U.S. ATOMIC ENERGY COMMISSION
REGION V
DIVISION OF COMPLIANCE

December 7, 1965

CO REPORT NO. 133/65-7

Title: PACIFIC GAS & ELECTRIC CO. (HUMBOLDT BAY)
LICENSE NO. DPR-7
Date of Visit: November 9, 10 and 11, 1965

By: A. D. Johnson, Reactor Inspector

SUMMARY

The Humboldt Bay Reactor facility was visited for the purpose of reviewing the licensee's activities during the current refueling outage. No items of noncompliance or of immediate safety significance were noted during the visit.

The core reloading operations were completed on October 11, 1965. The reload included 40 new Type-II Zircaloy clad fuel assemblies loaded in a one out of four scatter pattern. The 16 center control rods were replaced with new rods of lower reactivity value. The core reloading was done in accordance with a preplanned step-by-step procedure.

The modifications to the containment and reactor systems required by Change No. 17 to the technical specifications were scheduled to be completed during the week of November 15, 1965. Following completion of these modifications, a strength and integrated leak rate test of the containment system will be done.

Thirty-two of the 152 fuel assemblies tested with the dry sipper were determined to have clad failures. Examination of some of the assemblies showed that gross clad failures had occurred. The failures were attributed to stress corrosion rather than burnout. Significant amounts of scale were also noted on the fuel cladding. Flow tests were performed on several assemblies to determine the effect of the scale buildup on the flow characteristics of the fuel assemblies. The data have been forwarded to the General Electric Co. for evaluation. It was estimated that the largest effect noted was an approximate 15% reduction of flow through an assembly.

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Several facility changes were made during the outage. These modifications were undertaken in accordance with the provisions of 10 CFR 50.59.

An analysis of the atmosphere over the open reactor vessel indicated that the tritium concentration in the air was 4.7×10^{-8} μc/cc.

A recent malfunction of one of the control rods was attributed to the presence of some foreign material in the collet piston section of the drive system. The material is believed to have caused the collet fingers to cock.

DETAILS

I. Scope of Visit

A. D. Johnson, Reactor Inspector, Region V, Division of Compliance visited the Pacific Gas & Electric Humboldt Bay Nuclear Power Plant (Unit No. 3) at Eureka, California on November 9, 10 and 11, 1965. The purpose of the visit was to review core reloading operations and modifications being made to the containment and reactor systems as required by Change No. 17 to the license.

The visit included a tour of the facility, a review of blueprints, procedures, operating records and discussions with facility personnel.

Principal areas of discussion and review were:

1. Refueling operations
2. Fuel inspections
3. Control rods
4. Facility changes
5. Radiation Safety

Personnel contacted included:

D. Nix - Plant Superintendent

E. Weeks - Technical Supervisor

W. Raymond - Assistant Plant Superintendent

J. Shiffer - Nuclear Engineer

G. Allen - Radiation Protection Engineer

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II. Results of Visit

A. Core Reloading

The inspector discussed the reloading of the reactor core with Mr. Shiffer. Forty new Type-II Zircaloy clad fuel assemblies were inserted into the reactor core. Sixteen of the thirty-two control rods were replaced with new control rods which have a lower reactivity value. A detailed step-by-step procedure was used for the fuel reloading and repositioning operation. The fuel in the existing core was divided into five groups according to exposure as follows:

Assemblies	September 1, 1965 Exposures (Estimates) MWD/T
Group 1	_ 8,000
Group 2	_ 7,000 8,000
Group 3	_ 5,600 7,000
Group 4	_ 5,600
Group 5	-

Figure 1 shows the final reload status of the core.

Conditions established for the refueling operations were noted to be as follows:

1. A portable area radiation monitor equipped with an audible alarm was mounted over the reactor to provide continuous radiation monitoring of all refueling operations.

2. Prior to the refueling outage, the following operational tests were satisfactorily completed:

   a. Control rod coupling integrity check.

   b. Refueling building ventilation system automatic operation functional test

   c. Master reactor switch functional test

3. The reactor safety system was set up with the following scram sensors operable and capable of causing a reactor scram:

   a. Reactor low water level

   b. Loss of 115 volt preferred a.c.

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c. Scram dump tank, high level

d. High neutron flux (1 out of 3 coincidence)

e. Short period

f. Manual scram

4. Both startup channels (6 & 7) were in service and were monitoring a significant count rate (> 10 cps) above noise level. One of the detectors (a fission counter) was located in the core and was always maintained in a position within two bundles of the vicinity of the reactivity addition. The detector was located such that fuel was always between the detector and the neutron source. A change of neutron population was always indicated whenever reactivity changes were made, according to Shiffer.

