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

June 7, 1990

MEMORANDUM FOR: Thomas E. Murley, Director
Office of Nuclear Reactor Regulation

FROM: Martin G. Malsch
Deputy General Counsel for
Licensing & Regulation

SUBJECT: DEGREE OF DESIGN DETAIL REQUIRED FOR REACTOR DESIGN CERTIFICATION

This memo describes the pertinent requirements of 10 CFR Part 52 on the level of design detail required for reactor design certification. It is important at the outset to distinguish among three different kinds of requirements: the level of design detail required in the application for design certification; the level of design detail required to be available to NRC, if requested, during NRC's review of an application for design certification; and the level of design detail actually approved in the certification itself. As will be seen below, the rule addresses the first two, but is essentially silent on the third. This memo addresses only the level of design detail; Part 52 has separate provisions regarding the scope of design (e.g., inclusion of balance of plant).

1. Design Detail in a Certification Application

This is addressed specifically in 10 CFR § 52.47, principally 52.47(a)(1)(i), 52.47(a)(1)(v) and 52.47(a)(2). The first provision requires that an application for a design certification include:

> The technical information which is required of applicants for construction permits and operating licenses by 10 CFR Part 20, Part 50 and its appendices, and Parts 73 and 100, and which is technically relevant to the design and not site-specific;

We read this to say that a design certification application must include a level of detail that would satisfy the regulatory requirements for technical information in an FSAR, except to the extent that particular FSAR requirements are technically irrelevant (e.g., requirements for light water reactors may not apply to heavy water reactors), or are site specific. We understand that FSARs have been frequently amended during the course of OL review and a question has arisen whether the requirement in Section 52.47(a)(1)(i) refers to the FSAR as originally docketed or to the FSAR as amended just prior to licensing. The rule is silent on this question of interpretation. However, the level of detail must

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be such as to permit NRC to find that the regulations' technical information requirements for an FSAR are satisfied. Depending on completeness, this could be the FSAR as docketed or as later amended.

Section 50.47(a)(1)(v) requires that an application include a "design-specific probabilistic risk assessment". This means that the degree of design detail must be such as to permit a design-specific PRA to be completed.

Section 52.47(a)(2) provides that:

> The application must contain a level of design information sufficient to enable the Commission to judge the applicant's proposed means of assuring that construction conforms to the design and to reach a final conclusion on all safety questions associated with the design before the certification is granted. The information submitted for a design certification must include performance requirements and design information sufficiently detailed to permit the preparation of acceptance and inspection requirements by the NRC, and procurement specifications and construction and installation specifications by an applicant.

We read this to say three separate things about the level of design detail. First, the level of detail must permit NRC to reach a final conclusion on all safety questions associated with the design prior to certification. Second, the level of detail must be such as to allow the NRC to judge the applicant's proposed means of assuring that construction conforms to the design. This should probably be read along with Section 52.47(a)(1)(vi) which requires that the application include proposed inspections, tests, analyses, and acceptance criteria (ITAACS). So read, this requirement probably means only that the ITAACS must mesh or dovetail with the proposed design.

Third, this paragraph requires a level of detail sufficient to "permit the preparation of acceptance and inspection requirements by NRC and procurement specifications and construction and installation specifications by an applicant".1/

1/The rule preamble expresses what is meant by "procurement construction and installation specifications" as follows:

> Procurement specifications would have to identify the equipment and material performance requirements and include the necessary codes, standards, and other acceptance and performance criteria to which the equipment and materials will be fabricated and tested. Construction and installation specifications would have to identify the criteria and methods by which systems, structures and components are erected or installed in the facility and include acceptance, performance, inspection and testing requirements and criteria.

This language is similar to the proposed rule which required "performance requirements and design specifications sufficiently detailed to permit the preparation of procurement specifications and acceptance and inspection requirements". The Commission paper accompanying the draft proposed rule (SECY-88-169) characterized this as "stringent demands on the level of detail required in an application for design certification.

We believe that Section 52.47(a)(2), properly read, requires a level of detail in the application such that the application _itself_ would essentially permit the preparation of the specifications prior to certification. Otherwise the paragraph becomes meaningless as an application requirement, since any level of detail in a certification application, no matter how meager, can be _used_ to prepare specifications if there is no limit on the amount of design information, outside of the application and developed subsequently, which can be considered as well. However, some technical judgment will be required to decide whether specifications, as defined in the Part 52 preamble (see note) could reasonably be developed from the information in a given design certification.

