October 12, 1982

Docket No. 50-219

NOTE FOR: NRC Public Document Room
Local Public Document Room

FROM: Hazel Smith

SUBJECT: DRAFT SEP REPORT (NUREG-0822) - OYSTER CREEK

Enclosed is a set of evaluations for each SEP topic referenced in the above NUREG. It has been requested that these documents be filed together for the convenience of the public in connection with its review of NUREG-0822.

Original signed by/

Hazel Smith, Licensing Assistant
Operating Reactors Branch #5
Division of Licensing

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As stated

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

September 25, 1980

Docket No. 50-219

Mr. I. R. Finfrock, Jr.
Vice President - Generation
Jersey Central Power & Light Company
Madison Avenue at Punch Bowl Road
Morristown, New Jersey 07960

Dear Mr. Finfrock:

RE: SEP TOPICS V-10.B, V-11.A, V-11.B, VII-3 and IX-3 (SAFE SHUTDOWN SYSTEMS) - OYSTER CREEK NUCLEAR GENERATING STATION, UNIT NO.1

Enclosed is a copy of our current evaluation of Safe Shutdown Systems (Revision 1) for Oyster Creek Nuclear Generating Station, Unit No. 1. This assessment compares your facility, as described in Docket No. 50-219 with the criteria currently used by the regulatory staff for licensing new facilities. Please inform us if your as-built facility differs from the licensing basis assumed in our assessment within 90 days of receipt of this letter.

This evaluation will be a basic input to the integrated safety assessment for your facility unless you identify changes needed to reflect the as-built conditions at your facility. This assessment may be revised in the future if your facility design is changed or if NRC criteria relating to this subject is modified before the integrated assessment is completed.

I am also enclosing Staff Positions regarding the SEP Safe Shutdown Systems review for your facility.

Sincerely,

Dennis M. Crutchfield

Dennis M. Crutchfield, Chief
Operating Reactors Branch #5
Division of Licensing

Enclosures:
1. Completed SEP Topics - Safe Shutdown Systems
2. Staff Positions

cc w/enclosures:
See next page

~~PDR 8010310532~~

ENCLOSURE 1

SEP REVIEW

OF

SAFE SHUTDOWN SYSTEMS

FOR THE

OYSTER CREEK NUCLEAR POWER PLANT

REVISION 1

TABLE OF CONTENTS

1.0 INTRODUCTION

The Systematic Evaluation Program (SEP) review of the "safe shutdown" subject encompassed all or parts of the following SEP topics, which are among those identified in the November 25, 1977 NRC Office of Nuclear Reactor Regulation document entitled "Report on the Systematic Evaluation of Operating Facilities":

1. Residual Heat Removal System Reliability (Topic V-10.B)
2. Requirements for Isolation of High and Low Pressure Systems (Topic V-11.A)
3. RHR Interlock Requirements (Topic V-11.B)
4. Systems Required for Safe Shutdown (Topic VII-3)
5. Station Service and Cooling Water Systems (Topic IX-3)

The review was primarily performed during an on-site visit by a team of SEP personnel. This on-site effort, which was performed on August 8 & 9, 1978, afforded the team the opportunity to obtain current information and the licensee (Jersey Central Power & Light Company) the opportunity to provide input into the review.

The review included specific system and equipment requirements for remaining in a shutdown condition (defined in the Oyster Creek Technical Specifications as the reactor mode switch-being in the shutdown mode position) and for proceeding to a cold shutdown condition (defined as mode switch in shutdown mode position, all operable control rods fully inserted, and the reactor coolant system maintained at less than 212°F and vented). The review for transition from operating to shutdown considered the requirement that the capability exists to perform this operation from outside the control room. The review was augmented as necessary to assure resolution of the applicable topics, except as noted below:

Topic V-11.A (Requirements for Isolation of High and Low Pressure Systems) was examined only for application to the Shutdown Cooling System. Other high pressure/low pressure interfaces were not investigated. The Shutdown Cooling System is the Oyster Creek equivalent of an RHR system.

Topic VII-3 (Systems Required for Safe Shutdown) was completed except for determination of design adequacy of the systems.

Topic IX-3 (Station Service and Cooling Water Systems) was only reviewed to consider redundancy and seismic and quality classification of cooling water systems that are vital to the performance of safe shutdown system components. (No discussion of Topic IX-3 is included in this report. The information gathered during the safe shutdown review will be used to resolve this topic later in the SEP.)

The criteria against which the safe shutdown systems and components were compared in this review are taken from the: Standard Review Plan (SRP) 5.4.7, "Residual Heat Removal (RHR) System"; Branch Technical Position RSB 5-1 Rev. 1, "Design Requirements of the Residual Heat

Removal System" and, Regulatory Guide 1.139, "Guidance for Residual Heat Removal". These documents represent current staff criteria and are used in the review of facilities being processed for operating licenses.

This comparison of the existing systems against the current licensing criteria led naturally to at least a partial comparison of design criteria, which will be input to SEP Topic III-1, "Classification of Structures, Components and Systems (Seismic and Quality)". This report will also be reviewed for its application to the resolution of other topics.

As noted above, the five topics were examined while neglecting possible interactions with other topics and other systems and components not directly related to safe shutdown. For example, Topics II-3.B (Flooding Potential and Protection Requirements), II-3.C (Safety-Related Water Supply), III-4.C (Internally Generated Missiles), III-5.A (Effects of Pipe Break on Structures, Systems, and Components Inside Containment), III-6 (Seismic Design Considerations), III-10.A (Thermal-Overload Protection for Motors of Motor-Operated Valves), III-11 (Component Integrity), III-12 (Environmental Qualification of Safety-Related Equipment) and V-1 (Compliance with Codes and Standards) are among several topics which could be affected by-the results of the safe shutdown review or could have a safety impact upon the systems which were reviewed. These effects will be determined by later review. Further, this review did not cover in any significant detail the reactor protection system nor the electrical power distribution, both of which will also be reviewed later.

The staff considers that the ultimate decision concerning the safety of any of the SEP facilities depends upon the ability to withstand the Design Basis Events (DBEs). The SEP topics provide a major input to the SEP DBE review, both from the standpoint of assessing the probability of the event and that of determining the consequences of the event. As examples, the safe shutdown topics pertain to the listed DBEs (the extent of applicability will be determined during plant-specific review):

Topic	DBE Group	Impact Upon Probability Or Consequences of DBE
V-10.B	VII (Spectrum of Loss of Coolant (Accidents)	Consequences
V-11.A	VII (Defined above)	Probability
V-11.B	VII (Defined above)	Probability
VII-3	All (Defined as a generic topic)	Consequences
IX-3	III (Steam Line Break Inside Containment) (Steam Line Break Outside Containment)	Consequences
	IV (Loss of AC Power to Station Auxiliary) (Loss of all AC Power)	Consequences
	V (Loss of Forced Coolant Flow) (Primary Pump Rotor Seizure) (Primary Pump Shaft Break)	Probability
	VII (Defined above)	Consequences

Completion of the safe shutdown topic review (limited in scope as noted above), as documented in this report, provides significant input in assessing the existing safety margins at Oyster Creek.

Piping System Passive Failures

The NRC staff normally postulates piping system passive failures as 1) accident initiating events in accordance with staff positions on piping failures inside and outside containment, 2) system leaks during long term coolant recirculation following a LOCA, and 3) failures resulting from hazards such as earthquakes, tornado missiles, etc. In this evaluation, certain piping system passive failures have been assumed beyond those normally postulated by the staff, e.g., the catastrophic failure of moderate energy systems. These a sumptions were made to demonstrate safe shutdown system redundancy given the complete failure of these systems and to facilitate future SEP reviews of DBEs and other topics which will use the safe shutdown evaluation as a source of data for the SEP facilities. SRP 5.4.7 and BTP RSB 5-1 do not require the assumption of piping system passive failures.

Credit for Operating Procedures

For the safe shutdown evaluation, the staff may give credit for facility operating procedures as alternate means of meeting regulatory guidelines. Those procedural requirements identified as essential for acceptance of an SEP topic or DBE will be carried through the eview process and considered in the integrated assessment of the facility. At that time, we will: (1) decide which procedures are so important that they should be included in technical specifications and (2) establish an administrative procedure (e.g., FSAR changes) for ensuring that the other operating procedures are not changed without appropriate consideration of their importance to the topic or DBE evaluations.

