ATTN: J. English
July 18, 1980

SUMMARY OF CONCERNS

1. The Preoperational Test Program was conducted inadequately. This is evidenced by a large number of equipment malfunctions discovered during the Power Ascension Test Program. Large number of outstanding items added as condition of License also are indicative of unresolved Preoperational Test items.

2. Management controls over Power Ascension Test Program has been inadequate. There is evidence of "Jury-rigging" of systems to conduct tests. There is evidence that testing has been delayed to allow electrical generation. This action results in operation at significant power levels with untested systems even in light of events which show evidence of inadequate system performance (e.g. November 29, 1977 event).

3. There is evidence that the reactor design provides significantly less protection than other PWR reactor designs.

4. There have been numerous significant operator errors. The inspector notes that these errors are not being reduced in frequency.

5. There is evidence that significant design defects exists in the electrical distribution system.

Submitted to NRC chairman J.F. Ahearne by J.S. Creswell. Date unknown.

6. There are serious questions about conformance with several General Design Criteria.

7. NRC has conducted management meetings with the licensee identifying many repeat concerns. These concerns are not being adequately addressed.

8. A significant number of safety related FCR's (Facility Change Requests) remain outstanding. This item coupled with operator errors mentioned in Item 4 increases both the risks and consequences, of accidents at the facility.

9. There is evidence that when defects are identified in safety related systems the defects are not analyzed and corrected in a timely fashion.

EXAMPLES IN SUPPORT OF CONCERNS

1. References:

   a. Paragraph 8, Report 50-346/77-06

   b. Appendix A, to Report 50-346/78-19, Page 3(a)

   c. Paragraph 4, Report 50-346/78-17, Item H, March 7, 1979 transmittal

   d. Memo, Keppler to Mosely dated August 14, 1979

Commentary:

Reference a refers to how the SFAS testing was performed.
Reference b refers to the inadequacy of the SFAS testing.
Reference c refers to the conduct of electrical testing during preoperational testing and power ascension testing.
Reference d refers to allegation regarding SFAS testing by plant personnel.

2. References:

   a. IE Report 50-346/78-06, Appendix A

   b. IE Report 50-346/78-17, Appendix A and Paragraph 13

   c. IE Report 50-346/78-30, Paragraph 9

   d. IE Report 50-346/79-04, Appendix A (Item Q, this transmittal)

Commentary:

Reference a refers to management control over rod drop testing. The test deficiency is still unresolved.

Reference b refers to management control over determination of worst case core peaking factors and corporate management overview of the Startup Test Program.

Reference c refers to use of supplemental air supply during testing.

Reference d is to management controls over testing. Some items are repeat of Noncompliance issued in 50-346/78-06.

Some critical testing such as Natural Circulation Test, Loss of Offsite Power Test, Shutdown Outside Control Room Test and Load Rejection Test have only recently been completed after approximately one and one-half years after initial criticality.

3. References:

a. IE Report 50-346/78-30, Paragraph 13

b. Rancho Seco event, March 20, 1978

c. Licensee report on November 29, 1977 event (Item G, March 7, 1979 transmittal)

d. Additional Safety Evaluation of Transient Resulting from Inability of Operator to Control Steam Generator Level at 35 Inches (Docketed - Serial No. 475, December 22, 1978)

Commentary:

Reference a refers to no direct automatic tr... associated with loss of heat sink such as the Westinghouse low steam generator level reactor trip.

Reference b refers to the loss of power to non nuclear instrumentation which can result in severe thermal transients and extreme difficulty in controlling the plant.

Reference c and d refer to loss of pressurizer level indication and possible voiding of the pressurizer during anticipated operational transients.

There is a concern that large positive moderator temperature coefficients produce difficultly in controlling the plant. This item will be addressed further in Item 8.

There is a concern regarding the reduncancy and diversity of the auxiliary feed water system. Usually there are combinations of steam and electric driven auxiliary feed pumps. The Davis-Besse facility has two steam driven auxiliary feed pumps. There are indications that during some of the events steam pressure dropped to levels which affected the pumps operability.

4. References:

a. Vendor Report on September 24, 1977 event (Item [illegible], March 7, 1979 Transmittal)

b. Vendor Report on April 29, 1978 transient (Item P, this transmittal)

c. Reportable Occurrence 78-066 (in PDR)

d. Reportable Occurrence 78-067

Commentary:

Reference a refers to operator shutting off the Emergency Core Cooling System during the LOCA.

Reference b and c refers to poor operator performance particularly in view of withdrawing control rods when they should have been inserted.

Reference d is to operator repeating errors.

Additional examples can be furnished on request.

5. References:

a. Memo, Streeter to R. W. Woodruff dated June 9, 1978 (AITS F30385H2)

b. Daily Staff Notes, October 30, 1978.

c. Licensee Report on November 29, 1977 event (Item G of March, 1979 transmittal).

d. IE Report 50-346/78-06, Paragraph 3.

e. IE Report 50-346/78-17, Paragraphs 4, 5, and 12.

f. IE Report 50-346/78-30, Paragraphs 3, 9, and 10.

