SUPPLEMENT NO. 1

TO THE

SAFETY EVALUATION

BY THE

DIRECTORATE OF LICENSING

U. S. ATOMIC ENERGY COMMISSION

IN THE MATTER OF

TENNESSEE VALLEY AUTHORITY

BROWNS FERRY NUCLEAR PLANT

UNITS 1, 2 AND 3

DOCKET NOS. 50-259, 260 AND 296

Issuance Date: December 21, 1972

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APPENDICES

ABBREVIATIONS

a-c	alternating current
ACI	American Concrete Institute
ACRS	Advisory Committee on Reactor Safeguards
ADS	Automatic Depressurization System
AEC	United States Atomic Energy Commission
ANS	American Nuclear Society
ANSI	American National Standard Institute
ASME	American Society of Mechanical Engineers
ASTM	American Society for Testing and Materials
AWS	American Welding Society
BNL	Brookhaven National Laboratory
Btu/hr-ft^2	British thermal units per hour per square foot
Btu/lb	British thermal units per pound
BWR	Boiling Water Reactor
CAD	Containment Atmosphere Dilution
cfm	cubic feet per minute
cfs	cubic feet per second
Ci/sec	Curies per second
CSS	Core Spray System
DBA	Design Basis Accident
DBE	Design Basis Earthquake
d-c	direct current
ECCS	Emergency Core Cooling System

ESF	Engineered Safety Feature
EECW	Emergency Equipment Cooling Water System
°F	degrees Fahrenheit
ft^2	square feet
ft^3	cubic feet
FSAR	Final Safety Analysis Report
g	acceleration, 32.2 feet per second per second
GDC	AEC General Design Criteria for Nuclear Power Plant Construction Permits
GE	General Electric Company
GPM	gallons per minute
HEPA	High Efficiency, Particulate, Air
HPCIS	High Pressure Cooling Injection System
IEEE	Institute of Electrical and Electronics Engineers
in	inch
$\Delta k/k$	reactivity change
kV	kilovolts
kW/ft	kilowatts per foot
lb	pound
lb/hr	pounds per hour
LOCA	Loss-of-Coolant Accident
LPCIS	Low Pressure Coolant Injection System
LPCS	Low Pressure Core Spray
MCHFR	Minimum Critical Heat Flux Ratio
m	meters
mph	miles per hour

Mrem	Millirem
MSL	mean sea level
MSLIV	Main Steam Line Isolation Valve
MWD/ton	megawatt days per ton
MWe	megawatts electrical
MWt	megawatts thermal
NOAA	National Oceanic and Atmospheric Administration
NPSH	net positive suction head
OBE	Operating Basis Earthquake
PHS	Public Health Service
PMF	probable maximum flood
PORC	Plant Operations Review Committee
PSAR	Preliminary Safety Analysis Report
psi	pounds per square inch
psid	pounds per square inch differential
psig	pounds per square inch gauge
QA	Quality Assurance
QC	Quality Control
Rem	Roentgen equivalent man
SRB	Safety Review Board
TVA	Tennessee Valley Authority
R&D	Research and Development
RCICS	Reactor Core Isolation Cooling System
RCPB	Reactor Coolant Pressure Boundary
RHRS	Residual Heat Removal System
RSCS	Rod Sequence Control System

sec	second
SGTS	Standby Gas Treatment System
χ/Q	atmospheric diffusion factor (sec/m^3)
w/o	weight percent
10 CFR	AEC, Title 10, Code of Federal Regulations
Part 2	AEC Rules of Practice
Part 20	AEC Standards for Protection Against Radiation
Part 50	AEC Licensing of Production and Utilization Facilities
Part 100	AEC Reactor Site Criteria

1.0 INTRODUCTION

The Safety Evaluation Report of the Tennessee Valley Authority, Browns Ferry Nuclear Plants Units 1, 2 and 3 (SER) was issued on June 26, 1972. The SER identified unresolved issues requiring additional submittals by the applicant with subsequent staff evaluation. Since the date of issuance, there have been meetings with the applicant, additional amendments to the FSAR and meetings with the ACRS resulting in an ACRS letter on September 21, 1972.

The purpose of this supplement is to update the SER based on the staff evaluation work performed on this docket since June 26, 1972.

The sections of the SER which have been affected by the e[illegible]ion have been rewritten completely to make this document self supporting. The rewritten sections are identified by their SER section numbers.

Because of additional delays, the applicant requested and was granted extensions of CPPR-29 to December 1, 1973, CPPR-30 to August 1, 1974, and CPPR-48 to February 1, 1975.

In addition, the Supplement contains an updated chronology as Appendix A, the report of the ACRS as Appendix B, and an Errata to the SER as Appendix C. The list of abbreviations is repeated for ease of reference.

2.0 REVISED SECTIONS OF SAFETY EVALUATION REPORT ISSUED JUNE 26, 1972

2.3 Hydrology

The site is on the north side of the Tennessee River Wheeler Reservoir about 19 miles upstream of Wheeler Dam, 55 miles downstream of Guntersville Dam, and about 30 miles due west of Huntsville, Alabama. Normal reservoir level is elevation 556 ft. MSL, average ground elevation at the site is 580 ft. MSL, and plant grade adjacent to the reservoir is elevation 565 ft. MSL. Cooling water for the three units is supplied from a river bank intake structure. A single trifurcated conduit supplies water for each unit. The intake structure pumps are mounted outdoors above plant grade, and will draw water from the intake structure sump which has a bottom elevation of 518 ft. MSL and an excavated 25 foot wide approach channel at elevation 523 ft. MSL to deep water in Wheeler Reservoir. Cooling water is discharged into the reservoir via three corrugated metal pipes, each of which is perforated for diffusion in an existing deep channel of the Tennessee River. The pipes extend 1010, 1610, and 2210 feet from the shoreline, respectively, with the last 600 feet of each used for diffusion.

The applicant has evaluated flooding from three sources, the Tennessee River, a local tributary west of the site, and from plant drainage. Each potential flooding source is discussed separately below:

a. Tennessee River

Historical streamflow recorded 40 miles upstream since 1937 indicates the maximum Tennessee streamflow after TVA dam construction occurred in February 1957 and was 293,000 cfs. The minimum recorded streamflow was 400 cfs in July 1966 and is attributed primarily to upstream regulation. The maximum flood of record in the region occurred in 1897 prior to construction of TVA dams with an estimated local maximum runoff rate of about 470,000 cfs.

TVA's evaluation initially assumed a hypothetical Tennessee River Flood, which we consider inadequate, being only about half as severe as the probable maximum flood (PMF). Subsequently TVA performed an evaluation for a PMF based on the Weather Bureau's latest hydrometeorological estimates of probable maximum precipitation for the region, and determined that the peak runoff rate at the site would be about 1,200,000 cfs resulting in a river level elevation of about 572.5 ft. MSL. This PMF determination included an extensive study of the runoff capability of the upstream 27,130 square mile drainage area and was greatly complicated by the necessity for determining the effects of more than 22 major TVA and 6 privately owned reservoirs. TVA found that the reservoir and outlet capacity of the four Tennessee River dams immediately upstream of the site would be insufficient to pass a PMF and, therefore, included the effects of their potential failure in the PMF es[illegible]e. TV[illegible] also assumed a sustained wind speed of [illegible] mph [illegible]dent wi[illegible] [illegible]aximum PMF river level, and has estima[illegible] [illegible] ponding wind wave runup level could reach ele[illegible] [illegible] 574 ft. MSL. TVA assumed that the most likely month for a PMF was in March and used the mean March wind speed (14 mph) as the coincident wind. However, we have independently estimated the wind wave effects, using the guidance provided by the Corps of Engineers, for the plant area and accordingly estimate that a reasonably severe windstorm producing 45 mph sustained wind speeds could occur coincidentally with a PMF and produce a maximum wind wave runup level as high as an elevation of 578 ft. MSL.

The applicant has now provided, described in Amendment 40 to the FSAR, flood protection up to 578 ft. MSL so that the plant may be placed and maintained in a safe shutdown condition for all feasible combinations of wind and flood up to the hypothesized PMF condition. Specifically, the Reactor Building, Diesel-Generator Buildings, Radwaste Building and the Residual Heat Removal Service Water (RHRSW) pumps on the intake structure are protected against flotation and kept dry and operational up to 578 ft. MSL.

All protection is permanently in place. At a reservoir elevation of 558 MSL, the only other potential openings remaining into Class I areas, the outer door to the waste packaging area of the Radwaste Building and the outer set of double doors of the large equipment lock to the Reactor Building and its sliding gate will be closed if open and access denied to these areas.

The plant can continue normal operation, during flood conditions, until the water level reaches elevation 565 ft. MSL which is the elevation of the pumping station deck. When this elevation is reached, as indicated by redundant water level switches read out in the control room or by direct or TV visual observation, an orderly shutdown of the plant will be initiated. The circulating water pumps will be utilized as long as practicable. The protection provided to the RHRSW pumps, however, provides the capability for decay heat removal up to elevation 578 ft. MSL.

