structures necessary to safely shut the plant down and to mitigate the consequences of these postulated piping failures provides reasonable assurance of their ability to perform their safety function following a failure in high or moderate energy piping systems. The applicant's criteria comply with Standard Review Plan Section 3.6.2 and satisfy the applicable portions of General Design Criterion 4.

3.7.3 Seismic Subsystem Analysis

The review performed under Standard Review Plan Section 3.7.3 includes the applicant's dynamic analysis of all seismic Category I piping systems. In addition to operating transient loads, this analysis also considers abnormal loadings such as an earthquake.

At this time, the information in the FSAR is not adequate to verify that all the requirements of Section 3.7.3 of the Standard Review Plan for Seismic Subsystem Analysis have been met. Section 3.7.3 and the respective subsections which appear in the following paragraphs refer to the corresponding sections in the FSAR and are considered as open issues. The sections not discussed have

been found to have sufficient information and are sufficiently complete to meet the requirements of applicable sections of the SRP.

3.7.3.1 Seismic Analysis Methods

The following information is required as it pertains to the subsystem analysis before our review can be completed.

- (1) The method for determining that an adequate number of degrees of freedom were used in the dynamic modeling to determine the response of all Category I and applicable Non-Category I structures and plant equipment.
- (2) Justification that a sufficient number of modes were considered to assure participation of all significant modes.
- (3) The methods used to handle the relative displacements of Category I supports.
- (4) How significant effects such as piping interactions, externally applied structural restraints, hydrodynamic loads and nonlinear responses are accounted for.
- (5) If the equivalent static load method was used, justification must be provided that the system can be represented by a simple model and that the relative motion between support points is accounted for.

3.7.3.2 Determination of Number of Earthquake Cycles

No open issues.

3.7.3.3 Procedures Used for Analytical Modeling

The criteria and procedure given for the modeling of the seismic systems and the criteria for determining whether a component is analyzed as part of a system or independently requires amplification and inclusion of all information

required by the SRP. Before our review can be completed on this section, the criteria and procedures actually used must be described. This should include the modeling procedures used and the criteria for decoupling as outlined in SRP Section 3.7.2, paragraph III.3.

3.7.3.4 Basis for Selection of Frequencies

A discussion of the methods actually used in determining the fundamental frequencies is required in this FSAR Section. Also explain how the three ranges of equipment/support behavior (rigid, flexible, resonant) delineated are handled in the analysis. A statement or statements is required as to how these matters are considered in the analysis.

3.7.3.5 Use of Equivalent Static Load Method Analysis

Justification has been provided for the use of the equivalent static load method for piping systems. Similar justification is needed for all equipment for which this method was used. Also provide clarification on how the modified equivalent static load method differs from the equivalent static load method.

3.7.3.6 Three Components of Earthquake Motion

The loads corresponding to the three components of the ground motion are computed separately and the maximum co-directional responses are added by the square root of the sum of the squares (SRSS) method, as per Regulatory Guide 1.92 (Rev. 1), for obtaining combined response effects.

The approach for combining the three components of earthquake motion is satisfactory when the response spectra method of seismic analysis is used. Discuss the approach utilized for combining these components when the time history method of analysis is used.

3.7.3.7 Combination of Modal Responses

No open issues.

3.7.3.8 Analytical Procedures for Piping

Reference SER Sections 3.7.3.1, 3.7.3.5 and 3.7.3.9.

3.7.3.9 Multiple Supported Equipment & Components with Distinct Inputs

The criteria to be used in the analysis of multiple supported equipment and components meet the staff requirements as outlined in NRC Standard Review Plan 3.7.3 Section II-9 with the exception that a commitment be made to combine the support displacements in the most unfavorable combinations.

3.7.3.13 Intrraction of Other Piping with Seismic Category I Piping

This section concerning the interaction of other piping with seismic Category I piping adequately defines how these piping systems are handled when they are a part of the same system. However, information is required as to how Non-Category I piping systems are analyzed and/or isolated from Category I piping when the systems are in close proximity so that a failure of the Non-Category I piping would not damage the Category I piping.

3.7.3.14 Seismic Analysis of Reactor Internals

The discussion on the seismic analysis of the reactor internal structure, control element drive mechanism and control rod assemblies needs to be expanded in accordance with the requirements of Section II.1 and II.6 of SRP 3.7.2 concerning seismic analysis methods and the three components of earthquake motion respectively.

A description of the linear vertical analysis and nonlinear horizontal analysis is provided. Verify whether or not a vertical nonlinear analysis is used in the event that the linear vertical analysis indicates that the response of the core may be sufficiently large to lift off the core plate. In case it is used, provide a description of the analysis.



Provide a commitment that closely spaced modes are considered as per Regulatory Guide 1.92, in the analysis of the reactor internals and the core.

Upon resolution of the above open issues in Sections 3.7.3.1 through 3.7.3.14, we will report our findings in a supplement to the Safety Evaluation Report.

3.9 Mechanical Systems and Components

The review performed under Standard Review Plan Sections 3.9.1 through 3.9.6 pertains to the structural integrity and operability of various safety-related mechanical components in the plant. Our review is not limited to ASME Code components and supports, but is extended to other components such as control element drive mechanisms, certain reactor internals, ventilation ducting, cable trays, and any safety-related piping designed to industry standards other than the ASME Code. We review such issues as load combinations, allowable stresses, methods of analysis, summary of results, seismic qualification, preoperational testing, and inservice testing of pumps and valves. Our review must arrive at the conclusion that there is adequate assurance of a mechanical component performing its safety-related function under all postulated combinations of normal operating conditions, system operating transient, postulated pipe breaks, and seismic events. We have identified some open issues during our review. These issues are discussed in the appropriate paragraphs below.

