# Independent Design Review Panel

## REPORT

October 15, 1996

Phil Clark Chairman

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## I. Introduction

The Independent Design Review Panel was chartered by Dr. P. M. Beard, Senior Vice President, Nuclear Operations for Florida Power Corporation to perform an independent assessment of the Crystal River Unit 3 Design Bases and the adequacy of the design bases management control processes to provide reasonable assurance of safe plant operation. The CHARTER of the panel is provided as Appendix A.

Dr. Beard appointed the following industry personnel to comprise the panel:

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|---|---|---|
| Philip R. Clark (CHAIRMAN) | | Retired CEO, GPU Nuclear |
| Dr. Charles W. Pryor Jr. - (Vice Chairman) | | Retired CEO Framatome Technologies |
| Noman M. Cole. Jr. | | MPR Associates Inc. |
| Hugh A. Hammond | | Duke Power Company |
| James H. Lash | | Toledo Edison |
| Gordon R. Skillman | | GPU Nuclear |
| Fred W. Titus* | | Entergy Operations |
| Roger K. Wyrick | Liaison | INPO |
| Larry M. Lesniak | Liaison | Framatome Technologies |
| Robert E. Vaughn | Liaison | Parsons Power Group |
| Rolf C. Widell | Coordinator | Florida Power Corporation |

* For August and September meetings, Rick Lane, Director, Design Engineering(ANO), of Entergy Operations attended for Fred Titus.

## II. Executive Summary

### A. Formation and Operations of the Panel

During the formation of the Review Panel FPC management and the Panel Chairman discussed the appropriate qualifications and experience of the membership and the general scope. Subsequently, the Review Panel reviewed the proposed charter and satisfied itself that it was clear and practical, provided adequate flexibility and that the team composition was sound.

The Charter did not call for and the Panel did not attempt a technical review of the Design Basis information. The focus was on programs and process.

Samples of Design Basis information and documents were examined to provide understanding of context or a limited check on the application of processes and procedures. While FPC encouraged the Panel to interpret its Charter liberally, we did not knowingly go beyond Design Basis and closely related items.

The Panel exercised control of its agenda and operations and received from FPC complete and open access to documents and personnel. This report, its conclusions and recommendations, are the products of and unanimously supported by the Panel members.

The Panel used the seventeen questions stated in its Charter as a framework for its activities. The Panel divided into four subgroups, each of which pursued answers to several of the questions. The information developed, observations and recommendations were then discussed and final versions adopted by the entire Panel. The questions, answers and resulting recommendations are provided in Section IV.

The Panel members all worked part time but devoted a significant effort to this review including three 2-day meetings of the whole Panel at Crystal River, a one day meeting of the Panel in Alexandria, Virginia, five days of reviews at Crystal River by one or more members, review of numerous documents from Crystal River and some from other plants and interviews with more than thirty Crystal River employees from Executive Vice President to individual contributors. (See Appendix C for a detailed outline of Panel activities.)

## B. Design Basis

The regulatory requirements relating to design basis are very broad and general consisting primarily of; 1) a paragraph in Section 10 to the Code of Federal Regulations, Part 50.2 (10CFR50.2) which defines Design Basis Information, 2) a general requirement in Appendix B to 10CFR50, Criterion III Design Control, plus 3) requirements for notification and reporting of certain events or conditions affecting or violating the Design Basis stated in the Code of Federal Regulations and regulatory guidance documents.

There is some guidance on what constitutes Design Basis Information and appropriate documentation in NUMARC 90-12, which was accepted (with comments) as useful by NRC and in NUREG 1397 which presents results of an NRC survey of licensees with some NRC comments. However, significant variations are allowed and exist among the utilities represented on this Panel (and undoubtedly others) in the format, methods, and procedures used to develop, maintain and utilize Design Basis Information. Also, the detailed

regulatory requirements and commitments vary from plant to plant based on its licensing history.

Past initiatives by FPC relative to improving its knowledge and documentation of the CR-3 Design Basis were significant and were reviewed by the Panel. Current FPC initiatives relative to Design Basis information were also reviewed. These have identified and in many cases started to address a number of the areas addressed by the Panel. An attempt has been made to recognize these FPC initiatives in this report.

We note that as a result of the recent increased attention on Design Bases, Final Safety Analysis Report (FSAR), 10CFR50.59, etc., all of the licensees represented on the Panel are reviewing or planning to review some or all of their Design Basis related programs for possible improvement.

Late in its review, the Panel received copies of the NRC Integrated Performance Assessment Process (IPAP) report on Crystal River. A reading of the report did not show any direct conflicts with the Panel's recommendations. However, a thorough review of the NRC report was not attempted.

As noted above, the regulatory and industry guidance related to Design Basis is relatively limited. The Panel has recommended some significant changes and enhancement to FPC processes and practices relative to Design Basis. No implications are seen by the Panel or should be drawn as to other areas of CR-3.

## C. Licensee Event Reports Pertaining to "Design Basis"

Since the number of CR-3 Licensee Event Reports (LERs) pertaining to Design Basis was an immediate cause of the establishment of the Panel, these LERs were reviewed and analyzed by the Panel in a number of ways.

As a result of its review, the Panel concluded that the number of CR-3 LERs pertaining to Design Basis was due in part to:

a) FPC practice and process in recent years to classify potential plant "reportability" concerns differently than the other plants on the Panel which resulted in significantly more Design Basis LERs than would have been the case at the other plants.

b) FPC practice in recent years to submit an LER for "possible" Design Basis issues rather than wait for a final determination. Several issues

submitted as Design Basis LERs were subsequently determined not to be Design Basis LERs. (See Appendix D.)

c) Some significant plant modifications made at other B&W plants were not made at Crystal River. As a result of this and differences in the original plant designs, the design margins in some significant areas are lower at Crystal River. This has added to the number of Design Basis issues at Crystal River found by such programs as the set point and EOP programs.

The Panel also reviewed an FPC assessment of the individual and cumulative safety significance of the LERs. The Panel did not see a basis for a significant safety concern.

## D. Context for Recommendations

The recommendations of the Panel are interrelated and range from broad (Develop and promulgate a workable, understood definition of Design Bases) to specific suggested changes to a procedure. In the absence of an identified significant safety concern it seems appropriate to set an overall plan for responding to the recommendations in this report and any related NRC, INPO, FPC recommendations to recognize relative priority and logical sequencing of actions to minimize avoidable duplication or redoing of efforts.

The Panel developed its conclusions and recommendations against its judgment of excellence or best practices, not just against regulatory requirements. Thus, many of the recommendations go beyond meeting the regulations.

## E. Recommendations

**Each recommendation is followed by a citation to Section IV where the recommendation and the observations leading to it can be found.**

1. FPC should recognize the importance of the Design Basis and Licensing Basis and take steps to treat them as major programs with defined scope, clearly assigned ownership, and recognition throughout the organization. See Question 17, Recommendation A.

2. Establish a clear and complete definition of what documents and/or sections of documents contain the Design Basis for CR-3. Revise NEP-216, *Plant Design Basis Documents*, as required. See Question 1,

Recommendation A. (The issue of validation and accuracy of these documents is dealt with in recommendation 11F.)

3. Establish the legal and regulatory status of the Crystal River Unit 3 Final Safety Analysis Report in relation to utilizing the specific information for design engineering purposes. Also establish what was the independent review cycle for FSAR submittals. Review the appropriateness of page 141, Volume 1, paragraph 10 that excludes the FSAR as a design document. This may unnecessarily restrict the use of information contained in the FSAR for design purposes. See Question 1, Recommendation E.

4. Assign overall responsibility and ownership for maintaining the Design Basis to a specific manager. Also assign ownership of the design basis of defined systems, structures and components to specific positions within the appropriate nuclear organization. Make the individuals in these positions responsible to provide an independent review to ensure that any changes to procedures or hardware affecting their system or component are consistent with the design basis. This would enhance expert knowledge within the organization of key design basis information. See Question 1, Recommendation C.

5. Develop a training program for ensuring that FPC nuclear employees understand what documents constitute the design basis and the appropriate change control procedures for these documents, particularly when plant modifications are involved. In addition, the training program should cover the distinctive uses of design basis information, licensing basis information and the impact of non-safety SSC's on the plant's design basis." See Question 1, Recommendation B.

6. FPC should complete its efforts to better clarify the bases and approach to handling piping analysis and piping support design basis issues. See Question 2, Recommendation A.

7. Assure that the responsibility for assuring plant operation is consistent with the design basis is clearly defined and understood. See Question 1, Recommendation D.

8. The panel recommends and CR-3 had decided to install design upgrades to the HPI system to enhance operating margin in the event of certain postulated transients. See Question 5, Recommendation A.

9. The panel recommends and CR-3 has decided to conduct a thorough investigation to determine why similar balance-of-plant design features are limiting at CR-3 and not at TMI-1. See Question 5, Recommendation B.

10. FPC should complete their review of the July Framatome Technologies (FTI) report and take any appropriate action. See Question 5, Recommendation C.

11. FPC should take action to have the Steering Committee of the B&W Owners Group add the subject of Design Basis to its regular meeting agenda. Significant value could be gained by developing a more standardized approach to this important, emerging issue of design basis validation and control. See Question 6, Recommendation A.

12. The Panel recommends that when dealing with design basis issues, FPC take greater advantage of discussions with the other B&W utilities. See Question 6, Recommendation B.

13. The training on the design basis and the use of design basis information should be delivered to the appropriate individuals in organizations outside the engineering sections, and it should be covered in continuing training programs for those groups. See Question 7, Recommendation A.

14. Consider including all design basis documents (See Recommendation 2) at the designated plant locations for design bases documents and electronically on FulText. See Question 8, Recommendation A.

15. Obtain controlled copies of the Framatome/B&W type 52 reference documents for use by station engineering personnel with appropriate proprietary information protection. See Question 8, Recommendation B.

16. FPC management should revise resource planning and work management practices to ensure that adequate time and priority is provided to perform high quality engineering work and reviews. This should include an effective screening process to cancel engineering work requests of marginal value and to assign appropriate priority and schedules to all engineering work. See Question 9, Recommendation A.

17. Performance errors identified during design verification reviews should be monitored and remedial training and corrective actions provided for repetitive problems. See Question 9, Recommendation B.

18. Establish additional expectations for the plant's Design Review Panel to ensure that design basis implications of the modification have been adequately reviewed and considered. See Question 9, Recommendation C.

19. Consider revising AI-400C, *New Procedures and Procedure Change Requests*, to require a qualified review for all procedure changes by either nuclear plant technical support (system engineering) or nuclear engineering design. Alternatively, consider adding to AI-400C, *New Procedures and Procedure Change Requests*, more prescriptive conditions for when a qualified engineering review should be required. See Question 10, Recommendation A.

20. Revise AI-400F, *New Procedures And Procedure Change Processe. For Emergency Operating Procedures (EOPs), Abnormal Procedures (APs), and Verification Procedures (VPs)*, to require a qualified review by nuclear engineering design for abnormal procedures and verification procedures as well as for emergency operating procedures. See Question 10, Recommendation B.

