SCE&G
A SCANA COMPANY

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Washington DC 20555-0001

Gentlemen:

Gary J. Taylor
Vice President
Nuclear Operations

Subject: VIRGIL C. SUMMER NUCLEAR STATION
DOCKET NO. 50/395
OPERATING LICENSE NO. NPF-12
RESPONSE TO GENERIC LETTER 98-04
"Potential for Degradation of the Emergency Core Cooling System and the Containment Spray System After a Loss-of-Coolant Accident Because of Construction and Protective Coating Deficiencies and Foreign Material in Containment"

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Attachments:
1. Generic Letter 98-04 Requested Information
2. Containment Sump and its Effects on Long Term Cooling Following a LOCA
3. USNRC SER Excerpts for ECCS and RB Spray Systems

On July 14, 1998, the Nuclear Regulatory Commission issued the referenced generic letter addressing issues which have generic implications regarding the impact of potential coating debris on the operation of safety related systems, structures, and components (SSC) during a postulated design basis LOCA. Protective coatings are necessary inside containment to control radioactive contamination and to protect surfaces from corrosion. Detachment of the coatings from the substrate may make the ECCS unable to satisfy the requirement of 10 CFR 50.46(b)(5) to provide long-term cooling and may make the safety-related containment spray system (CSS) unable to satisfy the plant-specific licensing basis of controlling containment pressure and radioactivity releases following a LOCA. The generic letter requests information under 10 CFR 50.54(f) to evaluate the addressees' programs for ensuring that Service Level 1 protective coatings inside containment do not detach from their

substrate during a design basis LOCA and interfere with the operation of the ECCS and the CSS. The NRC intends to use this information to assess whether current regulatory requirements are being correctly implemented and whether these requirements need to be revised.

The generic letter requires, within 120 days, that licensees provide the information contained in the Requested Information section of the Generic Letter for each of their facilities. By this letter and its attachments, South Carolina Electric and Gas is providing the required 120-day response.

Should you have any questions, please contact Mr. Donald L. Jones at (803) 345-4480.

Very truly yours,

Gary J. Taylor

DLJ/GJT/dr
Attachments (3)

c: J. L. Skolds
W. F. Conway
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R. J. White
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RTS (LTR 980004)
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DMS (RC-98-0207)

STATE OF SOUTH CAROLINA :
: **TO WIT** :
COUNTY OF FAIRFIELD :

I hereby certify that on the 9th day of November 19 98, before me, the subscriber, a Notary Public of the State of South Carolina personally appeared Gary J. Taylor, being duly sworn, and states that he is Vice President, Nuclear Operations of the South Carolina Electric & Gas Company, a corporation of the State of South Carolina, that he provides the foregoing response for the purposes therein set forth, that the statements made are true and correct to the best of his knowledge, information, and belief, and that he was authorized to provide the response on behalf of said Corporation.

WITNESS my Hand and Notarial Seal

Michael J. Zaccone
Notary Public

My Commission Expires

My Commission Expires July 12, 2005
Date

Attachment 1
Generic Letter 98-04 Requested Information

The following is the South Carolina Electric and Gas response to the Request for Information section of Generic Letter 98-04 (questions from the letter are shown in bold italics):

(1) A summary description of the plant-specific program or programs implemented to ensure that Service Level 1 protective coatings used inside the containment are procured, applied, and maintained in compliance with applicable regulatory requirements and the plant-specific licensing basis for the facility. Include a discussion of how the plant-specific program meets the applicable criteria of 10 CFR Part 50, Appendix B, as well as information regarding any applicable standards, plant-specific procedures, or other guidance used for: (a) controlling the procurement of coatings and paints used at the facility, (b) the qualification testing of protective coatings, and (c) surface preparation, application, surveillance, and maintenance activities for protective coatings. Maintenance activities involve reworking degraded coatings, removing degraded coatings to sound coatings, correctly preparing the surfaces, applying new coatings, and verifying the quality of the coatings.