5. The liquid poison system was required to be in operation at all times during the refueling operation.

6. The following plant systems were required to be in service during the refueling operations:

   a. The shutdown cooling system was in operation to remove decay heat from the core.

   b. The cleanup system was in service until core work was completed.

   c. The ventilation system was set up for normal refueling operation.

   d. The stack gas monitor systems were in operation.

   e. The Refueling Building area and isolation monitoring systems were in service.

   f. Head sets and intercom systems were checked out and available for communication with personnel in the reactor control room.

7. Fuel handling tools were checked for proper use.

8. A short form instrumentation checklist was completed every 24 hours during core modifications except during critical test operations. The normally required instrumentation

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checks were made whenever the critical tests were performed.

9. Reactor core layout boards were set up in the reactor control room and the Refueling Building. These boards were maintained up to date throughout the core reloading operation.

10. The Refueling Building air was sampled and analyzed for iodine at least once a shift. No significant iodine was detected.

Because the neutron flux in the large control rod water gap is approximately a factor of 1.38 higher than the average neutron flux, the corner fuel rod next to the water gap of each fuel assembly has the highest burnup of any fuel rod in the assembly. In order to extend the core life of these fuel rods all fuel assemblies with greater than 5600 MWD/T exposure were rotated 180 degrees in their positions. This rotated the corner rod with the highest burnup to the smaller water gap existing between fuel assemblies where there is no control rod blade.

The core was reloaded a quarter at a time in a clockwise direction starting with the northwest corner, until the core was fully loaded. Shutdown margin checks were performed following the quarter core, half core, and full core loadings. The shutdown margin checks showed that in any array, the two strongest rods could be fully withdrawn without reaching criticality.

B. Fuel Inspection

Thirty-two of the 152 fuel assemblies that were scheduled to be "sipped" were determined to have cladding failures. Inspection of the fuel assemblies showed that gross cladding failures had occurred. The inspector reviewed some photos of the failed fuel provided by Mr. Raymond. These photos showed that sections of fuel rods were actually missing. The conclusion reached by the licensee and G-E, based on their inspection, was that the clad failures resulted from stress corrosion and not burnout. Complete results of the fuel inspection were not available at the time of this visit since only about 10 assemblies had been inspected.

It was noted during the inspection of the fuel that scale was present on the fuel rods. The amount of scale (8 to 10 mils) appeared to be directly related to the exposure and the relative heat flux that the element had been subjected to during operation. Measurements were

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made of flow characteristics in fuel assemblies with exposure histories similar to those presently in the core. According to Weeks, the data obtained indicated that some fuel assemblies showed an approximate 15% reduction in flow. The data have been forwarded to the General Electric Co. for an analysis of the effects of the flow restriction on existing power operation thermal limits. The licensee does not plan to operate the reactor at full power until the results of G-E's evaluation are known. Mr. Weeks indicated that in all probability the reactor would not be operated above 52 Mwe during the next operating period regardless of G-E's evaluation. This reduced power operation will be done in an attempt to stretch the life of the stainless steel clad fuel.

C. Control Rods

According to the requirements of the technical specifications, control rod drive friction tests were due to be performed on or before August 17, 1965. However, on August 13, 1965 by letter from R. L. Doan, DRL, to R. H. Peterson, PG&E, the licensee received permission to delay the tests for a few days. These tests were conducted on September 21, 1965. A comparison of these data with data obtained from previous tests did not disclose any significant or unusual friction values. The average pressure required to start a rod to move was about 100 lbs psig. The pressure required to keep the rod moving at a constant rate of speed was $\pm$ 4 to 5 psig from the initial pressure required to start the rod moving.

A recurrence of the control rod drift-out problem (see Compliance Report No. 133/65-4 Section A.4) was experienced on October 6 when rod F-3 was being exercised. In this instance it was definitely shown that all Aktomatic valves were closed and that a sticky collet piston was the source of the malfunction. The fuel was unloaded from cell F-3 and drive F-3 was replaced with a spare after the core work was completed. Inspection of the rod drive showed the rod to be in excellent condition according to Raymond. However, evidence was found which indicated that a piece of foreign material had rubbed on the collet finger and had possibly cocked the collet finger.