21. Include the Enclosure 11 guidance from AI-400C, *New Procedures and Procedure Change Requests*, in AI-400F, *New Procedures And Procedure Change Processes For Emergency Operating Procedures (EOPs), Abnormal Procedures (APs), and Verification Procedures (VPs)*, to assist the originator in determining if the procedure change affects design conditions or design requirements. Consider revising Enclosure 11 to be a check-off list to ensure consideration of design basis issues. Provide guidance to identify on the check-off list the specific sections of the FSAR, technical specifications, COLR, or design basis documents that may require revision. See Question 10, Recommendation C.

22. Develop specific guidance and training for engineering personnel for conducting technical reviews of procedure changes. This should include emphasis on evaluating the effect of procedure changes on design basis parameters and design requirements and on identifying possible revisions of the FSAR, technical specifications, COLR, and other design basis documents. See Question 10, Recommendation D.

23. Engineering management should clearly communicate and reinforce the need to revise design basis documents promptly following implementation

of a plant modification or procedure change. Procedure NEP-216, *Plant Design Basis Documents*, should be revised to provide specific guidance for the maximum time allowed to make temporary changes to the design basis documents and to incorporate temporary changes as permanent revisions. Other nuclear plants typically require permanent revision of design basis documents within 30 - 60 days of implementing a plant change. See Question 11, Recommendation A.

24. Revise procedure NEP-213, *Design Analyses/Calculations*, to provide specific guidance or a check list for ensuring that design basis documents are reviewed and revised as necessary as a result of an analysis or calculation. See Question 11, Recommendation B.

25. NOD-11, *Maintenance of the Current Licensing Basis*, should be revised to include specific guidance for ensuring that FSAR changes are reviewed by engineering to identify and then make any appropriate revisions to design basis documents. See Question 11, Recommendation C.

26. FPC should consider promulgating a procedure for the control of design/licensing basis information and documentation that applies to the entire Crystal River 3 nuclear organization. This procedure should require any change to the plant, plant documentation or procedures be evaluated for effect on design basis information and documentation. See Question 11, Recommendation D.

27. To help ensure that inaccuracies in design basis information and documentation are identified and corrected over time, establish clear expectations that engineering personnel are accountable for reviewing and correcting design information and requirements. Also establish that all personnel are expected to report design basis deficiencies or questions so they can be addressed. See Question 11, Recommendation E.

28. FPC should proceed with the planned selective SSFIs to further assess and ensure the adequacy of the design basis information. See Question 11, Recommendation F.

29. Station management should consider including within the NGRC annual assessment, the Quarterly Manager Assessment, or PRC reviews an assessment of the cumulative effect of design basis and design discrepancies on plant safety. See Question 12, Recommendation A.

30. The assessment of recent design basis LERs (Tanguay report) on plant safety should be revised to include an explicit conclusion regarding the

cumulative effect of the 44 LERs reviewed. See Question 12, Recommendation B.

31. FPC should establish a "stand alone" Safety Evaluation (SE) format that requires an integrated discussion of the proposed change, its effect on safety, and the USQ determination. See Question 13, Recommendation A.

32. A subcommittee of the NGRC should review SEs after the fact sufficiently to address quality, trends, and issues and report the results to the full NGRC. See Question 13, Recommendation B.

33. Consider limiting the number of personnel allowed to perform, review, and approve SEs. This should result in sustaining a level of proficiency in the process and thereby improving the overall quality of SEs. See Question 13, Recommendation C.

34. Develop a station SE training and qualification program that will result in a consistent understanding of SE process requirements, by all personnel who perform, review and approve SEs. See Question 13, Recommendation D.

35. The plan for the annual Engineering Phase 3 (continuing) training should be broken into three or four sessions and the Curriculum Review Committee should provide topic input to each session. This would allow Phase 3 training to be responsive to current issues. See Question 14, Recommendation A.

36. The Phase 3 continuing training curriculum should be more reflective of current design basis and licensing basis issues. See Question 14, Recommendation B.

37. Reinforce engineering ownership of its design basis training program. See Question 14, Recommendation C. (See Recommendations 5 and 13 on training for areas other than engineering.)

38. Change NOD-52, *Commitment Processing and Management of Programmatic Commitments*, to clearly identify Design Bases as an important element of the overall Configuration Management Program, and to require actions to ensure thorough assessment and treatment of changes, including the supporting or underlying analyses and assumptions thereto, as well as hardware issues (Structures, Systems and Components). See Question 15, Recommendation A.

39. Change procedures AI-404A, *Review of Technical Information*, and AI-404B, *Review of Industry Operating Experience*, to ensure Design Bases

issues are clearly considered, identified, and thoroughly treated. See Question 15, Recommendation B.

40. Change the Modification Approval Record (MAR) process document (NEP-210) to ensure Design Bases issues are clearly considered, identified and thoroughly treated consistent with and similarly to NOD-52, *Commitment Processing and Management of Programmatic Commitments*, and AI-404A, *Review of Technical Information*, and AI-404B, *Review of Industry Operating Experience*. See Question 15, Recommendation C.

41. Ensure procedure CP-150, *Identifying and Processing Operability Concerns*, has an appropriate "process" to ensure that only appropriate items are notified and reported. See Question 16, Recommendation A.

42. Ensure procedure CP-150, *Identifying and Processing Operability Concerns*, includes appropriate guidance from NUREG-1022 to take advantage of the allowed time frame to evaluate an issue prior to notification. See Question 16, Recommendation B.

43. Revise CP-150, *Identifying and Processing Operability Concerns*, to be consistent with industry practice for the shift supervisor to seek assistance in determining the need to make a notification. See Question 16, Recommendation C.

44. FPC undertake action to periodically assess the adequacy of the Design Bases Program and to report on it in a manner that is visible to management and requires organizational response. See Question 17, Recommendation B.

45. The Panel encourages FPC to continue implementation of a graded approach to instrument error calculations. See Question 4, Recommendation A.

46. The Panel's charter was "Design Basis"; however, it found that in some regards similar problems existed in the Licensing Basis. As implied in Recommendation 1, FPC should take steps to improve the definition, understanding, and use of the Licensing Basis and consider at least Recommendations 2, 4, 5 and 7 in that regard. See Question 1, Recommendation F.

47. Develop guidance for operations' personnel that establishes expectations for their review of engineering analyses and calculations. It should be clear that the operations' review is to validate input assumptions related to how the plant is operated and to review outputs for effect on plant operation,

not to provide a full engineering verification. (See Question 7, Recommendation B.)

## III. Crystal River Design Basis Programs

FPC provided to the panel descriptions of prior and current design basis programs as follows:

Crystal River Unit 3 was designed in the late 1960s and early 1970s. The plant was constructed in the mid to late 1970s and began commercial operation in March 1977. During the period of construction, Florida Power Corporation experienced financial stresses that affected the completion schedule of the unit and the construction close out activities. Florida Power Corporation did not foresee the extensive effort that would be required to support the operation of a nuclear power plant. As a result, there was no recognized need to assure complete detailed design documentation was turned over from the Architect Engineer and NSSS vendor.

During the early years of operation, plant modifications were designed primarily by the A/E and/or NSSS vendor. The design of the post-TMI modifications was also handled in this manner. The change in the nuclear environment that occurred during this period brought recognition that the ready availability of design and design basis information was a key element in the ability to adequately support nuclear plant operation. Florida Power Corporation began efforts to pull together this information in the mid 1980s with several focused initiatives.

The first of these initiatives was development of the Analysis Basis Document. The objective of this effort was to obtain and catalog the basis and assumptions that went into the FSAR Chapter 14 Safety Analyses and then to validate this information against the operation of the plant. The product was completed in 1989.

Shortly after beginning the Analysis Basis Document, FPC undertook development of a set of Design Basis Documents. At the time there was limited guidance on the content of this type of document. This resulted in a product that was more descriptive and contained a lot of information about the systems, but did not fully document the design basis to the extent or depth that was later found to be needed.

A follow on initiative was the Enhanced Design Basis Document which built on the original DBD, but went into more depth and detail on 37 of the most safety significant systems. This initiative began in the late 1980s and was completed in 1992. The objective was to pull the key design information and parameters into the DBD and to provide references to where the information came from. The documents were validated against the source documents and the physical configuration in the plant at that time. However, there was no effort to verify that

the source documentation was correct or accurate in relation to the way the plant was operated.

As a result of an Operational Safety Team Inspection performed by the NRC in 1987, significant deficiencies in the electrical design calculations for the plant were identified. A program to totally reconstitute the electrical calculations was completed in 1992.

In 1988 the Configuration Management Program was initiated. The objective of this program was to develop a data base containing the design information for the plant on a component level basis that would be available throughout the nuclear organization to support design, maintenance and operation. This program became operational in 1990 and is still in use.

In the late 1980s a significant effort was made by the industry and the NRC to develop new Standard Technical Specifications for each class of plants from each NSSS vendor. Crystal River Unit 3 volunteered to be the lead plant for the B&W plants. A major part of the development of the Improved Technical Specifications was to develop more complete and accurate bases for the specifications. The Improved Technical Specifications were implemented at CR-3 in early 1993.

CR-3 began moving toward 24 month fuel cycles in the late 1980s and by 1990 began to operate on this extended cycle. The need to resolve major equipment problems resulted in planned mid-cycle outages which deferred the need to address instrument calibration requirements related to longer cycles. In 1994 FPC began the Instrument Setpoint Program to establish calibration requirements for the extended cycle. This program involved a commitment by FPC to new restrictive standards for the calculation of instrument error. This program is continuing.

In 1995 FPC initiated an effort to perform a thorough review of the Emergency Operating Procedures with one of the objectives being to verify that the mitigation strategies utilized were adequately supported by the plant design basis. This initiative will be completed in late 1996.

The problems identified at several other nuclear plants led FPC to begin a review of the FSAR for Crystal River Unit 3 in 1996. The objective was to identify deviations between the plant configuration and documentation and the FSAR. This initiative is currently planned to be completed in early 1997.

Overall, during the period of time from the mid 1980s to present, largely as a result of the programs described above, numerous design and design basis issues have been identified.

IV. Answers to Questions in the Panel Charter

*Question 1: Is FPC's interpretation of "design basis" as stated in 10CFR50.2 consistent with the industry?*

Scope of Review

Interviews were conducted with senior station managers and plant personnel in design engineering, configuration management, system engineering, operations, maintenance, and licensing. Additionally, station documents and procedures regarding the plant's design basis were reviewed. These included procedure NEP-216, *Plant Design Basis Documents*, the Final Safety Analysis Report (FSAR), design basis documents, enhanced design basis documents, analysis basis documents, topical design basis documents, technical specifications, engineering calculations and analyses, core operating limit reports, technical basis documents for emergency operating procedures, and various references contained in design basis documents.

Reference documents reviewed by the Panel included: 10CFR50.2, 10CFR50.72, NEP-216, NUMARC 90-12, NUREG 1397, NUREG 1022, FPC's Tanguay Report

Observations

Regulation 10CFR50.2 provides a very broad and general description of what constitutes "design basis." FPC's interpretation of this is not inconsistent with that of industry.