In sum, the level of design detail required by Part 52 for a certification _application_ must be such that:

- NRC can find that the application meets the relevant technical information requirements for an FSAR, except for site-specific matters;

- a design specific PRA can be completed;

- NRC can reach a final conclusion on all safety questions;

- NRC can prepare acceptance and inspection requirements, and the applicant can prepare procurement and construction and installation specifications. This requirement should be met without recourse to any significant design information outside the application; and

- the design meshes or dovetails with the ITAACs.

I agree

2. _Design Detail to be Available if Requested by NRC_

10 CFR 52.47(a)(2) provides that:

> The Commission will require, prior to design certification, that information normally contained in certain procurement specifications and construction and installation specifications be completed and available for audit if such information is necessary for the Commission to make its safety determination.

This is largely self-explanatory. It does permit the inference that other parts of the rule could in particular cases require that actual procurement specifications be reviewed by NRC.[2]

3. Design Detail In a Certification

Part 52 is silent on the degree of detail contemplated in the certification or design approval itself, even though it is the certification itself, rather than the certification application, that controls the degree of standardization achieved by the regulation.[3] The rule preamble states specifically that "just how much [detail] is present will be an issue which will have to be resolved in each certification rulemaking".

There were two principal reasons for the rule's silence on this important issue. First, certification is to be by rule and rules must be public. Since some of the design details in the application were expected to be proprietary, it was clear that the entire application could not be certified, but how much was not clear. Second, NRC had never drafted a certification rule and it was thought unwise to be prescriptive as to what the certification must contain.

However, several inferences might be drawn from the rule Part 52 rulemaking. First, obviously the degree of detail in the certification cannot be greater (but may be less) than the application itself. Second, the rule preamble discusses and rejects a NUMARC comment that the certification itself should include a provision like 10 CFR 50.59 which would give utilities the latitude to make changes in the design without prior NRC approval, on the ground that the certified design "is likely to encompass roughly the same design features that § 50.59 prohibits changing without prior NRC approval". The preamble further states that the level of detail "should afford licensees the opportunity to take advantage of improvements in equipment".

(Seems to indicate Com'n's intent not to permit a 50.59 type provision in cert. rule.)

In sum, the level of detail in the approved design or certification itself is to be decided on a case-by-case basis and there is substantial leeway for judgment depending on the degree of standardization desired. However, the degree of standardization:

- cannot be greater than the level of detail in the certification application.

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[2] In this regard, it should be recognized that Part 52 as a whole indicates clearly that standardization has a safety benefit.

[3] All of the provisions of the rule on finality and limitations on design changes by vendors, utilities or NRC are keyed to the design as certified, not to the design as submitted in the application.

- was anticipated to allow some flexibility in making improvements in equipment.
- was anticipated to allow some flexibility in allowing changes which do not present unreviewed safety questions (Section 50.59).

Martin G. Malsch
Deputy General Counsel for
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SUGGESTED QUESTIONS
BRIEFING ON LEVEL OF DESIGN DETAIL - PART 52
DECEMBER 7, 1990

1. Are the present standards for changes allowable under Part 50.59 acceptable for the types of changes envisioned for evolutionary or advanced plants: (1) Between design certification and COL, (2) after operation begins? That is, how will the standards deal with the severe accident insights?

2. Should the staff be notified about 50.59-type changes between COL and authorization for operation because of the potential for a hearing?

3. How would risks that are projected through the design-specific PRA for the facility be maintained through the life of the plant?

4. Given technical controversy about severe accident phenomena and issues, will the level of detail suggested by the staff assure that the issues raised in SECY-90-016 will be resolved? For instance, the staff's position, approved by the Commission, on hydrogen control is that 100% oxidation of the cladding surrounding the active fuel is to be assumed and planned for. How will the suggested level of detail allow the staff to be assured that three times that amount, that is, oxidation of all the zirconium in a BWR, will not have to be dealt with?