2.0 DISCUSSION

2.1 Normal Plant Shutdown and Cooldown

Power is reduced from its operating value during commencement of shutdown by first simultaneously reducing recirculation flow in all loops to a specified value, about 50,000 gpm. Reactor coolant system pressure is controlled by the electrical pressure regulator/mechanical pressure regulator (EPR/MPR) and maintained between 980 and 1020 psig to the hot shutdown condition. Thus, as power is reduced, the turbine control valves are correspondingly closed to "hold" reactor pressure. After achieving the desired recirculation flow, control rods are selectively inserted in a prescribed pattern to continue the power decrease. Power is reduced at a rate compatible with the load dispatcher's requirements.

Feedwater heaters are removed from service when power reaches about 200 MWe. Feedwater is controlled in "automatic" with the master controller until high flow is no longer needed. As steam flow is reduced, feedwater matches and when all 3 pumps are no longer needed, a feedwater train is placed on its individual flow controller and manually controlled until flow is stopped and the train is secured. Flow is reduced until one pump operates and its controller is placed in "manual", controlling via the low flow control valve around the main feedwater control valve when flow reaches about 1000 gpm. Finally, during cooldown when the last operating feed pump is no longer needed for reactor water level control, the low flow control valve is fully closed and the pump is tripped. Then as RCS pressure is lowered, the vessel can be fed via a condensate pump. While shutting down, water continues to be added, about 70 gpm, to the RCS inventory by the control rod drive (CRD) hydraulic pumps.

When power reaches about 100 MWe, the station electrical feed is switched from the auxiliary transformer (station generator) to the startup transformers. Power is reduced further and the turbine generator is removed from service. With the turbine no longer extracting energy, steam is bypassed to the main condenser at very low power and the heat is transferred to the circulating bay water. The reactor is at pressure (>860 psig) and critical at low power, minimum RCS recirculation flow is on, steam is being bypassed to the condenser with a feedwater train in service and thus the reactor is at hot standby.

Cooldown is now accomplished, if desired, by continuing with control rod insertion and with a feedwater train on controlling reactor water level in "manual" via the lowflow valve and bypassing steam to the main condenser. This is continued, establishing a cool down rate not to exceed 100°F/hr. or a metal to flange ΔT (vessel or head) of 200°F. When RCS temperature reaches 350°F, the Shutdown Cooling System (SCS) is placed in service. Reactor Building Closed Cooling Water (RBCCW) flow to the S' heat exchangers is established and service water flow to the RBCCW heat exchangers is already established thereby creating the heat transport path to the bay.

Normally all recirculation pumps continue to run until vessel cooldown is complete then shutdown as desired. When all control rods are fully inserted, RCS temperature is less than 212°F, the mode switch is in "shutdown", and the reactor is in cold shutdown.

2.2 Shutdown and Cooldown With Loss of Offsite Power

On loss of offsite power the main condenser is unavailable for heat removal following reactor trip. The reactor can stay in the hot condition briefly while pressure is controlled with relief valves. The two isolation condensers activate on sustained high RCS pressure or may be "manually" activated. The single closed valve in each condenser system, in the condensate return lines, is opened and main steam passes through the isolation condenser tubes and boils off the secondary side water in the condenser. Makeup water is provided to the condensers from the condensate storage tank by transfer pumps powered from onsite sources or by the fire protection system using diesel fire pumps. The reactor is cooled by the isolation condenser until the SCS interlock temperature is reached. The SCS may then be put in service as above since it, the RBCCW, and Service Water Systems are powered by onsite electrical sources. Cooldown is accomplished as described in Section 2.1.

If the isolation condensers were unavailable, depressurization of RCS by operation of relief valves and activation of core spray at the lowered pressure would provide an alternate means of decay heat removal and cooldown to the cold shutdown condition.

3.0 SHUTDOWN AND COOLDOWN FUNCTIONS AND METHODS

This section will describe the existing systems available at Oyster Creek to accomplish the necessary functions for the safe shutdown of the reactor following either the loss of offsite power or the loss of onsite AC power. Seismic and Quality Group Classifications of the pertinent equipment (based upon USNRC Regulatory Guides 1.26 and 1.29) will be discussed in Section 4.0. The minimum list of safe shutdown systems is also provided in Section 4.0.

The losses of offsite and onsite AC power are not considered to be concurrent or sequential events, but rather, for the purposes of this evaluation, are taken as wholly independent occurrences.

Offsite power is supplied for startup through startup transformers SA and SB, each of which is supplied from a separate substation. These transformers are capable of supplying all electrical auxiliaries with the main generator producing full power. When the turbine-generator output reaches approximately 50 MWe during the startup, electrical power for "house" loads is transferred from the startup transformers to the auxiliary transformer, which has one primary winding and two secondary windings. Should this transformer fail (loss of onsite power) power is automatically transferred back to the startup transformers, preserving power to essential auxiliaries and minimizi j the consequences of a loss of onsite power. Transfer to either startup transformer will also occur upon loss of its "companion" secondary winding on the auxiliary transformer.

Note that the above discussion omitted use of the diesel generators for supplying power to the 4160V emergency switchgear buses 1C and 1D (which in turn supply 460V unit substations carrying additional essential electrical loads). Either diesel is capable of supplying sufficient power for the safe shutdown of the reactor. Additionally, although the total and simultaneous loss of offsite and onsite AC power (including diesel generators) is considered an extremely low probability event, Jersey Central Power and Light Company (JCP&L) management has prepared for such an occurrence by providing a procedure to be followed by plant operators in the event of complete loss of AC power.

Assuming a total loss of offsite power (this has never occurred at Oyster Creek) with the reactor at full power (1930 MWt, 650 MWe), a reactor scram would follow due to interruption of the protection system power supply. At 1050 psig, two electromagnetic relief valves (DC-powered) would lift to relieve the pressure. Each valve is rated at 600,000 pounds/hour capacity. At 1060 psig, the two isolation condenser systems automatically would initiate after a time delay of less than 3 seconds, rapidly decreasing reactor coolant system pressure as natural circulation flow from the reactor through the condensers returns cold water to the reactor vessel.

Each isolation condenser shell contains a minimum (as per Technical Specifications) of 22,730 gallons, which represents 11,060 gallons of water above the tubes. Both condensers in operation can absorb reactor decay heat for one (1) hour and forty (40) minutes without replenishment of the water supply, while one (1) condenser alone can remove decay heat for forty-five minutes without replenishment. These figures are based upon an analyzed scram from 1950 MWt, which is greater than maximum allowable power and is therefore conservative. Makeup to the isolation condensers is provided by either the condensate transfer system (normal source) or the fire protection system. The condensate transfer pumps (two) receive their power from the same "vital" bus. Power can be supplied by the diesel generators, but the pumps will not start automatically and must be brought onto the diesel bus manually. Even in the unlikely event that these pumps fail, the two diesel-driven fire pumps can be utilized. These diesels have separate battery supplied starter systems and can be relied upon to provide makeup water. All valves in the makeup path from either source (condensate or firewater) are local manual valves with the exception of the inlet valves to the condensers themselves. These valves, which are normally air-operated remote-manually from the control room, can easily be overridden locally to provide makeup flow to the condensers.

Each isolation condenser is provided steam inlet from its own single-use nozzle in the reactor vessel. This line contains, in series, two motor-operated valves (one AC, one DC) which are maintained open during operation at reactor coolant temperature greater than 212°F (Technical Specifications allow one isolation condenser to be out of service for a period not to exceed seven days, provided augmented surveillance of the operable condenser is performed). Thus, the piping and isolation condenser tubes are always pressurized at reactor pressure. The return line from each condenser to the reactor (one to recirculation loop A, one to loop E) contains two valves, the first of which is DC-powered and is normally closed (opened only to initiate flow), the second of which is AC-powered, normally open, and is the only one of the four valves of each condenser system to be inside containment.

Power to the two AC-powered valves for each condenser is provided from vital motor control center (MCC) 1AB2, which can be powered from either emergency diesel generator through an automatic bus transfer switch to either motor control center 1A2 or 1B2. The adequacy of this electrical arrangement will be further examined under SEP topic VII-7. Vital MCC 1AB2 is important in that it powers not only the isolation condenser AC valves, but also provides power to shutdown cooling system and core spray system valves, among others. The failure of this MCC will have no effect upon the normal operation of the isolation condensers since the AC valves are normally open and will fail "as is" upon loss of electrical power.