There is not a clear, complete definition and consistent understanding of what constitutes "design basis" information within the FPC nuclear organization. FPC's NEP-216 defines Design Basis documents as :

* design basis documents (DBDs)
* enhanced design basis documents (EDBD's)
* analysis basis documents
* topical design basis documents

This list is not complete since it does not include relevant documents such as the FSAR, the Technical Specifications, System Descriptions, Core Operating Limits

Report, Setpoint Bases, Drawings, Technical Basis for EOPs, etc. Further, it is not clear that FPC intends for this list to be their definition of "design basis". This conclusion has also been reached by FPC as documented in a recent report to management written by Paul Tanguay. Additionally, page 141, Volume 1, paragraph 10 of the FSAR states that Florida Power Corporation does not require the FSAR to be a design document. However, the various system design basis documents make frequent references to the FSAR as a source of design information.

There was consistent response from personnel that Design Engineering owns the design basis documents with respect to technical adequacy, and Configuration Management administratively controls the Design Basis Documents; however, there are documents that contain design basis information that are not controlled in the engineering organization and there has been no ownership assigned for design basis information at a system or component level. FPC has used the "area" owner and procedure owner concept successfully to improve and maintain plant materiel condition and procedure quality. This concept could also be applied to improve the technical content of the design basis information and improve the technical expertise on the various systems.

Additionally, the station has not adequately recognized the difference and overlap between design basis information and licensing basis information. This has contributed to uncertainty as to when regulatory commitments are affected by changes in design information and parameters.

Design Basis information is important to any nuclear power plant for the following functions:

1) Maintaining compliance with regulations and commitments.

2) Facilitating control of design information.

In addition, design basis information is used to determine whether there has been a violation of the design basis requiring NRC notification or reporting.

The use of design basis information is different for each of the three functions. FPC has not adequately defined the differences among these usages, which has lead to inconsistencies.

**Recommendations**

1A. Establish a clear and complete definition of what documents and/or sections of documents contain the Design Basis for CR-3. Revise NEP-216

as required. The issue of validation and accuracy of these documents is dealt with in recommendation 11F.

1B. Develop a training program for ensuring that FPC nuclear employees understand what documents constitute the design basis and the appropriate change control procedures for these documents, particularly when plant modifications are involved. In addition, the training program should cover the distinctive uses of design basis information, licensing basis information and the impact of non-safety SSC's on the plant's design basis."

1C. Assign overall responsibility and ownership for maintaining the Design Basis to a specific manager. Also assign ownership of the design basis of defined systems. structures and components to specific positions within the appropriate nuclear organization. Make the individuals in these positions responsible to provide an independent review to ensure that any changes to procedure or hardware affecting their system or component are consistent with the design basis. This would enhance expert knowledge within the organization of key design basis information.

1D. Assure that the responsibility for assuring plant operation is consistent with the design basis is clearly defined and understood.

1E. Establish the legal and regulatory status of the Crystal River Unit 3 Final Safety Analysis Report in relation to utilizing the specific information for design engineering purposes. Also establish what was the independent review cycle for FSAR submittals. Review the appropriateness of page 141, Volume 1, paragraph 10 that excludes the FSAR as a design document. This may unnecessarily restrict the use of information contained in the FSAR for design purposes.

1F. The Panel's charter was "Design Basis"; however, it found that in some regards similar problems existed in the Licensing Basis. As implied in Recommendation 1, FPC should take steps to improve the definition, understanding, and use of the Licensing Basis and consider at least Recommendations 1A, B, C and D in that regard.

*Question 2: Is FPC's design bases documentation consistent with industry practices and regulatory guidance in defining the requirements, limits, and parameters for the systems and components?*

### Scope of Review

Regulatory guidance was considered to be represented by NUMARC 90-12 and its qualified endorsement contained in the NRC paper SECY 90-365. Information regarding industry practices was provided by NUMARC 90-12 and NUREG 1397, as well as by each of the member utilities on the panel. Other sources of information reviewed include the NRC Inspection Module for Design Bases Review, CR-3 design bases documents, and interviews with FPC personnel. A technical review of design bases information was not performed.

### Observations

Design bases information is spread throughout a number of documents at CR-3, as noted in response to Question 1, *[i.e., FSAR, Tech Specs, DBDs, EDBDs, SERs, Responses to RAIs, Core Operating Limits Report, Analysis Basis Documents, Technical Bases Documents, Topical Design Basis Documents, Regulatory Requirements (e.g., 10CFR50), adopted codes and standards, Topical Reports, etc.]* Documents were reviewed by the panel with special emphasis placed on the Enhanced Design Basis Documents.

The panel also notes that industry practice in documenting design bases varies considerably from utility to utility. Although variations do exist between utility FSARs, Technical Specifications, Design Bases Documents, and other documents that contain design bases information, the total set of information at CR-3 is typical of that at other utilities. A strength of CR-3 is the quality of the Enhanced Design Basis Documents, which clearly present the major elements of design bases information as identified in NUMARC 90-12.

Although no technical review was performed, interviews with design and systems engineers indicate that, with certain exceptions, the composite of all the information available as described above provides them with reasonable confidence that their work is within the bounds of the design bases requirements. Exceptions were noted in the area of piping analysis and piping support design (where FPC is already taking action) and occasional unretrievability of formal documentation of early design assumptions; this latter is similar to the situation at other plants.

Review indicates the scope and content of the CR-3 design basis documentation is generally consistent with industry practices and regulatory guidance except as

noted above. The accuracy of the information is addressed in question 11 and the retrievability is addressed in question 8.

Recommendation

2A. FPC should complete its efforts to better clarify the bases and approach to handling piping analysis and piping support design basis issues.

*Question 3: Are past and current upgrade programs adequate to give reasonable assurance that design bases issues affecting safe shutdown of the plant will be discovered?*

Scope of Review

Past and current upgrade programs identified for review by the Panel included the following:

Setpoint Program (surveillance for 24-month cycle)
Emergency Operating Procedure (EOP) upgrade
Enhanced Design Basis Document program
Improved Technical Specifications

The panel review was based on the following resources:
NEP 213, *Design Analyses / Calculations*
NEP 216, *Plant Design Bases Documents*
Action Plan: *Surveillance Requirement Extension to 24 Months*
Action Plan: *EOP Enhancement Program (Phase 2)*
Interoffice Correspondence: 7/22/96, NED96-0428, *Graded Approach Methodology for Instrument Uncertainty*
Interviews with FPC Personnel

Observations

The panel concluded that the scope and breadth of the CR-3 upgrade programs as identified and discussed above do provide reasonable assurance that design bases issues affecting safe shutdown would be identified.

Recommendations

None.

*Question 4: Are past and current upgrade programs adequate to give reasonable assurance that plant operating documentation is consistent with the design bases?*

## Scope of Review

The scope of review was the same as described in response to Question 3.

## Observations

The panel concluded that the upgrade programs generally were designed such that they would identify and fix areas where plant operating documentation should be revised to be consistent with the design bases as known at that time. However, this doesn't answer the question of whether the current operating documentation is consistent with the plant design basis. This is addressed by Panel recommendations 4, 7, 27 and 28 which address ownership of the design basis, responsibility for assuring operation of the plant is consistent with the design basis, ensuring that inaccuracies in design basis information and documentation are identified and corrected over time and proceeding with the planned selected SSFIs.

The following relevant points were noted about those programs reviewed:

Setpoint Program
Calculation of setpoints is controlled by procedure NEP-213, *Design Analyses / Calculations*. Special additional requirements are included for instrument setpoints and string error calculations. All changes to setpoints must undergo 10CFR50.59 screening, a formal review of design inputs and formal cross-reviews of results. Changes are subject to specified configuration control procedures.

Improved Technical Specifications
The focus of the ITS was not directed towards changing the plant (relatively few changes resulted). Problem resolution was less formal than with other upgrade programs due to the more direct focus on ensuring that the new Tech Specs did not negatively impact plant operations.

EOP Enhancement Program
The very nature of this program is to ensure plant operating documentation (EOPs) is consistent with the design bases of the plant. It compares critical parameter values and setpoints in the EOP to their design bases sources to ensure they match. Discrepancies are evaluated and processed as required using procedures CP-111 and CP-150.

Enhanced Design Bases Document Program
The EDBD program included an extensive field validation effort. This effort included verification of consistency in the areas of system design,

operations, surveillance testing, and the physical configuration of the plant. Parameter Validation Sheets and Reference Validation Sheets were prepared to systematically verify consistency between the EDBDs and other documentation. The program included a formal process for identifying and resolving Open Items.

**Recommendations**

4A. The Panel encourages FPC to continue implementation of a graded approach to instrument error calculations.

***Question 5: Are there generic design basis issues from other B&W plants that FPC needs to address?***

**Scope of Review**

July 1996 Comparison of B&W Plants, Framatome Technologies, Inc.
Discussions among panel members from other B&W plants
Interviews with FPC personnel

**Observations**

The reference report was prepared as part of the panel's work and addresses this question. FPC has actions underway to evaluate this report and has made decisions in some cases to upgrade the CR-3 plant with modifications which will bring the plant in line with other B&W-designed plants where generic design differences existed.

The panel concluded that some of these design differences result in less operating margin for CR-3 under certain postulated conditions. This is believed to have contributed to design basis issues. For example, CR-3's programs to review Emergency Operating Procedures (EOP's) with attention to setpoints and instrument error ranges has identified a number of design basis issues which may not have existed or been noteworthy had more margin existed in the plant as the result of design upgrades which other B&W plants have implemented.

For example, all B&W plants of the lowered loop design (Davis Besse is a raised loop design) except CR-3 have installed cross-connect lines in their HPI systems to allow for flow balancing in the event of a postulated HPI line break. Also, all the other lowered loop design B&W plants have installed cavitating venturis or similar flow-limiting devices which restrict flow through postulated HPI line breaks. The lack of these features at CR-3 significantly reduces operating margin and provides little tolerance for instrument errors.

The closest plant in design to CR-3 is TMI-1. The panel discussed at length the similarity of diesel generators and decay heat (DH) pumps at the two plants. However, at CR-3 the DH pump and the emergency feedwater pump (EFP)cannot be simultaneously loaded on the emergency diesel generator, while they can at TMI-1. Also, the shut-off head of the CR-3 DH pumps is less than that assumed in the original safety analysis and this requires a longer time delay on the HPI system in the lower range of a postulated LOCA. TMI-1 has the same DH pump but does not have this limitation.

**Recommendations**

5A. The panel recommends and CR-3 has decided to install design upgrades to the HPI system to enhance operating margin in the event of certain postulated transients.

5B. The panel recommends and CR-3 has decided to conduct a thorough investigation to determine why similar balance-of-plant design features are limiting at CR-3 and not at TMI-1.

5C. FPC should complete their review of the July Framatome Technologies (FTI) report and take any appropriate action.