RESPONSE:

South Carolina Electric and Gas has implemented controls for the procurement, application, and maintenance of Service Level 1 protective coatings used inside the containment in a manner that is consistent with the licensing basis and regulatory requirements applicable to the V. C. Summer Nuclear Station. The requirements of 10 CFR Part 50 Appendix B are implemented through specification of appropriate technical and quality requirements for the Service Level 1 coatings program which includes ongoing maintenance activities.

The V. C. Summer Nuclear Station meets the requirements of RG 1.54, "Quality Assurance Requirements for Protective Coatings Applied to Water Cooled Nuclear Power Plants", Revision 0, dated June, 1973. As such, Service Level 1 coatings are subject to the requirements of ANSI N101.2-72, ANSI N101.4-72, and ANSI N5.12-74.

For the Westinghouse scope of supply (NSSS equipment and components), an alternate methodology for meeting the requirements of RG 1.54 was employed. The VCSNS FSAR states:

"For the Westinghouse scope of supply, Westinghouse employs process specifications and the Westinghouse Quality Assurance Program, including quality assurance surveillance and auditing, to provide adequate confidence that coating work within Westinghouse scope will perform satisfactorily in service."

An alternate method of compliance with this regulatory guide has been submitted to the NRC (via letter NS-CE-1352, dated February 1, 1977, to Mr. C. J. Heltemes, Jr., Quality Assurance Branch, NRC, from Mr. C. Eicheldinger, Westinghouse PWRSD, Nuclear Safety Department) and accepted (via letter, dated April 27, 1977, to Mr. C. Eicheldinger from Mr. C. J. Heltemes, Jr.).

Additionally, a small amount of unqualified coatings was identified inside containment. The VCSNS FSAR states:

"The estimated quantity of unqualified paint inside containment is 0.18 cubic feet. This is based upon the conservative estimate of 0.25 percent of all painted surfaces inside the Reactor Building including both the concrete and steel. For an average dry film thickness of 8 mils, this represents an area of 270 sq ft."

This quantity of unqualified coatings was confirmed in a survey conducted prior to power operations in November 1981.

The SER for VCSNS states:

The applicant has proposed to select protective coatings which satisfy ANSI N101.4-1972, "Quality Assurance for Protective Coatings Applied to Nuclear Facilities." The applicant has also committed additional quality assurance requirements in a letter from C. Eicheldinger, Westinghouse to C. Heltemes, NRC, dated February 1, 1977, which we have reviewed and found acceptable. Accordingly, we conclude that the protective coatings selected by the applicant will meet the recommendations of Regulatory Guide 1.54, "Quality Assurance Requirements for Protective Coatings Applied to Water-Cooled Nuclear Power Plants," with the exception of a small quantity (0.18 cubic foot) of unqualified paints. The quantity of unqualified paints is sufficiently small that potential decomposition products from this source will not pose a safety problem for the facility.

Adequate assurance that the applicable requirements for the procurement, application, inspection, and maintenance are implemented is provided by plant specific procedures and programmatic controls, approved under the South Carolina Electric and Gas Quality Assurance program.

SCE&G is evaluating the guidance provided in EPRI TR-109937 "Guideline on Nuclear Safety-Related Coatings" and, as appropriate, improvements to our existing programs and procedures for Service Level 1 coatings will be implemented upon completion of the evaluation.

(a) Service Level 1 coatings used for new applications or repair/replacement activities are procured from a vendor with a quality assurance program meeting the applicable requirements of 10 CFR Part 50 Appendix B. The applicable technical and quality requirements that the vendor is required to meet are specified by SCE&G in procurement documents. Acceptance activities are conducted in accordance with procedures that are consistent with ANSI N 45.2 requirements (e.g., receipt inspection, source surveillance, etc.). This specification of required technical and quality requirements combined with appropriate acceptance activities provides adequate assurance that the coatings received meet the requirements of the procurement documents.