D. Facility Changes

The inspector reviewed facility changes made under the provisions of 10 CFR 50.59 with Mr. Weeks. The changes reviewed are as follows:

1. Compensated Ion Chambers

Compliance Report 133/65-1 Section D. discusses difficulties experienced with overcompensation of the original CIC's used at the facility. Two new CIC's were installed for Channels 2 & 4 (log $_N$s) during the refueling outage in August of 1964. Since the performance of these two chambers has been satisfactory, the three ion chambers for the three

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safety channels were also replaced with the new compensated ion chambers during the refueling outage covered in this report.

2. Control Rod Hydraulic Flow Control Valves

All of the true-flow hydraulic control valves for the control rod drive system have been replaced with a new type of flow control valve known as color-flow valves. The new needle type flow control valves were acquired to eliminate the difficulties experienced with the old true-flow valves. Two of the new type flow control valves were installed in the summer of 1964 and no difficulty has been experienced with this type of valve to date.

3. Master Scram Valves

Two auxiliary master scram valves have been installed in parallel with the existing master scram valves. If difficulty occurs with the master scram valves, the control system can now be switched to the auxiliary set of master scram valves without shutting the reactor down for repair. According to the licensee, any malfunction of these valves would result in a scram of the reactor. The master scram valves serve as a backup to the individual control rod scram valves.

4. Individual Scram Valves

Air shutoff valves have been installed in the individual scram valve air supplies. This change will enable repairs to be made to an individual scram valve during operation. Loss of air to these valves results in a scram of the individual control rod. The improved pilot valve sub-assemblies discussed in CO Report 133/65-2 Section A have been installed on all of the individual control rod scram valves.

5. Dry Well Cooling

The dry well air cooling capacity has been increased by approximately 50%. This was accomplished by installing two new cooling fans. The additional cooling capacity was provided to insure that the dry well air temperatures will be maintained below the technical specification limit of 175°F during full power operation.

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6. Reactor Vessel Safety Valve Vacuum Breakers

The tie-in of the vacuum breakers on the reactor vessel safety valve lines was moved from the suppression chamber side of the dry well vacuum breakers to the dry well side. (See Figure 56 of the Final Hazards Summary Report) This change eliminates the potential passage of air between the dry well and suppression chamber.

7. Technical Specifications - Change No. 17

The inspector toured the facility and observed the modifications that were being done as required by Change No. 17 to the facility's technical specifications. In addition, the changes were discussed and the blueprints were reviewed with Mr. Weeks. The status of the modifications at the time of the visit were as follows:

a. Additional Support for the Radial End Walls of the Suppression Chamber

The I-beams required to provide additional support for the radial end walls of the suppression chamber were observed to be installed as shown in Figure I of the licensee's Addendum B to Proposed Change No. 17. It was noted that it was necessary to cut the I-beams in half in order to get them into the suppression chamber. The beams were then welded back together and placed into position. Mr. Wollak, the Civil Engineer responsible for structural design of the modification, reported to the on-site review committee that he was satisfied that the modification had been made in accordance with design, with only one exception. The one exception was in the suppression chamber where the full bearing surface of the I-beam end pads were not being fully utilized on some of the deflected beams. There was some concern that the resulting concentration of force over too small an area might result in crushing the concrete behind the chamber liner. This problem was taken care of by filling the space between the pads and the chamber lining with plastic steel.

In addition to adding support for the radial end walls of the suppression chamber, the roof of the suppression chamber was reinforced by keying a concrete beam into the west wall of the shutdown room and floor. This beam was approximately 15"x18"x12' in length. The suppression chamber roof was also reinforced on the southeast side by installing a 24-inch wide flange, 130 pound per foot, standard I-beam which was keyed

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to the cement walls. These two additional reinforcing members were installed to give additional assurance to the integrity of the suppression chambers at these locations.

At the time of the visit the licensee was measuring the thickness of the suppression chamber roof to assure that it was approximately 3-feet thick. The reason this check was being made was because a Bechtel civil engineer, who was on site during construction of the plant, mentioned in a discussion with Mr. J. C. Carroll that the actual thickness of concrete in the area of concern was about 2½ feet. Proof tests of the system could not be run until the actual thickness of the roof was determined and calculations made to insure that an adequate margin of safety exists.

b. Reactor Vessel Vent System

The modification required by Addendum B had not been completed at the time of the visit because the two air actuated valves had not arrived from the vendor.

c. High Pressure Core Flooding System

A manual valve has been added between the hotwell makeup line and the condensate pumps to enable the use of the full capacity of the condensate storage tanks. This modification was required by Section III.D of Proposed Change 17.

d. Low Pressure Core Flooding System

A second core flooding system was being installed to provide back up for the core spray and the feedwater condensate systems. The system consists of a connection between the yard fire water system loop and the shutdown system supply line to the reactor vessel, as shown in Figure 2 of Proposed Change No. 17. At the time of the visit the system was completed except for the installation of the automatic valve.

