MASTER

400-MWe CONSOLIDATED NUCLEAR STEAM SYSTEM (CNSS)

— 1200-MWt PHASE 2A INTERIM STUDIES —

BAW-1508
ORNL/Sub-4390/6
September 1978

Babcock & Wilcox

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— 1200-MWt Phase 2A Interim Studies —

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BABCOCK & WILCOX
Power Generation Group
Nuclear Power Generation Division
P. O. Box 1260
Lynchburg, Virginia 24505

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1. SUMMARY

The Phase 2A interim studies of the Consolidated Nuclear Steam System (CNSS) consisted of a number of separate task studies addressing the design concepts developed during the Phase 1 study reported in BAW-1445.* The purpose of the interim studies was to better establish overall concept feasibility from both a hardware and economic standpoint, to make modification and additions to the design where appropriate, and to understand and reduce the technical risks in critical areas of the design.

The work on these task studies included input from Barberton, Mt. Vernon, and the Alliance Research Center as well as United Engineers and Constructors (UE&C). The UE&C work was carried out under a separate DOE contract.

Sections 3 through 8 of the report cover task studies which were included in a draft report issued in November 1977 under Subcontract 4390 - Supplemental Agreement No. 6. Sections 9 through 11 of the report cover the task studies which were included in a draft report issued in July 1978 under Subcontract 4390 - Supplemental Agreement No. 6.

The important results from these task studies are given below.

1. The single integral steam generator developed in the Phase 1 study has been replaced by 10 integral steam generator modules similar in design to those used in the Maritime CNSG design.
2. The pressure suppression containment considered in the Phase 1 study has been replaced by a dry containment with no significant changes in the costs or construction time reported in the Phase 1 study.
3. Using current regulatory siting criteria, the plant arrangement may permit a greater degree of close-in siting.

* 400-MWe Consolidated Nuclear Steam System (CNSS) — 1200-MWt/Conceptual Design, BAW-1445, ORNL/Sub-4390/4, Babcock & Wilcox, Lynchburg, Virginia, June 1977.

4. A reactor coolant system pump with integral flywheel was found to be feasible.
5. Although not included in these task studies, experts on steam generator technology within B&W have reviewed the secondary system operating mode proposed for the CNSS ("assisted" recirculation with exit steam at 80% quality). They have concluded that the assisted recirculation mode of operation of these units may result in tube corrosion problems, and that a once-through or high-recirculation-ratio mode of operation should be adopted to preclude or minimize this problem. A once-through operational mode appears feasible and attractive and is currently being further evaluated.

2. INTRODUCTION

The Phase 2A Interim Studies are a continuation of the Phase 1 conceptual design study performed by the Babcock & Wilcox Company (B&W) under contract to Union Carbide Corporation (Oak Ridge National Laboratory) for the United States Department of Energy (DOE). In these studies B&W worked closely with United Engineers and Constructors (UE&C) in a continuation of the Phase 1 cooperative studies. UE&C worked under a separate DOE contract. The Phase 1 work was reported in ORNL/Sub-4390/4.* Its objective was to develop a conceptual design for an intermediate-sized nuclear unit (1200 Mwt) suitable for dual-purpose industrial steam/electric power generation use, for use as a thermal grid energy source, or for electric power generation.

The work reported here consists of a number of separate tasks addressing various aspects of the CNSS design concept in order to better establish overall concept feasibility. This work also provides added assurance that the technical risk associated with the concept is better understood in critical areas of design, and that technical risk in undertaking further development is lessened. The CNSS concept presents a novel configuration of the conventional loop-type pressurized water reactor (PWR) which permits a high degree of modularization and shop-fabrication standardization of the reactor and the reactor/containment complex. This is related directly to the minimal pipe sizes of the reactor coolant system external to the reactor vessel, and the resultant nominal transient resulting from a design basis loss-of-coolant accident (LOCA).

Previous studies have shown that the modular nature of the CNSS assures maximum shop fabrication input under strict shop quality assurance procedures, and minimizes field erection schedule and associated interest during construction and escalation expense. More conventional PWR designs do not demonstrate

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* 400-MWe Consolidated Nuclear Steam System (CNSS) — 1200 Mwt/Conceptual Design, BAW-1445. ORNL/Sub-4390/4, Babcock & Wilcox, Lynchburg, Virginia, June 1977.

these characteristics. B&W market survey and market contact activities have demonstrated a high degree of utility interest in an intermediate-sized nuclear unit having these characteristics, assuming the economics and construction schedule resulting from our joint studies can be achieved.

The prospective development program for the CNSS unit currently being reviewed within the Babcock & Wilcox Company is structured as a private-enterprise program, with elements of utility, industry, and government (DOE) support. It is intended to be a program of common NRC licensing and first-cost financing by a group of utilities, with B&W ultimately bearing all development expense except for portions of engineering tests and evaluation programs, which will be shared with government and industry groups if the proposed program format is approved and accepted. The tasks reported here and the additional subcontract work studies currently proposed are antecedent to and necessary to the development of this overall program plan. Although the tasks discussed appear as independent work efforts, each bears a direct relationship to the overall plant for CNSS plant development.

3. TASK 1 — PRELIMINARY REACTOR VESSEL STRESS ANALYSIS

3.1. Summary

A preliminary finite element analysis indicated acceptable stress levels in the Phase 1 CNSS concept arrangement. However, this task was de-emphasized after the selection of the modular steam generator concept in the Task 3 work activity. An equivalent additional scope of work was completed in sizing a reactor vessel to house the modular steam generators and the reactor coolant pumps (Figure 3-1 and 3-2).

3.2. Objectives and Scope

The objective of this task was to show that the single steam generator integral with the reactor vessel was acceptable from a stress standpoint. The unusual configuration of the reactor vessel with the integral steam generator shell and their interrelationships required a preliminary stress analysis. The resultant stresses were to be compared to code allowables to indicate permissible operational parameters and design configuration acceptability.

3.3. Results and Conclusions

In the initial scope of work, this task was important to the single annular steam generator design since the annular tubesheets integral with the reactor vessel shell appeared to have some unique stress problems. Although other steam generator design concepts were being studied under Task 3, it was decided to proceed with the reactor vessel analysis in case alternate steam generator concepts proved undesirable.

The reactor vessel analysis effort concentrated on the interaction of the tubes with the primary and secondary shell. In particular, the annular tubesheets were sized and the temperature distribution was determined throughout the heat exchanger region. This preliminary study indicated that the stresses in the structure were acceptable. It was at this stage of the analysis that the modular steam generator concept was selected as the reference design and further analysis of the annular concept was discontinued.

The design effort for this task was redirected toward sizing the reactor vessel to house the modular steam generators, core, and reactor coolant pumps in the closure head. Sizing calculations and layout drawings indicated that the modules would fit inside a 195-inch ID reactor vessel. Except for an additional length requirement, this is the reactor vessel size developed for a B&W standard 241 fuel assembly plant design several years ago. Licensing of that NSS design has since been suspended at B&W request in favor of our current 205 fuel assembly standard design.

The larger reactor vessel posed possible problems with the erection procedure and schedule developed by UE&C during the Phase 1 study. Therefore, special emphasis was given to minimizing the reactor vessel weight and size. Because of the reactor vessel lift limitations and erection procedures in the field, the steam generator module was designed to permit installation after the reactor vessel is set in place. Another weight saving idea was to use welded inserts for the interface between the reactor vessel and module rather than machining the opening out of a heavy wall forging. This effort plus investigations by UE&C with their hoist vendor indicate that the reactor vessel weight was within lift capability. Shop fabrication and shipping of the reactor vessel was also reviewed and found acceptable.

The eight wet rotor reactor coolant pumps located in the closure head are based on conceptual studies only. The pump inertia requirements for coastdown are unknown at this time and several continuing Phase 2A tasks have been proposed to help resolve this question. The congestion and space limitations in the closure head and maintenance considerations of the reactor coolant components would require substantial follow-on work activities with the pump design to determine the accessibility in this region.

Figure 3-1. Reactor Vessel — Elevation View

Figure 3-2. Reactor Vessel — Sectional Plan View

4. TASK 2 — SHOP FABRICATION PROCESS/ SPANS/BUDGETARY COSTS

4.1. Summary

Preliminary cost estimates for the modified reactor coolant system components were about \$3.4 million higher than estimated during the Phase 1 study. The fabrication span times for these components were essentially the same as in the Phase 1 study so that the short construction schedule for the plant was maintained.

4.2. Objectives and Scope

The objective of this task was to develop more detailed shop estimates to confirm the schedules and cost estimates used in the Phase 1 study. During conceptual design, shop spans, and fabrication schedules were estimated based on utility reactor component fabrication experience and CNSG shop estimates. A more detailed shop budgetary-scope estimate for the concept was required in order to confirm schedule and cost estimates and resulting economics of the conceptual design study.

4.3. Results and Conclusions

New budgetary cost estimates and fabrication span times were required for the reactor coolant components as a result of selecting the modular steam generator concept as the reference CNSS design. The modular steam generator concept requires larger reactor coolant components and, therefore, an increase in cost. For the major components: reactor vessel, steam generator modules, reactor vessel internals, and pressurizer; the preliminary cost estimate increased by about \$3.4 million not including first-of-a-kind manufacturing costs. This increase is considered reasonable for the design modification resulting from Tasks 1 and 3.

A reactor vessel fabrication span time of 50 months was reported in the Phase 1 study. This is somewhat longer than the "standard" reactor vessel shop span

time of about 36 months, because the single steam generator integral with the reactor vessel was on the critical path. The reactor vessel fabrication span time for the modular steam generator concept is estimated to be 46 months, but additional time will be required to install the steam generator modules after the reactor vessel is set in place. Assuming the earlier delivery will be offset by the assembly requirements, the overall span times for the Phase 1 and Phase 2A study concepts should be about the same. Some reduction in the 46 months span time can be expected for nth-of-a-kind (NOAK) units which should improve the overall plant construction sequence.

5. TASK 3 — STEAM GENERATOR SUBDIVISION/ REPLACEMENT FEASIBILITY

5.1. Summary

The selection of modular steam generators (Figure 5-1) with larger reactor vessel resulted in a major modification to the design concept permitting isolation of individual modules and continued plant operation if required. Replacement of modules was also considered feasible. Although the sizes of the reactor coolant system components were modified, the erection techniques developed in the Phase 1 study are still applicable.

5.2. Objectives and Scope

The purpose of this task was to study the feasibility of design modifications to the single integral steam generator to permit subdividing the steam generator into independent steam generating sections for added redundancy and to permit isolation capability if steam generator tube leak occurred. Also included in the study was the feasibility of modifications to permit removal/replacement of steam generator heat exchanger surface. Steam generator subdivision or replaceability, while not absolutely essential to the concept, would tend to increase the marketability and operational flexibility of the concept.

5.3. Results and Conclusions

The steam generator selected during the Phase 1 study was of the single annular type with tubesheets integral with the reactor vessel shell. A review of this design concept during the early stages of this task study indicated that subdividing and/or replacing the annular steam generator heating surface was not practical and that this aspect of the design was marginal with respect to steam generator maintenance and reactor availability. Attention was then directed toward using modular steam generators similar to the design developed for the CNSG Maritime Program, but with modules which could be removed if necessary.

The initial concern with the CNSS design modification was that the reactor vessel size would become too large to house the required steam generator heating surface, because modular steam generators could not be fitted geometrically into the annular region as efficiently. A preliminary thermal hydraulic analysis was performed to provide flow and thermal parameters necessary to determine the steam generator heating surface and module size and number. Supporting studies in Task 1 indicated that these modules would fit in the 195-inch ID reactor vessel developed several years ago for the B&W standard 241 fuel assembly plant design if saturated steam conditions were used as a design basis. One advantage of using saturated steam conditions rather than superheat is that less steam generator heating surface is required thus permitting the use of the 195-inch ID reactor vessel.

The modular concept therefore met the primary objective of the task and was chosen as the reference configuration for follow-on design studies. Table 5-1 provides a comparison of the design parameters between the Phase 1 and 2A studies.

5.3.1. Steam Generator Module Description

The steam generator module is a vertical straight tube and shell heat exchanger located within the reactor vessel. The shell is a right circular cylinder closed at each end by tubesheets and containing inconel tubes welded to the tubesheets. The shell and tubesheet material is mag-moly which is clad with stainless steel on the external surface since this surface is exposed to primary fluid. Overall dimensions of the module are shown in Figure 5-1.

Ten modules are spaced adjacent to each other above the core in the annulus between the outside of the internals plenum cylinder and the inside of the reactor vessel shell. The secondary side nozzle at the upper end of each module extends through and is secured from the outside to individual reactor vessel nozzles by bolting and seal welding. This provides the vertical and lateral support for the module; and additional lateral support is also provided at the lower end of the module. External housings are bolted to the reactor vessel nozzles to close off the secondary side and provide feedwater inlet and steam outlet connections for each module. Subcooled feedwater is carried to the bottom of the module in a pipe inside of an essentially triangular chamber. Saturated steam exits at the top of the module and flows to moisture separating equipment.

5.3.2. Steam Generator Operation

The saturated steam mode of operation was selected in order to minimize steam generator and reactor vessel height, and the height and weight of related components plus other advantages. This concept is based on dividing the 10 modules into two groups of five, each group feeding a moisture separating component which is located outside the containment building. Either group can be operated independently. Each module includes an isolation valve(s) so that a module with a tube leak can be valved out of service.

Two options are available for system design and control over the load range. Each requires essentially the same steam generator heating surface and major component dimensions. These options are the "assisted recirculation" and the "zero superheat" method of control. Follow-on studies would be required to select the optimum method of control.

For the present study, the "assisted recirculation" scheme was selected because it incorporated operation and control means quite similar to fossil fueled experience using essentially state-of-the-art technology. The basis of this concept is operation of each steam generator at a specified quality level (60 to 80%) at 100% power with pumped recirculation of water from the moisture-separator reservoir to be mixed with main feedwater upstream of the steam generator. The recirculated flow would be essentially constant volume over the load range resulting in much lower quality at the steam generator outlet at lower loads.

5.3.3. Steam Generator Isolation

A major advantage of the modular steam generator concept is that immediate reactor shutdown is not required in the event of a primary-to-secondary steam generator tube leak. For this case, the module with the leak is valved out of service allowing continued operation with a small reduction in power output until such time as plant shutdown is desirable. The secondary side, up to and including the isolation valve(s), is designed for primary conditions to permit this mode of operation. Locating tube leaks requires that a method be developed for monitoring leaks from each module. Methods such as radiation monitors or flow rate measurements should be investigated during preliminary design studies.

Continued reactor operation with one or more steam generator modules out of service requires that a fluid mixer be installed in the reactor vessel below each module to reduce core inlet temperature upsets that result from unbalanced load distribution among the modules. The mixer consists of a set of turning vanes which will turn half of the fluid in one direction and half in the opposite direction, thus producing turbulent eddies and adequate mixing. Model flow tests for the CNSG (maritime) module/mixer of equivalent design has confirmed that the fluid mixers are required and that they will maintain core inlet temperature upsets within acceptable limits.

5.3.4. Steam Generator Replacement

Replacement of modules may be an option where tubing has deteriorated under abnormal service conditions.

As shown in Figure 5-2, the steam generator modules are bolted and seal welded rather than welded into the reactor vessel nozzles. This provides for improved maintenance capability of the modules and also maintains reactor vessel shipping and erection weights within acceptable limits since the modules will be shipped separately from the vessel and installed after the reactor vessel is in place.

The steam generator module can be removed from the reactor vessel after removing the internals plenum assembly. The external secondary side module housing is then unbolted and removed from the reactor vessel to expose the module-to-reactor vessel bolts and nozzle seal weld. After cutting the seal weld, and unbolting, the module can be translated laterally to the center of the reactor vessel and then lifted out of the vessel.

Table 5-1. Design Parameter Comparison Between
Phase 1 and Phase 2 Study

	CNSS	
	1	2A
<u>Hydraulic and Thermal Design</u>		
Core heat output, MWt	1200	1200
Steam pressure, psia	2250	2250
Total primary flow, 10 ⁶ lb/h	57	67.5
SG outlet temperature, F	571	572.3
SG average temperature, F	596.5	593.3
SG inlet temperature, F	622	615.5
Core average thermal output, kW/ft	5.79	4.97
<u>Reactor Vessel Design</u>		
Primary volume, ft ³	4735	6982
Design pressure, psig	2500	2500
Design temperature, F	650	650
Shell ID, in.	182	195
Height inside heads, ft-in.	43-6	48-10
<u>Steam Generator</u>		
Net steam flow, 10 ⁶ lb/h	5.29	5.29
Steam pressure, psia	935	935
Steam temperature, F	Sat.	Sat.
Feedwater temperature, F	440	440
Tube length, ft	15	20
Tube OD, in.	0.5	0.5
Number of tubes (total)	30,807	20,460
Number of modules	1	10
<u>Core Design</u>		
Core height (active), ft	10	10
Number fuel assemblies	97	89
Number fuel rods	21,136	23,496
Fuel rod OD/pitch, in.	0.430/0.568	0.379/0.501
Number control rod assemblies	37	37
Number control pins/assemblies	16	24

Figure 5-1. Steam Generator Module

Figure 5-2. Steam Generator/Reactor Vessel Bolted Connection

6. TASK 4 — CONTAINMENT STUDY

6.1. Summary

A dry containment size slightly larger than the pressure suppression containment was found to be feasible. Also, this design retains the short construction schedule and erection methods developed during the Phase 1 study. The dry containment was adopted as the reference design as a result of this study.

6.2. Objectives and Scope

The objective of this task was to study the dry containment as an alternate to the pressure-suppression containment. The maritime reactor design which was adopted as the design basis for the 365-MWt Process Energy — Consolidated Nuclear Steam Generator (PE-CNSG) reactor, incorporated a close-coupled pressure-suppression containment system. This containment type was retained in the development of the CNSS concept so that the schedule and erection techniques previously worked out could be retained. Since heavy first-of-a-kind (FOAK) costs in development and testing of the pressure-suppression containment and potential regulatory delays were anticipated, the study of a dry containment concept was believed to be warranted.

6.3. Results and Conclusions

6.3.1. Introduction

In 1976, UE&C in conjunction with B&W, developed for ERDA a preliminary cost estimate for a 400-MWe CNSS for the hypothetical Middletown site. These estimates were based primarily on the cost estimates developed previously for a 110-MWe CNSG and reflected a pressure-suppression containment concept. A free standing steel containment (approximately 45-ft ID, 100-ft high with a 1½-inch wall thickness) based on a conservative design pressure of 105 psig was used. This was surrounded by a 3-foot thick reinforced concrete structure which served as a biological shield and missile barrier. Use of this containment

concept combined with construction techniques described in BAW-1445* resulted in a favorable overall schedule of about 6 to 7 years for nth-of-a-kind (NOAK) plants.

While the overall schedule of about 7 years is very attractive as compared to about 10 years for a typical PWR plant, the pressure-suppression containment concept would have substantial development costs associated with it. It was, therefore, considered desirable to evaluate dry containment concepts, the objective being to identify a dry containment design that would lend itself to accelerated erection in the field with minimum interference with site work.

6.3.2. Conclusion

A dry containment design that can be developed for accelerated erection in the field is found to be feasible. A free standing steel dry containment with 50-ft inside diameter and 100-ft springline height is recommended. This structure will be surrounded by a 4-ft thick reinforced concrete structure serving the dual purpose of biological shield and missile barrier. This concept retains the short construction schedule previously estimated for the CNSS.

6.3.3. Containment Peak Pressure Vs Free Volume

Containment peak pressure Vs free volume curve supplied by B&W (Figure 6-1) is based on a double-ended break for the 4-inch ID pressurizer surge line. This is the largest diameter primary fluid carrying pipeline in the containment subject to rupture. Initial conditions used in the containment were as follows:

Initial pressure = 15.2 psig
Initial temperature = 120F
Relative humidity = 30%

A containment spray system which would possibly reduce the peak pressure was not used in the analysis to obtain the peak pressure curve shown in Figure 6-1. Consideration for air cooler operation is included in the curve without which the pressure would be higher by approximately 0.5 psi.

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*400-MWe Consolidated Nuclear Steam System (CNSS) — 1200-MWt/Conceptual Design, BAW-1445, ORNL/Sub-4390/4, Babcock & Wilcox, Lynchburg, Virginia, June 1977.

The peak pressure curve shown in Figure 6-1 is based on 100% surface area of internal structures as defined in Table 6-1. The total surface area shown in this table is approximately 500,000 ft² and is based on the 3600-MWt standard plant data (except for item 1 which is varied in accordance with the free volume). Figures 6-2 and 6-3 give the multipliers to be applied to peak pressures obtained from Figure 6-1 as a function of the surface area of internal structures. Figure 6-2 is based on a containment free volume of 1,000,000 ft³ and will give higher than actual multipliers for higher free volumes. Figure 6-3 is based on a containment free volume of 200,000 ft³ and will give higher than actual multipliers for higher free volumes.

6.3.4. Equipment in Dry Containment

Table 6-2 shows a list of equipment and associated volumes in the CNSS dry containment. The largest item is the reactor vessel which comprises over half the total equipment volume. It includes the reactor coolant pumps and the steam generators. The small maximum credible break is believed to eliminate the need for core flood tanks. The polar crane and hoist are not included because they service the containment building area from the outside as part of the reactor service building. The rest of the equipment listed is mostly the same as the standard equipment in most dry containments for central station nuclear power plants.

The list of equipment shown in Table 6-2 is not represented to be an exhaustive list and future studies may provide for additional equipment. Most major pieces of equipment are, however, shown in Table 6-2 and some additional equipment will not have a significant impact on the results of this study.

6.3.5. Evaluation of Alternate Dry Containments

The most common dry containment configurations currently being employed are:

1. Parallelepiped.
2. Steel cylinder.
3. Reinforced concrete steel lined cylinder.
4. Prestressed concrete steel lined cylinder.
5. Steel sphere.

Various design features, such as the use of subatmospheric containment or secondary containment, can be associated with the above containment types.