3.9.1 Special Topics For Mechanical Components

The review performed under Standard Review Plan Section 3.9.1 pertains to the design transients, computer programs, experimental stress analysis and elastic-plastic analysis methods that were used in the analysis of seismic Category 1 ASME Code and non-Code items. The applicant has provided a complete list of transients to be used in the design and fatigue analysis of all Code Class 1 and CS components and of components supports and reactor internals within the reactor coolant pressure boundary. The number of events postulated for each transient has been included and is acceptable.

Computer programs were used in the analysis of specific components. A list of the computer programs used in the dynamic and static analyses to determine the structural and functional integrity of these components is included in the FSAR along with a brief description of each program. Design control measures, which are required by 10 CFR Part 50, Appendix B, require that verification of the computer programs be included. The applicant has provided verification for all of the listed computer programs.

The applicant has stated in the FSAR that experimental stress analysis methods were not utilized in the design of the St. Lucie, Unit 2 plant.

The criteria used in defining the applicable transients, computer codes and analytical methods used in the analyses of all seismic Category I ASME Code Class 1, 2, and 3 components, component supports, reactor internals, and other non-Code items provide assurance that the calculations of stresses, strains, and displacements for the above-noted items conform with the current state-of-the-art, are adequate for the design of these items and are in conformance with Standard Review Plan Section 3.9.1 and satisfy the applicable portions of General Design Criteria 2, 4, 14, and 15.

3.9.2 Dynamic Testing and Analysis of Systems, Components, and Equipment

The review performed under Standard Review Plan Section 3.9.2 pertains to the criteria, testing procedures, and dynamic analyses employed by the applicant to assure the structural integrity and operability of piping systems, mechanical equipment, reactor internals and their supports under vibratory loadings. This review is divided into three parts, each of which is discussed briefly below.

3.9.2.1 Piping Preoperational and Startup Testing Program

Piping vibration, thermal expansion, and dynamic effects testing will be conducted during the St. Lucie plant's preoperational and startup testing program. The purpose of these tests is to confirm that the piping, components, restraints, and supports have been designed to withstand the dynamic loadings and operational transient conditions that will be encountered during service



as required by the ASME Section III Code and to confirm that no unacceptable restraint of normal thermal motion occurs. We have identified the following open issues in our review. The issues are identified by sections of the FSAR.

Many of the items required by the Standard Review Plan (SRP), Section 3.9.2 are covered only briefly or not at all in this section. The SRP Acceptance Requirements II.1a through f and items a through d of the Review Procedures should be addressed before this FSAR Section can be considered acceptable. The staff requires a commitment to test all high energy piping and all seismic Category I moderate energy piping, including supports and restraints for thermal expansion, steady state vibration, dynamic and transient loads.

Subject to resolution of these open issues, our findings will be as follows:

The vibration, thermal expansion, and dynamic effects test program which will be conducted during startup and initial operation on specified high and moderate energy piping, and all associated systems, restraints and supports is an acceptable program. The tests provide adequate assurance that the piping and piping restraints of the system have been designed to withstand vibrational dynamic effects due to valve closures, pump trips, and other operating modes associated with the design basis flow conditions.

In addition, the tests provide assurance that adequate clearances and free movement of snubbers exist for unrestrained thermal movement of piping and supports during normal system heatup and cooldown operations. The planned tests will develop loads similar to those experienced during reactor operation. This test program complies with Standard Review Plan Section 3.9.2 and constitutes an acceptable basis for fulfilling, in part, the requirements of General Design Criteria 14 and 15.

3.9.2.2 Dynamic Analysis of Reactor Internals

Dynamic system analyses should be performed to confirm the structural design adequacy and ability, with no loss of function, of the reactor internals and

unbroken loops of the reactor coolant piping to withstand the loads from a loss-of-coolant accident (LOCA) in combination with the SSE. Our review covers the methods of analysis, the considerations in defining the mathematical models, and descriptions of the forcing function, the calculational scheme, the acceptance criteria, and the interpretation of analytical results.

The blowdown analysis requires further amplification and clarification. Specifically, the staff will require that the applicant (1) justify decoupling of the horizontal and vertical components of the responses to blowdown loads, (2) justify the use of results of linear analyses for the inherent nonlinear problem; and (3) present a discussion outlining the effects of system flow upon mass and flexibility properties.

Subject to resolution of the open issues, our findings are as follows:

The dynamic system analysis performed by the applicant provides an acceptable basis for confirming the structural design adequacy of the reactor internals and unbroken piping loops to withstand the combined dynamic loads of a postulated loss of coolant accident (LOCA) and the safe shutdown earthquake (SSE). The analysis provides adequate assurance that the combined stresses and strains in the components of the reactor coolant system and reactor internals do not exceed the allowable stress and strain limits for the materials of construction, and that the resulting deflections or displacements at any structural element of the reactor internals will not distort the reactor internals geometry to the extent that core cooling may be impaired. The methods used for component analysis have been found to be compatible with those used for the systems analysis. The proposed combinations of component and system analyses are, therefore, acceptable. The assurance of structural integrity of the reactor internals under combined LOCA and SSE conditions provides added confidence that the design will withstand a spectrum of lesser pipe breaks and seismic events. Accomplishment of the dynamic system analysis constitutes an acceptable basis for complying with Standard Review Plan Section 3.9.2 and for satisfying the applicable requirements of General Design Criteria 2 and 4.