Long periods of advance warning of flood or wind conditions are not required and we now conclude that the plant can achieve and maintain a safe, cold shutdown under all potential flood conditions up to and including those of probable maximum severity. The additional physical protection will be installed prior to Unit 1 exceeding 1% of full power.

b. Local Tributary

During the construction of the plant, a local tributary was diverted into Wheeler Reservoir west of the site. The applicant was requested to provide an analysis of the capability of the tributary to flood safety related plant facilities as a result of a local PMF. TVA found the existing diversion channel and bridge incapable of passing floods up to the severity of local PMF (with a maximum runoff rate of about 14,000 cubic feet per second) without inundating the plant. Consequently, TVA has proposed modifying the diversion channel to safely pass a local PMF and in Amendment 33, has provided details of the relocated diversion channel including typical sections, plans, and water surface profiles and minimum grade levels between the channel and the plant. The channel will pass the maximum runoff rate from a local PMF of 14,000 cfs with maximum water surface elevations below the ground, the dike, and the road which protect the plant and cooling tower areas from flooding.

We conclude that the design bases for the diversion channel are adequate to safely pass a local PMF without affecting safety related plant facilities.

c. Plant Drainage

The applicant has also evaluated the flooding potential from surface drainage and the extensive roof surface area of the facility. The applicant determined that the roof and its drainage are adequate for severe storms, but indicated that modifications would be required to three service building doors and their seals to prevent flooding of the radwaste building. This requirement was eliminated by the applicant in Amendment 40 wherein the Radwaste Building was protected against the PMF. This protection includes water tight entrances from the Service and Turbine Buildings, thereby negating the need for Service Building protection. We conclude that safety related facilities are adequately protected against surface and roof drainage.

d. Ground Water

Ground water at the site is derived from local precipitation, part of which percolates into the residuum. Deep regional ground water movement is prevented from reaching the site by local anticlinal and synclinal bedrock structures. All local ground water, as reported by the applicant, flows directly into Wheeler Reservoir. The 32 public ground water supplies within 20 miles of the site are not expected to be affected by plant operation. Since the onsite liquid radioactive waste storage is contained entirely within the radwaste building concrete structure which will be watertight and capable of the requirements of a Class I (seismic) structure, we conclude that there is little likelihood of accidental release of liquid radwastes to the ground. The eight private wells within one mile of the site have been surveyed and the applicant has stated that special local monitoring will be carried out in the event of any unusual release, even though there is also little likelihood of their contamination.

e. Water Supply

Cooling water is to be taken directly from Wheeler Reservoir. Adequate water supply is available for normal operation. However, we considered the limiting water supply condition that would occur following the effects of an assumed failure of the downstream Wheeler Dam. The applicant has estimated under these assumptions that the volume of water available in a large natural depression in the river bottom, coincident with minimum runoff, would still provide an adequate source of cooling water for safe shutdown cooling water requirements (45 cfs) for all three units. We conclude that adequate shutdown cooling water is available.

The Tennessee River from 12 miles upstream of the site to 49 miles downstream serves five public water supplies. Four intakes are downstream of the site, three of which are owned, operated, and controlled by TVA. TVA has stated that it will monitor both public and private supplies periodically. We concur with the applicant that there is little likelihood of contaminating public or private surface or ground water supplies based on conditions of storage and control of radioactive liquid effluent discussed in Section 8.2 herein, and that a suitable monitoring program is (as indicated by the applicant) a desirable safeguard for warning potable water users in the unlikely event of a spill.

3.4 Reactivity Control

Reactor power can be controlled by either movement of control rods or variation in reactor coolant recirculation system flow rate. A standby liquid control syster is also provided as a backup shutdown system.

There are 185 control rods which are used to bring the reactor through the full range of power (from shutdown to full power operation), to shape the reactor power distribution, and to compensate for changes in reactivity resulting from fuel burnup. Each control rod drive has separate control and rapid insertion (scram) devices. A common hydraulic pressure source for normal operation and a common dump volume for scram operation are used for the drives. On the basis of our review of the drive system design and the supporting evidence accumulated from operation of similar systems in other General Electric reactors, we conclude that the installed system will meet the functional performance requirements for each reactor in a safe manner.

During operation at power levels between zero to 10% of the rated power, control rod reactivity worths are limited by the rod worth minimizer (RWM), a device which utilizes a computer to restrict control rod patterns such that the total worth of any insequence rod that can be moved will be no more than 1% delta k. For reactor power levels in excess of 10% of the rated power, RWM operability is not required. (See Section 9.4)

The AEC has been reevaluating the modeling and the consequences of the postulated Control Rod Drop Accident (See Section 9.4). We have concluded that modifications are required which would provide means to augment the RWM so that the probability of occurrence of the postulated accident is negligibly low and/or that the consequences are consistent with the guidelines of 10 CFR Part 100.

Accordingly, the applicant has proposed, in Amendments 43 and 44, the installation of a Rod Sequence Control System (RSCS) as a backup to the RWM. The RSCS independently restricts the selection and withdrawal of control rods up through 50% rod density (checkerboard pattern, 50% of the control rods full out and 50% full in).

At power levels exceeding 50% rod density, the applicant states that the consequences of a control-rod-drop accident indicate that the peak fuel enthalpy is below the threshold value (280 cal/gm) assumed to cause rapid fuel dispersal and damaging pressure pulses to the reactor core and that the radiological doses at the site boundary from the estimated fuel cladding failures are well within the guidelines of 10 CFR Part 100.

The RSCS is a hard wired system which is electrically independent from the RWM and utilizes inputs from the full-in and full-out switches in the rod position indicator probes and rod sequence selector switches. It wires the rod select relays into groups that control four sequence patterns. A relay either inhibits or permits movement of all of the rods assigned to a sequence pattern. Our evaluation of the RSCS is discussed in Section 9.4 Control Rod Drop Accident.

A control-rod-ejection accident is precluded by a control rod housing support structure located below the reactor pressure vessel, similar to that installed on the other large General Electric reactors. This structure limits the distance that a ruptured control rod drive housing could be displaced. The applicant concluded, and we agree, that the control rod displacement would be so small in this event that any resulting nuclear transient could not be sufficient to cause fuel rod failure.

Rapid control rod withdrawal is prevented by the control rod velocity limiter which limits the free fall of a rod to 5 ft/sec but does not retard scram action.

Reactor power can also be controlled through changes in the primary coolant recirculation flow rate. The recirculation flow control system is the normal control method used to adjust reactor power level to station load demand whenever the reactor is operating between approximately 60% to 100% rated power. The recirculation flow control system is designed to allow either manual or automatic control of reactor power. This method of reactor power control has been demonstrated to be acceptable in the Dresden Units 2 and 3, Monticello and Millstone I facilities.

The standby liquid control system is designed to bring the reactor to a cold shutdown condition from the full power steady-state operating condition at any time in core life, independent of the control rod system capabilities. The injection rate of the system is adequate to compensate for the effects of xenon burnup and decay.

Each of the foregoing design features with the exception of the RSCS is similar to the corresponding features provided in plants we have previously reviewed. On the basis of our previous review of similar designs and of satisfactory operating experience with similar systems in other operating BWRs, we conclude that the mechanical, thermal and hydraulic, and reactivity control features of each reactor is acceptable.

3.5.3 Vibration Control

The applicant has planned for vibration tests of reactor internals in Units 1, 2 and 3 during plant start-up. During these tests, for Unit 1, the displacement of the shroud and a jet pump relative to the reactor pressure vessel wall will be monitored the separator motions will be recorded with accelerometers, strain levels will be recorded from a jet pump riser brace and the guide tube. These measurements will be provided for Units 2 and 3 as appropriate. The data obtained should be sufficient to verify that the steady state and cyclic stresses in the components, as determined by analyses, are within the acceptable design limits set forth in the design specifications and ASME Code Section III requirements. The applicant has stated that the criterion used in the Browns Ferry internals analysis is to limit the alternating peak stress intensity, including all stress concentration factors to a value of $\pm$ 10,000 psi which represents an additional factor of safety of 2.5 below the value permitted by ASME Codes.

The applicant has proposed a vibration testing program for Units 1, 2 and 3 such that Unit 1 can serve as the prototype for Units 2 and 3. The tests proposed for Units 2 and 3 are confirmatory tests. The program meets the intent of AEC Safety Guide 20, "Vibration Measurements on Reactor Internals."

The program for Unit 1 consists of three phases:

1) a cold flow test monitored with installed vibration monitoring instrumentation; the tests have been described by the applicant in

Amendment 44.

2) an in place inspection of reactor vessel internals; the inspection has been described by the applicant in Amendment 44.

3) a hot flow test monitored with installed vibration monitoring instrumentation. A predictive vibration analysis will be submitted prior to conducting the test program to provide a basis for evaluating the results.

In order to qualify Browns Ferry Unit 1 as the prototype for reactor internals vibration testing:

1) the responses measured from the hot flow test should be compatible in magnitude and characteristics to the responses measured from the cold flow test,

2) the in place inspection following cold flow testing demonstrates no component degradation,

3) the analytical prediction of the response at sensor locations compare favorably with the measured responses, and

4) the forcing functions and the analytical methods providing predictions are confirmed by the measured response.