To the extent achievable, the various types of dry containment concepts were evaluated with the primary objective of selecting one that can be developed for accelerated erection in the field with minimum interference with site work. The free standing steel containment is believed to be a leading contender for meeting the stated objective. Further analysis, as discussed below, shows that the size of a free standing steel containment could be kept within the limits necessary to achieve the shorter construction schedule advantages of the CNSS. For these reasons, a free standing steel containment was selected over other containment types.

6.3.6. Analysis of Various Sizes of Free Standing Steel Containment

This section presents an analysis of various sizes for a free standing steel containment. Table 6-3 shows the surface area of internal structures, free volumes, and design pressures associated with various sizes of free standing steel containment. A 50-ft inside diameter and a 100-ft springline height was taken as about the minimum size. This was based on layout considerations for equipment located in the containment. Also, a hemi-ellipsoidal dome with reinforced flanges was selected to minimize the size of the reactor service building which houses the containment.

In Table 6-3 the surface area of internal structures and free volume of containment were estimated for each size selected. Based on this free volume, a peak pressure was obtained from Figure 6-1. This peak pressure is adjusted by the surface area multiplier obtained from Figures 6-2 or 6-3 depending on whether the free volume is closer to 200,000 ft³ or 1,000,000 ft³, respectively. The design pressure for the containment was taken as 15% higher than the adjusted peak pressure to allow for theoretical uncertainties and safety margins.

This results in a design pressure of about 65 psig for a containment 50-ft ID and 100-ft springline height. This results in a containment wall thickness of less than 1½-inches which eliminates the need for stress relieving of field welds. Since the smallest containment size of 50-ft ID and 100-ft springline height meets the primary objective of containment selection, this is the recommended containment size for the CNSS.

The containment size selected is only slightly larger than that for a pressure-suppression containment evaluated in the previous study. This was unexpected,

but B&W believes that the design pressure of 105 psig used for the pressure-suppression containment was overly conservative and resulted from scaling factors applied to a much smaller unit. The methods and assumptions used in the containment pressurization analysis are in compliance with the rules set forth in the NRC Standard Review Plan for the dry containment DBA analysis. The CONTEMPT computer program (B&W version) was used with the Uchida heat transfer coefficients and the initial conditions given in section 6.3.3. The mass and energy release rates based on the 3600-MWt PWR were used in the pressurization analysis; however, the calculated peak pressure is adjusted to account for the reduction in mass and energy releases to the containment for the 1200-MWt CNSS. (Table 6-4 compares the Phase 1 and 2A containment.)

The recommended free standing steel containment will be surrounded by a 4-ft-thick reinforced concrete structure serving the dual purpose of biological shield and missile barrier.

6.3.7. Construction Schedule

The construction schedule for the 400-MWe CNSS is based on that developed for the CNSS (with a pressure-suppression containment) in a previous study. This schedule, shown in Figure 6-4, has been modified to reflect a dry containment for the CNSS. Only the schedule for the reactor service building has changed, the rest of the schedule remaining essentially the same. The overall construction schedule is still the same as developed in the previous study.

B&W is now considering a CNSS utilizing a 195-inch ID reactor pressure vessel (RPV) with 10 modular steam generators separately installed at the site. The estimated RPV weight (without internals and steam generator modules) is 1,017,000 lb with an inside diameter of 195-inches and an inside height of 48-ft, 10-in. In the previous concept, the CNSS utilized a 3800-MWt RPV with integral steam generators and internals to be installed later. The estimated weight, inside diameter, and inside height were 883,244 lb, 182 in. and 43-ft 6-in., respectively.

The increase in reactor vessel diameter and height are reflected in the design of the dry containment for the CNSS. The increase in weight (about 67 tons) led to a re-evaluation of the lifting capacity of American Hoist's S-80 Stiffleg Derrick to carry the heavier load.

To put the problem in perspective, American Hoist's chart ratings based on a 90° sill angle using 85-ft sills, do not include the weight of the tackle (anything hanging off boom tip: blocks, wire rope, etc.) which must be deducted from the rated capacity. The tackle weight could range from 60,000 to 100,000 lb, depending on the size of hook and blocks, number of parts, and feet of cable (such as requirement to go "x" feet below grade). Added to the base load to be lifted are the rigging and the upending device ("J" frame, rocker or trunnion) particularly applicable if the total assembly is lifted before the device is removed.

Based on these ground rules and the data provided by B&W, the approximate test load can be estimated as follows:

	<u>Pounds</u>
RPV weight (excluding steam generators and internals)	1,017,000
5% allowance for rigging	51,000
Allowance for upending service	<u>60,000</u>
Subtotal	1,128,000
Add 25% for test load	<u>282,000</u>
Approximate total test load	1,410,000

The maximum capacity shown on the S-80 charts is 1,400,000 lb at a 55-ft radius with a 160-ft boom. After making an allowance for the weight of the tackle, the maximum capacity is reduced to approximately 1,300,000 to 1,340,000 lb which is lower than the test load shown above.

If the upending device weight does not have to be included as part of the test load, the test load would be approximately 1,335,000 lb which would seem to be in the range of the maximum capacity shown above. However, the actual lift would be at a radius of greater than 55-ft. It appears marginal, therefore, whether the crane can be fully qualified to make the test lift and traverse through the path of the actual lift required.

American Hoist claims the crane capacity can be increased in several ways, as follows:

1. Change sill angle to 60° . This increases capacity by about 15% at 65-ft radius, 16% at 70 ft, 14% at 80 ft and 16% at 90 ft.
2. The crane can be put on a fixed tower (as opposed to the movable gantry) and the sill length changed to 90 ft. This increases capacity by 5.88%.
3. A larger crane could possibly be considered. It may also be possible to reduce the weight of the boom.

Based on this, if the sill angle is changed to 60° (sill length kept at 85 ft), the crane can handle about 1,350,000 lb net (after deducting the weight of the tackle) at a 70-ft radius and about 1,400,000 lb net at a 65-ft radius, with a 195-ft boom which is required if the crane is to service most of the reactor service building. This would allow for a test load at the 65-ft radius and an actual lift at the 70-ft radius. It should be noted that the test load is still at the limit of the crane capacity at the 65-ft radius.

While points 2 and 3 allow for some further increase in crane capacity, we believe other factors should be considered before making use of those techniques. For example, if the sill length is increased to 90 ft to increase capacity by 5.88%, a fixed tower will be utilized. This may not be a significant penalty as only one unit is to be built at a site. However, if more than one unit is to be built at a site, a gantry mounted crane is preferred because it permits the same crane to move between the units. Another advantage is that the gantry permits assembly and disassembly away from the work area. This allows earlier assembly if the foundations are not in, and safer disassembly in an area removed from the finished structures.

In summary, it seems that the S-80 with a 60° sill angle will meet the requirements. The test load, however, will be at the limit of the crane capacity at about the 65-ft radius. The only way to answer this problem with a high degree of assurance is to know the weights involved more accurately and to assess the corresponding changes in plant layout and their effect on the required radius and boom length for the crane. For the purposes of construction schedule shown here, it is assumed that the S-80 Stiffleg Derrick with a 60° sill angle will be able to meet the requirements.

The following are the additional assumptions implicit in the construction schedule shown:

1. A commitment to engineer, procure, and fabricate the containment provides no restraint. In other words, whatever lead time is required to meet field need dates, that commitment to gamble on receipt of NRC approval for construction will be, if necessary, made by the owner.
2. Schedule indicates in-situ durations (no unloading, storing, etc.).
3. Stiffleg Derrick on tower will be used to handle heavy lifts.
4. The containment vessel is fabricated off-site and delivered in the following sections:
 - a. Four-inch bottom plate with shell extension and any other weldments to 4-inch plate are to be shop welded and stress-relieved in shop. Extensions are to go up a certain number of feet (to be determined in a detailed study) so that automatic welding equipment can be used for circumferential welds.
 - b. Barrel section of the containment vessel in two pieces to the spring-line with shipping stiffeners.
 - c. Dome and head section of containment vessel.
5. Pressurizer and quench tank are supported from the 4-inch base plate. Should horizontal supports to the containment shell be necessary to provide lateral stiffness for these structural supports, the stiffeners will be welded to the shell prior to the air test.
6. Containment vessel is tested using air.
7. Reactor vessel is set after containment test.

The schedule savings are made possible by a reactor with integral prefit steam generators, a free standing containment that can be mostly fabricated offsite, and a derrick strategically located to handle the major lifts and service the main construction areas from one location. The effect on plant cost due to changes in the lift requirements were not considered in this study.

Table 6-1. Passive Heat Sinks

<u>Slab description</u>	<u>Exposed surface area, ft²</u>	<u>Material</u>	<u>Material thickness, ft</u>
Containment wall	97,120.0	Paint Steel Concrete	0.00083 0.02083 3.479
Containment dome	21,200.0	Paint Steel Concrete	0.00083 0.02083 2.979
Floor	20,920.0	Paint Concrete Steel Concrete	0.00083 3.0 0.02083 3.0
Internal concrete	127,880.0	Paint Concrete	0.00083 1.75
Equipment hatch & personnel locks	400.0	Paint Steel	0.00083 0.1667
Crane grider with trolley, floor framing, & columns	120,000.0	Paint Steel	0.00083 0.0417
Stairs	5,000.0	Paint Steel	0.00083 0.0104
Duct work & piping (1/8-inch thickness)	47,323.0	Steel	0.0104
Duct work & piping (1/4-inch thickness)	8,522.0	Steel	0.02083
Duct work & piping (1/2-inch thickness)	9,916.0	Steel	0.0417
Miscellaneous steel	52,000.0	Paint Steel	0.00083 0.02083

**Table 6-2. Equipment and Associated Volumes
in CNSS Dry Containment**

<u>Equipment</u>	<u>Volume, ft³</u>
Reactor vessel	10,127
Spray headers	50
Fan cooler(s)	200
Ducts for heating and/or cooling	50
Pressurizer	1,778
Pressurizer relief tank	2,000
Instrument racks	100
Personnel and equipment elevators and hatches	1,000
Containment sump pumps	100
Air compressors	200
Containment filter units	200
Electrical conduits and control panels	200
Hydraulic power package	<u>100</u>
Subtotal	16,100
Contingency	<u>3,900</u>
Total design equipment volume	20,000

Table 6-3. Containment Parameters for Various Sizes of Free Standing Steel Containment

Inside diameter, ft	50	60	70	80	90	60	70	80
Springline height, ft	100	100	110	120	130	120	130	100
Wall surface, ft ²	15,710	18,850	24,200	30,150	36,760	22,600	28,600	25,150
Dome surface, ft ² (a)	2,945	4,240	5,770	7,540	9,540	4,240	5,770	7,540
Floor surface, ft ²	1,965	2,830	3,850	5,027	6,360	2,830	3,850	5,027
Inside volume ft ³ (b)	212,700	311,000	468,200	670,300	922,200	367,900	545,400	570,000
Inside equipment volume ft ³	20,000	20,000	20,000	20,000	20,000	20,000	20,000	20,000
Volume of concrete walls ft ³	18,000	20,000	25,000	30,000	35,000	22,000	27,000	28,000
Containment free volume	175,000	271,000	423,000	620,000	867,000	326,000	498,000	522,000
Peak pressure, psig	21.6	19	16.5	14.4	12.2	18	15.5	15.3
Surface area of internal structures, %	14.8	18.6	24.3	30.7	37.8	21.3	27.4	27.1
Surface area multiplier	2.6	2.34	2.11	1.96 1.29	1.8 1.25	2.21	2.03	2.04 1.31
Adjusted peak pressure, psig	56.2	44.5	34.8	23.4	18.6	39.8	31.5	25.6
Design pressure, 15% margin	64.6	51.2	40.0	26.9	21.4	45.8	36.2	29.4

(a) Dome surface = $1.5 \times$ floor surface.

(b) Dome surface = $0.5 \times$ hemispherical dome volume.

Table 6-4. Containment Size Comparison for
1200 MWt CNSS

Type	CNSS Phase 1	CNSS Phase 2A
	Pressure suppression, free standing steel	Dry, free standing steel
Inside diameter, ft	45	50
Height – springline, ft	90	100
Shell thickness, in.	< 1½	< 1½
Design pressure, psig	105	65

Figure 6-1. Containment Peak Pressure Vs Free
Volumed (Based on 100% Surface
Area of Internal Structures)

Figure 6-2. Pressure Multiplier Vs Surface Area of Internal Structures
(Containment Free Volume = 1,000,000 ft³)

Figure 6-3. Pressure Multiplier Vs Surface Area of Internal Structures
(Containment Free Volume = 200,000 ft³)

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7. TASK 5 — STANDARD BALANCE-OF-PLANT REQUIREMENTS

7.1. Summary

The division of responsibility (DOR) scope and balance of plant (BOP) requirements documents were developed and forwarded to UE&C for comment and agreement. It is expected that future work in the preliminary design phase be compatible with these requirements.

Also included in this task was the development of information required by UE&C, such as the weight and size of the modified reactor coolant system components to aid in their Task 4 studies.

7.2. Objectives and Scope

The objective of this task was to interface with the A/E regarding: (1) typical division of responsibility documents, (2) typical scope and content of balance-of-plant requirements, and (3) equipment weight and sizes.

This portion of the Phase 2A study consisted of three areas which are important to plant designers before considering the preliminary design phase (Phase 2B). These areas are to: (1) participate with the A/E in developing a typical Division of Responsibility (DOR) document for design and procurement, (2) identify to the A/E typical scope and content of BOP requirements to ensure interface compatibility between the CNSS and balance-of-plant, and (3) provide the A/E with a preliminary listing of typical equipment weight, sizes, etc., for those items which are provided under the CNSS vendor scope of supply so that the A/E can reassess his construction schedule. Where possible, available results from other studies underway in Phase 2A will be factored into the efforts described above.

In a preliminary design phase, it is expected that the work performed by the A/E would be compatible with these requirements, and these requirements would be the basis for any later A/E proposal activity for a demonstration plant.

7.3. Results and Conclusions

The Division of Responsibility Document (DOR) was prepared by B&W and forwarded to UE&C for review and comment. This document defines the organization responsible for preliminary, or detail design, and specifications and procurement. The DOR document attached to this report as Appendix B incorporates the modifications recommended by UE&C.

Technical changes were made to the B&W NSS balance-of-plant (BOP) criteria documents to make these documents compatible with the CNSS design. A listing of these documents is shown in Table 7-1. This listing was reviewed by UE&C and Table 7-2 shows the suggested additional systems where the input responsibility is either by B&W or shared with the A/E.

Also included in this task was the development of typical equipment, sizes, weights, surface areas, and etc., for the equipment included under the B&W scope of supply. This information was forwarded to UE&C for the Task 4 study to size the containment and reassess the erection schedule reported under the Phase 1 study.

Table 7-1. B&W CNSS Balance-of-Plant Criteria Documents

- Automatic Sump Switchover Instrumentation
- Protective Coatings
- Plant Electric Systems
- Recommended Design Considerations for a CRD Equipment and Service Room
- RCS Makeup and Initial Fill
- The Spent Fuel Cooling System
- Essential Raw Cooling Water System
- Sampling, Water Chemistry Monitoring Instrumentation, and Lab Design
- Containment Spray System
- Hydrogen Supplies
- Nitrogen Supplies
- Missile Protection
- RC ECCS Sump
- Instrument Air System
- Radiation Monitoring System
- Reactor Building Crane
- RB Cooling Loads
- Control Room Design
- Plant Ventilation System
- Environmental Protection
- RC Pump Monitors
- Secondary Plant Water Chemistry
- Borated Water Storage Tank
- Separation of Class 1E Systems and Equipment
- Radioactive Waste Drains and Vents

Table 7-2. Additional Systems Needed to Support CNSS

	<u>Input Responsibility</u>
Reserve shutdown system	B&W
Miscellaneous liquid waste management system	A-E/B&W
Miscellaneous gaseous waste management system	A-E/B&W
Miscellaneous solid waste management system	A-E/B&W
Radioactive gas recovery system	A-E/B&W
Fuel handling system	B&W
Fuel handling purge and evacuation system	B&W
Steam/water dump system	A-E/B&W
Nuclear inservice inspection system	A-E/B&W

8. FUTURE STUDIES

The following three tasks are proposed as additional Phase 2A study areas requiring further design or economic definition.

8.1. Task 1.0 – CNSS Siting Study

The objective of this task is to perform some fundamental engineering evaluations of the CNSS as a nuclear system, and (with assistance from an Architect/Engineer) of the containment, biological shield space, and building structures and arrangements which will permit an assessment of the siting credits and advantages which are believed to be inherent in the modular reactor/containment/auxiliary building complex. These factors would have to be considered with respect to cogeneration (transient worker population) and district heating (near center-of-population) siting applications which may require special NRC siting guidelines and evaluations.

This study will assess in scoping fashion a spectrum of accidents of the type generally found to be of interest in determination of site boundaries, and will analyze the expected improvement or degradation (if any) projected for the CNSS plant as assessed. It will also include in this scoping estimate certain non-design-basis accidents (beyond the normal limits of mechanistic probability), and will assess CNSS siting impact for these occurrences as well.

The study is expected to result in a scoping estimate of accidents impacts which will:

1. Identify on a relative basis those events which would be most likely to influence site boundary determination.
2. Identify system, structure, or equipment modifications to minimize site boundary analysis impact of events studied.
3. Identify current regulatory siting requirements considered to be unduly restrictive or arbitrary which may require more detailed consideration

with NRC (later) in the context of the CNSS plant and specific applications (e.g., cogeneration, district heating, power generation).

8.2. Task 2.0 — RC Pump Inertia Requirements

The reactor coolant pumps provide the pumping force for the reactor coolant passing through the core and throughout the reactor coolant flow path. The classical transient for the loss of all electrical power to these pumps is of vital importance to the NSS design. The modified CNSS concept requires an evaluation of the pump inertia requirement to accommodate this loss-of-power transient and subsequent pump coastdown. The output from this task will be the determination of the pump/motor inertia which will produce an acceptable minimum DNBR.

8.3. Task 3.0 — Electrically Coupled External Pump Flywheel

Configuration optimization of the CNSS could be improved if necessary pump inertia, required for an assumed loss-of-power transient, could be incorporated outside the pump itself. This task will be to study the feasibility and conceptual design of electrically coupled flywheels outside the pressure barrier and possibly outside the containment. Equipment requirements, mechanical configuration and electrical circuitry that satisfy both design and regulatory requirements will be investigated. The task will result in a concept description for accomplishing the electrically coupled flywheel which can later be reviewed informally for regulatory acceptability.

Other study areas were also identified. Below is a partial listing of some of the more important study areas which would receive high priority during preliminary design work:

1. Reboiler elimination.
2. Reactor coolant system (RCS) components maintenance and inspection procedures.
3. RCS pump design and interface with reactor vessel.
4. Secondary system design and control.
5. Moisture separator concepts and design.

9. TASK 6 — CNSS SITING STUDY

9.1. Summary

The CNSS siting study was performed to help quantify the potential for siting CNSS units on small sites near metropolitan load centers. The ability to locate close to load centers is an important economic factor for any power station but is especially important for cogeneration plants designed to provide both electricity and either process steam or hot water. The CNSS is particularly well suited for metropolitan siting because of its small size (low power), the consolidation of the nuclear steam system within one pressure vessel, and the use of a building arrangement which provides multiple barriers to the escape of radioactivity following a loss-of-coolant accident (LOCA). A limited engineering evaluation of a spectrum of design basis accidents was performed to determine, in a scoping fashion, the effect of various CNSS design features on the radiological consequences of these accidents which, in turn, affect the minimum required distances to the exclusion area boundary and to the outer boundary of the low population zone and the maximum allowable population surrounding the site. The details of the analysis of each accident are presented in Appendix A.

Since this siting study is intended as an aid for the preliminary evaluation of potential sites and for the preliminary economic trade-off studies or cost-benefit analysis of site features and balance-of-plant design features, attempts were made to eliminate dependence on site specific parameters. Since the radiological dose consequences are directly dependent on the atmospheric dispersion factor (x/Q), the study is based on the (x/Q) values presented in Regulatory Guides 1.3 and 1.4 (the values used by the NRC when no site specific data are available).

The minimum acceptable site size is determined by the required distance to the exclusion area boundary (EAB) which is set by the 10 CFR 100 limits on the 2-hour doses to the thyroid and whole-body of offsite individuals with the

highest exposures. These two doses are shown as a function of distance for the most important accidents (Figures 9-1 and 9-2). Based on the Regulatory Guide 1.4 dose limits for a construction permit (150 rem thyroid and 20 rem whole-body), a CNSS unit may be placed on a site with an EAB between 160 and 460 meters. In comparison, the EAB distances for several typical large electrical generating nuclear plants range from 800 to 1900 meters.

The minimum low-population zone (LPZ) distance is set by the 10 CFR 100 limits on the 30-day doses to the thyroid and whole-body of offsite individuals with the highest exposures. These doses are shown as a function of distance for the most important accidents (Figures 9-5 and 9-6). Based on the Regulatory Guide 1.4 limits stated above, a CNSS unit requires an LPZ of less than 700 meters. Typical large nuclear plants have an LPZ distance ranging from 3200 to 6400 meters.

Although there are no specific regulations on population density in the vicinity of nuclear plants, the recommended value in Regulatory Guide 4.7 is 500 people per square mile within 30 miles of the site. According to WASH-1308, 95% of the acceptable sites docketed prior to its publication have site population densities less than 1000 people per square mile within the first 50 miles. Indian Point has 1490 people per square mile within a 50-mile radius which is the largest site population currently licensed. For this study, the density of 1000 people per square mile within a 50-mile radius was used in the population dose calculations. The man-rem doses based on whole-body gamma irradiation and the thyroid-rem doses based on iodine inhalation are presented in such a manner they can easily be applied to a specific site under consideration.

