Docket No. 52-001

May 10, 1992

APPLICANT: GE Nuclear Energy (GE)

PROJECT: Advanced Boiling Water Reactor (ABWR)

SUBJECT: AUDIT SUMMARY - ADVANCED BOILING WATER REACTOR (ABWR) INSPECTIONS, TESTS, ANALYSES, AND ACCEPTANCE CRITERIA (ITAAC) ON REACTOR VESSEL INTERNALS

As a followup action from January 28 and 29, 1992, meeting with GE, the staff of the Structural and Geosciences Branch conducted an audit at GE's San Jose, California, office on February 10-12, 1992. The purpose was to review documentation and the bases for the establishment of ITAAC for the ABWR reactor pressure vessel (RPV) internals. The RPV internals system is one of the nine ABWR Pilot ITAACs submitted by GE to the Nuclear Regulatory Commission (NRC) staff on January 17, 1992.

The audit reviewed the need to have a vibration prediction analysis to assure the ability of the RPV internals to withstand flow-induced vibrations and the need to specify key dimensions and major design parameters of the internals to assure that future plants which reference the ABWR standard design will be built and will operate in accordance with the certified design.

As a result of this audit, the staff found that vibration prediction analyses were done for a typical ABWR in Japan, and GE agreed to revise the specific ITAAC, as suggested by the staff, which will include information on key dimensions and major design parameters for the RPV internals. GE indicated that the revision will be contained in their official response to the NRC staff audit and will be submitted to the NRC staff by the end of February 1992. A summary of the audit is enclosed.

Original Signed By:
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Enclosure:
As stated

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AUDIT SUMMARY - ABWR RPV AND INTERNALS

OBJECTIVES:

On February 10-12, 1992, the NRC staff of the Advanced Reactors Engineering Section of the Structural and Geosciences Branch conducted an audit at GE's office in San Jose, California. The audit was a followup action from the NRC/GE management meeting held on January 28-29, 1992. The purpose was to review the documentation and the bases for the establishment of inspections, tests, analyses, and their acceptance criteria (ITAAC) for the ABWR reactor pressure vessel (RPV) and internals. The meeting agenda (Enclosure 1) and list of meeting participants (Enclosure 2) are attached.

BACKGROUND:

The staff review of the specific pilot ITAAC for RPV and internals found that it did not address key dimensions and major design parameters of RPV and internals. This information is needed to be verified by ITAAC to assure that future ABWRs which reference the design will be built and will operate in accordance with the certified design. In addition, vibration prediction analysis of RPV internals is needed prior to final design approval for the staff to reach a safety determination of the ability of the ABWR RPV internals to withstand flow-induced vibrations.

FINDINGS AND CONCLUSIONS:

During the three-day audit, the staff discussed the above concerns with GE personnel and reviewed relevant documents for resolving our concerns. The staff reached a tentative agreement with GE based on a proposed revision of the specific ITAAC. Pending verification by an official GE response to staff audit on the pilot ITAAC, which as indicated by GE, will be submitted to NRC by the end of February, 1992, the staff considers the RPV and internals ITAAC to be acceptable.

The following is a summary of audit findings and conclusions:

1. GE agreed that the pilot ITAAC related to the RPV system, as shown in Table 2.1.1 of the GE submittal dated January 17, 1992 will be revised as marked in Enclosure 3 to address the staff concerns.

2. As suggested by the staff, GE agreed to add new tables and figures (see Table 2.1.1-1, 2.1.1-2, 2.1.1-3 and Figure 2.1.1-2a, 2.1.1-2b in Enclosure 3) to the ITAAC, and revise the existing Figure 2.1.1-1. Thus, our concern on key dimensions and major design parameters of the RPV and internals is

resolved. However, construction tolerences of the listed key dimensions as shown in the column of "Variation" in Table 2.1.1-3 remain to be defined. GE indicated that values of variation will be proposed to the staff within two weeks.

3. GE indicated that the first and also the typical ABWR, Kashiwazaki Unit 6 (K-6), is currently under construction in Japan. A set of documents (References 1 to 7) related to the flow-induced vibration (FIV) assessment program for the K-6 reactor internals was presented. The information consists of an analysis for vibration prediction, the basis and details of instrumentation for vibration monitoring, specification for conducting the preoperational and startup tests, specifications for the installation and removal of the monitoring system, and a full-scale, 60-degree flow test of the ABWR reactor internal pump system conducted in Japan.

