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VIRGIL C. SUMMER NUCLEAR STATION NUCLEAR STATION NATURAL CIRCULATION EVALUATION PROGRAM REPORT

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1.0 INTRODUCTION

This natural circulation evaluation program has been developed to evaluate the natural circulation flow and the boron mixing capability of the Virgil C. Summer Nuclear Station relative to the requirements of Branch Technical Position RSB 5-1, Design Requirements for Decay Heat Removal Systems (Reference 1). The program will perform a systems comparison and evaluate the boron mixing capability of the Virgil C. Summer Nuclear Station under natural circulation conditions. To address the case of a single failure in the safety grade systems, an evaluation of natural circulation capabilities under single failure conditions will also be performed.

The Virgil C. Summer Nuclear Station natural circulation evaluation program consists of the following five program activities:

1) A general comparison between the Virgil C. Summer Nuclear Station and Diablo Canyon Unit 1 of the plant systems and equipment that affect natural circulation and boron mixing.

2) A qualitative natural circulation comparison between the Virgil C. Summer Nuclear Station and Diablo Canyon for items which may affect natural circulation flow.

3) A single failure evaluation of the Virgil C. Summer Nuclear Station natural circulation capabilities.

4) A comparison of the Virgil C. Summer Nuclear Station natural circulation testing to other plants, based on low power testing results.

5) A boron mixing calculation to demonstrate the capability of the Virgil C. Summer Nuclear Station to achieve a boration delta similar to the Diablo Canyon test results.

2.0 SYSTEM COMPARISON AND EVALUATION

This section of the report describes and compares the natural circulation and boron mixing capabilities of the Virgil C. Summer Nuclear Station with those of Diablo Canyon Unit 1, as identified in the Diablo Canyon final test report. The final report for the Diablo Canyon natural circulation test is provided in the Diablo Canyon Units 1 and 2 Natural Circulation/Boron Mixing/Cooldown Test Final Post Test Report (Reference 2).

Subsection 2.1 provides a general comparison between the system design features of the Virgil C. Summer Nuclear Station and Diablo Canyon Unit 1.

Subsection 2.2 provides an evaluation of the applicability of the Diablo Canyon test results to the Virgil C. Summer Nuclear Station.

Subsection 2.3 provides a single failure evaluation of the Virgil C. Summer Nuclear Station natural circulation capabilities.

2.1 Comparison of Diablo Canyon with Virgil C. Summer Nuclear Station

This subsection compares the systems and equipment that affect natural circulation and boron mixing of the Virgil C. Summer Nuclear Station to those of Diablo Canyon Unit 1 in sufficient detail to evaluate systems capabilities.

Reactor Coolant System

The general configuration of the piping and components in the reactor coolant loop is the same in both the Virgil C. Summer Nuclear Station and Diablo Canyon. The Virgil C. Summer Nuclear Station has three heat transfer loops, while Diablo Canyon has four loops for heat transfer. Each heat transfer loop contains a steam generator and a reactor coolant pump (RCP). Both plants have the same model reactor coolant pumps (Model 93A). Also, one loop at each plant is equipped with a pressurizer. The Virgil C. Summer Nuclear Station uses a Model D3-1 steam generator while Diablo Canyon uses a Model 51 steam generator.

With respect to natural circulation conditions, the Model D3-1 steam generator has a shorter tube bundle elevation and incorporates a preheater in the lower tube bundle region in comparison to the Model 51 steam generator. The only steam generator design difference between the two plants that may influence the actual natural circulation flow rates is the variance in the steam generator tube bundle elevation. An applicability study is performed in Section 2.2 to show that the SG tube bundle elevation differences have minimal effect on the natural circulation flow capabilities for the Virgil C. Summer Nuclear Station.

