HADDAM NECK NUCLEAR POWER PLANT, REACTOR CONTAINMENT
(Connecticut Yankee Nuclear Power Plant, Reactor Containment)
362 Injun Hollow Road
Haddam
Middlesex County
Connecticut

HAER CT-185-B
HAER CT-185-B

WRITTEN HISTORICAL AND DESCRIPTIVE DATA

REDUCED COPIES OF MEASURED DRAWINGS

FIELD RECORDS

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(Connecticut Yankee Nuclear Power Plant, Reactor Containment)
HAER NO. CT-185-B

Location:	362 Injun Hollow Road Haddam, Middlesex County, Connecticut U.S. Geological Survey Deep River Quadrangles UTM Coordinates 18.708795.4595127
Dates of Construction:	1964-1966
Engineers:	Westinghouse Electric Company (nuclear steam supply system equipment), Stone & Webster Engineering Corporation (foundations and superstructure)
Present Owners:	Connecticut Yankee Atomic Power Company (CYAPCO) 362 Injun Hollow Road Haddam Neck CT 06424-3022
Present Use:	Demolished with some foundations left in place
Significance:	The Haddam Neck Nuclear Power Plant was one of the earliest commercial scale nuclear power stations in the United States, and was eligible for the National Register of Historic Places. Reactor containment was one of the two most significant structures at the complex, along with the Turbine Building and produced the steam which was converted to electrical energy in the Turbine Building. Reactor Containment equipment design was typical of contemporary choices in reactors, steam generators and other major nuclear steam supply components. The steam generators had problems common to plants of this vintage.
Project Information:	CYAPCO ceased electrical generation at the Haddam Neck plant in 1996 and initiated decommissioning operations in 1997, subject to authority of the Nuclear Regulatory Commission (NRC). NRC authority brought the project under the purview of federal acts and regulation protecting significant cultural resources from adverse project effects.* This documentation was requested by the Connecticut State Historic Preservation Office to preclude the possibility of any adverse project effects.

* National Historic Preservation Act of 1966 (PL 89-655), the National Environmental Policy Act of 1969 (PL 91-190), the Archaeological and Historical Preservation Act (PL 93-291), Executive Order 11593, Procedures for the Protection of Historic and Cultural Properties (36 CFR Part 800).

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General Design Considerations

The Reactor Containment at the Connecticut Yankee plant was designed and constructed to provide a structurally stable, passive, and environmentally-secure housing for the nuclear steam supply system (NSSS), auxiliaries for normal operation, and some of the active engineered safety systems (ESS) associated with NSSS operations. With an inside diameter of 135 feet and total height of approximately 218 feet (with 170 feet above ground level), the internal volume was over 2.6 million cubic feet. The containment envelope consisted of a steel vapor barrier liner to resist gaseous emissions surrounded by and in contact with a reinforced-concrete biological shield to attenuate radiation.[1] Containment design represented a progression from the stand-alone steel spheres pioneered at Knolls Atomic Power Laboratory in Milton, New York in the mid-1950s, later structures with complete concrete encasement at ground level,[2] and the Indian Point 1 design with a separate steel barrier inside a concrete shell.[a] Probably the earliest American nuclear power plant with a unified steel and concrete containment was the Carolinas -Virginia Nuclear Power Associates (CVNPA) plant [b] in South Carolina, plant, also engineered by Stone and Webster, in which the erected steel shell served as the form work for the inside surface of the outer concrete shield.[3]

The principal load which Reactor Containment was designed to resist was internal pressure caused by a loss of coolant accident or a hydrogen explosion.[4] The structure also had to resist earthquake loads, atmospheric wind loading, and damage from internally-generated flying-component impacts, and have specified levels of safety against fire, flood, and electrical storms. Plant designers believed that there was an insignificant amount of combustible material, such as pump and motor lubricating oil and electrical cable insulation.[c] Earlier flood control programs in the Connecticut River maintained normal river

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[a] There was diversity in the containment types during the late 1960s. San Onofre had the steel vapor barrier as a separate structure from the concrete shell (Golden 2006). While a number of plants that followed Connecticut Yankee had steel-lined concrete containments, the design was not universal. Three 550- mw reactors at Prairie Island, MN, and Kewaunee, WI reverted to a separate system like Indian Point 1 called "Multibarrier Containment" which claimed to save costs (Koehler, Leppke, and Corcoran 1969: 214) For the never-constructed Malibu plant near a major population center — Los Angeles — Stone and Webster proposed a "double containment" system: an inner steel liner backed by low density cement followed by another steel liner attached to the outer reinforced concrete shell. With an extensive air suction and filtration system, the design goal was "zero leakage" (Stern 1964: 248).