   For vibration prediction analysis, a statistical approach was taken to estimate the range of responses of major RPV internal components in their first few fundamental modes, based on correlation of measured responses of a selected group of existing BWRs with similar configurations. Parameters used to estimate sample responses consist of flow, power, stiffness, etc. Both U. S. (i.e. RG 1.20 and ASME Boiler and Pressure Vessel Code' d Japan regulatory requirements were included in the acceptance criteria.

   The staff found that the information is comprehensive and generally in conformance with guidelines and requirements of the regulatory positions stated in SRP 3.9 and RG 1.20. It provides reasonable assurance that the RPV internals of the certified ABWR will withstand flow-induced vibrations. The staff also agreed with GE that the analytically predicted values may be upgraded when future test data become available, such as data from preoperational and startup tests of the K-6 facility.

4. The staff verified the availability and completeness of documents (References 8 to 9) related to the design specification for the K-6 RPV system (specifically the core support structures). The information consists of system and component design requirements, including applicable codes and standards, loads and load combinations, design limits, vibration assessment program, material and processing requirements.

5. The staff reviewed documents (References 10 to 12) related to the evaluation of the K-6 reactor internals under postulated accidents, such as the event of main steam line break at the RPV nozzle. The information consists of calculated pressure differentials and assessment of effects to the reactor internals. Mathematical models with details were presented in Reference 11 for dynamic analysis, which were not performed due to non-dynamic nature (slow variation) of pressure

differentials during the event. The staff observed the calculated pressure differential curves and concurred that, with large separation of component structural frequencies and the excitation frequencies, amplification of loads are unlikely.

6. The staff verified the availability of documents (References 13 and 14) related to purchase specifications for the K-6 facility. The information in Reference 13 defines the general administrative requirements on procurement, including material control, fabrication control, documentation requirements, and various QA/QC programs. Specific procurement requirements for the reactor internals are addressed in Reference 14.

7. GE agreed to provide the staff an updated MPL-18NS07A03 list, the master list of documents applicable to ABWR Standard Plant, to include References 1-12. References 13 and 14 related to procurement specifications will not be included due to potential variation in requirements among utilities.

REFERENCES:

The following consists a list of documents presented by GE to the staff during the audit:

1. GE Report 23A6701, Revision 0, "Reactor Internals Vibration Prediction," a design report for Japanese K-6 plant, November 29, 1989.

2. GE Report 23A6253, Revision 0, "Reactor Internals Vibration Instrumentation," a design specification for Japanese K-6 plant, July 14, 1989.

3. GE Report 23A6715, Revision 0, "Reactor Internals Vibration Instrumentation," a stress analysis report for Japanese K-6 plant, May 2, 1989.

4. GE Report 23A6315, Revision 0, "Reactor Internals Vibration Monitoring System," a test specification for Japanese K-6 plant, June 21, 1989.

5. GE Report 23A6255, Revision 0, "Reactor Internals Vibration Monitoring System Removal," a specification for Japanese K-6 plant, June 16, 1989.

6. GE Report 23A6254, Revision 0, "Reactor Internals Vibration Monitoring Equipment," a installation specification for Japanese K-6 plant, Jujy 14, 1989.

7. Katsuta Engineering Laboratory Report, "Verification Test of Reactor Internal Pump System for BWR."

8. GE Report 23A6012, Revision 3, "Reactor Pressure Vessel System," a design specification for Japanese plant K-6, April

23, 1991.

9. GE Report 23A1437, Revision 2, "Core Support Structures," a design specification for Japanese plant K-6, May 22,1991.

10. GE Report 386HA984, Revision 3, "Reactor Internal Pressure Differences," a design requirement document for Japanese plant K-6, June 10, 1988.

11. GE Report 23A1482, Revision 3, "Reactor Dynamic Model," a design specification for Japanese K-6 plant, Jujy 11, 1990.

12. GE internal letter, N. T. Patel to C. W. Dillmann, "Ampliphication of loads due to Dynamic Pressure Differentials on Reactor Internals during LOCA," August 26, 1991.

13. GE Report 23A6124, Revision 4, "Purchase Specification," a purchase specification for Japanese plant K-6, June 19, 1991.