Pressure control is available at both Diablo Canyon and the Virgil C. Summer Nuclear Station using the normal pressurizer spray valves or the pressurizer auxiliary spray systems. If both the normal and auxiliary spray valves are unavailable, the pressurizer PORVs are available at each plant for RCS depressurization. At the Virgil C. Summer Nuclear Station, the pressurizer spray valves and auxiliary spray valves are not safety grade. The pressurizer PORVs are available as a backup method for depressurization. The PORVs are powered from separate vital AC electrical power supplies and have two seismic Category I air supply accumulators. These auxiliary air accumulators provide a backup method to depressurize the RCS.

Auxiliary Feedwater System

The auxiliary feedwater systems at both Diablo Canyon and the Virgil C. Summer Nuclear Station are capable of supplying cooling to all steam generators using the auxiliary feedwater pumps during the natural circulation cooldown. The systems will provide water to the SGs from large storage tanks. The condensate storage tank provides this water source at Diablo Canyon, while the Virgil C. Summer Nuclear Station uses the Seismic Category I condensate storage tank which has 150,000 gallons. Additional backup is available from the Seismic Category I service water system.

Main Steam System

The steam generators at both plants have PORVs which are utilized for the plant cooldown. At the Virgil C. Summer Nuclear Station, a safety grade PORV is provided with each steam generator. The PORVs are equipped with handwheels to locally control the cooldown rate, if the remote controls are not available.

Chemical and Volume Control System (CVCS)

Injection of boric acid into the RCS is required to offset xenon decay and the reactivity change which occurs during plant cooldown. The Diablo Canyon natural circulation cooldown test utilized the charging pumps to charge through the boron injection tank (at 20000 ppm boron) in the Safety Injection System. Subsequent charging was aligned from the volume control tank in the CVCS. The boron concentration in the volume control tank was adjusted to 2000 ppm to simulate charging from the refueling water storage tank (RWST).

At the Virgil C. Summer Nuclear Station the boron injection tank has been eliminated. The primary boration source is four weight percent boric acid that is pumped from the safety grade boric acid tanks (at 7000 ppm boron) by the boric acid transfer pumps to the suction of the centrifugal charging pumps. The BAT pumps are powered from different emergency power trains and either pump can provide sufficient boric acid flow. A backup source of boric acid is available from the RWST (at 2000 ppm boron). The borated water is then injected to the RCS via the normal charging line and the RCP seals. As noted in the boron mixing discussion, Section 2.2, the Virgil C. Summer Nuclear Station boration requires a larger quantity of borated water over a longer period of time than the Diablo Canyon test.

To accommodate the borated water addition to the RCS, letdown capability is normally provided by the non-safety grade normal and excess letdown lines to the CVCS. If both the normal and excess letdown lines are unavailable, letdown is provided by the safety grade reactor vessel head vent letdown line to the pressurizer relief tank. Control of the head vent letdown is provided by two redundant parallel safety grade Class 1 motor-operated valves.

Residual Heat Removal (RHR) System

The RHR systems at both Diablo Canyon and the Virgil C. Summer Nuclear Station are low pressure heat removal systems consisting of RHR pumps and heat exchangers. They are designed to lower the temperature of the RCS from 350^{0}F to cold shutdown conditions at a controlled rate. Residual heat is transferred during this stage from the reactor core through the RHR system to the component cooling water circulating through the shell side of the RHR heat exchangers.

Following cooldown to RHR initiation conditions, the RHR system is brought into operation by accessing one or both of the redundant RHR trains. Start-up of the RHR system includes a warmup period during which time reactor coolant flow through the heat exchangers is limited to minimize thermal shock on the RCS components. This flow is regulated by flow control valves downstream of the residual heat exchangers. A bypass line around the residual heat exchanger contains a flow control valve which maintains a constant total return flow to the RCS. Should any of these flow control valves fail, adequate RHR flow can be provided by manual operator control of the RHR pumps.

