5.0 UTILITY PARTICIPATION AND INDEPENDENT REVIEW TEAM

5.1 IPE ORGANIZATION

The organizational structure for the IPE is shown in Figure 5-1. The team was put together to optimize the Virginia Power resources while meeting the requirements of Generic Letter 88-20 (NRC 1988). A consultant was retained to provide probabilistic risk assessment (PRA) technology transfer in order to produce the results in a thorough, yet efficient, fashion. Three engineers from the corporate staff were assigned to be team members. However, the consultant, Halliburton NUS Corporation, retained overall responsibility for the technical aspects of the work. The Virginia Power team members were full time participants in the process while the Halliburton NUS team members participated on as needed basis. This approach helped to optimize the IPE process because Virginia Power resources were utilized efficiently, technology transfer was achieved, and the process was completed in accordance with a schedule approved by NRC.

Each Virginia Power team member participated in several of the tasks. A breakdown of the tasks and the Virginia Power participants is provided in Figure 5-2. As shown in the table, each of the tasks had at least one significant Virginia Power participant. Therefore, technology transfer has been accomplished in an effective manner.

5.2 INDEPENDENT REVIEW TEAM AND PROCESS

As shown in Figure 5-1 the independent review team consisted of station personnel, corporate staff, and consultants. The consultants were retained for two reasons. First, among the Virginia Power members of the PRA team there was little prior experience with either core damage or accident progression aspects of a modified Level 2 PRA. Second, because the consultants possessed the detailed knowledge of PRA analysis, it was reasonable to have them act as the coordinators of the independent review. Therefore, a team of two senior analysts from Science Applications International Corporation (SAIC) were contracted to act as chairpersons of the independent review committee and to have overall responsibility for the preparation of the independent review reports. The second consultant was employed to perform the independent review of the accident progression analysis. Since Stone & Webster Engineering Corporation (SWEC) was the architect/engineer for the North Anna Power Station and has maintained cognizance of the station through numerous design change projects, it was logical to employ their services for a review of the Containment analysis. The reviewer from SWEC is a senior analyst who is very familiar with the North Anna design, with severe accident analyses, and with the MAAP code. The scope of his review included all of the accident progression analyses and a check of a limited number of MAAP runs made for success criteria analyses, accident progression analyses, and source term analyses.

The corporate staff involvement was through the normal channel for independent reviews of proposed changes at the station. The group is called Corporate Nuclear Safety (CNS) and it is organizationally independent of the Engineering group. CNS participated in the independent reviews using a variety of corporate staff including members who had worked at North Anna for several years.

The members of the independent review team from the station included licensed Senior Reactor Operators, Control Room Operators, a shift technical advisor, and a member of the procedures group. The system engineering group was represented mostly on an as needed basis during the independent review. The team members participated in a one week review conducted at North Anna. The members of the team are listed in Table 5-1.

The independent review took place during the month of August 1992. At this time the review team had access to each of the analysis files produced by the project. The list of files is presented in Table 5-2. In addition to the set of analysis files, the team had access to the draft final report issued on July 15, 1992.

In early August SAIC and SWEC personnel reviewed the draft report and prepared an outline of the team meeting. The team met at North Anna for the entire week beginning August 17th. The one week session consisted of a brief PRA training session followed by breaking up into groups lead by Messrs. Holderness and Singer. Each team member was then assigned an analysis file(s) to review. Comments were recorded on the standard review form used throughout the project.

The meeting at North Anna focused on the Level I analysis, although the interface between the Level I and the Level II analysis was also considered. Thus, the Containment Building performance analysis review was conducted during the same period of time as the Level I independent review.

In addition to the formal independent review team meetings discussed above, the models were reviewed by other corporate and station personnel at various stages of the project. For example, after the completion of each analysis file, it was reviewed and signed-off by another member of the PRA team. In addition, the system engineers at North Anna participated in a limited review of the system models prior to the independent review. Similarly, personnel from the training department participated in a review of the human reliability analysis. Finally, an STA participated in a one day review of the accident sequence delineation analysis file prior to the final sequence quantification.

5.3 AREAS OF REVIEW AND MAJOR COMMENTS

As stated above, the project analysis files were supplied to the independent review team. The analysis files were then divided among the team members for review. Document review forms were used to document the individual review comments. The overall review is documented in a report which summarizes the significant comments in addition to providing the individual document review forms. Once the document review forms were received, the PRA team responded to each comment and made the appropriate changes in the models. The document review forms were then compiled in a separate analysis file to become part of the IPE documentation.

Significant comments, summarized from the Level I independent review report (SAIC 1992), are presented below:

1. The scope of the study and the level of detail appear to meet or exceed the requirements for an IPE.

2. The models and results generally reflect the current North Anna plant configuration. There are exceptions, however. The starting point for much of the North Anna study appears to be the Surry IPE. As a result, some of the models contain references to design/operational features of Surry. In a few cases, it has been noted that these features are unique to Surry and do not apply to North Anna.

