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Chapter 5 - REACTOR COOLANT SYSTEM AND CONNECTED SYSTEMS

CHAPTER 5

5.0 REACTOR COOLANT SYSTEM AND CONNECTED SYSTEMS

5.1 SUMMARY DESCRIPTION

The Primary Coolant (recirculation) System (PCS) consists of the reactor vessel, the steam drum, the reactor recirculation pumps, the interconnecting piping and valves, and the steam drum safety relief valves. A summary description of these components and a simplified schematic is provided in Chapter 1 of this report.

The pressure boundary and system flow are shown on BRP Drawing 0740G40121, Piping and Instrument Diagram - Nuclear Steam Supply System (NSSS).

5.2 INTEGRITY OF REACTOR COOLANT PRESSURE BOUNDARY

5.2.1 COMPLIANCE WITH CODES AND CODE CASES

This subject was initially identified as Systematic Evaluation Program (SEP) Topic V-1, Compliance with Codes and Standards and Topic V-2, Applicability of Code Cases. Topic V-1 was deleted as an active SEP topic by NRC letter dated November 27, 1981 and Topic V-2 was determined to be not applicable to BRP by NRC letter dated November 16, 1979.

5.2.1.1 Compliance With 10 CFR 50.55 (a) Codes and Standards

Table 3-1 in Chapter 3 of this Updated FHSR provides a listing of selected components required to be ASME Section III, Class 1 components and Section 3.9 of Chapter 3 of this Updated FHSR addresses the Inservice Inspection (ISI) Requirements and Inservice Testing (IST) Requirements of ASME Section XI in order to meet 10 CFR 50.55(a). The BRP 40 Year Inservice Inspection Plan and the ISI/IST Program described in Section 3.9 of this Updated FHSR provides the basis for compliance with 10 CFR 50.55(a) Codes and Standards and identifies necessary relief requests.

5.2.1.2 Applicable Code Cases

Table 3-1 in Chapter 3 of this Updated FHSR provides a listing of selected components and the ASME Design and Fabrication Code Cases that were applied.

The BRP 40 Year Inservice Inspection Plan provides information on ASME ISI Code Cases applied during Nondestructive Testing Activities.

5.2.2 OVERPRESSURIZATION PROTECTION

To protect the primary system from potentially damaging over-pressurization transients during plant heatup and during system hydrostatic test operation, administrative controls have been instituted to assure pressurization during these periods remains within the limiting pressure temperature limits (as a function of neutron fluence of the material at the reactor vessel beltline) as prescribed in the BRP Technical Specifications. Because the control rod drive pumps which are used in performing system hydrostatic tests are positive displacement pumps, the possibility of overpressurization in violation of Technical Specification Nil Ductility Transition (NDT) Temperature requirements exists when the system is solid, if the Steam Drum Safety Relief Valves were gagged during the system hydro. Currently, the BRP Steam Drum Safety Relief valves are not gagged, as the hydrostatic test pressure required is below the relief setting of the first valve and below the design pressure of the Primary Coolant System. Thus, the potential for inadvertent over-pressurization during hydrostatic testing is very small.

5.2.3 REACTOR COOLANT PRESSURE BOUNDARY MATERIALS

5.2.3.1 Material Specifications

The Primary Coolant System (PCS) contains both cast stainless steel (CF8M) and wrought stainless steel (SS304 and SS316). The CPCo letter dated May 3, 1962 (Reference 1) Amendment 10, Addenda to Amendment 8 of the FHSR, in tab "M" response to Atomic Energy Commission (AEC) question number 15, provided Certificates of Inspection for the Recirculating and Reactor Riser piping. The details of various tube, piping, vents, reducers, castings, couplings, weldolets, thermal sleeves, reinforcing saddles, elbows, tees, lugs, consumable inserts, and weld rods utilized are provided.

The Certificates of Inspection also provide details of the material specifications and heat numbers.

5.2.3.2 Sensitized Stainless Steel Components

In response to an Atomic Energy Commission (AEC) August 6, 1970 request for information on furnace-sensitized stainless steel components in the reactor Primary Coolant System (PCS), CPCo provided a listing of all known sensitized stainless steel components of and within the reactor coolant pressure boundary including portions of piping. The listing attached to the September 11, 1970 response provided the location, type of material and sensitization process. The results of field measurements of piping displacement from cold and empty conditions to the filled-with-water 365° temperature condition were also provided in this letter. (Reference 2).

Stress Levels

Maximum stress levels for sensitized stainless steel components were provided by CPCo letter dated January 12, 1971 (Reference 3). Table 5.1 below summarizes these levels.

Table 5.1 - Maximum Stress Levels for Sensitized Stainless Steel Components (Reference 3)

Component (2)	Stress Intensity (ksi) (1)(3) Membrane	Peak	Alternating
Reactor Vessel			
Instrument Nozzle Extension (795-4)	9.1	8.7	6.2
Steam Outlet Nozzle Extension (795-13)	10.0	9.6	6.3
Letdown Nozzle Extension (795-17)	9.0	8.7	6.4
Recirculation Nozzle Extension (796-3)	10.5	10.1	6.6
Poison Nozzle Extension (796-8)	9.1	8.8	4.9
Vent Nozzle Flange (807-3)	7.2	5.8	5.7
Core Support Bracket (802-16)	4.0	11.8	<11.8
Core Support Plate Bracket (802-18)	1.4	15.7	<15.7
Diffuser Bracket (802-32)	0.5	3.9	< 3.9
Steam Drum			
Downcomer Nozzle Extension (103-2)	11.2	10.1	13.1
Rizer Nozzle Extension (103-8)	9.9	8.9	5.3
Vent Nozzle Extension (104-7)	5.1	5.7	10.2

Notes:

1. These stress values were calculated by Combustion Engineering, Inc at the request of Consumers Power Company. A copy of this report (Report 6460-A), less the detailed calculation sheets, is attached to Reference 3. A summary of the design practices for the steam drum internal brackets is included in the attached report. Stress levels for the reactor vessel stub tubes were previously submitted with Amendment 8 to the Final Hazards Summary Report (Reference 1) and are summarized in Section 5.3.1.4 of this Updated FHSR.

2. Due to a coordination error on the part of Consumers Power Company, stress levels for the reactor vessel upper support bracket (798-8, 12) were inadvertently not calculated. Therefore, these stress levels are not included.

3. All calculated stresses are less than the allowable values of 14.8 ksi for membrane stress intensity, 16.2 ksi for peak stress intensity, and the material endurance limit of 18.1 ksi.

5.2.3.3 Material Selection and Processing Guidelines for BWR Coolant Pressure Boundary Piping

NRC Generic Letter 81-04 dated February 26, 1981 requested Consumers Power Company to review coolant pressure boundary piping at the Big Rock Point Plant (BRP) to determine if material selection, testing and processing guidelines set forth in NUREG-0313, Rev 1, "Technical Report on Material Selection and Processing Guidelines for BWR Coolant Pressure Boundary Piping (Generic Task A-42)," dated July 1980, are met. It also requested identification of materials that do not meet these guidelines and our basis for actions proposed to conform to the guidelines, including Technical Specifications changes, if necessary.

CPCo, by letter dated June 30, 1981 (Reference 4) provided the evaluation of the coolant pressure boundary piping in accordance with NUREG-0313, Rev 1 guidelines. The results of this evaluation are provided by Attachment 1 to Reference 4. This evaluation concluded that intergranular stress corrosion cracking (IGSCC) is not a safety issue for BRP which is in agreement with the NUREG-0531 conclusion. Therefore, Consumers Power Company does not intend to replace BRP coolant pressure boundary piping or make Technical Specification changes.

<u>CPCo Evaluation</u> (Reference 4)

<u>Maintenance History Relating to IGSCC</u>

The maintenance history relating to IGSCC is very favorable in support of the argument that BRP has a reduced susceptibility to IGSCC. The Inservice Inspection Program has resulted in examinations of portions of those lines purported to be susceptible in accordance with NUREG-0531 and NUREG-0313. The only identified instances of IGSCC occurred in small diameter vent lines of the Reactor Cleanup System. The first of these occurred in 1978 and resulted in a leak. The second and third occurred in 1979 and 1980 and were located by UT examination. Both the first and second instances were confirmed to be IGSCC by metallurgical examination while the third instance was not destructively examined but was similar to the other instances. All of these instances were suspected to be assisted by fatigue in addition to the IGSCC and were associated with an oversized branch connection weld. The remaining three vent lines in the Reactor Cleanup System were also examined but no indications were found.

The Primary Coolant System ISI examinations performed have not identified IGSCC which lends additional support to our theory that BRP has a reduced susceptibility to IGSCC. Six reactor vessel steam outlet nozzles have been examined using mechanized UT from the nozzle ID. Nearly all of the reactor nozzles and safe ends were examined visually from the ID using a closed circuit TV (CCTV) system. The core spray sparger was also examined using the CCTV system. We are confident that any IGSCC indications would have been identified in the components examined.

Material Variables

The stainless steel material variables which must be considered to determine susceptibility to IGSCC include the carbon content, the chromium and other alloy contents, the manufacturing process, the heat treatment and the ferrite content. The BRP coolant pressure boundary piping consists of 304, 316 and CF-8M austenitic stainless steels.

NUREG-0313 has identified cast austenitic stainless steels with more than 5% ferrite content as being resistant to IGSCC in BWRs. The 17 inch downcomer piping and the 20 and 24 inch main recirculation piping to the reactor vessel have been identified as CF-8M cast austenitic stainless steel piping. Although no data exists to verify the ferrite content, Schaeffler diagram calculations establish the minimum ferrite content to be no less than 5%. We therefore consider the downcomer piping and the main recirculation piping to the reactor vessel as conforming to the guidelines of NUREG-0313.

Records are not readily available which identify the actual chemical composition of much of the remaining austenitic stainless steel piping in the coolant pressure boundary. Available records indicate that most of the wrought austenitic stainless steel piping is Type 304, except the 14 inch risers which are Type 316. The safe ends on the reactor nozzles are Type 304 and have apparently been furnace sensitized. The steam drum safe ends are Type 316 and are also furnace sensitized. In accordance with the guidelines of NUREG-0313, all of this piping would be considered nonconforming or indeterminate.

Environment

The water chemistry at BRP has benefited by the use of deaeration procedures during start-up since approximately 1978.

Most of the piping and safe ends are relatively free of highly oxygenated stagnant conditions which increases the susceptibility to IGSCC. Some apparent exceptions are the Reactor Cleanup System vent lines (which have an IGSCC failure history as discussed earlier) and the creviced area behind the thermal sleeves on the primary core spray and the liquid poison nozzles.

Stress

The stress conditions of the coolant pressure boundary piping are expected to be the most favorable factor in the theory that BRP has reduced susceptibility to IGSCC. Research has shown that all three contributing factors of material variables, environment and stress must be present in sufficient levels to obtain IGSCC. The history of IGSCC in BWRs has shown that most cracking, where the three factors are present, will start to occur in a relatively short time (ie, a few years). The excellent Inservice Inspection (ISI) history of BRP with regard to IGSCC is very likely due to low stress levels. The IGSCC failures which have occurred have all been associated with oversized welds and vibrational loads which are not representative of the systems in general. Therefore, stress levels are probably below the threshold value of stress necessary for IGSCC initiation in the BRP coolant pressure boundary piping, except as previously noted for portions of the Reactor Cleanup System.

CPCo Evaluation Conclusion

Consumers Power Company concludes that IGSCC at the Big Rock Point Plant poses no safety hazard to the public. Stainless steel is an inherently tough material and the piping systems at BRP have adequate structural integrity to eliminate safety problems should IGSCC occur. This is the same conclusion reached by NUREG-0531. However, we do not postulate IGSCC occurring in most of the stainless steel coolant pressure boundary systems at BRP. Our basis for arriving at this conclusion is that BRP has an extensive operating history where both susceptible material variables and an enabling environment exist. In spite of this, our experience with IGSCC remains excellent. We are attempting to confirm by analysis that the stress levels in the reactor vessel safe ends are below threshold values necessary for IGSCC initiation. (This analysis confirmation is addressed in Section 5.2.3.4 below.)

5.2.3.4 Furnace Sensitized Stainless Steel Safe-Ends

Background

During the fabrication of nuclear pressure vessels, a small section of austenitic stainless steel piping is welded to each vessel nozzle. This pipe section, called a safe-end, is added so that the subsequent attachment of piping to the vessel will be a piping-to-safe-end weld rather than a piping-to-nozzle weld. This arrangement eliminates the need to stress relieve the nozzle in the field which would otherwise be required. Unfortunately, the vessel stress relief heat treatment may, depending on the safe-end material, environment and service loads, render the safe-end susceptible to intergranular stress corrosion cracking (IGSCC). More specifically, the vessel stress relief heat treatment tends to deplete the chromium content near the grain boundaries in standard grades of Type 304 and 316 stainless steel (commonly used in Liquid Water Reactors (LWRs) by a chromium carbide precipitation mechanism. This process is called sensitization. Safe-ends which are sensitized by the vessel stress relief heat treatment are said to be furnace sensitized. Welding of these materials which causes a more local and less severe degree of sensitization is called weld sensitization.

Big Rock Point was built prior to the realization that BWR safe-ends, sensitized during shop post-weld heat treatments, would have a high susceptibility to intergranular stress corrosion cracking.

The NRC, by letter dated April 26, 1982 (Reference 5) notified CPCo that Systematic Evaluation Program (SEP) Topic V-4, "Piping and Safe-End Integrity," was being reopened for BRP. However, the review was limited to only the issue of stress corrosion cracking of sensitized stainless steel safe-ends.

CPCo letter dated May 28, 1982 (Reference 6) provided the technical basis and detailed response to the NRC April 26, 1982 information request. That response and attachments contain 1) a description of the inaccessibility of the safe-ends for inspections; 2) a request for relief from performing the required inspections; 3) an assessment of the Big Rock Point furnace sensitized safe-ends which includes: a) a Stress Rule Index Evaluation; b) a Failure Analysis Diagram Calculation Example; c) IGSCC Damage Index Calculations; and d) a Tearing Instability Analysis Example; and 4) a conclusion that IGSCC is not a safety issue at Big Rock Point.

Stress Calculation (Reference 6)

A report of the assessment of the possibility of IGSCC damage to the Big Rock Point furnace sensitized safe-ends has been completed by S Levy, Inc. (see Attachment IV of the May 28, 1982 CPCo letter). Specifically, likelihood of cracking, possible severity and safety significance of cracking have been explored in general terms and the specific geometries, materials, damage and safety indexes for the individual safe ends have been evaluated. Based on this assessment, it is concluded that Big Rock Point furnace sensitized safe ends are not typical of the population of BWR furnace sensitized components which have experienced cracking. Except possibly for two creviced safe end geometries, if a safe end crack should occur, its safety significance should not be different from that of a pipe crack.

NRC Safety Evaluation Conclusion (Reference 7)

The NRC provided a final evaluation of SEP Topic V-4 by letter dated December 30, 1982. The review guidelines and acceptance criteria for Topic V-4 are found in NUREG-0313, Rev 1. In summary, the acceptance criteria applicable to sensitized BWR safe-ends are:

1. The affected safe-ends should be modified, to the extent practical, to meet current materials selection requirements;

2. For those affected safe-ends not modified, leak detection systems should be provided that will detect leaks in accordance with Regulatory Guide 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems," and an augmented inservice inspection program capable of detecting flaws in austenitic stainless steel should be implemented.

Conclusion

Pending completion of NUREG-0313, Rev 2, the staff concludes that the examinations of the Big Rock Point safe-ends as proposed by the licensee and the additional examination of recirculation inlet nozzle 796-1A agreed to by the licensee (as supplemented by letter dated February 18, 1983) will provide assurance of the structural integrity of primary pressure boundary. The leakage detection requirements for Big Rock Point will be reviewed under SEP Topic V-5 (refer to Section 5.2.5 of this Updated FHSR) and the frequency of future inspections is being reviewed generically and will be resolved with the implementation of NUREG-0313, Rev 2.

5.2.3.5 Inspections of BWR Stainless Steel Piping - Generic Letter 84-11 (Reference 8)

By letter dated April 19, 1984, Generic Letter 84-11 provides recommended actions, including augmented inspections, that would be considered an acceptable response to current IGSCC concerns, and requests responses to questions regarding inspections, examiners, previously noted indications and remedial measures.

CPCo responded to this request by letter dated May 25, 1984 (Reference 9). Further information concerning this response and our IGSCC Program of Inspections are addressed in Section 3.9.3 of this Updated FHSR.

NRC Safety Evaluation Conclusion (Reference 10)

By letter dated June 26, 1984, the NRC transmitted their Safety Evaluation of the implementation of NUREG-0313, Revision 1. A correction to the evaluation to include pages of the Idaho National Engineering Laboratory EG&G Technical Evaluation Report which were inadvertently omitted was transmitted to CPCo July 25, 1984.

Conclusion

Based on a review of the contractor's evaluation, the staff concludes that the licensee's responses to reduce the IGSCC susceptibility of the austenitic stainless steel piping in their Big Rock Point Plant did not fully meet the guidelines set forth in NUREG-0313, Revision 1. However, because of the recent industry-wide experience and the other current industry-wide activities to mitigate the IGSCC (including the inspection conducted in responses to regulatory actions; eg, Bulletins 82-03, 83-02 and the confirmatory orders) that are not reflected in the licensee's responses, we conclude that the question regarding whether or not NUREG-0313, Revision 1 guidelines are being met is moot at this time. Further staff actions will depend upon the licensee's response to Generic Letters 84-07 and 84-11.

CPCo Resolution

Resolution of Generic Letter 84-07, "Procedural Guidance for Pipe Replacement at BWRs," is concerned with guidance for 10 CFR 50.59, "Changes, Tests, and Experiments," in the event that a decision is made to replace Reactor Coolant Pressure Boundary or Recirculating System Piping. Implementation of this guidance is not applicable until such time as a decision is made that the affected piping must be replaced or modified.

Resolution of Generic Letter 84-11 "Inspection of BWR Stainless Steel Piping," is being accomplished via the BRP Intergranular Stress Corrosion Cracking Inspection Program in Section 3.9.3 of this Updated FHSR for component inspections and in Section 5.2.5 for the Leakage Detection and Leakage Limitation requirements.

5.2.4 INSERVICE INSPECTION AND TESTING OF REACTOR COOLANT PRESSURE BOUNDARY

Refer to Section 3.9.2 of this Updated FHSR.

5.2.5 DETECTION OF LEAKAGE THROUGH REACTOR COOLANT PRESSURE BOUNDARY (RCPB)

5.2.5.1 Primary Coolant System Leakage Limits

Leakage limits for the BRP RCPB are specified in the Technical Specifications.

5.2.5.2 Leak Detection Systems (Reference 9 and 11)

CPCo provided a revised evaluation of Plant Leakage Detection Systems by letter dated June 6, 1983 (Reference 11), and in response to Generic Letter 84-11 provided clarification of the leakage detection systems utilized (Reference 9).

Guidance (Reference 9)

Regulatory Guide 1.45 (May, '73) recommends that at least three separate detection systems be installed in a Nuclear Power Plant to detect unidentified leakage from the Reactor Coolant Pressure Boundary to the Primary Containment of one gallon per minute within one hour. Leakage from identified sources must be isolated so that the flow rates may be monitored separately from unidentified leakage. The detection systems should be capable of being monitored in the control room.

Of the three separate leak detection methods recommended, two of the methods should be 1) sump level and flow monitoring and 2) airborne particulate radioactivity monitoring. The third method may be either monitoring of condensate flow rate from air coolers, or monitoring of containment atmosphere and pressure, and should be considered as alarms of indirect indication of leakage to the containment. In addition, provisions should be made to monitor systems interfacing with the reactor coolant pressure boundary for signs of intersystem leakage through methods such as detection of radioactivity and water level or flow monitoring.

Systems (Reference 9)

The Big Rock Point Plant utilizes the following systems which provide the necessary monitoring to assure compliance with 10 CFR 50 Appendix A, Criterion 30, which states: "Means shall be provided for detecting and, to the extent practical, identifying the location of the source of reactor coolant leakage." This is accomplished by utilizing the leakage detection system that are recommended by Regulatory Guide 1.45.

- Sump Level Monitoring
- Sump Pump Run Times (Flow Rate)
- Containment Humidity Monitoring
- Containment Temperature Monitoring
- Containment Pressure Monitoring
- Acoustic Monitors
- Reactor Vessel Head Leak Detection
- Airborne Radioactive Particulate Monitoring (via Continuous Air Monitoring Devices)
- Airborne Radioactive Gaseous Monitoring (via Stack Gas Monitor)
- Intersystem Leakage Detection (via Liquid Process Monitors and Reactor Cooling Water Tank Observations)

5.2.5.3 Description of Leakage Detection Systems (Reference 11)

A detailed description of the leakage detection systems and their ability to detect RCPB leakage is provided in the June 6, 1983 submittal and the following excerpts are provided:

5.2.5.3.1 Sump Level Monitoring

Three sumps are located in the Big Rock Point containment to monitor leakage from the RCPB. The rod drive room sump (RDR) monitors primarily in the reactor vessel area including any leakage from the control rod drive and incore penetrations. The enclosure clean sump (ECS) monitors the "identified" leak rate (sample stations, scram dump tank, RCP seal leakage, safety valve discharge, etc). The enclosure dirty sump (EDS) monitors the "unidentified" leak rate through all the floor drains in the containment. This sump would detect leakage caused from any form of degradation of the RCPB. Accumulation in the RDR sump is also pumped to the EDS to permit processing.

The level of all three sumps is monitored via manometers installed in the sumps. Based upon the dimensions of the sumps, volume sensitivities have been determined and with the manometers installed via Facility Change FC-219 during the 1982 refueling, readings to the nearest tenth inch are available.

Besides the manometers located in containment, sump level transmitters are installed on the ECS and EDS with remote indication in the radwaste area. These are available to monitor transients when containment access is not available. High level alarms are available in radwaste area with a remote alarm installed in the main control room. This alarm would notify control room operators of any leakage condition greater than the capacity of the sump pumps (50 gallons per minute) or of sump pump circuit failure.

Preventive maintenance and calibration is performed on instrumentation included in this scheme. Additionally, a primary system leak rate surveillance test using the sump manometers is performed daily. This provides a functional check that the sump level instrumentation is performing properly.

5.2.5.3.2 Sump Pump Run Times

Each sump pump (2 ECS, 2 EDS) is equipped with run time meters located in the control room. The meters indicate run times in minutes to the nearest tenth. Based on a sump pump rated capacity of 50 gallons per minute, sensitivity of the run times is calculated at five gallons per increment.

Sump pump run times are recorded and trended by the control room operators. Auxiliary operators also monitor radwaste tank levels which provide indication of pump run times. Trending of pump run times provides the operators information necessary to detect significant RCPB leakage increase and to permit corrective action.

These meters do not require maintenance/calibration. However, with readings taken and trended an adequate functional check is being performed.

Based upon EDS pump run times, investigations are initiated to determine the source of leakage when expected run times are exceeded. Should the source of leakage be classified as RCPB leakage, leak rate tests are performed to ensure plant operation is within the Technical Specification limit.