Each condenser will be totally isolated from the reactor, and thus inoperable, with closure of all valves (two AC and two DC), upon receipt of a high flow signal from sensors in its own steam supply and/or condensate return lines. Actuation of the sensors, and subsequent isolation, has occurred in the past upon initiation of isolation condenser flow. This condition was apparently due to the sensitivity of the flow sensors and has since been corrected, with the result that during the one condenser initiation event since the alteration no problem occurred.

The isolation of both condensers upon initiation, already an unlikely event after the modification mentioned above, would only become a problem with the highly unlikely concurrent loss of MCC 1AB2. Loss of this MCC would result in the inability to open the one valve in each system which is located inside containment and is thus inaccessible. Jersey Central Power & Light Co. is prepared for this unusual circumstance by having provided a procedure for reactor shutdown with attendant loss of normal reactor cooling mechanisms, including loss of the isolation condensers. This is discussed later.

Power to the two DC-operated valves for condenser A and the DC-operated inlet valve for condenser B is provided from MCC DC-1. Power to the DC-operated outlet valve of condenser B is provided by MCC DC-2. MCC DC-2 receives power from DC distribution center C. MCC DC-1 receives power from DC distribution center A or B through an automatic transfer switch. All other DC isolation valves, such as those for the shutdown cooling system, are on this MCC. Modifications to the DC electrical distribution center have been completed and are under review by the NRC staff. These modifications are described in the licensee's letter of April 4, 1978 (Reference 4).

Assuming that one or both isolation condenser systems function properly, the next system of concern during a shutdc ı following loss of offsite power is the shutdown cooling system. This system has a single suction line from recirculation loop E and a single discharge line to recirculation loop E. Very little of this system is inside containment.

However, two AC motor-operated isolation valves, one on the suction side, the other on the discharge, are located inside containment, receive power from vital MCC 1AB2, and upon the unlikely loss of the MCC would be inoperable. This was also noted above in the discussion of the isolation condenser systems. Also, a single failure of one of these valves to open would result in inoperability of the entire shutdown cooling system.

Outside the drywell, the shutdown cooling system branches into three headers, each containing (as major equipment) a DC motor-operated suction valve, pump, heat exchanger, and a DC-motor-operated discharge valve. These headers then return to a common line through the AC discharge valve mentioned above. (Thus diversity of isolation power is provided). The DC power-operated valves are powered from MCC DC-1. This system was designed for 1250 psig (Reactor Coolant System pressure) at 350°F, which is less than reactor temperature. However, it would take multiple failures of the valves (all of which are normally shut) and interlocks to initiate flow at temperatures greater than 350°F. The interlocks prevent AC valve opening at temperatures greater than 350°F and will isolate the system if it is in operation, upon reaching 350°F. Each of the five recirculation loops must be less than 350°F to satisfy this interlock.

Additionally, each pump is interlocked such that starting is prohibited unless suction pressure exceeds 4 psi and temperature in the suction line is less than 350°F.

In order to accommodate decay heat after shutdown and cooldown to its initiation limits, the shutdown cooling system requires only two of the three branches (i.e., two pumps and two heat exchangers) to be in service.

Cooling flow to the shutdown cooling system heat exchangers is provided by the Reactor Building Closed Cooling Water (RBCCW) system which is in turn cooled by the Service Water (SW) system. Only one of two RBCCW pumps and heat exchangers need be in service to provide shutdown cooling, provided no RBCCW flow is being supplied to the reactor water cleanup (RWCU) system non-regenerative heat exchanger and the five reactor coolant recirculation pumps.

Power to shutdown cooling pump A is provided from 460 volt unit substation 1A2, with pumps B and C being supplied from substation 1B2. Upon loss of offsite power, these substations are provided power from the emergency diesel generators.

The RBCCW system, which provides cooling water to the shutdown cooling system, consists of two pumps and two heat exchangers, which are in turn cooled by the SW system as noted above. Power to the pumps is provided from 460 volt unit substations 1A2 (pump 1-1) and 1B2 (pump 1-2). These substations are supplied by the emergency-diesels, assuring that even in the event of single bus or single diesel failure, one pump will be available.

The RBCCW system has a common suction line for both pumps and a common discharge line from the heat exchangers. Passive failure of either line would result in loss of RBCCW and subsequent heatup of the reactor coolant to the point of shutdown cooling system isolation. Alternate means for core cooling for this circumstance are discussed below.

There is only one motor-operated valve which isolates RBCCW flow from the shutdown cooling system. This valve, which is AC-powered, is on the common discharge of the three shutdown cooling heat exchangers. It is outside containment and is accessible for manual operation. The need for such operation is unlikely, since valve power is supplied from MCC 1B21A, which can be supplied from emergency diesel generator number 2.

The RBCCW system is cooled by the SW system, which consists of two pumps and associated valving and piping. The SW system provides flow primarily to the RBCCW system heat exchangers and is also utilized to maintain the emergency service water system (for containment spray heat exchanger cooling) filled. However, it is also an alternate means of cooling for the turbine building closed cooling water heat exchangers and is used as such when main condenser circulating water pumps are not operating. All valves in the SW system are manually operated, and power to the pumps is provided by 460 volt unit substations 1A3-(pump 1-1) and 1B3 (pump 1-2). These substations are powered respectively from 4160 volt emergency switchgear buses 1C and 1D, which are provided power from the diesel generators (DG1 to 1C; DG2 to 1D) in an emergency.

The shutdown cooling system operates at a higher pressure than the RBCCW, which in turn is at a higher pressure than the SW system. A leak in both the shutdown cooling and RBCCW heat exchangers and a failure to take proper action would be required to release radioactivity to the environment. The RBCCW and SW systems both incorporate radiation detectors to alert operators to leakage, and the RBCCW system includes a surge tank with high and low level alarms.

The only significant failure mode of the SW system would be passive failure of the common pump discharge header, which would entirely disable the system. As with the loss of RBCCW discussed before, this subject will be treated below.

The above discussion has dealt with systems which would normally be used for the safe shutdown of the reactor upon loss of offsite power. There are, of course, alternate means utilizing other equipment, which will permit safe shutdown should any or all of the above-mentioned systems fail.

As a review of the above, loss of offsite power results in turbine trip, reactor scram, loss of condenser circulating water pumps (rendering the condensers useless for cooling), and loss-of feedwater and condensate pumps. In such a situation, the redundant isolation condenser systems would be relied upon to reduce reactor pressure and temperature to the point of shutdown cooling initiation.

However, even when hypothesizing the failure of both independent isolation condensers, Oyster Creek has sufficient capability in the reactor pressure relief system to assure depressurization. There are five electromatic relief valves (EMRVs) which are located on the main steam lines in the drywell and which discharge into the pressure suppression pool (torus). Although these valves would function automatically to relieve pressure (two valves lift at 1050 psig, three at 1070 psig), they can be actuated by remote-manual means from the control room. They are DC-powered and receive power from either 125 volt DC power panel D (battery room) or panel F (460 switchgear room).

As is noted in a JCP&L Oyster Creek procedure concerning loss of reactor cooling mechanisms during reactor shutdown, the control rod drive hydraulic system may be used to maintain vessel level while maintaining only enough blowdown to limit reactor pressure to acceptable levels. However, as noted in another procedure concerning complete loss of AC power, the EMRVs may be opened to reduce vessel pressure and temperature to the levels at which fire water to the core spray system can be utilized, assuming shutdown cooling and core spray were not available (which is not the case here). Oyster Creek FDSAR Amendment 10 states that the automatic depressurization system would function upon receiving the appropriate low low low reactor water level, high drywell pressure, and core spray booster pump discharge pressure signals. This system will allow depressurization in sufficient time to add a substantial amount of core spray and prevent fuel clad temperature from exceeding 2200°F. FDSAR figure VI-6-6 does show that the core is temporarily and partially uncovered.

Under the circumstances of this analysis we have assumed, in addition to the loss of offsite power, the loss of both isolation condensers. In this case, the operator, as noted above, could choose to remain at hot standby, maintaining level with the control rod drive system while relieving pressure through the relief valves. If plant conditions dictated the need to immediately decrease pressure and cool the system, the use of the relief valves would serve this purpose, and would probably accomplish the necessary depressurization prior to uncovering the top of the core. However, even were the level to decrease to the low-low-low setpoint prior to blowdown initiation, the FDSAR analysis concludes that no clad melting would occur. We find the temporary and partial uncovery of the core, in this scenario, to be an acceptable event, given first that we have assumed multiple failures to achieve the scenario and second that no fuel melting would occur, as previously calculated, since a large influx of cooling water would be available upon completion of the depressurization.