The low doses and small EAB and LPZ distances are attributable to the inherent design features of the CNSS and to the additional protection provided by the multiple barriers that prevent radioactivity from leaking directly to the atmosphere since the plant arrangement surrounds the reactor vessel with the containment, the biological shield, and the service (auxiliary) buildings. In the CNSS arrangement (see Figure 9-10), any radioactivity which leaks out of the containment would become trapped in the annulus between the containment and the biological shield. From the annulus, the activity can be released to the atmosphere at a controlled rate after most of the radioiodine activity has been removed by charcoal filters. Since the entire biological shield is surrounded by the service building, if any fission product activity were to

escape from or circumvent the annulus, it would be trapped within the service building (which would not be the case for conventional doubly-contained plants). Activity leaking into the service building would eventually be released via the filters in the building's exhaust ventilation system. Figure 9-10 schematically shows the various pathways for radioactivity to escape from the containment.

This siting study will enable the identification of the limiting or constraining factors associated with any site under consideration. Once these factors have been identified, more detailed consideration can be given to the cost-benefit ratio of various alternatives. The evaluation of various alternatives will be aided by the detailed description of the bases used to evaluate the consequences of each accident which are provided in Appendix A. The study has identified various design features which merit more detailed economic evaluation, e.g.:

1. Charcoal filters in the fuel handling area.
2. Iodine removal sprays in the containment.
3. Design leak rate of the containment.
4. A stack for elevated release of activity.
5. A quench tank or scrubber to remove iodine from steam safety valve discharges.
6. Underground containment.
7. Enclosing the biological shield within the service building.

An economic trade-off study can be made between each of these design features and the cost of the specific site which, of course, is dependent on the site size and location.

9.2. Objectives and Scope

The objective of this task is to perform a siting study which will quantify the potential for siting CNSS units on small sites near metropolitan load centers. The CNSS is expected to have unique advantages for "close-in" siting because of its small size, the consolidation of the nuclear steam system within one pressure vessel, and the use of a building arrangement which surrounds the reactor vessel with the containment, the biological shield, and the service (auxiliary) building.

A limited engineering evaluation of a spectrum of design basis accidents will be performed to assess the merits of various CNSS design features and to serve as a basis for future economic trade-off studies or cost-benefit evaluations. Since the accident evaluations will be based on preliminary design information, the results of the siting study are intended to be of a scoping nature and will have to be confirmed by more detailed analyses as the CNSS design progresses. This preliminary evaluation, however, should provide a quantitative basis for screening tentative sites and for identifying major factors which require more detailed design or economic consideration.

In order to assure that the CNSS siting study reflected the current industry-wide practices of nuclear power plant siting, a literature search concentrating on siting methodology was performed. The fission product activities in the reactor core and coolant were calculated using currently available CNSS data and, where necessary, data from other plants were scaled to the CNSS power level of 1200 MWt. A range of moderate frequency events and design basis accidents was surveyed, and those events and accidents which were considered to be most critical with regard to the radioactive releases were then analyzed in detail. The accidents and events analyzed were:

1. Loss of all offsite power.
2. Rod ejection accident.
3. Fuel handling accident.
4. Design basis loss-of-coolant accident (per Regulatory Guide 1.4).
5. Steam generator tube rupture.
6. Steam line break.
7. Pressurizer surge line break.
8. Rupture of the waste gas decay tank.
9. Liquid waste system failure.

The radioactive releases associated with the above accidents were used together with the x/Q values from Regulatory Guides 1.3 and 1.4 to determine the radiation exposure to the maximum exposed individual as a function of distance (from 200-1000 meters). (The detailed analyses of these accidents are presented in Appendix A.) Studies were also done to determine the effects of stack height and containment leak rates on the dose versus distance curves for key accidents. The accidents which are the most critical in determining the EAB and LPZ were then studied to determine whether a design change could be made that would

lower the radiological consequences of these accidents and thereby reduce the required distances to the EAB and LPZ. The study also considered the impact of population density and population dose on site selection.

9.3. Results and Conclusions

The CNSS siting study concentrated primarily on determining the most limiting accidents for the plant and what additions or modifications to the plant would mitigate the resulting doses and facilitate siting. The dose consequences of the accidents analyzed are expressed in terms of the resulting thyroid dose and γ whole-body dose as a function of distance. Both doses are evaluated as 2-hour doses which determine the minimum distance to the EAB and as 30-day doses which determine the minimum radius of the LPZ.

9.3.1. Site Size

Figure 9-1 shows the most critical 2-hour thyroid doses as a function of distance. Clearly, the fuel handling accident is the most limiting, but if charcoal filters are added to the emergency exhaust ventilation system in the fuel handling area, the rod ejection accident (with an assumed 1 gpm primary-to-secondary leak releasing activity to the atmosphere via the steam safety valves due to the loss of condenser) becomes the most limiting accident. The rod ejection accident exceeds the Regulatory Guide 1.4 thyroid dose guidelines of 150 rem (at the construction permit stage of licensing) at distances less than 250 meters, but meets the 10 CFR 100 thyroid dose limit of 300 rem with an EAB of only 200 meters. If it became desirable to further reduce the thyroid doses by reducing the amount of iodine activity released via the steam safety valves in the event that the condenser is not available during a control rod ejection accident or a steam generator tube rupture accident, the most effective method of attaining such a reduction is by the addition of a quench tank which would condense the steam and remove most of the iodine, thereby lowering the thyroid dose.

The 2-hour whole-body doses are shown in Figure 9-2. Clearly, the doses at distances of 200 meters and beyond are, for all accidents, below the Regulatory Guide 1.4 limit of 20 rem for PSAR stage and the 10 CFR 100 limit of 25 rem. Since no dose mitigation seems to be needed and since the addition of iodine removal equipment, such as the charcoal filters for the fuel handling area or the quench tank for the steam safety valves, will only lower the whole-body

dose a small amount, the cost-benefit ratio of this equipment could not be supported by any whole-body dose reduction.

Based on Figures 9-1 and 9-2, it can be seen that under no conditions would the site size have to be larger than 460 meters. However, by simply adding charcoal filters on the emergency exhaust ventilation from the fuel handling area, the required site size can be reduced to approximately 200 to 250 meters. Further reductions in site size can be achieved by the addition of a quench tank on the discharge of the steam safety valves.

The addition of a stack to the plant site would also significantly lower both the thyroid and whole-body doses of those accidents whose activity release can be routed through the stack. Figure 9-3 shows the 2-hour thyroid doses (using the same activity releases used for the accidents in Figure 9-1) that result when the activity is released via a 75-meter stack. The rod ejection accident, the steam generator tube rupture, and the steam line break, which appear in Figure 9-1 but are not shown in Figure 9-3, have steam releases which, with the present CNSS design, cannot be readily vented up a stack. However, if a quench tank is added to the steam safety valves, the steam releases from the rod ejection accident and steam generator tube rupture would then be released to the quench tank and the off-gas from the quench tank could be vented to the stack, yielding a double benefit.

The dose reduction benefit of the stack is clearly shown by comparing the most limiting accident with Figure 9-1, e.g., the fuel handling accident with no iodine removal filtration on the ventilation system. The doses at 200 meters (Figure 9-3) are at least a factor of 100 less than the comparable doses at 200 meters in Figure 9-1, but even at greater distances, the 2-hour thyroid doses which result from the activities released via the stack remain at least a factor of 15 below the doses that result when the same activities are released at ground level.

The 2-hour whole-body doses which result when the activities resulting from the accidents are released via a 75-meter stack are shown in Figure 9-4. The whole-body dose reduction due to the 75-meter stack is a factor of 100 at 200 meters, and at least a factor of 15 at the other distances for those accidents with activities released via the stack.

Figures 9-3 and 9-4 show that a 75-meter stack would significantly lower the doses resulting from the accidents that could have activity releases discharged via a stack. However, it should be noted that although the doses are reduced by the addition of a stack, the amount of activity released to the environment is the same as shown in Figures 9-1 and 9-2. The stack changes the release point elevation and permits greater atmospheric dispersion. A stack could serve to reduce the amount of released activity if the stack is equipped with an iodine removal spray scrubber. A stack with an iodine scrubber could be substituted for a quench tank.

Table 9-6 is a summary showing how the special additional design features affect the EAB distance for several of the most limiting accidents in the plant. From the table it can be seen that with the addition of charcoal filters to the fuel handling area, the minimum exclusion area boundary would be decreased from 460 to 250 meters since the rod ejection accident would become the controlling accident. Furthermore, it appears that the addition of a quench tank or quench tank plus a stack to further reduce the minimum EAB may not be justifiable considering cost versus benefit. The cost of the stack and/or quench tank may well exceed the cost of the additional site area required.

9.3.2. Low Population Zone Size

Figure 9-5 shows the most critical 30-day thyroid doses as a function of distance. The most limiting accident for this dose is the design basis LOCA covered by Regulatory Guide 1.4. If the LOCA is analyzed assuming the fission product release and x/Q required by Regulatory Guide 1.4 and assuming a containment leak rate of 0.3% per day, no containment sprays, no annulus recirculation, and a high annulus exhaust rate, the required LPZ would be in the range of 450 to 700 meters. The doses which determine the LPZ size can be lowered by lowering the amount of iodine released, by reducing the containment leak rate, adding iodine removal sprays in the containment, adding annulus air cleaning capability by recirculating the air through charcoal filters, and by reducing the annulus ventilation exhaust rate. There are a family of LOCA dose curves which lie between the two limiting LOCA cases shown in Figure 9-5. Table 9-1 shows the dose results for the dose points at 200 meters for the family of curves with various combinations of the four parameters mentioned above. If the LOCA case 6 or an equivalent case is chosen, the rod ejection accident (REA) and steam generator tube rupture (SGTR) again become the

controlling accidents provided that charcoal filters are added to the fuel handling area. However, the addition of the quench tank to the steam safety valves could be used to lower the thyroid doses for the REA and SGTR accidents. If the quench tank is added to the steam safety relief valves, the doses resulting from the REA and SGTR accidents will be less than the doses from a fuel handling accident mitigated by charcoal filters.

The 30-day whole-body doses are shown in Figure 9-6. The LOCA activity release via the containment is the dose controlling accident. It can be seen from Table 9-1 that in order to meet the 25 rem 30-day whole-body dose limit of 10 CFR 100 at 200 meters from the plant, the containment leak rate must be 0.2% volume per day or less. With a 0.2% volume per day leak rate, the annulus exhaust rate must be less than 3.0 air changes per hour to meet the 30-day whole-body dose limits; but with a 0.1% volume per day leak rate, the annulus exhaust rate may be as great as 30 air changes per hour. When the 30-day whole-body dose is evaluated along with the 30-day thyroid dose, the LOCA cases 1 through 7, 9, 13, and 19 become the most viable choices of the 36 cases.

Based on Figures 9-5 and 9-6, it can be seen that under no condition would the LPZ size have to be larger than 700 meters. However, the addition of charcoal filters to the fuel handling accident will reduce the LPZ size to less than 300 meters. The one plant modification which effectively reduces the accident doses is the addition of a stack. Figure 9-7 shows the effect of using a 75 meter stack to lower the 30-day thyroid doses of accidents that could have their activity releases discharged via a stack. Even though the amount of activity released to the atmosphere was identical for the accidents shown in Figures 9-5 and 9-7, the doses shown in Figure 9-7 are significantly lower due to the greater amount of atmospheric dispersion provided by the stack. The rod ejection accident and the steam generator tube rupture accident do not appear on Figure 9-7 because the activity release is not amenable to being discharged via the stack since it is via the steam safety valves. However, with appropriate design modifications the steam safety valves could be designed to discharge either directly into a stack or indirectly via a quench tank. If the safety valves discharge directly into the stack, the stack could be designed with an iodine removal spray to condense some of the steam which would remove most of the iodine activity.

The dose reduction benefit of the stack is clearly shown by comparing the most limiting accident, the LOCA of Figures 9-5 and 9-7. The doses at 200 meters in Figure 9-7 are approximately a factor of 100 less than the comparable doses at 200 meters in Figure 9-5, but even at greater distances, the 30-day thyroid doses which result from the activities released via the stack are at least a factor of 15 less than the doses when the same activities are released at ground level.

The 30-day whole-body doses which result when the activities released from the accidents are released via a 75 meter stack are shown in Figure 9-8. The dose reduction effect of the 75 meter stack is a factor of 100 at 200 meters and at least a factor of 15 at the other distances for the accidents with activities released via the stack.

In Figures 9-7 and 9-8, as in Figures 9-3 and 9-4, it is shown that a 75-meter stack would significantly lower the doses resulting from the accidents that have activities released via the stack. It should be noted, however, that although the doses are reduced by the stack, the amount of activity released to the environment remains the same. If the stack is equipped with an iodine removal spray scrubber, the stack could then serve to reduce the amount of iodine activity.

The LOCA doses presented in Table 9-1 and Figures 9-1 through 9-8 are the doses which result from radioactivity which leaks from the containment into the annulus and is then released to the environment via the annulus exhaust system. No consideration was given to the doses which would result from activity which leaks directly from the containment to the environment (by-pass leakage) via containment penetrations. Table 9-2 shows the LOCA doses at 200 meters which are associated with a 0.01% containment volume per day leak rate bypassing the annulus and leaking directly to the environment. These LOCA doses are in addition to the LOCA doses presented in Table 9-1. The necessity of containment sprays for iodine removal is clearly shown in Table 9-2 by the very high thyroid doses that result when no iodine removal system is available. Since LOCA cases 7, 9, and 19 do not include containment sprays, these cases must be eliminated from the list of viable alternatives. Thus, the list of viable LOCA cases decreases to only seven of the 36 original cases; they are cases 1-6 and case 13. Of course, other viable alternatives could be created by defining new cases with intermediate containment leak rates, that is leak rates between 0.1 and 0.2% per day.

This study does not specifically consider the merits of the auxiliary building surrounding the annulus or of an underground containment, however, the study does inherently consider the merits of these two alternatives by considering two bounding cases. The dose values presented in Table 9-1 are based on the assumption that the auxiliary building or the underground containment is a perfect barrier and permits no activity which may bypass the annulus to escape into the atmosphere; whereas the dose values in Table 9-2 show the impact of 0.01% of the leakage from the primary containment penetrating the additional barriers provided by the auxiliary building or underground containment. Obviously, many other arbitrary cases could be defined and analyzed; however, the high doses shown in Table 9-2 provide sufficient insight into the problem to indicate the need for a detailed study of all the pathways for activity to escape to the atmosphere without being trapped within the annulus. After the pathways have been identified, the relative probability of leakage via each pathway must be assessed before a meaningful cost-benefit study can be performed on the siting benefits of an underground containment or of an auxiliary building designed to enclose the containment and annulus.

This study has concentrated on the design basis accidents presented in Chapter 15 of the Safety Analysis Report (per Regulatory Guide 1.70) which normally determines site suitability. However, because of the inherent design features of the CNSS and the additional protection offered by the multiple barriers, the accident releases are considerably lower than found in typically large electrical generating nuclear plants. Therefore, the siting may well be limited by the 10 CFR 50 Appendix I limits on the offsite doses associated with the radioactive release during normal reactor operation and anticipated operational occurrence. Before optimizing the balance of plant equipment for mitigating accident doses, a site evaluation should be performed for normal plant operations (based on Regulatory Guides 1.109 and 1.112) to see what balance-of-plant equipment is needed to justify normal operation of a CNSS unit on the site under consideration.

9.3.3. Population Density

The population doses (in man-rem or in thyroid-rem) associated with hypothesized accidents are functions of the amount, type, and duration of radioactivity release, the site meteorology, and the population distribution surrounding the proposed site. The radioactive releases associated with the CNSS accidents

analyzed in Appendix A were used to calculate the dose to an individual located at the centroid of each of the 160 regions formed by 16 22.5° sectors centered on the 16 points of the compass and 10 concentric circles 5 miles apart. The population doses received within the region were estimated by multiplying the dose received by this individual by the population within that region. The total population dose for the site was determined by summing the population doses for all regions. Figure 9-9 helps to visualize the location of the 160 regions; and Table 9-4 gives the distance to the centroid of each region. Since this is a general siting study, site specific meteorology could not be used effectively; therefore, the accident meteorology given in Regulatory Guide 1.4 was used to obtain typical (x/Q) values. The dose calculations were corrected for the amount of radioactive decay that occurred enroute to the centroidal dose point based on a wind speed of 1 m/s. The site population distribution for this hypothetical site was assumed to be uniform with a population density of 1000 people per square mile. The resultant population doses for this hypothetical site are given in Table 9-3. (Note that the values in Table 9-3 are based on the assumption that the wind is also uniformly distributed, i.e., it blows in each of the 16 sector's direction 1/16 of the time. Thus, the values in Table 9-3 may also be interpreted as sector population doses for the case where the wind always blows in that one sector. This dual interpretation is an artifact of the uniform distribution of both the wind and the population.) Since the population doses in Table 9-3 are for a hypothetical site, they are of little value except to show the relative impact of the various accidents; however, using the following technique together with the site population distribution and meteorological data for a specific site, the population dose can be easily calculated for any specific site.

The dose to the centroidal individual in any given time period of constant (x/Q) can be expressed as:

$$D_{\Delta t} = \left(\frac{x}{Q} \right)_{\Delta t} \sum_i \left[K_i (A_i)_{\Delta t} e^{-\lambda_i \tau_r} \right] \quad (9-1)$$

where

- $D_{\Delta t}$ = dose to the centroidal individual in time period Δt ,
- $(x/Q)_{\Delta t}$ = atmospheric dispersion factor for the time period Δt and for the sector and distance to the centroid of the region of interest,
- $(A_i)_{\Delta t}$ = activity of nuclide (i) released to environment during time period Δt ,

K_i = dose conversion factor for nuclide (i) combined with other constants required for correct units,

τ_r = travel time of the released activity to the centroidal dose point,

λ_i = radioactive decay constant for nuclide (i).

To simplify future notations, the symbol $(Z_{\Delta t})_r$ is defined as the "individual dose factor," i.e.:

$$(Z_{\Delta t})_r = \sum_i \left[K_i (A_i)_{\Delta t} e^{-\lambda_i \tau_r} \right] = \frac{D_{\Delta t}}{(\bar{x}/Q)_{\Delta t}} \quad (9-2)$$

If the activity released occurs for an extended period of time, the $(\bar{x}/Q)$ will change and the total dose $(D_t)_r$ must be accumulated over several time periods.

$$(D_t)_r = \sum_{\Delta t} \left[\left(\frac{\bar{x}}{Q} \right)_{\Delta t} (Z_{\Delta t})_r \right] \quad (9-3)$$

where

$(D_t)_r$ = total dose to the centroidal individual over some total time period $t = \Delta t_1 + \Delta t_2 + \dots$,

$(Z_{\Delta t})_r$ = the individual dose factor for the centroidal individual over the time period Δt .

The population dose in any region is estimated by

$$(PD)_r = P_r (D_t)_r \quad (9-4)$$

where

$(PD)_r$ = the population dose in a specific region,

P_r = the population in the region under consideration.

The population dose in a sector is obtained by summing the value of $(PD)_r$ for all the regions in the sector and multiplying the total by the frequency of the wind blowing in the direction of that sector.

$$(PD)_s = F_s \sum_{r=1}^{10} (PD)_r \quad (9-5)$$

where

$(PD)_s$ = the population dose for one 22.5° sector,

F_s = the frequency with which the wind blows in the direction of the sector under consideration.

The total population dose for the site is obtained by summing the $(PD)_s$ for all 16 sectors surrounding the site.

$$(PD)_T = \sum_{s=1}^{16} (PD)_s \quad (9-6)$$

where $(PD)_T$ = the total population dose for the site.

By substituting equations 9-3 through 9-5 into equations 9-5 and 9-6, the following equations result for the sector population dose $(PD)_s$ and the total population dose $(PD)_T$.

$$(PD)_s = F_s \sum_{r=1}^{10} \left\{ P_r \sum_{\Delta t}^4 \left[\left(\frac{x}{Q} \right)_{\Delta t} (Z_{\Delta t})_r \right] \right\} \quad (9-7)$$

$$(PD)_T = \sum_{s=1}^{16} \left[F_s \sum_{r=1}^{10} \left\{ P_r \sum_{\Delta t}^4 \left[\left(\frac{x}{Q} \right)_{\Delta t} (Z_{\Delta t})_r \right] \right\} \right] \quad (9-8)$$

Note that the values of F_s , P_r , and $(x/Q)_{\Delta t}$ are all site specific parameters and must be obtained for the site under consideration. The value of $(Z_{\Delta t})_r$ is, however, independent of site characteristics and is only a function of the CNSS plant design and accident analyses [except for the minor point that to correct for decay enroute a 1 m/s wind speed was incorporated into the value of $(Z_{\Delta t})_r$]. The values of $(Z_{\Delta t})_r$ for all the key accidents analyzed in this siting study are presented in Tables 9-4 and 9-5. With the aid of these tables and with the necessary meteorological and population data for a specific site, the site population doses can be easily calculated using equations (9-7) or (9-8).

Site population dose evaluations should not only consider the total population dose for the site but should also identify which sector is contributing most of the population dose and then re-evaluate the population dose for that sector assuming a unidirectional wind, i.e., $F_s = 1.0$.

Presently there are no NRC regulations which provide specific siting criteria on population dose from which to judge the acceptability of a given site. To date, population densities seem to have been the criteria. Sites with less than 500 people per square mile are usually acceptable, whereas sites in the 500-1500 range are questionable, and sites with more than 1500 people per square mile are usually not acceptable. Since the population distribution

at a specific site is usually very nonuniform, the calculation of the number of people per square mile is not truly reflective of the population dose that would result from an accident. In an attempt to compensate for this, the term "site population factor" (SPF) was developed. The SPF is the ratio of the population dose for the site under consideration to the population dose for a reference site with a uniform population distribution of 1000 people per square mile. In other words, a site population factor of 1.2 can be interpreted as a site with an equivalent population density of 1200 people per square mile.

Since the values of population dose provided in Table 9-3 are for the reference site, the SPF for a site under consideration can be easily obtained by dividing the population dose determined from equations 9-7 and 9-8 by the respective population dose from Table 9-3.