5.2.5.3.3 Containment Humidity Monitoring

Dewpoint instrumentation is installed at five locations within the Big Rock Point containment to detect steam leaks. These locations are the pipeway exhaust duct, personnel lock, new fuel storage, emergency condenser, and sphere exhaust air.

The last four locations are monitored and recorded in the control room. The recorder which also monitors containment temperature at these locations alarms in the control room. High temperatures or dewpoints actuate the alarm to enable investigation and corrective action.

Although sensitivity values for this system cannot be determined since response is governed by proximity to the source, the system has exhibited extremely good responsiveness to changes in containment atmospheric conditions.

The pipeway exhaust duct steam leak monitor utilizes the ventilation system as a medium to detect RCPB leakage. The pipeway and steam drum area, ion tubes, and reactor annulus are provided with two 1/2 capacity cooling units in parallel located at elevation 616 feet. Each unit is equipped with isolation dampers, water-type coils, filters and automatic controls. A variable amount of air, approximately 4000 - 6000 cfm, is continuously bled from this system and exhausted to the plant stack. An equal amount of air is introduced into the system by infiltration through minor openings and a make-up damper which is automatically controlled to maintain a slight negative pressure in the pipeway. Location of the dewcell in the pipeway exhaust duct places the dewcell in the only air stream leaving the pipeway area which provides adequate coverage of all areas where an RCPB leak could occur. All air leaving the steam drum, recirculating pump room, control rod drive room, and reactor annulus area must pass through the pipeway exhaust duct to reach the exhaust plenum.

This detection scheme functions in comparative manner looking at the dewpoint of the exhaust duct atmosphere and the dewpoint of containment atmosphere near the discharge of the supply air fans. These two dewpoint signals are compared and should a differential dewpoint setpoint be reached, the "pipeway steam leak" alarm, located in containment is actuated. Upon actuation of this alarm, a remote alarm is annunciated in the control room. At this time, operators can consult other leak detection information (temperature, sump run times, humidity) to quantify the leak or dispatch an auxiliary operator to the differential dewpoint recorder also located in containment.

Tests performed on the pipeway exhaust dewcell in previous years indicate that the dewcell will respond to a change in dewpoint of 1°F in less than 90 seconds (51°F to 52°F at ≅4600 cfm).

This is extremely good sensitivity for moisture detection but two limitations exist due to characteristics inherent with dewcell design and operation. The exhaust air temperature during summer operation with both reactor recirculation pumps in service goes above the 120°F level and is out of the range for accurate interpretation. Also, at times the dewcell is difficult to interpret because of the lag in response when outside air moisture content changes. We have observed downscale readings on this reference type system when outside air increases in moisture content. Although the above limitations exist, it is not considered that they significantly affect the overall ability to detect leaks. Dewcell system inaccuracy due to such high or changing temperatures occurs infrequently. Also, additional systems exist which can detect leakage in the event that the dewcells become inaccurate.

The dewcells are checked, cleaned, saturated and calibration is performed on an established frequency.

5.2.5.3.4 Containment Temperature Monitoring

Containment temperature is monitored at five locations with RTD type sensors:

1. Personnel lock inside the sphere
2. New fuel storage area
3. Emergency condenser area
4. Pipeway return to cooler
5. Pipeway exhaust

Although sensitivity values have not been determined for the system, it's detection capability exhibited during minor leakage occurrences is extremely good. Actual monitor response is dependent on its proximity to the source of the leakage. Through experience this system has provided considerable confirmations in detection and location of small leaks in the containment.

As discussed earlier, the containment temperature sensors are trended utilizing the multi-channel temperature recorder in the control room. Temperatures exceeding 100°F/120°F actuate an alarm which is also located in the control room.

The system is functionally checked during power operation.

5.2.5.3.5 Containment Pressure Monitoring

Containment pressure is monitored by two channels consisting of a transmitter and recorder located in the control room. The range of these channels is from -5 to 115 psig in one pound increments. This instrumentation was installed in 1981 via Facility Change FC-498 to meet NUREG-0737 requirements. The equipment is environmentally qualified.

Although alarms are not installed as part of this instrumentation scheme, four additional pressure switches provide inputs to the reactor protection system. These switches actuate an alarm and initiate reactor trip at a containment pressure of 1 psig increasing.

Since the Big Rock Point containment ventilation system maintains containment pressure at a slight vacuum during operation, a pressure response to RCPB leakage would only occur for large failures. Under these conditions it is felt that other detection systems would identify the leakage much sooner.

The instrumentation in this scheme is calibrated and functionally checked during refueling.

5.2.5.3.6 Acoustic Emissions

The steam drum safety valve monitoring system is an acoustic based system which will monitor the valves and provide the operator with information as to whether the valve is open or closed. The valve monitoring system uses accelerometers mounted on the valve to detect noise caused by flow through the valve. This noise signal is conditioned and applied to an alarm monitor, indicator, and audio monitor located in the control room. This system can distinguish between normal background noise (as when the valve is closed) and a much higher noise when the valve is open.

This system was installed in 1980 via Facility Change FC-489, as part of Big Rock Point response to Three Mile Island Lessons Learned, Item 2.1.3.a, "Direct Indication of Power Operated Relief Valve and Safety Valve Position." NRC acceptability of this system is documented in a letter from NRC to DPHoffman dated May 2, 1980.

Calibration is performed during refueling. Functional channel checks are performed by operators on an established frequency.

5.2.5.3.7 Reactor Vessel Head Leak Detection

A sensing line is connected to the space between the two O-rings on the reactor vessel head flange for the detection of leakage through the seal. Connected to this line are a pressure switch and a volume chamber with an integral float type level switch. A large leak will cause pressure to increase in the line and activate a high pressure alarm located in the control room. A small leak will cause the water level to rise in the volume chamber and activate an alarm located in the control room at a preset level. Sensitivity of this float arrangement is very good.

However, time to reach activation varies with the leak rate. Based upon approximate volumes of the chamber, the alarm is actuated in approximately ten seconds for a one gallon per minute leak rate. However, lower leak rates would lengthen the actuation time.

Preventive maintenance and calibration is performed on an established frequency.

5.2.5.3.8 Airborne and Particulate Monitoring

Airborne and particulate activity is trended via a continuous air monitor (CAM) installed in the exhaust line from the containment ventilation system. This CAM uses a fixed activated charcoal filter to concentrate iodine and a moving paper pre-filter to collect particulates prior to the charcoal filter. The use of a pre-filter reduces interference with iodine detection by various particulates. In addition to the iodine and particulate filters and their associated detectors, the CAM includes iodine and particulate count-rate meters, graphic recorders, and alarm system. The high count-rate alarm system is wired to a "sphere exhaust CAM high activity" alarm located in containment. Upon actuation of this alarm, a remote alarm is annunciated in the control room. At that time operators can consult the graphic recorders to qualify and evaluate the source of the alarm.

The detector, used in conjunction with the activated charcoal filter, is a sodium-iodide scintillation crystal. After passing through the pre-filter, xenon and iodine are selectively trapped in the activated charcoal filter. Iodine-131, with a major gamma peak of .364 Mev, is detected by the sodium iodide crystal which was selected as an optimum fit to the characteristics of the iodine gamma.

Using the methodology contained in a submittal from Dairyland Power Cooperative dated April 16, 1968 concerning the primary system leak detection capabilities at LACBWR, a response to a one gallon per minute RCPB leak rate on the iodine monitor can be expected (Reference 11).

The exhaust CAM is testable during power operation and a functional check of the monitor and calibration are performed on an established frequency.

5.2.5.3.9 Intersystem Leakage Detection

Three systems interfacing with the RCPB at Big Rock Point are of concern and are monitored for signs of leakage:

1. Liquid Poison System
2. Core Spray/Fire System
3. Shutdown Cooling System

Leakage to the liquid poison system is monitored by a low range (0-25 psig) pressure gauge installed in the main control room. This parameter is checked and recorded in the control room once a shift and monitored during hourly readings. Any RCPB leakage would cause a detectable pressure increase.

Leakage to the core spray system is monitored by "tell-tale" lines installed between the motor operated valves in both core spray lines. Leakage from these "tell-tales" is monitored by auxiliary operators during bi-hourly rounds. Additionally, during monthly core spray valve operability tests, leakage from these "tell-tales" is checked to ensure proper valve closure.

Leakage to the shutdown cooling system during operation is monitored by pressure switches installed between the motor-operated isolation valves on the inlet and outlet lines to the shutdown cooling system. Any major leakage would cause a pressure switch actuation and a control room alarm. Any minor leakage from these lines is routed to the enclosure dirty sump and would be detected during the daily unidentified leak rate calculation.

5.2.5.4 Operability and Surveillance Requirements

NRC Standard Technical Specification 3.4.3.1 requires the following systems to be operable:

- Gaseous or Particulate Radioactivity Monitor
- Containment Sump Flow System
- Containment Air Coolers Flow Monitor

This specification also imposes the following surveillance requirements on the above systems:

- Channel Check every 12 hours
- Channel Functional Test every 31 days
- Channel Calibration every 18 months

The basis for the above operability requirements is to ensure that instrumentation necessary to determine that the plant is operating within the RCPB leakage limits (Standard Technical Specification 3.4.3.2).

Although the Big Rock Point Technical Specifications for RCPB leakage limits do not conform to the standard format, they do adequately meet the above basis and ensure the plant is operating within leakage limits. As discussed previously, the containment sump level monitoring and the containment temperature/dewpoint systems are used to determine the RCPB identified and unidentified leakage rates. As stated in the Consumers Power Company submittal to the United States Atomic Energy Commission dated March 27, 1974, the Primary System Leak test will be performed every 24 hours to ensure that the RCPB leakage limits of Big Rock Point Technical Specification 4.1.2.(c) are not exceeded. This Technical Specification also identifies the corrective action requirements of the standard format should leakage limits be reached.

Even though the operability of other systems/schemes described earlier are not governed by Technical Specification requirements, these systems are only used to detect the presence of leakage and are not used to quantify the leakage for comparison to the Technical Specification 4.1.2.(c) leakage limits. Consumers Power Company's position is that the current Big Rock Point Technical Specification format and surveillance requirements adequately control instrumentation operability to ensure that plant operation is in compliance with the 4.1.2.(c) leakage limits.

For additional operability requirements in the event of a confirmed seismic event, refer to Section 5.2.5.6 of this Updated FHSR.

5.2.5.5 Probabilistic Risk Assessment (PRA) (Reference 11)

A PRA evaluation was performed in order to resolve SEP Topic V-5, "RCPB Leakage Detection," and to determine whether additional leak detection systems, over and above those currently in use, are cost beneficial from a risk reduction standpoint. Based upon the analysis performed, it was apparent that the addition of an acoustic monitoring system (proposed change for evaluation purposes) is not economical from the viewpoint of risk reduction.

5.2.5.6 NRC Safety Evaluations (Reference 12 and 13)

By letter dated June 13, 1983 the NRC provided a final evaluation of SEP Topic V-5, RCPB Leakage Detection.

Conclusions (Reference 12)

Our review indicated that systems employed at Big Rock Point to measure reactor coolant pressure boundary leakage do not meet all the recommendations of Regulatory Guide 1.45. Specifically, none of the systems are seismically qualified. All of the other recommendations have been met or equivalent alternatives have been provided.

The necessity for any leakage detection system modifications will be considered during the integrated safety assessment.

Integrated Safety Assessment Conclusions (Reference 13)

NUREG-0828, May 1984, Section 4.16 provided the NRC assessment of SEP Topic V-5, RCPB Leakage Detection as follows:

The staff review of this topic indicates that Big Rock Point satisfies current criteria with the exception of seismic requirements. The licensee's Technical Review Group has concluded that the emergency operating procedures will be revised to require a leak test in the event of a confirmed seismic event. Further, if the leak detection equipment is inoperable (after the event), Big Rock Point Plant would be shut down (limiting condition for operation) until such time that the equipment can be returned to service. The licensee has committed to complete these changes by the end of June 1984. The staff finds this commitment to be an acceptable resolution.

CPCo Resolution

The BRP Operating Procedures - Emergency for Earthquake, have been revised and currently require the following "Subsequent Operator Actions" in the event earth vibration or movement is felt at the plant site:

Determine primary system leak rate per procedure. If the primary system leak rate cannot be determined because of leak detection equipment being inoperable, the Plant shall be brought to the hot shutdown condition within 12 hours, and to the cold shutdown condition within the following 24 hours.

5.2.5.7 Generic Letter 84-11 - Inspections of BWR Stainless Steel Piping (Leak Detection) (Reference 8)

Reference 9 provided the CPCo response for the Generic Letter. An NRC inspection of the implementation of the actions set forth in Generic Letter 84-11 dated September 8, 1987 included the following comments:

Leak Detection and Leakage Limits

The Big Rock Point (BRP) Technical Specifications for reactor coolant leakage surveillance requirements provide for sump level monitoring every 24 hours. The guidelines provided in Generic Letter 84-11 however, require that sump level be monitored at 4 hour intervals. The BRP unidentified leakage limits require a plant shutdown if the coolant system leakage exceeds 1 gpm. This is considered to be more restrictive than Generic Letter 84-11 requirements and is considered acceptable.

5.2.6 THERMAL STRESSES IN PIPING CONNECTED TO REACTOR COOLANT SYSTEMS

CPCo, by letter dated September 26, 1988 in response to Nuclear Regulatory Commission Bulletin 88-08, Thermal Stresses in Piping Connected to Reactor Coolant Systems, dated June 22, 1988, and Supplements 1 and 2 to the bulletin, required Consumers Power Company to review unisolable piping connected to the Reactor Coolant System to identify where temperature distributions could result in unacceptable stresses and to take action where such piping is identified. The bulletin also required written confirmation that the actions have been completed. A description of the results of the review are provided below:

A review of systems connected to the Reactor Coolant System (Primary Coolant System - PCS) at Big Rock Point, was performed. Concerns identified by the bulletin were considered during the system evaluation. Based on industry experience, the Nuclear Regulatory Commission identified the potential for thermal fatigue in unisolable stagnant piping connected to the PCS. Specifically, undesirable stresses resulted when water, which was significantly cooler and at higher pressure than the primary system, leaked through normally closed valves into a stagnant portion of the PCS. The subsequent temperature stratification produced thermal stress cracking in the immediate area.

For the purpose of this evaluation the area of interest was considered to be all primary system ASME Class 1 piping. Interfacing systems within this boundary were reviewed. Vents and drains as well as passive piping, such as instrument and sample lines, were considered not applicable and were excluded from the evaluation. Three systems were determined to fall within the criteria of the stated concern. These systems and their potential to initiate thermal stresses as described in the bulletin are addressed below.

- Feedwater System (FWS)

This system is an extension of the PCS. Its design and operation considers the injection of cooler, high pressure water into the primary system. The results of our evaluation has determined that temperature stratification will not occur at the system interface, therefore, the Feedwater System is not subject to the concerns stated in the bulletin.

- Liquid Poison System (LPS)

This system is designed to inject a sodium pentaborate solution into the PCS. The solution is maintained at a higher pressure and cooler temperature than the PCS. Leakage into the PCS is not probable by design. System inlet and outlet are provided with positive acting squib valves which preclude inadvertent leakage. Additionally, the inlet is supplied by a check valve to prevent reverse flow, while the outlet includes a control valve capable of isolating LPS injection. The LPS piping is designed to operate at the same pressures and temperatures as the primary system. Any leakage of solution into the primary system would be readily detectable as a result of the negative reactivity effects of the sodium pentaborate as well as various alarms associated with the system. The results of our evaluation has determined that temperature stratification will not occur at the system interface, therefore, the Liquid Poison System is not subject to the concerns stated in the bulletin.

- Control Rod Drive System (CRD)

Various CRD system flow paths operate at higher than reactor pressure. The two flow paths of interest are the cooling path through the CRD mechanisms to the bottom of the reactor and the return line to the PCS utilized during control rod manipulation. The cooling line is adjusted to reactor pressure plus 30 psi. It provides 0.1 gpm to 0.5 gpm of cooling water to the control rod drive to prevent temperatures from exceeding 250°F. The cooling water passes through the drive mechanisms into the bottom of the reactor vessel. The reduction of thermal stress established by this flow is an inherent design of both the system and reactor vessel. In addition, the CRD mechanism is capable of being isolated.

The return line provides a path for displaced hydraulic water during the normal manipulation of a control rod drive. Its pressure is adjusted to reactor pressure plus 200 psi. It discharges into the return line of the Reactor Cleanup System. A thermal sleeve is installed at this branch connection with the cleanup system because of previous concerns similar to those raised by the bulletin. This design consideration precludes establishing the conditions needed to induce cracking. In addition, this piping is capable of being isolated by various valves. The results of our evaluation has determined that temperature stratification will not occur at the system interface, therefore, the Control Rod Drive System is not subject to the concerns stated in the bulletin.

Conclusion

In conclusion, it is determined that for all systems evaluated, either thermally induced stress was a design consideration, or the conditions needed to initiate cracking as stated in NRC Bulletin 88-08 does not exist. Subsequently, and in accordance with the direction provided by the bulletin, no additional action is required.

5.3 REACTOR VESSEL

5.3.1 GENERAL DESCRIPTION

The reactor vessel was procured and designed for research and development as well as electric power production/generation.

Combustion Engineering Incorporated designed, fabricated, and tested the vessel in accordance with the requirements of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section 1 - 1959 Edition utilizing Code Cases 1270N, 1271N, and 1273N. ASME Code Section VIII was used for the vessel internal cladding weld overlay. The design was in accordance with General Electric Company Specification DP-19881, and in cases where the ASME Code was not applicable, the design was evaluated from Navy Bureau of Ships Publication, "Tentative Structural Design Basis for Reactor Pressure Vessels and Associated Components, PB-15987, 1 April, 1959."

The general arrangement of the vessel and internal elements are provided in the following Figures:

Figure 5.1 Core Configuration (Section Through Fuel)

Figure 5.2 Reactor Vessel Schematic

Figure 5.3 Reactor Vessel and Internals

FIGURE 5.1
CORE CONFIGURATION (SECTION THROUGH FUEL)