Torus cooling may be desired by the operator utilizing the containment spray and emergency service water systems. The pumps and motor-operated valves of these systems are powered from sources which can be supplied by the emergency diesel generators. As was discussed before, Oyster Creek has the capability, through use of either isolation condenser, to remain hot while removing core decay heat upon loss of offsite AC power. The isolation condensers may also be used to cool down. The discussion immediately above shows that there is redundant capability for depressurization and cooldown, utilizing the EMRV's and the core spray/fire water systems.

Taken to an extreme, the method above (EMRVs and Core Spray) could function as a closed loop by filling the vessel with Core Spray, overflowing hot water to the pressure-suppression chamber (torus) through the relief valves. The torus provides water to the Core Spray system and cooling for such water would be provided by the containment spray system.

The cycling of the water through the core and through the relief valves to the torus and back again would only be limited by the design of the relief valves themselves. These valves incorporate a spring which must be overridden by system pressure. The spring will shut the valve at approximately 50 psig (FDSAR Page VI-6-1) and will hold it shut until the core heats up again and raises pressure or until pressure is increased by the Core Spray pumps.

The core spray system contains four main pumps and four booster pumps. Each of the main pumps provides 1700 gpm each to the reactor. These pumps and all motor-operated valves of the system are powered from emergency AC buses. Like the other systems discussed above, the component trains and their power supplies are arranged such that failure of one emergency diesel generator will not disable the entire system.

The core spray system is interlocked such that it will not provide water to the reactor until reactor pressure has decreased to 285 psig.

However, the system automatically initiates upon a low-low reactor water level signal, with a minimum flow line returning flow to the torus until the pressure interlock is satisfied and the system discharge valves open. Although the torus (nominally 85,000 cubic feet chromated water) is the preferred source of supply, water can also be drawn from the condensate storage tank (250,000 gallons minimum) or the demineralized water tank (30,000 gallons). Vessel level can be maintained by manual operation of easily-accessible valves, if required.

If, for any reason, the core spray pumps are inoperable, fire water system flow can be supplied to core spray and the reactor kept cooled. The interconnection valves are readily-accessible for manual operation, flow is provided by diesel-driven pumps which are battery-started, and JCP&L has even provided a procedure which includes using a fire truck from a local fire department to pressurize the core spray system.

As an alternative to the core spray system, the shutdown cooling system could be utilized to provide cooling after the EMRV depressurization. RBCCW and SW systems would then be necessary. If either of these systems fail, rendering shutdown cooling inoperable, a means still exists to maintain the core covered and cooled. Letdown from the reactor can be accomplished through the reactor water cleanup demineralizer system, although a temperature interlock to prevent resin damage would have to be disabled. Flow of cool water to the vessel could be obtained through core spray, as noted above. The letdown flow from the cleanup demineralizer could be routed to the condensate storage tank or to the radioactive waste processing system.

If the RBCCW and SW systems are operable but shutdown cooling is not, some cooling could be maintained by increasing RBCCW flow to the cleanup demineralizer system's non-regenerative heat exchanger. This capability has been accounted for in an Oyster Creek procedure.

CONCLUSION

As can be readily seen from the foregoing discussion, Oyster Creek has the ability to withstand multiple failures and still retain the capability to depressurize and cool the reactor core. Problems with systems to be primarily relied upon have been noted, as was the yet unreviewed change to the 125 volt DC system.

We are satisfied that Oyster Creek can be safely shutdown upon loss of onsite or offsite AC power, even considering failure of a single major component.

4.0 COMPARISON OF SAFE SHUTDOWN SYSTEMS WITH CURRENT NRC CRITERIA

The current criteria used in the evaluation of the design of systems required to achieve cold shutdown for a new facility are listed in the Standard Review Plan (SRP) Section 5.4.7 and Branch Technical Position RSB 5-1 (or proposed Regulatory Guide 1.139). This section discusses the comparison of these criteria with the safe shutdown systems of the Oyster Creek nuclear power plant. This comparison will be done by quoting a section of the Branch Technical Position RSB 5-1 and then discussing the degree to which Oyster Creek meets the requirements of that particular section.

"A. Functional Requirements
The system(s) which can be used to take the reactor from normal operating conditions to cold shutdown* shall satisfy the functional requirements listed below.

1. The design shall be such that the reactor can be taken from normal operating conditions to cold shutdown* using only safety-grade systems. These systems shall satisfy General Design Criteria 1 through 5.

2. The system(s) shall have suitable redundancy in components and features, and suitable interconnections, leak detection, and isolation capabilities to assure that for onsite electrical power system operation (assuming offsite power is not available) and for offsite electrical power system operation (assuming onsite power is not available) the system function can be accomplished assuming a single failure.

3. The system(s) shall be capable of being operated from the control room with either only onsite or only offsite power available with an assumed single failure. In demonstrating that the system can perform its function assuming a single failure, limited operator action outside of the control room would be considered acceptable if suitably justified.

*Processes involved in cooldown are heat removal, depressurization, flow circulation, and reactivity control. The cold shutdown conditions, as described in the Standard Technical Specifications, refers to a subcritical reactor with a reactor coolant temperature no greater than 200°F for a PWR and 212°F for a BWR.

4. The system(s) shall be capable of bringing the reactor to a cold shutdown condition, with only offsite or onsite power available, within a reasonable period of time following shutdown, assuming the most limiting single failure."

Background

A "safety grade" system is defined, in the NUREG 0138 (Reference 1) discussion of issue #1, as one which is designed to seismic Category I (Regulatory Guide 1.29), quality group C or better (Regulatory Guide 1.26), and is operated by electrical instruments and controls that meet Institute of Electrical and Electronics Engineers Criteria for Nuclear Power Plant Protection Systems, (IEEE 279). Oyster Creek nuclear power plant was constructed prior to the issuance of Regulatory Guides 1.26 and 1.29 (as Safety Guides 26 and 29 on 3/23/72 and 6/7/72 respectively). Also Proposed IEEE 279, dated August 30, 1968, was issued late in the construction phase of the facility.

General Design Criteria 1 requires that these systems be designed, fabricated, erected and tested to quality standards, that a quality assurance (QA) program be implemented to assure that these systems perform their safety functions and that an appropriate record of design, fabrication, erection and testing be kept. At the time that Oyster Creek was licensed, the NRC (then AEC) criteria for QA were under development. Since that time, various QA related regulations and criteria have been instituted by the NRC, and the QA program for operation of the plant was approved by the staff on November 5, 1976.

The Oyster Creek Technical Specification and QA program require appropriate QA records to be kept.

General Design Criterion 2 requires that structures and equipment important to safety be designed to withstand the effects of natural phenomena without loss of capability to perform their safety function.

The original Staff SER (Reference 2) states that the Oyster Creek power plant can safely survive a flood level of 23 feet above mean sea level (MSL). The maximum flood height at the Oyster Creek site has been 4.5 ft.

The licensee's seismic design bases specify that for a ground acceleration of 0.22g, there will be no loss of function of critical structures and components necessary to ensure a safe and orderly shutdown.

These seismic design bases were approved by the Staff in the original SER and will be re-reviewed as part of several SEP tasks.

General Design Criterion 3 requires that structures, systems and components important to safety be designed and located to minimize the effects of fires and explosions.

The Staff has completed an evaluation of the fire safety requirements of the Oyster Creek nuclear power plant. The results of this evaluation are given in Reference 3.

General Design Criterion 4 requires that equipment important to safety be designed to withstand the effects of environmental conditions for normal operation, maintenance, testing and accidents. Equipment should also be protected against dynamic effects such as internal and external missiles, pipe whip and fluid impingement.

The SEP will evaluate the extent to which Oyster Creek conforms to GDC 4 when reviewing topics III-12, "Environmental Qualification of Safety-Related Equipment," III-5.A, "Effects of Pipe Breaks Inside Containment," III-5B, "Pipe Breaks Outside Containment," and III-4, "Missile Generation and Protection."

General Design Criterion 5 relates to the sharing of structures, systems and components important to nuclear safety among nuclear units. Since the Oyster Creek nuclear plant is the single unit at the site, GDC 5 does not apply.