***Question 6: Is the coordination and cooperation among the B&W owners on design basis issues appropriate?***

**Scope of Review**

Discussion among Panel members and with FPC personnel
B&WOG Long Range Plan

**Observations**

While there is strong cooperation among B&W plant owners, there is little to no cooperation specifically on design basis issues either formally through the B&W Owners Group or informally.

A major benefit to FPC of the panel's work has been the knowledge gained from learning the approach of other B&W owners to the design basis issue.

### Recommendations

6A. FPC should take action to have the Steering Committee of the B&W Owners Group add the subject of Design Basis to its regular meeting agenda. Significant value could be gained by developing a more standardized approach to this important, emerging issue of design basis validation and control.

6B. The Panel recommends that when dealing with design basis issues, FPC take greater advantage of discussions with the other B&W utilities.

***Question 7: Is there adequate knowledge and understanding of the design bases information within the nuclear organization?***

### Scope of Review

The panel conducted interviews with a number of plant personnel representing licensing, design, systems, operations, and plant management. Responses were compared with documents referred to in the interviews, primarily including procedures CP-111, *Initiation and Processing of the Precursor Cards and Problem Reports* and CP-150, *Identifying and Processing Operability Concerns*, and NEP 216, *Plant Design Basis Documents*. The panel pursued two avenues to address this subject. One avenue was to determine if personnel had an adequate understanding of what constituted design bases information. The other avenue was to determine whether personnel are knowledgeable of and adequately understand the specific content of the design basis information available to them regardless of whether or not it was identified as design basis documentation in NEP-216, *Plant Design Basis Documents*.

A detailed technical review was not performed by the panel.

Interviews (focused primarily on systems and design engineers) included the following types of questions to gauge engineers' familiarity with design bases information:

* Explain how your work requires the use of design bases information.
* What are your primary references or sources for determining design bases?
* What is your confidence level in those sources?
* How much benefit would you expect to gain from design bases training?

* When drawing conclusions about design bases, what level of confidence do you have that your conclusion is solid and defensible?
* Are there particular weaknesses in design bases information that need improvement?

**Observations**

The panel concluded that, other than in a few specific areas, design bases information is well understood by systems and design engineering personnel. The following points summarize what was concluded from panel interviews.

* There was broad agreement that the combination of EDBDs, FSAR, and Technical Specifications (along with occasional reliance on other sources) provides a high confidence level that most any design bases issue can be completely addressed. EDBDs and Tech Specs are generally perceived as being the most accurate and/or complete of the primary sources for design bases information. Note under Question 1 above that the FSAR and Tech Specs are not included in NEP-216 definition of Design Basis.

* Design bases issues that are most difficult to address are where early information and assumptions were not thoroughly documented during early plant design. It was also noted that piping support design bases and non-safety ventilation system information in particular was not well documented.

* Examples of engineers' work that were reviewed demonstrated a good grasp of the relevant design bases associated with the assignment. Several clearly also understood the larger picture that distinguishes between descriptive information, commitments, and legal requirements contained in the FSAR, the EDBDs, and the Technical Specifications.

* Engineers have a good depth of understanding of design bases information. Most design and systems engineers have more than 5 years experience. Engineers believe that training programs are helpful for orienting engineers to available information. However, the worthwhile understanding only comes from experience with the issues.

The Panel addressed the knowledge and understanding of individuals outside the engineering organization to only a limited extent. Panel Recommendation 7A below calls for initial and ongoing training on Design Basis for all appropriate personnel.

Recommendations

7A. The training on the design basis and the use of design basis information should be delivered to the appropriate individuals in organizations outside the engineering sections, and it should be covered in continuing training programs for those groups.

7B. Develop guidance for operations' personnel that establishes expectations for their review of engineering analyses and calculations. It should be clear that the operations' review is to validate input assumptions related to how the plant is operated and to review outputs for effect on plant operation, not to provide a full engineering verification.

*Question 8: Is the design basis information adequately disseminated to those who need to use it?*

Scope of Review

Interviews were conducted with plant personnel in design engineering, configuration management, system engineering, operations, maintenance, and licensing to determine how design basis information is disseminated and used. Additionally, the availability of design information in various plant documents was evaluated. Documents reviewed included design basis documents, enhanced design basis documents, analysis basis documents, topical design basis documents, the final safety analysis report, technical specifications, engineering calculations and analyses, core operating limit reports, technical basis documents for emergency operating procedures, and various references contained in design basis documents.

Observations

Design basis documents and design information are available to station personnel who need them. However, some weaknesses exist regarding the accuracy, timeliness, and appropriateness of the available information. These weaknesses are addressed in the response to review Question 11. Also, the scope of design basis documents is not consistently and clearly defined. Issues regarding the definition of design basis documentation are addressed in the response to Question 1.

Controlled paper copies of the design basis documents as defined by NEP-216, the FSAR, and the Improved Tech Specs are easily accessible by personnel in various designated plant departments. This includes nuclear engineering design, nuclear configuration management, nuclear plant technical support, operations, maintenance, and licensing. However, other such design/licensing bases documents as the COLR, the Setpoint Basis, EOP Design Bases Documents, etc. are not included at these designated locations. Only the Enhanced Design Bases Documents, FSAR, and the Improved Tech Spec are available electronically via the FulText Program for search of design and license bases information via the plant's computer network. References in the Design Basis and Enhanced Design Bases Documents, such as calculations, memoranda, and other documents, are maintained in hard copy titled "Field Validation Reports" and on microfilm in the nuclear engineering design department. However, some proprietary references such as the Framatome/B&W type 52 documents are not available at the station.

**Recommendations**

8A. Consider including all design basis documents (see question 1) at the designated plant locations for design bases documents and electronically on FulText.

8B. Obtain controlled copies of the Framatome/B&W type 52 reference documents for use by station engineering personnel with appropriate proprietary information protection.

***Question 9: Is the process for implementing plant design modifications adequate to ensure the design bases are maintained?***

**Scope of Review**

Interviews were conducted with managers, supervisors, and engineers in the nuclear engineering design and the nuclear configuration management departments. The station's procedure for implementing design modifications, NEP-210, *Modification Approval Records*, was reviewed to determine the adequacy of processes for identifying and implementing revisions to design basis information and documentation.

**Observations**

Generally, the process for implementing modifications is adequate to ensure that the design basis is maintained. However, some enhancements regarding engineering work management practices and the verification of design engineering

work should be considered to improve the process. Additionally, the timeliness of changes to design basis documents needs to be improved. See Question 11 regarding timeliness.

Procedure NEP-210, *Modification Approval Records*, provides adequate guidance in exhibit 5, item 20 that a revision to the plant design basis documents is required if the modification changes a component/system function, functional requirement, or design requirement as delineated in the existing design basis document.

A number of engineering personnel indicated that the quality of engineering design work and associated verification reviews have been adversely affected by insufficient time for these activities due to work loads and conflicting work priorities. Additionally, engineering has not used the verification review process to identify and trend performance errors in engineering work. Reviews of CR-3 LER's in the 1992 to early 1996 time period indicate that approximately 45% of the LERs reported as design bases issues involved problems associated with analyses and their review.

Pending changes to Design Bases Documents due to plant modification are controlled by temporary changes per the process required by NEP-216, Section V.D. A spot-check of design bases documents showed that "Temporary Change" notices were attached to many of the sections of design bases documents. The timeliness of incorporating these changes into DBD is covered in Question 11. Work management practices have also affected the timeliness of revisions to design basis documents.

Additionally, a plant Design Review Panel was established several years ago to review modification packages. However, the review group has not been effective in identifying problems with many modifications. The station has recently assigned more senior plant personnel to this panel.

**Recommendations**

9A. FPC management should revise resource planning and work management practices to ensure that adequate time and priority is provided to perform high quality engineering work and reviews. This should include an effective screening process to cancel engineering work requests of marginal value and to assign appropriate priority and schedules to all engineering work.

9B. Performance errors identified during design verification reviews should be monitored and remedial training and corrective actions provided for repetitive problems.

9C. Establish additional expectations for the plant's Design Review Panel to ensure that design basis implications of the modification have been adequately reviewed and considered.

*Question 10: Is the process for making procedure changes adequate to ensure design basis information in the procedures is appropriately maintained or revised?*

Scope of Review

Interviews were conducted with station personnel involved with procedure revision and review in the operations, maintenance, nuclear engineering design, and nuclear plant technical support departments. The adequacy of processes in administrative instructions AI-400C, *New Procedures And Procedure Change Processes*, and AI-400F, *New Procedures And Procedure Change Processes For Emergency Operating Procedures (EOPs), Abnormal Procedures (APs), and Verification Procedures (VPs)* were reviewed to determine if design basis information is appropriately maintained or revised.

Observations

Generally, the process for making procedure changes is adequate to ensure that design basis information in the procedures is appropriately maintained or revised. However, some enhancements and improvements should be considered to ensure that engineering is appropriately involved in reviewing the effects of procedure changes on design parameters and assumptions.

Guidance for changing normal station procedures is provided in administrative instructions AI-400C, *New Procedures And Procedure Change Processes*. AI-400C requires that the originator of a procedure change and an independent qualified reviewer determine if the change can affect the plant design basis. The originator is guided in this determination by enclosure 11 of AI-400C that asks if there are any changes to design basis document parameters. The independent reviewer is often another member of the originator's department. If the procedure change affects any design condition or design requirement, then an additional qualified review by nuclear engineering design is required. A weakness in the process is that engineering personnel are not necessarily involved in determining if the change can affect the design basis.

Procedure AI-400F, *New Procedures And Procedure Change Processes For Emergency Operating Procedures (EOPs), Abnormal Procedures (APs), and Verification Procedures (VPs)*, is structurally different from AI-400C and it requires that all EOP changes have a qualified review from nuclear engineering design.

However, AI-400F does not require a qualified review by nuclear engineering design in procedure for changes to abnormal and verification procedures. The decision for an engineering review is left to the originator of the procedure change. Also, procedure AI-400F lacks the guidance provided in Enclosure 11 to procedure AI-400C, Enclosure 11, for assisting the originator in identifying if the procedure change affects design conditions or requirements.

Only 2-3 engineering personnel in each engineering section have been trained as qualified reviewers for procedure changes. In some engineering groups, only the supervisor is a qualified reviewer. However, any engineer may perform a technical review of a procedure change for a qualified reviewer. The qualified reviewer signs that it is correct. Weaknesses in the process are that engineering design personnel have little specific guidance or training in the review of procedure changes and persons with lesser qualifications are allowed to perform qualified reviews.

**Recommendations**

10A. Consider revising AI-400C to require a qualified review for all procedure changes by either nuclear plant technical support (system engineering) or nuclear engineering design. Alternatively, consider adding to AI-400C more prescriptive conditions for when a qualified engineering review should be required.

10B. Revise AI-400F to require a qualified review by nuclear engineering design for abnormal procedures and verification procedures as well as for emergency operating procedures.

10C. Include the Enclosure 11 guidance from AI-400C in AI-400F to assist the originator in determining if the procedure change affects design conditions or design requirements. Consider revising Enclosure 11 to be a check-off list to ensure consideration of design basis issues. Provide guidance to identify on the check-off list the specific sections of the FSAR, technical specifications, COLR, or design basis documents that may require revision.