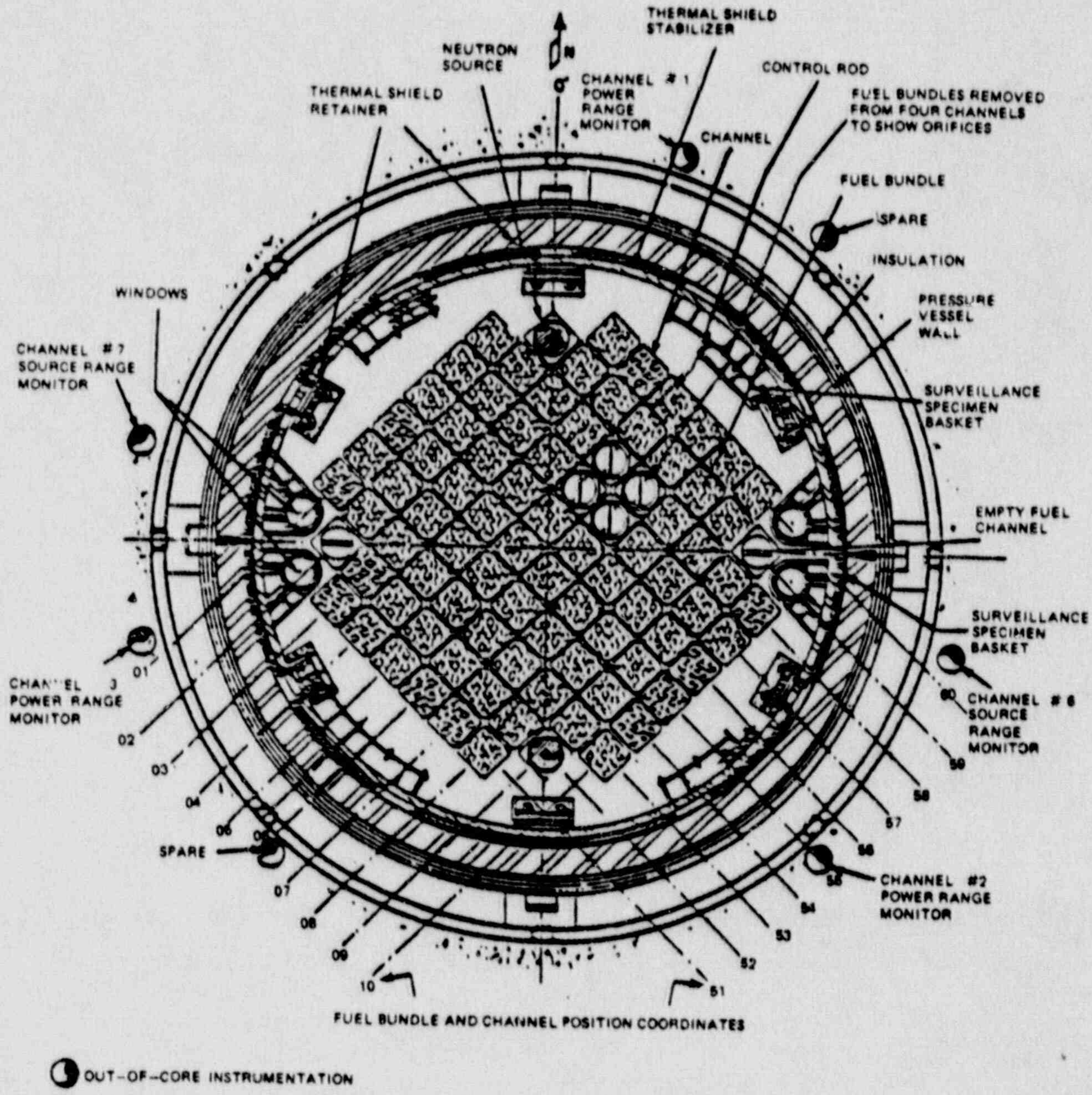

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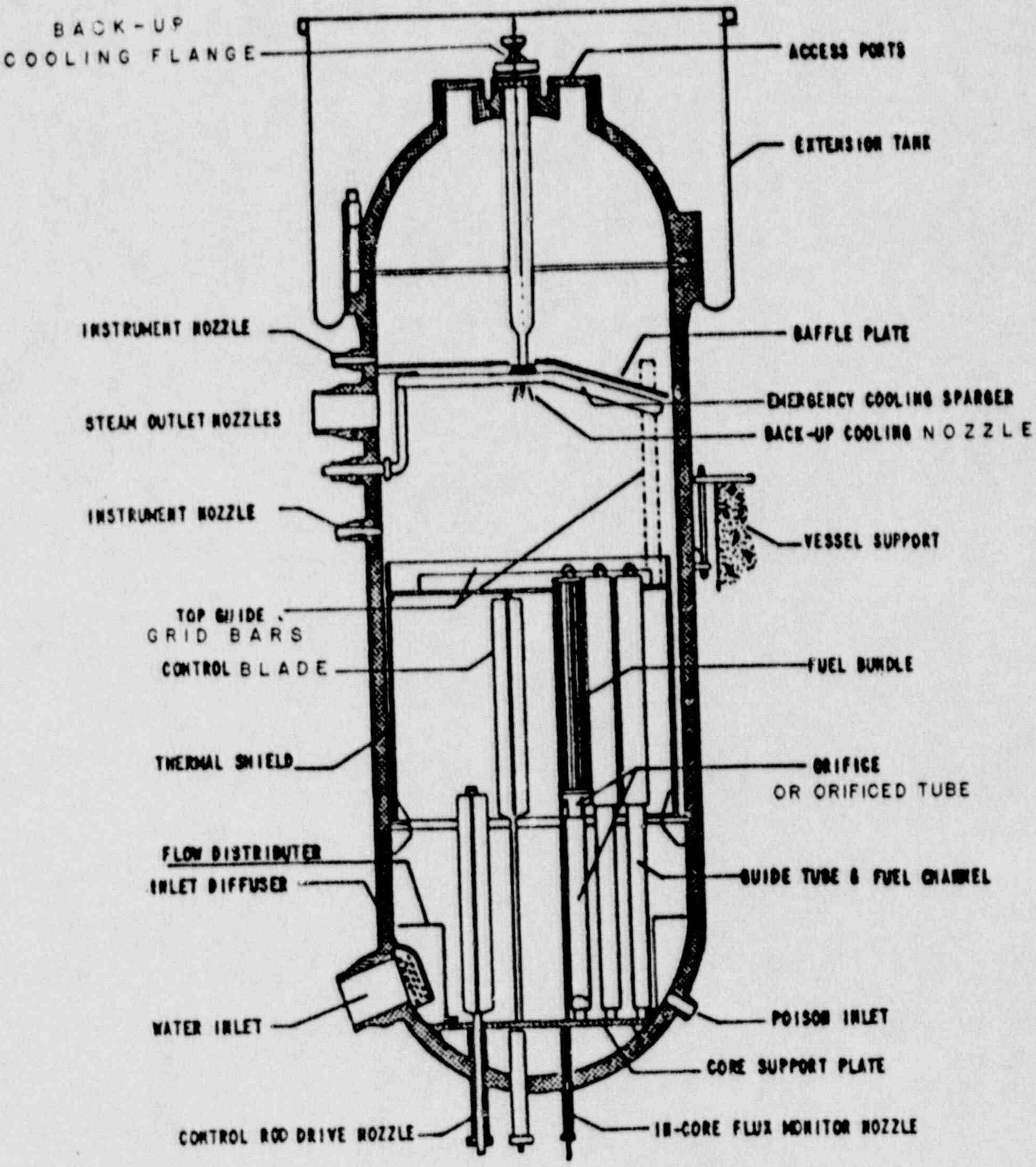
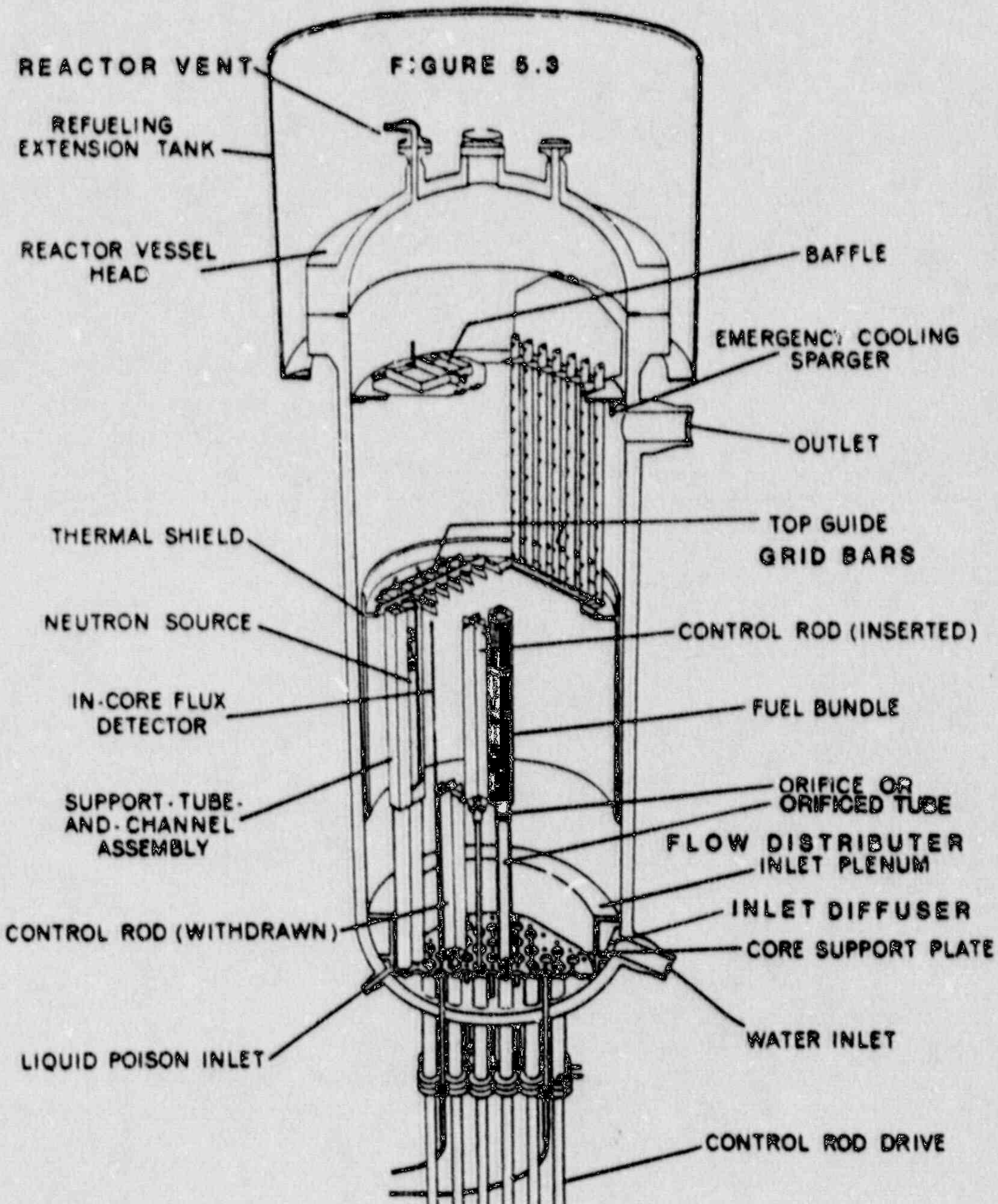
FIGURE 5.2
REACTOR VESSEL SCHEMATIC



Reactor Vessel Schematic

FIGURE 5.3
REACTOR VESSEL AND INTERNALS



REACTOR VESSEL & INTERNALS

5.3.1 GENERAL DESCRIPTION (contd)

The vessel is a vertically mounted cylindrical unit with a hemispherical lower head integral with the vessel shell plates and a removable upper closure head.

Table 5.2 Reactor Vessel General Characteristics

Length, overall	30 ft.
Inside diameter	106 in.
Wall thickness (excluding clad min)	5 1/4 in.
Cladding thickness (min)	5/32 in.
Removable head wt	28 tons
Total vessel weight, including head but excluding H_2O or internals	120 tons
Design pressure	1700 psig
Design temperature (on I.D. of vessel)	650°F
Maximum operating pressure	1485 psig
Nominal operating pressure	1335 psig
Nominal operating temperature	582°F
Approximate Initial Nil Ductility Transition Temperature	10°F
Initial Hydrostatic Test Pressure	2550 psig[1]

Notes: 1. After fabrication and prior to shipment. (The current hydrostatic test pressure/temperature limits are included in the Technical Specifications.)

5.3.1.1 Materials of Construction

Table 5.3 Reactor Vessel Principal Materials of Construction

Component	Material	Specification
Vessel Shell and Heads	Steel	ASTM SA-302 Grade B
Flanges and Nozzles	Steel Forging	ASTM SA-336 (Code Case 1236-1)
Cladding	Stainless Steel	Types 308 and 309 Weld Rod
Head Studs	Steel	ASTM A193, AISI 4340
Head Nuts and Washers	Steel	ASTM SA-194, AISI 4340

Other Material

Stainless steel type 304 meeting various ASTM specifications, depending on whether it is plate, pipe, tubing or forgings, is used for flange covers, nozzle extensions, drive housings and other parts coming in contact with the reactor coolant.

The flange faces in the area of the "O" ring seating surfaces are weld clad with Inconel A to provide a slightly harder surface than can be obtained with type 304 stainless steel.

The Primary Core Spray Sparger and Steam Baffle Plate were replaced with ASTM 316 Stainless Steel with a maximum carbon content of 0.02%. Refer to General Electric Design Report NEDC-21974, November 1978 submitted to the NRC November 21, 1978.

The reactor vessel cylindrical shell is made up from shell plate courses. Each shell course is made from two plates joined together by two longitudinal welds. The two shell courses are joined by a circumferential weld. The upper shell course is joined to a flange with another circumferential weld. The plate material was purchased from Lukens Steel. For information concerning Plate Heat Number, chemistry, Charpy Tests, Heat Treatment, Tensile Properties, and welding, refer to CPCo June 12, 1978 letter (Reference 13) and USNRC NUREG-0569, "Evaluation of the Integrity of SEP Reactor Vessels," published December 1979. Additional information on reactor vessel weld material was submitted to the NRC by letter dated April 26, 1979 by General Electric Company. This information was docketed by CPCo letter dated July 31, 1979 in response to IE Bulletin 78-012B.

5.3.1.2 Vessel Closure

The top closure head is a bolted flange with 106-inch inside diameter, bolted in place with 42 4 3/4 inch diameter studs. The studs have a minimum yield strength of 120,000 psi and are stressed to tensions with a maximum permissible range of 10% maximum to minimum variation of each other by using hydraulic stud tensioners which eliminate all torsional stress in the studs. The stress level in the studs is below the allowable 1/3 of yield stress at operating temperature. The outer seal used is a self-energizing stainless steel, silver plated and polished "O" ring backed up by a silverplated and polished "O" ring of 718 alloy, (Reference Specification Change SC-88-034. A leak off between seals is used for detection of inner seal leaks. The closure flanges of the reactor vessel were designed in accordance with the ASME Boiler Code Section VIII, Paragraph UA 45-55. A closure study was made to determine the ability of the design to withstand steady state and cyclic loading conditions. The closure flanges were analyzed to determine stresses resulting from internal pressure, bolt loads, and thermal loads occurring during 100° F/hr start-up and 300° F/hr cool-down. Thermal distribution in the flanges was determined by a finite difference relaxation method. (Reference 1)

5.3.1.3 Reactor Vessel Radiation and Stress Basis

An outline of the stress basis (Reference 1) used in determining the maximum combined pressure membrane and thermal stresses for 100% power level in accordance with ASME Code Case 1273N indicates the highest thermal stress will occur at the elevation of the core center line; therefore this location was analyzed. There are no nozzles or other attachments in the gamma energy flux region of significant magnitude. The peak thermal stress value is 2600 psi and when combined with pressure membrane amounts to a maximum of 20,800 psi. The peak thermal stress value is based on 2.94×10^4 Btu hr $^{-1}$ ft $^{-3}$ at the inside surface of the vessel.

The original stress effects basis as reported in the 1961 FHSR are provided below. Analysis and evaluation of current radiation effects and pressure-temperature limits are discussed in Sections 3.9.4, 5.3.2 and 5.3.3 of this Updated FHSR.

Stress Effects (Original Design Bases)

The "as-fabricated" reactor vessel material has the following mechanical properties:

Yield Strength, psi	56,900
Tensile Strength, psi	84,200
Elongation in 2", %	23

At Big Rock Point reactor's design pressure and temperature, the pressure produced reactor vessel membrane stress is about 38% of the unirradiated yield point stress. Calculations indicate that the temperature difference across the reactor vessel wall at the core midplane at 240 Mwt power will be about 14°F. This temperature distribution would produce tangential and longitudinal stresses on the outside face of about 2600 psi tension. The ASME Code allowable steady-state thermal stress is 10,000 psi. Combining this thermal stress with the pressure produced membrane stress at design pressure produces a stress of only about 44% of the unirradiated yield point stress at design temperature.

During startup, the transient thermal stresses are basically compression on the inner face and tension on the outer face, and tend to be neutralized by the gamma heating stresses. During shutdown, the transient thermal stresses are reversed, but the gamma heating is 6% or less of the gamma heating at normal full power so the additive stress combination is absent.

To minimize the thermal stresses in the reactor vessel walls, the rate of change of power is adjusted during startup or normal shutdown to maintain a temperature rate change not in excess of 100°F per hour in the reactor vessel walls.

Conclusion

Based upon the results of Sections 3.9.4 - "Reactor Vessel Material Surveillance Program"; 5.3.2 - "Pressure Temperature Limits"; and 5.3.3 - "Reactor Vessel Integrity," of this Updated FHSR, the original belief that the reactor vessel is adequately designed against the possibility of brittle failure during its 40 year operation is reinforced by further knowledge and experience.

5.3.1.4 Reactor Vessel Nozzles

Table 5.4 Reactor Vessel Penetration Nozzles

Nozzles	Number	Diameter, Inches	Location
Coolant Water Inlets	2	20	Bottom
Steam-Water Mixture Outlets	6	14	Shell
Outlet to Shutdown Heat Exchanger	1	6	Shell
Access Ports in Top Head (Center Port-Backup Core Spray)	3	10	Head
Control Rod Drive Penetrations	32	4	Bottom
Liquid Poison Inlet	1	3	Bottom
Emergency Core Spray Inlet	1	3	Shell
Vessel Vents (only one is piped)	2	3	Head
In-Core Flux Monitor Penetrations	8	2	Bottom
Instrument Nozzles	4	3	Shell
Seal Leak Monitor	1	1/2	Head Flange

5.3.1.5 Design Cycles (Reference 1)

Design cycles allowed under a normal start-up and fast shutdown are provided in Table 5.5. Calculations for fast start-up have not been made since the rate of start-up will be limited to 100°F/hr. We did not specify the minimum number of cycles which the vessel would be capable of withstanding. Each nozzle was analyzed independently and the results from these analyses are given in Table 5.4. The high number of emergency cycles allowable are far more than anticipated during the life of the vessel.

5.3.1.5 Design Cycles (contd)

Table 5.5 Reactor Vessel Cyclic Stresses (Reference 1)

Location	Transient	S_{alt}[1] (ksi)	S_m[2] (ksi)	N[3] (10^3)
Closure Flange	End of 100°F/Hr Start-Up	37.6	19.7	8.4
Seal Skirt	to End of 300°F/Hr	35.4	.6	16.0
Outlet Nozzle	Shutdown	21.0	8.9	∞
Inlet Nozzle		29.3	0	72.0
Poison Nozzle		54.3	21.5	22.4
Control Rod Drive Nozzle		42.5	42.5	8.0

Notes: 1. S_{alt} = Alternating Stress

2. S_m = Mean Stress

3. N = Allowable Number of Cycles (0.8 Usage Factor Included)

The limiting case is a control rod drive nozzle, a stress analysis for this nozzle was accomplished (Reference 1) and the calculation results verified the structural integrity of the control rod drive mechanism housing design.

Reactor Vessel Thermal Sleeves (Reference 1)

All thirty-two control rod nozzles contain a thermal sleeve. The three-inch poison inlet nozzle and the three-inch emergency cooling nozzle also contain thermal sleeves. These thermal sleeves were taken into account in the stress analysis.

5.3.1.6 Reactor Vessel Cladding (Reference 14)

The internal surfaces which are in contact with the coolant are clad with a minimum of 5/32 inches of corrosion resistant 304 stainless steel 0.08% maximum carbon at the surface, and have a finish of 250 microinches or better.

5.3.1.7 <u>External Reactor Vessel Supports, Brackets, and Insulation</u>

The vessel is supported by 12 brackets attached to the exterior vessel shell. Twenty-four 2.5 inch diameter hanger rods attached to these brackets will transmit the vessel weight to supports anchored in the supporting concrete. Individual supports will carry two or four hanger rods. There are eight stabilizing brackets attached to the exterior vessel shell. Four are located near the vessel bottom head, and four are located near the vessel support brackets. The stabilizing brackets allow expansion of the vessel, both in the radial and longitudinal direction, but prevent movement of the vessel central axis. These vessel supports are designed to adequately withstand the horizontal loadings on the reactor vessel from the reactive forces as may be developed with the rupture of any single line to the vessel. Thermal expansion of the vessel and connected piping is accommodated by these supports and by flexibility of the piping systems.

<u>Other Brackets</u>

Insulation fasteners and thermocouple attachments are provided on the exterior of the vessel to secure the stainless steel insulation and thermocouples.

<u>Reactor Insulation</u>

The reactor insulation is an all metallic insulation three inches thick. It is attached to the vessel by banding and is supported by small brackets welded to the vessel prior to final stress relief. The complete outside surface of the vessel is covered except for the drive housings located on the bottom head.

5.3.1.8 <u>Internal Reactor Vessel Supports</u>

<u>Internal Supports</u> (Refer to Figure 5.1)

Four sets of brackets are welded to the vessel internal wall to provide supports for the reactor internals. These are core supports, thermal shield supports, baffle supports, and inlet diffuser brackets. The core support brackets are located on the bottom head and consist of four plates welded to the base metal and are constructed of stainless steel type 304. These four brackets align and support the core support plate.

The six thermal shield brackets are located below the active fuel section of the core. They are welded to the vessel base metal and are made of 304 stainless steel. These brackets support and align the vessel thermal shield.

Just above the 14" steam-water mixture outlets are eight brackets welded directly to the vessel cladding. These are to support the steam baffle and the emergency core spray sparger.

The only other attachments on the vessel interior are brackets which hold the inlet diffusers in place in the bottom head. These also are welded directly to the vessel cladding.

5.3.1.9 Reactor Vessel Internals

Three major assemblies - the core support, thermal shield and top guide - provide support and alignment of fuel and guidance of control rods. These and other vessel internals are shown in Figure 5.1.

5.3.1.9.1 Core Support Plate

The core support, a circular plate 1 1/2" thick, rests near the bottom of the reactor vessel on four support pads which are welded to the vessel. It has four alignment brackets which engage the four support pads to accurately position the plate relative to the control rod drive penetrations in the vessel bottom head. A minimum radial clearance of 1/8" exists where each control rod drive nozzle penetrates the plate. Eighty-eight support tube adapters, bolted to the core support plate, accurately align the base of the support-tube-and-channel assemblies relative to the control rod penetrations. Large, evenly spaced holes through the plate permit coolant flow from beneath. The core support plate supports the support (guide) tubes, fuel channels, fuel assemblies, and flow distributor. The following provides the General Core Composition and Total Weight Supported by the Core Support Plate:

General Core Composition

a. Enrichment of Fuel - Approximate weight percent (w/o) U-235 from 2.6 to 5.2, inclusive. Approximate weight percent of plutonium (fissile Pu-239 and Pu-241) 1.0 to 10 in normal (0.7 w/o U-235) UO_2.

b. General Core Data

Number of Fuel Bundles in Core, A Maximum of	86
Total Nominal Weight of UO_2 Plus UO_2PuO_2 in 84 Bundles, Lb	29,300
Moderator to Fuel Volume Ratio	2.65
Equivalent Core Diameter (Approximate), Inches	77

c. Channels

Number of 304 SS and/or Zr-2	88
Wall Thickness, Inches:	
304 SS	0.075
Zr-2	0.100
Inside Width, Inches:	
304 SS	6.57
Zr-2	6.54
Length, Inches:	
304 SS	79-5/8
Zr-2	79-3/4

• Total Weight Supported by Core Support Plate

84 Fuel Bundles @ Approximately 440 Lb/Bundle	36,960 Lb
A Total of 88 Support Tube and Channel Assemblies Consisting of:	
1. Up to 86 Support Tube and Channel Assemblies With Orifice Bucket & 110 Lb/Assembly, or	9,460 Lb
2. Up to 86 Support Tube and Channel Assemblies With Modified Transition and Orifice Insert @ 107 Lb/Assembly, and	9,202 Lb
3. Two Support Tube and Channel Assemblies With Channel Plugs @ 110 Lb	220 Lb
One Flow Distributor Assembly	2,500 Lb
Total Weight	48,882-49,140 Lb

Dimensional drawings of the type 304 stainless steel core support plate were provided to the NRC in (Reference 1), as were the original stress calculations. The Horizontal Earthquake shock of 5% of total (core assembly + thermal shield) gravity load was re-analyzed to the 0.12g Safe Shutdown Earthquake. Refer to Chapter 3 Table 3.1 of this Updated FHSR.

The core support plate was modified as described in the CPCo letter dated October 9, 1963, "Report of Changes, Tests, and Experiments" (Reference 15) and in the CPCo letter dated July 22, 1963, Special Report - "Modifications to Core Support System (Reference 16).

- The 32 holes for control rod drive housings were fitted with sleeves to reduce the funnel effect which permitted foreign material to enter the conical drive thimbles.

- A new flow distributor was added to permit even distribution of flow and supplement the two original flow diffusers. The distributor reduced the flow induced vibrations of the support (guide) - tube-and-channel assemblies.

- A more positive locking and keeper system for the cap screws at the transition piece that mates the support tube with its channel was provided.

5.3.1.9.2 Support Tube and Channel Assemblies

The support-tube-and-channel assemblies support the fuel bundles, guide the control rods, and provide a means for controlling the flow of coolant through the fuel. Basically, each assembly consists of a round cross-section support tube and a square cross-section channel joined by either a bolted flange or transition piece.

- The bolted flange channel assemblies contain an orifice assembly (orifice bucket) attached to the support tube at the elevation of the flange and is locked into place by turning. Positive protection against unlocking is provided by interference between the feet of the inserted fuel bundle and the orifice handle. Two spring latches on the orifice assembly also act to prevent inadvertent unlocking prior to fuel loading.

- The modified transition piece channel assemblies contain a fixed orifice insert within the support tube at an approximate elevation of 48 inches below the lower tie plate. Information on the modified design was submitted by CPCo letter dated January 21, 1972.

- Empty support tube and channel assemblies are fitted with channel plugs and stiffeners when operating with less than a full core configuration, (Reference 15).

- Two bolted flange channel assemblies contain the neutron source cylinders which rest on a special orifice in the standard support-tube-and-channel assembly.

The wall thickness of the zircaloy channels is 0.100 inch, this thickness being required to withstand the possible pressure differential between the inside and outside of the channels. The wall thickness of the stiffer stainless steel channel is 0.075 inch.

The flow orifices provide flexibility for meeting various operating conditions and allow flow through channels to be varied across the core.

The fuel channels surround each fuel bundle, thus positioning each bundle, provide guide surfaces for the control rods and make possible flow orificing. The clearance between the fuel channels and the upper guide grid limits the between-channel flow to the required amount.

The fuel channels are individually attached to and supported by the support tubes by a flange-type bolted joint or a transition piece, and prevented from lifting out of the core by the weight of the channel-support-tube assembly, and by the beams of the upper guide grid.