The BTP RSB 5-1 functional requirements focus on the safety grade systems that can be used to take the reactor from operating conditions to cold shutdown. The staff and licensee developed a "minimum list" of systems necessary to perform this task. Although other systems may be used to perform shutdown and cooldown functions, the following list is the minimum number of system required to fulfill the BTP RSB 5-1 criteria:

1. Reactor Control and Protection System
2. Isolation Condensers
3. Condensate Transfer System (for isolation condenser makeup)
4. Electromatic Relief Valves (all 5 of which consititute the Automatic Depressurization System of the ECCS)
5. Core Spray System
6. Emergency Service Water System and Containment Spray System (for containment cooling)
7. Instrumentation for shutdown and cooldown*
8. Emergency Power (AC and DC) and control power for the above systems and equipment.

In addition to these systems, other systems may function as backup for the above systems and components. The preceding discussion in Section 3 described both these systems and the systems which may function as backup. Table 4.1 lists the minimum safe shutdown systems for the Oyster Creek Nuclear Power Plant along with the comparison of present criteria with the criteria to which these components and subsystems were designed. Table 4.3 provides the power supplies and locations of these systems.

4.1 Functional Requirements

The Reactor Control and Protection System (RCPS) is designed on a channelized basis to provide physical and electrical isolation between

*Safe shutdown instruments are identified in Table 4.2.

redundant reactor trip channels. Each channel is functionally independent of every other channel and receives power from two independent sources. The power source for the RCPS is the instrument buses which can receive power from either onsite or offsite sources. The RCPS fails safe (tripped) on loss of power. The system can be manually tripped both from the control room and from other locations outside the control room. The RCPS is designed so that a single failure will not prevent a reactor trip. Initiation of a reactor trip causes the insertion of sufficient reactor control rods to make the core subcritical from any credible operating condition assuming the most reactive control rod remains in the fully withdrawn position.

The design of the RCPS, as well as safe shutdown related electrical control and power systems, will be evaluated later in the SEP.

The normal shutdown systems and alternate systems have been reviewed in Section 3. The isolation condensers would be relied upon for cooling from full power conditions upon loss of the main condenser which is not available upon loss of offsite power. The isolation condensers are capable of cooling the reactor to near cold shutdown conditions. After reactor system pressure is reduced to the cut-in pressure of the core spray system, this system could be manually initiated and would take the reactor to cold shutdown conditions. The reactor can be maintained in cold shutdown conditions using the core spray, ADS, emergency service water, and containment spray systems.

The functional requirement to achieve cold shutdown conditions within a reasonable period of time is evaluated in Appendix A.

4.2 RHR System Isolation Requirements

The RHR system shall satisfy the isolation requirements listed below.

1. The following shall be provided in the suction side of the RHR system to isolate it from the RCS.

   (a) Isolation shall be provided by at least two power-operated valves in series. The valve positions shall be indicated in the control room.

   (b) The valves shall have independent diverse interlocks to prevent the valves from being opened unless the RCS pressure is below the RHR system design pressure. Failure of a power supply shall not cause any valve to change position.

   (c) The valves shall have independent diverse interlocks to protect against one or both valves being open during an RCS increase above the design pressure of the RHR system.

The purpose of these requirements is to provide assurance that a low pressure shutdown cooling system will not be exposed, either through a single operator error or failure of a single valve, to a pressure greater than design pressure.

The Oyster Creek Shutdown Cooling System is designed, as stated in Section 3, for reactor coolant system design pressure, 1250 psig. The design temperature is 350°F which is lower than the reactor coolant system design temperature (575°F). The licensee is evaluating the ability of the SCS to withstand exposure to these high temperature conditions on a one-time basis; however, as pointed out in Section 3, multiple failures of valves (all of which are normally shut) and interlocks would be necessary in order for this situation to exist.

Section 3 described the interlock on the RHR system which prevents opening of the suction and discharge valves on the SCS if the reactor coolant temperature in any of the five coolant loops is greater than 350°F. Also noted was the fact that system isolation will occur upon increase in temperature to 350°F. Although redundant diverse pressure interlocks do not control the SCS isolation valves, the temperature interlock on the AC valves and the high design pressure of the SCS provide adequate protection for the SCS.

The valves are motor operated and would fail in their "as-is" condition (which would be closed unless the SCS were in operation).

Thus, the Oyster Creek SCS acceptably meets the present criteria for SCS system isolation.

2. One of the following shall be provided on the discharge side of the RHR system to isolate it from the RCS:

   (a) The valves, position indicators, and interlocks described in item 1 (a) - (c).

   (b) One or more check valves in series with a normally closed power-operated valve. The power-operated valve position shall be indicated in the control room. If the RHR system discharge line is used for an ECCS function the power-operated valve is to be opened upon receipt of a safety injection signal once the reactor coolant pressure has decreased below the ECCS design pressure.

   (c) Three check valves in series, or

(d) Two check valves in series, provided that there are design provisions to permit periodic testing of the check valves for leak tightness and the testing is performed at least annually.

The isolation on the discharge side of the SCS is identical to that on the suction side, and, as discussed above, adequately meets the present criteria for SCS system isolation.

4.3 Pressure Relief Requirements

The RHR system shall satisfy the pressure relief requirements listed below.

1. To protect the RHR system against accidental overpressurization when it is in operation (not isolated from the RCS), pressure relief in the RHR system shall be provided with relieving capacity in accordance with the ASME Boiler and Pressure Vessel Code. The most limiting pressure transient during the plant operating condition when the RHR system is not isolated from the RCS shall be considered when selecting the pressure relieving capacity of the RHR system. For example, during shutdown cooling in a PWR with no steam bubble in the pressurizer, inadvertent operation of an additional charging pump or inadvertent opening of an ECCS accumulator valve should be considered in selection of the design bases.

2. Fluid discharged through the RHR system pressure relief valves must be collected and contained such that a stuck open relief valve will not:

   a. Result in flooding of any safety-related equipment.

   b. Reduce the capability of the ECCS below that needed to mitigate the consequences of a postulated LOCA.

   c. Result in a non-isolable situation in which the water provided to the RCS to maintain the core in a safe condition is discharged outside of the containment.

3. If interlocks are provided to automatically close the isolation valves when the RCS pressure exceeds the RHR system design pressure, adequate relief capacity shall be provided during the time period while the valves are closing.

The SCS relief valves discharge to the reactor building equipment drain tank (RBEDT). Overflow from this tank is directed to the reactor building drain sump. Therefore, a stuck-open relief valve would not result in the flooding of any safety related equipment. A high level alarm in the RBEDT would alert the operator to a potential problem which could be then diagnosed and corrected.

At Oyster Creek the SCS is independent of the ECCS. Therefore a failure of the SCS would not affect the ECCS.

Since the Shutdown Cooling System is designed for reactor design pressure, the reactor vessel relief valves will provide adequate SCS overpressure protection.

4.4 Pump Protection Requirements

The design and operating procedures of any RHR system shall have provisions to prevent damage to the RHR system pumps due to overheating, cavitation or loss of adequate pump suction fluid.

The SCS pumps are provided with bypass lines which return the pump discharge flow to the pump suction. Thus, even if the downstream valve were closed while the pump was running, the pump would be protected from overheating.

Cavitation protection is provided by the interlock which trips the pump if the suction pressure falls below 4 psig. The pump also trips on a coolant temperature greater than 350°F.

4.5 Test Requirements

The isolation valve operability and interlock circuits must be designed so as to permit on line testing when operating in the RHR mode. Testability shall meet requirements of IEEE Standard 338 and Regulatory Guide 1.22. (This is discussed in Section 5 of this report.)

The preoperational and initial startup test program shall be in conformance with Regulatory Guide 1.68. The programs for PWRs shall include tests with supporting analysis to (a) confirm that adequate mixing of borated water added prior to or during cooldown can be achieved under natural circulation conditions and permit estimation of the times required to achieve such mixing, and (b) confirm that the cooldown under natural circulation conditions can be achieved within the limits specified in the emergency operating procedures. Comparison with performance of previously tested plants of similar design may be substituted for these tests.

Regulatory Guide 1.68 was not in effect when Oyster Creek was designed and built. However, the licensee committed to preoperational tests to confirm operability and many uses have shown the system to be reliable for removing decay heat.

As part of the startup testing for Oyster Creek, the isolation condensers were placed in operation and the heat removal rates were measured and found to be in excess of design capacity. Similarly, the relief valve capacities were measured and found to be within tolerance of their design flow rates.

The licensee does not perform tests of SCS isolation feature for reactor coolant temperature greater than 350°F. The isolation of the SCS due to low-low water level is tested during the refueling outage.