To increase the reliability of the fire system to provide core flooding water, a third motor driven fire pump has been installed. It was noted that the new pump's rating was 500 gpm at a 275-foot head. The

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pump was tied to load center #9 on Unit #2. The instrumentation required for the automatic operation of the core flooding system motor operated valves was noted to have been installed.

The diesel driven fire pump has been modified so that the pump is now provided with an automatic starting feature which will start the pump if the pressure in the system falls below 150 psig. These modifications were all found to be consistent with Change No. 17 to the license.

e. Reactor Coolant Cleanup System

The reactor coolant cleanup system has been modified in order to provide a second post accident cooling system. An air operated valve has been installed on the cooling water inlet to the cleanup (nonregenerative) heat exchanger. Since the flow is normally throttled, this will permit the cooling water to be increased to full flow for post accident cooling. Also, two air operated valves have been installed as shown in Figure 2 of Proposed Change No. 17 to allow bypassing the cleanup regenerative heat exchangers and the cleanup demineralizer. In addition, valve 19 in Figure 3 was replaced with an air operated valve that will automatically close on initiation of operation of this system, to prevent water from recirculating back through the regenerative heat exchangers. All of the added automatic valves can be operated remotely from the control room.

An alternate 480 volt feeder to the cleanup pump motor and air operated valves has been installed. The alternate power supply requires a manual transfer.

f. Dry Well Water Level Alarms

A second level switch has been installed to insure initiation of an alarm in the event the dry well water level approaches the elevation of the 40" vent pipe entrances during post accident dry well flooding.

Mr. Raymond informed the inspector that Mr. G. D. Rice of the Hartford Insurance Co. is inspecting all welds that have or will have anything to do with containment penetration. The completed inspection reports of the welds will be reviewed during the next visit to the facility. Mr. Raymond also stated that all systems involved in the recent modifications will be thoroughly tested for proper operation prior to operation of the re

E. Reactor Vessel Flange Cracks

Mr. Raymond showed the inspector photos of three small radial cracks which had been found in the southeast quarter of the reactor vessel flange. According to Raymond, they were ground out and repaired by weld inlays of Inconel-82. The three cracks averaged approximately 3/4 inch in length and about 250 mils in depth. The cracks did not extend into the main part of the reactor vessel and appeared to be confined to the Inconel overlay.

F. Refueling Building Airlocks

Two airlocks are provided for personnel entry into the Refueling Building. One airlock is located on the south side of the Refueling Building at the turbine auxiliary building operating level. The other airlock is at the northwest corner of the Refueling Building, opening into the yard area. Each airlock consists of two mechanically interlocked doors with weather strip type seals around the edges. The door interlocks require one door to be closed before the other may be opened. This provision maintains the negative pressure of the Refueling Building as personnel enter and leave. The interlock on either door can be easily overridden by manually actuating the interlock device on the door, regardless of the position of the door. However, since the primary containment scheme for the facility during power operation is the drywell-pressure suppression system, containment of the Refueling Building is not required at all times. Section III.B of the technical specifications lists the conditions under which Refueling Building containment may be broken.

G. Containment System Strength and Integrated Leak Rate Testing

Messrs. Shiffer and Weeks discussed the procedure to be used in the strength and integrated leak rate testing of the containment system with the inspector. Test plans were as follows:

1. A dry construction proof test will be run at 28.75 psig air pressure to:

   a. Verify that the upper wall and ceiling of the suppression chamber can withstand the test pressure, and

   b. Check that there is no gross leakage in any of the new weld joints made during the outage. (Leakage at weld joints below the chamber operating water level would not be detected during a wet proof test.) Following this test, the suppression chamber will be filled with water to the normal operating level (18 feet) and a pneumatic strength test will be made of the entire

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containment system at 28.75 psig free air space pressure. Following the strength tests, containment leak rate tests of the dry well and suppression chambers will be conducted at 10 psig and 25 psig, respectively, for a duration of 24 hours each. Both the absolute method and the reference chamber method of determining the leak rate of the systems will be used. Following the leak rate tests the accuracy of the measurements will be verified by performing a test with a controlled leakage rate superimposed upon the measured leakage rate, and determining the composite result.