(b) The qualification testing of Service Level 1 coatings used for new applications or repair/replacement activities inside containment meets the applicable requirements contained in the standards and regulatory commitments referenced above. These coatings, including maintenance and repair coatings, have been evaluated to meet the applicable standards and regulatory requirements previously referenced.

(c) The surface preparation, application and surveillance during installation of Service Level 1 coatings used for new applications or repair/replacement activities inside containment meet the applicable portions of the standards and regulatory commitments referenced above. Documentation of completion of these activities is performed consistent with the applicable requirements.

SCE&G periodically conducts inspections and condition assessments of Service Level 1 coatings inside containment. Inspections are conducted as part of containment structural integrity verification, Maintenance Rule monitoring, general maintenance planning and, to a limited degree, during recovery and restoration from Refueling Outages. Coatings condition assessments are conducted on an as needed basis as part of non-conformance corrective actions. As localized areas of degraded coatings are identified, those areas are evaluated and scheduled, usually in the following outage, for repair or replacement, as necessary. The periodic inspections, condition assessments, and the resulting repair/replacement activities, assure that the amount of Service Level 1 coatings which may be susceptible to detachment from the substrate during a LOCA event is minimized.

(1) Information demonstrating compliance with item (I) or item (ii):

(i) For plants with licensing-basis requirements for tracking the amount of unqualified coatings inside the containment and for assessing the impact of potential coating debris on the operation of safety-related SSCs during a postulated design basis LOCA, the following information shall be provided to demonstrate compliance:

(a) The date and findings of the last assessment of coatings, and the planned date of the next assessment of coatings.

RESPONSE:

This article is not applicable to the V. C. Summer Nuclear Station. Please see article (ii).

(b) The limit for the amount of unqualified protective coatings allowed in the containment and how this limit is determined. Discuss any conservatism in the method used to determine this limit.

RESPONSE:

This article is not applicable to the V. C. Summer Nuclear Station. Please see article (ii).

(c) If a commercial-grade dedication program is being used at your facility for dedicating commercial-grade coatings for Service Level 1 applications inside the containment, discuss how the program adequately qualifies such a coating for Service Level 1 service. Identify which standards or other guidance are currently being used to dedicate containment coatings at your facility; or,

RESPONSE:

SCE&G does not currently employ commercial grade dedication for Service Level 1 coatings used inside containment at VCSNS. Therefore, this article is not applicable to the V. C. Summer Nuclear Plant.

(ii) ***For plants without the above licensing-basis requirements, information shall be provided to demonstrate compliance with the requirements of 10CFR50.46b(5), "Long-term cooling" and the functional capability of the safety-related CSS as set forth in your licensing basis. If a licensee can demonstrate this compliance without quantifying the amount of unqualified coatings, this is acceptable.***

RESPONSE:

The following description and referenced materials describe the licensing basis for VCSNS relative to conformance with 10 CFR. 50.46(b)(5), "Long-term cooling," specifically with regard to VCSNS's ability to provide extended decay heat removal including related assumptions for debris that could block containment emergency sump screens.

The ECCS components are designed such that a minimum of two accumulators, one charging pump and one residual heat removal pump together with their associated valves and piping will assure adequate core cooling in the event of a Design Basis Accident. The redundant onsite emergency diesels assure adequate emergency power to all electrically operated components in the event that a loss of offsite power occurs simultaneously with a loss of coolant accident, even assuming a single failure in the emergency power system such as the failure of one diesel to start. During the recirculation phase of cooling, the ECCS systems take suction from the Reactor Building recirculating sumps.

The design of the recirculation sumps includes specific features for minimizing the potential for clogging of the screens. These design features include the following:

1. An outer trash rack and curb to protect the fine screens from large pieces of debris.
2. Vertical orientation of the fine screens.
3. The establishment of low velocity settling areas for each of the four individual deep sumps.