[b] Later known as the Carolinas-Virginia Tube Reactor on account of its use of pressure tubes, natural uranium and heavy water moderation as in the Candu series.

[c] The 1975 Browns Ferry fire in Alabama showed that cable runs were vulnerable to serious fires that could disable critical plant systems (Ford 1982: 161).

levels below the plant surface elevation of 21 feet NGVD, and Reactor Containment was heavy enough to resist buoyancy at a flood stage of 70 feet. Calculations for hypothetical internal pressure were based in part on the total energy content in BTUs of the reactor coolant system water released into the structure from a double-ended rupture of a main reactor coolant pipe. The total pressure on the structure was determined from this figure, plus the effect of engineered safety systems (ESS) and heat sinks (components that stored, dissipated or transferred heat), to give a design pressure of 40 pounds per square inch gage (psig),[d] a maximum event pressure of 30 psig, and liner temperature of 260F. The maximum event pressure had to be maintained with all ESS operated from the backup diesel generators. Designers also provided for a small negative pressure, but considered this an inconceivable risk. The grade of liner steel was chosen for weldability and resistance to cold-induced brittle fracture to allow winter fabrication, and for sufficient strength to stand alone for pressure-leak testing before concrete encapsulation. The seismic loading was designed for a ground acceleration of 0.17g in the horizontal direction only, using calculations based on methodology developed for multi-story reinforced concrete factory or office buildings.[5] It was expected that the steel vapor liner alone would resist horizontal earthquake loads.[6] Wind loading was taken as 150 mph, but calculations for damage from external missiles such as tornado-driven telephone poles, flying automobiles, or projected parts resulting from the disintegration of the main turbine (see HAER No. CT-185-C) were not considered.[e] Potential damage from internally-generated missiles such as valves or chunks of concrete from exploding components was calculated from Army ballistic tests, and it was concluded that the internal concrete shield walls and the charging floor above them would deflect most missiles. Concrete slabs over the reactor control rod housings prevented damage to components if the rods were ejected during an accident, and pressurizer valves were protected by concrete slabs.[7]

The reactor vessel, the steam generators and the pressurizer were considered immune from damage due to their thick shells, but it was acknowledged that it would be impossible to make the liner or piping steel damage-proof against all postulated missiles.[8] At the time Connecticut Yankee was on the drawing board there were no building codes specifically aimed at reinforced concrete containment structures, so American Concrete Institute, American Society of Civil Engineers (ASCE), national and Connecticut codes were used.[9] Concrete had been used for internal shielding for years, and presumably the company

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[d] Steam or gas pressure was stated as pounds per square inch gage (psig) which was pressure over nominal atmospheric pressure at sea level of 14.7 pounds per square inch (psi). Pressure over true 0 was known as pounds per square inch absolute (psia).

[e] The calculations for the 1970 Indian Point 2 plant (one of the last built before pre-stressing of concrete containment structures became common) called for 300 mph wind loading with ability to resist sections of wood decks flying at 200 mph, 2-ton autos traveling at 50 mph, and provided for 0.15g of horizontal and 0.5 g vertical earthquake acceleration (Haga and Stevenson 1970: 153).

specified a mix that complied with the 1962 ASCE published guidelines for the types of concrete that most effectively resisted the damaging effects of Gamma rays and heating.[10]

Building Design and Construction[f]

Reactor Containment had a largely sub-grade reactor location inside a 120.5-foot-high straight-walled reinforced-concrete cylinder with a 70-foot-high hemispherical top known as a "right circular cylinder."[11] The steel vapor shell was attached to the inside surface of the outer concrete wall. Five principal full or partial levels inside or below the cylinder supported the reactor; four steam generators and coolant pumps; pressurizer, emergency core cooling system (ECCS), ventilation and filter equipment; sump; refueling systems; and overhead crane. The reactor was enclosed in a separate concrete chamber (reactor cavity) by a primary shield wall which isolated the coolant pumps and steam generators from radiation, allowing access shortly after shutdown.[12] Additional protective walls separated the pumps and steam generators into pairs.

A second concentric circular concrete wall (the secondary shield) further isolated the primary system from the containment shell and provided the support for the "polar" overhead crane that rotated and traversed to cover all the equipment areas. A concrete floor between the walls of the reactor cavity and the secondary shield provided a surface for access to the reactor head for refueling. Auxiliary systems were installed between the secondary shield and the outer Reactor Containment wall (Drawings HAER Nos. CT-185-B-1 through CT-185-B-7).