4.6 Operational Procedures

The operational procedures for bringing the plant from normal operating power to cold shutdown shall be in conformance with Regulatory Guide 1.33. For pressurized water reactors, the operational procedures shall include specific procedures and information for cooldown under natural circulation conditions.

The licensee has procedures to perform safe shutdown operations including shutdown to hot standby, operation at hot standby, hot shutdown, operation at hot shutdown and cold shutdown including long-term decay heat removal. The licensee has also provided the operating staff with procedures covering off-normal and emergency conditions for reactor shutdown and decay heat removal under conditions of loss of systems or parts of system functions normally needed for shutdown and cooling the core. Procedures for operation of systems used in safely shutting down the reactor are also included in the plant operating procedures. These procedures include provisions identified in Regulatory Guide 1.33. These procedures were reviewed and are in conformance with Regulatory Guide 1.33.

Certain operations were identified to the reviewers which constitute alternate ways and paths to achieve cooling water source alignment or heat sink alignment. Some of these methods are not included in their procedure system.

4.7 Auxiliary Feedwater Supply

The seismic Category I water supply for the auxiliary feedwater system for a PWR shall have sufficient inventory to permit operation at hot shutdown for at least 4 hours, followed by cooldown to the conditions permitting operation of the RHR system. The inventory needed for cooldown shall be based on the longest cooldown time needed with either only onsite or only offsite power available with an assumed single failure.

Boiling Water Reactors such as Oyster Creek do not have an auxiliary feed system. However, the cooling water inventory requirements for a safe shutdown of the facility, using the systems identified in Section 4.0, are evaluated in Appendix A.

TABLE 4.1 CLASSIFICATION OF SAFE SHUTDOWN SYSTEMS OYSTER CREEK

	Quality Group		Seismic		
Components/Subsystems	R.G. 1.26	Plant Design	R.G. 1.29	Plant Design	Remarks
Isolation Condensers (tube side)	ASME III Class 1	ASME III Class C	Category I	Class I	Except for the reactor coolant pressure boundary, the system boundaries are defined as those portions of the system required to accomplish the system safety function and connected piping up to and including the first valve that is either normally closed or capable of automatic closure when the safety function is required. (See R.G. 1.26)
(shell side)	ASME III Class 2	ASME VIII			
Iso. Condenser piping & valves	ASME III Class 1	ASME I 1965			
Core Spray System					
pumps (4)	ASME III Class 2	ASME III Class C			
Piping and valves in the system boundary		ASA B31.1			
Containment Torus		ASME VIII & Nuclear Code Cases			
Automatic Depressurization System					
Relief valves (5)	ASME III Class 1	ASME I			
Condensate Transfer System					Supplies condensate for Isolation Condensers.
pumps (2)	ASME III Class 3	?		Class II	

TABLE 4.1 (Continued)

Components/Subsystems	Quality Group R.G. 1.26	Quality Group Plant Design	Seismic R.G. 1.29	Seismic Plant Design	Remarks
Piping and valves	ASME III	?	Category I	Class II	
Condensate Storage Tank		?			
Emergency Service Water					
pumps (4)	ASME III Class 3	?			
Piping and valves		ASA B31.1			
heat exchangers (4)					
(shell side)		?			
(tube side) (contain spray)	ASME III Class 2	?			
Containment Spray System					
pumps (4)	ASME III Class 2	?		Class I	
Piping and valves		ASA B31.1		Class I	

TABLE 4.1 (Continued)

Components/Subsystems	Quality Group: R.G. 1.26	Quality Group: Plant Design	Seismic: R.G. 1.29	Seismic: Plant Design	Remarks
Emergency Power System					
Diesel generators	N/A	--	Category I	Class I	
DC batteries		--			
Distribution lines, switchgear, control boards, motor control centers		--			
Diesel mechanical auxiliaries	ASME III Class 3	?		?	
Reactor Control & Protection System	N/A	--	Category I	Class I	
Safe Shutdown System Instrumentation & Control	N/A	--			

TABLE 4.2 LIST OF SAFE SHUTDOWN INSTRUMENTS

Component/System	Instrument	Instrument Location	Reference
Reactor Recirculation System	Reactor Vessel level (LT 1A12, LITS RE05-19 A&B LI 1A13, LI RE21A&B)	LT & LITS - Reactor Building (RK01 & RK02) LI-Control Room (5F/6F)	DWG 148F712
	Reactor Vessel pressure (PIT 1D45, PIT 1D46A&B, PR/FR ID 77, PRIA08)	PIT - Reactor Building (RK01 & RK02) PR/FR, PR - Control Room	Plant Description Manual (PDM)
Isolation Condenser	Secondary level (LT IG06 A&B, LI IG07 A&B)	LT - Reactor Building LI - Control Room (2F)	DWG 148F262
Condensate Transfer System	Pump discharge pressure ( )		
	Cond. Storage Tank level (LT)	LT - At base of tank, east side LI - Control Room	Site visit
Core Spray System	CS flow (FTRV 27A&B, FI RV 27A&B)	FT - Reactor Building FI - Control Room (IF)	PDM
Pressure Suppression System (Torus)	Cont. Spray suction temperature (TE 1903-40A, TR IP01)	TE - Reactor Building TR - Control Room	PDM
Emergency Service Water System	ESW - Cont. Spray D/P (dPT IP05A, B, C, D & dPI IP06A, B, C, D)	dPT - Reactor Building dPI - Control Room	PDM
Containment Spray System	Cont. Spray flow (FTIP03A & B, FI IP04 A& B)	FT - Reactor Building FI - Control Room	PDM
Diesel Generator No. 1 and Diesel Generator No. 2	Diesel Gen. output voltage and current	Control Room	
DC Power Div. 1 and DC Power Div. B	Voltage and current, Div. 1 - Batteries A & B Div. 2 - Battery C	Control Room (8F/9F, 9XF)	DWG D-3033-1A DWG 3028-11A Site visit

TABLE 4.3 SAFE SHUTDOWN SYSTEMS POWER SUPPLY AND LOCATION

Component/System	Power Supply	Location
Condensate Transfer System pumps	480V MCC 1B32 via 1B3 from 4180V Bus 1D	Main transformer and condensate area (west of turb. build., 23')
Electromatic Relief Valves	125 VDC Control Power	Inside Drywell
Core Spray System pumps A, B, C, D	A, D - 4160V Bus 1C B, C - 4160V Bus 1D	Reactor Build. (pumps A, B, C, D (NW, SW Corner Rooms)
booster pumps A, B, C, D	B, C - 460V Substa. 1B2 A, D - 460V Substa. 1A2	(booster pumps A, C 51', B, D 23')
Containment Spray System pumps A, B, C, D	A, B - 460V Substa. 1A2 C, D - 460V Substa. 1B2	Reactor Build. (NE, SE Corner Rooms)
heat exchangers	-	Reactor Build. (23' NE, SE)
Emergency Service Water System pumps A, B, C, D	A, B - 4160V Bus. 1C C, D - 4160V Bus. 1D	Cir. Water Intake Structure
Diesel Generators No. 1 and 2	-	Diesel Generator Rooms
4160V Bus 1C	Offsite power or Diesel generator No. 1	Turbine Build. Mezzanine
4160V Bus 1D	Offsite power or Diesel generator No. 2	Turbine Build. Mezzanine
460V Substation 1A2	4160V Bus 1C	Office Build. North (23')
460V Substation 1B2	4160V Bus 1D	Office Build. North (23')
125V Batteries (A, B, C)	-	A, B - Battery Room (office build. 35') C - Enclosure Turbine Build. Mezzanine

5.0 RESOLUTION OF SEP TOPICS

The SEP topics associated with safe shutdown have been identified in the INTRODUCTION to this assessment. The following is a discussion of how Oyster Creek meets the safety objectives of these topics.

5.1 Topic V-10.B RHR System Reliability

The safety objective for this topic is to ensure reliable plant cooldown capability using safety-grade equipment subject to the guidelines of SRP 5.4.7 and BTP RSB 5-1. The Oyster Creek systems have been compared with these criteria, and the results of these comparisons are discussed in Section 4.0 of this assessment. Based on these discussions, we have concluded that the Oyster Creek systems fulfill the topic safety objective with the following comments:

1. The Shutdown Cooling System is not a safety-grade system by our definition. However, various ECCS systems, including ADS and core spray, can be utilized to effect reactor cooldown.