Table 9-1. Dose Consequences of a Design Basis LOCA at 200 Meters

Case No.	Containment design parameters		Annulus design parameters		2-hour doses, rem		30-day doses, rem	
	Leak rate, % volume per day	Sprays	Exhaust rate, h ⁻¹	Recirculation, h ⁻¹	Thyroid	Whole body	Thyroid	Whole body
1	0.1	(a)	0.3	6.0	3.76(-1) ^(b)	1.08	2.41	1.00(+1)
2		(a)	0.3	0	2.36	1.08	4.09(+1)	1.00(+1)
3		(a)	3.0	6.0	2.65	3.96	1.67(+1)	1.46(+1)
4		(a)	3.0	0	7.06	3.97	4.76(+1)	1.46(+1)
5		(a)	30.0	6.0	6.71	5.09	4.07(+1)	1.57(+1)
6		(a)	30.0	0	7.98	5.09	4.84(+1)	1.58(+1)
7			0.3	6.0	1.95	1.08	1.91(+1)	1.00(+1)
8			0.3	0	1.07(+1)	1.10	3.33(+2)	1.03(+1)
9			3.0	6.0	1.39(+1)	3.98	1.33(+2)	1.47(+1)
10			3.0	0	3.62(+1)	4.04	3.87(+2)	1.50(+1)
11			30.0	6.0	3.60(+1)	5.16	3.30(+2)	1.60(+1)
12	0.2		30.0	0	4.28(+1)	5.18	3.94(+2)	1.61(+1)
13		(a)	0.3	6.0	7.52(-1)	2.16	4.82	2.00(+1)
14		(a)	0.3	0	4.72	2.17	8.16(+1)	2.01(+1)
15		(a)	3.0	6.0	5.31	7.91	3.31(+1)	2.92(+1)
16		(a)	3.0	0	1.41(+1)	7.94	9.51(+1)	2.92(+1)
17		(a)	30.0	6.0	1.34(+1)	1.02(+1)	8.13(+1)	3.15(+1)
18		(a)	30.0	0	1.60(+1)	1.02(+1)	9.66(+1)	3.15(+1)
19			0.3	6.0	3.90	2.17	3.81(+1)	2.00(+1)
20			0.3	0	2.14(+1)	2.21	6.65(+2)	2.06(+1)
21			3.0	6.0	2.78(+1)	7.97	2.65(+2)	2.94(+1)
22			3.0	0	7.23(+1)	8.07	7.73(+2)	2.99(+1)
23	0.3		30.0	6.0	7.19(+1)	1.03(+1)	6.58(+2)	3.21(+1)
24			30.0	0	8.57(+1)	1.04(+1)	7.86(+2)	3.22(+1)
25		(a)	0.3	6.0	1.13	3.24	7.22	3.00(+1)
26		(a)	0.3	0	7.07	3.25	1.22(+2)	3.01(+1)
27		(a)	3.0	6.0	7.96	1.19(+1)	4.98(+1)	4.37(+1)
28		(a)	3.0	0	2.12(+1)	1.19(+1)	1.42(+2)	4.38(+1)
29		(a)	30.0	6.0	2.01(+1)	1.53(+1)	1.22(+2)	4.72(+1)
30		(a)	30.0	0	2.39(+1)	1.53(+1)	1.45(+2)	4.72(+1)
31			0.3	6.0	5.85	3.25	5.71(+1)	3.00(+1)
32			0.3	0	3.21(+1)	3.31	9.96(+2)	3.08(+1)
33			3.0	6.0	4.18(+1)	1.20(+1)	3.98(+2)	4.41(+1)
34			3.0	0	1.09(+2)	1.21(+1)	1.16(+3)	4.48(+1)
35			30.0	6.0	1.08(+2)	1.55(+1)	9.86(+2)	4.81(+1)
36			30.0	0	1.29(+2)	1.55(+1)	1.18(+3)	4.83(+1)

(a) Indicates chemical addition for iodine removal.

(b) 3.76(-1) = 3.76×10^{-1}

Table 9-2. Additional Dose Associated With 0.01%/Day of Airborne Activity in Containment^(a)

<u>Distance, meters</u>	<u>Ground Level Release</u>			
	<u>2-Hour Doses, rem</u>		<u>30-Day Doses, rem</u>	
	<u>Thyroid</u>	<u>Whole body</u>	<u>Thyroid</u>	<u>Whole body</u>
<u>With iodine removal by containment sprays</u>				
200	24.3	0.594	96.3	1.72
<u>Without iodine removal by containment sprays</u>				
200	202.0	1.03	1,830	3.54

(a) Following a design basis LOCA bypassing the annulus and leaking directly to the atmosphere.

Table 9-3. Population Dose

(Basis: 1000 People/mi² (SPF = 1.0))

<u>Distance,</u> <u>mi</u>	<u>Fuel handling</u> <u>accident(a)</u>	<u>LOCA</u> <u>case-4</u>	<u>Rod ejection,</u> <u>accident(b)</u>	<u>Steam</u> <u>generator</u> <u>tube rupture</u>
<u>Thyroid population dose, cumulative thyroid-rem</u>				
5	7.61(+3) ^(c)	2.06(+3)	1.93(+4)	1.09(+4)
10	1.66(+4)	4.33(+3)	4.15(+4)	2.37(+4)
15	2.33(+4)	6.12(+3)	5.74(+4)	3.31(+4)
20	2.94(+4)	7.59(+3)	7.17(+4)	4.16(+4)
25	3.56(+4)	9.05(+3)	8.59(+4)	5.02(+4)
30	4.15(+4)	1.04(+4)	9.92(+4)	5.83(+4)
35	4.74(+4)	1.17(+4)	1.12(+5)	6.62(+4)
40	5.33(+4)	1.30(+4)	1.25(+5)	7.41(+4)
45	5.89(+4)	1.42(+4)	1.37(+5)	8.16(+4)
50	6.41(+4)	1.53(+4)	1.48(+5)	8.85(+4)
<u>Whole body population dose, cumulative man-rem</u>				
5	3.72(+2)	6.09(+2)	5.60(+1)	5.11(+1)
10	8.12(+2)	1.13(+3)	9.89(+1)	1.07(+2)
15	1.13(+3)	1.45(+3)	1.22(+2)	1.53(+2)
20	1.43(+3)	1.66(+3)	1.38(+2)	1.87(+2)
25	1.73(+3)	1.86(+3)	1.52(+2)	2.21(+2)
30	2.01(+3)	2.02(+3)	1.63(+2)	2.52(+2)
35	2.29(+3)	2.16(+3)	1.73(+2)	2.83(+2)
40	2.57(+3)	2.30(+3)	1.83(+2)	3.13(+2)
45	2.84(+3)	2.42(+3)	1.91(+2)	3.41(+2)
50	3.08(+3)	2.54(+3)	1.97(+2)	3.67(+2)

(a) Includes charcoal filters on the exhaust system.

(b) Considers only release via secondary system.

(c) 7.61(+3) = 7.61×10^3 .

Table 9-4. Individual Dose Factors for Calculating
LOCA (Case-4) Population Doses

Annular ring, miles	Centroid distance, miles		(a)				
			$(Z_{\Delta t})_r$ 0-2 h	$(Z_{\Delta t})_r$ 2-8 h	$(Z_{\Delta t})_r$ 8-24 h	$(Z_{\Delta t})_r$ 1-4 days	$(Z_{\Delta t})_r$ 4-30 days
5	3.936	Thyroid	2.28+3 ^(b)	4.22+3	4.50+3	9.50+3	2.63+4
		Whole Body	8.77+2	1.25+3	1.13+3	1.38+4	2.64+3
10	7.906	Thyroid	2.19+3	3.87+3	4.35+3	9.36+3	2.61+4
		Whole Body	5.68+2	9.00+2	1.01+3	1.36+3	2.61+3
15	12.748	Thyroid	2.10+3	3.73+3	4.25+3	9.27+3	2.58+4
		Whole Body	3.77+2	6.70+2	9.25+2	1.34+3	2.58+3
20	17.68	Thyroid	2.02+3	3.62+3	4.15+3	9.15+3	2.57+4
		Whole Body	2.67+2	5.30+2	8.64+2	1.32+3	2.54+3
25	22.64	Thyroid	1.96+3	3.52+3	4.06+3	9.04+3	2.54+4
		Whole Body	2.10+2	4.65+2	8.71+2	1.40+3	2.92+3
30	27.61	Thyroid	1.90+3	3.42+3	3.98+3	8.95+3	2.53+4
		Whole Body	1.64+2	3.92+2	7.92+2	1.28+3	2.49+3
35	32.60	Thyroid	1.84+3	3.31+3	3.90+3	8.80+3	2.50+4
		Whole Body	1.46+2	3.59+2	7.66+2	1.26+3	2.46+3
40	37.58	Thyroid	1.79+3	3.24+3	3.83+3	8.69+3	2.49+4
		Whole Body	1.24+2	3.31+2	7.45+2	1.25+3	2.42+3
45	42.57	Thyroid	1.75+3	3.16+3	3.74+3	8.58+3	2.46+4
		Whole Body	1.14+2	3.15+2	7.29+2	1.23+3	2.40+3
50	47.57	Thyroid	1.70+3	3.09+3	3.67+3	8.25+3	2.44+4
		Whole Body	1.07+2	3.02+2	7.13+2	1.21+3	2.37+3

(a) $Z_{\Delta t} = (\text{dose})_{\Delta t} / (x/Q)_{\Delta t}$.

(b) $2.28+3 = 2.28 \times 10^3$.

Table 9-5. Individual Dose Factors for Calculating Population Doses

Annular ring, miles	Centroid distance, miles		$(Z_{\Delta t})_r^{(a)}$		
			Fuel handling accident	Steam generator tube rupture	Rod ejection accident
5	3.936	Thyroid	3.10+4 ^(b)	4.44+4	7.88+4
		Whole Body	1.52+3	2.08+2	2.28+2
10	7.906	Thyroid	3.05+4	4.32+4	7.50+4
		Whole Body	1.49+3	1.89+2	1.45+2
15	12.748	Thyroid	3.01+4	4.22+4	7.20+4
		Whole Body	1.46+3	1.75+2	1.04+2
20	17.68	Thyroid	2.97+4	4.12+4	6.93+4
		Whole Body	1.43+3	1.65+2	8.03+1
25	22.64	Thyroid	2.94+4	4.04+4	6.69+4
		Whole Body	1.41+3	1.59+2	6.58+1
30	27.61	Thyroid	2.90+4	3.95+4	6.50+4
		Whole Body	1.39+3	1.53+2	5.55+1
35	32.60	Thyroid	2.87+4	3.88+4	6.31+4
		Whole Body	1.37+3	1.48+2	4.84+1
40	37.58	Thyroid	2.84+4	3.82+4	6.14+4
		Whole Body	1.35+3	1.45+2	4.57+1
45	42.57	Thyroid	2.81+4	3.74+4	5.96+4
		Whole Body	1.33+3	1.42+2	4.00+1
50	47.57	Thyroid	2.78+4	3.68+4	5.85+4
		Whole Body	1.31+3	1.39+2	3.60+1

(a) $Z_{\Delta t} = (\text{dose})_{\Delta t} / (x/Q)_{\Delta t}$.

(b) $3.10+4 = 3.10 \times 10^4$.

Table 9-6. CNSS Exclusion Area Boundary Distance Meters
Based on Regulatory Guide 1.4

<u>Special design features</u>	<u>Type of accident</u>		
	<u>Fuel handling</u>	<u>Rod ejection</u>	<u>SG tube rupture</u>
None	460	250	180
Charcoal filters in fuel handling area	160	250	180
Quench tank	460	<<250 ^(a)	<<180 ^(a)
Stack	<160 ^(a)	250	180
Stack and quench tank	<160 ^(a)	<<250 ^(a)	<<180 ^(a)

(a) Estimated.

Figure 9-1. 2-Hour Thyroid Dose

Figure 9-2. 2-Hour Whole Body Dose

(GROUND LEVEL RELEASE)

Figure 9-3. 2-Hour Thyroid Dose
(RELEASE VIA 75 METER STACK)

Figure 9-4. 2-Hour Whole Body Dose

Figure 9-5. 30-Day Thyroid Dose

Figure 9-6. 30-Day Whole Body Dose

Figure 9-7. 30-Day Thyroid Dose
RELEASE VIA 75 METER STACK

Figure 9-8. 30-Day Whole Body Dose

REM

(RELEASE VIA 75 METER STACK)

Figure 9-9. Location of Dose Points and Population Dose Regions Surrounding Reactor Site

Figure 9-10. CNSS Multi-Barrier Design Concept

10. TASK 7 — REACTOR COOLANT SYSTEM PUMP INERTIA REQUIREMENTS

10.1. Summary

The flow versus time coastdown curves for various pump inertias are shown in Figure 10-4. The results of the analyses to determine the departure from nucleate boiling ratio (DNBR) show that a pump inertia of 5000 lb-ft² is required to maintain an acceptable DNBR.

10.2. Objectives and Scope

During the Phase 2A Interim Studies, the pump inertia was estimated based on pump inertia requirements for the CNSG and the 2650-MWt loop-type NSS. The magnitude of the estimated value and the resultant flywheel size indicated that formal calculations were required to reduce the conservatism in the estimate and to provide a more realistic flywheel size. Therefore, the purpose of this task was to determine the minimum pump inertia that would produce an acceptable DNBR during pump coastdown.

Part 1 of this task was to develop reactor coolant system (RCS) flow versus time coastdown curves for a range of pump inertias. This required modeling the RCS to determine the pressure distribution and the total dynamic head (TDH) of the reactor coolant pumps. Discussions with a pump vendor were necessary to establish a head capacity curve and a hydraulic torque curve. Finally, the development of a calculational method for performing the coastdown analysis was required since the standard techniques used for the loop-type NSS were not applicable for the CNSS geometry.

Part 2 of this task was to determine which of the coastdown curves developed in Part 1 yielded an acceptable minimum DNBR, thereby a specific value for pump inertia. This required the determination of the steady state minimum DNBR prior to the transient where power is lost and the pumps coastdown. Existing computer codes were used for the steady state and transient analysis with input to the transient analysis consisting of the RCS flow versus time,

and reactor power versus time. The minimum DNBR during the transient was calculated using the critical heat flux correlation used for other B&W reactors.

10.3. Results and Conclusions

Part 1

The design parameters used for the CNSS RCS are shown in Table 10-1. The core pressure drop was obtained from the preliminary thermal-hydraulic analysis performed during the Phase 2A interim studies. Layout drawings previously developed were used to define the primary system geometry. Figure 10-1 shows the geometry modeled and the model description used in the SAVER computer code to obtain the RCS pressure distribution. The resultant system TDH was calculated as 225 feet. The pressure drop results are shown in Table 10-2 and the input listing is summarized in Appendix C.

The pump characteristic curve was obtained from communication with Hayward Tyler and Company, LTD, on a non-contractual basis. The recommended curve is shown in Figure 10-2. This characteristic curve was generated for the Maritime-CNSG program; however, the pump vendor indicated RCS pumps of this size and application would have generally the same characteristics. Based on this curve and the pump head calculated from the SAVER results, the curve shown in Figure 10-3 was obtained. The torque curve was also determined from the estimated pump curve. A summary of the RCS pump parameters is shown in Table 10-3.

An analytical method was developed to perform the pump coastdown analysis using the MIMIC Digital Simulation Language. The resultant code listing is attached as Appendix D and is designated the PINS (Pump Inertia Study) program. This program calculates the total flow coastdown versus time curves for various pump inertia values. This program was benchmarked on the Standard 205-FA plant, four-pump coastdown curve obtained using the analog-hybrid PUMP program.

The CNSS data were input to the PINS program, and inertia values ranging from 500 lb-ft² to 25,000 lb-ft² were analyzed. The resultant flow coastdown curves (in part) are plotted as percent total flow versus time in Figure 10-4. This figure indicates the impact that pump inertia has on the flow coastdown.

Part 2

The eight-pump coastdown is the loss of flow transient that is most directly affected by the reactor coolant pump inertia. The rate at which the reactor coolant pumps coast down after loss of power to the pump motors determines the amount of flow available for core cooling until the control rods start into the core. The flow available is one of the parameters which affects the minimum DNBR during the transient. The minimum DNBR must be greater than the design limit value during steady state or anticipated transient conditions. Thus, the eight-pump coastdown is used to determine the minimum pump inertia that will produce acceptable minimum DNBR results.

The initial conditions used for the eight-pump coastdown DNBR analysis are shown in Table 10-4. Temperature and flow conditions are for rated conditions. The temperature and flow used in the analysis were adjusted for 102% power based on a constant vessel average temperature of 594F. A +2F temperature error and a -45 psi pressure error were included in the final inlet temperature and system pressure. These inlet conditions are consistent with standard B&W practice and the NRC requirements for licensing as given in the NRC Standard Format and Content Guide, Rev. 2.

The initial steady state minimum DNBR is determined through the use of computer codes which account for interbundle and inter-subchannel divergent crossflow. The thermal-hydraulic conditions and minimum DNBR are calculated for the most limiting (hot) subchannel. These hot channel conditions are then used for input to the transient analysis.

The transient analysis includes the steady state input from Table 10-4 and the hot channel flow and minimum DNBR from the steady state analysis. The transient input consists of reactor coolant flow versus time and reactor power versus time. Power to the reactor coolant pumps is assumed lost at time zero and the pumps begin to coast down with the rate of flow coastdown dependent on pump inertia. Several different pump inertias were used to produce a curve of minimum DNBR versus pump inertia. Calculations were based on a reactor trip with a 0.7 s delay time before the control rods start into the core. This delay time is typical for other B&W reactors.

The BXC critical heat flux correlation used for other B&W reactors was used to calculate the minimum DNBR during the transient. A minimum DNBR of 1.37

was chosen to determine the preliminary pump inertia. This value provides a margin above the calculated minimum DNBR.

A reactor coolant pump inertia of 5000 lb-ft² is required to meet the minimum DNBR value of 1.37.

10.4. Future Work

Consultation with a qualified pump vendor(s) to establish the pump/motor design which will satisfy the inertia requirements. If the pump/motor dimensions become too large, further work related to the electrically coupled external pump flywheel reported under Task 8 could be implemented.

The above work activity would lead to the development of a pump specification for use in obtaining a proposal from a qualified pump vendor(s). This activity would be included in the Phase 2B preliminary design study.

Table 10-1. CNSS Design Parameters

Hydraulic and Thermal Design

Core heat output, MWt	1200
Primary pressure, psia	2250
Total primary flow, 10 ⁶ lb/h	67.5
SG outlet temperature, F	572.3
SG average temperature, F	593.3
SG inlet temperature, F	615.5
Average thermal output, kW/ft	4.97

Reactor Vessel Design

Primary volume, ft ³	6982
Design pressure, psig	2500
Design temperature, F	650
Shell ID, in.	195
Height inside heads, ft-in.	48-10

Steam Generator

Net steam flow, 10 ⁶ lb/h	5.29
Steam pressure, psia	935
Steam temperature, F	Sat.
Feedwater temperature, F	440
Tube length, ft	20
Tube OD, in.	0.5
Number of tubes (total)	20,460
Number of modules	10

Core Design

Core height (active), ft	10
Number of fuel assemblies	89
Number of fuel rods	23,496
Fuel rod OD/pitch, in.	0.379/0.501
Number of control rod assemblies	37
Number of control pins/assembly	24

Table 10-2. Pressure Drop Summary

Path No.	From node	To node	Path type	Flow, lbm/h	Pressure drops, psi				Momentum ^(a) K-factor
					Total	Unrecoverable	Momentum	Elevation	
1	1	2	3	0.675000E+08	-1.0007	0.740	-1.161	-0.580	0.1378E-06
2	2	3	3	0.675000E+08	19.4458	26.263	-0.017	-6.800	0.2069E-08
3	3	4	3	0.675000E+08	-0.4742	1.888	0.000	-2.362	0.
4	4	5	3	0.675000E+08	-1.4742	0.447	0.284	-2.204	-0.3657E-07
5	5	6	3	0.566589E+08	2.2127	2.206	0.007	0.000	-0.1295E-08
6	6	7	2	0.566589E+08	2.3761	2.376	0.000	0.000	0.
7	5	7	3	0.108411E+08	4.5888	4.585	0.004	0.000	-0.2078E-07
8	7	8	3	0.675000E+08	2.2620	1.381	0.251	0.630	-0.3234E-07
9	8	9	3	0.675000E+08	22.9889	19.197	-0.138	3.929	0.1703E-07
10	9	10	3	0.675000E+08	4.8482	1.811	-0.152	3.189	0.1804E-07
11	10	11	3	0.675000E+08	6.6598	1.804	-0.072	4.928	0.8595E-08
12	11	12	3	0.675000E+08	9.9098	6.957	2.953	0.000	-0.3506E-06
13	12	13	3	0.675000E+08	-3.1416	0.012	-1.994	-1.160	0.2368E-06
14	13	1	9	0.675000E+08	64.6124	0.000	0.000	0.000	0.

(a) The units of the momentum K-factor are (lbf-s²)/(in.²-ft³-lbm).

Table 10-3. RCS Pump Parameters

Pump flow, gpm	25,200
Pump TDH, ft	225
Pump speed, rpm	1750
Reactor outlet specific volume, ft ³ /lbm (hot leg pump)	0.02396
Specific speed	4782
Pump efficiency, %	84
Brake horsepower (BHP)	1142
Rated torque, ft-lbs	3426

Table 10-4. Input Conditions for DNBR Analysis

Power, MWt	102% of 1200
Inlet temp., °F (@100% pwr)	572.3
System flow, 10 ⁶ lbm/h	67.5
Core flow, 10 ⁶ lbm/h (assumes 8% bypass)	62.1
System pressure, psia	2250

Figure 10-1. SAVER Nodal Description

Figure 10-2. Pump Characteristics Curve, Estimated

Figure 10-3. Estimated Pump Characteristics Curve (Based on
Hayward Tyler & Co., Ltd Curve No. E77940)

Figure 10-4. CNSS Eight-Pump Coastdown — Effect of Pump-Motor Inertia (I)

11. TASK 8 — ELECTRICALLY COUPLED EXTERNAL PUMP FLYWHEEL

11.1. Summary

This task was a study to determine the feasibility of locating the reactor coolant pump flywheel external to the pump/motor pressure housing. The external flywheel would provide a backup position in case the design of a flywheel integral with the pump/motor housing proved unacceptable. A review of several reports was made to determine industry experience with flywheels for this application. The conclusion from this review and subsequent study indicates that the concept of an external flywheel is feasible.