14. GE Report 23A6120BA, Revision 1, "Reactor Internals," a purchase specification for Japanese plant K-6, June 19, 1991.

ENCLOSURE 1

NRC Audit of GE on

ABWR ITAAC OF REACTOR VESSEL AND INTERNALS

February 10-12,1992

AGENDA

I. Discuss revision of RPV System ITAAC as shown on Table 2.1.1 of the GE submital dated January 17, 1992.

II. Discuss Tier 1 information regarding dimensions of key structural details and parameters controling design of the RPV system.

III. Review vibration prediction analysis of reactor internals to assure design adequacy against flow-induced vibrations during plant normal operation.

IV. Review faulted condition analysis to ensure operability and structural integraty of the RPV systems, specifically the consequences under a postulated steam line break.

V. Review design and procurement specifications

ENCLOSURE 2

NRC Audit of GE on

ABWR ITAAC OF REACTOR VESSEL AND INTERNALS

February 10-12,1992

ATTENDANCE LIST

Rebecca L. Nease	NRC	Project Manager, PDST
Shou-nien Hou	NRC	Sr. Mechanical Engineer, ESGB
Jack N. Fox	GE	ABWR Licensing
Tony James	GE	ABWR ITAAC Program Manager
Nilhanth Patel	GE	ABWR Senior Engineer
Ian See	GE	Engineer
T. J. Judge	GE	ABWR QA on Procurement

Enclosure 3

Key dimensions and design details of RPVS components are presented in Table 2.1.1-2 and Figure 2.1.1-2, respectively. The major plant design parameters are listed in Table 2.1.1-1

2.1.1

Table 2.1.1: REACTOR PRESSURE VESSEL SYSTEM

Tables 2.1.1-1, 2.1.1-2 and 2.1.1-3, and

Inspections, Tests, Analyses and Acceptance Criteria

Certified Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. System configuration of the reactor pressure vessel system (RPVS) as described in Section 2.1.1 is shown on Figure 2.1.1-1.	1. Visual field inspections will be conducted of the installed RPVS key components identified in Section 2.1.1 and Figure 2.1.1.	1. The installed configuration of the RPVS will be considered acceptable if it complies with Figure 2.1.1 -1 and 2.1.1-2, ~~and~~ Section 2.1.1
2. The reactor coolant pressure boundary (RCPB) portion of the RPV and appurtenances and their supports are classified as Quality Group A, Seismic Category I. These components are designed, fabricated, examined and hydrotested in accordance with the rules of ASME Code Class 1 vessel or component support, and are code stamped accordingly. The core support structures are Quality Group C, Seismic Category I, and are designed, fabricated and examined in accordance with the rules of ASME Code Class CS structures, and are code-stamped accordingly.	2. Inspections will be conducted of ASME Code required documents and the Code stamp on the components.	2. Existence of necessary ASME Code required documents and the code stamps on the components confirm that the components in the RCPB of the RPV and the supports, and the core support structures are designed, fabricated and examined as ASME Code Class 1 and CS respectively. This also confirms that the RPV is hydrotested per the ASME Code Class 1 requirements.
3. The RCPB of the RPVS retains its integrity under internal pressure that will be experienced during the service.	3. A hydrostatic test of the RCPB will be conducted in accordance with the ASME Code requirements.	3. The results of the hydrostatic test must conform with the requirements in the ASME Code.
4. The materials used for RCPB portion of the RPV and appurtenances are certain proven low and high alloy steels with certain additional requirements for construction, as identified in Section 2.1.1. Special controls are exercised when austenitic stainless steel is used for construction of RPV internals in order to avoid cracking during service.	4. Inspection will be conducted of the records of materials, fabrication, and examination used in construction of the RCPB and austenitic stainless steel reactor internals.	4. Records of the materials and processes must confirm that the requirements specified for the RCPB in Section 2.1.1 are satisfied and that the manufacture and fabrication of the RPV internals made of austenitic stainless steel avoid potential for cracking in service.