The first low velocity settling area is established at the entrance to the 1/2 inch mesh screens by directing the downward flow path around each 6 foot by 6 foot standpipe to

horizontal flow through the 1/2 inch screens. Based upon the available flow area around each 6 foot by 6 foot standpipe and the RHR pump design flow rate of 3750 gpm, the calculated fluid velocity at the entrance to the 1/2 inch mesh screens is 0.156 ft/sec.

The second low velocity settling area is provided between the 1/2 inch mesh screens and the 1/4 inch mesh screens. The location of the vertically mounted 1/4 inch mesh screens results in an upward flow path between the 1/2 inch and 1/4 inch mesh screens with the two required changes in flow direction to obtain horizontal flow through both sets of screens. Based upon the flow area between the screens and the RHR pump design flow rate (3750 gpm), the calculated fluid velocity in this second low velocity settling area is 0.42 ft/sec.

The total amount of fine screen in each recirculation sump standpipe provides enough total free area to ensure that the resultant pressure drop has no appreciable effect on the net positive suction head available to each RHR pump. For the postulated condition of partially clogged screens, the total pressure drop was conservatively calculated using only half of the free area of the screens. This calculated pressure drop for flow through both sets of screens is 0.2 feet for the RHR pump design flow rate of 3750 gpm. This pressure drop represents a reduction of only 0.77% in the net positive suction head available to the RHR pumps.

Additional features incorporated into the plant design to minimize clogging of the sumps are as follows:

1. Use of insulation described under Regulatory Guide 1.36 for equipment and piping inside the Reactor Building.

2. Use of coating systems as described under Regulatory Guide 1.54 for carbon steel and concrete surfaces.

The effects of LOCA generated debris on the Reactor Building recirculation sumps are discussed in Attachment 2.

The Virgil C. Summer Nuclear Station complies with the intent of the recommendations of Regulatory Guide 1.82. Under this commitment, VCSNS has assumed that the systems that draw from the sumps for emergency core cooling and containment spray systems may experience sump blockage of up to 50% of the effective sump area from debris generated as a result of a loss of coolant accident (LOCA). At the time VCSNS was licensed, no distinction was drawn between the various potential sources for post-LOCA debris; these systems were intended to function, even with debris partially obstructing the sumps, from whatever source derived. The analyses and testing results

submitted as part of the licensing basis for VCSNS demonstrate, however, that, even with this blockage, the emergency core cooling and containment spray systems will continue to provide sufficient cooling flow as to fulfill the long-term cooling functions required to conform with 10 CFR 50.46(b)(5).

The NRC accepted these analyses and these systems as meeting the requirements of 10 C.F.R. 50.46(b)(5) in NUREG 0717. Attachment 3 contains the applicable exerpts from the SER addressing the acceptability of the sumps for the emergency core cooling and containment spray systems.

The licensing basis for VCSNS as accepted by the NRC's SER, provides both the regulatory and safety basis for safety system performance. Coatings are not treated separately in the licensing basis for VCSNS because the 50% sump screen blockage assumption does not distinguish among the sources of the LOCA-generated debris. The potential sources of LOCA generated debris have been reviewed with respect to sump clogging and programmatic controls are in place to monitor and maintain protective coatings in the Reactor Building. These actions assure that the amount of LOCA generated debris that can actually migrate to the sumps is minimized. Therefore, we conclude that the 50% sump blockage criteria is bounded.

As the NRC noted in NRC Generic Letter 85-22, "Potential for Loss of Post-LOCA Recirculation Capability due to Insulation Debris Blockage," a change in regulatory guidance for the basis for sump screen blockage would constitute a generic backfit.

The following information shall be provided:

(a) If commercial-grade coatings are being used at your facility for Service Level 1 applications, and such coatings are not dedicated or controlled under your Appendix B Quality Assurance Program, provide the regulatory and safety basis for not controlling these coatings in accordance with such a program. Additionally, explain why the facility's licensing basis does not require such a program.

RESPONSE:

SCE&G does not currently employ commercial grade dedication for Service Level 1 coatings used inside containment at VCSNS. Therefore, this article is not applicable to the V. C. Summer Nuclear Plant.