In the event that Unit 1 is accepted as a prototype, Units 2 and 3 may perform instrumented confirmatory testing without subsequent inspection of the reactor internals provided that a comparison of the measured responses confirms the substantial similarity in vibration behavior between the tested internals and those of Unit 1. We conclude that the preoperational vibration test program for the Browns Ferry Plant is acceptable subject to receiving the predictive vibration analysis.

5.2.2 Missile and Pipe Whip Protection

The applicant has considered the effect of missiles ranging in size from nuts and bolts to valve bonnets, and concludes that no missile would have sufficient energy to penetrate the drywell wall. In addition, where possible, components are arranged so that the direction of flight of potential missiles is away from the containment wall.

If a high pressure pipe were to rupture within the drywell, the containment shell might be damaged in three different ways. These are direct impingement on the wall of the jet of fluid issuing from the broken pipe, the reaction forces of the jet acting on containment penetrations, and impact of a pipe that is moved by jet forces (pipe whipping). The plant design includes provisions in the design to reduce the possibility of containment failure as a result of these effects.

The direct impingement of a jet on the containment wall has been considered in the design of the containment, and adequate strength has been provided to prevent failure as a result of such impingement. Reaction loads acting on containment penetrations have also been considered in the design, and anchors and limit stops located outside the containment have been provided to limit pipe movement and prevent failure of the containment. To prevent pipe whip from causing failure of the containment, two design approaches have been taken. In the first approach the reactor coolant system recirculation lines have been provided with restraints which will prevent these lines from whipping in the event one ruptures. This design approach was not applied to the other lines within the drywell, such as the steam and feedwater lines. However, the applicant is protecting the lower spherical portion of the drywell wall with energy absorbing material. The material is a corrugated steel plate sandwich which can plastically deform to absorb the energy of a whipping pipe and is the same material previously proposed for and used in Vermont Yankee. This material provides protection to the containment against the effects of whipping of the main steam, feedwater, and RHR pipelines. In addition, TVA will inspect the critical welds of this unrestrained piping inside the drywell at a more frequent interval than that required by the inservice inspection program. The probability of failure of these lines is therefore minimized because of the accelerated inservice inspection program and because of the leak detection capabilities at the units. We therefore conclude that since the majority of the piping in the containment is either restrained or the containment is protected against its failure, and the remainder of the piping is of high quality, frequently inspected and continuously monitored for leakage, the probability of violating the integrity of the containment is acceptably low.

The reactor supplier, General Electric, has been conducting studies to analyze the potential for damage from missiles originating in a recirculation pump following a postulated pipe rupture. Topical Report NEDO-10677, "Analysis of Recirculation

Pump Overspeed in a Typical General Electric Boiling Water Reactor," dated October 1972, concludes that destructive pump and motor overspeed could occur for the double ended pipe break LOCA in either recirculation pump suction or discharge line with the potential for formation of damaging missiles.

The report recommends the use of a decoupling device between the pump and motor to prevent the pump driving the motor to destructive overspeed. The report recommends the use of additional pipe supports and restraints to prevent pump missiles capable of causing damage from escaping through the open end of the broken pipe.

The report is currently being evaluated by the staff.

The applicant has previously committed to implement any design measures required to prevent the generation of missiles from recirculation pump-motor overspeed if the General Electric studies indicated potential problems.

Following the staff's evaluation of the report, we will require the applicant to submit details of proposed design changes and a schedule for implementation. Resolution of this matter is not required for licensing Unit 1.

5.2.4 Containment Atmosphere Control

Following a loss-of-coolant accident (LOCA), (a) hydrogen gas could be generated inside the primary containment from a chemical reaction between the fuel rod cladding and steam (metal-water reaction), and (b) both hydrogen and oxygen would be generated as a result of radiolytic decomposition of recirculating coolant solutions. If a sufficient amount of the hydrogen is generated and oxygen is available in stoichiometric quantities, the subsequent reaction of hydrogen with oxygen can occur at rates rapid enough to lead to a significant pressure increase in the containment. This could cause damage to the containment and could lead to failure of the containment to maintain low leakage integrity.

General Design Criterion 41 of Appendix A to 10 CFR Part 50 requires that systems to to control hydrogen, oxygen and other substances which may be released into the primary containment be provided as necessary to control their concentrations following postulated accidents to ensure that containment integrity is maintained. In accordance with guidelines of the supplement to Safety Guide 7 "Control of Combustible Gas Concentrations in Containment Following a Loss of Coolant Accident," the applicant has proposed a Containment Atmospheric Dilution System (CAD).

Basically the CAD concept involves the maintenance of an oxygen deficient (inert) containment atmosphere in the post-LOCA period. This would be accomplished by addition of nitrogen gas from an external nitrogen makeup and supply system. As nitrogen is added, the containment pressure would rise in the post-LOCA period. However, even assuming a zero containment leakage rate in the post-LOCA period, the containment pressure would reach about 40 psig within 30 days following the accident. Assuming that no accident recovery actions were to be undertaken after the 30-day period, it would take about 2 months before the containment design pressure (56 psig) could be reached. Under this condition, containment purging under long-term controlled conditions would be necessary to prevent excess pressure rise and to allow the introduction of nitrogen to maintain the hydrogen-oxygen balance below the flammable limits. Resultant radiological doses would not exceed the 10 CFR Part 100 guideline values. If the containment is assumed to leak at a rate of 2 w/o per day, as is the situation postulated for analyses of the radiological consequences of a LOCA, the containment pressure would not exceed about 35 psig at any time during the post-LOCA period. Consequently, use of the CAD system as conceived should allow the control of combustible gases to be accomplished in the post-LOCA period, while at the same time its usage should not increase the presently predicted radiological consequences of a LOCA.

The applicant has provided design details and analysis of the system in Amendment 41. We have reviewed this information and find it acceptable. In answer to the ACRS recommendation that the applicant study means to assure that the repressurization pressure be limited to a value substantially below the containment design pressure, the applicant has proposed a containment repressurization limit of 30 psig which is about half of design pressure. The applicant has calculated that, using the assumptions of Safety Guide 7 with no containment leakage, purging at 10 days at a rate of 12 scfm would limit containment pressure to 25 psig. We have calculated the site boundary dose due to the 10 day hold up followed by a 12 scfm purge rate as 39 rem thyroid and .26 rem whole body.

We conclude that the containment repressurization limit is acceptable and that there is reasonable assurance that this limit can be maintained for acceptable operation of the CAD system in the unlikely event of a LOCA. We also conclude that the resultant doses from the proposed CAD operation are acceptable.

7.2.1 (1) Incident and Accident Surveillance Instrumentation

The BWR reactor protection and engineered safety feature instrumentation channels generally use blind sensors and, therefore, do not provide continuous readout in the control room of the parameters being monitored. The neutron monitoring and main steam line radiation monitoring systems are exceptions. The other vital parameters, however, are monitored by instrument channels associated with control systems. As such, these information readout channels are not designed to satisfy protection system criteria and availability and testing requirements are not included in the Technical Specifications.

Information readout channels are required by the operator to assess plant conditions during and subsequent to an anticipated operational occurrence or accident in order that he may determine whether to intervene in the operation of the Automatic Depressurization System (ADS) or to initiate containment spray. The applicant has provided a list of redundant channels that readout and, in some cases, are recorded in the control room. This listing was consistent with that of the Pilgrim design except that the applicant had not proposed redundant surveillance instrumentation for monitoring primary containment pressure. Amendment 39 proposed the installation of a second primary containment pressure monitoring instrument and we conclude that adequate information is provided to the operator.

7.2.1 (6) Anticipated Transient Without Scram (ATWS)

As further confirmation of the adequacy of design, we and the ACRS have requested the reactor supplier, General Electric, to study means for preventing common mode failures from negating scram action and design additional features to mitigate the consequences of failures to scram during anticipated transients. GE has submitted the results of these studies in two topical reports, NEDO-10189, "An Analysis of the Functional Common Mode Failures in GE BWR Protection and Control Instrumentation" dated July 1970 (submitted October 26, 1970), and NEDO-10349, "Analysis of Anticipated Transients Without Scram" dated March 1971 (submitted May 4, 1971). These reports are now under review by the regulatory staff and the applicant has agreed to install these systems when our review and the system design is complete.

The applicant, by Amendment 43, has submitted plans for installation of a recuirculation pump trip to mitigate the consequences of a failure to scram during anticipated transients. The pump trip is automatic on either a signal of high pressure or low water level. The pressure and level devices used for pump trip are not the same level and pressure devices that are used for scram. In addition, the pressure switches used to trip the recirculation pumps will be of a different type and manufacturer than those used for scram. The applicant has indicated that the Unit 1 modifications will be completed before Unit 1 exceeds 1% power.