5. Can you clarify how Tier 2 information and the warehouse will interface with ITAAC? I believe that staff should view the implementation of ITAAC as confirmatory only. Design issues should not be left "open" at Design Certification with the expectation that ITAAC implementation will resolve those issues.

6. What is the schedule for the draft of the Reg Guide to be issued for public comment? **We expect an answer of nine months, with a final of one year.** From reading the SECY paper, it seems that the staff has done a lot of the work necessary to produce the draft. I recognize that much of the debate surrounding SECY-90-377 concerns the examples in Appendix A. I believe that at this time the Commission should be focused on the policy issue of the staff's proposed **approach**. Much of the present debate, then, should take place in a dialogue with representatives of all industry parties and the public during development of the proposed Reg Guide.

7. I note that severe accident issues are not in the present SRP. What is the staff intending to use besides the existing SRP for the review of these designs?

8. I've heard the argument that some of the material is being requested for **standardization**. How can the staff make a **safety** determination without highly detailed information when the only review of the plant will be of a paper design (that is, without a physical plant), keeping in mind that there can be **no open items** apart from site-specific features?

IDI findings

THE IMPORTANCE OF DESIGN DETAIL IN ACHIEVING THE REGULATORY STABILITY AND PREDICTABILITY PROMISED BY 10 CFR 52.

Past experience has shown the importance of detailed nuclear power plant design activities -- what the industry has chosen to refer to as design implementation detail in the context of Part 52's requirement that certified designs be essentially complete. As examples, I would cite the attached results of several NRC-sponsored Independent Design Inspections at facilities licensed during the past decade. These inspections were performed well after the facilities' FSARs were first docketed and while construction activities were largely complete. These inspections disclosed numerous deficiencies which raised[in some cases potentially significant] safety questions concerning the adequacy of facility design.

Many of these deficiencies stemmed from improper interpretation of FSAR commitments or from otherwise technically incorrect methods of detailed design which were intended to implement well-understood FSAR commitments. This experience suggests that, if the NRC were to settle for, say, a level of detail comparable to that provided in an FSAR in order to certify a plant design, there is a considerable likelyhood that during the CP/OL licensing process or, much worse, just before facility operation, substantive issues relating to the adequacy of facility design could be raised.

In order to ensure that this does not happen under the Part 52 processes, it has been proposed that any unknowns associated with an FSAR level of design detail would be accommodated by a necessary and sufficient set of Inspections, Tests, Analyses, and Acceptance Criteria that, if met, would ensure that there remain no unresolved safety matters with the as-built, completed plant design. There are potential shortcomings with this approach, however. ITAAC implementation detail, like design implementation detail itself is, in fact, a matter of detailed design engineering. As a general proposition, the methods of ITAAC implementation can only be as detailed as the design itself. As a result, it is likely that ITAAC proposed for a design to be certified that has only been developed to an FSAR level of detail will be expressed at this same level of detail. In fact, the NUMARC draft ITAAC proposal confirms this to be the case.

If the Commission adopted the proposed approach concerning the level of design detail necessary to support design certification, the development of design and ITAAC implementation detail <u>after</u> design certification would involve the same kind of latitude for interpretation, or misinterpretation, of requirements and commitments associated with the certified design that has, in the past, led to the late identification of safety problems attributable to what can only be termed facility design. As a result, applicants proposing to build a design and CP/OL licensees wanting to operate their facilities will be exposed to virtually the same risk of litigation on the adequacy of facility

design that CP and OL applicants were faced with under Part 50 procedures, with a subtle twist.

A new class of design-related issues will emerge involving design implementation detail and ITAAC implementation detail. In CP/OL and preoperational hearings, these types of issues will supplant design issues explicitly addressed in the design certification and thus precluded from litigation. The language in Part 52 which states that the adequacy of the design will be established with finality during the design certification proceeding will be little more than an empty promise.

Byron IDI

In summary, in the mechanical systems discipline, the team recommends systematic review and corrective action programs to assure that (1) design work in the area of postulated breaks and cracks in high-energy and moderate-energy lines is complete, adequate and controlled and (2) necessary mechanical systems calculations are identified, performed and updated as necessary to support the current design. However, aside from meeting licensing commitments in the high-energy and moderate-energy line area, significant problems with the actual design were not found. The team attributes this to the experience of design personnel. For the Westinghouse balance-of-plant piping design work, the team recommends further examination to determine whether or not systematic weaknesses are present. In other respects, the problems found appeared to be confined to specific issues, such as electrical separation analyses, and the design process appeared to be controlled.