5.3.1.9.3 Flow Distributor (Reference 15 and 16)

The two original inlet flow diffusers were supplemented with a stainless steel flow distributor when vibration measurements confirmed that flow induced vibration of the support-tube-and-channel assemblies was occurring. Testing, subsequent to installation of the flow distributor, has proven that the distributor (which encompasses the support tubes) prevents direct impingement of the recirculation flow on the support tubes. The new two piece flow distributor is designed with a 360° vertical skirt approximately three feet high with an upper flange between the skirt and the vessel wall to prevent water flow up and over the distributor skirt. The flow distributor is mounted to the core support plate by eight bolts. Two cut off sections of the flange permit installation and these cut off sections are then bolted back onto the flange and the bolts safety-wired in place. Four inspection ports are provided in the 180° segment to enable inspection of the inlet diffuser fasteners.

5.3.1.9.4 Inlet Flow Diffusers

The two inlet nozzles, located immediately above the core support, are equipped with diffusers to impart lateral and downward velocity to the coolant, thus distributing the flow and sweeping the bottom head. The coolant flows both above and below the core support and enters the bottom of the support tubes through the support tube adapter. The adapter is designed to permit full flow up each channel and yet provide the support to the support tubes. The adapter is a machined casting with five support ribs connecting the inner adapter core to the outer adapter support ring. The space between ribs is clear for coolant flow therefore most of the area of the adapter is actually flow area. The water then flows up the support tube through the orifice and then through the fuel channel.

The inlet flow diffusers were modified in 1979 (Specification Field Change SFC 79-031). The flow diffusers were replaced with functionally identical components utilizing type 304 L and XM-19 plate and bracket material. Attachment holes in the reactor vessel diffuser support brackets (802-32) were enlarged to 25/32 inch diameter and new 3/4 inch XM-19 bolting material was installed. A new stress analysis for the changes verified the acceptability and is included in the design package.

5.3.1.9.5 Thermal Shield

The thermal shield is a 1 1/2 inch thick stainless steel cylinder located in the annular region between the reactor core and the reactor vessel. Functionally it separates the inlet plenum (flow distributor) and outlet plenum (steam baffle) of the reactor vessel. The thermal shield is aligned and supported by six equally spaced brackets welded to the reactor vessel base metal which are located below the active fuel section of the core. The thermal shield supports the top guide (grid bars). The thermal shield was modified in 1965, details on the modifications were submitted to the NRC by letters dated June 25, 1965 and December 24, 1965 (Reference 17 and 18). Changes to the thermal shield included (a) installation of six "stilts" to replace the original support brackets; (b) installation of twelve "stabilizer" weights in the annulus between the thermal shield and the reactor vessel to provide additional dampening to the thermal shield; (c) installation of a new segmented bypass-leakage seal at the top of the thermal shield; and (d) installation of six locking wedges to open and disable the lower seal ring.

The "stilt-type" support arrangement allows a flexible movement in the radial direction to provide for thermal expansion but deters lateral movement.

Drilled holes at the bottom of the thermal shield operate in conjunction with the disabled bottom seal and the segmented top bypass-leakage seal to prevent bulk boiling between the thermal shield and reactor vessel. The number and size of the holes were calculated on the basis of the 240 Mwt core. The weights serve to immobilize the original seal ring, the wedges provided an effective gap at the bottom seal of approximately 0.2 inch. The wedges assure that the pressure drop is controlled by the new top seal. General Electric Company report by Atomic Power Equipment Department - APED-5178, "BRP Nuclear Plant Thermal Shield Instability," May 1966, described the redesign and test program along with analog computer results of leakage gaps on shield stability.

5.3.1.9.6 Top Guide (Grid Bars)

The top of each fuel channel is held in alignment by the top guide (grid bars) assembly. The top guide consists of stainless steel beams, hinged on one end, and arranged so that they form a checker-board array.

A continuous ring and intermittent brackets near the top of the thermal shield provide a mounting surface for the top guide, which is aligned with the shield by two pins and is bolted in place. The top guide provides radial support and upper alignment for the support-tube-and-channel assemblies, and by virtue of the clearance between the channels and the top guide beams controls the leakage flow which bypasses the fuel bundles. When the top guide is in the assembled position, offsets on its beams prevent vertical movement of the support-tube-and-channel assemblies. The top guide beams are locked in the assembled position by a main beam assembly which can be released, allowing the beams to be individually lifted to a vertical position in order to free the support-tube-and-channel assemblies for removal.

The top grid bars, of the top guide assembly were modified by welding small buttons along the sides to provide better guidance to fuel bundles being inserted and removed. This modification prevented the edges of the fuel bundle from catching on the top of the fuel channel (Reference 15).

The top grid support bolts which hold the top grid to the thermal shield were replaced with inconel bolts as reported to the NRC October 24, 1966, and the 36 bolts on the hinge side were replaced with 304 stainless steel annealed and acid cleaned bolts. Refer to November 30, 1966 Atomic Energy Commission Inspection Report.

5.3.1.9.7 Steam Baffle Assembly

The steam baffle, located and supported by eight brackets above the steam outlet nozzles, prevents steam slugging in the risers between the reactor vessel and the steam drum. It is composed of four movable door segments hinged to a circular baffle support ring. For core reloading operations, the doors are opened and latched in the open position. During normal operations, the doors are latched shut with a latching mechanism. The back-up core spray nozzle extends through a square opening in the baffle center.

The latching mechanism was repaired and modified to prevent galling of components. Refer to Specification Field Change SFC 79-037.

The steam baffle (doors and support ring) assembly was replaced in 1979 and differed from the previous design only in material type which was changed to type 316 stainless steel. Information on this change was submitted to the NRC March 28, 1979 and was approved by Technical Specification Amendment No. 26, April 10, 1979. Details of the change are provided in Facility Change FC-464.

5.3.1.9.8 Emergency Core Cooling

There are two components within the reactor vessel providing emergency core cooling capability:

1. Primary Core Spray Sparger Assembly

   The primary core spray sparger ring is located below and is supported by the steam baffle assembly. The sparger is a two-inch schedule-40 pipe arranged into an octagonal ring with 36 attached spray nozzles aimed at the core. A portion of the sparger is canted downward at approximately a 19° angle to conform with the steam baffle contour. The original sparger was replaced in 1979 with a design consisting of type 316 versus type 304 stainless steel. The new sparger nozzle design permits precision nozzle aiming during fabrication and allowed for an optical aiming device to be utilized for verification of each nozzles orientation prior to installation. Once the sparger was installed, spot checks (eg, not every nozzle) of selected nozzle aiming points was checked.

The sparger water inlet pipe was redesigned to facilitate remote, underwater installation and the "J-bolts" which hold the sparger to the steam baffle were redesigned to facilitate remote assembly.

For additional details on the primary core spray sparger, design, fabrication, testing, and analysis, refer to CPCo January 16, 1979 submittal with corrections April 6, 1979. Further, refer to Technical Specification Amendment No. 26 dated April 10, 1979. The design changes were accomplished via Facility Change FC-464.

2. Back-Up Core Spray Nozzle Assembly

A back-up means of providing core spray was added in 1970 and information on the installation was provided in CPCo Technical Specification Change Request dated February 2, 1971. The original single nozzle provided emergency core cooling in the event the primary core spray sparger system was disabled. This single nozzle design was modified and replaced by a multi-nozzle assembly with 12 spray nozzles via Facility Change FC-444. This nozzle assembly is supported from the center access port in the reactor vessel head by a bolted flange. The multi-nozzles extend below the steam baffle assembly in the reactor vessel centerline. This design change was evaluated by the NRC as part of the October 17, 1977 Technical Specification Amendment No. 15.

5.3.1.9.9 Neutron Sources

The type, quantity, location, and physical description of reactor neutron sources are as follows and are limited by Technical Specifications:

Type	Antimony Beryllium
Quantity	Up to Two Initial (start-up) Sources Up to Four Auxiliary Sources

Neutron sources may be provided to assure neutron visibility is sufficient to satisfy the requirements of the Technical Specifications. If neutron sources are used to assist in providing this visibility, location of these sources shall be as follows:

Location

The initial (start-up) neutron sources are placed in core positions 02-59 and 09-02 in vacant fuel channels at the core periphery.

Up to four auxiliary neutron sources may be contained within fuel bundles in rod locations normally occupied by fuel rods or inert rods.

Physical Description

The initial (start-up) neutron sources consist of a steel-jacketed antimony pin, 1-inch diameter by 12 inches long, centrally located on the vertical axis of a steel-jacketed (Type 304 SS) beryllium cylinder 5 1/2 OD by 16 inches long. The entire assembly, including support structure, is a cylinder 79 7/16 inches long by 6 inches diameter which rests on a special orifice in a standard support-tube-and-channel assembly. A lifting bail is provided for handling purposes. The assembly design allows adequate cooling along the surface of the source pin and the outer surface of the assembly.

The auxiliary neutron sources each consist of a homogeneous 50-50 mixture of antimony-beryllium first encapsulated in a steel tube (Type 304L SS), then secondarily encapsulated in a zirconium alloy tube.

Initial (Start-Up) Neutron Source Design

The two initial (start-up) neutron sources having 1660 curies total minimum strength design details were originally submitted in 1962, (Reference 1). By letter dated March 26, 1974, CPCo addressed the replacement of the original design start-up neutron sources with sources of essentially the same design. The design changes were depicted on drawings attached to the letter. The change consisted of modifying the gamma source hold-down device to provide a means of irradiating a new antimony pin while utilizing the original neutron sources. This will enable changeover from the original neutron sources to the new neutron sources.

Auxiliary Neutron Source Design

Two additional auxiliary neutron sources contained in fuel bundles were approved for use by the Atomic Energy Commission (AEC) by Change Number 23 to the Technical Specifications, dated February 22, 1971 based upon CPCo Proposed Change dated January 18, 1971. The auxiliary neutron sources were proposed in order to improve the start-up count rate and improve the ability to measure (fission) neutron count rate. The design of these additional auxiliary neutron sources was also provided in the proposed change.

A design change of these sources was requested February 2, 1973 and approved by the AEC March 2, 1973 as Technical Specification Change No. 35. The change was necessary to make the sources compatible with new 11 x 11 rod array fuel bundles and to prevent secondary encapsulation weld failure. This change also allowed two new auxiliary sources (in addition to the two additional auxiliary sources allowed by Change No 23 above) for a total of four auxiliary neutron sources to be placed in the core until the activity of the new sources builds up to a useful level (1 to 3 years) at which time the two original auxiliary neutron sources will be removed. The location of these four sources was limited by the change also.

Thus, the Technical Specification basis for up to four auxiliary sources allows for activation of new sources and the number of auxiliary sources will normally be two.

A Technical Specification Change Request was submitted September 25, 1980 to remove the restrictions on fuel bundle auxiliary neutron source rod location and to allow additional fuel management flexibility with respect to future location of auxiliary neutron sources. This request addressed a change of auxiliary source rods in that replacement source rods will not be "removable" but will require bundle disassembly in order to move them from one bundle to another. This change request was approved January 12, 1981 via Amendment Number 36.

The amendment (1) removed the restrictions on allowable locations for auxiliary neutron sources, and (2) modifies the physical description of the auxiliary neutron sources to allow the source to be placed in the center of a fuel assembly rather than in the corner location previously used.

CPCo by letter dated January 18, 1971 provided information which indicates that the startup channels will respond to fission neutrons rather than source neutrons even when the auxiliary neutron sources are placed very close to the start-up detectors. This indicates that the restriction on the location of the fuel bundles with auxiliary neutron sources is unnecessary.

In terms of changing the source location to the center of the fuel assembly, the licensee has performed analyses to demonstrate that there will be no reduction in safety margin associated with the thermal hydraulic, fuel design limits (minimum critical heat flux ratio) or the ECCS performance analyses (maximum average planer linear heat generation rate).

Auxiliary Neutron Source Design (Reference 19)

The neutron source material is a homogeneous mixture of 50-50, by volume, antimony-beryllium compacted to a minimum packing fraction of 80%. The source material is first encapsulated in a 0.374 inch OD steel tube (Type 304L SS) with a 0.028 inch wall thickness. The overall length of the source tube is 70.110 inches with the source material located in the middle 44.26 inches, held there by a hollow, steel tube spacer at each end. The remaining space in the source tube is void volume. The source tube is encapsulated in a zirconium alloy fuel tube of the same quality and dimensions as tubing used for fuel rods.

Design Life (Reference 19)

The in-reactor design life of the auxiliary neutron sources is 15 years. Sufficient void volume has been incorporated into the design to attain this objective. Based on an assumption of 1.5×10^{13} n/cm^2 -s for the flux of neutrons with energies greater than the 2.7 MeV threshold for the (n, 2n) and (n, alpha) reactions in beryllium, approximately 2.5×10^{22} He atoms would be generated in 15 years. Using 799.5°F as the temperature of the outer surface of the stainless steel capsule and assuming conservative conductivity values, the peak temperature in the source material would be 870°F. The internal capsule pressure developed, after 15 years of irradiation, would be 1127 psia. The minimum wall thickness of 0.027 inch exceeds the minimum thickness specified by the ASME Pressure Vessel Code for 304L SS stressed under the above conditions of pressure and temperature. (Rules of Construction of Pressure Vessels, Division 1, 1971 Edition, ASME Boiler and Pressure Vessel Code Section VIII and supplements through summer 1972.)

5.3.1.9.10 Control Rod Blade Assemblies

The 32 cruciform-shaped control rods are guided and provided lateral support by the "fuel channel and support (guide) tube" assemblies. Vertical support is provided by the control rod drive mechanisms. These rods move up and down between the fuel channels and support tubes and are the primary means of controlling reactor power. Each control rod blade assembly is approximately 11 1/2 inches wide and 5/16 inch thick.

The neutron absorbing material is solid hafnium (Hf) or Carbide (B_4C) Powder and have an effective poison length of approximately 68 inches.

Types of Control Rod Blade Assemblies

Type 1 Blades (Peripheral Positions)

The sixteen peripheral control blades contain one hundred and four 304 stainless steel tubes filled with B_4C powder.

Type 2 Blades (Interior Positions)

The sixteen interior control blades contain sixty four 304 stainless steel tubes filled with B_4C powder and forty 304 stainless steel empty tubes open at each end. Each wing of the cruciform blade contains ten empty tubes and the outer sixteen tubes are B_4C filled. These blades are referred to as type 2A.

Also present are blades of a newer design utilized in the sixteen interior positions. These blades are referred to as Hybrid Control Rods and are type 2. These control blades contain sixty four 348 stainless steel tubes filled with B_4C and forty 348 stainless steel empty tubes open on each end. Each wing of the cruciform blade contains ten empty tubes and the outer fourteen tubes are B_4C filled followed by two exterior tubes consisting of B_4C filled tubes (bottom 75% and solid hafnium (Hf) metal rodlets (top 25%). Approval for utilization of these Hybrid control blades was provided by Technical Specification Amendment Number 88 dated February 17, 1987. Use of hafnium and 348 stainless will provide longer blade life.

Sheath Material

All control rod blade assemblies are enclosed in a perforated 304 stainless steel sheath welded to a central tie rod.

Control Rod Blade Rollers and Pins

Each control rod contains a maximum of eight (8) rollers to a minimum of four (4) rollers of either a nominal 0.485 inch or 0.567 inch diameter. The bottom four (4) rollers, which can be eliminated, move in a minimum interfuel channel space of 0.628 inch.

After the loss of several bottom rollers, (described in the February 11, 1965 Technical Specification Change), a decision was made to remove the bottom four (4) rollers and/or to reduce the diameter of the rollers for new control rods. The function of the rollers on the control rod is to reduce the metal-to-metal contact between the control rod sheath and the support-tube-and-channel assemblies and thus minimize long term wear. A reduction in the diameter of these rollers has not increased the wear noticeably. Also, the operation of control rods with bottom rollers missing has not changed the wear pattern significantly and has had no adverse effect on scram time or normal operating characteristics of the control rods.

Technical Specification Amendment Number 6 dated July 18, 1974 allowed removal of all four bottom rollers on the peripheral (type 1) blades. When new type 1 blades were installed for cycle 18 core reload, the bottom rollers were removed via Specification Field Change 82-004.

The type 1 and 2A control blades utilize Haynes 25 Pins and Stellite 3 Rollers. The Hybrid type 2 control blades utilize PH13-8 Mo Pins and Inconel x 750 rollers.

Control Rod Blade Poison Tubes

Poison tubes are type 304 or 348 stainless steel tubes, with welded end plugs and with approximately 68" poison length of natural boron carbide powder or 51" boron carbide powder plus 17" Hafnium. The poison tubes also contain steel balls, crimped in position at regular intervals to compartmentalize the boron carbide and minimize the possible effects of densification or settling of the B_4C powder.

The poison tubes are contained in a structure composed of a central core and four sheaths which form the cruciform shape. This cruciform, along with a handle, and a connector which contains the coupling to the drive, make up the control rod. Holes are placed in the sheaths to allow coolant to flow by the poison tubes.

Control Rod Stress and Distortion Analysis

The probable limit to the life of the control rod is internal pressure build-up due to release of helium formed by B^{10} (n, α) Li^7 reaction.

The pressure build-up and stress in each individual poison tube of each control rod will depend on its integrated exposure.

In order to give an indication of minimum life expected for any individual control rod, hoop stress in the worst tube due to internal pressure has been calculated as a function of time based on the following assumptions:

a. Internal pressure is present due to 1500 ppm volatile content in the B_4C (assumed to be H_2O which subsequently dissociates completely to H_2 and O_2), and helium which is introduced during fabrication.

b. The control rod is inserted continuously in the highest flux region of the reactor (1.3 times average flux), being fully inserted for a fraction of each operating cycle and being gradually withdrawn at the end of each operating cycle. (The operating cycle is the time between reactivity additions - refueling or steel channel removal.)

c. Reactor is operating at .8 load factor.

d. Of the He atoms formed, 30% are released from the B_4C powder and contribute to the internal pressure within the poison tubes.

If a control rod is inserted in the highest flux region continuously as described above, the resultant life, or time for the hoop stress in the worst tube to reach 50,000 psi (90% of expected yield strength), is greater than 1 year.

The stress in the worst tube has also been calculated as a function of time for "normal operation." In "normal operation" all control rods are used to control excess reactivity for burn-up and fission product poisoning such that the worst control rod captures 1.3 times as many neutrons as the average control rod and the worst poison tube in the worst control rod captures 2.9 times as many neutrons as the average tube in that rod. Assumptions for helium release from B_4C, initial pressure in tubes, and plant load factor are as given above. Resultant control rod life if limited by internal pressure is greater than 10 years.

An analysis was made to determine whether temperature gradients could exist in the structure of the control rod sufficient to cause thermal distortions. It was calculated that even with the control rod bowed close to the fuel channel in the worst expected tolerance condition (1/16" gap between control rod and fuel channel along their full length) there was sufficient natural circulation flow (with local boiling) to keep all surfaces of the control rod at essentially uniform temperature.

5.3.1.9.11 Fuel Bundles

The fuel bundles used in the reactor core are described here only in general terms. Each bundle weighs about 440 pounds and has an active fuel length of about 70 inches. Present fuel bundles use 121 rods in an 11 by 11 array. The enrichment in each rod varies depending on the intended positions of the rod within the bundle. A normal core will contain 84 fuel bundles. For a detailed description refer to Chapter 4.

Fuel cladding will, in addition to 304 stainless steel and Incaloy-800, include Zircaloy 2, Inconel-600, and Zr-3Nb-1Sn.

The fuel (Sintered Pellets or Compressed Powder) are UO_2 or UO_2-PuO_2.

5.3.1.9.12 In-Core Flux Detector Assembly

The eight in-core flux monitoring detector assemblies are mounted through a nozzle and encasement, which penetrates through the bottom of the reactor vessel. The in-core flux detectors are encased in guide tubes located in eight radial positions located throughout the core and are used to evaluate, under varying power conditions, the predicted neutron flux profile throughout the reactor core. Each assembly consists of three individual fission chambers located at different elevations. Calibration tubes run inside the incore flux detector assemblies. The calibration flux wire system provides the flux level data for comparison with predicted reactor core conditions. The detector assemblies are ~19.5 feet long and are inserted from the top of the core and are supported by the incore flux monitor nozzles. The detector assemblies are guided by the channels within the reactor core. The detector element is a fission chamber consisting of a fissile coating on the cathode separated from the anode by a gas gap.

5.3.1.9.13 Neutron Window Assemblies

Four 304 Stainless Steel neutron windows are supported by the thermal shield within storage baskets located at the core periphery and positioned approximately 90° from the neutron sources and near the location of the source range channel monitors. The windows are 6 inch schedule 160 pipe with end caps and lifting handles. The windows were slightly modified from original design. Refer to Specification Field Change SFC 79-035.

5.3.1.10 Biological Shield Cooling and Reactor Shielding

A cooling jacket is provided at the inner face of the reactor shield structure. The coolant flowing through the jacket removes the major portion of heat lost by conduction and radiation from the reactor vessel and the heat generated within the shield due to energy absorption. The jacket is water cooled with a design inlet water temperature of 68°F; cooling water is supplied from the closed loop reactor cooling water system. The cooling water system is designed to remove 60,000 Btu per hour at this design inlet water temperature. In the event a leak should develop, it will be possible to convert to air as the cooling medium.

The cooling jacket is a carbon steel, annular tank divided into eight segments. It extends vertically from a point opposite the bottom of the reactor vessel to an elevation just below the reactor supports. There is a two inch annular water filled space between the inside and outside faces of the tank. Water enters the jacket at the bottom and leaves at the top.

The maximum expected temperature within the shielding is 110°F with temperature gradient of 13°F per foot within structural portions of the shielding. The maximum thermal gradient occurs within the inner 6 inches of the shield and is approximately 80°F per foot. Complete disintegration of the inner 6 inches of the concrete opposite the core can occur without affecting the structural elements.

Reactor shielding is ordinary concrete with a density of approximately 150 lb/ft^3. Thickness varies in plan and elevation to suit structural requirements. The shielding thickness directly opposite the core is approximately 9 feet, 6 inches. The control rod drive room, which is directly beneath the reactor, has ordinary concrete walls which are approximately 4 feet thick. A removable shield plug of a thickness 4 feet, 6 1/2 inches, consisting of 4 feet, 4 inches of concrete and 2 1/2 inches of lead, closes the opening above the top of the reactor.

5.3.2 REACTOR VESSEL PRESSURE-TEMPERATURE LIMITS

The Big Rock Point reactor pressure-temperature limits for Hydrostatic Test, Cooldown, and Heatup Conditions are included in License DPR-6, Docket No. 50-155, Appendix "A", Technical Specifications. These limits were based upon Amendment No. 66 dated April 12, 1984 as corrected September 24, 1984, in response to a CPCo request dated October 24, 1983. The CPCo request included an analysis and basis for the change to the reactor vessel pressure/temperature limits to account for accumulated neutron radiation dose to the vessel metal up to 18 Effective Full Power Years (EFPYs) which is approximately 1993. Based upon information provided in Section 3.9.4 of this Updated FHSR, "Reactor Vessel Material Surveillance Program," the two remaining surveillance capsules are not scheduled for removal and analysis until approximately 1995.

5.3.2.1 NRC Safety Evaluation (Reference 20)

The NRC Safety Evaluation for this issue was based upon the CPCo October 24, 1983 request and analysis. The NRC Staff revised the CPCo limits to meet their evaluation requirements and CPCo agreed with the revisions.