10D. Develop specific guidance and training for engineering personnel for conducting technical reviews of procedure changes. This should include emphasis on evaluating the effect of procedure changes on design basis parameters and design requirements and on identifying possible revisions of the FSAR, technical specifications, COLR, and other design basis documents.

*Question 11: Is the process for maintaining the design basis accurate, up-to-date, and appropriate? For non-modification changes to design bases (e.g. calculations, drawings, figures, tables, text, etc.), is there a process for adequately incorporating those changes into the FSAR and design basis documentation?*

## Scope of Review

Interviews were conducted with station managers, supervisors, and staff in the nuclear engineering design, nuclear plant technical support, nuclear configuration management, and nuclear licensing departments. Relevant documentation and station procedures reviewed included procedure NEP-216, *Plant Design Basis Documents*, NEP-210, *Modification Approval Records*, NEP-271, *As-Building Of Modification Approval Records*, NEP-213, *Design Analyses/Calculations*, NOD-11, *Maintenance Of The Current Licensing Basis*, various station problem reports and LERs, design basis documents, the enhanced design basis documents, the analysis basis documents, topical design basis documents, the FSAR, technical specifications, core operating limit reports, and technical basis documents for emergency operating procedures.

## Observations

Some weaknesses and areas for improvement were noted regarding the timeliness, accuracy, and appropriateness of some design basis information. Also, some enhancements regarding the processes for incorporating changes to design basis documents and the FSAR should be considered. A factor contributing to these weaknesses is that the scope of design basis information is not consistently and clearly defined. Issues regarding the definition and scope of design basis documentation are addressed in the response to Question 1.

In some cases, design basis documents are not updated in a timely manner. The following examples were noted:

- The station's process for revising design basis documents allows up to 150 days for issuing a temporary change to the document following implementation of a plant design change. A review of 147 design basis document changes indicated that for 52 of these, the temporary change to the document was issued after a modification was installed and operable. Of these, 25 design basis document temporary changes required greater than six months to be issued, 20 required greater than one year, 11 greater than 2 years, and 7 greater than 3 years. After a temporary change to a design basis document is issued, it often takes two years before the change is incorporated as a permanent revision to the document.

- The nuclear engineering and projects procedure NEP-216, *Plant Design Basis Documents*, does not provide specific guidance for the time required to revise design basis documents following implementation of a plant modification or procedure change. Some design engineers stated that they were unaware of any required time. Others stated that NEP-210, *Modification Approval Records*, allows a modification to be implemented with design basis document changes remaining as an open item. In this case, procedure NEP-271, *As-Building Of Modification Approval Records*, allows completion of the open items within 150 days.

- A July 1996 station problem report, PR 96-0230, noted two examples where the guidance in procedure NEP-216, *Plant Design Basis Documents*, for revision of design basis documents was not followed. In one example, a temporary change to the makeup system design basis document has not been issued for a plant modification installed in August 1993 that changed the hydrogen addition pressure regulator setpoint from 10 psig to 19.5 psig. In another example, the required 12-month review of the design basis documents had not been performed and documented. This review is intended to formally revise the design basis documents every two years if any outstanding temporary changes exist from the previous two years. Six temporary design basis document changes had not been incorporated within the required two year time period.

- Several design engineering personnel indicated that they experience difficulty in issuing timely revisions to design basis documents due to work loads and higher priority tasks.

A number of station personnel commented that the currently defined design basis documents when taken alone contain inaccuracies and that additional review is needed to ensure that information is correct when performing design activities or analyses. The design basis documents represent a compilation of available design information and references some of which have not been validated. Also, some original design basis information, such as the design bases for piping, structures, and pipe supports, is generally not available. A review of the design basis documents for the spent fuel pool cooling system and the fuel handing system indicated inconsistencies and confusion between the values and bases in these documents and the plant's technical specifications regarding the boric acid concentration of the spent fuel pool. The following inconsistencies were noted:

- The design basis document for the spent fuel pool cooling system states that boric acid concentration must be greater than 1925 ppm to ensure that $k_{eff}$ is less than or equal to 0.95 for sub-criticality. Current

licensing requirements state in one section of the IT that the pool must be sub-critical with no boric acid present and then other parts of the same section state boron is required for fuel drop accident at 1925 ppm (B3.7.14).

- The design basis document for the fuel handling system states that the water boron concentration is 3400 ppm.

- Technical specification 3.7.14, for spent fuel pool boron concentration, requires the concentration of dissolved boron to be greater than 1925 ppm based on maintaining sub-criticality during postulated events such as a dropped assembly. Technical specification 3.9.1, for refueling boron concentration, states that boron concentration shall be maintained within the limit specified in the core operating limits report. The COLR report specifies a value of greater than 3013 ppm for refueling.

- ITS 3.5.4 requires the BWST boron concentration to be > 2,270 and < 3,000 max.

Also, it was noted that the process for plant modifications or procedure changes requires identification of any needed revision to the FSAR or technical specifications, however, the COLR is not specifically covered. Changes to Design Bases Documents or the FSAR and ITS as the result of new limits and requirements in the COLR do not appear to be covered. Changes to the FSAR implemented by Licensing per NOD-11 do not necessarily require a review by engineering to assess the need to revise design basis documents. An on-going review of the station's FSAR by plant personnel has identified a number of errors and inconsistencies between the actual plant configuration and the FSAR and these are being addressed by the plant staff.

Procedure NEP-213, *Design Analyses/Calculations*, does not provide specific guidance or a check list to revise design basis documents based on the results of analyses or calculations. This contributed in part to the missed design basis document revision for the change to the hydrogen addition pressure regulator setpoint in August 1993 for the makeup system. The engineer performing this calculation forgot to revise the design basis document for the system. As indicated above, analyses for the Core Operating Limits Report performed per procedure NEP-213 do not routinely consider the effect on design basis documents.

The only station guidance for controlling and maintaining plant design basis documents, NEP-216, Plant Design Basis Documents, is an engineering department procedure. Other station groups such as licensing and operations do not have specific procedural guidance for maintaining design basis information.

Recommendations

11A. Engineering management should clearly communicate and reinforce the need to revise design basis documents promptly following implementation of a plant modification or procedure change. Procedure NEP-216, *Plant Design Basis Documents*, should be revised to provide specific guidance for the maximum time allowed to make temporary changes to the design basis documents and to incorporate temporary changes as permanent revisions. Other nuclear plants typically require permanent revision of design basis documents within 30 - 60 days of implementing a plant change.

11B. Revise procedure NEP-213, Design Analyses/Calculations, to provide specific guidance or a check list for ensuring that design basis documents are reviewed and revised as necessary as a result of an analysis or calculation.

11C. NOD-11 should be revised to include specific guidance for ensuring that FSAR changes are reviewed by engineering to identify and then make any appropriate revisions to design basis documents.

11D. FPC should consider promulgating a procedure for the control of design/licensing basis information and documentation that applies to the entire Crystal River 3 nuclear organization. This procedure should require any change to the plant, plant documentation or procedures be evaluated for effect on design basis information and documentation.

11E. To help ensure that inaccuracies in design basis information and documentation are identified and corrected over time, establish clear expectations that engineering personnel are accountable for reviewing and correcting design information and requirements. Also establish that all personnel are expected to report design basis deficiencies or questions so they can be addressed.

11F. FPC should proceed with the planned selective SSFIs to further assess and insure the adequacy of the design basis information.

*Question 12: Is there an adequate process for assessing the cumulative effect of design basis discrepancies on plant safety?*

**Scope of Review**

AI-300, *Plant Review Committee Charter*; Paul Tanguay to Pat Beard interoffice correspondence dated June 27, 1996.
Discussions with FPC personnel

**Observations**

Two assessment types were reviewed that could be used for this purpose, but neither have included the review of design basis discrepancies within their scope. They are the Nuclear General Review Committee (NGRC) annual assessment and the Quarterly Manager Assessment. In addition, the Plant Review Committee (PRC) Charter was reviewed and did not contain this type of assessment within the scope of its activities.

The assessment of the 44 more recent "design basis" LERs, performed by a team led by Paul R. Tanguay, was intended to provide a determination regarding the cumulative effects of those design basis discrepancies on plant safety. The assessment resulted in some insightful conclusions and provided several recommendations. However, it contains no specific conclusion as to the cumulative effect of the issues in these LERs.

**Recommendations**

12A. Station management should consider including within the NGRC annual assessment, the Quarterly Manager Assessment, or PRC reviews an assessment of the cumulative effect of design basis and design discrepancies on plant safety.

12B. The assessment of recent design basis LERs (Tanguay report)on plant safety should be revised to include an explicit conclusion regarding the cumulative effect of the 44 LERs reviewed.

*Question 13: Is the FPC 10CFR 50.59 and Safety Review process consistent with industry practices and regulatory guidance?*

**Scope of Review**

NOD-11, *Maintenance of the Current Licensing Basis*; AI-400C, *New Procedures and Procedure Change Requests*; MAR 96-02-09-01, HPI Flow Upgrade
Discussions with FPC personnel

**Observations**

The program, as described in station procedures, largely follows normal industry practices for 10CFR50.59 evaluations (i.e. adopts NSAC 125 guidance). However, it appears that FPC Safety Evaluations (SEs) are not stand-alone documents. Rather, SEs seem to be documents prepared by individuals who are familiar enough with the associated change package that summary paragraphs are deemed sufficient to establish an unreviewed safety question (USQ) determination. This is quite different from the practice used at some other plants, where a SE is virtually stand-alone and the thought processes leading to the summary paragraphs are detailed enough that the reviewer and approver can make a judgment without having to have the entire change package at hand.

It also is not evident that a mechanism exists for routinely assessing the SE process. As such, no performance indicator is available to management for periodically assessing the overall quality of SEs performed and whether management expectations for the process are being met.

There does not appear to be a station requirement for specific qualification or training of individuals in the SE process. This qualification and training is addressed on a department-specific basis, as determined by each manager. For example, demonstrating the ability to perform a SE is required by most of the Engineering Support Personnel position-specific qualification guides. Engineering supervisors are given the responsibility for verifying satisfactory completion of this qualification task.

The population of personnel qualified to perform, review, and approve SEs is not limited by management to those who are required to routinely perform these tasks. Essentially all engineering support personnel are allowed to perform SEs. Therefore, it is possible that SEs could be performed by personnel not proficient in the process.

## Recommendations

13A. FPC should establish a "stand alone" Safety Evaluation (SE) format that requires an integrated discussion of the proposed change, its effect on safety, and the USQ determination.

13B. A subcommittee of the NGRC should review SEs after the fact sufficiently to address quality, trends, and issues and report the results to the full NGRC.

13C. Consider limiting the number of personnel allowed to perform, review, and approve SEs. This should result in sustaining a level of proficiency in the process and thereby improving the overall quality of SEs.

13D. Develop a station SE training and qualification program that will result in a consistent understanding of SE process requirements, by all personnel who perform, review and approve SEs.