Callaway IDI

Section 1 of the report provides a summary of the results of the inspection and the conclusions reached by the inspection team. No pervasive breakdown in the design process was identified; however, your prompt attention is needed for resolution of the specific deficiencies identified.

The most significant negative findings or deficiencies are summarized as follows:

(1) There was a lack of formal control over Bechtel's use of plant design newsletters. Thus, these newsletters, which described acceptable modeling and stress analysis techniques, were not being applied uniformly to project design work (Section 3.1.2).

(2) The auxiliary feedwater pump turbine exhaust pipe was not classified as Seismic Category I and safety grade throughout its entire length. No justification was available. This represented incomplete detailed analysis to support pump operability requirements. A similar classification was identified in two other systems (Section 2.4).

(3) The ability of motor controllers to withstand fault currents had not been considered or assured. This represented an instance of improper detailed design (Section 5.2).

(4) The team identified needs for improvement in control of the design process at Bechtel in certain areas such as those related to high energy line break analyses (Section 2.4), guidance for two design groups (Sections 3.1.4 and 3.2.4), interface definitions (Section 4.4) and baseplate design (Section 4.5).

(5) Three instances were identified where specific FSAR commitments were not met, one of which involved the turbine exhaust pipe discussed above (Sections 2.3, 2.4, and 6.2).

With the exception of the matters identified in the findings and one observation concerning delay in resolving a design issue, the team considered the general project management to be a strength. Nearly all the detailed design information reviewed was adequate and consistent, indicating a controlled design process.

In accordance with 10 CFR 2.790(a), a copy of this letter and the enclosures will be placed in the NRC Public Document Room unless you notify this office, by telephone, within 15 days of the date of this letter and submit written application to withhold information contained herein within 30 days of the date of this letter. Such applications must be consistent with the requirements of 10 CFR 2.790(b)(1).

Perry IDI

Section 1 of the report provides a summary of the results of the inspection and the conclusions reached by the inspection team. There are five significant technical areas in the Perry design which indicate weaknesses in the design process and, therefore, raise the possibility that similar weaknesses will be found in systems other than those inspected by the team. We regard these systematic weaknesses as particularly significant. These are as follows:

(1) Piping stress analyses for the faulted condition did not always consider piping thermal stress in nozzle loadings on NSSS equipment, as required by licensing commitments.

(2) Piping stress analyses modeled equipment as rigid even when the equipment had a frequency more appropriate to dynamic modeling.

(3) Voltage drops due to motor starting, normal operations and excessive circuit lengths were not properly accounted for.

(4) Circuit breaker and fuse sizing was either contrary to licensing commitments or without adequate documented technical basis.

(5) Temperature profiles used as the basis to size room coolers and qualify electrical equipment were deficient.

We also identified in the report other systematic weaknesses in the design process, which are not as potentially significant to overall plant safety as those on the above list. Among these other weaknesses are programmatic deficiencies which are identified below.

(1) In some instances, the FSAR stipulated design criteria which were not implemented or were contrary to the plant design.

(2) In some instances, supporting documentation was inconsistent with the FSAR.

(3) In some instances, the final design, including design changes, was not traceable to design input. For example:

   (a) Analyses and calculations did not exist.

   (b) Calculations did not address all factors affecting the end result.

   (c) Judgments and assumptions were undocumented and unsubstantiated.

(4) The verification process was not as effective as could be expected, as indicated by the number of calculational deficiencies found by the team, including cases where the verifier found a deficiency also found by the inspection, but did not correctly identify its significance.

Harris IDI

Chapter 1 of the report provides a summary of the results of the inspection and the conclusions reached by the inspection team. There are several significant technical areas in the Shearon Harris design which indicate weaknesses in the design process and, therefore, raise the possibility that similar weaknesses will be found in systems other than those inspected by the team. These are identified below.

(1) Voltage drop considerations were not properly accounted for in a number of analyses.

(2) The design of the containment sump did not follow the guidance given in Regulatory Guide 1.82 which you committed to follow.