The staff agrees that the addition of the recirculation pump trip as proposed by the applicant represents a substantial improvement in protection of the reactor for anticipated transients without scram; however, the staff has not completed its review of all the transients as discussed in the General Electric Company Topical Report NEDO-10349. Completion of our review of this topic is pending receipt of and review of responses to additional information which was requested from General Electric in a letter dated June 13, 1972. The staff has not concluded that the proposed recirculation pump trip provides a completely acceptable degree of protection against anticipated transients without scram for reactors of this general type. This conclusion is pending our receipt and review of the outstanding information cited above. The General Electric Company has indicated that the information requested by the staff regarding anticipated transients without scram will be submitted as a topical report in early 1973.

7.2.1 (8) Operational Bypasses

The applicant has proposed that circuitry be included to provide a means for manually bypassing one of the initiating signals for the core spray and low pressure coolant injection system (i.e., high drywell pressure coincident with low reactor pressure) during integrated leak rate testing of any unit and during a blowdown of a unit to its suppression pool via power relief valves. The bypass provides for additional reduction in possible generation of false accident signals. The applicant, in Amendment 39 and 41, has submitted revised Functional Control Diagrams and a commitment that the revised circuitry will be designed to meet the intent of IEEE-279. We conclude that the proposed approach in providing operational bypasses is acceptable.

7.2.2 (2) Environmental Testing

In response to our request for test results establishing the suitability of electrical equipment and components within the containment to sustain accident or anticipated operational occurrence environments, the applicant stated that these equipment and components are identical to those used and found acceptable in Millstone 1. The applicant has in addition proposed to add circuitry to alert the operator to high primary containment temperatures. The operator will be instructed by operating procedures to initiate containment spray in order to ensure that primary containment for certain plant conditions does not exceed 281°F for 30 minutes or 35 psig high drywell pressure. The need for this operator action and the additional circuitry is to ensure, with margin, that the environmental capability of the instrumentation in containment is not exceeded.

The applicant, in Amendment 39, has described the additional circuitry to annunciate the containment conditions requiring manual actuation of the containment sprays. We have reviewed the description and have concluded that the added circuitry need not meet the single failure criterion. This is based on the annunciation being only a means to alert the operator. The instrumentation relied upon for monitoring the parameters involved (primary containment pressure and temperatures) is redundant and is read out in the control room.

7.2.3 Separation Criteria

The applicant's separation criteria were incomplete in some areas. One of these areas concerns the separation of redundant devices and the connection of redundant circuits to single devices in control room panels, boards, and racks. Consistent with our position in Pilgrim and Vermont Yankee, we require that redundant protection system circuits not be connected to a single device (switch) and that a minimum separation of 6 inches or physical barrier be provided between such devices. The applicant, in Amendment 39, committed to the required separation criteria.

Our review also revealed that the HPCI and RCIC steam supply line redundant high flow sensors are mounted on common racks. The applicant has now demonstrated acceptability of the high flow sensors mounted on common racks based on high temperature detectors along the steam lines providing diverse redundant protection.

Another area where the applicant's separation criteria were incomplete concerned cable routing. We identified the criteria which had been omitted and the applicant has responded by including these criteria with a minimum of exceptions. The exceptions are concerned with the degree of separation (9 vs 12 inches between cable trays). We do not consider this to be sufficiently significant to safety to warrant backfit and have determined that the applicant's design is acceptable.

7.3 Emergency Electrical Power System

7.3.1 Offsite Power

Units 1, 2 and 3 of the Browns Ferry Nuclear Plant will be interconnected to the transmission system through 500 kV circuits. Power from each unit generator will be fed via separate circuit containing a step-up transformer to the 500 kV switchyard. The 500 kV switchyard will be arranged in a modified breaker-and-a-half configuration. Six transmission circuits will emanate from the plant. These circuits are routed on separate rights-of-way.

Offsite power for plant startup, shutdown, and engineered safety features is supplied from a separate 161 kV switchyard. This switchyard is connected to the 161 kV grid by two circuits each of which is mounted on separate towers. While these two circuits share a common right-of-way for a short distance, there is sufficient separation to preclude one tower or line failure from affecting the other. The 161 kV switchyard is arranged in a simple two bus configuration interconnected with a single circuit breaker and motor operated switch to disconnect these buses in the event of circuit or bus faults.

The failure of this circuit breaker could result in the loss of offsite power to the plant. In response to our concern, the applicant has stated in Amendment 14 that his design provides protection against the most probable causes of failure with the following features:

(1) Two trip coils are provided; one coil tripped from

normal relays, the other tripped by backup relays.

(2) The trip coils are continuously monitored from the control room.

(3) The circuit breaker is provided with manual mechanical trip device.

(4) Fail-to-trip relaying has been added to trip incoming (supply) breakers at their source if the breaker does not trip.

The applicant has provided, in conjunction with "(4)" above, an analysis which shows that the control room operator can activate the motor operated disconnect switches and isolate the fault in sufficient time to re-energize one bus in the switchyard to ensure that the system meets the requirements of General Design Criterion 17 of Appendix A to 10 CFR Part 50 as published in the Federal Register on February 20, 1971. On the basis of our review, we conclude that the design of the offsite power to the 161 kV switchyard satisfies General Design Criterion 17 in this regard and is acceptable.

Two circuits interconnect the 161 kV switchyard to the plant emergency distribution system. Each circuit is routed on separate towers through a redundant 161/4.16 kV common station service transformer to the 4160 volt distribution system shutdown buses. There is one tower immediately adjacent to the plant whose failure could result in the loss of the redundant 161 kV circuit. The applicant has agreed to increase the separation and, by Amendment 39, has shown the relocation of the tower and has committed to completion of the modification prior to licensing of Unit 1. We now find the circuit routing to be acceptable.

The offsite power available to the shutdown boards is limited by the size of the circuit breakers. Less than half of the installed cooling equipment can be operated with offsite power sources. This results in an inordinate amount of operator action to provide for safe shutdown of the facility. The applicant has agreed to increase the shutdown capability of the plant with offsite power sources, however, the designs are not complete. The unacceptable aspects of the design are only related to multiple facility operation. The system is acceptable for Unit 1 operation only. This item will be considered as a condition to the licensing of multi-unit operation and will be resolved prior to licensing of Unit 2.

Our review of the offsite power system design reveals that the design pending satisfactory resolution of the above mentioned matters meets the requirements of General Design Criterion 17 and IEEE-308 and is acceptable.

7.3.2 Onsite Power

The initial submittal for the emergency standby a-c power system for the plant consists of four diesel generator sets each assigned to power one 4160 volt shutdown board. The engineered safety feature (ESF) and shutdown loads for all three units were distributed among these shutdown boards and attendant distribution systems. The intent of this arrangement was to ensure that any three of the four diesel generator sets or shutdown boards would supply minimum ESF loads in one unit and safe shutdown loads in the remaining two units.

The applicant had attempted to respond to the concerns of the ACRS as expressed in the Committee's letter dated May 15, 1968 issued in connection with its review of the application for a construction permit for Brown's Ferry Unit 3. These concerns were with regard to the improvement of the marginally acceptable onsite power system with respect to capacity of diesel generator sets and the need for paralleling of these generators. The applicant attempted to improve the design by eliminating the need for paralleling the diesel generators. However, these attempted design improvements resulted in the development of a more complex design that required extensive interrelationship among the units' control circuits, required automatic transfer of load groups, resulted in excessive diesel generator loadings and required an excessive amount of operator coordination.

Our review of the system revealed that single circuit failures, maintenance operations or testing operations in one unit would affect all or at least half of the ESF in the remaining two units. This was due to the need to shed and lockout non-essential loads in the accident unit and ESF of the non-accident units made necessary because of the limited capacity of the totally shared standby a.c. power supply. The control circuits which accomplished this shedding and lockout were initiated by the accident signals and effect the block or lockout in the ECCS circuits of each unit. Therefore, with regard to this control scheme, the ECCS circuits of each unit were inter-connected. This interrelationship was such that the testing of a channel of one unit and another channel in another unit could disable automatic ECCS actuation in all three units. This design interrelationship was not consistent with our requirements for independence in the design of engineered safety feature control circuits. We, therefore, concluded that the controls needed to be modified to provide additional independence prior to issuance of an operating license for Unit 2.

We could not conclude that the capacity of the onsite a-c power system was adequate to provide safe and orderly shutdown of the plant as required by General Design Criterion 5 of Appendix A to 10 CFR Part 50. The diesel generators did not have the capacity to power a sufficient number of Class I seismically qualified cooling components to allow safe and orderly shutdown of the plant without exceeding the guidelines of Safety Guide 9, "Selection of Diesel Generator Set Capacity For Standby Power Supplies." Further, the associated electrical distribution design of the onsite a-c power system was extensively shared among the three units which resulted in a complex design requiring extensive electrical interlocks and an excessive amount of operator control. We concluded that the design of the onsite a-c power system although acceptable for operation of Unit 1 was unacceptable for multiple unit operation.