2. Component redundancy and single-failure-proof requirements are not met in the case of the shutdown cooling system, in that failure of either AC-powered valve inside containment (system suction or discharge) would result in loss of the system. However, the ECCS systems would still be available.

3. Component redundancy (and single-failure-proof) requirements are also not met in the case of the two isolation condensers. Each condenser's discharge isolation valve inside containment is supplied

AC power from the same vital bus. However, these valves are normally open and fail open on loss of electrical power. As noted in Section 3, it would take simultaneous spurious isolation of both condensers plus loss of the power supply to create any problem. Additionally, even if this highly-unlikely scenario were to occur, the ECCS systems would still be available.

4. No procedure exists to perform a shutdown and cooldown to cold conditions with the systems identified in Section 4.0. The licensee will be required to develop such a procedure.

5.2 Topic V-11.A Requirements for Isolation of High and Low Pressure Systems

The safety objective of this topic is to assure that adequate measures are taken to protect low pressure systems connected to the primary system from being subjected to excessive pressure which could cause failures and in some cases potentially cause a LOCA outside of containment. As noted in Section I, only the shutdown cooling system was examined. The shutdown cooling system is designed for full reactor pressure but less than full reactor temperature. Therefore interlocks (with the exception of the pump suction low pressure interlock) are based upon temperature considerations.

System operation cannot begin until temperature in all five reactor coolant recirculation loops and the shutdown cooling system suction lines is less than 350°F (and pump suction pressure exceeds 4 psig). This will enable system inlet and outlet valve-and-pump-permissive interlocks and allow the system to be started. Additionally, the inlet and outlet valves will shut, isolating the system, if temperature should rise to 350°F when the system is in operation.

Because of the systems full-pressure design and the incorporated interlocks (even though they are temperature-based), we consider the applicable requirements to have been met. However, there are no testing requirements for those interlocks. The need for such requirements will be addressed in the integrated assessment at the completion of the SEP review.

5.3 Topic V-11.B RHR Interlock Requirements

The safety objective of this topic is identical to that of Topic V-11.A. The staff conclusion regarding the Oyster Creek Shutdown Cooling System valve interlocks, as discussed in Section 5.2, is that adequate interlocks exist.

5.4 Topic VII-3 Systems Required For Safe Shutdown

The Safety objectives of this topic are:

1. To assure the design adequacy of the safe shutdown system to (a) initiate automatically the operation of appropriate systems, including the reactivity control systems, such that specified acceptable fuel design limits are not exceeded a a result of anticipated operational occurrences or postulated accidents, and (b) initiate the operation of systems and components required to bring the plant to a safe shutdown.

2. To assure that the required systems and equipment, including necessary instrumentation and controls to maintain the unit in a safe condition during hot shutdown are located at appropriate locations outside the control room and have a potential capability for subsequent cold shutdown of the reactor through the use of suitable procedures.

3. To assure that only safety grade equipment is required for a plant to bring the reactor coolant system from a high pressure to a low pressure cooling condition.

Safety objective 1(a) will be resolved in the SEP Design Basic Event reviews. These reviews will determine the acceptability of the plant response, including automatic initiation of safe shutdown related systems, to various Design Basis Events, i.e., accidents and transients.

Objective 1(b) relates to availability in the control room of the control and instrumentation systems needed to initiate the operation of the safe shutdown systems and assures that the control and instrumentation systems in the control room are capable of following the plant shutdown from its initiation to its conclusion at cold shutdown conditions. The ability of Oyster Creek to fulfill objective 1(b) is discussed in the preceding sections of this report. Based on these discussions, we conclude that safety objective 1(b) is met by the safe shutdown systems at Oyster Creek subject to the findings of related SEP Electrical, Instrumentation, and Control topic reviews.

safety objective 2 would require the capability to shutdown to both hot shutdown and cold shutdown conditions using systems, instrumentation, and controls located outside the control room. An Oyster Creek procedure concerning fire in the control room provides the steps for operation of the necessary equipment to shut the plant down, initiate the isolation condenser, and monitor necessary parameters. It does not include specific steps for achieving cold shutdown conditions. The ongoing fire protection review will require the licensee to develop a procedure to achieve cold shutdown conditions from outside the control room.

The adequacy of the safety grade classification of safe shutdown systems at Oyster Creek, to show conformance with safety objective 3, will be completed in part under SEP Topic III-1, "Classification of Structures, Components, and Systems (Seismic and Quality)," and in part under the Design Basis Event reviews. Table 4.1 of this report will be used as input to Topic III-1.

6.0 REFERENCES

1. Staff Discussion of Fifteen Technical Issues listed in Attachment to November 3, 1976 Memorandum from Director, NRR to NRR Staff, NUREG-0138, November 1976.

2. Letter from USAEC to Jersey Central Power and Light transmitting Safety Evaluation Report by Division of Reactor Licensing, December 23, 1978.

3. Oyster Creek Fire Protection Safety Evaluation Report, March 3, 1978.

4. Letter from JCP&L to NRC, transmitting 125 VDC electrical modifications description, April 4, 1978.

APPENDIX A

SAFE SHUTDOWN WATER REQUIREMENTS

Introduction

Standard Review Plan (SRP) 5.4.7, "Residual Heat Removal (RHR) System" and Branch Technical Position (BTP) RSB 5-1, Rev. 1, "Design Requirements of the Residual Heat Removal System" are the current criteria used in the Systematic Evaluation Program (SEP) evaluation of systems required for safe shutdown. BTP RSB 5-1 Section A.4 states that the safe shutdown system shall be capable of bringing the reactor to a cold shutdown condition, with only offsite or onsite power available, within a reasonable period of time following shutdown, assuming the most limiting single failure. BTP RSB 5-1 Section G, which applies specifically to the amount of auxiliary feed system (AFS) water of a pressurized water reactor available for steam generator feeding, requires the seismic Category I water supply for the AFS to have sufficient inventory to permit operation at hot shutdown for at least four hours, followed by cooldown to the conditions permitting operation of the RHR system. The inventory needed for cooldown shall be based on the longest cooldown time needed with either only onsite or only offsite power available with an assumed single failure. A reasonable period of time to achieve cold shutdown conditions, as stated in SRP 5.4.7 Section III.5, is 36 hours. For a reactor plant cooldown, the transfer of heat from the plant to the environs is accomplished by using water as the heat transfer medium. Two modes of heat removal are available. The first mode involves the use of reactor plant heat to boil water with the resulting steam vented to the atmosphere. The water for this process is typically demineralized, "pure" water stored onsite and, therefore, is available only in limited quantities. The systems designed to use this type of heat removal process (boiloff) are the steam generators for a pressurized water reactor (PWR) or the emergency (isolation) condenser for a boiling water reactor (BWR). The second heat removal mode involves the use of power operated relief valves to remove heat in the form of steam energy directly from the reactor coolant system. Since it is not acceptable to vent the reactor coolant system directly to the atmosphere following certain accidents, the steam is typically vented to the containment building from where it is removed by containment heat removal systems. The containment heat removal systems are in turn cooled by a cooling water system which transfers the heat to an ultimate heat sink - usually a river, lake, or ocean. When using the blowdown mode, reactor coolant system makeup water must be continuously supplied to keep the reactor core covered with coolant as blowdown reduces the coolant inventory. Systems employing the blowdown heat removal mode have been designed into or backfitted onto most BWR's. The efficacy of the blowdown mode for PWR's has received increased staff attention since the Three Mile Island Unit 2 accident in March 1979. Additional studies of the viability of this mode for PWR's are in progress or planned.

This evaluation of cooling water requirements for safe shutdown (and cooldown) is based on the use of the system identified in the SEP Review of Safe Shutdown Systems which has been completed for each SEP facility. The Review of Safe Shutdown Systems used SRP 5.4.7 and BTP RSB 5-1 as a review basis. It should be noted that the SEP Design Basis Events (DBE) reviews, which are currently in progress, may require the use of systems other than those which are evaluated in this report for reactor plant shutdown and cooldown. In those cases, the water requirements for safe shutdown will have to be evaluated using the assumptions of the DBE review.

Discussion

The requirement that a plant achieve cold shutdown conditions within approximately 36 hours, as proffered in BTP RSB 5-1 and SRP 5.4.7, is based mainly on the fact that the amount of onsite-stored water for the AFS of a PWR is limited, and it is desirable to be able to place the RHR system in operation and transfer the plant heat to an ultimate heat sink prior to the exhaustion of the onsite-stored pure water supply. Remaining in a hot shutdown condition, with reactor coolant system temperature and pressure in excess of RHR initiation limits, requires the continued expenditure of pure water via the boiloff mode to remove reactor core decay heat. A BWR relying on the emergency condenser system for cooldown would also be susceptible to the potential exhaustion of onsite-stored pure water.