3. The documentation of the study is well organized and nearly complete at the time of the review. Many of the supporting work packages were prepared much earlier in the study and have not been maintained up to date. For example, there are references to future work activities (which have now been completed). The supporting documents also contained more erroneous references to Surry features (see above) than were actually observed in the final NAPS IPE models.

Several specific technical findings are listed in Section 2 of the final independent review report. These findings are the more significant of the comments from the document review forms. An example is the comment for the Emergency Diesel Generator and Electrical Power Distribution Systems in Section 2.3:

> Operator action is required to reclose the 4160 V stub bus breakers if a CDA signal was generated or if an RHR pump was running. Trees E1H2, E1J2, E2H2, etc., should include the CDA interlock requiring manual action if a CDA signal is expected to occur.

Each of the individual comments have been reviewed by the appropriate PRA team member to determine an appropriate resolution.

Interfaces between the core damage and accident progression analyses have been reviewed by both groups of experts. A portion of the independent review team meeting was dedicated to a discussion of this interface. The SWEC consultant was in attendance for these discussions. A sample of an interface finding is the following comment from Section 2.9 of the independent review report:

> The plant damage states for SGTR sequences P13 through P16 are listed as PDS number 25. This plant damage state apparently includes failure of SG isolation. Yet in sequences P13 through P16, the SG has been isolated.

The independent review for the North Anna Level 2 IPE is similar to that conducted for the Surry plant approximately two years ago. The review of the Surry analysis concentrated on the Level 1 support aspects (success criteria), the Plant Damage States (PDSs), and the Containment Event Trees (CETs). Since these aspects were reviewed in great detail for Surry, and since the North Anna Level 2 IPE is very similar to the Surry Level 2 IPE in regard to those aspects, the North Anna review concentrated on the release categories and source terms. However, all parts of the analysis were reviewed to the level of detail presented in Section 4 and Appendix F of the draft North Anna IPE report.

In addition to the relevant IPE report sections and analysis files, twelve supporting MAAP analyses were reviewed for North Anna, eight of them being steam generator tube rupture (SGTR) analyses (six for Level 1 success criteria and two for Level 2 source terms), and four other analyses for Level 2 source terms. The four "other" analyses are as follows:

- Case 29 - Station Blackout with 200 gpm seal leak, no Auxiliary Feedwater, early Containment rupture
- Case 33 - V Sequence (2.6" cold leg)
- Case 37 - Station Blackout with 200 gpm seal leak, no Auxiliary Feedwater, early Containment leak
- Case 39 - Station Blackout with 200 gpm seal leak, 2" Containment Isolation failure with late rupture

The accident progression analysis review was documented in a similar fashion to the Level I review. Individual review comments were generated during the review of the analysis files and these comments were included in a report which also summarizes the review (SWEC 1992). The significant comments from this review are listed below:

1. Given the importance of SGTR and the use of MAAP to analyze success criteria for these sequences, a

suggestion was made to verify that the MAAP code is capable of these calculations.

2. Caution should be used in applying flooded/unflooded split fractions from NUREG-1150 (developed for Surry) to the North Anna V Sequence analysis. There are many factors: likely break locations, location of below-grade openings to adjacent structures, sump pump location and capacity, etc., which can affect this likelihood. This split is also based on other phenomenological aspects which may be affected by how the above differences affect accident progression. It isn't clear that the North Anna analysis has been sufficiently detailed or plant-unique in this area.

3. The PDS rule for RCS pressure is a unique function of the accident sequence type, i.e., large LOCA, small/medium LOCA or transient. The assignment of CET split fractions related to both direct containment heating and in-vessel steam explosion ("alpha") containment failure modes is related to this pressure/accident sequence type designation. The problem is that the relevant RCS pressure for alpha containment failure mode is best characterized as that at the time of the initial large relocation of core debris into the lower plenum while for direct containment heating (DCH) the relevant pressure that at the time of lower head failure. The two pressures are not necessarily the same, particularly for small/medium LOCAs.

4. The source term information presented on Draft IPE report Tables 4.7.3-2 and 4.7.3-3 is incomplete in that no timing or energy of release information is provided.

5.4 RESOLUTION OF COMMENTS

The comments presented in each of the independent review reports discussed above have been resolved. The process of resolving the comments consisted of the following events. Each comment was assigned to the PRA team member responsible for the development of the model/calculation in question. The resolution of a review comment could consist of either a model change or a discussion of why the comment is not important or applicable. When the resolutions were determined, a review of the resolution was made by the PRA project manager. The review forms were then compiled in an analysis file.