***Question 14: Is there initial and periodic training on understanding the relationships and implications between 10CFR50, design bases (50.2), industry codes and FPC's administrative programs controlling configuration and design?***

## Scope of Review

TDP-308 Training Needs Assessment Study, December, 1993; Tech Staff and Manager Training Program position specific qualification guides; INPO final accreditation evaluation report, December, 1995; Phase 3 curriculum guides and lesson plans for 1993-1996. Discussions with FPC personnel

## Observations

The FPC Engineering Support Personnel Training Program is divided into three (3) phases. Phase 1 is orientation training and addresses the above subject areas under administrative training topics. Phase 2 is position-specific training and qualification. The position-specific qualification guide lists the tasks and references that are used as standards for performance of each task. Completion of a position-specific training and qualification guide does require knowledge in the subject areas. Phase 3 is continuing training for engineering support personnel. Phase 3 training has provided limited coverage of the subject topics. For the period of 1993 to 1995, it included the following tropics relevant to this question;

Enhanced Design Basis Document (1995)
Probabilistic Safety Analysis (1995)
Operability Determination (1995)
Reportability Requirements(1995)
10CFR50.59 (1993, 1996)
FSAR (1996)

Implementation of the Engineering Support Personnel Training Program is the responsibility of its Curriculum Review Committee (CRC). This committee is comprised of the relevant engineering managers and chaired by the Engineering Director. Discussions with personnel responsible for training coordination, within engineering, indicated that there did not exist complete ownership of the program. The engineering workload was so great that training was not frequently given priority.

The CRC specifically is responsible for selecting topics for the Phase 3 curriculum. Phase 3 training is designed to cover a period of one year, therefore, its curriculum is developed based on an annual input from the CRC.

**Recommendations**

14A. The plan for the annual Engineering Phase 3 (continuing) training should be broken into three or four sessions and the Curriculum Review Committee should provide topic input to each session. This would allow Phase 3 training to be responsive to current issues.

14B. The Phase 3 continuing training curriculum should be more reflective of current design basis and licensing basis issues.

14C. Reinforce engineering ownership of its design basis training program. (See Recommendations 5 and 13 on training for areas other than engineering.)

***Question 15: Is there a process to evaluate the effect of using new or different design criteria that are in addition to or go beyond the existing plant design bases?***

**Scope of Review**

The documents reviewed include NOD-52, *Commitment Processing and Management of Programmatic Commitments*, CP-111, *Initiation and Processing of the Precursor Cards and Problem Reports*, **CP-150, *Identifying and Processing Operability Concerns*** and administrative procedures AI-404A, *Review of Technical Information* and AI-404B, *Review of Industry Operating Experience*

Discussions with FPC personnel.

**Observations**

Florida Power Corporation has processes to evaluate changes to programmatic commitments, but those processes do not require identifying whether or not the change is new or different (Revision B of criteria versus Revision A of criteria). The processes are simply silent as to the exact nature of the change. The processes don't directly ask, "How does this new change alter the plant's Design Bases?".

Crystal River's Nuclear Operations Document 52, *Commitment Processing and Management of Programmatic Commitments* (NOD52, dated 1/31/96) identifies the process by which Florida Power Corporation (FPC) manages and processes programmatic commitments. On Page 9, Item D, Item (a) discusses determining if there is a codified change process and that in turn points to provisions of 50.59 or 50. 54; Item (b) covers determining whether or not the change is safety significant. Item (c) discusses determining if the original commitment is necessary to achieve compliance with an obligation. Item (d) discusses determining if the NRC relied upon that original commitment being considered for change. Item (e) discusses determining whether the original commitment was made to minimize recurrence of an adverse condition.

Crystal River's Administrative Procedures AI 404 A& B pertain to the review of technical information and refer to CP-111, *Initiation and Processing of the Precursor Cards and Problem Reports* and CP-150, *Identifying and Processing Operability Concerns*.

a) Administrative Procedure AI-404A prescribes the method by which incoming information received by FPC is to be reviewed and the process by which action items are triggered as the result of that review. Close scrutiny of AI-404A does not show that incoming information would be reviewed for impact on the Design Bases unless the reviewer has the intuition to believe it might be so or if in a subsequent review (see Enclosure 2 of AI-404A), one of the other organizations determined that the information might affect the Design Bases. AI-404A in and of itself does not call for addressing the question of whether or not the design bases would be affected as a result of the receipt of the information.

b) Administrative Procedure AI-404B specifies, in Paragraph 3.3.1 on Page 4, that the procedure intends to provide a thorough review to identify items which may affect design. Paragraph 4.2.2.7.1 requires (for those items for which the reviewer has checked Tech Spec block) review of the Technical Specifications as well as review of reference documents. This could trigger

recognition and understanding of a Design Bases issue. The knowledge and sensitivity of the people making the call relative to Design Bases issues is crucial to this conclusion.

**Recommendations:**

15A. Change NOD-52 to clearly identify Design Bases as an important element of the overall Configuration Management Program, and to require actions to ensure thorough assessment and treatment of changes, including the supporting or underlying analyses and assumptions thereto, as well as hardware issues (Structures, Systems and Components).

15B. Change procedures AI-404A and B to ensure Design Bases issues are clearly considered, identified, and thoroughly treated.

15C. Change the Modification Approval Record (MAR) process document NEP-210 to ensure Design Bases issues are clearly considered, identified and thoroughly treated consistent with and similarly to NOD-52 and AI-404 A and B.

*Question 16: Are the current processes for determining operability, notifying the NRC, and reporting (10CFR50.73) design bases issues consistent with industry practice and in compliance with regulatory requirements?*

**Scope of Review**

CP-111, *Initiation and Processing of the Precursor Cards and Problem Reports* and CP-150, *Identifying and Processing Operability Concerns*.
Discussions with FPC personnel

**Observations**

Florida Power Corporation's approach in years prior to 1996 is described in Compliance Procedure CP-111, while in more recent months (Mid 1996) CP-150 is being used.

Under CP-111, ("Responsibilities," Paragraph 3.2 on Page 9), the originator can be any individual, Florida Power Corporation employee or contract employee, who observes or becomes aware of a concern. The procedure directs that individual to document the concern, incident, or condition. Further, it obligates whomever

found this item to notify the Shift Manager immediately. (Note emphasis on timeliness.) The Nuclear Shift Manager reviews the issue, initiates and assigns the problem report. The Nuclear Shift Supervisor on duty (who may be involved in concurrent significant plant operating issues) reviews the problem report to assure proper implementation of Tech Spec actions. Then, the Shift Technical Advisor determines reportability, the requirements of the problem report, and performs the appropriate notifications. This requirement for determination by the single on shift STA is different from what is done elsewhere. Elsewhere in the industry, the Shift Technical Advisor (or others involved in similar processes) advises the Shift Supervisor/Shift Foremen or Operations Manager/Director, but does not have the duty to initiate 10CFR50.72 one-hour phone call. It is important to recognize that the notification (50.72) nearly always commits the company to subsequent (50.73) reportability. To be more consistent with industry practice the procedure should guide the Shift Technical Advisor to seek assistance in determining the need to make a notification should the conditions warrant. In addition, Procedure CP-111 covers a wide variety of events, from routine to major, and it places responsibility on each employee who has a concern to evaluate whether it is a safety issue, a reportability or an operability question (Paragraph 3.2.1). Some level of review or supportive screening so as to assist in clarifying the impact on plant safety by knowledgeable personnel before the item is brought to the Shift Manager seems appropriate.

The recently issued CP-150 (5/7/96) is now being used. CP-150 gives recognition to levels of risks. (See page 5 and 6). The statement of the operability philosophy, Paragraph 3.3.1,directs the SSOD to make an immediate disposition with the information in hand as conditionally operable, potentially inoperable, or inoperable, and directs appropriate compensatory actions and justification for continued operation.

The process in CP-150 allows some degree of flexibility for the SSOD in making operability calls and allows consideration of the risk issue. This addresses the comment made in preceding paragraphs relative to CP-111 (above).

It appears that Florida Power Corporation has made important changes in its approach towards determinations of operability. While the upgraded process seems to be a major improvement, it may need further enhancements to enable the process to objectively take credit for levels of redundancies, availability of backups, and other appropriate risk reduction measures to avoid declaring the plant is outside its design envelope when it isn't.

Recommendations

16A. Ensure procedure CP-150 has an appropriate "process" to ensure that only appropriate items are notified and reported.

16B. Ensure procedure CP-150 includes appropriate guidance from NUREG-1022 to take advantage of the allowed time frame to evaluate an issue prior to notification.

16C. Revise CP-150 to be consistent with industry practice for the shift supervisor to seek assistance in determining the need to make a notification.

*Question 17: Is there a formal management oversight of Design Base Programs including accountability to ensure program adequacy and taking effective actions to prevent recurring events?*

Scope of Review

Interviews with FPC staff and management.
Recent Managers Assessments
Recent Safety Assessment Department Annual Review

Observations

Crystal River does not presently have a defined process for management oversight of the Design Bases and related program. The station has not established an effective oversight process to ensure that design basis information is adequately considered and that changes to design requirements are promptly reflected in design basis documentation.

FPC does have Quality Action Plans (by division and by department) as corporate initiatives which are headed in the right direction. The annual review, conducted by the Safety Assessment Department, is intended to provide general oversight but it does not directly address Design Bases issues. The Manager's Assessment also tries to get to this but does not directly address it.

Additionally there is not, at the present time, an overall periodic audit of Design Bases Programs.

Recommendations

17A. FPC should recognize the importance of Design Basis and Licensing Basis and take steps to treat them as major programs with defined scope, clearly assigned ownership, and recognition throughout the organization.

17B. FPC undertake action to periodically assess the adequacy of the Design Bases Program and to report on it in a manner that is visible to management and requires organizational response.

# APPENDICES

Appendix A

# Crystal River Unit 3
# Independent Design Review Panel

## CHARTER

Background:

Over the last several years, CR-3 has reported many design related issues, apparently more than industry experience. Therefore, Florida Power Corporation decided to perform this review.

Purpose:

Perform an independent assessment of the Crystal River Unit 3 Design Bases and the adequacy of the design bases management control processes to provide reasonable assurance of safe plant operation.

Identify other related areas that warrant further consideration.

Provide recommendations for improvement.

Scope:

The scope of the review should address the following areas and other areas pertinent to the purpose.

Assessment of Design Bases

1. Is FPC's interpretation of "design bases" as stated in 10CFR50.2 consistent with the industry?
2. Is FPC's design bases documentation consistent with industry practices and regulatory guidance in defining the requirements, limits and parameters for the systems and components?
3. Are past and current upgrade programs adequate to give reasonable assurance that design bases issues affecting safe shutdown of the plant will be discovered?
4. Are past and current upgrade programs adequate to give reasonable assurance that plant operating documentation is consistent with the design bases?
5. Are there generic design bases issues from other B&W plants that FPC needs to address?
6. Is the coordination and cooperation among the B&W owners on design bases issues appropriate?
7. Is there adequate knowledge and understanding of the design bases information within the nuclear organization?