3.9.2.3 Preoperational Flow-Induced Vibration Testing of Reactor Internals

Flow-induced vibration testing of reactor internals should be conducted during the preoperational and start-up test program. The purpose of this test is to demonstrate that flow-induced vibrations similar to those expected during operation will not cause unanticipated flow-induced vibrations of significant magnitude or structural damage. The Maine Yankee and Fort Calhoun precritical vibration monitoring programs together constitute a valid prototype design for St. Lucie Unit 2. The St. Lucie Unit 2 reactor has been designated as non-prototype seismic Category 1 design. The applicant is proceeding to implement a preoperational vibration monitoring program for St. Lucie Unit 2 consistent with the recommendations of Regulatory Guide 1.20 as it relates to nonprototype seismic Category I Units. A summary of the significant hydraulic and structural design parameters for the Maine Yankee, Fort Calhoun, and St. Lucie, Unit 2 plants have been provided in the FSAR. The reactor vessel internals of St. Lucie will be subjected during the preoperational and functional testing program to the significant flow modes of normal reactor operation for a sufficient period of time to determine whether the reactor vessel internals exhibit any unexpected vibration problems. We find this program acceptable provided the applicant submits a correlation of the St. Lucie Unit 2 observed vibrational characteristics with the results from the prototype reactors. If the comparison of the observed vibrational characteristics of St. Lucie with those of the prototype plants indicate the need for any corrective action, the staff will review the applicant's proposed corrective action for St. Lucie Unit 2 and provide its evaluation in a supplement to this SER.

Subject to resolution of these open issues, our findings are as follows:

The preoperational vibration program planned for the reactor internals provides an acceptable basis for verifying the design adequacy of these internals under test loading conditions comparable to those that will be experienced during operation. The combination of tests, predictive analysis, and post-test inspection provide adequate assurance that the reactor internals will, during their service lifetime, withstand the flow-induced vibrations of reactor operation without loss of structural integrity. The integrity of the reactor

internals in service is essential to assure the proper positioning of reactor fuel assemblies and unimpaired operation of the control rod assemblies to permit safe reactor operation and shutdown. The conduct of the preoperational vibration tests is in conformance with the provisions of Regulatory Guide 1.20 and Standard Review Plan, Section 3.9.2 and satisfies the applicable requirements of General Design Criteria 1 and 4.

3.9.3 ASME Code Class 1, 2, and 3 Components, Component Supports, and Core Support Structures

Our review under Standard Review Plan Section 3.9.3 is concerned with the structural integrity of pressure-retaining components, their supports, and core support structures which are designed in accordance with the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section III, or earlier industry standards. This review is divided into four parts, each of which is discussed briefly below.

3.9.3.1 Loading Combinations, Design Transients and Stress Limits

For Section 3.9.3 of the FSAR to be acceptable, the following issues need to be resolved.

- (1) The discussion of plant conditions in Section 3.9.3.1 of the FSAR requires clarification. The loading combinations method of response combination and allowable limits should be provided for all ASME Class 1, 2 and 3 components and their supports for each design and service condition.
- (2) The methods of combining responses to the various loads listed in Sections 3.9.3.1 of the FSAR are not defined. We will require a description of the methods used for the combinations of responses to all dynamic loads for all NSSS and BOP supplied ASME Class 1, 2, and 3 equipment, components and their supports. Our position on this issue is outlined in NUREG-0484, "Methodology for Combining Dynamic Responses," Revision 1 dated May, 1980.

- (3) The response of certain reactor coolant system components and their supports to postulated asymmetric LOCA loads needs to be addressed in accordance with NUREG-0609.
- (4) Provide stress limits and criteria to limit deformation and assure functional capability for Class 2 and 3 austenitic pipe bend and elbows.

We have contracted with the Brookhaven National Laboratory to perform an independent analysis of a sample piping system in the St. Lucie, Unit No. 2 plant. This analysis will not only verify that the sample piping system meets the applicable ASME Code requirements, but will also provide a check on the applicant's ability to correctly model and analyze its piping systems. The results of the above evaluations will be presented in a future supplement to this report.

Subject to resolution of the above open issues, our findings are as follows:

The specified design and service combinations of loadings as applied to ASME Code Class 1, 2 and 3 pressure retaining components in systems designed to meet seismic Category I standards are such as to provide assurance that, in the event of an earthquake affecting the site or other service loadings due to postulated events or system operating transients, the resulting combined stresses imposed on system components will not exceed allowable stress and strain limits for the materials of construction. Limiting the stresses under such loading combinations provides a conservative basis for the design of system components to withstand the most adverse combination of loading events without loss of structural integrity. The design and service load combinations and associated stress and deformation limits specified for ASME Code Class 1, 2, and 3 components comply with Standard Review Plan Section 3.9.3 and satisfy the applicable portions of General Design Criteria 1, 2, and 4.