Commentary:

Commentary on these items will be delayed until results of recent events regarding loss of offsite power are reviewed further.

6. (a) Comments on GDC 11

Criterion 11 - Reactor inherent protection. The reactor core and associated coolant systems shall be designed so that in the power operating range the net effect of the prompt inherent nuclear feedback characteristics tends to compensate for a rapid increase in reactivity.

Commentary:

It does not appear to the inspector that the Davis Besse core design conforms to this minimum design requirement. A moderator temperature coefficient of + 0.7×10^{-4} % $\Delta K/k^{o}F$ was measured during the Cycle 1B startup. In addition, the inspector has a concern that the reactor was operated above 95% power with a positive temperature coefficient (see 50-346/79-04). Some effects noted from operation with the positive coefficient are noted below.

8/2/78 In preparation for 40% reactor physics testing, the six second rod insertion step for differential rod worth measurement was attempted. The rod movement resulted in a Reactor Coolant System upset. The positive temperature coefficient caused feedwater control of Tave to be unstable. A divergent oscillation in feedwater lead to overfeeding of the steam generators, and resulted in an RPS low pressure trip.

Reference: Supplement 3 of the Davis Besse Unit 1 Initial Startup Report Dated February 8, 1979.

8/31/78 Tech. Spec. 3.1.1.4 regarding minimum temperature for criticality exceeded due to positive moderator temperature coefficient.

Reference: R.O. 78-088

TP 800.08

The inspector notes that the Zion (Westinghouse PWR) Technical Specifications require:

3.2.1.C. Unit Startup

1. Immediately prior to startup, the reactor coolant temperature shall be shown to be greater than the temperature above which the moderator temperature coefficient is always negative and greater than 500°F, except during low power physics tests.

and the bases state:

During the early part of a fuel cycle, the moderator temperature coefficient may be slightly positive at coolant temperatures below the power operating range. (1)(2) The moderator coefficient at low temperatures will be most positive at the beginning of life of the fuel cycle, when the boron concentrations in the coolant is the greatest. Later in the cycle, the boron concentrations in the coolant will be lower and the moderator coefficient will be either less positive or will be negative. At all times, the moderator coefficient is negative in the power operating range. (1)(2) The maximum temperature at which the moderator coefficient is positive at the beginning of life of any fuel cycle with all control rods withdrawn, is determined during the lower power physics tests for that cycle.

(b) Comments on GDC 12

Criterion 12 - Suppression of reactor power oscillations. The reactor core and associated coolant, control, control, and protection system shall be designed to assure that power oscillations which can result in conditions exceeding specified acceptable fuel design limits are not possible or can be reliably and readily detected and suppressed.

Reference: I.E. Report 50-346/78-06 Paragraph 4

The reference refers to power oscillations observed at Davis Besse -1. The Oconee facility has experienced power oscillations of up to 7% power peak to peak. The inspector can find no definitive statement regarding the safety implications of these oscillations.

(c) Criterion 13 - Instrumentation and Control. Instrumentation shall be provided to monitor variables and systems over their anticipated ranges for normal operation, for anticipated operational occurrences, and for the accident conditions as appropriate to assure adequate safety, including those variables and systems that can affect the fission process, the integrity of the reactor core, the reactor coolant pressure boundary, and the containment and its associated systems. Appropriate controls shall be provided to maintain these variables and systems within prescribed operating ranges.

Reference: I.E. Report 50-346/78-06, Paragraph 2

During the November 29, 1977 event there was a loss of pressurizer level indication for five minutes. In addition there is concern about monitoring makeup flow and T cold during a thermal transient of this type.

Further information on GDC 17 and GDC 33 will be furnished at a later date when more information becomes available.

7. References:

   a. Licensee minutes of October 1977, Management Meeting (Item E, March 7, 1979 transmittal)

   b. NRC Report on August 1978, Management Meeting (Item J, March 7, 1979 transmittal)

   c. Licensee Minutes of August 1978 Management Meeting (Item K, March 7, 1979 transmittal)

Commentary: The inspector feels that a review of the references address the concern but if further information is needed the inspector will gladly discuss the matter.

8. Reference: I.E. Report 50-346/79-05 Paragraph 6.d

Commentary:

Supplementary information obtained from another inspector regarding the status of the FCR's (Facility Change Requests) indicates the following:

a. 516 FCR's are not ready for implementation

b. 145 are in the implementation stage

c. 245 are in the followup stage

d. 162 are closed

Additional information is forthcoming as a result of my request for an investigation.

9. References:

a. LER 77-11

b. LER 77-53

c. LER 77-61

d. LER 77-17

e. LER 77-80

f. LER 77-83

g. LER 77-110

h. LER 77-113

i. LER 77-116

j. LER 78-05

Commentary:

The auxiliary feed pumps have had extensive difficulties in speed control. In July and August, 1977, repeated speed control relay failures rendered the auxiliary feed pumps inoperable. On August 10, 1977, a design modification was implemented which added a second set of identical speed relays in parallel to reduce the current carried by each relay. This did not totally eliminate the speed control failures and in January, 1978, the relays of the speed circuit were replaced with relays of a larger current carrying capacity. (Exerpt from startup test report)