(Note: Essentials already in SAR 5.3.1.4 & 5.2.3.4.1 and also duplicated in SAR 4.5.2.4)

Validation Attributes:

The following special controls are exercised when austenitic stainless steel is used in manufacture and fabrication of

Table 2.1.1: REACTOR PRESSURE VESSEL SYSTEM (Continued)

Inspections, Tests, Analyses and Acceptance Criteria

Certified Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
		4. (Continued) RPVS internals. Where stainless steel surfaces are exposed to water at temperatures above 93°C, low carbon (0.020% maximum) or nuclear grade materials (maximum C=0.020% with nitrogen added) or CF3 type castings are used. All materials are supplied in the solution heat treated condition. Sensitization tests are applied to assure that the material is in the annealed condition. During fabrication, any heating operation (except welding) between 427-982°C is avoided, unless followed by solution heat treatment. During welding, heat input and interpass temperature are controlled. Weld filler material used is Type 308L/316L/309L or equivalent. All weld filler materials used have a minimum of 8 FN average (ferrite number) determined on undiluted weld pads by magnetic measuring instruments. During fabrication, cold work is controlled by applying limits in hardness, bend radii and surface finish on ground surfaces. Process controls are exercised during all stages of component manufacturing, fabrication and installation to minimize contaminants. Surface contaminants are removed prior to any heating operations.
5. The ferritic materials used in RCPB portion of the RPV and appurtenances are not susceptable to brittle fracture under pressure during the service.	5. Fracture toughness tests of the ferritic base, weld and heat affected zone (HAZ) metal used in the RCPB will be conducted in accordance with the requirements for ASME Class 1 components.	5. Records of the fracture toughness data of the RCPB ferritic materials must confirm that 1) the requirements of the ASME Code are met, and 2) the reactor vessel beltline materials will not be susceptible to brittle fracture during the service.

Table 2.1.1: REACTOR PRESSURE VESSEL SYSTEM (Continued)

Insp_ctions, Tests, Analyses and Acceptance Criteria

Certified Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
	(Note: Contents are from 10CFR Part 50, Appendix G Sec. IV.1 and IV.5)	5. (Continued) ~~Validation~~ Attributes: ~~a. The minimum upper shelf energy level for base and weld metal in reactor vessel beltline must be 10.4 kg-m.~~ (≥75 ft-#) ~~b. The predicted minimum upper shelf energy level for base and weld metal in reactor vessel beltline at end of life must be 6.9 kg-m.~~ (50 ft-#) ~~c. The predicted value of adjusted reference temperature, RT_{NDT}, of base and weld metal in reactor vessel beltline at end of life must be 93 C or less.~~
6. Specimens for the surveillance program are selected from the vessel base metal and weld metal.	6. Inspection will be conducted of the records of the specimens selected from the reactor beltline region.	6. Records of the specimens with respect to location and orientation, types (tensile or Charpy V notch), and quantities must meet the requirements of ASTM E 185.
Analysis for vibration prediction is performed to assure that 7. Design and construction of the RPV internals ~~assure that the internals~~ can withstand the effects of flow induced vibration (FIV). This design analysis is based on predicted values of FIV loads which may be upgraded by available test data.	7. A vibration test will be conducted of the reactor internals to verify the adequacy of the internals design, manufacture, and assembly with respect to the potential effects of FIV. The first of a kind prototype internals will be flow tested by vibration instrumentation followed by inspection for damage. The internals in subsequent plants will be flow tested, but without vibration instrumentation, followed by inspection for damage.	7. Reactor vessel internals vibration is considered acceptable when results of the ~~vibration analysis,~~ vibration measurement ~~testing and~~ inspection of the internals indicate no sign of damage, loose parts, or excessive wear in the prototype test. The vibration of reactor internals in subsequent plants is considered acceptable when inspection of the internals indicate no sign of damage, loose parts, or excessive wear is compared with results of the vibration prediction analysis to verify compliance with design limits, and when

Table 2.1.1: REACTOR PRESSURE VESSEL SYSTEM (Continued)