3.9.3.2 Design and Installation of Pressure Relief Devices

The design and installation criteria applicable to the mounting of pressure relief devices (safety and relief valves) for the overpressure protection of

ASME Class 1, 2, and 3 components are reviewed. To be acceptable, the following issues need to be resolved:

- (1) Section 3.9.3.3 of the FSAR should include a more detailed description of the calculation procedures, which were used in the parametric studies for closed discharge systems.
- (2) Information should be provided in Section 3.9.3.3 of the FSAR relating to the various design and service loading conditions and combinations thereof, and the corresponding stress criteria used in the design for the mounting of pressure relief valves.
- (3) The method of evaluating the structural response of the piping and support system stiffness in the dynamic analysis of these mountings should be discussed in Section 3.9.3.3 of the FSAR.

Based upon our review of FSAR Section 3.9.3.3 and contingent upon the satisfactory resolution of the above open items, our findings will be as follows:

The criteria used in the design and installation of ASME Class 1, 2, and 3 safety and relief valves provide adequate assurance that, under discharging conditions, the resulting stresses will not exceed allowable stress and strain limits for the materials of construction. Limiting the stresses under the loading combinations associated with the actuation of these pressure relief devices provides a conservative basis for the design and installation of the devices to withstand these loads without loss of structural integrity or impairment of the overpressure protection function. The criteria used for the design and installation of ASME Class 1, 2, and 3 overpressure relief devices constitute an acceptable basis for meeting the applicable requirements of General Design Criteria 1, 2, 4, 14, and 15 and are consistent with those specified in Regulatory Guide 1.67 and Standard Review Plan Section 3.9.3.

3.9.3.4 Component Supports

The review of information submitted by the applicant includes an evaluation of Code Class 1, 2, and 3 component supports. The review includes an assessment of the design and structural integrity of the supports and their evaluation.

Both BOP and NSSS supplied supports for Code Class 1, 2, and 3 component supports which were procured prior to July 1, 1974 were designed in accordance with the criteria in Section 3.9.3.4 of the FSAR. Component supports procured after July 1974 will comply with the requirements of ASME Section III Subsection NF "Component Supports." Since the criteria in the FSAR may not be as conservative as that in Subsection NF, we require more information for those supports procured prior to July 1, 1974. A discussion which demonstrates that those components designed to the FSAR criteria have an adequate margin of safety should be submitted in the FSAR. In addition, the applicant should verify that the allowable stresses of MSS-SP-58, "Pipe Hangers and Supports" are used without the addition of a shape factor to account for bending stresses.

Provide in a tabular form for both BOP and NSSS Code Class 1, 2 and 3 component supports the load combinations, stress limits for various plant conditions.

Provide the allowable buckling limits for ASME Class 1 linear and plate and shell type component supports subjected to faulted condition load. Also provide additional information concerning the design of support bolts and bolted connections.

In addition, assurances must be provided that stresses due to thermal expansion, thermal shock and differential support movements have been included.

We will also require an acceptable response to our request for preservice inspection and testing information on snubbers.

Subject to resolution of the above open issues, our findings are as follows:

The specified design and service loading combinations used for the design of ASME Code Class 1, 2, and 3 component supports in systems classified as seismic Category I provide assurance that, in the event of an earthquake or other service loadings due to postulated events or system operating transients, the resulting combined stresses imposed on system components will not exceed allowable stress and strain limits for the materials of construction. Limiting the stresses under such loading combinations provides a conservative basis for the design of support components to withstand the most adverse combination of loading events without loss of structural integrity or supported component operability. The design and service load combinations and associated stress and deformation limits specified for ASME Code Class 1, 2, and 3 component supports comply with Standard Review Plan Section 3.9.3 and satisfy the applicable portions of General Design Criteria 1, 2, and 4.

3.9.4 Control Element Drive Mechanisms

Our review under Standard Review Plan Section 3.9.4 covered the design of the hydraulic control rod drive system up to its interface with the control rods. We reviewed the analyses and tests performed to assure the structural integrity and operability of this system during normal operation and under accident conditions. We also reviewed the life-cycle testing performed to demonstrate the reliability of the control rod drive system over its 40 year life.

The review indicates that additional information is required on the design criteria for the nonpressurized components. The thermal deflection problem of dissimilar materials is not covered and there is no information as to the allowable and actual deflections due to the various loading conditions. Design margins for stress, deformation, and fatigue should be presented and should be shown to be equal to or greater than those of other plants of similar design having a period of successful operation.

Subject to resolution of the above open issues, our findings are as follows:

The design criteria and the testing program conducted in verification of the mechanical operability and life cycle capabilities of the control rod drive

system are in conformance with Standard Review Plan Section 3.9.4. The use of these criteria provide reasonable assurance that the system will function reliably when required, and form an acceptable basis for satisfying the mechanical reliability stipulations of General Design Criterion 27.

3.9.5 Reactor Pressure Vessel Internals

Our review under Standard Review Plan (SRP) Section 3.9.5 is concerned with the load combinations, allowable stress limits, and other criteria used in the design of the St. Lucie Unit 2 reactor internals.

In addition to the issues discussed in Section 3.9.2.4 of this Safety Evaluation Report, resolution of the following issues is also required:

- (1) Verify that the Control Element Assemblies (CEA's) can be inserted for an inlet break and the reaction can be stopped for an outlet break.
- (2) Identify the highest usage factor and the location where it occurs in the reactor internals.