The construction process involved the alternating erection of concrete and steel portions. The lower connecting reinforcing bars for the outer shell and internal shield walls were placed before the main slab was poured. The approximately two-inch-thick bars were of special steel, and reinforcing bar connections were either welded or sleeve cast. Over 6,200 tons of reinforcing steel was used in the containment structure (Figures 2-5; Drawings HAER Nos. CT-185-B-5, CT-185-B-6).[13]

The main foundation slab, with an upper elevation of -0.5-feet, was a circular 9-foot-thick mat of heavily reinforced concrete. In the center of the slab was a keyhole-shaped depressed center well with a floor at elevation -20.5-feet, lined with 1/2-inch welded carbon steel which provided space for the lower reactor components, and served as sump from which spilled system water could be pumped.[14] The bottom of the 7-foot thick sump slab was at -27-feet or 48-feet below the site ground level. The mat was designed to rest on bed rock requiring deeper than desired excavation in some areas. The surface rock was uneven requiring voids to be filled with concrete for an even bearing (Drawings HAER Nos. CT-

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[f] Parenthetical references to Drawings are to large-format reproductions of original drawings; [drawings] in endnotes refer to Section A in Sources of Information/Bibliography.

185-B-1, CT-185-B-5, CT- 185-B-6).[15] The surface of the main slab was completely covered with 1/4-inch-thick carbon-steel liner plate upon which another 2 feet of concrete was laid. Above the main slab, the first courses of the inner vertical 3/8-inch-thick steel liner were welded to the periphery of the steel floor liner and rose up forming a right circular cylinder.

All welds on the steel shell were either radiographed or dye penetrant checked in accordance with the ASME Boiler and Pressure Vessel Code and the standards set by the American Society for Testing Materials (ASTM).[16] The weld seams were then gas tested for leakage.[17] Insulation was added to the lower portion of the liner to limit hot reactor water spilled in an accident from reducing the earthquake resistance of the steel.[18] While the steel vapor shell was being erected, the 4.5-foot-thick concrete portion of the primary shield system that surrounded the main part of the reactor vessel was poured.[19] The circular wall around the reactor was recessed to provide clearance for the coolant nozzles and the encircling, water-filled neutron shield tank which supported the reactor at about elevation -3-feet and protected the adjacent concrete from neutron radiation and heat damage. The concrete around the reactor vessel was also protected by a 1/4-inch-thick steel "dust" liner.[20] The reactor head cavity above the primary shield at elevation 22-feet had a 3/8-inch-thick steel liner, and was large enough to provide lay down areas on two sides of the reactor for components removed during refueling. Certain elements of the primary shield were inside the reactor.[21] The four reactor coolant pumps, the four steam generators, the pressurizer and the coolant loop piping were arrayed around the primary shield at various elevations. Concrete divider walls 1.5-feet thick and 47-feet high separated the components into two groups along the east/west axis (Figure 3; Drawings HAER Nos. CT-185-B-3 through CT-185-B-6).

The fuel transfer canal next to the east side of the reactor head cavity, and the transfer tube, were probably formed and poured at the same time, with a bottom elevation of about 15 feet. Concrete was also poured to form the secondary shield which consisted of a 3-foot-thick, 47-foot-high ring around the coolant loop areas to provide biological shielding, protection from explosively projected coolant loop components, and a foundation for the revolving polar crane at elevation 48.5-feet.[22] The secondary shield also formed a chamber (annulus) with the outer wall around containment for auxiliary equipment at elevation 22-feet which controlled containment atmosphere (fans, ducting and filters) during normal operations or during accident conditions. The area in the annulus below elevation 22-feet served as a chase for most of the high-pressure steam piping, and was floored over to make the area accessible by personnel during reactor operation (Drawings HAER Nos. CT-185-B-3, CT-185-B-5 through CT-185-B-6, and CT-185-B-12 through CT-185- B-13).[23] Observation ports in the shield walls allowed visual inspection of parts of the NSSS during normal operations. A 2-foot-thick concrete charging floor at elevation 48.5-feet surrounded the reactor cavity opening and was attached to the top of the secondary shield wall and

divider walls with rebar connections, covering the reactor coolant loops. In addition to providing missile protection, the floor supported the manipulator crane and personnel during refueling and maintenance operations. Openings were left in the floor for the projecting steam generators, pressurizer, and piping/ductwork runs. Access to the components was by removable slabs and shield blocks.[24] Additional form work was provided to construct a large equipment hatch at the north side of the structure and east of it for the refueling canal and transfer tube.[25] The reactor cavity was left uncovered, and was flooded during refueling as described in HAER No. CT-185.