Should the onsite-stored water supply at a plant be expended, the capability usually exists to use raw water from a river, lake, or ocean, for example, to supply the boiloff systems. However, use of raw water can lead to the degradation, through corrosion, of the boiloff system materials, i.e., steam generator and emergency condenser tubes. This degradation can occur rapidly even if fresh water makeup is used. If seawater were used, chloride stress corrosion cracking of the tubes could occur well within one week.* If raw fresh water were used, caustic stress corrosion cracking of tube materials could occur in less than 72 hours for both stainless steel and inconel tube materials through NaOH concentration.* A plant cooldown and depressurization

*"vanRooyen, Daniel and Martin W. Kendig, 'Impure Water in Steam Generators and Isolation Generators.' BNL-NUREG-28147, Informal Report. June 1980."

would help reduce the rate of tube cracking by reducing the stresses in the tube materials. Also, the leakage rate of reactor coolant through potential cracks in the tubes would be reduced if the plant were in a cool, depressurzied state.

The original design criteria for the SEP facilities did not require the ability to achieve cold shutdown conditions. For these plants, and for the majority of operating plants, safe shutdown was defined as hot shutdown. Therefore, the design of the systems used to achieve cold shutdown condition was determined by the reactor plant vendor and was not based on any safety concern. Our safe shutdown reviews have pointed out a difference in the vendor approach to system design for cold shutdown. This difference is reflected in the Standard Technical Specification definition of cold shutdown. For a BWR, cold shutdown requires reactor coolant temperature to be $\leq$212 degrees Farenheit. For a PWR, cold shutdown requires reactor coolant temperature to be $\leq$200 degrees Farenheit. These differences in cold shutdown temperatures require the use of additional systems to achieve cold shutdown for a PWR over and above the systems needed for a BWR. For example, a BWR could use an isolation condenser alone to reach 212 degrees Farenheit (although the approach to 212 degrees Farenheit would be asymptotic); but a PWR, in addition to the steam generators, must use an RHR and supporting systems to get below 200 degrees Farenheit.

Evaluation

Table 1 provides plant specific data and assumptions used in the staff calculation of safe shutdown water requirements for the Oyster Creek nuclear plant. Table 2 provides the results of the calculation. The temperature profile for the cooldown is shown in Figure 1.

After the reactor trip, the reactor system pressure and temperature increase to the relief valve pressure setpoint because the main condenser is not operable following an assumed loss of offsite power. The emergency condensers are automatically initiated after reactor pressure exceeds 1060 psig for 15 seconds; however, one of the condensers is assumed to be inoperable because this single failure assumption results in the longest cooldown time and is most limiting from the standpoint of pure water consumption. The operator is assumed to maintain reactor system pressure near normal operating pressures, by cycling one of the emergency condenser condensate valves, for a period of four hours prior to commencing the cooldown. The four hour delay is based on BTP RSB 5-1 Section G and again is intended to maximize pure water consumption. Emergency condenser pure water makeup is supplied by the condensate transfer system; the level of makeup water in the emergency condensers is controlled by the control room operator by means of water level transmitters and remotely controlled, air-operated makeup valves. The cooldown data are presented in Table 2 (and on Figure 1). Since the plant compressed air systems are not on the safe shutdown system list, control of the emergency condenser makeup valves would be accomplished by manual operation of the handwheels on the valves.

As the cooldown progresses, the reactor system fluid contracts and the need for reactor system makeup exists to keep the level of coolant above the fuel in the reactor core. The reactor feed system, which is normally used to inject water into the reactor system at high pressures (greater than 285 psig), is not available because it depends on offsite power. The Control Rod Drive (CRD) hydraulic system, which can also supply high pressure water, is not considered to be available because it was not designed as a safe shutdown (safety) system and, therefore, is not included on the safe shutdown system list for Oyster Creek. Without these high pressure reactor makeup systems, the operator must rely on the core spray (CS) system to supply reactor coolant. The CS system is a low pressure system (shutoff head 285 psig); and, therefore, if reactor pressure is not below 285 psig, the operator must initiate the Automatic Depressurization System (ADS) to lower the pressure sufficiently for CS flow into the reactor system to occur. In fact, the ADS can be manually initiated at any time during the cooldown sequence following reactor trip, provided the reactor vessel level at ADS initiation is at or above the triple-low level (4'8" above the core); and the CS system will provide adequate core cooling. Thus, the ADS and emergency condensers are redundant to each other for the function of plant cooldown. The main reason that the emergency condensers are included on the safe shutdown list is to provide a core cooling method which does not reduce the reactor system coolant inventory since Oyster Creek does not have the high pressure coolant injection capability that most other boiling water reactors have.

In the course of the cooldown, the operator must eventually use the ADS, CS, and containment heat removal systems (containment spray and emergency service water) for long term cooling of the plant. The core heat and stored heat in the reactor system materials is transferred to the containment by the CS and ADS. The containment heat removal systems transfer the heat to the ultimate heat sink. Normally, long term heat removal would be accomplished by the

Shutdown Cooling System (SCS). If this system and its auxiliary systems are available, it could be started at a reactor system temperature below 350°F (approx. 8.5 hours on Table 2). However, since the SCS and its auxiliaries were not designed and constructed with the quality of the plant safety systems, the ADS, CS, and containment cooling systems are relied on for long term cooling of the plant.

The quantity of makeup water consumed by the operable emergency condenser is a function of the time at which the long term cooling heat removal mode begins transferring heat to the ultimate heat sink. The condensate storage tank contains 2,085,000 lbs of water. At the end of the four hour delay period before cooldown starts, 334,000 lbs. of water would be expended. To cooldown to the conditions required for CS initiation (285 psig), 750,000 lbs. of condensate would be expended. And, if the SCS were available, it could be started at a reactor system temperature of 350°F after 955,000 lbs. of densate were consumed.

Based on our calculations, sufficient emergency condenser makeup inventory capacity is available in the condensate storage tank to conduct a plant cooldown in accordance with BTP RSB 5-1. However, the Oyster Creek plant technical specifications should be modified to require the plant operators to maintain sufficient condensate storage tank inventory to cond t the cooldown (approximately 750,000 lbs) in addition to the inventory requirements for the emergency core cooling systems.

TABLE 1

Plant: Oyster Creek Power (MW) : 1930

Normal Operating Temp. (°F): 547

Safety valve lift (psig): 1070

Initial secondary inventory (lbm): 92240 (in each of 2 emergency condensers)

Secondary makeup water temp. (°F): 80

S/RV flow area (ft 2): NA

Emerg. Condenser total ht. xfer. coeff. (BTU/hr-°F): 6.1E5 at 1070 psig

Stored sensible heat (BTU/°F): fuel - 27,000
metal - 224,000
water - 1,540,000

RHR Parameters: Not applicable

Pure water onsite (lbm): 2,085,000 in the Condensate Storage Tank*
250,200 in the Demineralized Water Storage Tank*

Cooldown assumptions:

1. At t=0 reactor trips.
2. Decay heat is in accordance with proposed ANS 5.1 (1973).
3. Plant remains at hot shutdown for four hrs. prior to cooldown.
4. The secondary (steam generator or emerg. condenser) is considered dry when the initial secondary inventory is boiled away.
5. Emergency condenser total heat transfer coefficient is assumed to be constant.

*These quantities are not included in plant technical specifications.

TABLE 2

Plant: Oyster Creek

Phase I (reactor trip to safety lift):

Time to safety valve lift (sec): approx. 0

Phase II (safety valve lift to cooldown start):

Time to boil secondary dry, assume no makeup (min): 40 (for one isolation condenser)
Decay heat generated prior to cooldown start (BTU): 324E6
Feedwater expended prior to cooldown start (lbm): 334,000

Phase III (cooldown): (1 emergency condenser)

Time (hrs)	Temperature (°F)	Pressure (psia)	Decay heat generated (BTU)
4	553	1085	324E6
5	478	563	380E6
6	425	327	443E6
6.5	403	258	459E6
8	357	147	531E6
8.5	345	129	554E6
10	320	94	620E6
12	299	79	703E6
24	267	55	1170E6

FIGURE 1 REACTOR SYSTEM TEMPERATURE VS TIME