Inspections, Tests, Analyses and Acceptance Criteria

Certified Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
8. Access for examinations of the RPV is incorporated into the design of the vessel, biological shield wall and vessel insulation.	8. Visual inspection will be conducted of accessibility for examinations of the vessel and welds.	8. Provisions for access in the design of the vessel, biological shield wall, and vessel insulation shall be, in the minimum, as follows: The shield wall and vessel insulation behind the shield wall must be spaced away from the RPV outside surface. Access for the insertion of automated devices must be provided through removable insulation panels at the top of the shield wall and at access ports at reactor vessel nozzles. Access to the reactor pressure vessel welds above the top of the biological shield wall must be provided by removable insulation panels. The closure head must have removable insulation to provide access for manual ultrasonic examinations of its welds. Access to the bottom head to shell weld must be provided through openings in the RPV support pedestal and removable insulation panels around the cylindrical lower portion of the vessel. Access must be provided to partial penetration nozzle welds, i.e., CRD penetrations, instrumentation nozzles and recirculation internal pump penetration welds, for performance of the visual examinations. Access must be provided for examination of the attachment weld between the support skirt knuckle (forged integrally on the shell ring) and the RPV support skirt. Access must be provided to the balance of the support skirt for performance of visual examination.

Table 2.1.1-1
KEY DIMENSIONS OF RPVS COMPONENTS

Description	Elev./Dim. (Figure 2.1.1-1)	Value (mm)
RPV ID	G	7061
RPV wall thickness in beltline	H	190.5
RPV bottom head inside invert	A	0.00
Top of RPV flange	F	17703.0
RPV skirt support bottom	B	3200.0
RPV Stabilizer connection	E	13766.0
Shroud OD	L	5550.0
Shroud wall thickness	M	50.8
Steam nozzle OD	K	700.0
Steam nozzle flow element throat dia.	J	353.8
Core plate support	C	4695.2
Top guide support	D	9351.2
Shroud support legs (Fig. 2.1.1-2a)	N	151.0
Control rod guide tube OD	P	273.05

Table 2.1.1-2

MAJOR PLANT DESIGN PARAMETERS

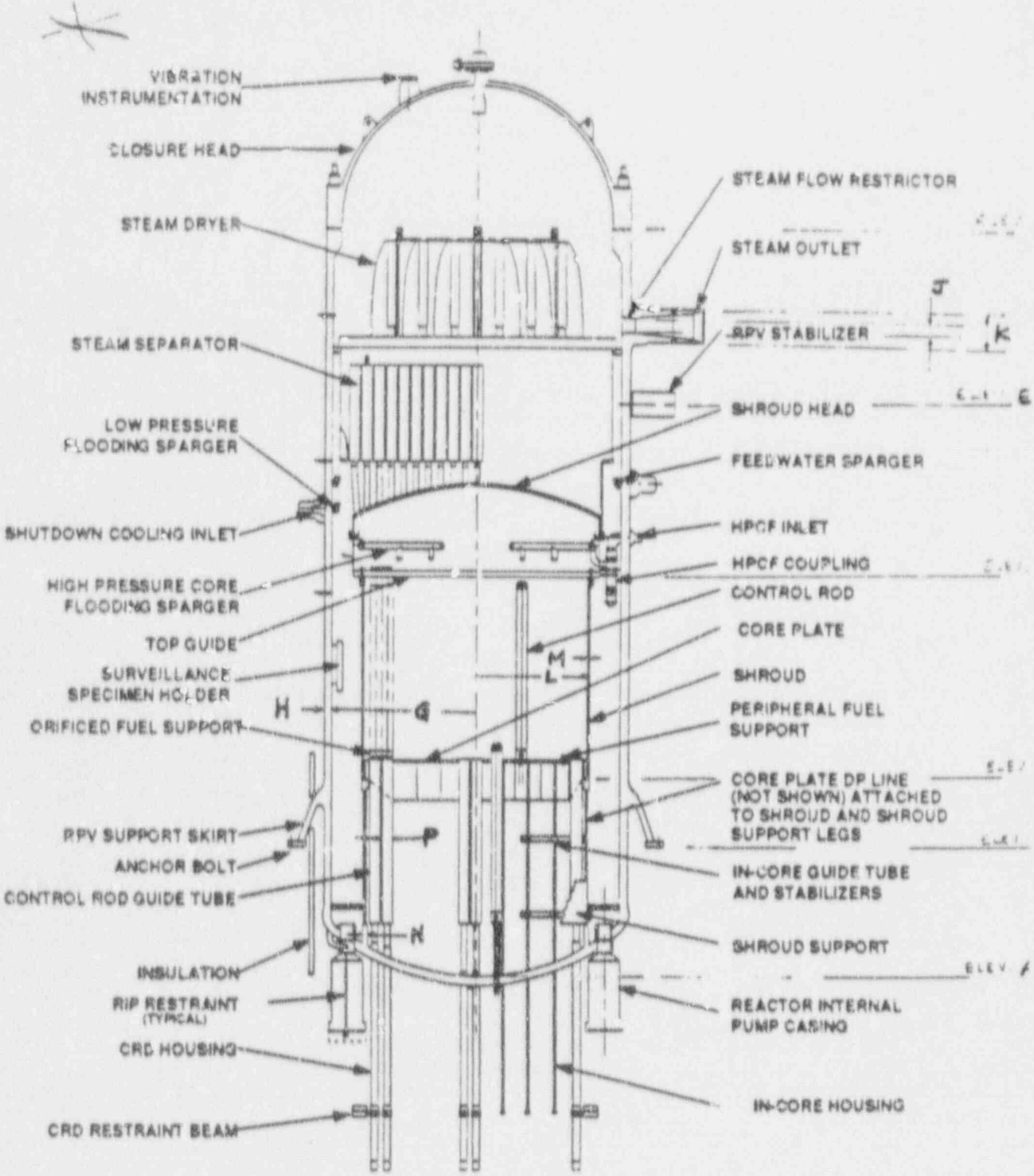
Description	Value
Design Rated Power (3926 MWt)	See Section 2.0
Core Coolant Flow Rate (115.1 Mlb/hr))	Same
Steam flow rate at full power (16.843 Mlb/hr)	Same
Pump flow rate at rated core flow (30430 gpm)	Same
Number of fuel asemblies (872)	Same
Number of control rods	204
Number of internal pumps	10
RPV design pressure	87.9 kg/cm^2g
RPV design temperature	302 C