11.2. Objective and Scope

The objective of this task was to establish the feasibility of using an external flywheel driven generator to extend the pump coastdown. The scope was to review the report made for the Hayward Tyler Company and the FSAR for the Shippingport Atomic Power Station which uses externally mounted flywheel generators.

11.3. Results and Conclusions

Literature Review

A feasibility study of a flywheel motor-generator coupled electrically to a glandless pump was made by Kennedy & Donkin, Consulting Engineers for the Hayward Tyler Company. The purpose was to address the practical aspects of the concept including reliability, costs, and manufacturing implications. The consultants also carried out analytical studies to examine the behavior of the conceptual motor-generator. The conclusion from the report was that the concept is feasible. A review and study of this report resulted in a similar conclusion.

A review of the Shippingport Atomic Power Station SAR shows that the reactor coolant system for that plant has four pumps with flywheel generators. The use of this concept in an operating plant confirms its feasibility.

Concept Studied

The concept studied for operation was that shown in the Final Safety Analysis Report for the Shippingport Atomic Power Station (see Figure 11-1).

The flywheel generator was stated to have sufficient capacity and inertia to produce the flow coastdown required.

The Shippingport operation shows that the concept is feasible and can improve the pump coastdown inertia during loss of power. The following is a brief description of the operation:

1. The pump is started at operating conditions before the flywheel set is started. The flywheel set is mounted externally to the containment building.
2. The flywheel set is started and synchronized on a power bus that is independent from the pump motor.
3. The flywheel set is tied to the pump motor after synchronization.
4. A pump coastdown would be obtained when the pump motor breaker is operated.

Reliability

One thing that must be kept in mind is that an external pump flywheel is more complex and, therefore, less reliable than an integral flywheel. Because of this, an assumption must be made that any time there is a loss-of-power condition one of the pumps could have a coastdown failure. It is also assumed that a shutdown failure of one pump will result in a run back if the power is above a set value. It can then be postulated that at the onset of this run-back, a loss-of-power condition could occur. This means that on this loss-of-power occurrence, two pumps will have a coastdown failure. It will be required to show that: (1) the reactor can accept two coastdown failures, and (2) that no more than two will occur. The reason for the assumption that one failure may occur on loss-of-power is:

1. The voltage swings that can be expected as the flywheel set takes control could open a circuit breaker (spurious trip).
2. A failure of the pump motor and the flywheel set to synchronize.
3. A hardware failure.

The problem that exists here is that a compromise is required between two conflicting requirements imposed on the pump motor. One is that it has an emergency role during coastdown, and therefore, should have a minimum of protection against possible damage during this period which might shut off the equipment during the period of emergency. Second is that it is an environment that requires a maximum of protection during normal operation to minimize maintenance.

Further Work

1. Complete a conceptual design to incorporate the following:
 - a. Control scheme
 - b. Type of pump motor
 - c. Type of flywheel motor
 - d. Method of excitation
2. Conduct an analytical and reliability study of the completed concept.
3. Develop the maximum and minimum coastdown curves of speed Vs time. This should be available for 1 above.
4. Develop coastdown failures that the reactor can stand.

This work along with the hydraulic requirements of the pump should enable B&W to develop the documents needed to procure a preliminary design (supported by analysis) and a cost estimate.

5. Obtain additional design data on Shippingport and study it for feasibility.

Figure 11-1. Electronically Coupled External Pump Flywheel Concept

APPENDIX A
CNSS Siting Study

1. Radioactive Source Terms

Since much of the fuel data needed to calculate source term activities is not available at this stage of the CNSS preliminary core design, the source term data used in the CNSS siting study were estimated by scaling some of the needed parameters from 3600 MWt nuclear units. From CNSG information, the first fuel cycle is assumed to be 460 effective full power days (EFPD), and the equilibrium fuel cycles are assumed to be 356 EFPD. The refueling shutdowns are assumed to be 50 days each. The data used to calculate the source term activities is given in Table A-1. The source terms used to evaluate the dose consequences of accidents are shown in Table A-2; these fission product source terms are based on uranium-235 and plutonium-239 fission yields and the appropriate number of uranium and plutonium fissions during three core cycles (1172 EFPD). The source terms for reactor coolant activities are listed in Table A-3. Tritium production data calculated for the CNSS are shown in Table A-4.

Table A-1. Calculational Bases

Power, MWt	1200
Flux, n/cm ² /s	6.00+13 ^(a)
Reactor coolant volume, ft ³	6.98+03
Reactor coolant density, g/cc	0.71
Reactor coolant pressure, lbs	2250
Average reactor coolant temperature, F	594
Failed fuel, %	1.0
Primary-to-secondary leak rate, gal./min.	1.0

(a) $6.00+13 = 6.00 \times 10^{13}$

Table A-2. Radioactive Source Terms

<u>Isotope</u>	<u>Core Activity</u>	<u>Primary Coolant Activity</u>	<u>Secondary System Activity</u>	<u>Maximum Fission Product Inventory in a single fuel assembly after 72-hour decay</u>
	Ci	μCi/cc	μCi/cc	Ci
I131	3.49+07 ^(a)	3.53	1.41-05	3.79+05
132	4.04+07	1.05	4.20-06	2.99+05
133	5.95+07	3.14	1.26-05	8.88+04
134	7.51+07	3.98-01	1.59-06	0.00
135	5.95+07	1.66	6.64-06	4.74+02
Kr83m	4.29+06	2.61-01	7.83-07	2.83-04
85m	1.01+07	1.14	3.42-06	1.84
85	2.75+05	1.58+01	4.74-05	4.29+03
87	1.84+07	5.90-01	1.77-06	0.00
88	2.59+07	1.84	5.52-06	7.20-03
Xe131m	2.08+05	1.56	6.24-06	2.83+03
133m	1.42+06	2.08	8.32-06	1.24+04
133	5.91+07	1.92+02	7.68-04	6.87+05
135m	1.50+07	2.58-01	5.16-06	4.29+01
135	8.58+06	4.08	1.63-05	8.40+03
138	5.46+07	3.58-01	1.43-06	0.00

^(a) 3.49 + 07 = 3.49 x 10 +07

Table A-3. Liquid Source Activities for Design
Operation With 1.0% Failed Fuel

Isotope	RCS
Br84	1.97(-2) ^(a)
85	2.59(-3)
Kr83m	2.61(-1)
85m	1.14
85	1.58(+1)
87	5.90(-1)
88	1.84
Rb88	1.86
Sr89	3.27(-2)
90	2.71(-3)
91	1.80(-2)
92	5.97(-3)
Y90	7.08(-4)
91	2.35(-3)
Mo99	9.97(-1)
Ru106	3.19(-1)
Xe131m	1.56
133m	2.08
133	1.92(+2)
135m	2.58(-1)
135	4.08
138	3.58(-1)
I129	8.42(-8)
131	3.53
132	1.05
133	3.14
134	3.98(-1)
135	1.66
Cs134	2.84(-2)
136	8.88(-2)
137	4.13(-1)
138	5.29(-1)
Ba137m	3.85(-1)
140	5.79(-2)
La140	2.03(-2)
Ce144	1.92(-3)
Cr51	3.56(-3)
Mn54	3.97(-4)
56	1.16(-2)
Fe55	2.05(-3)
59	3.97(-4)
Co60	2.74(-3)
Zr95	3.42(-4)

(a) 1.97(-2) = 1.97×10^{-2}

Table A-4. Tritium Production

Curies per equilibrium cycle

<u>Tritium source</u>	<u>Design Basis</u>
Ternary fission	174(a)
Boron activation	247(b)
Lithium activation	<u>55</u> (c)
Total	476

- (a) 2.0% of the total ternary fission-product tritium ($D = 0.02$), i.e., 1.0% by diffusion through the cladding and 1.0% from defective fuel rods.
- (b) Assume boron concentration decreases with time and initial boron concentration is 1800 ppm.
- (c) The lithium hydroxide is 99.9% highly enriched lithium-7. Assume constant coolant concentration of 2 ppm.

2. Dose Calculations

2.1. Meteorology

Since a specific site is not being evaluated in this siting study, the X/Q values used in the study were taken from Regulatory Guides 1.3 and 1.4.

The building wake factor for ground level releases increases the amount of atmospheric dispersion and reduces the 0-8 hour X/Q value. The value of the building wake factor is a function of the minimum cross-sectional area of the building and the distance from the building. For the CNSS, the minimum cross-sectional area is $2.19 \times 10^4 \text{ ft}^2$. The resultant building wake factors are listed in Table A-5. The X/Q values used in the dose calculations of the CNSS siting study are listed in Table A-6.

Table A-5. Building Wake Factors

<u>Distance from building, meters</u>	<u>Building wake factor</u>
200	3.00
500	3.00
700	2.10
800	1.80
1000	1.60
1500	1.37
2000	1.25
2500	1.20
3000	1.15

Table A-6. x/Q Values^(a)

	0-8 hours (m^3/s) ⁻¹	8-24 hours (m^3/s) ⁻¹	1-4 days (m/s) ⁻¹	4-30 days (m^3/s) ⁻¹
<u>200m</u>				
Ground Release ^(b)	3.33(-3) ^(c)	2.5(13)	9.0(-4)	2.2(-4)
50m stack	6.5(-5)	9.5(-5)	1.9(-5)	6.0(-6)
75m stack	1.5(-5)	2.0(-5)	4.0(-6)	3.0(-6)
<u>500m</u>				
Ground Release ^(b)	6.67(-4)	5.0(-4)	1.8(-4)	4.0(-5)
50m stack	6.0(-5)	5.0(-5)	1.5(-5)	1.5(-6)
75m stack	3.0(-5)	3.4(-5)	8.0(-6)	2.6(-6)
<u>700m</u>				
Ground Release ^(b)	5.24(-4)	2.7(-4)	1.0(-4)	2.2(-5)
50m stack	5.5(-5)	3.5(-5)	1.1(-5)	3.5(-6)
75m stack	2.6(-5)	2.4(-5)	6.0(-6)	2.0(-6)
<u>1000m</u>				
Ground Release ^(b)	4.06(-4)	1.5(-4)	5.4(-5)	1.2(-5)
50m stack	4.85(-5)	2.5(-5)	8.5(-6)	2.5(-6)
75m stack	2.4(-5)	1.7(-5)	4.5(-6)	1.5(-6)

(a) These values are taken from the graphs in RG 1.3 and 1.4.

(b) Includes building wake factor.

(c) $3.33(-3) = 3.33 \times 10^{-3}$

2.2. Dose Calculational Method

The accident doses calculated in this study are based on the atmospheric release of the following gaseous nuclides:

1. Iodine 131, 132, 133, 134, and 135.
2. Xenon 131m, 133m, 133, 135m, 135, and 138.
3. Krypton 83m, 85m, 85, 87, and 88.

Consideration of where the activities were released was also included in the dose calculations. Some of the hypothetical accidents occurred inside the primary containment, some occurred in the reactor service building, and some of the accidents resulted in the direct release of radioactivity to the environment.

The release of radioactivity to the primary containment could be partially cleaned up by iodine removal sprays operating in the containment. Once the activity escaped from the containment into the annulus, consideration was given to its exhaust rate from the annulus and cleanup of the iodine isotopes in the annulus by a charcoal filtration system.

The release of radioactivity to the reactor service building can be partially cleaned up by adding a charcoal filtration system to remove the iodine isotopes. Those hypothetical accidents which released activity directly to the atmosphere were considered to have no cleanup of the releases.

3. Accident Analyses

A spectrum of accidents and transients have been evaluated for their impact on radiological releases to the containment and the environment. The transients evaluated generally can be classified into three categories depending on their relative probability of occurrence and associated radiological release. Incidents of moderate frequency, such as inadvertent rod withdrawals and pump coastdown, do not result in a radiological release to the containment or environment above normal operating levels and, therefore, are not considered relevant to siting criteria. Incidents generally classified as "infrequent" may result in radiological releases above normal operating levels and, therefore, are considered pertinent. Design basis events (relatively low probability events) may result in large radiological releases. Such events would include the LOCA, pressurizer surge line break, control rod ejection, steam generator tube failure, and the steam line break event.

Other events that are not reactor core transients are also considered (such as a fuel handling accident or a waste gas tank rupture) because they could have significant radiological consequences.

The events and accidents which are considered as a part of the accident analyses for the CNSS siting study are:

1. Fuel handling accident.
2. Loss of offsite power.
3. LOCA release of core activity to the containment.
4. The rod ejection accident.
5. The steam generator tube rupture accident.
6. The steam line break.
7. The surge line break.
8. Rupture of the waste gas decay tank.
9. Waste liquid system failure.

3.1. Fuel Handling Accident

Description of Fuel Handling Accident — Fuel handling accident is classified as an infrequent incident. When transferring irradiated fuel from the core to the spent fuel pool for storage, the reactor cavity and refueling cavity are filled with borated water at a boron concentration which ensures subcritical conditions in the core even if all control rod assemblies are withdrawn. The

fuel handling bridges are designed so that fuel cannot be raised above the minimum depth of water required for adequate shielding and protection of operating personnel.

The containment, auxiliary building, refueling cavity, fuel transfer canal, and spent fuel pool are designed in accordance with Seismic Category 1 requirements, which prevent the structures themselves from failing in the event of a safe shutdown earthquake. The spent fuel storage racks are designed to prevent any credible external missile from damaging the stored irradiated fuel.

Even though the probability of a fuel handling accident is very low because of the safety features, it is postulated that a fuel assembly is damaged during a refueling operation in the spent fuel pool breaching the cladding of the fuel rods and releasing the volatile fission products contained in the gap region of the fuel rod.

Fuel Handling Accident

Radiological Consequences — The design analysis of the fuel handling accident is based on the guidelines set forth in Regulatory Guide 1.25. The guidelines used in this analysis include:

1. All fuel pins in the fuel assembly fail.
2. Overall effective decontamination factor of 100 for iodine and one for noble gases.
3. Iodine species above the spent fuel pool are 75% inorganic iodine and 25% organic iodine.
4. If charcoal filters are installed on the exhaust of the fuel handling area, they will have a removal efficiency of 90% for inorganic iodine and 70% for organic iodine.
5. The activity released from the assembly is based on the maximum fission product inventory in the spent fuel assembly after 72 hours of decay (from Table A-2).

The activities released to the fuel handling building are listed in Table A-7 and the radiological consequences are listed in Table A-8.

Table A-7. Activity Releases From a
Fuel Handling Accident

<u>Isotope</u>	<u>Activity Released to the Fuel Handling Area, Ci</u>	
	<u>Inorganic</u>	<u>Organic</u>
I131	2.84+02 ^(a)	9.47+01
132	2.24+02	7.47+01
133	6.25+01	2.09+01
134	0.00	0.00
135	3.55-01	1.19-01
Total		
Kr 83m	2.83-05	
85m	1.84-01	
85	1.29+03	
87	0.00	
88	7.20-04	
Xe131m	2.83+02	
133m	1.24+03	
133	6.87+04	
135m	4.29	
135	8.40+02	
138	0.00	

(a) $2.84 + 02 = 2.84 \times 10^{+02}$

Table A-8. Dose Results of Fuel Handling Accident

	<u>2-hour and 30-day doses with no filtration, rem</u>		<u>2-hour and 30-day doses with iodine filtration, rem</u>	
	<u>Thyroid</u>	<u>Whole Body</u>	<u>Thyroid</u>	<u>Whole Body</u>
<u>Ground level release^(b)</u>				
200	7.05+02 ^(a)	5.76	1.06+02	5.15
500	1.41+02	1.15	2.12+01	1.03
700	1.11+02	9.07-01	1.66+01	8.10-01
1000	8.59+01	7.02-01	1.29+01	6.28-01
<u>50 meter stack</u>				
200	1.38+01	1.13+01	2.06	1.01-01
500	1.27+01	1.04-01	1.91	9.28-02
700	1.16+01	9.52-02	1.75	8.50-02
1000	1.02+01	8.30-02	1.54	7.40-02
<u>75 meter stack</u>				
200	3.17	2.60-02	4.76-01	2.32-02
500	6.35	5.19-02	9.52-01	4.64-02
700	5.50	4.50-02	8.25-01	4.02-02
1000	5.08	4.15-02	7.62-01	3.71-02

(a) $7.05+02 = 7.05 \times 10^2$.

(b) Building wake factor included.

3.2. Loss of AC Power to Station Auxiliaries (Station Blackout)

Description of Loss of AC Power Event

The loss of AC power event is defined to be the loss of AC power to the station auxiliaries, exclusive of the vital instrumentation and control power system and diesel generators. A loss of AC power will occur whenever the turbine generator is tripped, for any reason, and all other offsite power sources also fail. Following a loss of AC power, the vital instrumentation and control power system will continue to supply power to essential instrumentation and control circuits. The emergency generator units in the standby power supply will be started and loaded with essential AC loads as required, but in any event the reactor coolant pumps and the condenser cooling water pumps will not have power.

The sequence of events and the evaluation of consequences for this accident are as follows:

1. A turbine trip results in a reactor trip (gravity insertion of the control rods), closure of the turbine stop valves, and the loss of the reactor coolant pumps and condenser cooling water pumps.
2. After the turbine stop valves close, excessive temperatures and pressures in the RC system are automatically prevented by relieving excess steam through the modulating atmospheric dump valves and the bank of secondary side code safety valves with the lowest setpoint. Condenser dump valve steam relief is unavailable due to the loss of power to the condenser cooling water pumps.

The modulating atmospheric dump valves are opened automatically by the control system. The secondary side code safety valves are self-actuated by high steam pressure. Excess steam is relieved until the RC system pressure is below the pressure corresponding to the lowest setpoint of the steam safety valves. Thereafter, the modulating atmospheric dump valves are used to remove decay and system heat until a-c power is restored by the emergency generators. The operator can then reduce the reactor coolant temperature to 305F and place the decay heat removal system into operation to continue the cooldown to ambient conditions.

3. Following loss of power to the reactor coolant pumps, the RC system flow decays without the occurrence of fuel damage. The energy stored in the main turbine is not required to supply power to the reactor coolant pumps following the loss of power. Decay heat removal after coastdown of the reactor coolant pumps is provided by the natural circulation characteristics of the RC system.
4. The auxiliary feedwater system provides cooling water to the steam generators.

The total mass of secondary steam relieved to the atmosphere and the assumed venting time is provided in Table A-9.

The loss of all offsite power concurrent with loss of the unit generator does not result in a radiation exposure to the public or to on-site personnel because only secondary steam is discharged to the atmosphere. However, radiation releases do occur when the unit is operated with defective fuel cladding and steam generator tube leakage, which is the case considered in conjunction with this accident. The following sequence of events is considered:

1. It is assumed that the operator takes action to ensure full opening of the modulating atmospheric dump valves 10 minutes after the loss of power. This action provides cooling in excess of the decay heat rate. This action shortens the release time and helps to minimize the doses which result from the atmospheric release. These valves are automatically actuated and have reset capability by operator adjustment of setpoint pressure.
2. Cooling down at the maximum available rate requires additional time to reach a temperature below the saturation temperature corresponding to the setpoint pressure for the steam safety valve having the lowest setting.
3. The steam generator with tube leakage is completely isolated by closing its modulating atmospheric dump valve and the main steam isolation valves. The leaking steam generator is identified by a high radiation level reading on the radiation monitors (N-16) which must be on the system. The other steam generators are used to remove decay heat. The steam flow required to remove decay heat and sensible heat up to this point is assumed to be released directly to the atmosphere, resulting in the most conservative values for activity release.

Table A-9.

Steam relieved to the atmosphere, lbm	200,000
Steam venting time to atmosphere, min. (a)	55

(a) All coolant from 1 gpm leak for 55 minutes is assumed to pass directly to the atmosphere.

Radiological Consequences

Assumptions for dose calculations:

1. Leakage rate from primary coolant to secondary system of 1 gallon per minute is vented directly to the atmosphere for 55 minutes (as a puff release) due to the assumed loss of condenser.
2. Mass of steam released to the atmosphere is 200,000 lb.

Thus, from the above two assumptions, the total activity released would be the reactor coolant leakage plus the secondary system leakage. Table A-10 shows the activities released as a result of a loss of offsite power event, and Table A-11 shows the dose consequences of these activities being released.

Table A-10. Activities Released to Atmosphere as a Result of Offsite Power (Station Blackout)

<u>Isotope</u>	<u>Activity Released, Ci</u>
I131	7.35E-01 ^(a)
132	2.09E-01
133	6.54E-01
134	8.30E-02
135	3.45E-01
Kr83m	5.44E-02
85m	2.37E-01
85	3.29
87	1.23E-01
88	3.84E-01
Xe131m	3.25E-01
133m	4.34E-01
133	4.00E+01
135m	5.37E-01
135	8.49E-01
138	7.45E-02

(a) $7.35-01 = 7.35 \times 10^{-1}$

Table A-11. Dose Consequences of Loss of All Offsite Power (Station Blackout)

<u>Distance (meters)</u>	<u>Doses (short term release), rem</u>	
	<u>Thyroid</u>	<u>Whole Body</u>
<u>Ground Level Release</u> ^(b)		
200	1.62	5.46-03
500	3.25-01 ^(a)	1.09-03
700	2.56-01	8.59-02
1000	1.98-01	6.65-04

(a) $3.25-01 = 3.25 \times 10^{-01}$.

(b) It is assumed the steam safety valves release to the atmosphere and did not vent to a stack. Building wake factor is included in x/Q values used in these calculations.