Subject to resolution of these issues, our findings are as follows:

The specified transients, design and service loadings, and combinations of loadings as applied to the design of the St. Lucie Unit 2 reactor internals provide reasonable assurance that in the event of an earthquake or of a system transient during normal plant operation, the resulting deflections and associated stresses imposed on these reactor internals would not exceed allowable stresses and deformation limits for the materials of construction. Limiting the stresses and deformations under such loading combinations provides an acceptable basis for the design of these reactor internals to withstand the most adverse loading events which have been postulated to occur during service lifetime without loss of structural integrity or impairment of function. The design procedures and criteria used by the applicant in the design of the St. Lucie Unit 2 reactor internals comply with Standard Review Plan

Section 3.9.5 and constitute an acceptable basis for satisfying the applicable requirements of General Design Criteria 1, 2, 4, and 10.

3.9.6 Inservice Testing of Pumps and Valves

In Section 3.9.3 of this Safety Evaluation Report, we discussed the design and operability of safety-related pumps and valves in the St. Lucie Unit 2. The design of these pumps and valves is intended to demonstrate that they will be capable of performing their safety function (open, close, start, etc.) at any time during the plant life. However, to provide added assurance of the reliability of these components, the applicant will periodically test all its safety-related pumps and valves. These tests are performed in general accordance with the rules of Section XI of the ASME Code. These tests verify that these pumps and valves operate successfully when called upon.

Additionally, periodic measurements are made of various parameters and compared to baseline measurements in order to detect long-term degradation of the pump or valve performance. Our review under Standard Review Plan Section 3.9.6 covers the applicant's program for preservice and inservice testing of pumps and valves. We give particular attention to those areas of the test program for which the applicant requests relief from the requirements of Section XI of the ASME Code.

The information presented in Section 3.9.6 of the FSAR does not contain sufficient detail to demonstrate how the applicant intends to implement the inservice testing of pumps and valves requirements of ASME Section XI, "Rule for Inservice Inspection of Nuclear Power Plant Components." We will require a description of the applicant's proposed program on this subject. Guidelines on the type of information that we require is contained in Attachment 1 of this SER.

In addition, we will require an acceptable response to our request for additional information on periodic leak testing of pressure isolation valves.

We will report on the resolution of these issues in a supplement to this SER.

ATTACHMENT 1

NRC STAFF COMMENTS ON INSERVICE PUMP AND VALVE TESTING PROGRAMS AND RELIEF REQUESTS

The NRC staff, after reviewing a number of pump and valve testing programs, has determined that further guidance might be helpful to illustrate the type and extent of information we feel is necessary to expedite the review of these programs. We feel that the Licensee can, by incorporating these guidelines into each program submittal, reduce considerably the staff's review time and time spent by the Licensee in responding to NRC staff requests for additional information.

The pump testing program should include all safety related* Class 1, 2, and 3 pumps which are installed in water cooled nuclear power plants and which are provided with an emergency power source.

The valve testing program should include all the safety related valves in the following systems excluding valves used for operating convenience only, such as manual vent, drain, instrument, and test valves, and valves used for maintenance only.

PWR

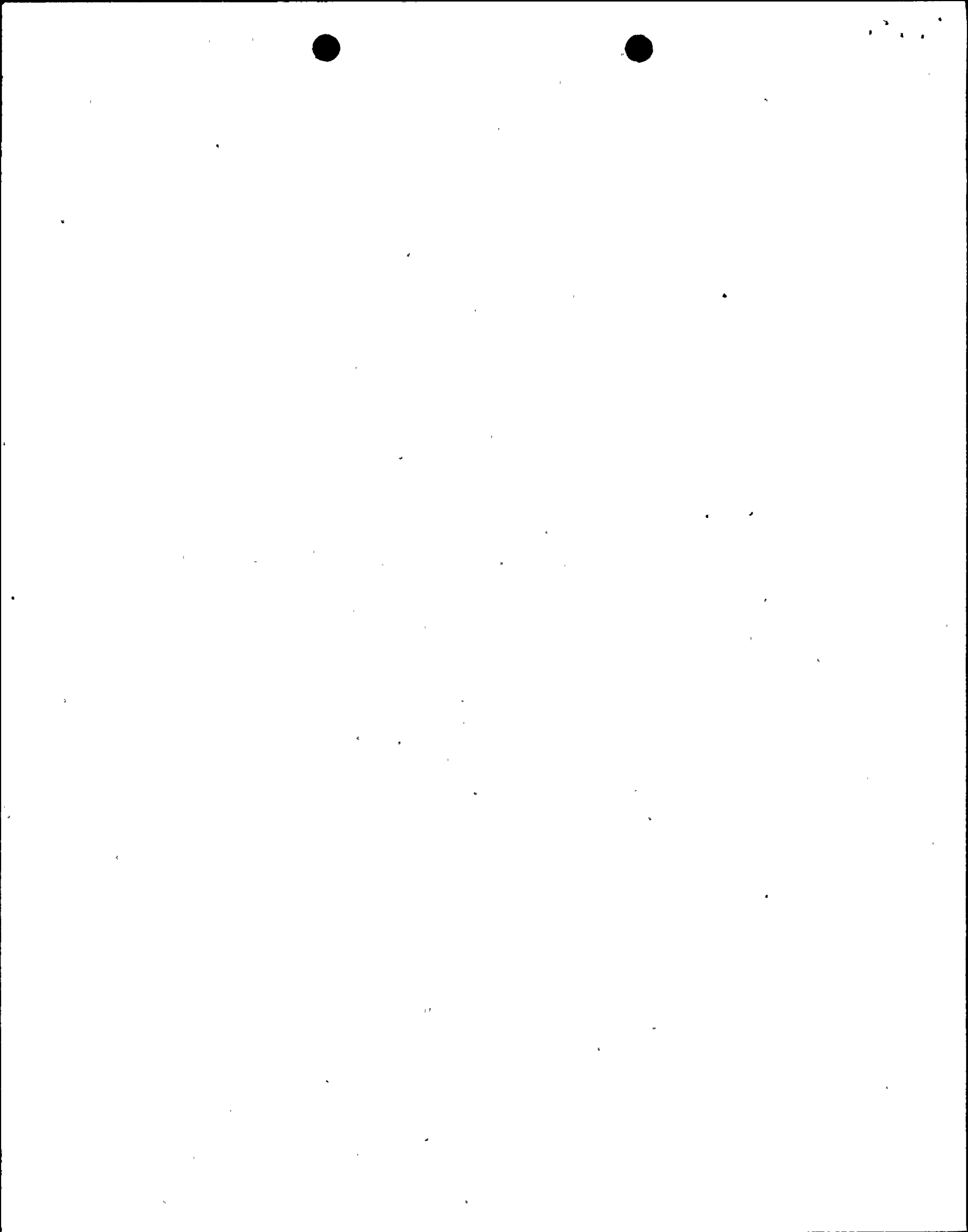
- a. High Pressure Injection System
- b. Low Pressure Injection System
- c. Accumulator Systems
- d. Containment Spray System