Process for Assuring Consistency of Design Bases and Plant Documentation

8. Is the design bases information adequately disseminated to those who need to use it?

9. Is the process for implementing plant design modifications adequate to ensure the design bases are maintained?

10. Is the process for making procedure changes adequate to ensure design bases information in the procedures is appropriately maintained or revised.

11. Is the process for maintaining the design bases accurate and up to date, appropriate? For non-modification changes to the design bases (e.g., calculation, drawings, figures, tables, text, etc.), is there a process for adequately incorporating these changes into the FSAR?

Management of Control Processes

12. Is there an adequate process for assessing the cumulative effect of design basis discrepancies on plant safety?

13. Is the FPC 10CFR50.59 and Safety Review process consistent with industry practice and regulatory guidance?

14. Is there initial and periodic training on understanding the relationships and implications between 10CFR50, design bases (50.2), industry codes and FPC's administrative programs controlling configuration and design?

15. Is there a process to evaluate the effect of utilizing new or different design criteria that are in addition to or go beyond the existing plant design bases?

16. Are current processes for determining operability, notifying (10CFR50.72), and reporting (10CFR50.73) design bases issues consistent with industry practice and in compliance with regulatory requirements?

17. Is there formal management oversight of design bases programs, including accountability to ensure program adequacy and taking effective actions to prevent recurring events?

Review Material:

1. 4 years of Design Bases LERs

2. SSFI and system audits conducted since 1989

3. Design Bases Control Procedure

4. Design Bases Documents

5. Outline of Design Bases accidents (not just FSAR Chapter 14 accidents, but all accidents that CR-3 must be designed to handle)

6. Engineering Audits for last 4 years

7. NRC Information Notice 96-17

8. NRC Inspector General Report - Millstone Spent Fuel Cooling

9. NRC Policy Issue - Design Document Reconstitution Programs Initiated by Utilities, SECY-90-365

10. NUMARC Design Basis Program Guidelines, NUMARC 90-12

11. NRC Inspection Module for Design Bases Review

12. Problem Reports associated with Design Bases LERs

13. Sampling of 10CFR50.59 reviews and safety evaluations

14. Functional Organization Chart

15. Safety Listing

16. EOP Upgrade Action Plan

17. Improved Technical Specifications

18. Action Plan for 18 to 24 month surveillance program

19. Power Level Upgrade Documents

20. CR-3 Time Line History

21. NRC Inspection Manual Part 9900 (draft)

Team:

The Senior Vice President, Nuclear Operations, shall appoint a team composed primarily of external professionals to perform the assessment. Team members should have knowledge and experience with commercial nuclear power plants of the same vintage as CR-3. Expertise shall be in the areas of design engineering, plant operations, licensing and management/organizational analysis.

The team leader and the team members shall have no vested interest in the results of the evaluation.

There should be an FPC representative assigned as full-time liaison. Other FPC personnel may be assigned on an as-needed basis.

Team Composition

Team Leader: Recognized leader in nuclear power industry
Team Member: Knowledgeable of AE design practices
Team Member: Knowledgeable of NSSS design practices
4 Team Members: Engineers from other B&W units knowledgeable of utility design practices
Liaison contact with Parsons (Gilbert/Commonwealth)
Liaison contact with Framatome Technologies, Inc.
Liaison with INPO

Duration of the Review:

The review shall be performed between May 15, 1996, and November 15, 1996, with a final report provided to the Senior Vice President on or before November 18, 1996.

P Beard Jr — 6/1/96

P. M. Beard, Jr. — Date

# BIOGRAPHIES

## PHIL CLARK

Philip R. Clark, a 42-year veteran in the field of nuclear power, retired in December 1995 from the position of president and chief executive office of GPU Nuclear Corporation. GPU Nuclear is a wholly owned subsidiary of General Public Utilities Corporation and operates the Oyster Creek Nuclear Generating Station in New Jersey and Three Mile Island Unit 1 in Pennsylvania and was responsible for the cleanup of the damaged Unit 2 reactor at Three Mile Island.

He was also a director of GPU Nuclear, GPU Service Corporation, Saxton Nuclear Experimental Corporation, and the Institute of Nuclear Power Operation (INPO), a non-profit organization of utilities in the United States devoted to excellence in commercial nuclear power operations.

He was elected to the National Academy of Engineering in 1993 and is a fellow of the American Nuclear Society.

Before retiring from government service in August 1979, Mr. Clark worked as associate director, Reactors, Naval Reactor Division, U.S. Department of Energy, and as chief, Reactor Engineering Division, Nuclear Power Directorate, Naval Sea Systems Command, Department of the Navy. In these positions, he reported directly to Admiral Hyman G. Rickover and directed a major element of the U.S. Naval Nuclear Propulsion Program.

He received the Navy Distinguished Civilian Service Award in 1972 and the U.S. Energy Research and Development Administration Special Achievement Award in 1976.

Mr. Clark earned a Bachelor's Degree in Civil Engineering in 1951 from Polytechnic Institute of Brooklyn, NY, where he also did graduate study in 1951-1953. He attended Oak Ridge School of Reactor Technology in 1953-54.

NOMAN M. COLE, JR.

Mr. Noman Cole is a principal of MPR Associates and since 1955 has been involved in the power production field, working with both fossil and nuclear plants for central power stations and marine applications. These experiences have involved a wide range of technologies, systems engineering and project management activities in both the military and civilian power production fields. This experience included serving as a project manager for a developmental submarine project reporting directly to Admiral H. G. Rickover. This experience covered the design, fabrication and operation of reactors, material irradiation facilities, expended core storage and disposal facility, reactor refueling and fuel handling systems, spent fuel shipping casks. This experience has involved many projects dealing with high levels of radioactivity such as steam generator and reactor internals repairs, TMI-2 defueling and sampling the TMI-2 reactor vessel after the accident. Mr. Cole has also been a member of management review boards at TMI-2 after the accident, Davis-Besse, Fort Saint Vrain Decommissioning, Crystal River #3 and Chernobyl stabilization.

Mr. Cole attended the University of Florida and earned a B.S. degree in Mechanical Engineering in 1955. In 1957, he graduated from Reactor Engineering School, Bettis Laboratory, Naval Reactors, U.S. AEC.

HUGH A. HAMMOND

Mr. Hugh Hammond has over sixteen (16) years experience in the nuclear industry; twelve of these have been in direct support of Duke Power's Oconee Nuclear Station. He is currently serving as the project manager of an effort to clarify QA scope and design basis issues for Oconee and between Duke and the NRC. Hugh's previous Oconee experience includes design engineering and licensing support, including Maintenance Rule program development, operability evaluations, design of major modifications, 50.59 reviews, Generic Letter responses, and other engineering responsibilities. Hugh has a degree in Mechanical Engineering and is a registered Professional Engineer. As an employee of Duke Power subsidiary DE&S, Inc., Hugh has also been responsible for supporting other domestic nuclear power plants, and for licensing and engineering activities for a planned uranium enrichment facility based on centrifuge technology.

RICK D. LANE

Mr. Rick Lane is currently the Director - Design Engineering at Entergy Operations, Inc. in Russellville, Arkansas. In this role, he is responsible for Design Engineering support for both units at Arkansas Nuclear One. Functional

groups include Mechanical, Civil & Structural, Electrical and I&C, Engineering Programs, Nuclear Engineering Design and Engineering Support. Engineering studies/evaluations, plant modifications, programs administration/ implementation, etc. are involved in these groups on a daily and long-term basis. Prior to his becoming Director of Design Engineering, he was Manager of MCS Design Engineering at Arkansas Nuclear One in Russellville, Arkansas.

Mr. Lane has been employed with Entergy Operations, Inc. since 1972 where he progressed from Assistant Engineer to Supervisor to Management.

Mr. Lane attended the University of Arkansas receiving a B.S. in Mechanical Engineering. He is a registered Professional Engineer in Mechanical.

His nuclear experience began in 1972 where he was responsible for the design interface associated with the nuclear steam supply system on ANO 2 during the design and construction phase of the plant.

## JAMES H. LASH

Mr. James Lash is the Plant Manager for Davis-Besse.. His previous assignment was the Director, Engineering and Services Department where he directed the activities of the Plant Engineering, Nuclear Engineering, Business Services, and Regulatory Affairs Sections. With more than 22 years experience in the nuclear industry, Jim was a commissioned officer in the U.S. Navy and has experience in the design and construction of nuclear generating stations with an architect engineering firm. Jim has a degree in Physical Oceanography from the U.S. Naval Academy. Jim's assignments have included Manager, Independent Safety Engineering; Manager, Design Engineering; and Manager, Plant Operations. He has held a Senior Reactor Operator License for the Davis-Besse Nuclear Power Station. Jim has been with Toledo Edison since 1989.

## LARRY M. LESNIAK

Mr. Larry Lesniak has been employed with Framatome Technologies, Inc. (FTI) and its predecessor companies since 1969. During this time, Larry has held various engineering leadership positions covering Neutron Flux Measurement Systems, Safety/Non-Safety Instrumentation/Control/Electrical Systems, Equipment Environmental and Seismic Qualification, Licensing, and Reactor/Reactor Support Fluid Systems Design. Since 1982, Larry has been involved in engineering, project management, and business development roles for FTI engineering and field service business for the Crystal River Unit 3 plant. Notable achievements include RC pump repairs and seal improvements, the RV/RCP/OTSG Long-Term

Maintenance Program, dedicated engineering project teams to support FPC engineering work, and the development of design bases and analysis basis documents. Larry Lesniak is a professional engineer, author of several technical papers, and earned a B.S. degree in electrical engineering from the University of Pittsburgh.

## CHARLES W. PRYOR, JR.

Dr. Charlie Pryor attended Virginia Tech where he received his B.S., M.S., and Ph.D. degrees in structural engineering. He graduated with honors and was recognized for outstanding graduate research work in the area of composite materials.

Dr. Pryor began his industrial career with McDonnell Douglas Corporation in St. Louis, Missouri. As a senior engineer there, he worked on the development of a boron-epoxy wing structure for the F-15 Air Superiority fighter. Dr. Pryor joined the Babcock & Wilcox Company in January of 1972 as a senior engineer in the Applied Mechanics Unit. He progressed to the position of President and CEO of B&W Nuclear Technologies (a wholly owned subsidiary of the French nuclear company, Framatome). Dr. Pryor was awarded the prestigious award "Chevalier of the Ordre National du Merite" by the republic of France. Thereafter, in 1993, he was named Virginia's Outstanding Industrialist of the Year.

Today, Dr. Pryor has formed his own company to offer management consulting services based on his many years of successful experience in corporate growth, increasing shareholder value and working in demanding domestic and international markets.