Equipment being installed at the time of the visit included instrumentation to measure the temperature and relative humidity in the chamber and fans to circulate the air within the chamber.

During the strength test the railroad doors of the Refueling Building will be open with someone standing by to close them if necessary. This will be done to insure that the Refueling Building will not become overpressurized in case of a major leak or rupture in the dry well-pressure suppression containment system.

I. Contamination Control

The radiation safety practices employed at the plant to prevent contamination from spreading to unrestricted areas were discussed with Mr. Allen. Mr. Allen told the inspector that no significant amounts of contamination have been found during the routine surveys of the unrestricted areas. All personnel coming from contaminated areas must pass through a controlled access way. At this point radiation survey instrumentation is available for personnel to check themselves for contamination. Once per shift, the step-off pads at the controlled access ways are checked for contamination. A review of these records showed that on several occasions the step-off pads were contaminated to a level of about 500-1000 counts per minute. On one occasion contamination was found in the general walkway through the control room. This area is surveyed daily by a radiation control technician. Mr. Allen said that it is a daily routine to also check the control room area and that if contamination is found it is cleaned up immediately.

Prior to leaving the reactor control room each individual is required to check himself on the hand and foot counter located at the exit. Personnel and their clothing must be free of detectable contamination (less than 100 counts per minute beta-gamma) before they are

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released to go home. Exceptions require the specific approval of the Radiation Protection Engineer. Mr. Allen stated that on one occasion an employee was permitted to go home with contamination on the heel of his shoe indicating 500 counts per minute. However, this was only after an attempt to remove the contamination was made and after it was concluded by the Radiation Protection Engineer that the contamination was fixed.

J. Radiation Areas

The inspector noted, while touring the facility with Mr. Weeks, that radiation areas in the radiation controlled access area were properly roped off and identified. It was also observed that shielding had been installed around the control rod drives in the dry well to reduce exposure to personnel during the installation of the additional new dry well air cooling equipment. The general whole body radiation level outside the temporary shielding was approximately 300 mrem/hr, according to Weeks, whereas the levels behind the shielding were several roentgens per hour.

K. Personnel Exposures

In response to a query regarding the maximum quarterly personnel exposures to date, Mr. Allen stated that an administrative limit of 2400 mrem/quarter had been established to insure that no personnel exposures occurred over the 3000 mrem/quarter permitted by 10 CFR 20. The highest quarterly exposure was approximately 2350 mrem for the past quarter, according to Allen.

L. Tritium

The tritium concentration in the atmosphere above the open reactor vessel during refueling operations has been evaluated by the licensee to be less than concentrations permitted in unrestricted areas as shown in Appendix B of 10 CFR 20. The measurement of the tritium air concentration was performed by passing a known amount of air through a cold trap which was maintained at a temperature of approximately -40°F. The condensed water was then analyzed for tritium and the results of the analysis used to calculate the tritium concentration in the air. The calculation indicated that the tritium concentration was 4.7×10^{-8} µc/cc. The temperature of the reactor water was approximately 90°F when the air sample was taken, according to Mr. Weeks.

M. Documentation of Facility Changes

The procedures used by the licensee in authorizing a facility change under the provision of 10 CFR 50.59 were discussed with Messrs. Nix,

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Raymond and Weeks. Mr. Raymond described their procedure as follows: If it is decided that a change should be made to the facility, the change is discussed by the on-site review committee at which time a decision is made as to whether the change constitutes an unreviewed safety question or a change in the technical specifications. If the change does not require Commission approval, the change is approved by the Committee and is so indicated in the Committee meeting minutes. After the change is completed a description of the change is then documented for submittal to the Commission in the semiannual report. However, a review of the meeting minutes indicated that it would be difficult to conclude that an adequate safety review of a proposed change had been conducted since the minutes did not contain a written description of the purpose of the change, description of the change, or reasons indicating why the change did not constitute an unreviewed safety question or a change in the facility's technical specifications. When this was pointed out to the group, Mr. Nix commented that they would review their documentation practice concerning facility changes.