The concerns for operation of Unit 1 only were in regard to automatic bus transfer features of the standby a-c and d-c systems. The applicant was advised that automatic bus transfer of a-c loads should be limited to only the low pressure coolant injection system valves to make the design more consistent with the guidelines of Safety Guide 6, "Independence Between Redundant Standby (Onsite) Power Sources and Between Their Distribution Systems" and that an additional battery should be added with associated changes to the d-c system to eliminate the need for automatic bus transfer of d-c loads as expressed by Safety Guide 6. The applicant, in Amendment 39, submitted a description of the design changes necessary to eliminate automatic transfer of d-c loads and to limit automatic transfer of a-c loads to the low pressure coolant injection valves. The applicant committed to completing these modifications prior to licensing of Unit 1.

The applicant has now provided, by Amendments 40 and 43, a redesigned emergency standby a-c power system. They propose to provide a new shutdown a-c power distribution system to Unit 3 which includes four additional diesel generators and four 4.16 kv additional shutdown distribution boards. The system for Unit 3 will be separated from that of Units 1 and 2. The new diesel generator units will be the same as the existing units. For flexibility of operation, provisions will be made to interconnect each 4.16 kv shutdown boards for Unit 3 to a corresponding 4.16 kv shutdown board for Units 1 and 2 through manually controlled breakers.

The system has the capability to provide emergency power to accommodate any combination of accident signals (real or spurious) in any unit without operator action for the short term (0-10 minutes) within the ECCS interim criteria established for calculating peak fuel cladding temperatures. The system can also accommodate the power needs to provide long term shutdown on all units.

All electrical modification work affecting the operation of Units 1 and 2 will be completed before Unit 1 operates. No connections will be required to shared systems that will require shutdown of Unit 1. All major construction work involved with the modification will be confined to the Unit 3 area.

The applicant's final design will be submitted as an Amendment to the FSAR in January 1973. We reviewed the proposed design at this time on the basis of its acceptability for operating Unit 1 only and to make an evaluation of the proposed system design criteria and design approach.

7.3.3 Conclusions

Our conclusions are separately grouped below for single unit operation (Unit 1) and multiunit operation.

1) Operation of Unit 1 only

We conclude that the emergency electrical power systems are acceptable for operation of Unit 1.

2) Multiunit Operation

We conclude that 1) the implementation of the proposed modifications for multi-unit operation will not adversely affect the operation of Unit 1, 2) the design bases of the proposed system appear acceptable for multi-unit operation and 3) the design approach, subject to properly implementing the criteria and subject to our final review of the circuitry regarding system interaction, appears to be an acceptable concept for multi-unit operation.

We will review the detailed design, presently scheduled for January 1973 submittal, and prepare a Supplement to the SER prior to licensing Unit 2.

8.4 Control Room Ventilation Systems

The applicant proposes to meet General Design Criterion No. 19, Control Room, of Appendix A to 10 CFR Part 50, by filtering inlet air to the control room. (Includes entire upper level of the auxiliary building which contains the control panels of Units 1, 2 and 3). Reference is made to Amendments 38 and 40 of the Browns Ferry Application which describes the system as modified. An accident signal from any one of the three reactor units or a high activity indication from control room radiation monitors would actuate the filter trains. Two 500 cfm clean-up trains maintain a positive pressure within the control room and preclude unfiltered inleakage. Each train consists of an isolation damper, a HEPA filter, two-two-inch deep charcoal beds in series, a fan, and a backflow damper. Each fan unit has sufficient capacity to maintain the control room at a slight positive pressure of approximately 1/4 in. water. The operator may manually select, from a number of ventilation modes, that one which best fits the circumstances.

The overall system design has been analyzed by the Staff. Doses to the control room operators have been calculated assuming conservative iodine source terms from a LOCA or a main steam line break accident. The LOCA is the controlling accident. Based on an assumed 95% iodine removal efficiency for elemental and 90% for organic by the charcoal filters, the dose rates are within Criterion 19 guidelines.

Although redundant filter trains are provided the applicant is relying on single valves for isolation of the Control Room from the normal ventilation paths. Failure of one of these valves to close would reduce the effectiveness of the proposed system. The applicant has stated that the isolation valves are all located in the Control Room area with local position indicators and means for manually closing a failed open valve. We conclude that sufficient time is available for the operators to assure isolation from the normal ventilation system and that the Control Room Ventilation System, as proposed, is acceptable.

9.4 Control Rod Drop Accident

For the postulated design basis control rod drop accident, it is assumed that a bottom entry control rod has been fully inserted and has stuck in this position, the drive becomes uncoupled and withdrawn from the rod. Subsequently, it is assumed that the rod falls out of the core inserting an amount of reactivity corresponding to the worth of the rod.

From the standpoint of radiological consequences, when the reactor is in the hot standby condition at zero power is the worst situation at which a rod drop accident could occur because a high energy release is calculated for this condition and because a path for the unfiltered release of fission products could exist through the turbine-condenser systems.

The reactor is designed to reduce the probability of this accident and engineered safety features are provided to limit the consequences of the accident. For example, the control rod worth minimizer is designed to limit the reactivity worth of any control rod during the startup phase of reactor operation. The control rod velocity limiter will limit the velocity during free fall to less than five feet per second. The steam line radiation monitor will detect excessive radioactivity and isolate the main turbine and condenser by closing isolation valves in the condenser mechanical vacuum pump system before the radioactive steam can travel from the detector to these isolation valves. Because of the operation of these engineered safety features, the fission products that escape to the environment would be only those which leak from the isolated turbine and condenser.

In evaluating the radiological consequences of this accident, we have made assumptions based upon the applicant's analytical model as presented in the Final Safety Analysis Report. As discussed in the subsequent paragraphs, the analysis techniques for this particular accident have been revised by General Electric and we have required modifications, in addition to those presently provided, to mitigate the potential consequences.

The Atomic Energy Commission has for some time utilized Brookhaven National Laboratory (BNL) as its consultant as part of the regulatory assistance program. For some time, personnel at BNL have been performing independent calculations of boiling water reactor control rod worths and potential consequences of a design basis control rod drop accident. As a consequence of the work performed to date at BNL,* it appears that the model used by General Electric to evaluate the design basis control rod drop accident should be revised.

Specifically, the assumed rate of negative reactivity insertion from control rod scram is not suitably conservative

*BNL 16717-RP1021, "Rod Drop and Scram in Boiling Water Reactors," dated April 1972 since it uses insertion characteristics now considered to be not readily attainable in large boiling water reactors. In addition, the actual reactivity insertion rates are not linear as assumed.

The General Electric Company has now revised the analysis of the effects of a control rod drop accident and has submitted topical reports** to the regulatory staff. The analysis presented in the reports applies to those reactor plants using control curtains in the core for initial reactivity control with a supplement applicable to the multiple enrichment cores with axial gadolinium (Browns Ferry Class). The regulatory staff with the assistance of BNL is currently evaluating the adequacy of the revised model and the resultant consequences of this postulated accident. Included in the revised analyses are, among other features, a change in the method for modeling considering flux shape factors which affect the rate of negative reactivity insertion from a control rod scram.

The analyses provided for the multiple enrichment cores with axial gadolinium indicate unacceptable results for the maximum out-of-sequence rod drop accident below about 10% of rated power level for the most reactive part of the fuel cycle, i.e., the resultant peak fuel enthalpy exceeds the threshold value (280 cal/gm) assumed to cause rapid fuel dispersal and damaging pressure pulses to the reactor core.

**NEDO-10527, "Rod Drop Accident Analysis for Large Boiling Water Reactors," dated March 1972, and Supplement 1 to NEDO-10527, dated July 1972.

We have concluded that modifications are required to augment the Rod Worth Minimizer (RWM) to make the probability of occurrence of this postulated accident negligibly low.

In Amendments 43 and 44, the applicant has proposed the installation of a Rod Sequence Control System (RSCS) as a backup to the RWM as described in Section 3.4. The RSCS is designed to prevent the operator from withdrawing an out-of-sequence control rod during start up or shutdown in the 100% to 50% rod density*** range. Beyond 50% rod densit the peak enthalpies relulting from a rod drop accident are less than 280 cal/gm.

The applicant has stated that the peak enthalpy value of 276 cal/gm for exposed core conditions presented in Amendments 43 and 44 are preliminary. The final calculated values will be submitted in February 1973.

The accident calculations in the SER using the previous model assumed the most reactive control rod assembly to drop out of the core during a startup from hot standby 30 minutes after shutdown, causing 330 fuel rods to exceed a calculated energy input of 170 cals/gm. These rods were assumed to perforate, releasing 100% of the contained noble gases and 50% of the contained halogens to the reactor coolant system. Of the halogens released from the affected rods, 90% are assumed to be retained in the primary system

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***Rod density is defined as the percent of control rods fully inserted in the core.

and one-half of the remaining halogens are assumed to be removed by plate-out. All of the noble gases and 2.5% of the halogens are assumed to be released from the primary system through the condenser system to the atmosphere. A conservative ground level release from the turbine building was assumed. A wake factor of 0.5, a turbine building area of $2400m^2$, and Safety Guide 3 meteorology assumptions are used for diffusion calculations.