The resolution of the specific comments discussed above in Section 5.3 are presented below in the order they were introduced. The PRA analyst disagreed with the review team regarding the Electrical Power Distribution System comment. The operator action was not felt to be required since the stub bus supplies the CC and RH systems. These systems are only modeled in the SGTR event tree which will not generate a CDA signal.

The review of the interface between the two analyses produced some findings. The response to the sample finding listed above was that these sequences, along with P22 through P28, are lumped together in PDS 25 because the source term impact is about the same.

In reviewing the accident sequence analysis, several comments were offered. The sample comments listed above were resolved as indicated in the following text:

1. The NAPS accident progression analysis considered SGTR cases. The results were consistent with operational data.

2. The North Anna Safeguards Building was evaluated and compared to the Surry Safeguards Building. As a result of this evaluation, the split fraction used in the Surry Analysis was found to be applicable to North Anna. This result is to be expected since the general layout of the buildings is the same.

3. MAAP analysis made for the Surry IPE indicate that the two pressures are reasonably close. The same analyses were performed for North Anna, resulting in the same conclusion.

4. The analysts agreed that the timing and energy of release information should be added to Tables 4.7.3-2 and 4.7.3-3. The tables were updated to include this information.

5.5 REFERENCES

NRC (U.S. Nuclear Regulatory Commission), Individual Plant Examination for Severe Accident Vulnerabilities, 10CFR50.54(f) (Generic Letter 88-20), Washington, D.C., 1988.

SAIC, Holderness, J. H. and Singer, B. S., Virginia Electric and Power Company, North Anna Power Station, Individual Plant Examination Independent Review, September 1992.

SWEC (Stone and Webster Engineering Corporation), Peer Review of the North Anna Nuclear Power Station Level 2 Individual Plant Examination (IPE), September 1992.

TABLE 5-1

INDEPENDENT REVIEW TEAM COMPOSITION

James H. Holderness, SAIC

Blake S. Singer, SAIC

James E. Metcalf, SWEC

James R. Roth, Virginia Power - Corporate Nuclear Safety, Shift Technician Advisor, SRO licensed.

Dave C. Hawkins, Virginia Power - North Anna Operations Department, Operations Coordinator, SRO licensed.

Donald L. Reid, Virginia Power - North Anna Nuclear Safety Engineering, Shift Technical Advisor, SRO licensed.

Robert E. Rink, Virginia Power - North Anna Training Department, CRO/SRO Simulator Instructor, CRO licensed.

Robert M. Garver, Virginia Power - Station Engineering, System Engineer, Shift Technical Advisor qualified.

Ross C. Anderson, Virginia Power - Nuclear Safety Engineering, Shift Technical Advisor.

John W. Daily, Virginia Power - Operations Department, Procedure Writer.

Robert M. Neil, Virginia Power - Corporate Nuclear Safety, North Anna Licensing Engineer.

TABLE 5-2
NORTH ANNA IPE INDEPENDENT REVIEW
ANALYSIS FILE LIST

File Number	Subject
319MAF.N.1	Development of a Generic Database
319MAF.N.2	Development of Plant Specific Data (Vols. I and II)
320MAF.N.1.1	System Modeling - Accumulators
320MAF.N.1.2	System Modeling - HHSI/HHSR
320MAF.N.1.3	System Modeling - LHSI/LHSR
320MAF.N.1.5	System Modeling - SI Actuation
320MAF.N.1.6	System Modeling - CDA
320MAF.N.2	System Modeling - Charging
320MAF.N.3	System Modeling - Quench Spray
320MAF.N.4	System Modeling - Recirculation Spray
320MAF.N.5	System Modeling - Containment Isolation
320MAF.N.6.1	System Modeling - Auxiliary Feedwater
320MAF.N.6.2	System Modeling - Main Feedwater
320MAF.N.7	System Modeling - Instrument Air
320MAF.N.8	System Modeling - Main Steam
320MAF.N.9	System Modeling - Primary System Pressure
320MAF.N.10	System Modeling - Emergency Electrical
320MAF.N.11	System Modeling - Emergency Diesels
320MAF.N.12.1	System Modeling - Reactor Protection
320MAF.N.12.2	System Modeling - AMSAC
320MAF.N.13	System Modeling - Service Water
320MAF.N.14	System Modeling - Component Cooling
320MAF.N.15	System Modeling - Residual Heat Removal
320MAF.N.16	System Modeling - Containment Structure
320MAF.N.17	System Modeling - Ventilation
321MAF.1.N	Accident Sequence Delineation
322MAF.N.1	Development of Success Criteria
322MAF.N.2	Identification of Special Initiators
322MAF.N.3	deleted
322MAF.N.4	Transient Analysis
323MAF.N.1	Common Cause Analysis
324MAF.N.1	Human Error Probabilities
325MAF.N.1	Initial Sequence Quantification (Vols. I and II)
325MAF.N.2	Final Sequence Quantification (Vols. I and II)
325MAF.N.	Recovery Actions