2.2 Applicability of the Diablo Canyon Test Results to Virgil C. Summer Nuclear Station

Natural Circulation

The Diablo Canyon natural circulation test evaluation verified that RCS natural circulation flow could be established, thereby permitting boron mixing, RCS cooldown and RCS depressurization to RHR system initiation conditions. This phase of the test had no specific acceptance criteria and it was evaluated based on the results of the boron mixing and cooldown/depressurization phases of the natural circulation cooldown test.

The Diablo Canyon test results indicated that natural circulation flow rates were adequate to ensure core decay heat removal, boron mixing and plant cooldown/depressurization capabilities throughout the test. The response of the RCS temperatures indicated stable natural circulation conditions existed throughout the test.

The Virgil C. Summer Nuclear Station and Diablo Canyon Unit 1 have been compared to ascertain any system differences between the two plants that could potentially affect natural circulation flow. The general configuration of the piping and components in each reactor coolant loop is the same in both the Virgil C. Summer Nuclear Station and Diablo Canyon Unit 1. The elevation head represented by these components and the system piping is similar in both plants. Steam generator units (Model 51 for Diablo Canyon vs. D3-1 for the Virgil C. Summer Nuclear Station) were also compared to ascertain any variation that could affect natural circulation capability by changing the effective elevation of the heat sink or the hydraulic resistance seen by the primary coolant. The primary design difference affecting natural circulation flow rates between the two steam generator models is that the Model D3-1 has a shorter tube bundle length that that of the Model 51. The length of the tube bundle region has an effect on the natural circulation driving head established by the system. The longer tube bundle in the Model 51 SG for Diablo Canyon Unit 1 would result in approximately a 10% higher driving head when compared to the Model D3-1 SG installed at the Virgil C. Summer Nuclear Station. This variance in net driving head is relatively small and does not significantly affect the natural circulation flow rate. Therefore, it can be concluded that there are no significant differences in the steam generator units between the two plants which would adversely affect the natural circulation flow characteristics for the Virgil C. Summer Nuclear Station.

To further compare the natural circulation flow capabilities of the Virgil C. Summer Nuclear Station and Diablo Canyon, the hydraulic resistance coefficients of the system piping were also compared. The coefficients were generated on a per loop basis. The hydraulic resistance coefficients applicable to normal flow conditions are shown in Table 2-1. The most signifigant difference is in the reactor core and internals. The resistances as tabulated are based on the flow in each loop.

The general configuration of the reactor core and internals for Diablo Canyon is similar to the Virgil C. Summer Nuclear Station. The remaining variation in the vessel hydraulic resistance coefficient is due to the specific design details of the reactor vessel and internals (i.e., flow area, upper/lower support plate designs, thermal design flow, elevations, etc.).

The flow losses in the reactor coolant loop piping and the steam generators are very similar for the two plants, with the increase in RCS loop piping for the Virgil C. Summer Nuclear Station are compensated for by the decrease in steam generator resistance. This similarity is reflected in the resistance coefficients calculated in Table 2-1.

The coefficients in Table 2-1 represent the resistance in one loop, excluding the resistance through the reactor coolant pump. Since the RCP impeller designs for Diablo Canyon and the Virgil C. Summer Nuclear Station are nearly identical, the flow ratio reported in Table 2-1 would be closer to one if the added RCP flow resistances under full "Locked Rotor" configuration were included. The "Locked Rotor" conditions represents the highest resistance configuration which can result from the RCPs. However, during the RCP coastdown to the "Locked Rotor" configuration, differences in the RCP coastdown characteristics can result in small differences in natural circulation flows between the two units. The overall hydraulic flow coefficient for the Virgil C. Summer Nuclear Station is lower than that of Diablo Canyon, under the full "Locked Rotor" configuration, resulting in an increased natural circulation flow rate capability.