Attachment 2
Containment Sump and its Effect on Long Term Cooling Following a LOCA
Taken from FSAR Section 6.3.2.6.1

The Virgil C. Summer Nuclear Station Technical Specifications require that each Emergency Core Cooling System (ECCS) subsystem be demonstrated operable by a visual inspection. This visual inspection verifies that no loose debris is present in the Reactor Building which could be transported to the RHR and RB Spray sumps to cause restriction of their pump suctions. This inspection is performed immediately prior to establishing containment integrity for:

1. The Reactor Building areas affected during the outage, and

2. All accessible areas of the Reactor Building.

The suction inlets and sump components are inspected to ensure that debris is not present and that no evidence exists of structural distress or corrosion.

Station Surveillance Test Procedures specify these requirements.

An instruction dealing with possible ECCS sump blockage is contained in the Virgil C. Summer Nuclear Station Emergency Operating Procedures (EOP's). This procedure addresses long term cooling and requires the operator to monitor pump flow and motor amps when in the sump recirculation mode. It also cautions the operator that if pump flow and/or amps decrease, the cause may be sump blockage. If there is an indication of sump blockage, the operator is instructed to monitor the pumps for signs of cavitation and the piping for water hammer. If suction is completely lost, the operator is instructed to stop the affected pump. During the classroom training session on EOP's for station operators, possible actions that may be taken in the event of a blocked screen are covered. For example, a blocked sump could be backflushed using water which may remain in the RWST.

Four independent ECCS sumps, 2 RHR and 2 RB Spray, are provided in the Virgil C. Summer Nuclear Station. FSAR Figure 1.2-4 shows the physical separation between these sumps and the Reactor Building drainage sumps. The Reactor Building drainage system as well as the Refueling Canal drain are located such that direct water flow does not impinge in the sump areas. There are no high energy lines in the areas which will subject the sumps to pipe or jet impingement loads and cause subsequent damage. There is also a 6 inch curb around the upper floors to keep water from cascading down the containment wall to the sumps below. Water from the upper floors will tend to go down the stair wells.

The four sumps are located below the working floor elevation and are surrounded by 6 inch high trash curbs. The LOCA condition water level when recirculation begins is approximately 6 to 7 feet above this working floor elevation. A horizontal grating acts as a trash rack over each sump and 2 vertical screens, one with 1/2 inch openings and the other with 1/4 inch openings, act as the suction strainers. An inner sump extends below the strainers an additional 8 feet to the suction inlet pipe. A horizontal top plate covers the strainer screens and the inner sump and includes a hatch for inspection purposes. The hatch clearances provide air vent gaps for the top cover plate. Components of the ECCS sumps are seismically designed and are constructed of stainless steel except the grating which is of painted carbon steel.

Three (3) types of insulation are used inside the reactor building: all stainless steel reflective insulation and two types of mass insulation encapsulated in stainless steel. The stainless steel reflective insulation is used primarily on piping while one type of mass insulation encapsulated in stainless steel is used only around the reactor pressure vessel loop inlet and outlet nozzles and the portions of reactor coolant piping that penetrate the primary shield wall. The potential for mirror insulation to be torn away from its pipe and causing a blockage problem is limited by:

1. The method of attachment to the pipe (metal bands and riveted buckles),

2. The fact that it tends not to break up after separation from the pipe,

3. The fact that it will sink to the bottom of the containment pool, and

4. Only a small quantity is located outside of the missile shield wall in the area of the sumps.

The other mass type encapsulated in stainless steel is used on the pressurizer and steam generator level and flow instrument tubing. The mass type insulating material is attached by a stainless steel jacket that is riveted in place. Approximately 520 square feet of 2 inch thick insulation cover the steam generator reference legs. For the insulation in a single steam generator compartment to become a blockage source, it would have to separate from the pipe and jacket and then maneuver through the missile shield openings and around to the sump areas. An additional 440 square feet of 1 inch thick Temp Mat covers the pressurizer equipment cooling duct as it branches from the Reactor Building cooling duct and enters the pressurizer cubicle. Over half of this insulation is actually inside the cubicle. For the ductwork insulation to become a potential problem, a pipe rupture of the main steam loop A line or a line inside the pressurizer cubicle would have to occur. Then either a jet impingement or pipe impact load would have to tear the insulation and duct work apart. For the insulation outside

