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April 19, 1982

Mr. Harold R. Denton, Director
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D.C. 20555

Subject: Virgil C. Summer Nuclear Station
Docket No. 50/395
NRR Items Requiring Resolution

Dear Mr. Denton:

Through formal submittal of documentation, and continual meetings and discussions, South Carolina Electric and Gas Company (SCE&G) and the NRC have resolved a large number of items requiring resolution for the Virgil C. Summer Nuclear Station. This number is probably in the range of over a thousand. A good rapport was needed to accomplish this task. Many times honest differences of opinion or points of view occurred, but we were able to work out a resolution for most.

At this time seven items exist which in our opinion need additional discussions to resolve. We have attempted to resolve these on the reviewer level and with some items at the branch chief level. However, it is our position that the seven items described below require resolution at a higher management level within the NRC. Six of the items involve the plant Technical Specifications.

1. Remote Shutdown Instrumentation
   Technical Specification 3.3.3.5 (Page 3/4 3-53)

   This Specification will require us to shut down the Plant if any one of the instruments on the Control Room Evacuation Panel is inoperable for over seven (7) days. After the TMI accident, the NRC changed the action for this Specification from 30 days to 7 days.

   SCE&G contends that the loss of one instrument, although it may compromise operation from the panel, does not prevent the safe shutdown of the Plant. Therefore, the action should not require a shutdown of the Plant but an evaluation of the effect on the function of Control Room Evacuation Panel. A report of the problem with corrective action addressed would be made to the NRC and, if necessary, temporary measures would be provided to assure the plant can be safely shut down and maintained from outside the Control Room. SCE&G, in our September 25, 1981, letter to Mr. Virgilio, requested a change to this Specification.

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2. Accident Monitoring Instrumentation
   Technical Specification 3.3.3.6 (Page 3/4 3-56)

   This Specification applies to the Post-Accident Monitoring (PAM) Instrumentation and the contention concerning this Specification is the same as for Specification 3.3.3.5, above. The major difference is that for the PAM instrumentation, redundant channels are required. The action for the loss of one channel is the same as for Specification 3.3.3.5, i.e., shut the Plant down in 7 days. However, if both channels fail, one will have to be repaired within 48 hours or the Plant will have to be shut down.

   Exceptions to the redundant channels are the position indications for the pressurizer PORV's, PORV block valves, and safety valves. Only one position indication exists for each valve and if it fails and is not repaired within 48 hours, the Plant will be shut down. SCE&G, in our September 25, 1981, letter to Mr. Virgilio, requested a change to this Specification.

3. Steam Generator
   Technical Specification 3.4.5 (Page 3/4 4-13)

   This Specification requires that all steam generators be operable prior to increasing the average temperature of the reactor coolant above 200°F. The item of contention concerns the surveillances for this Specification which establishes the requirements for determining if the steam generators are operable.

   After a loss-of-coolant accident which causes a safety injection, Surveillance 4.4.5.3.c requires an unscheduled inspection of at least 3% of the tubes in each steam generator. This inspection will have to be completed before returning the Plant to power.

   SCE&G contends that if during a loss-of-coolant accident the safety injection pumps are able to maintain the reactor coolant system pressure above the secondary system pressure in the steam generators, there is no undue stress on the tubes. Therefore, no justifiable reason exists for the unscheduled inservice inspection which will require about 10 or 12 days of shutdown to perform. If a safety injection is caused by a loss-of-coolant accident, it is expected that in the majority of the cases the safety injection pumps will maintain the reactor coolant pressure above the secondary coolant pressure. SCE&G, in our September 25, 1981, letter to Mr. Virgilio, requested a change to this Specification.

4. Operational Leakeage
Technical Specification 3.4.6.2 (Page 3/4 4-19)

This Specification requires that the reactor coolant leakage from each reactor coolant system pressure isolation valve be limited to 1 GPM. The basis for this requirement is the NRC proposed scenario that the valves would fail and cause an overpressurization and rupture of the low pressure piping producing a loss-of-coolant accident outside of containment. The NRC contends that valve leakage greater than 1 GPM is indicative of an impending gross failure and that the leakage should be checked at least once per refueling and the valve repaired if the leakage is greater than 1 GPM.

SCE&G initially contended that this requirement was not needed because the Specification already requires that the reactor coolant system leakage be limited to 1 GPM unidentified leakage and 10 GPM of identified leakage and that although 1 GPM leakage through the reactor coolant system pressure boundary may be indicative of an impending gross failure, leakage of 1 GPM across a valve seat certainly is not indicative of impending gross failure of the valve. However, the former position was conceded and we agreed to include the reactor coolant system pressure isolation valves in this Specification with some major changes. Details of the SCE&G position are described in our December 8, 1981, letter to Mr. Virgilio.