## GORDON R. SKILLMAN

Mr. Gordon R. Skillman's experience includes thirty years of multi-disciplinary technical and management involvement in the U. S. Commercial Nuclear industry beginning with three (3) years as an Engineering officer on the N. S. Savannah, America's first and only nuclear powered merchant vessel. Major leadership roles in increasingly complex and demanding major assignments involving personnel, nuclear technology, and rigorous attention to detail relative to nuclear safety. Incorporated in these leadership roles are significant personal contributions to NSSS design, operation, maintenance, and support including sixteen (16) years of NSSS design experience with a NSSS supplier, intimate involvement in the first seven (7) years of the TMI-2 Accident Cleanup Program, and seven (7) years of day to day engineering and technical support of a highly successful commercial nuclear power plant. Formerly licensed as AEC/NRC SRO. Presently licensed as

Professional Engineer in both Pennsylvania and Virginia. Mr. Skillman is a 1966 graduate of the United States Merchant Marine Academy graduating with a BS degree in Marine Engineering.

## FRED W. TITUS

Mr. Fred W. Titus is currently the Vice President - Engineering at Entergy Operations, Inc. in Jackson, Mississippi. In this role, he is responsible for the Design Engineering function for the ANO, Grand Gulf, River Bend and Waterford 3 nuclear plants. Prior to this becoming Vice President of Engineering, he was Director of Nuclear Plant Engineering at the Grand Gulf Nuclear Station in Port Gibson, Mississippi.

Mr. Titus was employed by the Bechtel Power Corporation from 1973-1984 where he progressed from Design Engineer to Supervisor to Management. His nuclear plant experience includes Duane Arnold, Trojan, and Susquehanna.

Mr. Titus attended the University of Washington receiving a B.S. in Electrical Engineering and a M.B.A. he is a registered Professional Engineer in Mechanical and Nuclear. He also completed a Senior Reactor Operator Certification while at the Grand Gulf Nuclear Station.

His nuclear experience began in 1967 when he entered the Naval Nuclear Submarine Program where he became an Engineering Officer of the Watch on three reactor plants.

## ROBERT E. VAUGHN

Mr. Robert Vaughn's experience encompasses 28 years in project management, mechanical/nuclear engineering design, construction, licensing, startup and testing and project coordination involving major nuclear power generating facilities in the United States and internationally. Also included is experience in operation and maintenance of marine power and propulsion plants.

Mr. Vaughn has extensive and direct design experience associated with all aspects of Crystal River Unit No. 3 as the Nuclear/Mechanical Design, Startup and Modifications Project Engineer and Project Manager from 1971 to 1990 and 1994 to 1996.

ROLF WIDELL

Mr. Rolf C. Widell is currently the Director, Nuclear Operations Training for Florida Power Corporation's Crystal River Unit 3. In this capacity, he is responsible for the management of all Nuclear Training programs conducted for Florida Power Corporation's Crystal River Unit 3. Previously Mr. Widell was the Director, Nuciear Operations Site Support at Florida Power Corporation's Crystal River plant. In this capacity he was responsible for Site Nuclear Services, Radiological Emergency Planning, Nuclear Security, Nuclear Licensing and Nuclear Fuel Management and Safety Analysis.

Prior to this position, Mr. Widell acted in the capacity of Director, Nuclear Operations Engineering and Projects and in this position was responsible for all engineering activities performed in support of CR-3 including Site Nuclear Engineering Services, Nuclear Operations Engineering and Fuel Management.

Mr. Widell received a Bachelor of Science Degree in Mechanical Engineering from Lowell Technological Institute in Lowell, Massachusetts, and is a Registered Professional Engineer.

ROGER K. WYRICK

Mr. Roger Wyrick has 20 years experience in the nuclear power industry and nuclear reactor safety. His background includes fifteen years with the Institute of Nuclear Power Operations (INPO). Mr. Wyrick's positions at INPO include Senior Engineering Support Evaluation, Assistant Manager, World Association of Nuclear Operators (WANO) - Atlanta Center, Manager Plant Performance Analysis, and Manager Operational Data Analysis. Other experience consists of Supervising Engineer in Nuclear Analysis Department at Florida Power and Light Company and Senior Engineer at Combustion Engineering in Safety Analysis Design Group.

Mr. Wyrick earned his M.S. Degree in Nuclear Engineering from the University of Illinois; an M.A. Degree in Physics from the University of California, and a B. S. Degree in Physics from the Indiana University. He received his senior reactor operator certification at Palisades nuclear plant in 1994 and was an executive committee member of the American Nuclear Society's Nuclear Reactor Safety Division, 1982-1987.

## Major Activities of Team Members

- Panel Conference Call on May 8, 1996
- Panel Meeting on May 28-29, 1996
- Panel Conference Call June 18, 1996
- Panel Meeting on July 10-11, 1996
- Panel Meeting on August 19-20, 1996
- Panel Meeting on September 18, 1996
- Noman Cole Meeting with FPC staff - July 30-31, 1996
- Hugh Hammond Meeting with FPC staff August 21, 1996
- Roger Wyrick Meeting with FPC staff-July 30-August 1, 1996

Notes and minutes of the Panel meetings and conference calls are being retained in FPC files.

# Review Of Crystal River 3 Design Basis Licensee Event Reports (LERs) And Problem Reports (PRs)

The review panel performed two independent reviews of the Crystal River 3 LERs and associated station problem reports pertaining to design or design basis issues for the period from January 1992 to June 1996. The general results and conclusions of the two independent reviews were similar. Additionally, the results of the panel's reviews were also consistent with the conclusions of an LER design basis review conducted by the Crystal River 3 staff for the period from January 1994 to June 1996. Pertinent conclusions of these reviews are as follows. Numerical results from the two independent panel reviews are indicated by a value from one review followed parenthetically by the value of the other review.

1. The majority of the design basis related events at Crystal River 3 were not significant. Of 54 events reviewed for the period from 1/1/92 to 6/5/96, only two were screened significant by the INPO event screening process (LER-94-004 and LER-94-009-1).

2. Roughly 45% of the design basis events are associated with original design errors. The station therefore should expect that other undetected original design errors likely exist that will be identified in future reviews and engineering activities.

3. On the order of 40% of the events were associated with some aspect of modification activities. Approximately 20 of the events were associated with setpoints and instrumentation errors.

4. The dominant contributors to Crystal River 3 design basis events are human performance errors caused by ineffective work practices (44%) and incorrect procedures and documents (roughly 25%). The ineffective work practice issues are primarily associated with insufficient or ineffective analysis and calculational activities (roughly 50%), ineffective or inadequate review and verification of analysis and design activities (43%), and insufficient oversight of contractor work (on the order of 25 to 30%). In this regard, we note that FPC has made a major upgrade to their procedure (i.e., NEP-213, "Design Analysis/Calculation") governing the performance and the review of analysis and calculations. This procedural

upgrade specifically addresses design inputs and assumptions and significantly enhanced the process for independent review of assumptions before calculations are undertaken as well as the review of results of calculations before they are finalized. This procedural upgrade was issued June 26, 1995.

5. For a sample of 25 LERs reported by Crystal River 3 as operation outside of the design basis, the review panel did not consider 6 of these 25 LERs reportable in accordance with 10CFR50.73. Of the remaining 19 reportable LERs, the panel did not consider 7 of these as being design basis issues.

# REFERENCES

10CFR50.2

10CFR50.72

AI-300, *Plant Review Committee Charter*

AI-400C, *New Procedures and Procedure Change Requests*

AI-400F, *New Procedures And Procedure Change Processes For Emergency Operating Procedures (EOPs), Abnormal Procedures (APs), and Verification Procedures (VPs)*

AI-404A, *Review of Technical Information*

AI-404B, *Review of Industry Operating Experience*

July 1996 Comparison of B&W Plants, Framatome Technologies, Inc.

CP-111, *Initiation and Processing of the Precursor Cards and Problem Reports*

CP-150, *Identifying and Processing Operability Concerns*

EDBDs, *Enhanced Design Bases Documents*

FSAR, *Final Safety Analysis Report*

HPI Flow Upgrade

INPO final accreditation evaluation report, December, 1995

INPO-AP 905 *Configuration Change Process*

INPO-AP-906 *Design Change Process Description*

MAR 96-02-09-01

NEP-213, *Design Analyses/Calculations*

NEP-216, *Plant Design Basis Documents*

NOD-11, *Maintenance of the Current Licensing Basis*

NOD-52, *Commitment Processing and Management of Programmatic Commitments*

NUMARC 90-12, *Design Basis Program Guidelines*

NUREG 1397, *An Assessment of Design Control practices and Design Reconstitution Programs in the Nuclear Power Industry Dated February 1991*

NUREG 1022, *Event Reporting Guidelines 10CFR50.72 and 10CFR50.73*

Paul Tanguay to Pat Beard interoffice correspondence dated June 27, 1996.

Paul Tanguay Report

Phase 3 curriculum guides and lesson plans for 1993-1996.

TDP-308 *Training Needs Assessment Study*, December, 1993;

Tech Staff and Manager Training Program position specific qualification

# CRYSTAL RIVER 3 PERSONNEL INTERVIEWED

**July 10-11, 1996**

**July 30 to August 1, 1996**

| | |
|---|---|
| Andy Auner | Manager Nuclear Technical Training |
| Terry Austin | Supervisor Mechanical System Maintenance Crew |
| Ken Baker | Manager, Nuclear Configuration Management |
| W. K. Bandhauer | Manager, Nuclear Shift |
| Sterling Barnette | Nuclear Engineering Assistant |
| Pat Beard | Senior Vice President, Nuclear Operations |
| Gary Becker | Manager In Training |
| Gary Boldt | Vice President, Nuclear Production |
| Jeff Endsley | Supervisor, Nuclear Engineering-Electrical |
| Ed Gschwender | Senior Configuration Management Specialist |
| Brian Gutherman | Manager, Nuclear Licensing |
| Dan Jopling | Senior Nuclear Structural Engineer |
| Bob Knoll | Senior Nuclear Mechanical Engineer |
| Steve Koleff | Supervisor Nuclear Engineering |
| Thomas Lehman | Nuclear Projects Engineer |
| Mikell Lord | Senior Nuclear I&C Engineer |
| Robert Marckese | Senior Nuclear Electrical Engineer |
| Joe Maseda | Manager, Nuclear Engineering Programs<br>(formerly Manager, Nuclear Engineering Design) |
| Craig Miller | Supervisor, Nuclear Engineering-Mechanical |
| Dennis O'Shea | Manager, Nuclear Fuel Management and Safety Analysis |

| | |
|---|---|
| Tony Petrowsky | Supervisor, Nuclear Engineering-Civil/Structures |
| Roger Schmiedel | Senior Nuclear Electrical Engineer |
| Fran Sullivan | Manager, Nuclear Engineering Design |
| Paul Tanguay | Assistant to Vice President, Nuclear Production (formerly Director, Nuclear Engineering and Projects) |
| Bruce Taylor | Supervisor, Nuclear Configuration Management |
| John Taylor | Nuclear Project Engineer |
| Shawn Tyler | Nuclear Projects Specialist |
| Rollin VanAlstine | Senior Nuclear I&C Engineer |
| Art Washburn | Supervisor, Nuclear Plant Technical Support-Mechanical |
| Ken Wilson | Principal Nuclear Operations Engineer |
| Mike Winship | Nuclear Operations Specialist |
| Henry Wojtasinski | Nuclear Maintenance Specialist |