the cubicle, it would then have to go down three floors to get to the sump areas. Several grating floor elevations inside the pressurizer cubicle are between this insulation and the floor. The insulation would then have to pass through the 3' x 9' -6" door opening and then make its way from elevation 436 to the sump floor elevation 412.

Temp Mat tends to float due to entrained air. However, if it does sink, tests show it acts as a filter element. No significant effect on flow is indicated when a blanket of the material is placed over a sump drain.

Other potential blockage sources include:

1. Fifteen fibrous reinforced silicon rubber enclosures (approximately 4 square feet each) which provide forced air cooling of equipment inside the pressurizer cubicle,

2. Rubber expansion joints in the ring header duct,

3. Rubber boots for reactor nozzle and support feet ventilation (6 total at approximately 14 square feet each).

4. Kaowool wrapping (with stainless steel encasement sealed on edges with an elastomer sealant) for electrical conduit to meet separation criteria of electrical circuits for fire protection, and

5. M-Board (with stainless steel encasement sealed on edges with an elastomer sealant) for electrical conduit and cable trays to meet separation criteria of electrical circuits for fire protection.

The addition or deletion of potential blockage sources would be addressed by the VCSNS Design Control Program.

The blockage potential of the equipment covers is subject to the same location considerations as the Temp Mat inside the pressurizer cubicle. In addition, the covers are located high in the cubicle and would have to clear instruments, piping, and grating to reach the cubicle opening at the bottom. The ductwork expansion joints are so removed from the sumps and potential missiles that their blockage potential is very small. The reactor nozzle and support ventilation boots are bolted to the well and clamped to the pipe. However, if a boot became free due to its physical location, it would most likely remain in the incore instrument tunnel.

The Kaowool is totally enclosed by a stainless steel corrugated tube which is split along the axial direction and installed over the Kaowool. The split tube is held in place by half inch wide banding straps. Strap spacing is four (4) inches in the axial direction except at conduit supports. The ends of the encasement are sealed by an elastomer sealant to preclude entrance of any spray fluid into the encasement. The encasement assembly will also withstand seismic inputs to the installation without loss of structural integrity or sealing function.

The M-Board installations are designed with total encasement of each M-Board by stainless steel sheets with the edges sealed by overlap of the sheets or angle clamps to preclude entrance of spray fluid. The assemblies of the various M-Board shapes are joined by angles and channels by bolting for structural strength and rigidity. The various sections which make up a particular fire barrier are seismically qualified to remain in place. In the unlikely event sections should fall, the stainless steel encasement will remain intact and prevent the M-Board from becoming dissolved by the spray/flood water.

All of the above items are installed above post-LOCA flood level and most are located in areas not directly struck by the containment spray water. All materials used are non-reactive with the spray water and therefore do not produce hydrogen in the post-LOCA environment.

The potential for debris getting into the suction piping and causing blockage or damage to the pumps or other components is greatly reduced by the trash racks and screens. For the components in the ECCS flow path the Reactor Building Spray nozzles are the determining factor for sizing the smallest strainer screens. Strainer screens with 1/4 inch square openings will allow only those particles smaller than 1/4 inch square to pass completely through the system.

To perform a complete analysis, SCE&G contracted Alden Research Laboratories to perform a model study of the Virgil C. Summer Nuclear Station ECCS sumps and suction piping. The study investigated several design phenomena including swirling and vortexing under full flow and 50% block strainer conditions, losses leading to insufficient NPSH, and air entrainment. Scaling factors were also evaluated to ensure similarity between the model and prototype operating under LOCA conditions. The results of this study demonstrate that no significant vortex phenomena occurred. Therefore, it was determined that the existing sump design is acceptable and that no modifications are required.