If the effect of the increased natural circulation driving head (10%) for Diablo Canyon Unit 1, RCP differences, and the lower overall piping resistances for the Virgil C. Summer Nuclear Station are factored together, the total hydraulic flow ratio would range from approximately 0.98 to 1.03. Therefore, the natural circulation loop flow rate flow for the Virgil C. Summer Nuclear Station is expected to be nearly the same as for Diablo Canyon Unit 1. The differences in the per loop reactor power and decay heat levels between the two plants are not expected to alter this conclusion, and will, in fact, tend to increase the relative flow for the Virgil C. Summer Nuclear Station.

TABLE 2-1

DIABLO CANYON UNIT 1 VS. VIRGIL C. SUMMER NUCLEAR STATION HYDRAULIC RESISTANCE COEFFICIENTS FOR NORMAL FLOW CONDITIONS

COMPONENT	Diablo Canyon (ft/gpm**2)	Virgil C. Summer Nuclear Station (ft/gpm**2)
Reactor Core & Internals	129.0E-10	89.9E-10
Reactor Inlet & Outlet Nozzles	36.1E-10	33.3E-10
RCS Loop Piping	20.9E-10	26.6E-10
Steam Generators	112.0E-10	102.1E-10
Total Hydraulic Flow Coefficient	298.0E-10	251.9E-10

$$\text{Flow Ratio/loop} = \left[\frac{\text{Total Hydraulic Flow Coefficient for Diablo Canyon}}{\text{Total Hydraulic Flow Coefficient for V. C. Summer Nuclear Station}}\right]^{0.5}$$

$$= 1.087$$

** NOTE **

The Total Hydraulic Flow Coefficients listed above do not include RCP resistances.

Boron Mixing

The Diablo Canyon boron mixing test evaluation demonstrated adequate boron mixing under natural circulation conditions when highly borated water at low temperatures and low flowrates (relative to RCS temperature and flowrate) was injected into the RCS. It also evaluated the time delay associated with boron mixing under these conditions.

The acceptance criterion for this phase of the Diablo Canyon test was that RCS hot legs (loops 1 & 4) indicate that the active portions of the RCS were borated such that the boron concentration had increased by 250 ppm or more.

Boron injection was conducted at the Diablo Canyon test using the 20,000 ppm boron solution contained in the boron injection tank (BIT). The BIT's contents were flushed into the RCS and, within 12 minutes, natural circulation had provided adequate mixing to increase the boron concentration in the RCS by 340 ppm. Following injection, makeup to the Volume Control Tank (VCT) was set to provide 2000 ppm boron. This simulated suction of the charging pumps aligned to the RWST. The charging pump discharge was aligned to provide seal injection flow to each RCP and charging flow to one RCS loop. This alignment was continued throughout the remainder of the test causing the boron concentration to further increase.

For the Virgil C. Summer Nuclear Station, boron will normally be supplied from the 7000 ppm (minimum) boron solution of the boric acid tanks to the suction of the centrifugal charging pumps by the boric acid transfer pumps. Makeup in excess of that needed for boration can be provided from the RWST. The BAT boron concentration (7000 ppm minimum) at the Virgil C. Summer Nuclear Station is less than that used for the successful Diablo Canyon test, therefore, the addition of a larger quantity of borated water over a longer period of time will be required to achieve a similar change in boron concentration. However, because natural circulation flow at the Virgil C. Summer Nuclear Station is expected to be very similar to the flow obtained at Diablo Canyon, adequate mixing of the boron would also be provided for the Virgil C. Summer Nuclear Station. The ability to borate to the required concentration with the BAT will be discussed in the analysis presented in Section 4.0.

Reactor Coolant System Cooldown

The cooldown portion of the Diablo Canyon test demonstrated the capability to cool down the RCS to RHR system initiating conditions at approximately 25 ^{o}F/hour using all four steam generators for natural circulation. The RHR system was then used to cool the RCS to cold shutdown conditions. Plant cooldown was controlled within Technical Specification limits. All active portions of the RCS remained within 100^{o}F of the average core exit temperature. Also, both the steam generators and reactor vessel upper head were cooled to below 450^{o}F when the core exit temperature was 350^{o}F.