Exposure doses calculated for the whole body and for the thyroid at the Exclusion Area Boundary are less than one Rem and 3.6 Rem, respectively for the assumed two hours exposure, and at the Low Population Zone Boundary are less than one Rem and 5.9 Rem for 24 hours exposure assumed as the duration of the accident. The exposure doses for this accident are well within 10 CFR Part 100 guidelines.

The reactor supplier, General Electric, has stated that beyond 50% rod density, current estimates indicate that approximately 600 fuel rods will perforate following the assumed rod drop accident. The resultant exposure doses would be approximately double those listed above and would still be well within 10 CFR Part 100 guidelines.

The staff must complete its evaluation of the proposed system and the applicant must confirm that peak fuel enthalpies resulting from the rod drop accident beyond 50% rod density do not exceed 280 cal/gm. In the event the finalized calculations for the peak fuel enthalpies exceed our acceptance criterion of 280 cal/gm we will require the applicant to provide additional modifications to satisfy this Criterion. If found acceptable, the RSCS installation will be required prior to the reactor exceeding 1% of rated power. The Technical Specifications will require that the RSCS, when accepted and installed, be operable below 10% power level and that the control rod scram time (to 90% insertion) be less than 4.0 seconds, the value used in the revised rod drop accident analysis.

10.0 DESIGN BASES FOR STRUCTURES AND EQUIPMENT

The applicant has classified the plant structures and equipment into two categories dependent upon their relationship to safety.

Structures (e.g., reactor pressure vessel and internals, primary coolant system and the emergency core cooling system) whose failure could cause significant release of radioactivity or which are vital to a safe shutdown of the facility and the removal of decay heat have been classified as Class I. Class II structures and equipment are defined as those which are necessary for station operation but are not essential to a safe shutdown. We have reviewed the applicant's classification of structures and equipment and we conclude that they have been appropriately classified.

The Class I reactor building, concrete chimney and pumping station structures are founded on mats on bedrock. The Class I diesel generator building is founded on about 3 feet of earth backfill on top of 32 feet of crushed stone backfill. The Class I equipment access lock rests on a row of steel bearing piles to rock under each vertical wall and another row at the mid-point of the ground level slab. The Class I standby gas treatment structure bears on about 10 feet of earth backfill over the same crushed stone backfill as for the diesel generator building. The Class II turbine building is supported on steel H-piles to bedrock. As a result of some weathered rock in the foundation material the

Unit 1 reactor building was underpinned, while under the Unit 2 and 3 portions fill concrete was placed. Seam grouting was utilized under the turbine building for the bearing pile clusters. The foundations as designed are acceptable, and it can be concluded that their construction was in accordance with the design criteria.

Class I structures, as defined in Appendix C of the FSAR and listed in Section 12 of the FSAR are designed for normal dead and live loads, 100 mph wind, 300 mph tornado wind and 3 psi pressure drop, operating and design basis earthquakes of 0.1 g and 0.2 g maximum ground accelerations respectively. Soil, hydrostatic and missile loads have also been included.

For tornado design, the upper 320 feet of the chimney is designed to fail well before the lower 280 feet reaches its ultimate load capacity. Therefore, the chimney fall line under tornado winds does not reach any Class I structures, the nearest of which is 365 feet from the chimney. Pieces of concrete and an aircraft warning beacon are considered as potential missiles originating from the chimney in the spectrum of missiles for which Class I structures are analyzed.

The Radwaste Building, although not defined as a Class I (Seismic) structure, meets Class I (Seismic) structural design criteria under tornado or earthquake loading, and it can be concluded that it will satisfactorily perform its function under these loads.

The reactor vessel concrete support pedestal is capable of withstanding, within acceptable stress limits, either design basis accident, earthquake (OBE or DBE) or design basis accident combined with earthquake (OBE or DBE).

The applicant has described the consequences of a short duration peak temperature on the drywell steel shell of 340°F. No buckling is anticipated, and stresses remain within the allowable stress intensity value of the ASME code. Direct jet impingement on the drywell plate has been analyzed by the applicant and a determination made that containment integrity would not be endangered. We have reviewed the applicant's findings with respect to the effects on the containment of local or general high temperatures and find them acceptable.

Splicing of reinforcing bars by the Cadweld process, where used, was carried out with an acceptable testing program to ensure quality control.

The design strength of the concrete is generally 3000 psi with some 4000 psi. The reinforcing used conforms to ASTM A432 and has a yield point of 60,000 psi.

No unresolved construction items are under review, and the materials used in construction are considered to be acceptable.

The secondary containment building will be leak tested after construction to verify a minimum of 0.25 inch water gauge negative pressure at calm wind conditions at a flow rate of 9000 cfm (1.5 secondary containment volumes per day). Surveillance will be carried out as charted in Table 5.3-1 of the FSAR. Penetration testability has been reviewed and found to be acceptable.

Amendment 24 presented structural revisions to the intake and discharge structures which will be completed prior to Unit 1 operation. These are structural modifications for the future use of cooling towers, in place of once-through river cooling water. The only changes reviewed are those which will be carried out prior to construction, in order not to interfere with plant operation at a later date if cooling towers are to be used.

The Class I intake structure will have a cellular cofferdam installed, with an opening left in the center for continued flow of river water, but which can be closed off when cooling towers are installed. The design criteria have been reviewed and are acceptable.

The discharge structure (not Class I) will have future connection openings installed in the conduits, and gates placed and provided for in order to make it possible to reroute the discharge water when future cooling tower connections are made.

In evaluating the structural design of the Class I structures, systems, and equipment, our seismic design consultant (Nathan M. Newmark Consulting Engineering Services), whose report is enclosed as Appendix C, concluded that the design incorporates an acceptable range of margins of safety for the hazards considered and that the design could be considered adequate in terms of provision for safe shutdown for the design basis earthquake and capable otherwise of withstanding the effects of an operating basis earthquake.

Class I components for the mechanical fluid systems exclusive of the reactor coolant pressure boundary have been designed, fabricated and inspected in accordance with the following codes:

(a) Piping conforms to the requirements of the USAS B31.1.0-1967

(b) Pumps conform to the Class C requirements of Section III of the ASME Boiler Pressure Vessel Code.

(c) Valves conform to the B31.1.0-1967 Code for Pressure Piping.

We find the codes and standards specified for Category I mechanical fluid systems provide an acceptable quality level and are consistent with recently reviewed plants of this type.

All Class I systems, components, and equipment outside of the reactor coolant pressure boundary were designed to sustain the Operational Basis Earthquake within the appropriate code allowable stress limits and the Design Basis Earthquake within stress limits which are comparable to those associated with the emergency operating condition category of current component codes. We consider that these stress criteria provide an adequate margin of safety for Category I systems and components which may be subjected to seismic laodings.

Modal response spectrum multi-degree-of-freedom and normal mode-time history methods are used for the analysis of all class I structures, systems and components. Governing response parameters have been combined by the square root of the sum of the squares to obtain the modal maximums when the modal response spectrum method is used. The absolute sum of responses is used for closely spaced frequencies. Concurrently applied horizontal and vertical floor spectra inputs used for design and test verification of structures, systems and components were generated by semi-empirical methods and confirmed by the normal mode-time history method. Vertical ground accelerations were assumed to be 2/3 of the horizontal ground accelerations for items rigidly attached to structures. Constant vertical load factors were employed only where analysis show sufficient vertical rigidity to preclude significant vertical amplifications in the seismic system being analyzed. We and our seismic consultant conclude that the seismic-system dynamic methods and procedures proposed by the applicant provide an acceptable basis for the seismic design.

The basic seismic instrumentation program proposed for this facility corresponds to the recommendation of Safety Guide 12 with respect to the type, number, location and utilization of strong motion accelerographs to record seismic events and to provide data on the frequency, amplitude and phase relationship of the seismic response of the containment structures. In addition, the applicant has proposed, in Amendment 39 to the FSAR, the installation of supporting seismic instrumentation on selected mechanical components in order to provide data for the verification of the seismic responses determined analytically for those representative components.

We conclude that the basic seismic instrumentation program proposed by the applicant is acceptable.

15.0 REPORT OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

The report of the ACRS on this project is presented in the attached Appendix B. The report deals only with Unit 1. As discussed under Section 7.3 Emergency Electric Power System of this Supplement, the design details covering the proposed modifications for multi-unit operation will be reviewed by the Regulatory Staff and the ACRS prior to licensing of Units 2 and 3.

The other recommendations from the report and the appropriate Section of the Safety Evaluation Report which discusses the implementation of these recommendations are:

a) The repressurization pressure resulting from use of the CAD system during the post-LOCA period should be limited to a value substantially below the containment design pressure. (See Section 5.2.4 SER Supplement)

b) The applicant should continue to study means of assuring reactor vessel integrity in regions currently inaccessible for inspection. (See SER Section 4.8)

c) The recurculation pump trip, which has been proposed as a means of limiting the consequences of the unlikely occurrence of a failure to scram during an anticipated transient, should be provided prior to the start of commercial power operation. (See Section 7.2.1(6) SER Supplement)

d) The report requires that design changes to render the probability of the control rod drop accident (over the range wherein the results are unacceptable) negligibly low should be implemented prior to operation above 1% of rated power. This matter should be resolved in a manner satisfactory to the Staff and the ACRS. (See Sections 3.4 and 9.4 SER Supplement)

We consider that the applicant was responsive to the recommendations of the ACRS indicated in their report (Appendix B) and conclude that the matters raised have been or will be satisfactorily resolved as noted herein.