RHR and RB Spray pump flow beyond rated runout is another condition which required evaluation. Due to the different known and unknown conditions of operation, this evaluation is more important for the RHR system. Therefore, full scale tests of the

RHR pumps were performed at the Virgil C. Summer Nuclear Station for the different modes of injection and recirculation. As a result of these tests, flow restricting orifices were installed at the outlet of each heat exchanger.

A full scale test of the RB Spray discharge ring headers and spray nozzles is not feasible. However, this system is only subjected to two worst case operating conditions, injection and recirculation, as its design basis. Full scale tests have been performed using the suction and recirculation test returns to the RWST. Test data on the suction piping has been compared to the results of the design calculations as a measure of their accuracy. A detailed analysis was then performed of the Reactor Building Spray System's calculations to ensure that flow rates beyond runout are not possible with subsequent pump damage.

Attachment 3
USNRC SER Excerpts for ECCS and RB Spray Systems

The VCSNS SER section for the RHR system (6.3.6) states:

We have reviewed the drawings, component descriptions, design criteria, performance analyses, and testing of the emergency core cooling system. Based on this review, we have determined that the emergency core cooling system will meet the acceptance criteria and conform to the Commission's requirements as set forth in General Design Criteria, regulatory guides, and staff technical positions provided that matters discussed in Section 6.3.5 of this Safety Evaluation Report are resolved.

The "matters discussed in section 6.3.5 pertained to testing of the RHR and RB Spray sumps. The text of section 6.3.5 follows:

6.3.5 Tests and Inspections

The applicant has performed flow tests on the containment recirculation sump to meet the requirements of Regulatory Guides 1.68 and 1.79 to demonstrate the performance of the recirculation sump. However, in-plant sump tests did not accurately replicate expected post-loss-of-coolant accident conditions, and this did not fully confirm acceptable sump performance under emergency core cooling system recirculation conditions. Specifically, the plant tests only took suction from a single line, when there are two lines in each of two sumps. This resulted in approach flow velocities which were lower in the test than would be expected during actual loss-of-coolant accident conditions.

Additionally, various flow approach directions were not investigated to determine if undesirable rotation could be induced in the sump area, which could lead to vortex formation.

Finally, sump screen blockage due to debris entrainment was not considered, with the correspondingly higher screen velocities which also could aggravate vortex formation.

The applicant plans to conduct a scale model sump test to investigate these areas and to demonstrate that recirculation sump performance will be acceptable in the expected post loss-of-coolant accident environment. The applicant has Committed to make any modifications to the recirculation sump which are identified as necessary by the model test program. We will require that the results of the model test be submitted for our review. The resolution of this matter will be reported in a supplement to this Safety Evaluation Report.

The applicant has committed to perform routine periodic testing of the emergency core cooling system components and all necessary support systems with the plant at power. Valves that are required to operate after a loss-of-coolant accident will be operated through a complete cycle, and pumps will be operated individually on their miniflow lines. Test lines will be provided to perform periodic tests on emergency core cooling system check valve operability.

The staff has requested additional information on the applicant's leak testing program to verify the integrity of pressure isolation check valves. The applicant's response has been evaluated and is reported in Section 3.9.6 of this Safety Evaluation Report.

The additional testing and analysis described above was completed and reviewed by the NRC Staff. In Supplement 3 to the VCSNS SER, the Staff stated,

6.3.5 Tests and Inspections

In the Safety Evaluation Report, we stated that the applicant planned to test a scale model of the containment sump design. The following evaluation replaces the fourth paragraph of Section 6.3.5 of the Safety Evaluation Report.

At our request, the applicant conducted a scale model sump test to investigate these areas. The sump tests demonstrated that recirculation sump performance will be acceptable during the postulated post loss-of-coolant accident environment. We consider this matter resolved.