Table 2.1.1-3

Accetable Variations of Dimensions and Elevations

Description	Elev./Dim. (Figure 2.1.1-1)	Variation (mm)
RPV ID	G	
RPV wall thickness in beltline	H	
RPV bottom head inside invert	A	
Top of RPV flange	F	
RPV skirt support bottom	B	Remain to be defined
RPV Stabilizer connection	E	
Shroud OD	L	
Shroud wall thickness	M	
Steam nozzle OD	K	
Steam nozzle flow element throat dia.	J	
Core plate support	C	
Top guide support	D	
Shroud support legs (Fig. 2.1.1-2a)	N	
Control rod guide tube OD	P	

Figure 2.1.1-1 REACTOR PRESSURE VESSEL SYSTEM KEY FEATURES



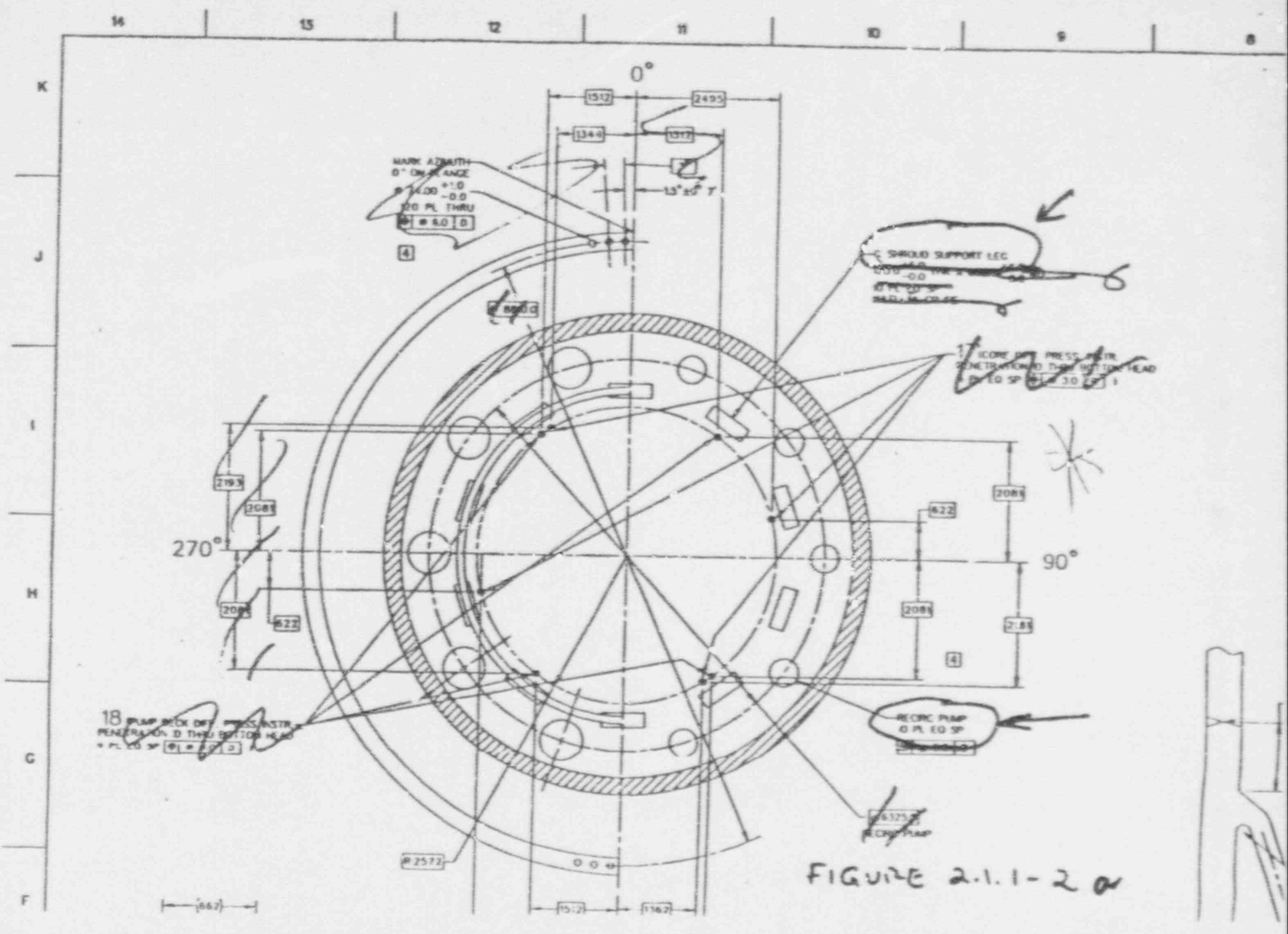


FIGURE 2.1.1-2a

NOTE: LOADING PATTERN IS SHOWN FOR QUARTER CORE ONLY. ROTATIONAL SYMMETRY APPLIES.

	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17
1															4	4	4
2												4	4	4	2	2	1
3											4	2	2	1	1	2	2