(3) Relay coordination was not effectively accomplished.

(4) The design of slender struts used to support piping systems did not address their dynamic excitation and eccentricity.

(5) Radiation protection analyses performed by EBASCO's Applied Physics Department had numerous non-conservatisms and errors.

(6) Seismic II/I (non-seismic piping, equipment, and components whose failure could affect seismic Category I equipment) analyses were incomplete and deficient, with a walkdown planned to resolve problems.

In addition to the conclusions in Chapter 1 which relate to particular disciplines the team identified in Chapter 7 of the report some concerns which were common to more than one discipline. These include:

(1) The EBASCO verification process was not as effective as could be expected, as indicated by the number and nature of the deficiencies found by the inspection. The team concluded that this indicates a lack of management attention to an important area of the design process.

(2) Certain deficiencies were found in the design performed on-site by Carolina Power and Light which will likely require design and hardware changes.

(3) Inappropriate use of the "minor design change" classification has resulted in situations where the team concluded that appropriate independent design verification reviews may have been ommitted.

It is our understanding that responsibility for performance of certain original design work is being shifted to your on-site engineering organization to smooth the transition from construction to operations, such that Carolina Power and Light will have more design capability during the operational phase.

We consider this intent to be commendable and are in complete agreement with the ultimate objective. However, it does not appear that some parts of the current organization possessed the requisite experience and training to undertake the expected responsibilities.

Seabrook IDI

Jther observations are identified where it was considered appropriate to call attention to matters for which there are no specific regulatory requirements, but which are recommended for your consideration.

Section 1 of the report provides a summary of the results of the inspection and the conclusions reached by the inspection team. The principal points from that summary are discussed below.

In the mechanical systems area, there were deficiencies in the method used by United Engineers to calculate available net positive suction head (NPSH) for the containment building spray pump. In addition, there is uncertainty as to the pressure drop across the inner screen of the containment recirculation sump and its effect upon NPSH. There is also uncertainty as to the required NPSH identified by tests because the pump was not tested with the motor to be used at Seabrook and United Engineers has not obtained test data establishing the torque capability of the Seabrook motor. The team independently calculated NPSH margin and determined that it may be less than required. The team also reviewed a Westinghouse calculation of NPSH for the residual heat removal pump and found deficiencies similar to those for the containment building spray pump. The team reviewed two NPSH calculations for the emergency feedwater pumps and found deficiencies with regard to the bases and validity of assumptions. Based on the number of deficiencies we found in NPSH calculations, involving three separate systems and two design organizations, there appears to be a systematic problem. Action needs to be taken to review NPSH calculations for other systems to determine if the designs are adequate and to determine the root cause of the deficiencies identified above. The team also found that work is being accomplished at a very late stage in assessing whether there is adequate protection of essent--l components from postulated pipe breaks and cracks in high and moderat energy piping. The design cannot be considered complete until the work is finished. In other respects, the design process in the mechanical systems area appeared to be controlled.

In the mechanical components area, there were items of technical significance which warrant additional design efforts. Waterhammer loads and modeling procedures should be addressed in certain piping reanalyses. Rapid closure of containment isolation valves during operation of containment building spray pumps should be eviewed taking into consideration the peak pressures that result. The functional adequacy of the containment building spray pump under specified thermal transient loadings should be confirmed. Bolted joints on certain valves should be assessed to be sure that their structural integrity is assured. In other respects, the design process in the mechanical components area is generally controlled.

In the civil-structural area, three areas of concern were found which warrant additional design efforts. Floor live loads are not included in load combinations which incorporate seismic loads. This is a violation of the basic structural design criteria approved for the plant. The classification of the structural elements of the tank farm structure with regard to seismic loadings and tornado loadings was found to be inconsistent within the project criteria and there were also inconsistencies between the project criteria and design calculations. Instances of improper modeling of the tank farm structure for

both the reinforced concrete portions and the structural steel portions were also noted. In other respects, the design process in the civil-structural area appeared to be controlled.

In the electric power area, the design process appeared to be generally controlled. However, the seismic and environmental qualification had not been satisfactorily demonstrated for a number of electric components.