*Safety related - necessary to safely shut down the plant, mitigate the consequences of an accident and maintain the plant in a safe shutdown condition.

- e. Primary and Secondary System Safety and Relief Valves
- f. Auxiliary Feedwater Systems
- g. Reactor Building Cooling System
- h. Active Components in Service Water and Instrument Air Systems
which are required to support safety system functions.
- i. Containment Isolation Valves required to change position to isolate
containment.
- j. Chemical & Volume Control System
- k. Other key components in Auxiliary Systems which are required to directly
support plant shutdown or safety system function.
- l. Residual Heat Removal System
- m. Reactor Coolant System

BWR

- a. High Pressure Core Injection System
- b. Low Pressure Core Injection System
- c. Residual Heat Removal System (Shutdown Cooling System)
- d. Emergency Condenser System (Isolation Condenser System)
- e. Low Pressure Core Spray System
- f. Containment Spray System
- g. Safety, Relief, and Safety/Relief Valves
- h. RCIC (Reactor Core Isolation Cooling) System
- i. Containment Cooling System
- j. Containment isolation valves required to change position to isolate
containment.



- k. Standby liquid control system (Boron System)
- l. Automatic Depressurization System (any pilot or control valves, associated hydraulic or pneumatic systems, etc.)
- m. Control Rod Drive Hydraulic System ("Scram" function)
- n. other key components in Auxiliary Systems which are required to directly support plant shutdown or safety system function.
- o. Reactor Coolant System

Inservice Pump and Valve Testing Program

- I. Information required for NRC Staff Review of the Pump and Valve Testing Program
 - A. Three sets of P&ID's, which include all of the systems listed above, with the code class and system boundaries clearly marked. The drawings should include all of the components present at the time of submittal and a legend of the P&ID symbols.
 - B. Identification of the applicable ASME Code Edition and Addenda
 - C. The period for which the program is applicable.
 - D. Identify the component code class.
 - E. For Pump testing: Identify
 - 1. Each pump required to be tested (name and number)
 - 2. The test parameters to be measured
 - 3. The test frequency

F. For valve testing: Identify

1. Each valve in ASME Section XI Categories A & B that will be exercised every three months during normal plant operation (indicate whether partial or full stroke exercise, and for power operated valves list the limiting value for stroke time.)
2. Each valve in ASME Section XI Category A that will be leak tested during refueling outages (Indicate the leak test procedure you intend to use)
3. Each valve in ASME Section XI Categories C, D, and E that will be tested, the type of test and the test frequency. For check valves, identify those that will be exercised every 3 months and those that will only be exercised during cold shutdown or refueling outages.

II. Additional Information that will be Helpful in Speeding Up the Review Process

- A. Include the valve location coordinates or other appropriate location information which will expedite our locating the valves on the P&IDs.
- B. Provide P&ID drawings that are large and clear enough to be read easily.
- C. Identify valves that are provided with an interlock to other components and a brief description of that function.

Relief Requests from Section XI Requirements

The largest area of concern for the NRC staff, in the review of an inservice valve and pump testing program, is in evaluating the basis for justifying relief from Section XI Requirements. It has been our experience that many requests for relief, submitted in these programs, do not provide adequate descriptive and detailed technical information. This explicit information is necessary to provide reasonable assurance that the burden imposed on the licensee in complying with the code requirements is not justified by the increased level of safety obtained.

Relief requests which are submitted with a justification such as "Impractical", "Inaccessible", or any other categorical basis, will require additional information, as illustrated in the enclosed examples, to allow our staff to make an evaluation of that relief request. The intention of this guidance is to illustrate the content and extent of information required by the NRC staff, in the request for relief, to make a proper evaluation and adequately document the basis for that relief in our safety evaluation report. The NRC staff feels that by receiving this information in the program submittal, subsequent requests for additional information and delays in completing our review can be considerably reduced or eliminated.