Evaluation

Pressure-temperature limits must be calculated in accordance with the requirements of revised Appendix G, 10 CFR 50, which became effective on July 26, 1983. Pressure temperature limits that are calculated in accordance with the requirements of Appendix G, 10 CFR 50 are dependent upon the initial RT_{NDT} for the limiting materials in the beltline and closure flange regions of the reactor vessel and the increase in RT_{NDT} resulting from neutron irradiation damage to the limiting beltline material.

The BRP reactor vessel was fabricated to ASME Code requirements, which did not specify fracture toughness testing to determine RT_{NDT} for each reactor vessel material. Hence, the initial RT_{NDT} for materials in the closure flange and beltline region of the BRP reactor vessel could not be determined in accordance with the test requirements of the ASME Code. Therefore, the initial RT_{NDT} for these materials must be estimated from material test data for other similar materials used for fabrication of reactor vessels in the nuclear industry. The licensee, in developing the pressure-temperature limits proposed in the October 24, 1983 submittal, estimated the initial RT_{NDT} of the limiting closure flange material as 30°F. The licensee indicated that the limiting closure flange region material is the base metal, which was fabricated to the ASME Code requirements of SA 336 Code Case 1236 and was heat treated to the quenched and tempered condition. The chemical composition and heat treatment requirements of ASME SA 336 Code Case 1236 material are similar to that of ASME Code SA 508 Class 2 material. Hence, a conservative estimate of the initial RT_{NDT} of the licensee's closure flange base material may be based upon a conservative estimate of RT_{NDT} for quenched and tempered SA 508 Class 2 material. According to Table 4.4 of NUREG-0577, "Potential for Low Fracture Toughness and Lamellar Tearing on PWR Steam Generator and Reactor Coolant Pump Supports," the upper bound RT_{NDT} for quenched and tempered ASME SA 508 Class 2 material is 40°F. Thus, the staff concluded that the initial RT_{NDT} of 30°F estimated by the licensee for the closure flange region was not conservative and unacceptable. Accordingly, the staff revised the proposed limits using an initial RT_{NDT} of 40°F.

The licensee indicated that the limiting beltline region material is weld material fabricated using Arcos B-5 flux, which has a chemical composition of .27 percent copper and .10 percent nickel. The licensee in their January 29, 1982 letter to D. M. Crutchfield indicated that Arcos B-5 flux weld material has a high initial upper shelf and an initial maximum RT_{NDT} of -50°F. The basis for this estimate was EPRI fracture toughness data for welds with high upper shelf properties. However, to conservatively estimate the initial RT_{NDT} for the Arcos B-5 flux weld material, the applicant has used the staff's estimate for Linde 0091 flux weld material which is reported in Appendix E of SECY-82-465, "Pressurized Thermal Shock." The estimated initial RT_{NDT} for this material was -56°F with a standard deviation of 30°F. Since Linde 0091 flux welds have high initial upper shelf properties and have an initial RT_{NDT} similar to that of Arcos B-5 flux weld materials, the staff concludes that SECY-82-465 material data for Linde 0091 flux welds will conservatively predict the initial RT_{NDT} of the Arcos B-5 weld materials.

The increase in RT_{NDT} resulting from neutron irradiation damage was estimated by the licensee using an empirical relationship, which was reported by Dr. Randall of the staff at the ANS Annual Meeting in Detroit, Michigan, on June 14, 1982. The empirical relationship reported by Dr. Randall depends upon the amount of neutron fluence, and the amount of copper and nickel in the weld material. This empirical relationship has a standard deviation of 30°F for weld metals. The BRP surveillance weld metal test results are reported in Table 5-5 of WCAP-9794 (Reference 21). The empirical relationship reported by Dr. Randall, provides a conservative estimate of the effect of neutron irradiation damage on weld material, because the increase in RT_{NDT} predicted by the mean empirical relationship exceeds that from the surveillance weld material for four out of five neutron fluences.

The applicant has estimated the neutron fluence to be received by the reactor vessel beltline materials in accordance with the methods described in Westinghouse Topical Report WCAP-9794. This method is currently under review by the staff. The licensee originally proposed the pressure/temperature limits in the form of a table.

After reviewing the table, the staff concluded 1) that the table did not accurately show the lower limit temperature restrictions imposed by Appendix G to 10 CFR Part 50; and 2) that table format was difficult to read and understand. Therefore, the staff revised the limits from a table format to a graph format and included the appropriate lower limit temperature restrictions.

The amount of time that pressure-temperature limits are effective depends upon the amount of neutron irradiation damage. The applicant has used the method described in Appendix E of SECY-82-465 to predict the amount of neutron irradiation damage. This method of predicting neutron irradiation damage depends upon the predicted amount of neutron fluence, the amount of nickel and copper in the weld, the standard deviation for the initial RT_{NDT} and the standard deviation for the empirical relationship, which was used to predict the amount of neutron irradiation damage. The staff concludes that the method used by the applicant for predicting neutron irradiation damage is acceptable and that the proposed pressure-temperature limit curves meet the safety margins of Appendix G, 10 CFR 50, for a period of time corresponding to 18 EFPY. Hence, the revised pressure/temperature limit curves are acceptable.

As indicated previously, the method of estimating neutron fluence is currently under review by the staff. Since there is considerable margin between the method utilized to predict radiation damage and the amount reported from the surveillance weld metal samples, the result of the staff's review of the licensee's method of predicting neutron fluence should not significantly impact the licensee's pressure-temperature limits curves for several years. If the staff's review of this method indicates that the predicted neutron fluence for the BRP reactor vessel are significantly non-conservative, the staff will revise the effective period for the licensee's pressure temperature limit curves.

Conclusion

The staff has further concluded, based on the considerations discussed above, that: 1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner; and 2) such activities will be conducted in compliance with the Commission's regulations and the issuance of these amendments will not be inimical to the common defense and security or to the health and safety of the public.

5.3.2.2 Operational Requirements (Reference 21)

Vessel metal temperature is normally measured by four (4) thermocouples located at 0°, 90°, 180°, and 270° at the 604' elevation. The thermocouples measure outside vessel temperature from which the pressure temperature limits are based upon. Temperature measurement of the reactor vessel with the above four thermocouples will normally govern heat up and cool down conditions.

Temperature measurement during NSSS Hydrostatic Testing will be governed by thermocouples on the reactor vessel and temperature measurement systems on the steam drum.

In both cases, conservative temperature margins exist to ensure the integrity of associated components.

It should be noted that there are fourteen thermocouples on the reactor vessel and six on the steam drum, it will not be necessary to assure that all are greater than the required temperature limit, thereby allowing for thermocouple failures. Access limitations to these thermocouples should failure occur necessitates allowance for failures (Reference 22).

Other operational limitations are as described in the Technical Specifications.

5.3.3 REACTOR VESSEL INTEGRITY (NUREG 0569)

The NRC performed a documented review of the integrity of the reactor pressure vessel in NUREG-0569, "Evaluation of the Integrity of SEP Reactor Vessels," published December 1979. Appendix "C" of the NUREG provided the BRP evaluation. The important supplementary requirements of the reactor design are as follows:

- The vessel stress analysis included analysis of thermal transient and fatigue effects. The method used was based upon the method of analysis developed for Naval Reactors. The method is given in PB-15987, "Tentative Structural Design Basis for Reactor Pressure Vessels and Associated Components." The vessel stress analysis performed to the procedures outlined in this document together with the Code and Code case requirements is essentially equivalent to that required by ASME Section III for Class 1 vessels.

- The vessel was constructed of SA-302, Grade B, plate and SA-336 forging material. These materials were Charpy V-notch impact tested. A minimum Charpy impact energy of 30 ft-lbs was required at a temperature of 10°F or lower. These materials are essentially equivalent to the SA-533, Grade B, Class 1, and SA-508, Class 2, materials being used today.

- All forging material in the vessel pressure boundary was magnetic particle and ultrasonicly inspected.

- All stainless steel cladding was dye penetrant inspected after final stress relief. In addition, the cladding was ultrasonicly inspected for bonding to the base metal.

- The surfaces of completed pressure boundary welds were magnetic particle or liquid penetrant inspected.

- The weld preparations in ferritic materials were magnetic particle inspected prior to deposi' (of weld metal.

- All welds in the beltline region were made by the submerged metal arc process. The post-weld heat treatment was 21 hours of total stress relief treatment at 1125°F ± 25°F. The nominal chemical composition of weld metal is 0.27% copper and 0.014% phosphorus. The chemical composition of plate metal in the beltline region is 0.10% copper and 0.016% phosphorus. No drop weight tests were conducted on these materials. Charpy tests were conducted on weld and plate material at one temperature, 10°F. Plate material was tested in both the transverse and longitudinal directions. The Charpy energy for weld metal was over 50 ft-lbs, which is considered very good. The Charpy energy for plate materials varied from 27 to 30 ft-lbs in the transverse (weak) direction. These values are considered to be about average for this type of steel.

- Based on chemistry and expected fluence, the limiting material is estimated to be weld metal. There is limited information (refer to Section 5.3.1 above) on the type or batch of filler metal or flux used to make the vessel welds. Therefore, at present we will consider all welds to be representative of the material surveillance weld and having the chemistry reported above. Based on data from unirradiated specimens in the material surveillance program, the initial value of RT_{NDT} of the weld material is about -50°F. The initial upper shelf energy of the weld metal is about 90 ft-lbs.

5.3.3.1 <u>Generic Safety Items Applicable to the Reactor Vessel</u> (NUREG-0569)

Generic safety items applicable to Big Rock Point are vessel material low upper shelf toughness and sensitized stainless steel safe ends. The feedwater nozzle and CRD return line nozzle cracking problems are not applicable to this plant. There is no CRD return line to the reactor vessel. The excess water from the control rod drive system flows into either the recirculation system or the cleanup system. The feedwater nozzles on Big Rock Point are located on the steam drum. Condensate from the turbines is pumped by the feedwater pumps to the steam drum. Water from the steam drum is pumped to the reactor vessel by the recirculation pumps. At normal operating conditions, the temperature of the water entering the vessel is 570°F. This is about 12°F lower than the vessel temperature so thermal stresses will be very low. For transient conditions the temperature differential between the inlet fluid and the vessel wall is also relatively low. Since the initial crack growth in feedwater nozzles is due to thermal stresses, Big Rock Point should have no problem regarding cracks in the recirculation nozzles on the reactor vessel (the feedwater inlet nozzles are called recirculation nozzles). To date, no flaws have been detected in the recirculation nozzles of Big Rock Point.

There are sensitized stainless steel safe ends on the Big Rock Point reactor vessel. These safe ends are made from 304 stainless steel. We requested information on these safe ends and Consumers Power Company responded by letter dated September 11, 1970. Through 1970, no flaws had been detected in these safe ends. The 304 stainless steel was made with low carbon content which increases its resistance to stress corrosion cracking. Since the 1970 review of the safe ends, no flaws or cracks have been found in the sensitized safe ends. We conclude that, since the vessel has been operating for 15 years (currently over 25), if a corrosion problem existed there would be throughwall flaws in these safe ends by now. We also realize that inservice examinations of these safe ends have been limited (as of the date of the NUREG).

However, from this present review it is concluded that there is no evidence of any stress corrosion cracking on these safe ends. Furthermore, we believe that there are no major flaws in these safe ends because of their low carbon content.

5.3.3.2 Evaluation Conclusions (NUREG-0569)

The Big Rock Point reactor vessel was designed to ASME Code Sections I and VIII. However, the requirements of these sections were supplemented by the requirements of Nuclear Code cases, the Navy Code and purchase specifications so that the quality control and design criteria utilized were essentially in accordance with the rules of ASME Code Section III. Therefore, the initial integrity of the vessel is considered acceptable. The primary stresses in the beltline region of the vessel are low, approximately 70% of those permitted by Section III. These low stresses, along with the use of materials with adequate fracture toughness, provide assurance that brittle fracture will not occur. Inservice examinations have been performed on components of the reactor vessel in accordance with ASME Code Section XI since 1973.

The reactor vessel is currently operating with pressure-temperature operating limits that are in accordance with Appendix G, 10 CFR Part 50. The staff will continue to review and update these operating limits to account for further radiation damage on vessel materials. The amount of radiation damage will be determined from the results of tests on Big Rock Point's surveillance specimens. The material surveillance program has been reviewed and is considered acceptable. The combination of inservice inspections, conservative operating limits, low vessel stresses and the use of materials having adequate fracture toughness properties provides assurance that the integrity of the reactor vessel will be maintained at acceptable levels throughout service life. The generic safety items applicable to Big Rock Point (low upper shelf energy and sensitized stainless steel safe ends) have been successfully resolved and will not adversely affect the vessel integrity.

For additional information (since issuance of NUREG 0569) on safe-ends, refer to Section 5.2.3.4 of this Updated FHSR.

5.4 COMPONENT AND SUBSYSTEM DESIGN

GENERAL DESCRIPTION OF REACTOR SYSTEM EQUIPMENT

The nuclear steam supply system is composed of the equipment in the main recirculation loop, plus those auxiliary systems required to provide a safe and operable system. The main steam and condensate system is shown schematically by Drawing 0740G40106 with Legends identified by Drawing 0740G40105.

The main recirculation loop consists of the reactor vessel and internals, including the core, control rods, and flow baffle; the steam drum, reactor recirculating pumps, the interconnecting piping and valves, and the safety relief valves. Piping and instrument diagram details are shown by Drawing 0740G40106.

The auxiliary systems are: the shutdown cooling system, emergency cooling system, reactor cleanup demineralizer system, and the liquid poison system, all of which are shown on Drawing 0740G40107.

The reactor vessel has been previously described in Section 5.3.

The design pressures and temperatures for the reactor system equipment are summarized below:

Equipment	Design Pressure (psia)	Design Temperature (°F)
Reactor Vessel	1715	650
Drum	1700	650
Piping	1715	625
Pump	1750	617
Valves	1765	615

Data on pumps and heat exchangers are shown on the following pages.

REACTOR SYSTEM EQUIPMENT - PUMPS
Data Sheet

Service	Type	No.	Design Pressure Psia	Capacity Gpm	Diff Head-Ft
Reactor Recirculating	Vert-Cent.	2	1,750	16,000	76
Reactor Shutdown	Vert-Cent.	2	315	500	75
Core Spray	Vert-Cent.	2	140	400	324
Reactor Water Cleanup	Hor-Canned	1	1,715	95	130
Fuel Pit	Hor-Cent.	2	140	250	110
Control Rod Drive	Triplex-Recip	2	2,815	(25 at	1,975 Psig)

REACTOR SYSTEM EQUIPMENT - HEAT EXCHANGERS
Data Sheet

		Shell Design				Tube Design			
Service	Duty Btu/Hr	Flow Lb/Hr	Psia	°F	Temp °F In & Out	Flow Lb/Hr	Psia	°F	Temp °F In & Out
Reactor Shutdown (Two Heat Exchangers)	6×10^6 (both)	250,000	90	320	85 - 115	250,000	315	425	190 - 160
Emergency Condenser (Two Tube Bundles)	Operating 15×10^6 (both)	15,700	42 (External)	250	100 Initial 215 Final		1,715	650	605 - 601
	Transient 32×10^6 (both)					26,900			601 - 215
Core Spray	7.8×10^6	225,251 (Note 1)	165	80	75 - 110	197,000	225	235	140 - 100
Non-Regen Reactor Cleanup	4.76×10^6	200,000	90	200	70 - 93.7	47,500	1,715	600	210 - 110
Regen Reactor Cleanup (Four In-Series)	2.1×10^7 (all)	47,500	1,715	600	110 - 522	47,500	1,715	600	595 - 210
Fuel Pit Coolers (Two Heat Exchangers)	3×10^6 (each)	125,000	90	150	70 - 94	125,000	90	150	119 - 95

[1]Shell Discharge Line Orifice Limits Flow to ≃ 211,000 Lb/Hr at 133 Psig Electric Fire Pump Pressure via Facility Change FC-479 to prevent tube damage.

5.4.1 REACTOR COOLANT RECIRCULATING PUMPS

Recirculation water from the steam drum is returned to the reactor vessel by two vertically mounted recirculating pumps normally operating in parallel. The pumps are single stage, centrifugal, double suction with an overhung impeller. A water-lubricated bearing is provided internally to reduce the overhang. The pumps are of a mechanical seal-limited leakage design, with a total estimated outflow of 16 gallons per hour maximum per pump to the waste system.

The pump rating is 16,000 gpm at 76 feet of head.

The 17,000 gpm volume flow rate from each pump is essentially constant with two operating. With one pump operating, flow increases about 20% over normal with a corresponding increase in net positive suction head (NPSH) required.

The net positive suction head (NPSH) requirements as determined by test on the actual pumps, "as built," and curves showing the effects of reduction of NPSH available from pump design conditions were provided to the NRC in Section 5 of Reference 1.

The pump driver is a vertical, drip-proof induction motor of a conventional design rated for 400 HP at 900 rpm at 2300 volts. An oil-lubricated Kingsbury thrust bearing in the motor will absorb all residual pump and motor thrust.

The pump casing design and construction is in accordance with the ASME Boiler and Pressure Vessel Code, Section VIII, 1959 edition, and the applicable latest interpretations of the nuclear code cases. The pump body, cover, and impeller are stainless-steel castings solution heat-treated to ASTM-A-381-58T, in compliance with ASTM-E71-52 standards for Class 2.

Pump Cavitation (Reference 1, Section 5)

During the reactor warm-up period at low power, from atmospheric to full pressure with essentially no feed-water flow, the recirculating water being pumped will be at pressure and saturation temperature. The steam drum elevation will supply the NPSH for the pumps during this period. At low pressures with one pump in operation, limited pump cavitation may occur due to high pressure drops in the piping before the pump suction. The consequences of such cavitation are not serious, unless they continue for long periods of time, measured in days or weeks. In the latter event, some loss of metal from the surface of the impeller could result.

Recirculating Pump Seals

Two mechanical seals operating in series are supplied with each unit. The first seal and the Carbon-A water-lubricated bearing operate at system pressure and are protected from the system temperature by a heat shield and a supply of lower temperature (125°F max) reactor water that is circulated (12-15 gpm) by a small shaft pump through an external cooler. The cavities between the first and second seal are cooled (125°F max) by reactor water drawn off (15 gph max) the pump suction, passed through a heat exchanger and reduced in pressure through a fixed orifice. The pressure reduction establishes that each seal sees only one half of the total reactor operating pressure across its faces at operating pressure. However, each seal is designed to take total reactor pressure across its faces if, for any reason, the other seal malfunctions. An instrumentation system (Drawing 0740G40237) is supplied to monitor temperature and pressure at each seal and will warn of any abnormal condition. An evaluation of the consequences of loss of cooling water to the pump seals is provided in Section 5.4.9 of this Updated FHSR.

The pumps are welded into and are partially supported by the recirculation piping, but the principal supports are two constant support hangers with subsidiary spring hangers. A vibration dampener at each pump is also supplied. A switch is mounted on each pump to warn of excessive vibration.

5.4.1.1 Recirculating Pump Trip (RPT) (Reference 23)

Section 50.62(c)(5) of 10 CFR Part 50 requires that each boiling water reactor (BWR) must have equipment to trip the reactor coolant recirculating pumps automatically under conditions indicative of an Anticipated Transient Without Scram (ATWS).

CPCo provided a Probabilistic Risk Assessment (PRA) February 26, 1981 addressing the RPT modification which concluded that the automatic RPT is a relatively ineffective method of reducing core damage during ATWS sequences primarily due to the large primary system safety valve capacity at the facility. Other plant specific design features such as the absence of a continuous source of high pressure safety injection and relatively quick effectiveness of the liquid poison system contribute in the mitigation of ATWS consequences.

In May 1984, the Commission issued NUREG-0828, "Integrated Plant Safety Assessment, Systematic Evaluation Program" (IPSAR). The staff concluded that unlike larger BWR plants, Big Rock Point does not need an automatic RPT feature to compensate for positive pressure reactivity at the end of core life. In view of the small risk reduction potential, the staff also concluded that an automatic RPT modification is probably not warranted.

By letter dated March 20, 1986, in response to a CPCo October 14, 1985 request, the NRC issued an exemption from 10 CFR 50.62(c)(5), such that installation of an automatic recirculating trip system is not required at Big Rock Point.

5.4.2 STEAM DRUM AND STEAM DRUM RELIEF VALVES

The steam drum, with its piping, is mounted high up inside the enclosure to perform the following functions:

Separate the steam from the steam-water mixture generated in the reactor core. The design criteria calls for drum exit steam quality of 99.9%.

Provide water storage to accommodate surges of water level and pressure between the reactor vessel and the drum.

Provide natural circulation driving head to maintain flow in case the recirculating pumps are inoperative. It has been calculated that it will be possible to run at over 50% load on natural circulation alone with both pumps inoperative but free to rotate.

Assure net positive suction head for the recirculating pumps to meet their design requirements. Drum water level is 65 feet above the center line of the pump suctions. The static head is sufficient to maintain flow during normal operation without pump cavitation; during transient conditions limited pump cavitation may occur.

Serves as a mixing tank for the cooler feedwater and hot recirculating water. This aids in smoothing (or absorbing) part of the reactivity changes due to moderator temperature changes.

Approximately 500 cubic feet of water are stored in the drum. If, at full load operation, all steam voids in the core, reactor vessel and riser piping collapsed, the water storage available is sufficient to keep the downcomer piping inlets covered. Operational transients and pump vibration will not occur as a result of steam drawn into the pump suction, and the supply of reactor recirculating water will be maintained.

5.4.2.1 Design Information

Combustion Engineering (CE) Incorporated designed, fabricated and tested the steam drum in accordance with the requirements of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section 1 - 1959 edition utilizing Code Cases 1270N and 1273N and Part UCL of Section VIII - 1959 edition for internal cladding weld overlay. The design was in accordance with General Electric Company Specification DP-19890, Revision 0, with modifications as shown on detailed drawings and as noted in C.E. Book No. 6460-D, September 1961, Instruction Manual - Primary Steam Drum.

The drum is a horizontally mounted "Code Stamped" cylindrical pressure vessel with internal steam drying and auxiliary equipment. Base material for shell and heads is SA-212-B, Fire Box, carbon steel clad with 5/32 inch (TP-304 stainless steel) minimum weld deposited with type 309 and type 308 stainless steel weld rod with equal to or better than 250 RMS surface finish on all internal surfaces. The cladding thickness is not considered in wall thickness calculations. Nozzles four inches and over are SA-105-Gr. II carbon steel forgings clad internally with stainless steel. Nozzles under four inches are solid inconel SB-166. The drum internals are essentially stainless steel and inconel plate and strip, A-167 and A-276 type 304 and SB-166 and SB-168. The Manway pad and cover are SA-105-GR. II carbon steel, hex nuts are A-194-2H and studs are A-193-B7 for the Manway closure.

Design Calculations

Design calculations for the drum have been made to cover the following:

ASME Code allowable stresses.

A detailed structural analysis of the shell nozzles and attachments to account for principal stresses and their combination for normal and transient power operation.

A transient analysis that concerns itself with the fatigue limits of the design (refer to CE Book No 6460D, Instruction Manual).