N. Fuel Sipping Procedures

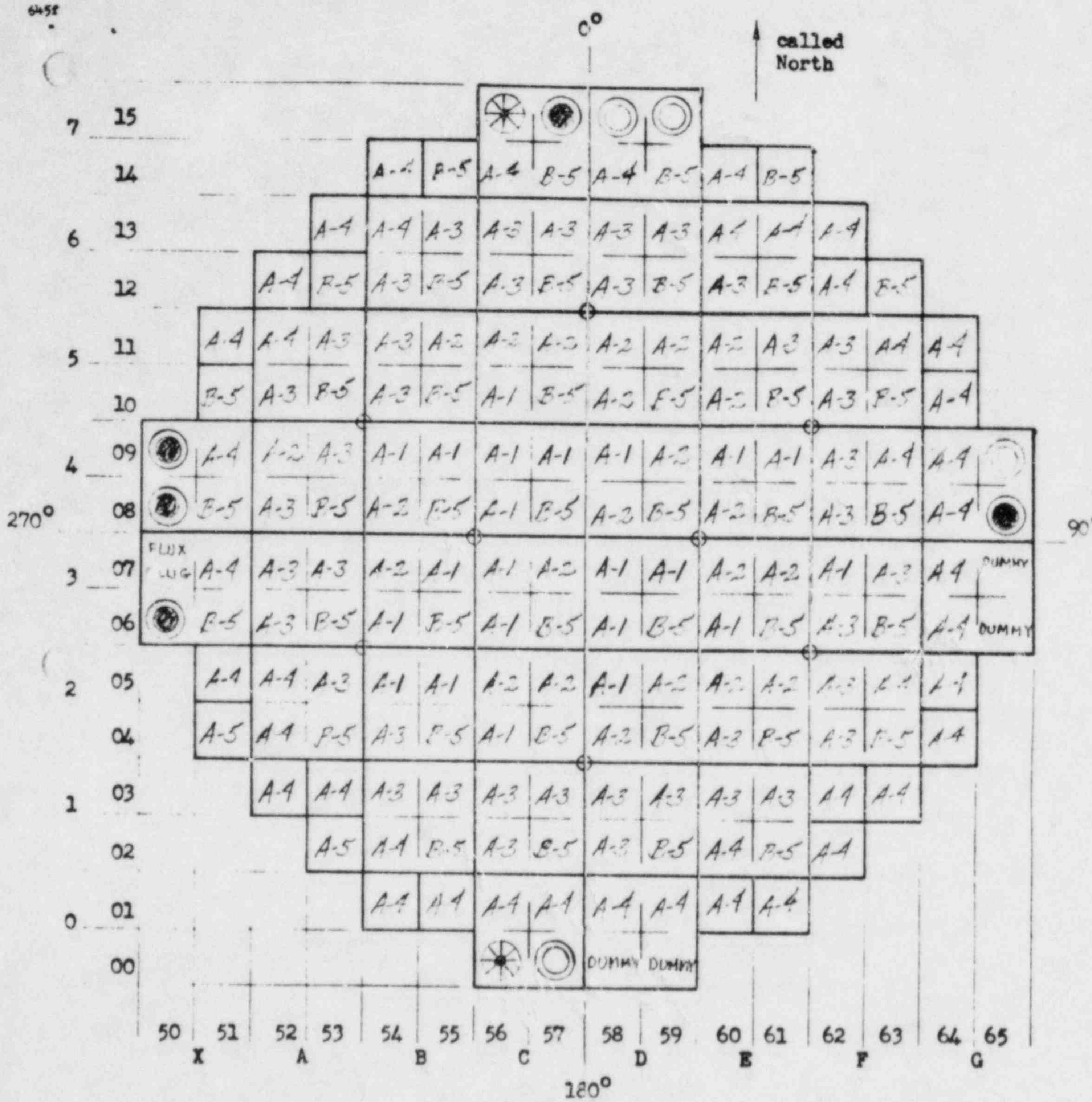
The inspector discussed the adequacy of the fuel sipping procedure with Messrs. Nix, Raymond and Weeks. Mr. Raymond stated that it was their position that the dry sipping procedure and equipment was safe. Mr. Raymond stated that a qualified man would always be present during fuel sipping activities. He defined a "qualified man" as one who would know how and when to initiate corrective action in the event of an emergency. The sipping procedure requires that a nuclear engineer or the shift foreman be present during all fuel sipping operations.

O. Exit Interview

No formal exit interview was held because Mr. Weeks and Mr. Raymond were present while discussions were held concerning the significant portions of the inspection.

Attachment:
Figure 1

FIGURE 1



CORE II-A FINAL ARRANGEMENT (10-11-65)

KEY:

= SOURCE

= CORE PLUG

= ... WINDOW

A = TYPE I FUEL

B = TYPE II FUEL (...)

NUMBERS 1, 2, 3, 4 AND 5 ...