4								4	4	4	1	2	1	1	2	1	2
5							4	2	2	1	2	1	2	3	1	2	1
6						4	2	1	1	2	1	2	2	1	2	3	2
7					4	2	2	1	2	1	2	3	1	2	3	1	3
8				4	2	1	1	2	1	1	2	1	2	3	1	2	1
9				4	2	1	2	1	3	3	1	2	3	1	2	1	3
10				4	1	2	1	1	3	3	1	3	1	2	2	1	3
11			4	1	2	1	2	2	1	1	2	1	2	3	1	2	1
12		4	2	2	1	2	3	1	2	3	1	2	1	1	2	3	2
13		4	2	1	2	2	1	2	3	1	2	1	3	3	2	1	3
14		4	1	1	3	1	2	3	1	2	3	1	3	3	1	2	1
15	4	2	1	2	1	2	3	1	2	2	1	2	2	1	2	3	2
16	4	1	2	1	2	3	1	2	1	1	2	3	1	2	3	1	2
17	4	2	1	2	3	1	2	2	3	3	2	1	2	3	1	2	3

J → 1 … 17

~~BUNDLE TYPE~~	~~ENRICHMENT~~	~~NUMBER OF BUNDLES~~
~~1~~	~~HIGH ENRICHMENT, 3.18 wt.%~~	~~308~~
~~2~~	~~MEDIUM ENRICHMENT, 2.19 wt.%~~	~~334~~
~~3~~	~~LOW ENRICHMENT, 1.23 wt.%~~	~~148~~
4	~~NATURAL URANIUM, 0.71 wt.%~~	~~82~~

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Figure ~~4.3-1~~ 2.1.1-2a CORE ~~LOADING MAP USED FOR RESPONSE ANALYSES~~ ARRANGEMENT