For the Virgil C. Summer Nuclear Station, cooldown capability will be similar to Diablo Canyon due to similarities in the design of the RCS, AFW, main steam and RHR systems. Initial plant cooldown will be accomplished via steam release from the main steam power-operated atmospheric relief valves. After RHR system initiation, the RHR system will be used to cool the plant down to cold shutdown temperatures. In terms of the upper head cooldown for the Virgil C. Summer Nuclear Station, the upper head region is expected to cool at a rate signifigantly greater than that of Diablo Canyon Unit 1. The upper head volume for the Virgil C. Summer Nuclear Station is approximately equal to that that for Diablo Canyon. However, the reactor vessel spray nozzle between the downcomer and the upper head region has a flow rate approximately 10 times larger for the Virgil C. Summer Nuclear Station for a given loop flow, thereby allowing better flow communication and mixing in the upper head during natural circulation cooldown. In fact, due to the enhanced flow mixing capability of a T-cold upper head design plant such as the Virgil C. Summer Nuclear Station, a maximum RCS natural circulation cooldown rate of 50^{o}F/hr may be employed under normal conditions. This rate is twice as fast as the recommended natural circulation cooldown rate of 25^{o}F/hr for T-hot upper head design plants such as Diablo Canyon. Also, the Virgil C. Summer Nuclear Station is not required to perform any upper head soak prior to placing the RHR system in service due to the added upper head cooldown capability of the T-cold head design. The ability of the Virgil C. Summer Nuclear Station to achieve cold shutdown while maintaining adequate subcooling in the upper head will be discussed further in Section 4.0.

2.3 Single Failure Evaluation

The V. C. Summer Nuclear Station FSAR (Reference 3) contains a detailed single failure evaluation of the V. C. Summer Nuclear Station natural circulation capabilities. The following key functions are evaluated:

- o Circulation of Reactor Coolant
- o Removal of Residual Heat
- o Boration and Makeup
- o Depressurization

These key functions can be performed during the hot standby period and natural circulation cooldown period. The normal cooldown methods and appropriate backup methods are evaluated, taking credit for local operator action. Based on a review of Reference 3, it is concluded that following a hot standby period, sufficient natural circulation capabilities exist to achieve RHR initiation conditions and then cold shutdown conditions.

3.0 NATURAL CIRCULATION FLOW TEST EVALUATION

This section presents a summary of applicable industry natural circulation tests and discusses the applicability of these tests to the Virgil C. Summer Nuclear Station.

A series of natural circulation tests have been performed by various utilities with Westinghouse designed plants. A typical list of the tests performed by the utilities are:

- Natural Circulation Verification
- Natural Circulation with Loss of Pressurizer Heaters
- Natural Circulation at Reduced Pressure
- Natural Circulation with Loss of Offsite Power
- Effect of Steam Generator Isolation on Natural Circulation
- Simulated Loss of All Onsite and Offsite Power
- Cooldown Capability of the Charging and Letdown System

For the purpose of this report, two of the typically performed tests have been evaluated and summarized. The two tests reviewed were the Natural Circulation Verification Test and the Simulated Loss of All Onsite and Offsite Power. Table 3.1 summarizes these two tests for several Westinghouse designed plants (North Anna 2, Sequoyah 1, Salem 2, McGuire 1, Diablo Canyon 1, and Virgil C. Summer Nuclear Station).

Natural Circulation Verification Test

The objective of this test was to establish, maintain and recover from natural circulation conditions while at low power. The following description applies to the six Westinghouse plants that performed the natural circulation verification tests, which are summarized by Table 3.1.

The initial conditions of the tests were as follows. The reactor was critical at approximately 3% reactor power. All reactor coolant pumps were in operation. The RCS was at normal temperature and pressure. Pressurizer level was at normal no-load conditions and steam generator narrow range levels were normal.