In the instrumentation and controls area, an adequate set of procedures is in place to assure that the design can be controlled in a satisfactory manner. Nevertheless, portions of the present instrumentation and controls design may not be adequate. Sufficient independence of certain control circuits that are essential to the operation of three engineered safety features has not been demonstrated. There is lack of automatic valve position control for certain valves in the Residual Heat Removal System to assure that the valves are in the proper position for automatic operation of the Emergency Core Cooling System during the recirculation phase. In addition, seismic and environmental qualification has not been satisfactorily demonstrated for certain instrumentation and control equipment. In spite of procedures in effect at Yankee Atomic and United Engineers, these deficiencies have not been found by the current quality assurance program.

The above items and many others are listed in the enclosed inspection report. In general, the problems found in the Seabrook design appeared to be confined to specific issues that did not seem to cross discipline boundaries. The overall design appeared to be adequately controlled.

In accordance with 10 CFR 2.790(a), a copy of this letter and the enclosures will be placed in the NRC Public Document Room unless you notify this office by telephone, within 10 days of the date of this letter, and submit written application to withhold information contained herein within 25 days of the date of this letter. Such applications shall be consistent with the requirements of 10 CFR 2.790(b)(1).

You are requested to respond in writing to the findings and unresolved items within 45 days after receipt of this letter. With respect to the deficiencies identified in findings, the response should address the cause, extent, corrective actions and any other information you consider relevant. For unresolved items, the response should provide information concerning acceptability of the specific feature or practice involved or indicate the extent to which corrective action is needed. In such cases the cause and corrective actions and any other information you consider relevant should also be included in the response. Finally, the response should include your position, and the bases therefore, with respect to the necessity for conducting additional audits of design implementation in areas other than those covered by our inspection so as to provide assurance that deficiencies of importance either do not exist or are corrected. The response should be addressed to this office.

As discussed in the report, the NRC's followup efforts will be managed by the Office of Inspection and Enforcement with assistance from the Region I Office

River Bend

1.5.1 Mechanical Systems

In the mechanical systems area three substantial deficiencies were identified.

Passive failure in the low pressure coolant injection pump suction lines, which might result either in draining the suppression pool water below the minimum acceptable level or flooding emergency core cooling system equipment, was not properly considered (Deficiencies D2.3-2 and D2.3-3).

Resistance orifices in the low pressure coolant injection system were not installed. This decision was based on an incorrect calculation for net positive suction head and low pressure coolant injection pump runout flow rate. Without the resistance orifices there is a possibility of cavitation in the low pressure coolant injection loops during a design basis event (Deficiency D2.3-4).

In the design of the pneumatic supply for the automatic depressurization system valves it was improperly concluded that the safety grade air compressors alone were adequate. The design process failed to identify that the accumulators, associated check valves, and a higher air pressure (provided by the non-safety grade air compressors) are also needed for consistency with the functional requirements in the plant safety analysis. This may be corrected by appropriate analysis, specification of accumulator pressure, and leak testing of check valves. In fact, test and operations personnel had raised questions that might well have led to proper resolution in this manner. However, these questions had not been properly resolved at the time of the inspection and might not have been properly resolved given the misconception in the design process (Deficiency D2.4-6).

There were also deficiencies of lesser individual significance related to the adequacy or control of specific analyses in this area. These deficiencies included documentation errors, calculational errors, and failures to meet interface specifications provided by the nuclear steam system supplier.

The team reviewed other calculations in which no substantial problems were uncovered. These calculations were controlled, comprehensive, and technically correct. As a result, the team concluded that the design process in the mechnical systems area appeared to be adequately controlled.

1.5.2 Mechanical Components

A significant problem was found in this area with the loadings on safety relief valve nozzle flanges and the use of ball joints to reduce these nozzle loadings. Exceeding these nozzle loads could result in reactor coolant system leakage at the inlet flange of the safety relief valves. (Deficiency D3.3-3) A breakdown in the design process was found in the specification and evaluation of the ball joints to be used in the safety relief valve discharge lines. (Deficiency D3.6-2) Problems were found with required tests in the specification, review of test reports submitted by the vendor, transmittal of information to stress analysts, use of the information in the analysis, and feedback for results of the evaluation. A significant reanalysis effort may be required to resolve this problem and some re-design may be required. Use of ball joints is not customary in safety grade piping of nuclear power plants and the problems encountered may be attributed to lack of experience in their

application. No similar problems of this extent were found in the design process for customary plant features.