I. Information Required for NRC Review of Relief Requests

A. Identify component for which relief is requested:

1. Name and number as given in FSAR
2. Function
3. ASME Section III Code Class
4. For valve testing, also specify the ASME Section XI valve category as defined in IWX-2000

- B. Specifically identify the ASME Code requirement that has been determined to be impractical for each component.
 - C. Provide information to support the determination that the requirement in (B) is impractical; i.e., state and explain the basis for requesting relief.
 - D. Specify the inservice testing that will be performed in lieu of the ASME Code Section XI requirements.
 - E. Provide the schedule for implementation of the procedure(s) in (D).
- II. Examples to Illustrate Several Possible Areas Where Relief May Be Granted and the Extent and Content of Information Necessary to Make An Evaluation
- A. Accessibility: The regulation specifically grants relief from the code requirement because of insufficient access provisions. However, a detailed discussion of actual physical arrangement of the component in question to illustrate the insufficiency of space for conducting the required test is necessary.

Discuss in detail the physical arrangement of the component in question to demonstrate that there is not sufficient space to perform the code required inservice testing.

What alternative surveillance means which will provide an acceptable level of safety have you considered and why are these means not feasible?

- B. Environmental Conditions (e.g., High radiation level, High temperature, High humidity, etc.)

Although it is prudent to maintain occupation radiation exposure for inspection personnel as low as practicable, the request for relief from the code requirements cannot be granted solely on the basis of high radiation levels alone. A balanced judgment between the hardships and compensating increase in the level of safety should be carefully established. If the health and safety of the public dictates the necessity of inservice testing, alternative means or even decontamination of the plant if necessary should be provided or developed.

Provide additional information regarding the radiation levels at the required test location. What alternative testing techniques which will provide an acceptable level of assurance of the integrity of the component in question have you considered and why are these techniques determined to be impractical?

C. Instrumentation is not originally provided

Provide information to justify that compliance with the code requirements would result in undue burden or hardships without a compensating increase in the level of plant safety. What alternative testing methods which will provide an acceptable level of safety have you considered and why are these methods determined to be impractical?

D. Valve Cycling During Plant Operation Could Put the Plant in an Unsafe Condition.

The licensee should explain in detail why exercising tests during plant operation could jeopardize the plant safety.

E. Valve Testing at Cold Shutdown or Refueling Intervals in Lieu of the 3 Month Required Interval

The licensee should explain in detail why each valve cannot be exercised during normal operation. Also, for the valves where a refueling interval is indicated, explain in detail why each valve cannot be exercised during cold shutdown intervals.

III. Acceptance Criteria for Relief Request

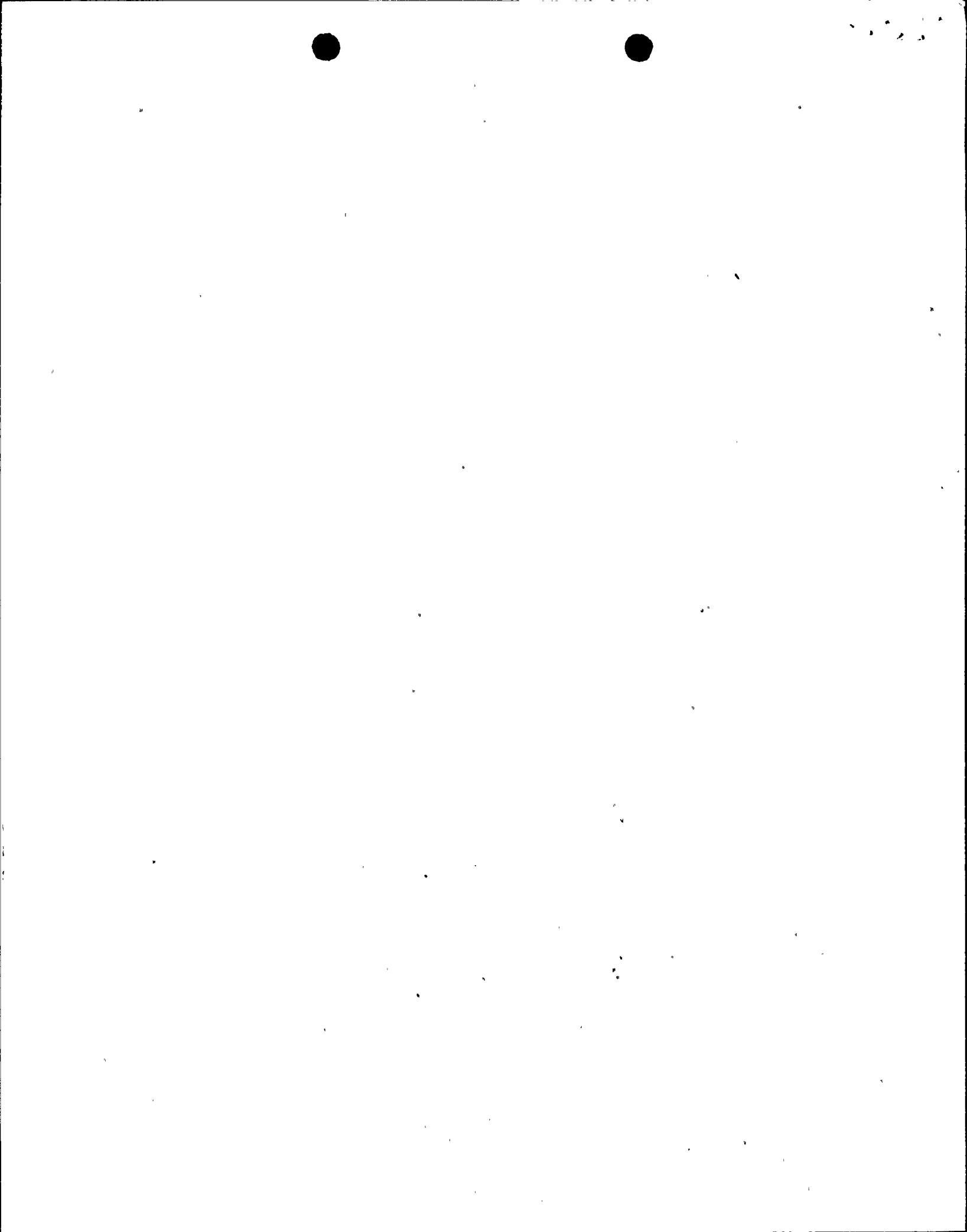
The Licensee must successfully demonstrate that:

1. Compliance with the code requirements would result in hardships or unusual difficulties without a compensating increase in the level of safety and noncompliance will provide an acceptable level of quality and safety, or

2. Proposed alternatives to the code requirements or portions thereof will provide an acceptable level of quality and safety.

Standard Format

A standard format, for the valve portion of the pump and valve testing program and relief requests, is included as an attachment to this Guidance. The NRC staff believes that this standard format will reduce the time spent by both the staff in our review and by the licensee in their preparation of the pump and valve testing program and submittals. The standard format includes examples of relief requests which are intended to illustrate the application of the standard format and are not necessarily a specific plant relief request.



ATTACHMENT

STANDARD FORMAT

VALVE INSERVICE TESTING PROGRAM SUBMITTAL

Valve Number	Class	Coordinates	Valve Category					Size (inches)	Valve Type	Actuator Type	Normal Position	Test Requirements	Relief Requests*	Testing Alternative
			A	B	C	D	E							
710	3	D-14					X	4	GA	M	LO	ET		
700	3	D-15				X		6	DE	NA	C	DT		
717	3	C-15			X			16	CK	SA	-	CV	X	CS
702C	3	C-15			X			16	CK	SA	-	CV		
707	3	E-14			X			3	REL	SA	-	CV		
834	3	D-11		X				4	GL	M	C	Q	X	ET
												MT		
722B	3	B-11			X			3/4	REL	SA	-	SRV		60 sec.
722C	3	B-11			X			3/4	REL	SA	-	SRV		
715	2	A-10			X			3	REL	SA	-	SRV		
729	2	B-10			X			3	REL	SA	-	SRV		
744B	2	D-14	X					10	GA	MO	C	Q		
												LT	X	
												MT		30 sec.

REMARKS
(Not to be used for relief basis)

60 sec.

30 sec.

Legend for Valve Testing Example Format

- Q - Exercise valve (full stroke) for operability every (3) months
- LT - Valves are leak tested per Section XI Article IW-3420
- MT - Stroke time measurements are taken and compared to the stroke time limiting value per Section XI Article IW-3410
- CV - Exercise check valves to the position required to fulfill their function every (3) months
- SRV - Safety and relief valves are tested per Section XI Article IW-3510
- DT - Test category D valves per Section XI Article IW-3600
- ET - Verify and record valve position before operations are performed and after operations are completed, and verify that valve is locked or sealed.
- CS - Exercise valve for operability every cold shutdown
- RR - Exercise valve for operability every reactor refueling

Relief Request Basis

System: Auxiliary Coolant System, Component Cooling

1. Valve: 717
Category: C
Class: 3
Function: Prevent backflow from the reactor coolant pump cooling coils

Impractical

test requirement: Exercise valve for operability every three months

Basis for relief: To test this valve would require interruption of cooling water to the reactor coolant pumps motor cooling coils. This action could result in damage to the reactor coolant pumps and thus place the plant in an unsafe mode of operation.

Alternative This valve will be exercised for operability.

Testing: during cold shutdowns.

2. Valve: 834
Category: 8-E
Class: 3
Function: Isolate the primary water from the component cooling surge tank during plant operation. It is normally in the closed position, but routine operation of this valve will occur during refueling and cold shutdowns.

Impractical Test Exercise valve (full stroke) for operability

Requirement: every three (3) months.

Basis for Relief: This valve is not required to change position during plant operation to accomplish its safety function. Exercising this valve will increase the possibility of surge tank line contamination.

Alternate Testing: Verify and record valve position before and after each valve operation.

3. Valve: 744B
Category: A
Class: 2
Function: Isolate the residual heat exchangers from the cold leg R.C.S. backflow and accumulator backflow.

Test Requirements: Seat leakage test

Basis for Relief: This valve is located in a high radiation field (2000 mr/hr) which would make the required seat leakage test hazardous to test personnel. We intend to seat leak test two other valves (875B and 876B) which are in series with this valve and will also prevent backflow. We feel that by complying with the seat leakage requirements we will not achieve a compensatory increase in the level of safety.

Alternative Testing: No alternative seat leak testing is proposed.

200-1000
100-1000

JUN 15 1981