With the reactor stabilized at approximately 3%, all reactor coolant pumps were tripped in rapid succession. Plant conditions were monitored and recorded during the subsequent stabilization of plant parameters. Adjustments were made to RCP seal water flow rate, charging rate and auxiliary feedwater flow rate in order to maintain stable plants conditions. After stabilized conditions occurred, natural circulation was verified by monitoring the wide range hot and cold leg temperatures and the resultant loop ΔT as shown in Table 3.1. Recovery from natural circulation was achieved by restarting all RCPs at the end of the test.

Simulated Loss of All Onsite and Offsite Power

The purpose of this test is to demonstrate that natural circulation conditions can be supported under simulated loss of power conditions. Simulated Loss of All Onsite and Offsite AC Power, was initiated by selectively deenergizing specified equipment to simulate the conditions that would exist if a loss of all AC power occurred. The plant was at $T_{no-load}$, 2235 psig, pressurizer level and steam generator levels at no-load values. Power was removed to lighting busses, motor driven auxiliary feedwater pumps, and all non-essential equipment and valves. Some of the plants maintained power to the RCPs, the charging pumps, the component cooling water, service water, and lube oil systems. The plant was maintained by manual operator control (remote) of atmospheric steam dump valves and (local) steam driven auxiliary feedwater pumps. The tests demonstrated manual control capability under adverse lighting and communication conditions.

Test Results and Conclusions

The test data was reviewed for the natural circulation verification and loss of power tests that were performed for the Westinghouse designed plants identified in Table 3.1. Based upon this review the test results for all of these natural circulation tests were found to be very similar. Specifically, these tests demonstrated that natural circulation could be established and maintained with all reactor coolant loops in service and that natural circulation could be supported under simulated loss of power conditions. Also, all plant responses were consistent with expectations and results seen at other similar Westinghouse designed plants.

TABLE 3.1

SUMMARY OF NATURAL CIRCULATION TESTS

NATURAL CIRCULATION VERIFICATION TEST SUMMARY

	North Anna 2	Sequoyah 1	Salem 2	McGuire 1	Diablo Canyon 1	V.C. Summer
Test Number	ST-8	1	90.1	7.2	1.1	ZPT 9.1
Reactor Power	3%	3%	3%	3%	3%	3%
N.C. Flow Verified	Yes	Yes	Yes	Yes	Yes	Yes
Loop ΔT	36-40^{o}F	36^{o}F	40^{o}F	28^{o}F	35-40^{o}F	33.9^{o}F

SIMULATED LOSS OF ALL ONSITE AND OFFSITE POWER

	North Anna 2	Sequoyah 1	Salem 2	McGuire 1	Diablo Canyon 1	V.C. Summer
Test Number	2-ST-9	7	SUP 90.7	7.1	44.3	ZPT-9.2
Reactor Power	1%	1%	3%	0%	0%	0%
RCPs Tripped	Yes	Yes	Yes	No	No	No
Loop ΔT	20^{o}F	17^{o}F	-	N/A	N/A	N/A
TD AFW Pump	Yes	Yes	Yes	Yes	Yes	Yes
PRZR Htrs.	No	No	No	No	No	No
Manual AFW Control	Yes	Yes	Yes	Yes	Yes	Yes
S/G Level	-	-	-	30-70%	Normal	31-47%

N/A - Item is not applicable
 \- - Data is not available

4.0 NATURAL CIRCULATION BORON MIXING EVALUATION

The mixing of borated water with the circulating water in the loop can be divided into two distinct concerns:

1) Mixing at the point of injection relative to the composition of the fluid across the fluid flow path.

2) The boron concentration of the stream measured at various points along the flow loop as a function of time.