The team also identified deficiencies in the pipe and equipment supports area. Calculations for the stiffness of pipe anchors and restraints attached to structural steel did not include the flexibility of the structural steel. This omission affects the analysis of selected supports. These calculations should be redone to obtain revised stiffnesses and these values should then be used in the final analysis cycle for the as-built piping configuration (Deficiency D3.4-4). Large seismic Category I equipment supported on structural steel was not analyzed in accordance with the FSAR commitment to verify such components can be legitimately decoupled from the response of the structure. Corrective action may require analysis to ensure decoupling is justified, if appropriate (Deficiency D3.5-3).

Despite the substantial amount of analytical rework that apparently will result from our inspection in the mechanical components area, we do not believe that significant hardware changes will be required after confirming the integrity of the existing design. The team was generally impressed by the availability of procedures, the documentation of work, and the ready availability of that documentation. The team concluded that the design process in the mechanical components area appeared to be adequately controlled.

1.5.3 Civil/Structural

The Stone and Webster design for radial shear did not incorporate required "hooks" for the end anchorage of reinforcement steel. Although this was a violation of the ACI Code, the structures were adequate because they either contained sufficient embedment to be considered fully developed or the need for such reinforcement was eliminated by subsequent design changes. The item was not an adverse finding with respect to design control for the River Bend project since the decision to select this design was a knowing decision that was documented, reviewed and accepted by project personnel. However, it does appear to reflect on the generic approach by Stone and Webster to embedment length of the inclined stirrups and indicates a need for assurance that the concrete structures built elsewhere have adequate capacity to resist radial shear. The NRC staff is pursuing this aspect on a generic basis.

In the Auxiliary Building two deficiencies were identified concerning the shear design of the mat (See Deficiencies D4.11-1 and D4.11-2). It is expected that a reanalysis of the mat will demonstrate the adequacy of the structure for shear. Two findings also require further investigation to confirm that the exclusion of control rod drive piping masses in the shield wall calculation and failure to address decoupling of the control rod drive piping from the pipe supports do not have an adverse effect on the design (Deficiency D4.3-2 and Unresolved Item U4.16-1).

There were instances of isolated weak points associated with some of the elements of design control as noted in the particular deficiencies and observations. The team was generally impressed with the quality of design products in the civil/structural area. The personnel interviewed in the Structural Division and the Geotechnical Division were highly qualified engineers with many years of experience on the River Bend Project. The calculations were clear, well referenced and organized. Based on the facts gained during the integrated

design inspection, the team concluded that the design process in the civil structural area appeared to be adequately controlled.

1.5.4 Electrical Power

Two deficiencies were identified with respect to terminal block qualification. The documentation for terminal blocks inside containment was inadequate to support qualification. Also, Stone & Webster did not meet General Electric requirements for sealing conduit on terminal boxes inside containment (Deficiencies D5.10-1 and D5.10-2). The problem with terminal block qualification was identified by Stone and Webster prior to the inspection, however, the team felt that the problem was not being addressed in a timely fashion prior to the inspection. These terminal blocks are essential because they are used in redundant safety-related circuits within containment. During the inspection, Gulf States made the decision to bypass these terminal blocks which eliminated the concern regarding terminal block qualification.

A data sheet error in the specification for the Loops A and B low pressure injection valves resulted in the installation of incorrect (unqualified) motor actuators inside the containment (Detail A5.3-1). This problem also had been identified by the licensee prior to the inspection. However, it was obvious in nature and had remained undetected after several previous review of the documentation.

Overall, the results of the inspection in the electrical power discipline were positive. The design change process for the main control room was well controlled and documented. In general, the team was impressed with the accuracy and consistency that Stone & Webster maintained among numerous related electrical design documents. The team was also impressed with the project problem tracking scheme. Calculations and analyses were performed in a competent and professional manner. The team concluded that the design process in the electrical power area appeared to be adequately controlled.

1.5.5 Instrumentation and Control

In the instrumentation and control area the sample systems were expanded to include the Standby Service Water System and a portion of the High Pressure Core Spray System involving cooling of its diesel generator. Significant findings are described below.