APPENDIX A

CHRONOLOGY

Regulatory Review of Tennessee Valley Authority Browns Ferry Nuclear Power Station Units 1, 2 and 3

September 9, 1969	Meeting with Tennessee Valley Authority representatives to discuss organization and quality levels for the completion of fabrication of Units 2 and 3 reactor vessels.
July 6, 1970	AEC letter requests environmental impact information to be submitted when filing the Final Safety Analysis Report.
July 17, 1970	Meeting with TVA representatives to discuss conformance to proposed rule on codes and standards and use of furnace-sensitized stainless steel.
September 25, 1970	TVA submits Final Safety Analysis Report on financial information as Amendment No. 9 to the application.
September 28, 1970	TVA letter containing proposed procedures of environmental statement.
August 26, 1970	Meeting with TVA representatives to discuss procedures for preparation of environmental statement.
October 1, 1970	Receive TVA proposed procedures for preparation of environmental statement.
October 9, 1970	Meeting with TVA representatives to discuss onsite electrical power system.
November 12, 1970	AEC letter commenting on proposed procedures for the preparation of environmental statement.
November 24, 1970	TVA submits Amendment 10 containing Proposed Technical Specifications and reactor thermal-hydraulic information.
January 29, 1971	Meeting with TVA representatives to discuss review schedule and items requiring additional information.
March 1, 1971	TVA submits Amendment 11 containing Unit 1 Reactor Pressure Vessel Report and revised fuel design information.

March 25, 1971	AEC letter requests additional information.
April 16, 1971	Meeting with TVA representatives to discuss preparation of environmental statement.
May 7, 1971	TVA submits Amendment 12 containing proprietary information on fuel design.
May 22, 1971	AEC letter requests additional information.
June 17, 1971	Meeting at site to discuss hydrology and controls and instrumentation.
June 30, 1971	AEC letter confirming procedure for the preparation and issuance of the environmental statement.
July 15, 1971	TVA issues Draft Environmental Statement for comment.
July 30, 1971	AEC letter requests additional analyses consistent with AEC interim criteria for the performance of emergency core cooling system.
August 3, 1971	TVA submits Amendment 13 containing revised and supplementary information in response to 3-25-71 DRL letter.
October 12, 1971	AEC letter requests additional information relative to the requirements of Safety Guide 7.
October 18, 1971	TVA submits show cause information.
November 2, 1971	TVA submits Amendment 14 containing partial responses to AEC letters dated 3-25-71 and 5-22-71 and all the information requested in DRL letter dated 7-30-71.
November 8, 1971	TVA submits supplements and additions to Draft Environmental Statement for comment.
November 11, 1971	TVA submits Amendment 15 containing revised and supplementary information in response to 3-25-71 and 5-22-71 AEC letter.
November 24, 1971	AEC publishes Show Cause Determination and discussion and findings not to suspend construction of the Browns Ferry Nuclear Plant.
December 1, 1971	TVA submits Amendment 16 containing revised Proposed Technical Specifications and information in response to 3-25-71 DRL letter.

December 6, 1971	AEC letter requests additional information.
December 14 & 15, 1971	Meeting with applicant to discuss hydrology, conduct of operations, radwaste systems, electrical power and instrumentation and control system and review schedule.
January 4 & 5, 1972	Meeting to discuss onsite electrical power system, instrumentation, and control system and future proposed fuel modification.
January 19, 20 & 21, 1972	Meeting with applicant to discuss controls and instrumentation systems.
January 26 & 27, 1972	Meeting with applicant to discuss Proposed Technical Specifications.
January 26, 1972	TVA submits Amendment 17 containing proprietary information on fuel design.
February 1, 1972	TVA submits Amendment 18 containing supplemental and revised information.
February 3, 1972	Meeting with TVA to discuss quality assurance program, emergency operating procedures, hydrology, pipe whip protection; and the standby gas treatment system.
February 10, 1972	AEC letter commenting on radiological matters of the TVA Draft Environmental Statement.
February 10, 1972	Meeting with TVA to discuss review schedule and affects of stem bypassing of suppression pool.
February 11, 1972	Meeting with TVA to discuss Proposed Technical Specification.
February 14, 1972	TVA submits Amendment 19 containing supplemental and revised information.
February 23, 1972	TVA submits Amendment 20 containing proprietary information related to revised fuel design.
February 28, 1972	TVA submits Amendment 21 containing revised information related to fuel design.
February 28, 1972	TVA submits Amendment 22 containing revised information; responses to AEC letter dated 10-12-71 and partial responses to AEC letter dated 12-6-71.

March 2 & 3, 1972	Meeting with TVA to discuss Proposed Technical Specifications.
March 6, 1972	TVA submits Amendment 23 containing proprietary information related to industrial security plans.
March 13, 1972	AEC letter requesting additional analyses of tower structural components.
March 14, 1972	Meeting with TVA to discuss onsite and offsite electrical power system.
March 20, 1972	TVA submits Amendment 24 revising information.
March 22, 23 & 24, 1972	Meeting with TVA to discuss Proposed Technical Specifications.
March 27, 1972	TVA submits Amendment 25 revising information.
March 29, 1972	AEC letter requests additional information on combustible gas control system.
April 18, 1972	TVA submits response to AEC letter dated 3-13-72.
April 26, 1972	AEC letter requesting additional financial information.
April 26, 1972	TVA submits Amendment 26 containing revised Proposed Technical Specifications, Unit 2 Reactor Pressure Vessel Report and revised responses to DRL requests for information.
	TVA submits Amendment 27 containing revised proprietary information related to industrial security.
May 10, 1972	Meeting with TVA to discuss fuel design.
May 11, 1972	TVA submits Amendment 28 containing revised information to the Proposed Technical Specifications, revised answers to previous DRL requested information.
May 12, 1972	Meeting at site to discuss emergency power system.
May 19, 1972	TVA submits Amendment 29 containing revised information and additional financial information requested by AEC letter dated April 26, 1972.

May 25, 1972	TVA submits Amendment 30 containing revised proprietary information related to "Protection Against Industrial Sabotage" report.
May 25, 1972	TVA submits Amendment 31 containing revised responses to AEC questions and additional emergency plan information.
June 2, 1972	AEC letter requesting additional information on fuel design.
June 7, 1972	Meeting with applicant on outstanding review items needed to complete Safety Evaluation Report, including electrical power system concerns.
June 12, 1972	TVA letter transmitting Amendment No. 32, consisting of revision to TVA's Radiological Emergency Plan.
June 13, 1972	TVA letter advising that they will be able to load fuel in Unit 1 near 11-1-72.
June 14, 1972	TVA letter requesting that the formal plant name be "Browns Ferry Nuclear Plant" and transmitting a list relating to design changes, including a schedule for completion.
June 20, 1972	ACRS Subcommittee Site Meeting
June 22, 1972	TVA letter submitting Amendment No. 33, consisting of revised information for Appendix 2.4A, which has been revised to document the effects of final plant grading on local flooding.
June 26, 1972	AEC Safety Evaluation sent to ACRS.
June 27, 1972	AEC letter requesting additional information in regard to the proposed emergency electrical power systems, which confirms items discussed in meeting of June 7, 1972.
July 3, 1972	TVA letter transmitting Amendment No. 34, consisting of miscellaneous page change corrections to the FSAR.
July 12, 1972	ACRS Subcommittee Meeting.
July 13, 1972	ACRS Full Committee Meeting.

August 24, 1972	TVA letter advising that items 2 and 5 of Amendment No. 40 constitute response to AEC letters of 6-27-72 and 8-11-72.
August 25, 1972	TVA letter requesting that CPPR-29 (Unit 1), CPPR-30 (Unit 2), and CPPR-48 (Unit 3) be extended to 12-1-73, 8-1-74 and 2-1-75, respectively.
September 1, 1972	TVA letter furnishing a schedule for completion of additional design items (ECCS signal bypass; redundant drywell pressure instrument; alarm related to initiation of containment spray; and seismic monitoring equipment).
September 6, 1972	Meeting at Bethesda with TVA representatives to discuss technical specifications.
September 12, 1972	ACRS Subcommittee meeting.
September 15, 1972	ACRS Full Committee meeting.
September 15, 1972	TVA letter, supplementing their request of 8-25-72, providing information on specific reasons for delay of construction of Units 1, 2 and 3.
September 20, 1972	AEC letter transmitting Notice of Consideration of Issuance of Facility Operating Licenses and Notice of Opportunity for Hearing Pursuant to 10 CFR Part 50 Appendix D, Section C, dated 9-15-72, which provides an opportunity for hearing with respect to 1) whether construction permits for Units 2 and 3 should be continued or appropriately conditioned to protect environmental values; and 2) issuance of licenses for Units 1, 2 and 3.
September 26, 1972	AEC letter transmitting letter from ACRS, dated 9-21-72, reporting on its review of TVA's application for authorization to operate Browns Ferry Nuclear Plant Unit 1.
September 26, 1972	AEC letter transmitting (three) Orders extending the latest completion dates specified in Construction Permit Nos. CPPR-29 (Unit 1), CPPR-30 (Unit 2), and CPPR-48 (Unit 3) to 12/1/73, 8/1/74, and 2/1/75, respectively.