Once the charging floor concrete was poured, the 175-ton crane was erected. Electric-motor-powered wheels traveling on a rail or rails laid on top of the shield wall provided rotation. Twin welded 38-foot-high column legs on each end supported two welded plate cross girders upon which rode two hoist trolleys, one with 125-ton main and 15-ton auxiliary hooks and the other with a 125-ton main hook. One end of the crane girders overhung the support leg to cover the annulus area and serve as a base for an electrical collector which picked up power from a conductor mounted on brackets that ran around the liner of the outside Reactor Containment wall. The idea of a polar 360-degree revolving crane was not new to nuclear containment structures, having been used earlier in circular water pumping stations (Figure 7; Drawings HAER Nos. CT-185-B-4 and CT-185-B-5).[26]

Once the crane was erected, construction resumed on the steel liner. At elevation 121 feet, the plates for the hemispherical steel ½-inch-thick dome liner were welded to the side wall. A large opening was left at top of the dome until well after the turbine and auxiliary buildings were erected. The inside surface of the liner was protected from corrosion by vinyl paint but it was expected that the acidity of the adhering concrete would protect the outer surfaces (Figure 8).[27]
During 1966 the reactor components were barged up the Connecticut River. The reactor mounted on a shipping skid was slid into the building on a greased steel temporary floor through an opening left in the shell where the access hatch was to be located. Once inside both main hoists on the crane were attached to a lifting beam to upend the reactor and place it in position.[28] The steam generators and pressurizer were backed into the building on multi-wheel lowboy trailers and rigged onto wooden cribbing and sandbags for positioning (upending) under the lifting beam.[29]

It appears from company photographs that both fixed and slip-form pouring of concrete was used to construct the 4.5-foot-thick side walls and 2.5-foot-thick dome, with the steel shell serving as the inner form surface (Figure 9). Concreting was completed after major NSSS components were installed. The steel vapor shell was studded to the concrete to prevent buckling of the liner from high temperatures in an accident. The top of the dome had a ventilation hatch reached by an external ladder. That penetration and another on the east side were only used during construction but were checked for integrity during each

refueling. Containment engineering had to allow for loading in and out of large equipment, personnel entry and exit, and routing of electrical, steam, and water lines. Each penetration was a potential source of radiation streaming during normal operations or in the event of an accident. The largest penetration was the 23-foot diameter equipment hatch on the north side of the shell, where the main NSSS components were loaded in during construction. The opening was framed with an extra heavy concrete ring connected to the shell with reinforcing. The cover was double gasketed and bolted into the ring.[30] With the cover removed, the opening was large enough to allow the steam generators and pressurizer to be replaced without damaging the hatch frame concrete.[31] The equipment hatch cover was itself penetrated by an 86-inch-diameter access hatch, protected by an air lock, for personnel and smaller components. Interlocks were provided to prevent accidental creation of leak pathways.

The inner access hatch door also had an 18-inch-wide man way for emergency egress if the door locking system failed. It is undocumented whether the opening as framed in concrete was large enough to admit replacement of main components. On either side of the hatch were temporary penetrations that could be plugged with sealant for routing of hoses and cables which allowed the hatch to be closed during refueling. For full integrity during power production, bolted and welded flanges were added over the openings. Reactor Containment had 120 electrical service penetrations made with steel sleeves running through the concrete and welded to the steel shell. Conductors were housed in conduits were originally sealed with compression fittings and O-rings and later by "Conax" penetration assemblies.[32] There were also 80-100 mechanical penetrations for piping.[33] The main steam and feed water lines were routed in sleeves between containment and the Terry Turbine Building (HAER No. CT-185-E) while other pipe chases were on the north side. They were generally protected by various manual and automatic closure valves. The Component Cooling Water System (CCW) cooled process lines carrying heated fluids and vapors to prevent weakening the adjacent concrete from dehydration (see HAER No. CT-185).[34] The refueling transfer tube was a major penetration between containment and the fuel building. A purge system to clear out "activity" in containment before personnel access required a penetration for connection to the 175-foot-high vent stack attached to the Primary Auxiliary Building (HAER No. CT-185-G).[35] All the penetrations were leak tested on completion.