5.4.2.2 Steam Drum Characteristics

Table 5.6 Steam Drum General Characteristics [1]

Length, Overall, Feet [2]	40
Inside Diameter, Inches	78
Wall Thickness, Excluding Cladding, Inches	4-3/8
Cladding Thickness; Minimum, Inches	5/32
Design Pressure, Psia [3]	1700
Design Temperature, °F [3]	650
Weights, Lbs	
Dry Weight, Actual (including internals)	199,100
Wet Weight, Calculated @100% Load, @600°F [4]	225,100
Flooded Weight, Calculated @600°F	251,100
Hydrostatic Test Pressure, psig [5]	2,528
Cycles of normal start-up and shutdown [6]	2,000
Cycles of Emergency Shutdown [6]	100

1. Does not include manufacturing tolerances per "As Built" drawings.

2. As Built overall length including Manways is about 40 feet 9 inches.

3. Design Temperature at Design Pressure equals 614°F saturated.

4. 100% Load Design Pressure equals 1470 psia

5. After fabrication, and prior to shipment, the current hydrotest limits are contained in the Technical Specifications.

6. The drum will withstand a normal (100 degrees/hour - from and to 100% power and 594°F) start-up and shutdown approximately 2000 times, and approximately 100 emergency shutdowns from 100% power with a cooling rate of 6.4°F/min (384 degrees/hr). Normal cooling of the drum will be limited to 100 degrees/hour, but, 300 degrees/hour will be allowed in an emergency shutdown.

5.4.2.3 Steam Drum Penetration Nozzles

Table 5.7 Steam Drum Penetration Nozzles

Heads (2 Penetrations - 1 in Each End)	
2 Manway Openings	18" ID
Drum Shell (35 Penetrations)	
6 Steam Riser Nozzles A-105 Gr.II with TP-316 extensions	14"
2 Feedwater Inlet Nozzles A-105 Gr.II with type 304 stainless steel sleeves	8"
4 Steam Outlet Nozzles A-105 GR.II	8"
4 Downcomer Nozzles A-105 Gr.II with TP-316 extensions	17"
6 Safety Relief Valve Openings SB-166 Inconel with A-105Gr.II flange	3" ID
2 Condensate Return Nozzles (From Emer Cond) A-105 Gr.II with carbon steel extensions	4"
2 Remote Level Indicators (Upper) SB-166 Inconel with Carbon steel extensions	1 1/2"
Head (2 Penetrations - 1 in Each End)	
2 Remote Level Indicators (Lower) A-105 Gr.II with carbon steel extensions	4"
1 Vent From Reactor SB-166 Inconel with TP-304 extensions	1 1/2"
2 Vents From Steam Drum SB-166 Inconel with carbon steel extensions	1"
1 Sample Nozzle SB-166 Inconel with carbon steel extensions	1"
1 Decontaminating Nozzle SB-166 Inconel with carbon steel extensions	2"
2 Gage Glass Nozzles SB-166 Inconel with carbon steel extensions	1 1/2"

Nozzle Cyclic Stress Analysis (Reference 1 and 24)

A summary of the cyclic stress analysis for one of the most critical nozzles (17" downcomer), including the loading imposed on the nozzle from the piping was supplied to the NRC by letter dated May 3, 1962 and again by letter dated March 12, 1975 in response to a February 24, 1975 request for stress analyses. This February 24, 1975 letter also included the Manufacturers Data Report.

5.4.2.4 Steam Drum Support

Eighteen 1 1/2 inch support lugs are provided for supporting the drum during operation.

The drum is supported from the concrete overhead structure by 8 constant support hangers. Movement of the drum due to thermal expansion of the piping and reactor vessel is compensated for by a specially designed support system which makes the downcomers, anchored at the same elevation as the reactor vessel, into thermal rams to move the drum up as the rising temperature expands the components. Sidewise movement is controlled by guide rods that make the drum move on a line between the center of the drum and the reactor vessel anchor point. Additional rods keep the drum from rotating or skewing. The suspension and support system as designed for maintaining the position of the steam drum is capable of withstanding the forces developed by a riser or downcomer line break.

5.4.2.5 Miscellaneous Externals

Six peen pads and brackets are provided for attachment of thermocouples. Angles that extend circumferentially and longitudinally, are furnished for supporting the external three inch thick insulation.

5.4.2.6 Steam Drum Internals

The steam-water mixture from the reactor passes through the riser piping to the steam drum.

The internals of the drum provide for three stages of steam separation, with a steam exit quality of about 99.9%. Sixty stainless steel turboseparators provide the first stage separation. The turbo steam separators are located in two rows of 30 each on top of the riser baffle boxes. The riser baffle boxes direct the steamwater mixture from the six risers into the bottom of the turboseparators where the moisture is removed from the steam by centrifugal action. Secondary steam separation is provided by stainless steel steam dryers on top of the turboseparators. The final stage of steam separation is by the screen dryer assemblies located at the top of the steam drum through which the steam must pass to the steam outlet nozzles. The steam then passes to the turbine, while the water is returned to the bottom of the drum. The water passes down the downcomers into the recirculating pumps where it is then pumped into the reactor vessel.

Refer to Figure 5.4, Internal Arrangement - Steam Drum

Riser Baffle Box - A pressure containing box-type structure of inconel is provided on the inside of the steam drum at each of the six (6) riser nozzles. The box-type structures receive the steam-water mixture from the riser nozzles.

Turbo Separators - Sixty (60) TP-304 stainless steel steam separators are located in two (2) equal rows on the top of the riser baffle boxes. Each separator consists of a primary stage incorporating a turboblade to remove moisture by centrifugal action (cyclone moisture separator) and a secondary section (chevron steam dryer) which removes moisture by causing the steam to impinge upon corrugated steel plates. The steam-water mixture exits from the baffle boxes through the turbo separators.

Screen Dryers - Type 304 stainless steel screen dryers are provided at the top of the pressure vessel and extend the length of the cylindrical shell. Each of the four (4) outlet nozzles is provided with approximately twenty (20) square feet of screen area, through which the steam must pass prior to leaving the steam drum via the steam outlet nozzles.

Feedwater Pipe - Approximately thirty feet (30) of type 304 stainless steel, perforated, 5" sch. 40 piping is provided to distribute and mix the cool make-up water that enters through the feedwater nozzles with the warm recirculating water from the reactor by admitting it along the entire length of the drum. A Type 304 stainless steel sleeve in the feedwater nozzle connects to this feedwater piping.

Downcomer Screen Assembly - Each of the four (4) downcomer nozzles is provided with screens through which the water leaving the steam drum must pass. These screens prevent large metal particles, that might work themselves loose, from circulating through the system.

FIGURE 5.4

INTERNAL ARRANGEMENT STEAM DRUM

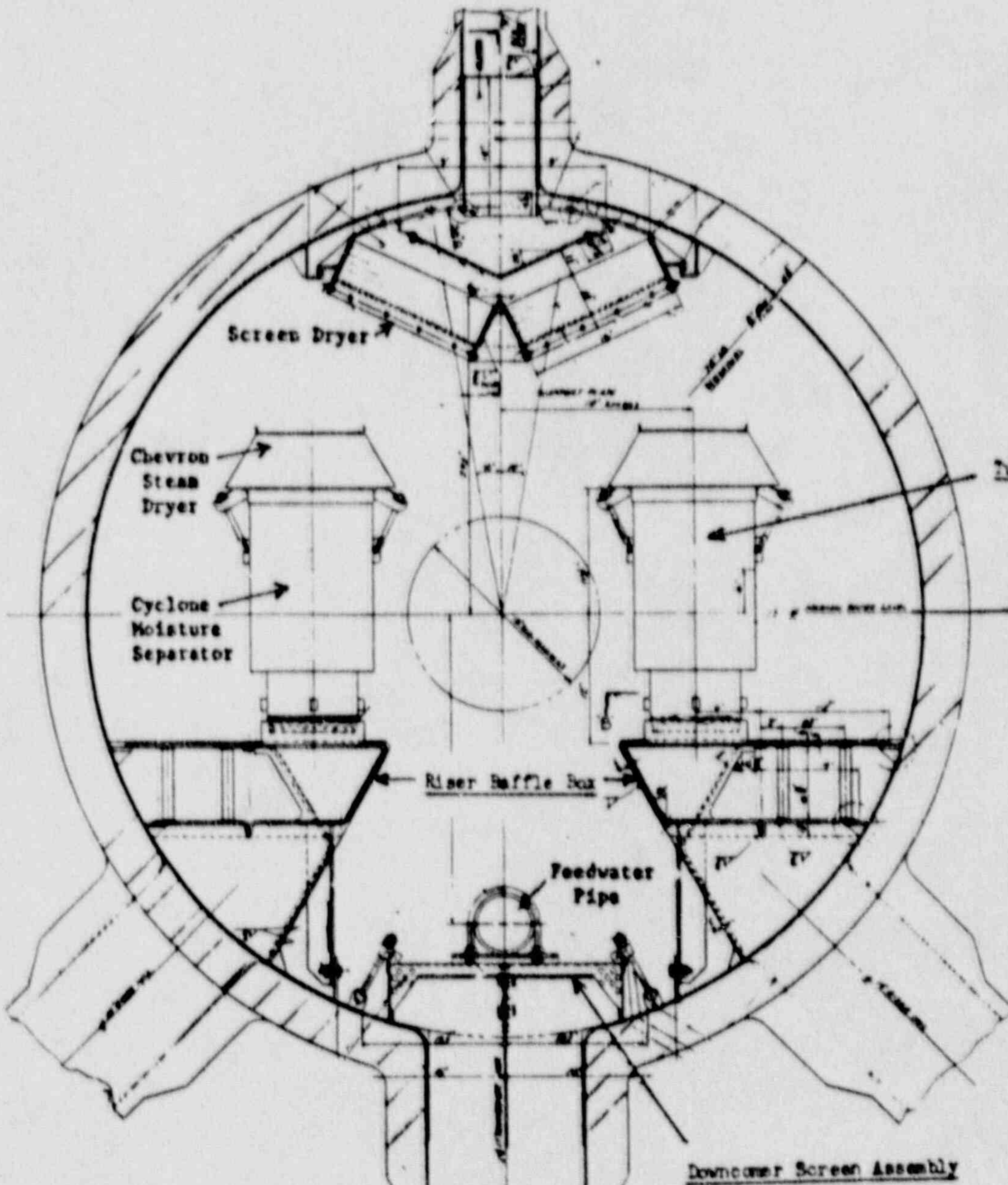


FIGURE 5.4

5.4.2.7 Steam Drum Safety Relief Valves (Reference 25 and 26)

The six spring loaded safety relief valves are located on the top of the steam drum on flanged three inch ID nozzles. The valves have a three inch inlet and six inch outlet. The outlet is also flanged and is attached to a flanged elbow to direct the discharge up in a vertical direction to the atmosphere within the steam drum enclosure. Each elbow is flanged at the discharge end and is fitted with a rupture disc. The purpose of the rupture disc is to contain any weeping from the valves. This weeping is piped via lines at the bottom of each discharge elbow to the enclosure clean sump through a common collection line. A high temperature alarm in the drain signals excessive seat leakage. The rupture discs are specified at 10 psig ± 3 psig. Detailed information on the safety valves was provided to the NRC in response to NUREG-0737, Item II.D.1 "Relief and Safety Valve Testing" by letters dated October 1, 1981, December 19, 1981, July 22, 1982 and June 18, 1984 to address the safety valve test program performed in cooperation with the Dairyland Power Cooperative. As a result of this test program, the valve springs were replaced utilizing Specification Field Change - SFC 83-012.

Operating Requirements

Technical Specifications requires the safety relief valves to be set approximately for all planned reactor operating pressures so that the allowable pressure of 1870 psia (1700 design plus 10%) in the nuclear steam supply system is not exceeded. The Technical Specifications further requires the following criteria for overpressure protection:

Safety Relief Valves:

Number	6
Type	Spring-Loaded
Maximum Setting of First Valve Including Rupture Disc, Psia	1700
Sequential Pressure Increment Setting of Remaining Valves, Psi	10
Minimum Capacity per Valve (1202 Psia Setting), Pounds per Hour	2.36×10^5
Valve Orifice Area, Square Inch	3.976
Acoustic Position Monitors	6

Steam Drum Relief Valve Operating Parameters

Accident Analyses performed as part of Chapter 15 of this Updated FHSR utilized a nominal setpoint range of 1550 to 1600 psia as an input in calculations performed. These values correspond to the current setpoint of 1535 to 1585±5 psig which is a function of nominal Reactor Pressure, ~1335 psig plus 200.

5.4.2.7.1 Steam Drum Relief Valve Setpoint (Reference 25)

Big Rock Point initially operating at a nominal pressure of approximately 1000 psia with relief valve set pressures of 1237-1288 psig. The ultimate design rating of the plant was specified for operation at 1500 psig with safety valve set pressures approximately 200 psi higher.

When the Atomic Energy Commission (AEC) demonstration program was finished in 1964, an optimum nominal operating pressure of approximately 1350 psia was selected with the relief valves set approximately 200 psi higher.

Section 12.5.8 of the November 14, 1961 FHSR postulated an accident of coincident steam shut off with failure to scram to establish the size and settings of the safety relief valves. The setting of the first safety valve was varied from 155 psi to 250 psi through computer analyses. The analysis indicates that the six steam drum safety valves have a total capacity of nearly twice the normal steaming rate at a 200 psi differential set pressure.

Section 4.5.8 of the November 14, 1961 FHSR assumed all steam drum safety valves blowing at 1870 psia when describing the liquid poison system capability to provide poison flow to the reactor.

Thus, the setpoint of the steam drum safety valves affects both the overpressure protection for the primary system and the operation of the standby liquid poison system.

It should be noted that full valve lift is not to be expected during any anticipated operational transient at Big Rock Point. Calculations have shown that the first relief valve will not lift for several minutes following reactor scram, and that a relief valve will only lift if either loop of the emergency condenser fails to operate. At the point of opening, the rate of pressure rise is expected to be so small that only one of the six valves (each of which has a rated capacity of approximately 35% of normal steaming rate) is more than adequate for relieving system pressure. In addition, the analyzed worst case event, the Anticipated Transient Without Scram (ATWS), establishes relief capacities that are equivalent to full lift of approximately four of the six valves.

Setpoint Evaluation (Reference 26)

In response to high initial as-found setpoints observed during surveillance testing, reported via Licensee Event Report 87-003, Rev 3, the effects of various safety valve settings on primary system pressure was evaluated.

First, various ranges of safety valve setpoints were used to evaluate overpressure response to a 100% ATWS event using a 1986 RETRAN model of Big Rock Point. This model conservatively assumes an initial power level of 244.5 MWt and primary pressure of 1350 psig. Additionally, no credit is taken for operation of the emergency condenser which serves as the initial pressure control system prior to actuation of the safety valves.

The results of these are tabulated below:

Safety valve settings: 1740, 1750, 1760, 1770, 1780, 1790 psia
Maximum system pressure: 1812.5 psia

Safety valve settings: 1750, 1760, 1770, 1780, 1790, 1800 psia
Maximum system pressure: 1823.2 psia

Safety valve settings: 1820, 1821, 1822, 1823, 1824, 1825 psia
Maximum system pressure: 1868.65 psia

Safety valve settings: 1652, 1653, 1654, 1655 psia* *(because it was found only four of six relief valves are needed to provide pressure relief; steam drum relief valve numbers 5 and 6 were set artificially high to assure they would not open)
Maximum system pressure: 1702 psia

It was also noted that during the first three runs with all six safety valves set near required range, that only two valves fully open and the third and fourth cycle to maintain system pressure. The fifth and sixth valves do not open. This margin in relief capacity concludes that only four safety valves are required to mitigate the ATWS transient.

From this evaluation, in order to exceed 1870 psia it is necessary that all six safety valves operate at setpoints greater than 1820 psia.

With respect to liquid poison system (LPS), the primary purpose of the nitrogen pressurization of the LPS is to ensure positive displacement of the poison solution when the reactor recirculation system is static, such as refueling, when there is no initial driving head to establish a siphon action in the poison tank. At maximum system pressures, the nitrogen volume is sufficient to displace enough solution to establish the siphon action needed to discharge the solution to the reactor recirculation system when the initial gas injection pressure is maintained within the Technical Specification operational limits discussed in Section 4.8 of this Updated FHSR.

NRC Safety Evaluation (Reference 27)

As part of the May 1984 "Final" Integrated Plant Safety Assessment - Systematic Evaluation Program, NUREG 0828 - Section 5.4.10 "Relief and Safety Valve Testing" to meet NUREG 0737 Item II.D.1, remained open for review completion by the NRC staff. The review was completed January 10, 1985. Conclusions from this evaluation state that:

"We find the information submitted demonstrates the ability of the reactor coolant system safety valves to function under expected operating conditions for design basis transients and accidents, thus completing Item II.D.1 for Big Rock Point."

5.4.2.7.2 Steam Drum Relief Valve Position Monitoring System

In response to NUREG 0737, Item II.D.3, BRP installed a Relief Valve Position Monitoring System (via Facility Change FC 489) to provide the control room operators with a reliable open/closed position indication of the steam drum relief valves. A detector is mounted on each relief valve. Each detector acts essentially as a sensitive microphone, with a readout on a panel in the main Control Room. The excessive noise generated by an open relief valve will be detected by the detector on the valve and cause an alarm in the Control Room. Redundancy is provided by the close proximity of pairs of Safety Valves.

Backup methods for determining that a safety relief valve is open include the relief valve common drain header high temperature alarm, containment high-pressure alarm, and indication and Dewcell temperature recorder for high humidity and pipeway high temperature. These indirect methods of recognizing an open valve are discussed in Plant Operating Procedures and Section 5.2.5 of this Updated FHSR.

NRC Evaluation(s)

- NRC Staff review of the relief valve monitor installation design was provided in their May 2, 1980 evaluation which concluded that BRP, "has met with our requirements for this (NUREG-0737) item," with the adequacy of installation, qualification and procedures to be reviewed by NRC Inspectors.

  By letters dated September 25, 1980 as clarified by the September 18, 1981, BRP requested Technical Specification Changes to incorporate the Relief Valve Monitor operability requirements.

- By letter dated October 9, 1981 Amendment 49 was issued, the amendment concluded that the revised Technical Specification meets NRC requirements and is acceptable.

- The NRC issued NUREG-0828, Integrated Plant Safety Assessment - Systematic Evaluation Program - Final Report, in May 1984. Within the report, section 5.3.1.4 addressed "Position Indication of Power-Operated Relief Valves as follows:

  As a result of the events at TMI, the NRC staff requires (NUREG-0737, Item II.D.3) direct indication of power-operated relief valve (PORV) position. Big Rock Point does not have PORVs for pressure control. The spring-operated safety valves have nonenvironmentally qualified position indication.

  The licensee has determined that it is not necessary to upgrade the safety valve position indication because there are no controls nor are there any gagging devices for these valves. Although such indications could identify a stuck-open or leaking valve, containment atmosphere monitors provide equivalent indicators. Therefore, the staff concludes that no further action is necessary.

CPCo Conclusion

The purpose of the position indication for safety relief valves on the steam drum is to provide the operator with unambiguous indication of valve position (open or closed) so that appropriate operator actions can be taken.

The appropriate actions to be taken for a stuck open or leaking SRV are the same as those for a Loss of Coolant Accident (LOCA) at Big Rock Point; since, 1) the SRVs are spring loaded with no remote actuation capability, 2) there are no isolation valves in the discharge piping from the steam drum and 3) no operator action can be taken to stop leakage or blowdown through an SRV.

Existing instrumentation required to recognize a LOCA is available.

The steam drum SRV Monitoring System serves only as redundant backup to existing instrumentation. A common drain header high temperature alarm to the SRVs also provides redundancy.

The NRC October 9, 1981 Safety Evaluation supporting Amendment 49, in Section 3.2 states that the Valve Position Indication - "system did not need to be safety grade provided that backup methods of determining valve position are available."

Consequently, the commitment to pursue environmental qualification for components of the system is not necessary.

Operational Requirements

Operating requirements for the steam drum relief valve monitoring instrumentation were incorporated into the Technical Specifications by Amendment Number 49, dated October 9, 1981.

The Amendment requires that at least three (3) steam drum safety valve position monitors shall be operable during power operation. Also, one of every two (2) adjacent monitors oriented in each north-south plane shall be operable. In the event that any of these monitoring channels become inoperable, they shall be made operable prior to startup following the next cold shutdown.

The basis for the operability stems from the CPCo September 18, 1981 submittal regarding the capability of adjacent position monitors to detect opening of the nearest adjacent valve through relief valve monitor cross-coupling. Cross-coupling will occur if one of the nearby valves lifts, the noise level from the lifted valve will be cross-coupled to the nearest (adjacent) valve.

5.4.2.8 Steam Drum Level and Temperature Elements

Level Elements

A level gage is installed on the drum for local visual indication. This level gage is not normally used during operation.

Two level elements at each end of the drum (four total) send level indication to two separate and independent Level Indication Systems - Bailey and Yarway. Yarway level indication of the steam drum reads out on four indicators on the south steam drum wall, near NSSS Local Panel C-30. The four level indicators on the drum are also transmitters which provide remote level indication on the main control panel and low level scram signals to the Reactor Protection System. Bailey level indication reads out on a level gage and a level recorder, both located on the main control panel. Pressure transmitters from each end of the steam drum send their signals to the main control panel.

The two Yarway Level Elements (RE08A and RE08B) on the steam drum and four Yarway Level Elements (RE09A, RE09B, RE09C and RE09D) are insulated and heat traced to maintain a constant temperature within the element (reference Facility Change FC-497). The level element reference legs must not exceed 250°F, when the containment sphere temperature approaches 235°F under accident conditions, if the instruments are to perform their safety related functions.

Information on the preliminary design of this modification to the level elements and a functional description of the steam drum level instruments was provided to the NRC by CPCo letter dated September 18, 1980. Further information on the modification and installation of reference column heating was submitted by CPCo December 19, 1980 letter concerning NUREG-0737 Item II.F.2 - Instrumentation for Detection of Inadequate Core Cooling.

The purpose of the modification was to stabilize the density of the water in the reference column of the steam drum level elements. Drum level instrumentation and alarm can then be calibrated with fixed temperature compensation.

Temperature Elements

There are six (6) temperature elements attached to the steam drum. These thermocouples are located (top and bottom) in both semi elliptical heads and the middle shell section of the steam drum. The thermocouples are inserted into drilled peen pads welded to the base metal. The temperature signal is routed to the main control panel in the control room and recorded.

5.4.3 REACTOR COOLANT PIPING AND VALVES

For additional information on this subject, refer to Section 5.1 and 5.2.