3.3. Design Basis Loss-of-Coolant Accident (LOCA)

The assumptions used in the LOCA analysis include:

1. 25% of the equilibrium radioactive iodine inventory developed from maximum full power operation of the core is assumed to be immediately available for leakage from the primary reactor containment. Of this iodine available for release, 91% is elemental iodine, 5% is particulate iodine, and 4% is organic iodides.
2. 100% of the equilibrium radioactive noble gas inventory developed from maximum full power operation of the core is assumed to be immediately available for leakage from the reactor containment.
3. Credit is taken for radiological decay during holdup in the containment and annulus.
4. The primary containment leak rate is assumed to be 0.1% volume per day, 0.2% volume per day, and 0.3% volume per day. (Each case is studied individually.) In each case, the containment leak rate is reduced by 50% after the first 24 hours for the duration of the accident.
5. The annulus exhaust rate is assumed to be 0.3, 3.0, and 30 air changes per hour through charcoal filters with iodine removal fractions of 0.98 for elemental and particulate iodine and 0.94 for organic iodine. Each exhaust rate value is studied individually.
6. The annulus recirculation rate is assumed to be 0 and 6 air changes per hour through charcoal filters with iodine removal fractions of 0.98 for elemental and particulate iodine and 0.94 for organic iodine. Each recirculation rate is studied individually.
7. The containment sprays are studied with and without iodine removal capability. The sprays with iodine removal capability are assumed to be sodium hydroxide sprays which operate for 2 hours. For elemental iodine, the maximum removal rate of sodium hydroxide sprays is 10.0 per hour. For particulate iodine, the removal rate is 0.45 per hour and 0.0 for organic iodine. Without the addition of sodium hydroxide to the containment sprays, there is no reduction in the amount of the iodine isotopes available for leakage from the containment.
8. The meteorology used in the LOCA analysis is from Regulatory Guide 1.4.

The annulus recirculation and exhaust system is shown in Figure A-1. The radiological consequences of this accident are calculated for 36 cases using various sets of containment system parameters and the (X/Q) for a ground level release at 200 meters. The variation of parameters and dose consequences are shown in Table A-12.

Figure A-1. Containment Design With Stack

Table A-12. Dose Consequences of a Design Basis LOCA at 200 Meters

Case Number	CONTAINMENT DESIGN PARAMETERS		ANNULUS DESIGN PARAMETERS		2-HOUR DOSES rem		30-DAY DOSES rem	
	Leak Rate, % Volume per Day	Sprays	Exhaust Rate, hr ⁻¹	Recirculation hr ⁻¹	Thyroid	Whole Body	Thyroid	Whole Body
1	.1	*	0.3	6.0	3.76(-1) ¹	1.08	2.41	1.00(+1)
2		*	0.3	0	2.36	1.08	4.09(+1)	1.00(+1)
3		*	3.0	6.0	2.65	3.96	1.67(+1)	1.46(+1)
4		*	3.0	0	7.06	3.97	4.76(+1)	1.46(+1)
5		*	30.0	6.0	6.71	5.09	4.07(+1)	1.57(+1)
6		*	30.0	0	7.98	5.09	4.84(+1)	1.58(+1)
7			0.3	6.0	1.95	1.08	1.91(+1)	1.00(+1)
8			0.3	0	1.07(+1)	1.10	3.33(+2)	1.03(+1)
9			3.0	6.0	1.39(+1)	3.98	1.33(+2)	1.47(+1)
10			3.0	0	3.62(+1)	4.04	3.87(+2)	1.50(+1)
11			30.0	6.0	3.60(+1)	5.16	3.30(+2)	1.60(+1)
12			30.0	0	4.28(+1)	5.18	3.94(+2)	1.61(+1)
13	.2	*	0.3	6.0	7.52(-1)	2.16	4.82	2.00(+1)
14		*	0.3	0	4.72	2.17	8.16(+1)	2.01(+1)
15		*	3.0	6.0	5.31	7.91	3.31(+1)	2.92(+1)
16		*	3.0	0	1.41(+1)	7.94	9.51(+1)	2.92(+1)
17		*	30.0	6.0	1.34(+1)	1.02(+1)	8.13(+1)	3.15(+1)
18		*	30.0	0	1.60(+1)	1.02(-1)	9.66(+1)	3.15(+1)
19			0.3	6.0	3.90	2.17	3.81(+1)	2.00(+1)
20			0.3	0	2.14(+1)	2.21	6.65(+2)	2.06(+1)
21			3.0	6.0	2.78(+1)	7.97	2.65(+2)	2.94(+1)
22			3.0	0	7.23(+1)	8.07	7.73(+2)	2.99(+1)
23			30.0	6.0	7.19(+1)	1.03(+1)	6.58(+2)	3.21(+1)
24			30.0	0	8.57(+1)	1.04(+1)	7.86(+2)	3.22(+1)
25	.3	*	0.3	6.0	1.13	3.24	7.22	3.00(+1)
26		*	0.3	0	7.07	3.25	1.22(+2)	3.01(+1)
27		*	3.0	6.0	7.96	1.19(+1)	4.98(+1)	4.37(+1)
28		*	3.0	0	2.12(+1)	1.19(+1)	1.42(+2)	4.38(+1)
29		*	30.0	6.0	2.01(+1)	1.53(+1)	1.22(+2)	4.72(+1)
30		*	30.0	0	2.39(+1)	1.53(+1)	1.45(+2)	4.72(+1)
31			0.3	6.0	5.85	3.25	5.71(+1)	3.00(+1)
32			0.3	0	3.21(+1)	3.31	9.96(+2)	3.08(+1)
33			3.0	6.0	4.18(+1)	1.20(+1)	3.98(+2)	4.41(+1)
34			3.0	0	1.09(+2)	1.21(+1)	1.16(+3)	4.48(+1)
35			30.0	6.0	1.08(+2)	1.55(+1)	9.86(+2)	4.81(+1)
36			30.0	0	1.29(+2)	1.55(+1)	1.18(+3)	4.83(+1)

*Indicates chemical addition for iodine removal

¹3.76(-1) = 3.76 x 10⁻¹

3.4. Rod Ejection Accident

Description of Rod Ejection Accident

The rod ejection accident is considered to be a limiting fault which has the potential for the release of significant amounts of radioactivity.

A rapid reactivity excursion is assumed to be caused by the physical failure of a pressure barrier component in the control rod drive assembly. Such a failure could cause a pressure differential to act on a control rod assembly and rapidly eject the assembly from the core region.

For the purposes of calculating radiological consequences resulting from the rod ejection accident, the rod ejection accident is assumed to result in a small LOCA on the top of the reactor vessel at the penetration of the CRDM lead screw.

An estimate of the total mass leaked through the resultant breach in the pressure vessel boundary caused by the ejected rod is based on the primary pressure response and rupture area.

The fuel rod damage resulting from a rod ejection accident depends largely on the rate at which the thermal energy resulting from the nuclear excursion is released to the coolant. If the fuel rods remain intact while the excursion is being terminated by the negative Doppler coefficient and by reactor trip, then the energy release rate is limited by a relatively low surface to volume ratio for heat transfer. The energy stored in the fuel rods will then be gradually released to the coolant (over a period of several seconds) at a rate that poses no threat to the integrity of the reactor coolant system. However, if the magnitude of the nuclear excursion is so great that the fuel rod cladding does not remain intact, then both fuel and cladding may be dispersed into the coolant to such an extent that the heat transfer rate increased significantly.

In determining the environmental consequences from this accident, a conservative approach is taken in computing the extent of departure from nucleate boiling (DNB) experienced in the fuel rods. All fuel rods that undergo DNB to any extent are assumed to experience cladding failure with subsequent release of all gap activity. Actually, most of the fuel rods will recover from DNB, and no fission product release will occur. The fuel rods that experience DNB at beginning of life (BOL) are assumed to have end of life (EOL) gap activities.

The percent of the core in DNB following the ejection of a 0.65% $\Delta k/k$ control rod while operating at full power is given in Table A-13. The mass of coolant released to containment following a control rod ejection is also provided in Table A-13.

Table A-13. Summary of Rod Ejection Accident
for Radiological Consequences

Mass of coolant released to containment	37,500 lbm
% core in DNB	43%

Radiological Consequences

For the rod ejection accident, a fraction of the core undergoes DNB, releasing radioactive iodine and noble gases to the containment. Since an ejected control rod is assumed to puncture the reactor vessel, all the reactor coolant activity is also assumed to be released to the containment atmosphere. The activity released to the containment and available for leakage to the environment is calculated for a design basis analysis. For conservatism, primary coolant is also leaked via a pre-existing 1 gpm leak to the secondary system. A total of 40 pounds of primary coolant is assumed to leak into the secondary system and be released directly to the environment. The assumptions used in the analysis of this accident are taken from Regulatory Guide 1.77. The assumptions used in this analysis include:

1. The amount of activity accumulated in the fuel-clad gap is assumed to be 10% of the iodines and noble gases in the reactor core.
2. All the noble gas activity and 50% of the iodine activity in the fuel-clad gap is available for release to the secondary system.
3. All the noble gas and iodine activities in the reactor coolant are evenly distributed. A total of 40 pounds of primary coolant is leaked to the secondary system and is then vented to the atmosphere.
4. Loss of offsite power is assumed; therefore, the decontamination factor is 1.0 for the iodines and noble gases.
5. 43% of the gap activity is released due to the percentage of the core in DNB.

The radiological consequences of the two types of releases which result from this accident in Table A-15 are based on the activity releases in Table A-14.

Table A-14. Activity Releases Due to Rod Ejection Accident

ISOTOPE	ACTIVITY RELEASE TO SECONDARY, Ci	ACTIVITY RELEASE TO CONTAINMENT, Ci		
		91% ELEMENTAL	5% PARTICULATE	4% ORGANIC
I131	9.61E+01 ^(a)	6.83E+05	3.75E+04	3.00E+04
132	8.63E+01	7.91E+05	4.35E+04	3.48E+04
133	1.65E+02	1.17E+06	6.40E+04	5.62E+04
134	2.08E+02	1.47E+06	8.05E+04	6.44E+04
135	1.65E+02	1.16E+06	6.40E+04	5.12E+04
Kr 83m	2.37E+01		1.84E+05	
85m	5.60E+01		4.34E+05	
85	1.92		1.49E+04	
87	1.02E+02		7.91E+05	
88	1.43E+02		1.11E+06	
Xe 131m	1.19		9.25E+03	
133m	7.93		6.15E+04	
133	3.33E+02		2.58E+06	
135m	8.32E+01		6.45E+05	
135	4.77E+01		3.70E+05	
138	3.03E+02		2.35E+06	

(a) $9.61\text{E}+01 = 9.61 \times 10^{-01}$.

Table A-15. Dose Consequences of a Rod Ejection Accident

Distance (meters)	Primary Release <u>Dose results, rem</u>				Secondary Release <u>Dose Results, rem</u>	
	<u>2-hour</u>		<u>30-day</u>		<u>2-hour & 30-day</u>	
	<u>Thyroid</u>	<u>Whole Body</u>	<u>Thyroid</u>	<u>Whole Body</u>	<u>Thyroid</u>	<u>Whole Body</u>
Ground Level Release ^(a)						
200	5.68-01 ^(b)	1.71-01	3.69	6.33-01	2.77+02	1.63
500	1.14-01	3.43-02	7.30-01	1.26-01	5.54+01	3.27-01
700	8.93-02	2.69-02	4.80-01	8.80-02	4.35+01	2.57-01
1000	6.92-02	2.09-02	3.24-01	6.36-02	3.37+01	1.99-01

(a) includes building wake factor

(b) 5.68-01 = 5.68×10^{-01}

3.5. Steam Generator Tube Rupture

Description of Steam Generator Tube Rupture

A double-ended rupture of one steam generator tube is assumed to occur while the reactor is operating at full power (with coolant contamination corresponding to continuous full power operation with 1% defective fuel). For this occurrence, radioactivity fission product activity contained in the reactor coolant would be released to the secondary system. Some of the radioactive noble gas and iodine activity entering the secondary system could be released to the atmosphere through the condenser air removal system, the steam safety valves, or the atmospheric dump valves. A double-ended rupture of the steam generator tube is considered a conservative assumption; the more probable mode of failure would be minor leaks in the tubes or welds.

In analyzing the consequences of this failure, the following sequence of events is assumed to occur:

1. The reactor is operating at full power, nominal conditions.
2. A double-ended rupture of one steam generator tube occurs with unrestricted discharge from each end to the secondary side of the steam generator.
3. The steam mixture containing reactor coolant activity passes through the turbine to the condenser. Some of the volatile fission product activity is passed out the condenser air ejector vent discharge causing a sudden increase in activity level indication and a monitor alarm.
4. The initial tube leak rate exceeds the normal makeup to the reactor coolant system and the system pressure decreases eventually causing a low reactor coolant system pressure trip. Offsite power is assumed to be lost at the time of reactor trip; thus losing the capability of a turbine bypass system to remove heat.
5. After reactor trip, the reactor coolant system pressure continues to decrease until the high pressure injection system is actuated. The capacity of one high pressure injection pump is sufficient to compensate for the tube leakage and to increase the pressurizer water level.
6. The ESFAS low primary system pressure signal initiates isolation of the affected steam generator by closure of the main steam and feedwater isolation valves. The N-16 monitors provide sufficient indication for the operator to determine the affected steam generator.

7. After secondary system isolation has occurred and the affected generator has been determined, some provision may be made for long term decay heat removal by reopening the main steam isolation valves on the unaffected steam generators to use the atmospheric dump valves or the turbine bypass system for heat removal.

In the event of a loss of site power, heat removal would be achieved via the pressurizer relief and safety valves or via steam relief valves downstream of the main steam isolation valves.

The times and leakage associated with the above sequence are summarized in Table A-16.

The sequence of events and release are based on the secondary steam relief valves having a setpoint the same as the primary side safety valves. This feature precludes the need for rapid cooldown of the primary system to minimize radiological consequences.

The leakage of reactor coolant from the primary side to the secondary system via the steam generator tube rupture is released to the atmosphere via the condenser prior to turbine and reactor trip. Since loss of offsite power is assumed to occur simultaneously with the turbine and reactor trip, all reactor coolant leakage between the trip and steam generator solution must be considered to pass directly to the atmosphere via downstream relief valves prior to the closure of the main steam isolation valve.

Table A-16. Summary of Steam Generator
Tube Failure Analysis

TIME SEQUENCE

• Rupture (minutes)	0
• Low Pressure Trip (Minutes)	8
• ESFAS Signal for Secondary System Isolation	18
• Final Steam Generator Isolation	18.1

REACTOR COOLANT LEAKAGE VALUES

• Reactor Coolant Leakage to Affected Steam Generator Prior to Reactor Trip (lbm)	24,230
• Reactor Coolant Leakage to Affected Steam Generator Between Reactor Trip and Affected Steam Generator Isolation (lbm)	29,550

Radiological Consequences

The total reactor coolant activity released to the environment consists of the following three sources:

Source 1 — Reactor coolant activity released to the environment from reactor coolant leakage to the affected steam generator which is vented to the condenser prior to reactor trip. Assume an iodine partition factor of 7.5(-3). Partition fraction for Xe,Kr is one.

Source 2 — Reactor coolant activity released directly to the environment from the reactor coolant leakage to the affected steam generator after reactor trip and prior to steam generator isolation.

Source 3 — Reactor coolant activity released to the environment through the secondary system from a 1 gpm primary-to-secondary leak rate in the unaffected steam generator. This reactor coolant leakage for the first 19 minutes is vented to the environment through the condenser. Due to loss of offsite power for the remaining 8 minutes (19 min. - 27 min.), the reactor coolant leakage is vented directly to the environment without iodine removal benefit of the condenser.

These activities are summarized in Table A-17. The resulting doses are in Table A-18.

Table A-17. Activity Release Due to Steam
Generator Tube Rupture

<u>Isotope</u>	<u>Total activity released to the environment, Ci</u>
I 131	6.82+01 ^(a)
132	2.04+01
133	6.11+01
134	7.74
135	3.22+01
Kr 83m	9.10
85m	3.97+01
85	2.82+02
87	2.05+01
88	6.40+01
Xe 131m	5.44+01
133m	7.24+01
133	6.69+03
135m	9.00
135	1.42+02
138	1.25+01

(a) $6.82+01 = 6.82 \times 10^{+01}$.

Table A-18. Radiological Consequences of a
Steam Generator Tube Rupture

<u>Distance, meters</u>	<u>2-hour & 30-day dose results, rem</u>	
	<u>Thyroid</u>	<u>Whole body</u>
Ground level release ^(a)		
200	1.51+02 ^(b)	7.99-01
500	3.02+01	1.60-01
700	2.38+01	1.26-01
1000	1.84+01	9.74-02

(a) Including building wake factor.

(b) $1.51+02 = 1.51 \times 10^{+02}$.

3.6. Steam Line Break

Description of Steam Line Break

The loss of secondary coolant due to a double-ended steam line rupture has been analyzed for its effects and consequences relative to radioactive releases.

The total mass of steam released during a double-ended steam line break downstream of the main steam isolation valve was calculated considering critical flow blowdown through a double-ended 12-inch ID steam line. A qualified isolation signal was assumed to occur, such as low steam line pressure, that would initiate closure of all main steam line isolation valves, thus terminating blowdown. The mass is released directly to the environment since the break is outside containment. This constitutes the most severe release of secondary steam, which is considered along with steam generator tube leakage for determining environmental consequences. It is also assumed that long term steam generator primary-to-secondary leakage can be released to the environment via the main steam safety valves during the normal cooldown process, if the main condenser is unavailable for steam relief.

No core damage is expected to occur for this event because the minimum DNBR is greater than 1.3.

The total mass released via the rupture is estimated to be 38,500 lbm of saturated steam.

It was assumed in this analysis that an engineered safety features actuation system signal would actuate on low steam line pressure causing main steam isolation valve closure. This signal presently does not exist on the CNSS design but must be incorporated in CNSS-2 design to prevent all steam generators from blowing down as a result of a steam line break outside containment. This analysis also assumed that all steam pipes from all 10 steam generators are headered together downstream of the main steam isolation valves.

Radiological Consequences

The activities released to the environment are listed in Table A-19 and consist of the activity in the saturated steam and in a 1 gpm primary-to-secondary leak. The maximum length of cooldown to decay heat removal temperature is assumed to be 8 days. Thus, a 1 gallon per minute primary-to-secondary leakage is assumed to be released to the atmosphere for the 8 days. This release is through the

safety relief valves due to loss of condenser; therefore, the decontamination factor for all isotopes is one.

Table A-19. Activity Release Due to
Steam Line Break

<u>Isotope</u>	<u>Activity in Saturated Steam, Ci</u>	<u>Activity in 8-day cooldown, Ci</u>
I 131	3.47-04 ^(a)	1.54+02
132	1.03-04	4.57+01
133	3.10-04	1.37+02
134	3.91-05	1.74+01
135	1.63-04	7.23+01
Kr 83m	1.93-05	1.14+01
85m	8.41-05	4.97+01
85	1.17-03	6.89+02
87	4.35-05	2.57+01
88	1.36-04	8.02+01
Xe131m	1.54-04	6.80+01
133m	2.05-04	9.07+01
133	1.89-02	8.37+03
135m	1.27-04	1.13+01
135	4.01-04	1.77+02
138	3.52-05	1.57+01

(a) 3.47-04 = 3.47×10^{-04}

Table A-20. Dose Consequences From
Steam Line Break

Distance^(a)
(meters)

	<u>2-hour & 30-day doses, rem</u> Puff of Steam		<u>2-hour doses, rem</u> 8-day cooldown	
	<u>Thyroid</u>	<u>Whole Body</u>	<u>Thyroid</u>	<u>Whole Body</u>
200	7.68-04 ^(b)	2.53-06	8.05+01	2.49-01
500	1.54-04	5.07-07	1.61+01	4.98-01
700	1.21-04	3.99-07	1.27+01	3.91-02
1000	9.36-05	3.09-07	9.81	3.03-02

30-day doses, rem
8-day cooldown

	<u>Thyroid</u>	<u>Whole Body</u>
200	2.52+02	7.67-01
500	5.05+01	1.54-01
700	3.79+01	1.13-01
1000	2.85+01	8.37-02

^(a)Ground level release with building wake factor included.

^(b)7.68-04 = 7.68×10^{-04}

3.7. Surge Line Break

Description of Surge Line Break

The surge line break is assumed to be a double-ended (200%) break of the pipe leading to the pressurizer which allows reactor coolant to be released to the containment. The activity released to the containment and available for leakage to the environment is calculated for a design analysis. It is assumed that all the noble gas and iodine activity contained in 3283 ft³ of reactor coolant is released to the containment atmosphere. The 3283 ft³ of reactor coolant comes from the mass-energy release rate for a 0.175 ft² double-ended surge line break. Furthermore, it is assumed that 50% of the iodines will plateout on the containment, and therefore, would not be available for leakage to the environment. The activity released from the secondary relief valves as a result of pressure transient is neglected.

The following assumptions were also used in the analysis of the pressurizer surge line break:

1. The containment leak rate is 0.1% volume per day.
2. There are sodium hydroxide sprays in the containment which operate for two hours. For elemental iodine, the maximum removal rate of these sprays is 10.0 per hour. For particulate iodine, the removal rate is 0.45 per hour and 0.0 for organic iodine.
3. The annulus exhaust rate is 3.0 air changes per hour through charcoal filters with iodine removal fractions of 0.98 for elemental and particulate iodine and 0.94 for organic iodine.
4. The annulus recirculation rate is 0.12 air changes per hour with iodine removal fractions of 0.98 for elemental and particulate iodine and 0.94 for organic iodine.

The activities released to the containment are listed in Table A-21 and the radiological consequences of these activities being released to the containment are in Table A-22.