Further protection for the largest penetrations was provided by external concrete shield walls constructed to block any radiation streaming from containment. The main steam and feed water lines were shielded by a 3-foot-thick by 20-foot-high concrete structure which formed the west wall of the Terry Turbine building (HAER No. CT-185-E). Another 3.5-foot-thick wall shielded the purge duct penetration. The containment access hatch was protected by a labyrinth of concrete walls and blocks but it was later removed because it

was determined that any radiation stream would by blocked by the hill east of containment.[g] After the containment cement had cured, the structure was pressure tested to 40-psig for 24 hours to ensure leakage was below the design rate of 0.1 % of volume (a rate actually below the limit set by the CFR).[36]

In operation, containment integrity had to be functional as soon as the reactor coolant loops began to heat up and pressurize. All penetrations had to be closed or their automatic trip devices working in accordance with an inspection procedure. With integrity complete, the building was slightly pressurized to facilitate leak monitoring.[37] Containment temperature, pressure, and humidity were continuously monitored during operation, but individual tests of the penetrations were only performed during refueling.

Containment Operating Systems

The circular neutron shield tank was rectangular in section, wrapping around the chamber walls that surrounded the core region, resting on a shelf near the bottom hemisphere of the reactor vessel. The 1-inch-thick steel tank walls formed a box 28-inches thick and 16-feet high[38] containing over 14,000 gallons of water to prevent gamma and neutron bombardment of the surrounding concrete and subsequent weakening by dehydration.[39] A natural circulation flowed though the tank and into a shell-and-tube heat exchanger cooler located above the tank outside the primary shield. A surge tank there maintained water level independent of thermal expansion and provided leak make-up. The top of the shield tank carried the entire weight of the reactor vessel from support blocks below four of the reactor coolant system loop nozzles.[40] The tank was penetrated for instrumentation outside the core (ex-core) on account of its controlled temperature (Drawings HAER Nos. CT-185-B-5, CT-185-B-6, and CT-185-B-9).[41] Many of the normal operating systems were located in the secondary shield/shell annulus including equipment to maintain atmospheric control by cooling, recirculating, filtering, heating, humidifying, and if necessary purging.[42] The Containment Air Recirculation (CAR) system (which also operated during accident conditions) consisted of four fan powered cooling coil units and associated duct work at elevation 22-feet. The system could move 100,000 cubic feet of air per minute (cfm) drawn from the annulus over coils cooled by the service water system. Ducting directed the air stream to the coolant loop areas, refueling cavity and the dome area. The system, which operated whenever the reactor coolant system was hot and pressurized, had backup power from the diesel generators in the event of a power failure.[43] In accident conditions, banks of filters were cut in to dry and purify the air (see engineered safeguards described in HAER No. CT-185-A).

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[g] Radiation streaming (as opposed to leaks projected in gaseous or water inventory accidents) was assumed to travel in a straight line out of an opening allowing containment by labyrinth structures or geological features at ground level (Connecticut Yankee Atomic Power Company 1987-1995: Chapter 7, page 20.)

A separate system provided cooling air for the control rod drives and their vicinity. A Containment Heating System used six steam coil-unit heaters to heat and humidify the building during long shutdowns, using steam from two oil fired auxiliary boilers in the Service Boiler Room (see HAER NO. CT-185-Q). Heating system condensate from which was sent to the Primary Auxiliary Building (HAER No. CT-185-G). The purge system was designed to clear out containment of "normal" activity, filter the air and ultimately release it from the primary vent stack[44]. Containment atmospheric radiation monitoring was done by separate gaseous and particulate monitors in the Primary Auxiliary building and high range radiation monitors in Containment. Two sump pits below the reactor vessel collected spilled water during normal operation or after water spill accidents. The in-core instrumentation (ICI) sump and pump were normally used to collect leakage from the neutron shield tank and instrumentation nozzle penetrations in the bottom of the reactor and pump them to the two part main containment sump. Two pumps there sent the water to the Liquid Waste Disposal System. In the event of a spill accident these sumps were vital to clean-up operations, and they also functioned with and forwarded water to the Emergency Core Cooling System (ECCS).[45] Water level instrumentation signaled the control room if the water level exceeded the ICI sumps capacity during an accident. The entire containment area had loudspeakers for paged messages and ensured complete alarm coverage. Uninterruptable communications systems were used between the control room and all fuel handling equipment stations. Personnel moved up and down on space saving spiral stairs. A 2.25-ton freight elevator near the equipment hatch accessed several levels and made it unnecessary to the use the main crane for smaller equipment transfers. The access doors were interlocked so that only one could be opened at a time, and facilitated entry during power operation. For prolonged shutdowns, personnel could defeat the interlocks and open both of the air lock doors for access.[46] After normal shutdown when radiation was still present, access was controlled by Special Radiation Work Permits. With containment integrity established and the coolant loops hot and pressurized, access was more strictly controlled to allow only needed maintenance and testing. The PIB specified that teams of three were required with one person remaining in the air lock as a "hatch watch, but that requirement may have been waived in practice.[47]