General features and characteristics of the Primary Coolant Recirculation System are as follows:

Design Features and Characteristics

Number of Recirculation Loops	2
Number of Recirculation Pumps per Loop	1
Approximate Internal Volume of System Excluding Reactor Core and Internals to Isolation Valves, Cubic Feet	3830
Approximate Volume of Coolant in System During 157 Mwt Operation, Cubic Feet	2689

Recirculating Valves, Each Loop

Location	Type	Mode of Operation	Size Inches	Opening Rate Inches/Minute
Pump Suction	Gate	Motor	24	12
Pump Discharge	Gate	Motor	20 w/15 port	3
Pump Discharge Bypass Valve	Gate	Motor	5	12
Pump Discharge	Butterfly	Electrically Disabled and Locked in the Full Open Position		

Recirculation System Piping

Location	Size	Number	Material
Risers (SS)	14" x 0.712 Wall	6	316 Stainless Steel
Downcomers	17" x 0.858 Wall	4	316 SS
Pump Suction	24" x 1.21 Wall	2	316 SS
Pump Discharge	20" x 1.009 Wall	2	316 SS

Reactor Power Operation Cooling

Coolant Material	Demineralized Water
Type of Cooling System	Forced Recirculation
System Pressurization	Boiling Water
Minimum Loops Operating Concurrently (or Equivalent)	1
Number of Passes Through Core	1
Flow Direction Through Core	Upward

5.4.3.1 Primary Loop Piping and Valves

5.4.3.1.1 Piping

The primary loop piping connects all the major components of the nuclear steam supply system. Six 14" risers carry the steam-water mixture from the reactor vessel to the steam drum. After the steam is separated, the recirculating water flows out of the drum down four 17" downcomers. The downcomers, in two groups of two each, join together into two 24" pump suction headers. The two pump discharges are each 20" and return to the bottom of the reactor vessel.

Piping design criteria and material selection are in conformance with the Code for Pressure Piping, ASA-B31.1-1955, latest revision and nuclear code cases N-1, N-7, N-9 and N-10; as specified by the laws of the State of Michigan. Fabrication and inspection of the pipe conforms to the applicable sections of Section I of the ASME Boiler and Pressure Vessel Code. Piping supports are designed in accordance with Section VI of the Code for Pressure Piping, ASA-B31.1-1955 latest revision.

The risers are stainless steel ASTM-A-358 type 316 seam welded pipe, which includes full radiograph and fluid penetrant test of the seam. The remainder of the pipe is stainless steel centrifugally cast ASTM-A-351 grade CF8M and meeting the requirements of nuclear code case N-9. The fittings are stainless steel castings ASTM-A-351 grade CF8M and meeting the requirements of nuclear code case N-10. Miscellaneous small piping satisfies to ASTM-A-376 type 316. All shop welds and field welds are fully radiographed and fluid penetrant tested. Those sections of pipe which have been hot formed during fabrication have been solution heat treated and quenched. After the piping system was erected it received a hydrostatic test to 1.5 times design pressure.

For the proper design of the piping system, a complete stress analysis was performed that took into account stresses due to expansion of the pipe and its connecting equipment, plus externally applied loads due to hangers, supports, anchors and vibration eliminators. Wall thickness design includes stresses due to normal operation and stresses due to over-pressures under transient conditions.

The piping is hung from constant support hangers at selected points, and anchored in the vertical direction at the same elevation as the reactor vessel supports. This effectively allows the pipe and other system components to expand freely up or down from the anchor point with very small resultant thermal expansion stresses. A system of vibration dampeners are fastened to the piping to reduce possible vibration stresses. An open bypass between the suctions of the two pumps will keep water flowing in the four downcomers even if one pump is out of service. This feature prevents tilting of the steam drum due to unequal thermal expansion in the two sets of downcomer piping.

5.4.3.1.2 <u>Valves</u>

Two 24" gate valves, one on each suction side, and two 20" gate valves (with reduced ports of 15 inches), one on each discharge side of the two recirculating pumps, serve as stop valves.

The two 20" gate valve motor operators were modified via Facility Change FC-245 to change the operating speed from 5"/minute to 3"/minute which is a 300 second travel time for the 15" reduced ports. Details on the change were provided to the NRC by letter dated December 6, 1973.

The pumps are of a limited leakage design and the valves are needed to assure shutoff if pump leakage becomes excessive for any reason. These valves are remote motor operated by extension shafts and can be hand operated from the far side of the shielding wall if necessary. Each pump has a startup bypass around the discharge stop valve with a 5" motor operated gate valve in it for control. These six valves are normally operated electrically from the control room, but local pushbutton stations are mounted with the starters outside the shielding wall.

These valves meet and exceed the standard of the ASA code for Pressure Piping for wall thickness. The valve bodies are stainless steel castings to ASTM-A351 Grade CF8M. All castings are fully radiographed after heat treatment to comply with ASTM-E71 standards for Class 2 castings, and are liquid penetrant tested. After assembly, the valves were hydrostatically tested at 1.5 times design pressure.

The joint between body and bonnet on the valves is designed for seal welding, if, for any reason, leakage from the joint becomes excessive. The steam seal is a stuffing box with two sets of packing with a leakoff to waste in between.

If a recirculating pump were started with the 5" bypass valve open and cold water in the pipe, the flow rate through the bypass valve would be approximately 160 lb/sec, as compared to the normal single loop recirculation flow rate of 1620 lb/sec. This bypass flow rate would empty the cold leg in approximately 75 seconds if it is assumed that 1) the cold leg consists of the volume of water between the recirculation pump suction valve and the reactor vessel nozzle; 2) this water is at a temperature of 100°F; and 3) the reactor is at 50% power when the cold loop is brought into service. The net reactivity insertion from this cold water addition would be approximately five cents. The resultant power would peak at about 75% in about 40 seconds after the bypass valve was opened, and power would settle out at about 53% (Reference 1, Section E.5).

5.4.3.1.3 Operating Requirements

Administrative controls are established to require that all suction and discharge valves be either fully open or fully closed, never throttled.

Current Technical Specifications require a minimum of one reactor recirculating loop to be used during all reactor power operations (ie, recirculating pump suction valve and 20" discharge valve shall remain open and pump shall be running). The maximum operating pressure and temperature shall be the same as the reactor vessel. The controlled rate of change of temperature in the reactor vessel shall be limited to 100°F per hour. All other components in the system shall be capable of following this temperature change rate.

Protection against a "cold-water accident" is provided by recirculation pumps and valve interlocking. The valves on either side of the recirculating pumps are interlocked with pump power such that each valve must be in its proper position before the pump motor can be started. If the suction valve to the pump is closed, the motor will be tripped. If the discharge valve and bypass valve are closed, the motor will be tripped.

The Technical Specifications require that on approximately 10% of full simultaneous closure of both discharge or both suction valves, or any combination of both of these valves, one in each loop during power operation will initiate reactor scram.

In addition to the above valves, two 20" butterfly valves, one on the discharge of each recirculating pump were installed to regulate pump discharge during the Research and Development Program period. Information on the previous operation of these valves can be obtained from Reference 1.

These motor operated butterfly valves have been electrically disabled and locked in the full open position. CPCo provided an analysis of inadvertent valve closure due to mechanical failure of the locked open valves in a Technical Specification Change Request dated June 15, 1973. The change was approved in Technical Specification Change Number 38 dated July 27, 1973.

Design and construction of these valves are to the same standards as the gate valves discussed above. The valve bodies are cast stainless steel ASTM-A351, Grade CF8.

5.4.4 NUCLEAR STEAM SUPPLY SYSTEM

The nuclear steam supply system is composed of the equipment in the main recirculation loop, plus those auxiliary systems required to provide a safe and operable system.

The purpose of the Nuclear Steam Supply System is to circulate water to remove heat from the core and provide a path for steam/water mixtures to rise to the steam drum.

A partial listing of the various systems which interface with the NSSS are as follows:

Main Steam System (MSS)
Primary Coolant System (PCS)
Condensate System (CDS)
Feedwater System (FWS)
Control Rod Drive System (CRD)
Reactor Vessel General (RVG)
Reactor Vessel Internals (RVI)
Steam Drum System (RSD)

Auxiliary Systems involved

Shutdown Cooling System (SCS)
Liquid Poison System (LPS)
Reactor Cleanup System (RCS)

Engineered Safety Feature Systems involved: Described in Chapter 6

Emergency Core Cooling System (ECCS)
Emergency Condenser System (ECS)
Post Incident System (PIS)
Reactor Depressurization System (RDS)

Information on each of these systems is provided in other sections of this Updated FHSR.

5.4.4.1 NSSS Stress Analysis (Reference 1 and 24)

Tab "M", response to question 13, of Reference 1 provided a brief outline of the NSSS Stress basis and is included below:

The stress analysis conducted on the Consumers Power Nuclear Steam Supply Piping System utilized a Structures Computer Program. It was performed by the Atomic Power Equipment Department of the General Electric Company. This program, called "Fast," analyzes statically indeterminate structures which may be divided into a series of straight and curved bars. Since the program may handle up to four hundred members at one time, the whole piping system including risers, drum, downcomers, and recirculating loops was analyzed in one problem.

The "Fast" Program uses as its basis Castigliano's Theorem. Coupled with the use of this principle are the general assumptions of small deflections, linear elasticity, and plane sections remaining plane. The program does not compute and sum "shape Coefficients" as commonly done in the Kellogg Flexibility Analysis outlined in the book, Design of Piping Systems, by the M. W. Kellogg Company. Rather, the program computes influence coefficients for each member (loads that cause unit deflections) and couples the members with the aid of matrix algebra. The advantage of the latter method is that the program may be used for many general structural configurations and not just piping systems.

The strain energy due to bending, torsion, and shear is computed for each member. The stress computation for pipes is computed according to the procedure outlined in the ASA B31.1-1955 Code for Pressure Piping, utilizing all of the proper flexibility and stress intensification factors for curved members.

Because of the complexity of this piping system, all weight loading was considered in addition to the thermal expansions and all restraints.

To insure meeting the ASA Code requirements the following three cases were analyzed:

a. System Cold

All weights, hanger loads, and restraints in the cold condition were applied to determine if the allowable stress, S_c, in the cold condition was exceeded.

The maximum computed stress in the riser system is 3750 psi.

S_c = 18,750 psi.

The maximum computed stress in the downcomers and recirculating loops is 7960 psi. S_c = 17,500 psi.

b. System Hot With No Thermal Expansion

Stresses resulting from weight, hanger loads, and restraints in the hot condition were added to the longitudinal pressure stress to determine if the allowable stress, S_h, in the hot condition was exceeded.

The maximum computed stress in the riser system is 13,000 psi.

S_h = 17,100 psi.

The maximum computed stress in downcomers and recirculating loops is 10,460 psi. S_h = 15,350 psi.

c. System Hot With Thermal Expansion (Operating Condition)

Stresses resulting from weight, hanger loads, and restraints in the hot condition, and from thermal expansion were added to the longitudinal pressure stress to determine if the allowable stress, 1.25 (S_c = S_h), was exceeded. It was assumed that the total number of full temperature cycles over the life of the system was less than seven thousand.

The maximum computed stress in the riser system is 13,230 psi. Allowable stress = 44,810 psi.

The maximum computed stress in the downcomers and recirculating loops is 10,115 psi. Allowable stress = 41,060 psi.

Hot condition calculations were made for the 75 Mw power level conditions with a 530°F temperature change and a system pressure of 1600 psi.

5.4.4.2 NSSS Risers and Downcomers Details (Reference 1 and 24)

The risers and downcomers in the nuclear steam supply system have been designed to have equivalent hydraulic lengths under operating conditions. Reference 1 and 24 contained a complete set of drawings which shows piping arrangement, the suspension system, vibration dampeners, and system anchor points.

141F788 - Recirculating Piping
198E180 - Downcomer & Riser Vibration Dampeners
198E182 - Suspension System Details
198E136 - Pump and Loop Suspension System
198E181 - Pump and Loop Vibration Dampeners
198E135 - Steam Drum Suspension System
198E909 - Risers

5.4.4.3 NSSS Shielding

The steam drum, riser and downcomers are primarily shielded by ordinary concrete walls which shall vary in thickness from 4 feet, 9 inches near the bottom to 3 feet, 3 inches at the top. A large section, 12 feet by 42 feet, of the steam drum enclosure wall serves as a blowout panel and shall contain high density, loose aggregate to a thickness of approximately 4 feet, 9 inches. This provides the shielding equivalent to 4 feet, 9 inches of ordinary concrete.

Blowout Panel

In designing the shielding around the nuclear steam supply system, it was necessary to restrict openings to such an extent that the pressure immediately following a rupture accident could cause the shield walls around the recirculating loop to collapse. To prevent this, a blowout panel has been provided to relieve such a sudden pressure buildup. The blowout panel consists of high density sand and gravel confined in place by light gauge metal siding. If the panel were blown free, it would separate into six sub-panels, each about 90 square feet in area and 4 1/2 feet thick. The gravel would no longer be confined at the edges, and each unit would immediately begin to fall apart. The trajectory would bring the panels down onto the floor directly over the reactor. From there some of the gravel might roll or bounce far enough to reach the shell of the sphere, but the maximum size missile would be 1 1/2 inch and would not have sufficient energy to be damaging. The blowout panel is held in place sufficiently that it would not be dislodged by blowing of the steam drum safety valves.

The entire Reactor Depressurization blowdown of 3.2×10^6 Lb/hr flow was not capable of effecting the pressure differential ($\simeq$ 2 psig) required to fail (blowout) the panels. (Reference Bechtel Power Corporation letter dated January 24, 1974 and attached calculations.)

5.4.4.4 Specific Activities of Coolant

During plant operation, the specific activity of the reactor coolant loop is predominately due to the equilibrium Nitrogen-16.

Average activities in the various regions of the external coolant loop based on plant operation at 240 Mwt, and based on the 6.5 Mev gamma energy of N-16 are tabulated below:

Primary Coolant System Equipment

	gammas/cm^3 - sec
Risers	1.6×10^6
Steam Drum	9.4×10^5 (water only - uncorrected for voids)
Reactor Vessel	2.5×10^6
Steam at Outlet Nozzle	7.8×10^4
Downcomers	5.6×10^5
Recirculating Pump	4.7×10^5
Pump Discharge Piping	3.8×10^5

The specific activity of system leakage and in the coolant loop after shutdown varies widely, depending upon the decay time and relative quantities of corrosion products and fission product leakage into the water. The activity is expected to range from 0.02 uc/ml to a high of 6 uc/ml shortly after shutdown; the basis of 6 uc/ml is an assumption of 1000 leaking fuel rods.

5.4.5 RESIDUAL HEAT REMOVAL (RHR) - SHUTDOWN COOLING SYSTEM (SCS)

At Big Rock Point, the Residual Heat Removal System is known as the Shutdown Cooling System (SCS) and the system is located inside containment. The SCS removes decay heat from the reactor by taking suction from the reactor vessel unloading nozzle near the top of the vessel through two cooling loops containing single pass heat exchangers. From the heat exchangers, the water flows to the shutdown pumps (one per loop) and is pumped to the discharge piping of the Number 1 Recirculating Pump.

Design Characteristics

Design Pressure, Psia	315
Design Temperature, °F	425
Number of Pumps	2
Number Heat Exchangers	2
Heat Removal Capacity per Loop, Btu/h	20×10^6

Operating Characteristics

The procedure following reactor shutdown is to cool the nuclear steam supply system to allow lowering the water level to below the reactor vessel flange in order that the head may be removed for refueling.

The rate of system cooling is limited to 100°F/hr. Drum water level is maintained and the reactor recirculating pumps are kept running to allow uniform cooling of the system.

The initial phase of cooling consists of bleeding steam into the main condenser via the main steam bypass valve. Refer to drawings 0740G40106 and 0740G40107. The valve is throttled by a remote manual control to maintain the proper cooling rate. Metal temperatures on the steam drum and reactor vessel, as well as system water temperature and pressure, are recorded in the control room for operation information.

When system pressure has dropped to 150 psig, the turbine seals become ineffective and it is necessary to close the bypass valve. The cooling is continued by recirculating reactor water through the shutdown cooling system. Changeover in the cooling operation from that through the bypass valve to that through the shutdown coolers may be accomplished between 300 psig and 150 psig.

The system will normally be placed in service when the primary loop pressure is about 200 psig and above 150 psig to have the system in operation before the turbine air ejectors become ineffective and the main heat sink is lost.

The shutdown cooling system consists of two vertical 500 gpm pumps and two heat exchangers (See Drawing 0740G40107). Normal power, and emergency power from the diesel generator, are available. Although a single system of one pump and one exchanger will safely remove core decay heat, two pumps and two exchangers are provided to assure system cooling reliability, and to provide for rapid cool-down of the system at the 100°F per hour rate to allow refueling to commence as soon as possible. The system is cooled to 212°F within about four hours after shutdown, at which time the vapor pressure in the reactor will be down to atmospheric and the reactor head bolts can be removed. Within eight hours after shutdown, the system is cooled to 120°F, and this temperature can be maintained by a single pump and exchanger.

As the shutdown system is rated at a lower pressure than the reactor, it is isolated during normal operation by double block valves at both the inlet to the reactor vessel and return connections to the recirculation lines. A bleed-off orifice with a pressure switch between each pair of valves provides indication of leakage.

The block valves are motor operated with pressure interlocks to prevent inadvertent opening while the reactor pressure is at 300 psig or higher.

Controls for both shutdown pumps and block valve control switches are located in the control room. In addition, a remote manual flow control valve is provided in the shutdown heat exchanger cooling water discharge which can be regulated to control the cooling rate. Thus, the entire cooling down operation can be carried out from the control room, however, it may be necessary to locally valve off the fuel pit heat exchangers during initial shutdown to route more cooling water flow through the exchangers when both are in operation.

System Modifications

- The SCS heat exchanger outlet valve leak detection pressure switches were replaced with switches with a 1500 psig rating via Facility Change FC-351.

- Tell-Tales were added on the Reactor Cooling Water side of the SCS heat exchangers to provide a method of determining specific relief valve leakage. Refer to FC-590.

5.4.5.1 Operating Requirements (Reference BRP Technical Specifications)

Power Operation

The shutdown cooling system shall be ready for service during power operations with the 480 volt circuit breakers for isolation valves

MO-7056, MO-7057, MO-7058, and MO-7059 checked "open" when reactor pressure is above 300 psig.

Refueling Operation

The shutdown cooling system shall be operable and ready for service during refueling operations and the breakers for MO-7070 and MO-7071 shall be tagged "open".

Extended Shutdown

The reactor shutdown cooling system shall be placed in operation whenever reactor pressure drops below a pressure sufficient to maintain turbine seals. This system will complete the cooling of the reactor water to 125°F.

5.4.5.2 Shutdown Cooling System Heat Exchanger Tube Failure

Systematic Evaluation Program (SEP) Topic V-10.A - Residual Heat Removal System Heat Exchanger Tube Failure was applicable for the BRP SCS. The NRC provided a revised safety assessment on this SEP topic by letter dated October 9, 1979.

The safety objective of this review is to assure that impurities from the cooling water system are not introduced into the primary coolant in the event of shutdown cooling system heat exchanger tube failure. This was expanded to assure that adequate monitoring exists to assure no leakage of radioactive material in the other direction - into the service water and thus to the environment.

Because the shutdown cooling system (SCS) heat exchangers are on the suction side of the shutdown cooling pumps, the primary system may be contaminated by a leak from the shutdown cooling system during cooling system operation and the primary system may leak into the cooling system during reactor operation.

The NRC revised Safety Assessment identified two concerns and provided two recommendations for resolution of this topic.

CPCo responded to these concerns by letter dated January 13, 1983 and resolution of the concerns was addressed by the NRC in NUREG-0828 (Reference 29), Section 4.17 as follows:

- As protection against undetected leakage into the primary system, the Big Rock Point RCW system water tank incorporates a low level alarm which will alert the plant operators to leakage through the SCS heat exchangers (or any of the other components cooled by the RCW system) when the RCW is in operation. In addition, the RCW system incorporates a radiation detector and alarm, as does the service water system which cools the RCW heat exchangers.

- The RCW system pressure at the two RCW heat exchangers varies from a few inches of water vacuum to a few pounds per square inch gage. Because the service water pressure at these heat exchangers varies from approximately 20 to 45 psig, the possibility exists for inleakage of contaminants from Lake Michigan into the RCW system. As noted above, such inleakage could find its way into the primary coolant system during SCS operation because of the differential pressures across the SCS heat exchanger. Although this scenario presumes failures of tubing in a combination of the SCS and RCW heat exchangers, such a combination, with resultant primary system contamination, cannot be ruled out, given that no inservice inspection of heat exchanger tubes has been performed and that differential pressures would aid such leakage. Big Rock Point procedures require twice weekly analysis of the RCW system and testing for dilution of chromates (a compound that is used in the RCW system as a corrosion inhibitor) and conductivity. These tests may detect inleakage from the service water system, but added defense and early warning could be obtained by the incorporation of a high level alarm in the RCW system water tank.

- Currently, only the low level alarm exists as protection in addition to the twice weekly sampling and operating procedures that require the level to be logged every shift on the control room log sheet.

- As defense against primary system contamination during power operation, Big Rock Point Technical Specification requires daily primary coolant sampling, which includes chlorides and conductivity.

- The limited Probabilistic Risk Assessment (PRA) for Big Rock Point rated this issue to be of low risk significance because of the need to fail two heat exchangers and because the relatively low corrosion rates limit the analysis sensitivity to sampling rates greater than the present twice weekly rate of the Technical Specifications.

- As noted under SEP Topics V-12.A (Section 4.18) and VI-1 (Section 4.19) of NUREG-0828, the staff has found the present chloride limits acceptable.

- Conclusion

  The periods of plant shutdown are relatively short. Any impurities that might develop in the primary coolant would be detected following plant startup, and the appropriate corrective action taken before any long-term degradation effects might begin. Therefore, the staff concludes that no further action is necessary.

5.4.5.3 Residual Heat Removal (RHR) - Shutdown Cooling System (SCS) Reliability, Isolation, and Interlock Requirements

Systematic Evaluation Program (SEP) Topics evaluated and applicable to BRP SCS are identified below:

- V-10.B - Residual Heat Removal System Reliability.
- V-11.A - Requirements for Isolation of High and Low Pressure Systems.
- V-11.B - Residual Heat Removal System Interlock Requirements.

5.4.5.3.1 SCS Reliability

The NRC Final Evaluation for Topic V-10.B (System Reliability) was provided by letter dated September 10, 1982, (Reference 30). The safety objective for this topic was to ensure reliable plant shutdown capability using "Safety Grade" equipment, (as defined in NUREG-0138). BRP systems were compared to the NRC Standard Review Plan (SRP) criteria and the SCS does not satisfy the review guidelines. However, the staff concluded that other systems at BRP fulfill the required safety functions and objectives.

5.4.5.3.2 SCS Isolation of High and Low Pressure Systems and Interlock Requirements

The NRC staff's revised Final Safety Evaluation Report on Topics V-11.A (Isolation) and V-11.B (Interlocks) were addressed in the December 15, 1982 letter, (Reference 31), as follows:

Evaluation

The SCS isolation valves do not have diverse and independent interlocks to prevent operation when PCS pressure exceeds SCS design pressure. However, redundant pressure switches (two in series) and relays (two in parallel) are provided. Relay contacts are segregated with one relay controlling the inboard valves and the other the outboard valves. Because the relays share a common power supply, they are not independent and could, theoretically, be damaged by an overvoltage or underfrequency transient. SEP Topic VIII-1.A examined such transients and found that suitable protection was provided at this plant. The system is designed to close the valves on high pressure (either switch opens) or loss of instrument power. All four valves are powered from the same bus. Power is locked out during power operations. Should the valve power fail during SCS operation the only source of pressure available is decay heat. The pressure buildup could be controlled by the use of the pressure relief system and the core spray system if the valves could not be closed manually in sufficient time.