Table A-21. Activity Release Due to
Surge Line Break

<u>Isotope</u>	<u>Activity released to the primary containment, Ci</u>
I 131	1.64+02 ^(a)
132	4.88+01
133	1.46+02
134	1.85+01
135	7.71+01
Kr 83m	2.43+01
85m	1.06+02
85	1.47+03
87	5.49+01
88	1.71+02
Xe131m	1.45+02
133m	1.93+02
133	1.79+04
135m	2.40+01
135	3.79+02
138	3.33+01

(a) $1.64+02 = 1.64 \times 10^{+02}$

Table A-22. Radiological Consequences of a Surge Line Break

<u>Distance</u> <u>(meters)</u>	<u>2-hour doses, rem</u>		<u>30-day doses, rem</u>	
	<u>Thyroid</u>	<u>Whole Body</u>	<u>Thyroid</u>	<u>Whole Body</u>
<u>Ground level release</u> ^(a)				
200	9.73-05 ^(b)	1.11-04	7.06-04	1.60-03
500	1.95-05	2.23-05	1.40-04	3.17-04
700	1.53-05	1.75-05	9.07-05	1.95-05
1000	1.19-05	1.36-05	6.04-05	1.24-05

(a) The x/Q values for these distances include the building wake factor.

(b) $9.73 - 05 = 9.73 \times 10^{-05}$.

3.8. Radwaste Gas System Failure

Description of Waste Gas Decay Tank Failure

The limiting accident associated with the radwaste gas system would be an unexpected and uncontrolled release to the atmosphere of the radioactive fission product gases that are stored in the tanks. The failure is postulated to occur due to an operator error or a valve malfunction which are infrequent incidents. The waste gas decay tanks (WGDTs) are quality group C and Seismic Category I and are isolated from one another during use.

The failure is postulated to occur during refueling shutdown immediately after all the gases are stripped from the reactor coolant system and transferred to the WGDT which is to be affected. In order to reduce the iodine activity, the reactor coolant is passed through two ion exchangers prior to degasification. In the degasifier, all of the noble gases and all of the volatile iodines are assumed to be removed and transferred to the WGDT. Five percent of the iodine in the coolant is assumed to be volatile. Assuming that the gas stripper is 100% efficient, the transfer time is conservatively calculated to be 7.62 hours. Only one tank releases its contents to the auxiliary building in the event of a failure.

Radiological Consequences

For the purpose of determining radiological consequences, the contents of the failed WGDT are assumed to be released to the environment as a puff ground level release from the auxiliary building. The maximum inventory of the WGDT is calculated using the total reactor coolant mass which has decayed for 30 hours. No credit for radioactive decay is taken during the release period.

The assumptions in Regulatory Guide 1.24 are used in calculating the radiological consequences of this hypothetical accident. The activities released as a result of this occurrence are in Table A-23, and the radiological consequences of these activities being released are in Table A-24.

Table A-23. Activities Released as a Result
of the Rupture of a Waste Gas
Decay Tank

<u>Isotope</u>	<u>Activity, Ci</u>
I 131	4.48-01 ^(a)
I 132	1.88-05
I 133	1.73-01
I 134	0.00
I 135	1.12-02
Kr 83m	6.91-04
Kr 85m	2.03
Kr 85	3.12+03
Kr 87	8.71-06
Kr 88	8.69-02
Xe131m	2.87+02
Xe133m	2.80+02
Xe133	3.23+04
Xe135m	0.00
Xe135	8.07+01
Xe138	0.00

(a) 4.48-01 = 4.48×10^{-01}

Table A-24. Dose Consequences of Waste Gas
Decay Tank Rupture

<u>Distance</u> <u>(meters)</u>	<u>2-hour & 30 day doses, rem</u>	
	<u>Thyroid</u>	<u>Whole Body</u>
Ground level release ^(a)		
200	8.85-01 ^(b)	2.27
500	1.77-01	4.55-01
700	1.39-01	3.57-01
1000	1.08-01	2.77-01

^(a)The x/Q valves for these doses include the building wake factor.

^(b) $8.85-01 = 8.85 \times 10^{-01}$

3.9. Liquid Waste System Failures

The radiological consequences of a release to the atmosphere of radioactive fission products resulting from a failure of tanks containing radioactive liquids were calculated by United Engineers and Constructors (UE&C). Results of this analysis indicate that the doses due to atmospheric releases from liquid tank failures are well below the 10CFR100 guidelines.

Although liquid releases and subsequent concentrations of radioactive effluents at potable water supplies via liquid pathways were not evaluated, there are certain inherent design features of the CNSS that tend to minimize the consequences of such accidents. Some of these features are discussed below.

During plant operation, radioactive liquid waste material is normally contained within the liquid waste system, the auxiliary systems, or other containers or vessels designed to prevent the uncontrolled release of radioactivity to the environment. The release of radioactivity from small spills or leaks of radioactive material within the plant would be significantly mitigated by the general facility design of the CNSS; i.e., the filtration and cleanup of exhaust ventilation and the general availability of contamination control equipment. Also, the concrete walls and floors of the compartments in which the tanks are mounted are coated to guard against radioactive fluid contamination with the floor sloped to a floor drain located in the compartment. Each compartment in which a tank containing potentially radioactive fluids is installed is designed such that the entire tank volume could be contained within that compartment. Therefore, any radioactive liquid leaks can be detected, contained, and treated within the reactor service building.

Based upon the analysis performed, it is concluded that waste liquid system failures do not pose siting problems for the CNSS plant.

APPENDIX B

DIVISION OF RESPONSIBILITY FOR SCOPE OF SUPPLY
FOR 400 MWE CNSS

<u>Item</u>	<u>Preliminary Design</u> <u>Design Review</u>		<u>Detail Design</u> <u>Design Review</u>		<u>Specifications</u> <u>& Procurement</u>
STRUCTURES					
1. <u>Reactor Service Building</u> (Includes Fuel Handling, Radioactive Waste Storage & Decontamination Areas)					
A. Basic Design Data	A/E	B&W			
B. Building Arrangement	A/E	B&W	A/E	B&W	A/E
C. Excavation & Backfill	A/E		A/E		A/E
D. Bearing Piles & Caissons	A/E		A/E		A/E
E. Substructure	A/E		A/E		A/E
F. Superstructure	A/E		A/E		A/E
G. Building Service					
(1) Plumbing & Drainage					
(a) Controlled	A/E		A/E		A/E
(b) Noncontrolled	A/E		A/E		A/E
(2) Heating, Ventilating, & Air Conditioning	A/E	B&W	A/E		A/E
(3) Lighting	A/E		A/E		A/E
(4) Fire Protection	A/E		A/E		A/E
H. Cranes & Hoists	A/E	B&W	A/E		A/E

<u>Item</u>	<u>Preliminary Design</u>		<u>Detail Design</u>		<u>Specifications</u>
	<u>Design Review</u>		<u>Design Review</u>		<u>& Procurement</u>
STRUCTURES (Continued)					
2. <u>Control Building</u> (Mechanical & Electrical)					
A. Arrangement (Building & Equipment)	A/E	B&W	A/E		
B. Excavation & Backfill	A/E		A/E		A/E
C. Substructure	A/E		A/E		A/E
D. Superstructure	A/E		A/E		A/E
E. Building Service					
(1) Plumbing & Drainage	A/E		A/E		A/E
(2) Heating, Ventilating, & Air Conditioning	A/E,	B&W	A/E		A/E
(3) Lighting	A/E		A/E		A/E
(4) Fire Protection	A/E	B&W	A/E		A/E
3. <u>Diesel Generator Building</u>					
A. Building Arrangement	A/E		A/E		A/E
B. Excavation & Backfill	A/E		A/E		A/E
C. Substructure	A/E		A/E		A/E
D. Superstructure & Miscellaneous Steel	A/E		A/E		A/E

<u>Item</u>	<u>Preliminary Design</u>		<u>Detail Design</u>		<u>Specifications</u>
	<u>Design</u>	<u>Review</u>	<u>Design</u>	<u>Review</u>	<u>& Procurement</u>
STRUCTURES (Continued)					
3. <u>Diesel Generator Building (Continued)</u>					
E. Building Services					
(1) Heating & Ventilating	A/E		A/E		A/E
(2) .Lighting	A/E		A/E		A/E
(3) Fire Protection	A/E		A/E		A/E
F. Standby Diesel Generators Controls & Auxiliaries					
	A/E	B&W	A/E		A/E
(1) Oil Storage Tanks	A/E		A/E		A/E
(2) Oil Transfer Pump	A/E		A/E		A/E
G. Electrical Loads	A/E-B&W	A/E	A/E	B&W	A/E
4. <u>NSS Containment Structure</u>					
A. Basic Design Criteria	A/E-B&W	A/E			
B. Building Arrangement	A/E-B&W	A/E	A/E		
C. Excavation & Backfill	A/E		A/E		A/E
D. Substructure	A/E	B&W	A/E		A/E
E. Superstructure	A/E	B&W	A/E		A/E
F. Containment Vessel Liner	A/E		A/E		A/E
G. Containment Pressure & Vacuum Relief System	A/E		A/E		A/E

<u>Item</u>	<u>Preliminary Design</u>		<u>Detail Design</u>		<u>Specifications</u>
	<u>Design</u>	<u>Review</u>	<u>Design</u>	<u>Review</u>	<u>& Procurement</u>
STRUCTURES (Continued)					
4. <u>NSS Containment Structure (Continued)</u>					
H. Containment Vessel Liner Insulation	A/E		A/E		A/E
I. Shielding & Concrete Structure Within Containment	A/E	B&W	A/E	B&W	A/E
J. Containment Recirculation Sump	A/E	B&W	A/E		A/E
K. Miscellaneous Steel					
(1) Refueling Cavity Liner	A/E-B&W	A/E-B&W	A/E	B&W	A/E
(2) Other Steel	A/E	B&W	A/E	B&W	A/E
(3) Missile Shielding	A/E-B&W	A/E	A/E	B&W	A/E
L. Penetrations & Seals	A/E-B&W	B&W	A/E	B&W	A/E
M. Building Service					
(1) Drainage	A/E		A/E		A/E
(2) Elevators	A/E		A/E		A/E
(3) Lighting	A/E		A/E		A/E
(4) Fire Protection	A/E		A/E		A/E
(5) Containment Ventilation System (See Reactor Aux. Emerg. & Safeguard System Section 3-J)	A/E	B&W	A/E		
N. Reactor Plant Crane & Hoists	A/E-B&W	A/E	A/E		A/E
5. <u>Stacks*</u>	A/E	B&W			
*Criteria including volume flow rates, compositions, activity, suggested height, & stack release rates					

<u>Item</u>	<u>Preliminary Design</u>		<u>Detail Design</u>		<u>Specifications</u>
	<u>Design</u>	<u>Review</u>	<u>Design</u>	<u>Review</u>	<u>& Procurement</u>
REACTOR PLANT EQUIPMENT					
1. <u>Reactor Equipment</u>					
A. Reactor Vessel					
(1) Vessel					
(a) Body & Attachments	B&W	A/E	B&W	A/E	B&W
(b) Head & Attachments	B&W		B&W		B&W
(c) Closure Studs, Nuts, Gaskets & Washers	B&W		B&W		B&W
(d) Mirror Insulation	B&W		B&W	A/E	B&W
(e) Mounts for Outside Thermocouples	B&W		B&W		B&W
(f) Attachments for Insulation	B&W		B&W	A/E	B&W
(g) Shipping Frame & Lifting Beam for Vessel Body & Head (Loan Basis)	B&W	A/E	B&W		B&W
(2) Vessel Supports					
(a) Foundation	B&W	A/E	A/E	B&W	A/E
(3) Vessel Internals					
(4) Steam Generator					
(1) Safety & Relief Valves	B&W	A/E	B&W	A/E	B&W

<u>Item</u>	<u>Preliminary Design</u>		<u>Detail Design</u>		<u>Specifications</u>
	<u>Design</u>	<u>Review</u>	<u>Design</u>	<u>Review</u>	<u>& Procurement</u>
REACTOR PLANT EQUIPMENT (Continued)					
(4) Steam Generator (Continued)					
(2) Turbine Bypass System	B&W	A/E	B&W	A/E	A/E
(a) Condenser Dump & Atmos. Dump Valves	B&W		B&W	A/E	B&W
(b) Standard Valves	B&W	A/E	A/E		A/E
(c) Piping	B&W	A/E	A/E		A/E
(d) Hangers & Supports	A/E		A/E		A/E
(e) Insulation	A/E		A/E		A/E
(3) Feedwater Control Valves	B&W	A/E	B&W	A/E	B&W
(4) Feedwater Flow Nozzle	B&W	A/E	B&W	A/E	B&W
(5) Main Steam System	B&W	A/E	A/E	B&W	A/E
(6) Main Feedwater System	B&W	A/E	A/E	B&W	A/E
(7) Auxiliary Feedwater System	B&W	A/E	A/E	B&W	A/E
B. Reactor Controls					
(1) Control Rods	B&W	A/E	B&W		B&W
(2) Drive Mechanisms or System	B&W	A/E	B&W		B&W
(3) CRDM Seismic Supports	B&W	A/E	B&W		B&W
(4) M-G Sets for Rod Power Supply	B&W	A/E	B&W	A/E	B&W

<u>Item</u>	<u>Preliminary Design</u>		<u>Detail Design</u>		<u>Specifications</u>
	<u>Design</u>	<u>Review</u>	<u>Design</u>	<u>Review</u>	<u>& Procurement</u>
REACTOR PLANT EQUIPMENT (Continued)					
1. Reactor Equipment (Continued)					
C. Reactor Shielding					
(1) Biological Shield					
(a) Concrete	A/E-B&W	A/E	A/E	B&W	A/E
(b) Steel (Forms or Liners)	A/E-B&W	A/E	A/E	B&W	A/E
2. <u>Reactor Coolant System</u>					
	B&W	A/E	B&W	A/E	
A. Reactor Coolant Pumps, Drives & Accessories					
	B&W	A/E	B&W	A/E	B&W
(1) Hangers, Supports & Restraints	B&W	A/E	A/E	B&W	A/E
(2) Insulation	B&W	A/E	B&W	A/E	B&W
B. Reactor Coolant Piping					
	B&W	A/E	B&W		B&W
(1) Hangers, Supports & Restraints (If required)	B&W	A/E	A/E	B&W	A/E
(2) Insulation	B&W	A/E	B&W	A/E	B&W
(3) Foundation	A/E-B&W	A/E	A/E		
C. Reactor Coolant Pressurizing & Relief System					
	B&W	A/E	B&W	A/E	B&W
(1) Pressurizer Vessel	B&W	A/E	B&W	A/E	B&W
(2) Electrical Heaters	B&W	A/E	B&W	A/E	B&W

<u>Item</u>	<u>Preliminary Design</u>		<u>Detail Design</u>		<u>Specifications</u>
	<u>Design</u>	<u>Review</u>	<u>Design</u>	<u>Review</u>	<u>& Procurement</u>
REACTOR PLANT EQUIPMENT (Continued)					
C. Reactor Coolant Pressurizing & Relief System (Continued)					
(3) Pressurizer Internal Spray System	B&W	A/E	B&W	A/E	B&W
(4) Quench Tank with Coolers & Internals	B&W	A/E	A/E	B&W	A/E
(5) Safety & Relief Valves	B&W	A/E	B&W	A/E	B&W
(6) Pressurizer Spray Isolation Valves	B&W	A/E	B&W	A/E	B&W
(7) Electromatic Pilot Relief Valve	B&W	A/E	B&W	A/E	B&W
(8) Standard Valves (Manual)	A/E	B&W	A/E	B&W	A/E
(9) Piping, Not including Surge & Spray Lines	B&W	A/E	B&W	A/E	B&W
(10) Pressurizer Surge Piping	B&W	A/E	B&W	A/E	B&W
(11) Pressurizer Spray Piping	B&W	A/E	B&W	A/E	B&W
(12) Hangers & Supports	B&W	A/E	A/E	B&W	A/E
(13) Insulation	B&W	A/E	B&W	A/E	B&W
(14) Vessel Attachments for Insulation	B&W		B&W	A/E	B&W
(15) Foundation	A/E		A/E		A/E

<u>Item</u>	<u>Preliminary Design</u>		<u>Detail Design</u>		<u>Specifications</u>
	<u>Design</u>	<u>Review</u>	<u>Design</u>	<u>Review</u>	<u>& Procurement</u>
REACTOR PLANT EQUIPMENT (Continued)					
3. <u>Reactor Auxiliary Emergency & Safeguard Systems</u>					
A. Makeup & Purification System	B&W	A/E	B&W	A/E	
(1) Tanks					
(a) Makeup Tank & Internals	B&W	A/E	B&W	A/E	B&W
(2) Makeup Pumps	B&W	A/E	B&W	A/E	B&W
(3) Heat Exchangers					
(a) Letdown Coolers	B&W	A/E	B&W	A/E	B&W
(b) Makeup Pump Recirculation Coolers	B&W	A/E	B&W	A/E	B&W
(4) Demineralizers					
(a) Purification Demineralizers w/First Charge Resin	B&W		B&W		B&W
(5) Filters, Orifices, Misc.					
(a) Letdown & MU Pump Recirc. Orifices	B&W	A/E	B&W	A/E	B&W
(b) Purification Filter	B&W	A/E	B&W	A/E	B&W
(c) Feed Flow Filter	B&W	A/E	B&W	A/E	B&W
(d) Flow Measuring Nozzles	B&W	A/E	B&W	A/E	B&W

<u>Item</u>	<u>Preliminary Design</u>		<u>Detail Design</u>		<u>Specifications</u>
	<u>Design</u>	<u>Review</u>	<u>Design</u>	<u>Review</u>	<u>& Procurement</u>
REACTOR PLANT EQUIPMENT (Continued)					
3. <u>Reactor Auxiliary Emergency & Safeguard Systems</u> (Continued)					
A. Makeup & Purification System (Cont'd)					
(6) Special Valves	B&W	A/E	B&W	A/E	B&W
(7) Standard Valves (Manual)	A/E	B&W	A/E		A/E
(8) Piping	A/E	B&W	A/E		A/E
(9) Supports & Hangers	A/E	B&W	A/E		A/E
(10) Insulation	A/E	B&W	A/E		A/E
(11) Foundation	A/E	B&W	A/E		A/E
(12) Heat Tracing	B&W	A/E	A/E		A/E
B. Decay Heat Removal System					
(1) Decay Heat Removal Pumps, Drives & Accessories	B&W	A/E	B&W	A/E	B&W
(2) Decay Heat Coolers	B&W	A/E	B&W	A/E	B&W
(3) Special Valves (including Cavitating Venturis)	B&W	A/E	B&W	A/E	B&W
(4) Standard Valves	A/E	B&W	A/E		A/E
(5) Piping	A/E	B&W	A/E		A/E
(6) Hangers & Supports	A/E	B&W	A/E		A/E

<u>Item</u>	<u>Preliminary Design</u>		<u>Detail Design</u>		<u>Specifications</u>
	<u>Design</u>	<u>Review</u>	<u>Design</u>	<u>Review</u>	<u>& Procurement</u>
REACTOR PLANT EQUIPMENT (Continued)					
3. <u>Reactor Auxiliary Emergency & Safeguard Systems (Continued)</u>					
B. Decay Heat Removal System (Continued)					
(7) Insulation	A/E	B&W	A/E		A/E
(8) Foundation	A/E	B&W	A/E		A/E
C. Core Flooding System	B&W	A/E	B&W	A/E	B&W
(1) Core Flooding Tanks & Nozzles	B&W	A/E	B&W	A/E	B&W
(2) Special Valves	B&W	A/E	B&W	A/E	A/E
(3) Standard Valves	A/E	B&W	A/E	B&W	A/E
(4) Piping	A/E	B&W	A/E		A/E
(5) Hangers & Supports	A/E	B&W	A/E		A/E
(6) Insulation	A/E	B&W	A/E		A/E
(7) Foundation	A/E	B&W	A/E	B&W	A/E
(8) Heat Tracing	B&W	A/E	A/E		A/E
D. Reactor Building Spray System, (Including Post Accident Iodine Removal System, if required)	A/E	B&W	A/E	B&W	A/E

<u>Item</u>	<u>Preliminary Design</u>		<u>Detail Design</u>		<u>Specifications</u>
	<u>Design</u>	<u>Review</u>	<u>Design</u>	<u>Review</u>	<u>& Procurement</u>
REACTOR PLANT EQUIPMENT (Continued)					
3. <u>Reactor Auxiliary Emergency & Safeguard Systems (Continued)</u>					
E. Spent Fuel Cooling & Purification System	A/E	B&W	A/E	B&W	A/E
F. Component Cooling System	A/E	B&W	A/E	B&W	
(1) Component Cooling Pumps, Drives & Accessories	A/E	B&W	A/E		A/E
(2) Component Cooling System Coolers	A/E	B&W	A/E	B&W	A/E
(3) Component Cooling Surge Tank	A/E	B&W	A/E	B&W	A/E
(4) Component Cooling System Filters	A/E	B&W	A/E	B&W	A/E
(5) Special Valves	A/E	B&W	A/E	B&W	A/E
(6) Standard Valves	A/E	B&W	A/E		A/E
(7) Piping	A/E	B&W	A/E		A/E
(8) Supports & Hangers	A/E	B&W	A/E		A/E
(9) Insulation	A/E	B&W	A/E		A/E
(10) Foundation	A/E	B&W	A/E		A/E
G. Chemical Sampling Systems	A/E	B&W	A/E		A/E

<u>Item</u>	<u>Preliminary Design</u>		<u>Detail Design</u>		<u>Specifications</u>
	<u>Design</u>	<u>Review</u>	<u>Design</u>	<u>Review</u>	<u>& Procurement</u>
REACTOR PLANT EQUIPMENT (Continued)					
3. <u>Reactor Auxiliary Emergency & Safeguard Systems (Continued)</u>					
H. Steam Gen. Auxiliary Feedwater System	B&W	A/E	A/E	B&W	
(a) Pump & Drive	B&W	A/E	A/E		A/E
(b) Special Valves	B&W	A/E	B&W		B&W
(c) Standard Valves	B&W	A/E	A/E	B&W	A/E
(d) Piping	A/E		A/E		A/E
(e) Hangers & Supports	A/E		A/E		A/E
(f) Insulation	A/E		A/E		A/E
(g) Foundation	A/E		A/E		A/E
I. Refueling Water System	A/E	B&W	A/E		
(1) Refueling Water Purification Pumps, Drives & Accessories	A/E		A/E		A/E
(2) Borated Water Storage Tanks	A/E	B&W	A/E	B&W	A/E
(3) Refueling Cavity Refueling Water Filter	A/E		A/E		A/E
(4) Refueling Cavity Refueling Water Pumps, Drives & Accessories	A/E		A/E		A/E
(5) Special Valves	A/E		A/E		A/E
(6) Standard Valves	A/E		A/E		A/E