5. Hot vs. Cold Shutdown
Specification 3.7.3 (Page 3/4 7-11)
3.7.4 (Page 3/4 7-12)
3.7.5 (Page 3/4 7-13)
3.8.1.1 (Page 3/4 8-1, 3/4 8-2)
3.8.2.1 (Page 3/4 8-9)
3.8.3.1 (Page 3/4 8-13)

These Specifications are for systems that are required to keep the Plant in cold shutdown. The NRC contends that cold shutdown is the safest condition for the Plant and requires that when the redundance of these systems is lost that the Plant be placed in cold shutdown.

SCE&G contends that when the reliability of systems needed to keep the Plant in cold shutdown is questionable, it is safer to maintain the Plant in hot shutdown. Hot shutdown provides the maximum number of options available to remove the decay heat of the core. Under this condition, 5 possible paths for heat removal exist: two residual heat removal loops and three reactor coolant loops. In cold shutdown the only method of removing heat is the two residual heat removal loops

because the reactor coolant temperature is less than 200°F and heat can no longer be removed by steaming the steam generators.

The above Specifications require the Plant to be placed in cold shutdown when any one of the systems (including electrical power) that is needed to remove heat via one of the residual heat removal loops fails. This means that the one remaining residual heat removal loop has to be used to maintain the reactor coolant system temperature below 200°F. However, if the Plant was maintained in hot shutdown (reactor coolant system temperature between 200°F and 350°F) in addition to having the one remaining residual heat removal loop, there would also be three reactor coolant loops, any one of which could be used to remove heat.

If the service water pond is lost, Specification 3.7.5 requires the Plant to be placed in cold shutdown. However, this pond is the ultimate heat sink for both residual heat removal loops without which there is no way to remove the decay heat or comply with the requirements of the Specification. However, the Plant could be maintained safely in hot shutdown under these conditions by steaming the steam generators. SCE&G, in our October 14, 1981, letter to Mr. Virgilio, requested a change to this Specification.

6. Steam Generators
   Technical Specification 3.4.5 (Page 3/4 4-14)

This criteria specifies a tube plugging margin of 40%. As discussed in the FSAR, SCE&G had Westinghouse determine the degraded steam generator tube plugging margin in accordance with Regulatory Guide 1.121. Applying the same design philosophy for the Virgil C. Summer Nuclear Station as previously used for RESAR-414, which was accepted by the NRC staff in NUREG 0491 (November 1978), pertinent portions attached, Westinghouse determined the tube plugging margin to be 55%. On November 30, 1981, SCE&G submitted WCAP 9912 (Properietary) and WCAP 9989 (Non-Proprietary) which provided technical justification to support a 55% steam generator tube plugging margin for the Virgil C. Summer Nuclear Station. Technical Specification 3.4.5 was changed and sent by SCE&G to the NRC reviewer on May 15, 1981.

However, in April 1982, the NRC rejected this analysis because a factor of safety of three was not used. Discussions revealed that the NRC intended for the factor of safety of two to apply only to the case of through-wall cracking and not to tube thinning (wastage). SCE&G pointed out that NUREG 0491 accepted the Westinghouse position with no exceptions. The NRC reviewer stated that even though NUREG 0491 did

not clearly state it, that this new interpretation was what the NRC meant.

SCE&G finds this difficult to understand. In good faith, and at considerable expense, we took written NRC positions to provide a basis for the Reg. Guide 1.121 analysis and support the 55% tube plugging limit. We are now being told that the NRC position stated in NUREG 0491 is invalid.

It is the position of SCE&G that the factor of safety of two is justified and acceptable. We find no valid basis for the subjective, contradictory interpretation of a previous NRC review.

7. Accumulator Discharge Valves (SER Licensing Condition 1.8.2)

   In Supplement 3 of the Safety Evaluation Report (SER), NUREG 0717, the NRC required SCE&G to install power lock-out devices as part of the controls for the accumulator discharge valves since the current design violates Branch Technical Position RSB 5-1 requirements. Supplement 3 of the SER also states that the modification is to be accomplished prior to startup after the first refueling outage.

   It is our position that as a Class 2 category plant within RSB 5-1, Virgil C. Summer Nuclear Station is not required to be in full compliance with the Staff document. As stated in our letter of March 15, 1982, SCE&G's position is that the present plant design and procedural methods for plant cooldown are an acceptable alternative to BTP RSB 5-1 requirements for a Class 2 plant. On April 2, 1982, the Staff indicated in a conference call that they did not accept our position as stated in the March 15, 1982 letter, and that resolution of this matter was required prior to fuel load.

   Both SCE&G and Westinghouse will be present to offer justification for the original position at our meeting with you.

SCE&G would like to meet with the Staff in Bethesda during the week of April 26, 1982, to resolve the above items. If this week is unacceptable, we suggest the week of May 3, 1982. We feel that with the proper levels of management present at this meeting, final disposition of these items can be accomplished.

If you have any questions, please let us know.

Very truly yours,

T. C. Nichols, Jr.

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RBC:TCN:lkb

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