Stone and Webster actuation logic for a LOCA for the standby diesel generators and certain Engineered Safety Feature loads required either off-site or on-site AC power to function. This application should have been provided noninterruptible AC or DC power. This design was changed during the inspection and is now considered acceptable. (Deficiency D6.5-2)

The team noted a design philosophy difference between the emergency core cooling systems designed by General Electric and their counterpart safety-related cooling systems designed by Stone and Webster. In the General Electric parts of design, generation of a LOCA signal causes the former systems to either begin accident mitigation or to operate in a mode that clearly demonstrates their immediate readiness to perform. In the Stone and Webster

parts of design, the LOCA signal is not used to initiate or demonstrate availability of the latter safety-related systems. The Stone and Webster design approach is considered acceptable provided that high availability of these cooling systems, such as the standby service water system, can be assured during plant operation (see Sections 6.7 and 6.12).

The design of main control room display indications and annunciation alarms was reviewed with emphasis on accident monitoring instrumentation and indication of bypassed or inoperable status of safety-related systems relative to Regulatory Guides 1.97 and 1.47 respectively. Stone and Webster makes only two equipment quality designations; namely, QA CAT. I (i.e., safety-related including Class 1E) or QA Cat. II (i.e., non-safety-related). Some system status accident monitoring variables do not fit either of these two categories, and may require upgrading of some QA Cat. II equipment to be consistent with commitments to implement Regulatory Guide 1.97. (Deficiencies D6.9-1 and D2.3-1)

Stone and Webster design procedures in the Instrumentation and Control area used at Cherry Hill and at the River Bend site appear to be complete and comprehensive. Their use in accomplishing design work at both locations was evident in all of the documents reviewed. The team concluded that the design process in the instrumentation and control area appeared to be adequately controlled.

1.5.6 Overall Conclusions

Several problems identified in the River Bend design appeared to be threads of similar types of findings that crossed discipline boundaries.

The team was concerned about the design process in light of the findings discussed above. The post LOCA flooding calculation failed to properly address the degree of protection against draining the suppression pool or flooding ECCS equipment. The flow restricting orifices that are usually needed to meet General Electric specifications to provide adequate NPSH were eliminated from the residual heat removal system based on an incorrect calculation. The design of the air supply for the automatic depressurization system actuators failed to identify that several elements were necessary for consistency with the safety analysis performed by General Electric. The stress analysis of the safety relief valve discharge line containing ball joints failed to address or meet the interface load limits provided by General Electric. The diesel starting logic failed to fully meet the FSAR commitment to start on either loss of off-site power or LOCA. Because a great deal of good work was also reviewed, the team does not consider that these findings warrant negative conclusions concerning adequacy of the overall design process. However, they do raise questions concerning the design verification process. The team's concern was heightened by the fact that system descriptions (or system design criteria) were not used to guide the design effort for the River Bend project. Use of such guidance is a standard approach at most architect/engineering firms, including Stone and Webster. The design verification process was not modified to accomodate the fact that the River Bend project did not use system descriptions. Based on these considerations and in light of the nature of the findings discussed above, the team concluded that a limited design review should be conducted by off-project Stone & Webster or Gulf States Utilities personnel of a sample of other safety

related systems to determine whether or not similar deficiencies can be expected elsewhere. The team recommends that these be selected from both nuclear steam supply systems pre-engineered systems and from Stone and Webster's scope of supply.

Another cross-discipline problem concerned the general quality of the calculation checking and design review documentation process being used by Stone and Webster. Many examples were identified of simple arithmetic errors, transcription problems, lack of ability to identify the individual responsible for detailed checking or review of the particular calculations, etc. Generally these deficiencies were minor in nature or had already been identified by other features of the design process. As a result, the team was not overly concerned with the River Bend design in this regard. The design organizations should investigate their practices in these areas and consider appropriate improvements (see Report Attachment A-1).

Although the aspects of the design process mentioned above raised concerns and questions, in other respects the team considered that the overall design process was well controlled. The team was very impressed with the control of documents within the project. The personnel assigned to the project were considered to be highly motivated and well qualified.

AdvLWR
Vol. 4 # 57.1