November 28, 1972	TVA letter transmitting request that proprietary booklet "Preoperational Flow Test and Inspection Program" be returned. This information was submitted as non-proprietary in Amendment 44.
December 6, 1972	Meeting at Bethesda with TVA representatives to discuss containment leak rate testing capabilities.
December 7 & 8, 1972	Meeting at Bethesda with TVA representatives to discuss Technical Specifications.

APPENDIX B

ADVISORY COMMITTEE ~~ON REACTOR~~ SAFEGUARDS
UNITED STATES ATOMIC ENERGY COMMISSION
WASHINGTON, D.C. 20545

September 21, 1972

Honorable James R. Schlesinger
Chairman
U. S. Atomic Energy Commission
Washington, D. C. 20545

Subject: REPORT ON BROWNS FERRY NUCLEAR PLANT, UNIT 1

Dear Dr. Schlesinger:

At its 149th meeting, on September 14-16, 1972, the Advisory Committee on Reactor Safeguards completed its review of the application by the Tennessee Valley Authority for authorization to operate Browns Ferry Nuclear Plant Unit 1 at power levels up to 3293 MW(t). This project had been considered previously at the Committee's 147th and 148th meetings, July 13-15 and August 10-12, 1972, and at Subcommittee meetings at the site on June 20, 1972, and in Washington, D. C., on July 12, August 8, and September 12, 1972. During its review, the Committee had the benefit of discussions with representatives of the Tennessee Valley Authority, General Electric Company, the AEC Regulatory Staff, and their consultants. The Committee also had the benefit of the documents listed. The Committee reported to the Commission on the construction of this unit in its report of March 14, 1967.

The Browns Ferry Nuclear Plant is located in a sparsely populated area on the shore of Wheeler Lake approximately 30 miles west of Huntsville, Alabama. The plant is composed of three units. The units employ General Electric boiling water reactor nuclear steam supply systems of essentially identical design. The estimated completion date of Unit 2 is approximately eight months after Unit 1, and that for Unit 3 is about seven months after Unit 2. This report deals only with Unit 1.

The Browns Ferry reactors have essentially the same power density and linear heat generation rate as the Vermont Yankee reactor (the Committee reported on operation of this reactor in its letter of March 9, 1971), but have the highest power level of any boiling water reactor reviewed for operation to date. The reactor core design has been substantially modified from that proposed at the construction permit stage, and employs five different fuel enrichments as well as gadolinia bearing rods for reactivity control augmentation (instead of boron-steel control curtains).

Some of the gadolinia rods are uniformly axially loaded and are similar to those used in the Quad Cities reactors (ACRS operating license report of March 9, 1971). Others, however, are non-uniformly loaded (part-length) and their use in Browns Ferry Unit 1 will represent the first application in a commercial reactor.

Analyses of postulated control rod drop accidents recently have been revised by the applicant to employ a more realistic rate of reactivity insertion than formerly assumed, and to account for the changes made in the core design, in particular the use of a number of fuel enrichments and employment of full and part-length gadolinia bearing fuel rods. These analyses indicate that, for accidents occurring during certain portions of the fuel cycle, the results are unacceptable. The applicant has proposed possible changes in plant design or operating procedures which he believes would render the probability of occurrence of such an accident negligibly low. The general approach appears feasible; however, details of the proposal are not yet available and will require thorough evaluation after submittal. This matter should be resolved in a manner satisfactory to the Regulatory Staff and the Committee. Approved measures should be placed into effect prior to operation above 1% of rated power.

Four diesel-generator sets have been provided for emergency power. Originally, the applicant planned to have these sets serve all three units of the plant. However, in order to reduce diesel-generator loadings, lessen the extensive interrelationship among the units' control circuits, and reduce the required amount of operator coordination, he recently proposed a modification which provides for sharing of these sets between Units 1 and 2 only, with four additional diesel-generator sets to be provided at a later date for Unit 3. The final design details of the proposed modification will not become available for several months. The Committee believes that the modified arrangement, which involves negligible change in Unit 1, is satisfactory for Unit 1 operation. After submittal of the system design details, review of the adequacy of the modification in regard to operation of Units 2 and 3 will be made by the Regulatory Staff and the Committee.

The applicant proposes to provide flood protection for the plant which includes protection to elevation 578.0 feet for the reactor building, radwaste building, diesel-generator building, and the Residual Heat Removal Service Water pumps on the intake structure. This elevation represents a probable maximum flood elevation of 572.5 feet plus wave runup associated with a sustained wind of 45 miles per hour. The applicant states that all flood protection features for Unit 1 will be completed prior to operation of Unit 1 above 1% power. The Committee finds these provisions satisfactory.

For control of combustible gas concentrations in the containment following a postulated loss-of-coolant accident, the applicant proposes use of a containment atmospheric dilution (CAD) system. With this system the desired dilution is accomplished by controlled addition of nitrogen, and results in the maintenance of higher containment pressure during a portion of the post-LOCA period than would otherwise exist. The Committee believes that, in general, use of such dilution schemes, which involve repressurization of the containment, is not desirable. However, as a backfitted provision on a plant well along in construction, use of this approach is believed by the Committee to be acceptable. The Committee nevertheless recommends that the applicant study means to assure that the peak repressurization pressure will be limited to a value substantially below the containment design pressure.

The inservice inspection program proposed for the reactor coolant system complies with Section XI of the ASME Boiler and Pressure Vessel Code to the extent permitted by the existing design. The Committee believes the program is acceptable, but recommends that the applicant continue to study means of assuring reactor vessel integrity in regions currently inaccessible for inspection.

The applicant proposes to employ recirculation pump trip as a means of limiting the consequences of the unlikely occurrence of a failure to scram during an anticipated transient. The Committee believes that this recirculation pump trip represents a substantial improvement and should be provided for Unit 1 prior to the start of commercial power operation. The specific means employed for implementing the pump trip should be resolved in a manner satisfactory to the Regulatory Staff.

Other problems relating to large water reactors which have been identified by the Regulatory Staff and the ACRS and cited in previous ACRS reports, should be dealt with appropriately by the Regulatory Staff and the applicant as suitable approaches are developed.

The Advisory Committee on Reactor Safeguards believes that, if due regard is given to the items mentioned above, and subject to satisfactory completion of construction and pre-operational testing, there is reasonable assurance that the Browns Ferry Nuclear Plant Unit 1 can be operated at power levels up to 3293 MW(t) without undue risk to the health and safety of the public.

Sincerely yours,

C. P. Siess

C. P. Siess
Chairman

References Attached.

<u>References</u>

1. Tennessee Valley Authority letter dated September 23, 1970 transmitting FSAR, Volumes 1 through 5 to Browns Ferry Nuclear Power Station Units 1, 2 and 3

2. Amendments 10 through 40 to the License Application for Browns Ferry Nuclear Power Station Units 1, 2 and 3

APPENDIX C

ERRATA TO SAFETY EVALUATION REPORT

1. Table of Contents - add "4.11 Main Steam Line Isolation Valve Leakage"
2. p. 1 line 6 - change "340 acre site" to "840 acre site"
3. p. 23 line 16 - change "treatment are" to "treatment and are"
4. p. 29 line 8 - change "fuel pins with five . . ." to "fuel pins with four . . ."
5. p. 31 line 6 - change "MCHFR $\leq$ 1.9" to "MCHFR $\geq$ 1.9"

   line 7 change "power density" to "linear heat generation rate"
6. p. 32 line 3 under 3.3 Core Thermal and Hydraulic Power - change "core power density is 18.35 kw/ft" to "core power density is 50.7 kw/liter"
7. p. 52 first paragraph - change "The reactor building has an air recirculation system and a Standby Gas Treatment System (SGTS) to mix and filter primary containment leakage prior to its discharge to the environment" to "The reactor building has a Standby Gas Treatment System (SGTS) to filter primary containment leakage prior to its discharge to the environment."
8. p. 56 line 15 - change "entrainment" to "containment"
9. p. 56 line 21 - change "because at" to "because of"
10. p. 57 last paragraph - delete words "In addition to agreeing to meet the requirements of proposed Appendix J"
11. p. 68 line 2 under Auto-Depressurization System - delete "and safety"
12. p. 71 under 6.8 Discussion of ECCS Review line 8 - change "2090°F" to "1990°F"

    line 14 - change "1930°F to "1850°F"
13. p. 99 line 3 under Emergency Equipment Cooling Water System - change "heat exchanger, diesel generator" to "heat exchangers, diesel generators"

14. p. 122 line 16 - change "verified" to "verification"