TABLE 5-2 (Continued)
NORTH ANNA IPE INDEPENDENT REVIEW
ANALYSIS FILE LIST

File Number	Subject
326MAF.N.1	MAAP Parameter File
326MAF.N.2	- na -
326MAF.N.3	Plant Damage State Logic
326MAF.N.4	Containment Event Trees
326MAF.N.5	Accident Progression Success Criteria (Vols. I, II and III)
326MAF.N.6	Release Category/Source Terms
326MAF.N.7	MAAP Level 2 Analyses
326MAF.N.8	Level 2 Sensitivity Studies
327MAF.N.1	Analysis of Internal Flooding (Vols. I and II)
328MAF.N.1	Level I Sensitivity Analysis

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FIGURE 5-1
PROJECT ORGANIZATION CHART

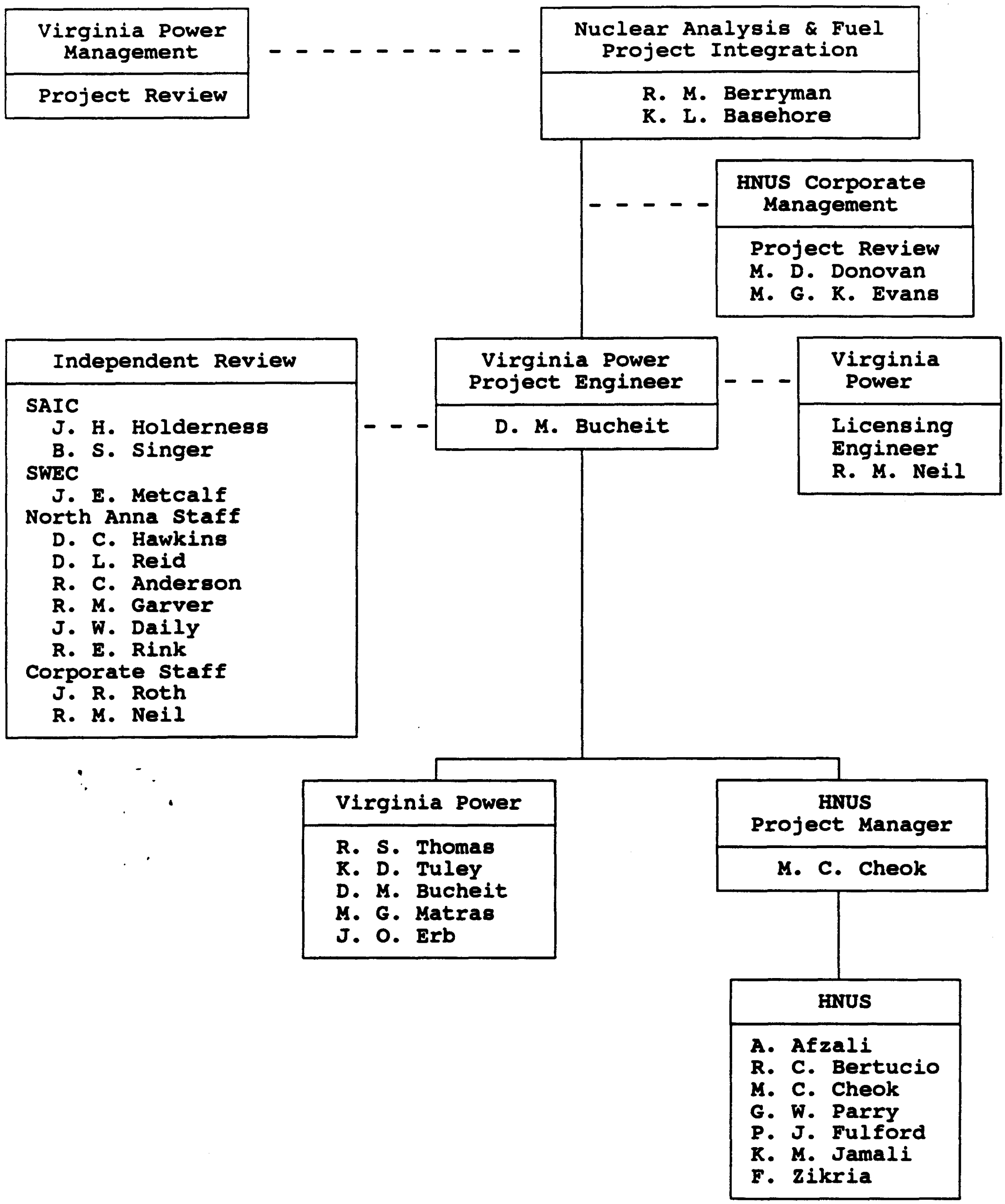

FIGURE 5-2
PROJECT TECHNICAL ORGANIZATION