The concern addressed in the first case has to do with whether or not the injected borated water mixes with the loop circulation at the point of injection. For the Virgil C. Summer Nuclear Station, it is assumed that the normal boration path is unavailable. As a result, a more limiting borated water flow of 5 gpm/loop through the RCP seals is assumed which represents the minimum borated water injection rate. Should the normal boration path be available, the period required to achieve the desired change in RCS boron concentration would be reduced. For this analysis effort, the borated water is injected through the RCS pump seals, and mixing actually takes place at the periphery of the pump rotor. This means that the injected stream is diffuse at the point of mixing with the circulating stream, and the flow is turbulent. Thus, any tendency for the denser injection stream to settle out of the circulating stream is overcome and the boron concentration will be essentially uniform over any cross section of the flow path taken a short distance down stream of the RCS pump.

For the second concern, the boron concentration along the flow path as a function of time is a transport problem. From the moment injection is initiated, a concentration front travels around the loop and the front reaches the injection point. At that time, the concentration at the injection point increases to a new level, and a new front is established. The delay time is determined as the loop swept volume divided by the total flow. As discussed in the following paragraph, when the loop flow is 10 times the injection flow or greater, the transport lag becomes negligible compared to the overall rate of change of concentration at the injection point, and the concentration at any time is essentially uniform around the loop. Under these conditions, the concentration is an exponential function of time with a time constant determined by the swept volume divided by the injection flow, and is independent of the circulating flow.

The CREARE 1/5th scale test facility, Reference 4, tested the degree of mixing of cold high pressure injection water with warmer loop and downcomer water assuming various loop and HPI flow rates. An important parameter used during this testing was the ratio of loop flow to HPI flow. From the test report it was determined that for a ratio of 10 or greater, excellent mixing of the HPI and loop flow exists. For the Virgil C. Summer Nuclear Station under natural circulation conditions, the best estimate of the loop flow is 600 times the injection flow rate based on a comparison with the test results obtained from Diablo Canyon Unit 1. Clearly, therefore, the boron concentration is independent of the actual value of the loop flow, and is determined solely by the injection flow and concentration. With an injection of 5 gpm per loop and a boron concentration of 7000 ppm, the time required to raise the loop concentration from 2000 ppm to 2300 ppm is 225 minutes, or 3.75 hours.

5.0 SUMMARY AND CONCLUSIONS

As stated in the introduction, the Virgil C. Summer Nuclear Station natural circulation evaluation program addresses five program activities for natural circulation and boron mixing. These five activities are noted in parentheses following each activity summarized below.

A general comparison of the plant systems and equipment that affect natural circulation flow and boron mixing has been made between the Virgil C. Summer Nuclear Station and Diablo Canyon Unit 1 plants (Activity 1). This comparison has demonstrated that the Virgil C. Summer Nuclear Station natural circulation flowrates and boron mixing system capabilities are comparable to those of Diablo Canyon Unit 1 (Activity 2). A single failure evaluation of the Virgil C. Summer Nuclear Station natural circulation capabilities demonstrates that in the event of a single failure, natural circulation capability is maintained (Activity 3). Available industry natural circulation test results for the Virgil C. Summer Nuclear Station and several Westinghouse designed plants have been reviewed. These tests included verification of natural circulation flow and demonstrating support of natural circulation with loss of AC power. Based upon the industry natural circulation test results reviewed it is concluded that natural circulation flows will be established for the Virgil C. Summer Nuclear Station (Activity 4). Finally, based upon comparing test results from several small scale mixing tests with the expected natural circulation flow responses for the Virgil C. Summer Nuclear Station, it is expected that borated fluid injected into the RCS will thoroughly mix throughout the system (Activity 5).

Therefore, it is concluded that for the natural circulation flow and boron mixing capabilities, the Virgil C. Summer Nuclear Station has addressed the requirements of Branch Technical Position RSB 5-1, Design Requirements for Decay Heat Removal Systems (Reference 1).

6.0 REFERENCES

1. Branch Technical Position RSB 5-1, "Design Requirements for Decay Heat Removal Systems", Nuclear Regulatory Commission, Revision 2, July 1981.

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