<u>Item</u>	<u>Preliminary Design</u>		<u>Detail Design</u>		<u>Specifications</u>
	<u>Design</u>	<u>Review</u>	<u>Design</u>	<u>Review</u>	<u>& Procurement</u>
REACTOR PLANT EQUIPMENT (Continued)					
3. <u>Reactor Auxiliary Emergency & Safeguard Systems (Continued)</u>					
I. Refueling Water System (Continued)					
(7) Piping	A/E		A/E		A/E
(8) Hangers & Supports	A/E		A/E		A/E
(9) Insulation	A/E		A/E		A/E
(10) Foundation	A/E		A/E		A/E
J. CNSS Ventilation Systems (Includes Containment Purge Supply & Exhaust System, Reactor Service Bldg. Ventilation & Cooling System, & Containment Cleanup System)					
	A/E	B&W	A/E	B&W	A/E
K. Chemical Addition & Boron Recovery System					
(1) Tanks					
(a) Regenerative Caustic Mix Tank	B&W		B&W	A/E	B&W
(b) Reactor Coolant Bleed Evaporator Distillate	B&W		B&W		B&W
(c) Concentrated Boric Acid Storage Tanks	B&W		B&W	A/E	B&W
(d) Boric Acid Mix Tank with Agita- tor	B&W		B&W	A/E	B&W

<u>Item</u>	<u>Preliminary Design</u>		<u>Detail Design</u>		<u>Specifications</u>
	<u>Design</u>	<u>Review</u>	<u>Design</u>	<u>Review</u>	<u>& Procurement</u>
REACTOR PLANT EQUIPMENT (Continued)					
3. <u>Reactor Auxiliary Emergency and</u> <u>Safeguard Systems (Continued)</u>					
K. Chemical Addition & Boron Recovery System (Continued)					
(1) Tanks (Continued)					
(e) Boric Acid Addition Tank	B&W	A/E	B&W	A/E	B&W
(f) Reactor Coolant Bleed Holdup Tanks	B&W	A/E	A/E	B&W	A/E
(2) Pumps, Drives & Accessories					
(a) RC Bleed Evaporator Pumps	B&W	A/E	B&W	A/E	B&W
(b) Regenerative Caustic Pump	B&W	A/E	B&W	A/E	B&W
(c) Boric Acid Pumps	B&W	A/E	B&W	A/E	B&W
(d) RC Distillate Transfer Pumps	B&W	A/E	B&W	A/E	B&W
(e) RC Distillate Test Tank Pumps	B&W	A/E	B&W	A/E	B&W
(3) Evaporators & Demineralizers					
(a) RC Bleed Evaporator Systems	B&W	A/E	B&W	A/E	B&W
(b) Distillate Demineralizers with First Charge Resin	B&W	A/E	B&W	A/E	B&W
(c) Deborating Demineralizers with First Charge Resin	B&W	A/E	B&W	A/E	B&W

<u>Item</u>	<u>Preliminary Design</u>		<u>Detail Design</u>		<u>Specifications</u>
	<u>Design</u>	<u>Review</u>	<u>Design</u>	<u>Review</u>	<u>& Procurement</u>
REACTOR PLANT EQUIPMENT (Continued)					
3. <u>Reactor Auxiliary Emergency & Safeguard Systems (Continued)</u>					
K. Chemical Addition & Boron Recovery System (Continued)					
(3) Evaporators & Demineralizers (Continued)					
(d) RC Bleed Evaporator Demineralizer with First Charge Resin	B&W	A/E	B&W	A/E	B&W
(4) Filters, Orifices, Miscellaneous					
(a) Caustic Flow Orifice	B&W	A/E	B&W	A/E	B&W
(b) RC Distillate Transfer Flow Orifice	B&W	A/E	B&W		B&W
(c) Boric Acid Flow Orifice	B&W	A/E	B&W		B&W
(d) RC Degasifier	B&W	A/E	B&W		B&W
(e) Boric Acid Filter	B&W	A/E	B&W		B&W
(5) Special Valves	B&W	A/E	B&W	A/E	B&W
(6) Standard Valves	A/E	B&W	A/E		A/E
(7) Piping	A/E	B&W	A/E		A/E
(8) Supports & Hangers	A/E	B&W	A/E		A/E

<u>Item</u>	<u>Preliminary Design</u> <u>Design Review</u>		<u>Detail Design</u> <u>Design Review</u>	<u>Specifications</u> <u>& Procurement</u>
REACTOR PLANT EQUIPMENT (Continued)				
3. <u>Reactor Auxiliary Emergency & Safeguard Systems (Continued)</u>				
K. Chemical Addition & Boron Recovery System (Continued)				
(9) Insulation	A/E	B&W	A/E	A/E
(10) Foundation	A/E	B&W	A/E	A/E
(11) Heat Tracing	B&W	A/E	A/E	A/E
L. Essential Raw Cooling Water System	A/E	B&W	A/E	A/E
M. Reactor Service Building Non-Accident Iodine Removal System (If Required)	A/E		A/E	A/E
N. Containment Hydrogen Removal System	A/E	B&W	A/E	A/E
O. Instrument Air System	A/E	B&W	A/E	A/E
4. <u>Radioactive Waste Treatment & Disposal System</u>				
A. Liquid Waste System	A/E	B&W	A/E	A/E
B. Gaseous Waste System	A/E	B&W	A/E	A/E
C. Solid Waste System	A/E	B&W	A/E	A/E
D. Reactor Coolant Waste System	A/E	B&W	A/E B&W	A/E

<u>Item</u>	<u>Preliminary Design</u>		<u>Detail Design</u>		<u>Specifications</u>
	<u>Design</u>	<u>Review</u>	<u>Design</u>	<u>Review</u>	<u>& Procurement</u>
REACTOR PLANT EQUIPMENT (Continued)					
5. <u>Nuclear Fuel-Handling & Storage Equipment*</u>					
A. Cranes & Hoisting Equipment					
(1) Stud Tensioner Sling	B&W	A/E	B&W	A/E	B&W
(2) Stud & Stud Tensioner Hoist	B&W	A/E	B&W	A/E	B&W
(3) Stud Nut Handling Fixture	B&W	A/E	B&W	A/E	B&W
(4) Stud & Nut Handling Sling	B&W	A/E	B&W	A/E	B&W
(5) Closure Head Service Structure	B&W	A/E	B&W	A/E	B&W
(6) Closure Head & Internals Handling Fixture	B&W	A/E	B&W	A/E	B&W
(7) New Fuel Transfer Container with Adapter Plate & Sling	B&W	A/E	B&W	A/E	B&W
(8) Main Fuel Handling Bridge with Track & Stops & Including Fuel & Control Rod Handling Mechanism, Grapples, Trolley & Hoist Controls	B&W	A/E	B&W	A/E	B&W
(9) Auxiliary Fuel Handling Bridge Including Fuel Handling Mechanism, Grapple, Bridge, Trolley & Hoist Controls	A/E	B&W	A/E	B&W	A/E

*All special tools and servicing equipment for complete vessel servicing and refueling requirements shall include but not be limited to the equipment listed.

<u>Item</u>	<u>Preliminary Design</u>		<u>Detail Design</u>		<u>Specifications</u>
	<u>Design</u>	<u>Review</u>	<u>Design</u>	<u>Review</u>	<u>& Procurement</u>
REACTOR PLANT EQUIPMENT (Continued)					
5. <u>Nuclear Fuel-Handling & Storage Equipment</u> (Continued)					
A. Cranes & Hoisting Equipment (Continued)					
(10) Fuel Storage Handling Bridge with Track & Stops and Including Control Rod & Fuel Handling Mechanisms, Grapple, Bridge, Trolley & Hoist Controls	B&W	A/E	B&W	A/E	B&W
(11) Spent Fuel Shipping Cask Crane	B&W	A/E	A/E	B&W	A/E
B. Special Tools, Handling Equipment, & Fixtures					
(1) Stud Tensioner with Elongation Gage	B&W	A/E	B&W	A/E	B&W
(2) Stud Nut Wrench	B&W	A/E	B&W	A/E	B&W
(3) Stud Hole Chase Tool	B&W	A/E	B&W	A/E	B&W
(4) Stud Hole Seal Plug	B&W	A/E	B&W	A/E	B&W
(5) Stud Impact Wrench with Adapter	B&W	A/E	B&W	A/E	B&W
(6) Stud Handling Tool with Adapter	B&W	A/E	B&W	A/E	B&W
(7) Stud Support Spacer	B&W	A/E	B&W	A/E	B&W
(8) Stud Storage Rack	B&W	A/E	B&W	A/E	B&W
(9) Elongation Rod	B&W	A/E	B&W	A/E	B&W
(10) Elongation Gage	B&W	A/E	B&W	A/E	B&W

<u>Item</u>	<u>Preliminary Design</u>		<u>Detail Design</u>		<u>Specifications</u> <u>& Procurement</u>
	<u>Design</u>	<u>Review</u>	<u>Design</u>	<u>Review</u>	
REACTOR PLANT EQUIPMENT (Continued)					
5. <u>Nuclear Fuel-Handling & Storage Equipment</u> (Continued)					
B. Special Tools, Handling Equipment, & Fixtures (Continued)					
(11) Alignment Stud	B&W	A/E	B&W	A/E	B&W
(12) Canal Seal Plate Assembly	B&W	A/E	B&W	A/E	B&W
(13) Shield Plate Assembly	B&W	A/E	B&W	A/E	B&W
(14) Closure Head Storage Stand	B&W	A/E	A/E	A/E	A/E
(15) Internals Storage Stand	B&W	A/E	A/E	A/E	A/E
(16) Internals Handling Extension	B&W	A/E	B&W	A/E	B&W
(17) Internals Handling Adapter	B&W	A/E	B&W	A/E	B&W
(18) Internals Indexing Fixture	B&W	A/E	B&W	A/E	B&W
(19) In-Core Instrument Tube Plug Tool	B&W	A/E	B&W	A/E	B&W
(20) New Fuel Handling Tool	B&W	A/E	B&W	A/E	B&W
(21) Rod Assembly Handling Tool	B&W	A/E	B&W	A/E	B&W
(22) Rod Assembly Handling Rack & Containers	B&W	A/E	B&W	A/E	B&W
(23) Long Handled Tools (Handles, Extensions, Hooks, Guide, & Wrenches)	B&W	A/E	B&W	A/E	B&W

<u>Item</u>	<u>Preliminary Design</u>		<u>Detail Design</u>		<u>Specifications</u>
	<u>Design</u>	<u>Review</u>	<u>Design</u>	<u>Review</u>	<u>& Procurement</u>
REACTOR PLANT EQUIPMENT (Continued)					
5. <u>Nuclear Fuel-Handling & Storage Equipment</u> (Continued)					
B. Special Tools, Handling Equipment, & Fixtures (Continued)					
(24) Fuel Transfer Mechanism Including Track & Track Supports, Transfer Carriage, Fuel Basket, Carriage Actuating System, Fuel Basket Rotating System, Control Panel, Etc.	B&W	A/E	B&W	A/E	B&W
(25) Fuel Transfer Tube Including Rails, Tube Valve, Cover Flange	B&W	A/E	B&W	A/E	B&W
(26) Boroscope	B&W	A/E	B&W	A/E	B&W
(27) Submarine Lights - Cavity, Canal, & Pit	A/E	A/E	A/E		A/E
(28) New & Spent Fuel Storage Racks	A/E	B&W	A/E		A/E
(29) Television, Other Viewing, & Special Lighting	B&W	A/E	B&W	A/E	B&W
(30) Failed Fuel Assembly Detecting Equipment	B&W	A/E	B&W	A/E	B&W
(31) Miscellaneous Tools & Handling Devices	A/E	B&W	A/E		A/E
(32) In-Core Detector Disposal Reel	B&W	A/E	A/E	B&W	A/E

<u>Item</u>	<u>Preliminary Design</u>		<u>Detail Design</u>		<u>Specifications</u>
	<u>Design</u>	<u>Review</u>	<u>Design</u>	<u>Review</u>	<u>& Procurement</u>
REACTOR PLANT EQUIPMENT (Continued)					
5. <u>Nuclear Fuel-Handling & Storage Equipment</u> (Continued)					
C. Spent Fuel Shipping Cask	B&W	A/E	A/E	B&W	A/E
6. <u>Nuclear Steam Supply System</u> <u>Instrumentation & Control</u>					
A. Reactor Coolant System	B&W		B&W		B&W
B&W will design & supply all process control (except motor control switches) and instrumentation for the RCS. The customer is responsible for the development of a RC pump monitoring system.					
B&W will design and supply all engineered features actuation (ESFAS) safety controls and instrumentation (SRCI) for the RCS.					
B. Secondary System	B&W	A/E	B&W	A/E	B&W/A/E
B&W will design and supply the Integrated Control System (ICS) and the SRCI equipment for the steam plant. The customer is responsible for the remainder of this system, i.e.					

<u>Item</u>	<u>Preliminary Design</u>		<u>Detail Design</u>		<u>Specifications</u>
	<u>Design</u>	<u>Review</u>	<u>Design</u>	<u>Review</u>	<u>& Procurement</u>
REACTOR PLANT EQUIPMENT (Continued)					
6. <u>Nuclear Steam Supply System</u>					
<u>Instrumentation & Control (Continued)</u>					
C. Process Instrumentation & Control					
(1) Reactor Coolant System	B&W		B&W		B&W
(2) Auxiliary, Emergency & Safeguard Systems					
(a) Makeup & Purification System	B&W	A/E	B&W	A/E	B&W
(b) Decay Heat Removal	B&W	A/E	B&W	A/E	B&W
(c) Core Flooding System	B&W	A/E	B&W	A/E	B&W
(d) Reactor Building Spray System	A/E	B&W	A/E	B&W	A/E
(e) Spent Fuel Cooling & Purification System	A/E	B&W	A/E		A/E
(f) Component Cooling System	A/E	B&W	A/E	B&W	B&W
(g) Chemical Sampling System	A/E	B&W	A/E	B&W	A/E
(h) Refueling Water System*	A/E	B&W	A/E	B&W	A/E
(i) Essential Raw Cooling Water System	A/E	B&W	A/E	B&W	A/E
(j) Containment Ventilation System	A/E	B&W	A/E	B&W	A/E
(k) Nonaccident Iodine Removal	A/E		A/E		A/E
*Can be combined with item 6.C(2)(e) if design permits.					

APPENDIX C

C O L U M N N U M B E R
 11111111112222222222223333333333334444444444445555555555556666666666667777777777776
 12345678901234567890123456789012345678901234567890123456789012345678901234567890

2250.0 CNSS REACTOR VESSEL UNRECOVERABLE PRESSURE LOSSES

THIS IS A PRELIMINARY CALCULATION OF THE PRESSURE LOSSES

PUMP DISCHARGE IS REFERENCE ELEVATION

PUMP DISCHARGE LOSS

1	2	3	1	67.5+6			
27.271			0.0	615.5	122.719	-2.00	615.5
	3	27.271		122.719			615.5

SG LOSS (FROM PUMP DISCHARGE PLENUM TO ANNULUS BELOW SG) E=163(RMS)

2	3	3	4	67.5+6			
122.719			-2.0	615.5	145.532	-24.5	572.3
	7	122.719		21.996	54.989	1.0	615.5
	3	54.989		20.156			615.5
	2	2.9	-4	0.425	20.0	20.156	593.3
	3	20.156		145.532			572.3

MIXER LOSS

3	4	3	4	67.5+6			
145.532			-24.5	572.3	145.532	-32.0	572.3
	3	145.532		36.383			572.3
	3	36.383		145.532			572.3
	4	145.532		36.383	90.0		572.3
	4	36.383		145.532	90.0		572.3

LOSS FROM MIXER EXIT TO CORE SUPPORT CYLINDER
 ASSUME AREA AT CS CYLINDER IS FOR H=22.0 INCH AND D=106 INCH

4	5	3	2	67.5+6			
145.532			-32.0	572.3	50.876	-39.0	572.3
	3	145.532		50.876			572.3
	4	145.532		50.876	90.0		572.3

FLCW THRU CORE SUPPORT CYLINDER HAS TWO PATHS - ONE THRU CYLINDER
 AND ONE PATH UNDER THE CYLINDER. ASSUME 20 PERCENT UNDER CYLINDER

PATH THRU CYLINDER

5	6	3	1	54.0+6			
40.701			-39.0	572.3	40.302	-39.0	572.3
	7	44.506		17.104	40.302	0.5	572.3

6	7	2	1	54.0+6			
	4	17.104		40.302	90.0		572.3

PATH UNDER CYLINDER (FLOW EXPANDS TO CYLINDER D=66 AND H=9)

11111111112222222222333333333344444444445555555555666666666677777777778
 C O L U M N N U M B E R
 12345678901234567890123456789012345678901234567890123456789012345678901234567890

C
 C
 C LAST CARD
 99 13 1

***** END OF INPUT LISTING *****

OVERLAY

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3ADDR
PUMP INERTIA STUDY
TASK 2 CNSS RC PUMP INERTIA REQUIREMENTS
1 NODE MODEL (BENCH-MARKED ON STD-205)
INPUT
      CON(RFRICT,RFORM,LA)
L/A VALUE IS LENGTH-WEIGHTED
      CON(B100,N100,WTOT0)
      CON(HRATC,RHOS,DT,TFIN)
      CON(REO,DAM)
      PAR(INERT)

PUMP CURVES
FF=REYNOLDS NUMBER VS. FRICTION FACTOR
      FF      CFN(10.)
HAN=(HEAD/HEAD-RATED)/SPEED E+2 VS. FLOW/SPEED
      HAN      CFN(25.)
BAN=(TORQ/TORQ-RATED)/SPEED E+2 VS. FLOW/SPEED
      BAN      CFN(25.)
      DWTOT    4633.1 * (DPP - DPFR - DPFORM) / LA
      WTOT     INT(DWTOT,WTOT0)
      FFC      FUN(FF,REO)
      RE       WTOT * DAM
      FFT      FUN(FF,RE)
FM=FRICTION FACTOR MODIFIER
      FM      FFT/FFC
      DPFORM   RFORM * WTOT * WTOT
      DPFR     RFRICT * WTOT * WTOT * FM
      NU       WTOT / WTOT0
      ALPHA    N/N100
      NA       NU/ALPHA
      HA2      FUN(HAN,NA)
      HRAT     HA2 * (ALPHA*ALPHA)
      H        HRAT * HRATC
      DPP      H * RHOS / 144.0
      DOMECA   (BEL - BHYD) * 32.2 / INERT
      OMEGA0   N100 * 2. * 3.14159 / 60.
      OMEGA    INT(DOMECA,OMEGA0)
      N        OMEGA * 60. / (2.0 * 3.14159)
      BA2      FUN(BAN,NA)
      BRAT     BA2 * ALPHA * ALPHA
      BHYD     BRAT * 0.100
      NSTART   FSW(T-2.0,FALSE,TRUE,TRUE)
      BEL      LSW(NSTART,0.0,B100)
      TC       LSW(NSTART,T-2.0,0.0)
      FIN(T,TFIN)
      OUT(T,TC,NU,ALPHA,FM)
      END

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APPENDIX D

OVERLAY

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$ADDR
PUMP INERTIA STUDY
TASK 2  CNSS RC PUMP INERTIA REQUIREMENTS
1 NODE MODEL (BENCH-MARKED ON STD-205)
INPUT
      CON(RFRICT,RFORM,LA)
L/A VALUE IS LENGTH-WEIGHTED
      CON(B100,N100,WTOT0)
      CON(HRAT0,RHOS,DT,TFIN)
      CON(RE0,DAM)
      PAR(INERT)

PUMP CURVES
FF=REYNOLDS NUMBER VS. FRICTION FACTOR
      FF      CFN(10.)
HAN=(HEAD/HEAD-RATED)/SPEED E+2  VS.  FLOW/SPEED
      HAN      CFN(25.)
BAN=(TORQ/TORQ-RATED)/SPEED E+2  VS.  FLOW/SPEED
      BAN      CFN(25.)
      DWTOT    4633.1 * (DPP - DPFR - DPFORM) / LA
      WTOT     INT(DWTOT,WTOT0)
      FFC      FUN(FF,RE0)
      RE       WTOT * DAM
      FFT      FUN(FF,RE)
FM=FRICTION FACTOR MODIFIER
      FM       FFT/FF0
      DPFORM   RFORM * WTOT * WTOT
      DPFR     RFRICT * WTOT * WTOT * FM
      NU       WTOT / WTOT0
      ALPHA    N/N100
      NA       NU/ALPHA
      HA2      FUN(HAN,NA)
      HRAT     HA2*(ALPHA*ALPHA)
      H        HRAT * HRAT0
      DPP      H * RHOS / 144.0
      DOMEGA   (DEL - BHYD) * 32.2 / INERT
      OMEGA0   N100 * 2. * 3.14159 / 60.
      OMEGA    INT(DOMEGA,OMEGA0)
      N        OMEGA * 60. / (2.0 * 3.14159)
      BA2      FUN(BAN,NA)
      BRAT     BA2 * ALPHA * ALPHA
      BHYD     BRAT * 9100
      NSTART   FSW(T-2.0,FALSE,TRUE,TRUE)
      DEL      LSW(NSTART,0.0,B100)
      TC       LSW(NSTART,T-2.0,0.0)
      FIN(T,TFIN)
      OUT(T,TC, NU,ALPHA,FM)
      END

```