Topics V-10.B, "RHR Reliability" and VII-3, "Systems Required for Safe Shutdown" also have examined the subject of SCS Interlocks and found them to be acceptable.

Conclusions

The staff does not recommend modification of the SCS because the present design satisfies the single failure criterion and further modifications to provide diversity or redundancy will not provide a significant improvement in the protection of the public health and safety.

5.4.6 REACTOR CLEANUP SYSTEM (RCS)

Description

The RCS is designed to maintain the required chemistry and purity of the Nuclear Steam Supply System.

The reactor cleanup system removes the corrosion products originating in the feedwater heaters, reactor vessel, recirculating loop, piping and shutdown system equipment to maintain reactor water purity at or below 0.5 ppm dissolved solids with total solids at or below 2 ppm. Excess water removal from the primary system is also accomplished by the cleanup system by providing a means to blow down the primary system.

The RCS is shown on Drawing 0740G40107.

Primary system water from the reactor is piped from the discharge of the No 2 recirculating pump to the regenerative heat exchangers. The water flows through the tubes of the regenerative heat exchangers where it is cooled from the reactor system operating temperature to an intermediate temperature of approximately 210°F by counter-current exchange with low temperature treated coolant water being returned to the reactor. The water then flows from the regenerative heat exchanger through a non-regenerative heat exchanger for further cooling to about 110°F by exchange with reactor demineralized cooling water in the non-regenerative heat exchanger.

From the outlet of the non-regenerative heat exchanger, the water flows to the suction of cleanup pump or through the pump bypass line. The water is then either pumped or allowed to flow via the bypass to the cleanup demineralizer.

Treated coolant leaving the demineralizer passes through the shell side of the regenerative heat exchangers, is reheated and re-enters the reactor from the suction of both recirculating pumps at about 485°F. This flow may also be routed to the discharge to the No 1 recirculating pump.

A remote manual flow control valve is supplied on the cleanup pump discharge to permit adjustment of the cleanup system flow rate. Bypass line flow may be throttled via a manual globe valve in-line.

As an alternate, reactor water can be made up with clean condensate while drawing off impure water to the radwaste demineralizer.

Spent resins from the demineralizer are not regenerated, but are discharged to the radioactive waste system resin disposal tank for storage prior to ultimate disposal. Fresh resins are added by sluicing from the condensate demineralizer regeneration facility into the cleanup demineralizer. After May 1970, condensate demineralizer resins were no longer regenerated to eliminate "spikes" noted in water chemistry. However, it is possible to regenerate cleanup demineralizer resins during initial startup operations by routing them to the condensate demineralizer regeneration system. This would be done only until the resins become too radioactive to be handled in the regeneration system.

5.4.6.1 Design Information

Cleanup Demineralizer

The cleanup demineralizer is designed to treat 95 gpm of reactor water at 110°F maximum. Resin Bed size is 24 cubic foot of mixed cation and anion exchange resins. The design of the vessel was to the 1959 ASME Code and Code Case 1270-N for a safety factor of four. Design Pressure is 1700 psig internal pressure at 110°F. The vessel is Unfired Pressure Vessel Code stamped and is the vertical type with 2 1/8" A-212 Grade "B" Shell and 2" A-212 Grade "B" heads. Original Hydrostatic Test Pressure was 2550 psig. The vessel is 42" diameter O.D. and about five feet long with an 18" manhole in the top.

Maximum influent solids content for design purposes is 1.0 ppm. Normal influent solids content will be less than 0.5 ppm.

The head loss through the cleanup demineralizer resin bed just prior to exhaustion will not exceed 10 psi at 95 gpm flow rate.

The design considered reactor water delivered to the demineralizer at 1485 psig and 110°F.

Operating weight of the unit is about 15,850 pounds.

The vessel is lined with 1/8" of 304 stainless cladding.

Reactor Cleanup System Resin Replacement Water Purity

RCS resin replacement was addressed in Systematic Evaluation Program Topic V-12.A, Water Purity of BWR Primary Coolant. In response to the NRC October 9, 1979 revised safety assessment on this topic, CPCo letter dated June 14, 1983 provided the following:

Resin Replacement Criteria

1. Number of days inservice on a resin bed.

2. Expected future operation of the plant - this may result in replacement in advance of that required by the other criteria and allows scheduling of the resin bed replacement when it is least likely to interfere with other planned activities.

3. Differential pressure across each bed - retention of crud (corrosion products) on the resin beds will cause an increase in differential pressure and necessitate resin replacement in advance of that dictated by resin exhaustion alone.

4. Resin bed effluent conductivity.

Plant experience has shown resin replacement normally occurs because of 1, 2, or 3 above (or a combination thereof) instead of 4.

The volume of resin in the cleanup demineralizer is 24 cubic feet. The resin anion and cation ratio is 1.1 and the demineralizer is operated at a flow rate of approximately 60 gallons per minute. Influent and effluent conductivity measurements are normally 0.1 and 0.08 umhos/cm, respectively at the end of an approximate 12-month run. The resins are replaced a week or two prior to a scheduled refueling shutdown to prevent release of accumulated crud in the demineralizer to the reactor coolant system during pressure reduction.

Calculations indicate that the cleanup demineralizer would function for approximately 2888 days with an influent conductivity of 0.1 umho/cm or 288 days with an influent conductivity of 1.0 umho/cm before complete anion exhaustion would occur. Such calculations are based on: 1) the conditions described in the preceding paragraph; 2) a reactor water coolant pH of 7.0; 3) a background conductivity of 0.056 umho/cm; 4) an anion exchange capacity of 0.8 milli-equivalence per liter (meq/l) (the resin manufacturer's certificate of compliance requires that the exchange capacity is greater than 1.0 meq/l), and a 75.5 mho per centimeter per gram equivalent per cm^3 for chlorine. Seventy percent exhaustion of the anion resin would occur in 2020 and 200 days with influent and effluent of 0.1 umhos/cm and 1.0 umhos/cm, respectively. An investigation would commence before ever reaching 1.0 umhos/cm due to the intrusion of organics into the primary system.

Given that the lake water contains only approximately 10 ppm chloride, chances for a gross and undetected chloride intrusion in the reactor vessel water are small.

Heat Exchangers

The non-regenerative and regenerative heat exchangers were replaced in 1972 due to minor tube leakage in the regenerative heat exchangers and excessive fuel crudding of which the original copper-nickel tubed heat exchangers were considered to be the source. The replacement was accomplished via General Work Order 8485 utilizing stainless steel tubes and construction equivalent to the 1959 ASME Unfired Pressure Vessel Code and Code Case 1270-N. The heat exchangers were designed under ASME Code (1968) Section IIIC with the additional more severe requirements of stress relief of all welds in the carbon steel material and ultrasonic test of the metal under nozzle welds as required in the original design for 1959 Section VIII ASME Code Case 1270-N.

The Heat Exchangers are ASME Unfired Pressure Vessel Code Stamped type BEU Vertical with the following design ratings:

Heat Exchanger		Pressure PSIG	Temperature Degrees F	Original Hydro Test PSIG
Regenerative	Tubes	1700	600	2550
	Shell	1700	600	2550
Non-Regenerative	Tubes	1700	600	2550
	Shell	75	200	113

The RCS Piping Design Rating is 1700 Psi at 625°F.

Cleanup Pump and Bypass

The cleanup pump was designed to pump 95 gpm of water at 130 feet dynamic head and for 1700 psig and 215°F. The pump was installed to overcome friction losses of piping and equipment.

A bypass around the pump was installed via Facility Change FC-598 permitting operation of the cleanup system from the differential pressure across the recirculating pumps. This modification was discussed in NUREG 0828 Section 5.3.9.1 (Reference 29) and in the Integrated Plan as Issue 48A. Because of the bypass installation the cleanup pump is normally used only during shutdown.

Testing operation with the recirculating pumps on, and cleanup pump off, in April 1983 indicated that the Differential Pressure across the cleanup demineralizer did not change significantly with cleanup flow rates with the cleanup pump operating.

The bypass installation was such that temperature protection of the cleanup resins is still provided and isolation capability of the cleanup loop remains.

A pressure switch prevents opening of the resin charging line before relieving the pressure in the demineralizer vessel.

Cleanup system flow is recorded and totalized and is normally limited to a maximum of 75 gpm to prevent erosion of the heat exchangers.

5.4.7 REACTOR COOLANT SYSTEM HIGH POINT VENTS

In response to NUREG-0737, Clarification of TMI Action Plan Requirements II.B.1 Reactor Coolant System Vents, and the September 13, 1979 NRC Implementation Orders for NUREG 0578 TMI Lessons Learned Task Force Status Report and Short Term Recommendations, BRP proceeded with Facility Change FC-490, High Point Vent Modification of the Primary Coolant System. The project was initiated to install Reactor Coolant System vents on the tube side of the Emergency Condenser. The piping system connection was completed under Specification Field Change SFC-81-008.

The modification was never completed and the FC was closed out. Exemption from the requirement for High Point Vents was granted by the NRC via a July 17, 1985 letter in response to CPCo request dated April 19, 1983. The Exemption was from the subsequent 10CFR50.44(c)(3)(iii) Reactor Coolant System High Point Vents Rule.

The basis for the exemption as briefly provided in the NRC Integrated Assessment, (NUREG-0828, Section 5.3.26 Reference 29):

1. The RDS could be used to vent the pressure vessel (via the main steam lines)

2. The likelihood of core uncovery (which is necessary to generate hydrogen) is very small.

3. The cost is too high.

On the basis of these considerations, the staff concludes that further modifications to place the system in operation are not warranted. However, suitable test connections and seismic supports should be provided or the valves should be removed.

The valves were subsequently removed, refer to FC-490 for close out actions taken.

5.4.8 PRIMARY COOLANT WATER PURITY AND LIMITS

The NRC provided evaluations for Systematic Evaluation Program (SEP) Topic V-12.A, Water Purity of BWR Primary Coolant by letters dated August 9, 1979 (Draft Evaluation), October 9, 1979 (Revised Safety Assessment including CPCo September 13, 1979 comments on the Draft Evaluation), and in NUREG 0828 Integrated Assessment Section 4.18 in 1984, (Reference 29). CPCo responded to the recommendations made in the above letters on February 28, 1983 and June 14, 1983.

The Topic V-12.A review was to verify BRP compliance with Regulatory Guide 1.56, Revision 1, "For Comment," 1978 version entitled "Maintenance of Water Purity in BWRs." 10CFR50, Appendix A, General Design Criteria 14, as implemented by Regulatory Guide 1.56, requires that the reactor coolant pressure boundary (RCPB) have minimal probability of rapidly propagating failure. This includes corrosion-induced failures from impurities in the reactor coolant system. The safety objective of this review is to ensure that the plant reactor coolant chemistry is adequately controlled to minimize the possibility of corrosion-induced failures. The staff's review of this topic identified the following two issues. (NUREG 0828 Section 4.18, Reference 29).

Water Chemistry Limits

The Big Rock Point Technical Specifications did not meet the limits established in Regulatory Guide 1.56 for conductivity, chlorides, and pH of the reactor vessel water and conductivity of the feedwater system. On the basis of BRP past operating experience, the staff has concluded that these differences are not significant.

Limiting Conditions for Operation

The topic evaluation concluded that the requirements of the plant operating procedures that govern 1) the sampling of the reactor water cleanup (RCS) system demineralizer in service and any subsequent shifting of flow and; 2) the measurement of flow every four hours through each condensate demineralizer in service and the daily calculation of unused capacity of each bed are not incorporated into the plant Technical Specifications. These requirements are desirable to avoid corrosion-induced failures in case of a condenser tube rupture.

The topic evaluation recommended that BRP provide new water chemistry limits and new limiting conditions for operation of the RCS system and condensate demineralizers unless it can be demonstrated that such changes are not necessary.

CPCo responded in a letter dated June 14, 1983 and maintains that 20 years of operating experience at Big Rock Point (which includes condenser tube failures) and the ongoing inservice inspection (ISI) program have demonstrated the adequacy of the existing limits and Technical Specifications. On the basis of this experience, the staff concludes that the licensee's existing procedures are adequate and incorporating these procedures into the Technical Specifications is not warranted.

5.4.8.1 Primary Coolant Limits (Reference BRP Technical Specifications)

The primary coolant shall be sampled and analyzed daily during periods of power operation. The following are absolute limits which if exceeded shall necessitate reactor shutdown. Corrective action will necessarily be taken at more stringent limits to minimize the possibility of the absolute limits ever being reached.

Conductivity (Micromho/cm)	
Maximum	5
Maximum Transient*	10
pH (Lower and Upper L)	4.0 and 10.0
Chloride Ion (Ppm)	1.0
Boron (Ppm)	100

* Conductivity is expected to increase temporarily after startups from cold shutdown. The maximum transient value here stated is the maximum permissible and applies only to the period subsequent to a cold shutdown between criticality and 24 hours after reaching 20% rated power.

Isotopic analysis of the primary coolant to determine the dose equivalent I-131 concentration shall be performed at least every 72 hours during periods of operation.

1. If the dose equivalent I-131 concentration exceeds 0.2 µCi/ml and is less than or equal to 4.0 µCi/ml, isotopic analysis to determine dose equivalent I-131 shall be performed every 24 hours until the activity is less than 0.2 µCi/ml.

2. If the dose equivalent I-131 exceeds 4.0 µCi/ml, the plant shall be placed in a shutdown condition with the main steam isolation valve closed within 12 hours.

5.4.9 REACTOR COOLING WATER (RCW) SYSTEM

The reactor cooling water system is a closed intermediate cooling loop utilizing demineralized water to remove heat from the following pieces of equipment:

Reactor Shield Cooling Panels
Reactor Cleanup Non-Regenerative Heat Exchanger
Reactor Shutdown Heat Exchangers
Fuel Pit Cooling Water Heat Exchangers
Miscellaneous Sample Coolers
Reactor Recirculating Pump Coolers

The two reactor cooling water heat exchangers are the equipment by which the removed heat is transferred to the service water system. The return header is monitored to indicate and alarm excess radioactivity buildup. Two full capacity, vertical motor driven pumps, each rated at 1500 gpm, are provided which take suction from the concrete cooling water return tank and recirculate the cooling water through the RCW heat exchangers, each rated at 9×10^6 Btu/h, to the various equipment services listed above.

5.4.9.1 Design Characteristics

The RCW System is shown on Drawing 0740G40111 and consists of a concrete tank of approximately 5000 to 6000 gallon capacity, two full capacity vertical two stage pumps, and two full capacity heat exchangers for heat transfer to the service water system.

Design flow of the shell (inhibited water) side of each RCW heat exchanger is 500,000 lb/hr with reactor cooling water making one pass and then returning to the cooling water tank. On the tube side, service water makes four passes through the heat exchanger at a design flow of 500,000 lb/hr. Heat exchanger design pressure is 100 psig on the tube side and 75 psig on the shell side, both at a design temperature of 180°F. The heat exchangers are Vertical Straight Tube type with removable bundles and are manufactured to TEMA Class "C" requirements.

Operating Characteristics

An emergency water supply to the tube side of the heat exchangers via a manual operated cross-connection with the plant fire system. The valve is located in containment, near the cable penetration area.

The Reactor Cooling Water System is monitored for temperature, pressure, water inventory, and radioactivity levels. The Liquid Process Radiation Monitor Recorders provide trending traces for the early detection and correction of increasing levels of radiation.

In the event that the reactor cooling water system is to be used as part of the Alternate Shutdown System (ASD), the No. 2 reactor cooling water pump motor will be fed by the emergency diesel generator via a 4 conductor cable from a panel located in containment.

A pressure switch on the pump discharge header serves a dual function of alarming on falling pressure in the control room and also starting the other RCW pump if its control switch is in the "standby" position. Controls for the pumps are located on the main control panel. Water level in the tank is monitored by level switches which alarm in the control room. Automatic demineralized water makeup capability exists, however, normally manual makeup is accomplished.

Total loss of RCW System would necessitate reactor shutdown to prevent loss of reactor recirculating pump seals due to high temperature. The most likely cause of total loss of RCW System would be a loss of station power.

5.4.9.3 Loss of RCW to the Reactor Recirculating Pump Seal Coolers

As part of the NUREG-0737, "Clarification of TMI Action Plan Requirement," Item Number II.K.3.25, "Effect of Loss of A-C Power on Pump Seals," CPCo completed a testing program to demonstrate the adequacy of the elastomer now used in the BRP recirculating pump seals to determine the consequences of loss of cooling water to the reactor recirculating pump seal coolers. CPCo letter dated July 9, 1981 and February 5, 1982 provided the results of the test program and the NRC completed a Safety Evaluation on this topic October 27, 1982.

NUREG-0737 Item II.K.3.25 required the pump seals to withstand a complete loss of AC power for at least two hours. Loss of AC power for this case is assumed to be loss of offsite power. As noted above, loss of AC power results in loss of RCW System pumps with resultant loss of RCW flow to the reactor recirculating pump seal coolers. Further, complete loss of AC power also results in loss of Service Water System (SWS) cooling on the tube side of the RCW heat exchangers.

- It should be noted that no credit is taken for the ability to manually supply fire water to the RCW heat exchangers, and if load conditions permit, the ability to connect one reactor cooling water pump to the Emergency Diesel Generator bus in the event of a loss of station power.

Test Results

The results showed that the O-rings maintained a water-tight seal at primary coolant conditions of 1512 (±3) psig and 583 (±2)F for six hours. 580F is normal operating temperature and thus the highest temperature the seals would experience if cooling were lost. Six hours was chosen as the test interval because it allows a conservative amount of time for plant cooldown following the loss of seal cooling.

NRC Safety Evaluation Conclusion (Reference 32)

We have concluded, based on review of CPCo submittals and evaluation of the results of the tests conducted, that the licensee has adequately demonstrated that the seals used in the RCPs will maintain their integrity for periods in excess of the NUREG-0737 criteria for Item II.K.3.25. We therefore find that the licensee's response to this item is acceptable and that no modifications to the seals are necessary.

CHAPTER 5 REFERENCES

1. CPCo letter dated May 3, 1962, Amendment No 10, Addenda To Final Hazards Summary Report, Technical Qualifications Amendment No 8 (CPCo response completed May 1, 1962 to the NRC letter dated April 18, 1962 request for information).

2. CPCo letter dated September 11, 1970, Response to AEC letter dated August 6, 1970 Identification of Sensitized Stainless Steel Components.

3. CPCo letter dated January 12, 1971, Completes CPCo response of September 11, 1970 to AEC letter dated August 6, 1970. (Provides Maximum Stress Levels for Sensitized Stainless Steel Components.)

4. CPCo letter dated June 30, 1982, Response to Generic Letter 81-04, Implementation of NUREG-0313, Rev 1, Technical Report on Material Selection and Processing Guidelines for BWR Coolant Pressure Boundary Piping (Generic Task A-42).

5. NRC letter dated April 26, 1982, Furnace Sensitized Stainless Steel Safe-Ends. (Reopening of Systematic Evaluation Program (SEP) Topic V-4, "Piping and Safe End Integrity").

6. CPCo letter dated May 28, 1982, Furnace Sensitized Stainless Steel Safe Ends. (Response to the April 26, 1982 NRC letter, and additional analysis in response to Generic Letter 81-04 for NUREG-0313 Rev 1.)

7. NRC letter dated December 30, 1982, Systematic Evaluation Program (SEP) Topic V-4, Piping and Safe-End Integrity (NRC final Safety Evaluation.)

8. NRC letter dated April 19, 1984, Inspections of BWR Stainless Steel Piping (Generic Letter 84-11).

9. CPCo letter dated May 25, 1984, Response to Generic Letter 84-11, Inspections of BWR Stainless Steel Piping.

10. NRC letter dated July 25, 1984, Correction to Safety Evaluation of the Implementation of NUREG-0313, Rev 1, for Big Rock Point.

11. CPCo letter dated June 6, 1983, Systematic Evaluation Program (SEP) Topic V-5 "Reactor Coolant Pressure Boundary (RCPB) Leakage Detection" - Revised Evaluation of Plant Leakage Detection Systems.

12. NRC letter dated June 13, 1983, SEP Topic V-5, Reactor Coolant Pressure Boundary Leakage Detection (Final Safety Evaluation).

13. CPCo letter dated June 12, 1978, Reactor Vessel Surveillance Addendum (Revised Response of CPCo July 29, 1977 concerning reactor vessel materials.)

14. Combustion Engineering Book No. 6460R, January 1962, Instruction Manual - Reactor Vessel.

15. CPCo letter dated October 9, 1963, Annual Report of Changes, Tests, and Experiments.

16. CPCo letter dated July 22, 1963, Special Report - Modifications to Core Support System.

17. CPCo letter dated June 25, 1965, Semi-Annual Report of Operation.

18. CPCo letter dated December 25, 1965, Semi-Annual Report of Operations.

19. CPCo letter dated February 2, 1973, Request for Change to the Technical Specifications, Proposed Change No. 36. (Auxiliary Neutron Source Replacement).

20. NRC letter dated April 12, 1984, Technical Specification Changes for Reactor Vessel Pressure/Temperature Limits. Amendment No. 66.

21. CPCo letter dated September 18, 1981, Technical Specification Change Request, Reactor Vessel Criticality, Cooldown, Heatup and Hydrotest Limitations.

22. CPCo letter dated January 29, 1982, Reactor Vessel Criticality, Cooldown, Heatup and Hydrotest Limits.

23. NRC letter dated March 20, 1986, Exemption from 10 CFR Part 50.62(c)(5) Automatic Recirculating Pump Trip.

24. CPCo letter dated March 12, 1975 Response to NRC February 12, 1975 Request for Information. (Provided a copy of Amendment 8 and an excerpt from Amendment 10 to CPCo Application for Construction Permit and Operating License for Docket 50-155, dated May 3, 1962.

25. CPCo letter dated July 22, 1982, NUREG-0737 Item II.D.1, Performance Testing of BWR Relief and Safety Valves.

26. CPCo letter dated November 24, 1987, Licensee Event Report 87-003, Rev 3 - Inoperable Primary System Safety Valves.

27. NRC letter dated January 10, 1985, Completion of NUREG-0737 Item II.D.1 - Safety and Relief Valve Testing. (Final Safety Evaluation).

28. CPCo Special Report Number 5 (SR-5) dated August 15, 1963 Covering Initial Criticality (September 27, 1962 Through April 17, 1963) Thermal Power Level of 157 Megawatts Demonstration Run. Report of Operating Experience.

29. NRC NUREG-0828, May 1984, Integrated Plant Safety Assessment Systematic Evaluation Program - Final Report.

30. NRC letter dated September 10, 1982, BRP-SEP Topic V-10.B, RHR Reliability, V-11.B, RHR Interlock Requirements and VII-3, Systems Required for Safe Shutdown (Final Evaluation).

31. NRC letter dated December 15, 1982, SEP Topics V-11.A, Requirements For Isolation of High and Low Pressure Systems and V-11.B, RHR Interlock Requirements (Revised Final Safety Evaluation Report).

32. NRC letter dated October 27, 1982, Resolution of NUREG-0737 Item II.K.3.25, Effect of Loss of AC Power on Pump Seals (NRC Safety Evaluation).