When the reactor was at power producing levels, Continuous Health Physics Technician Coverage was mandated for each entry. Access during pressurized leak testing and under accident situations required further precautions. The level of protective clothing, and the degree of radiation monitoring, varied with the strength of the radiation field and the amount of loose or airborne radioactive particulate contamination present.

Reactor Design, Construction, and Operation

Reactor Vessel Construction

The Connecticut Yankee nuclear reactor was the source of 1,825 thermal megawatts of heat (mwt) that produced the steam to drive the turbine/generator. It was an evolutionary development of the Westinghouse reactors that drove the Nautilus (approximately 30 mwt)[h] and powered the Shippingport (231 mwt) and Yankee Rowe (482 mwt) stations. The large leap in power output was the result of intensive engineering based on the belief that nuclear power stations had to be as large or larger than the their fossil-fueled partners on the grid to be economical. The increase in power required a larger reactor vessel, but one in which nuclear and mechanical advances produced over four times the power of Yankee Rowe in a unit less than three times the size of the Rowe reactor, which weighed over 200 tons exclusive of the fuel and control elements and was over 32 feet high and 10 feet wide. The Connecticut Yankee reactor weighed almost 500 tons (dry) and was over 41 feet high and 15 feet wide.[48]

The reactor vessel that contained the heat-producing nuclear core, control rods, support internals and coolant flow channels was designed in accordance with the ASME Boiler and Pressure Vessel Code for Unfired Pressure Vessels of 1956 or later and special nuclear rulings.[49] The code specified the thicknesses of materials to resist stresses imposed by the working pressure of 2,485 psig and temperature of 650° F. The reactor was fabricated by Combustion Engineering Company to Westinghouse specifications.[50] All material was tested at the steel mill for uniformity, ductility and impact resistance. Non-destructive testing (radiographic, magnetic particle, ultrasonic and dye penetrant) was performed on raw materials and finished components.

The vessel consisted of the main lower body topped by a removable head to access the core. The lower shell was a 10.6-inch-thick manganese-molybdenum steel-walled[51] cylinder with a 5.2-inch-thick welded hemispherical bottom section. This was an advance metallurgically over the earlier reactors which were constructed of ordinary carbon steel.[52] The material was chosen because it had good resistance to damage from high neutron and gamma radiation. It also had to resist the high pressures postulated, but because it had poor resistance to the corrosive chemicals used to control nuclear reactions, the interior surface was lined with stainless steel. This was a time-consuming process in which automatic welding machines and manual welders clad the surface with 5/32-inch-thick stainless steel of type 308 and 309.[53] At the top of the vessel a 19-inch-wide forged ring flange was welded to provide a mating surface for the upper head (closure head.)

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[h] Nautilus reactor output remains classified by the U.S. Navy.

Just below the flange, four cooling water input and four output nozzles were welded to the shell in alternating pairs. The nozzles were forgings, with inside diameters of 27.5 inches where they were connected to the system piping and widened to 33.75 inches inside the reactor to lower the coolant velocity to minimize impact on the internals (Drawing HAER No. CT-185-B-9). Two of the inlet and two of the outlet nozzles had welded pads resting on base plates on the top of the neutron shield tank from which the whole vessel was supported. The nozzle mounting system allowed some movement resulting from thermal expansion and contraction. The lower hemisphere was penetrated for nineteen incore flux detectors.

The 7.75-inch-thick hemispherical, manganese-molybdenum-steel/stainless steel-clad reactor vessel head had a welded 7.75-inch-thick flanged ring which mated with the lower body ring. The head was penetrated for forty-five control rod drive mechanisms and additional spare drive locations. Since the entryways for the control rod drives breached the boundary of the reactor coolant system discussed below, they had to be highly engineered to prevent coolant loss. Each penetration was a 4-inch-diameter Inconel tube welded to the inside of the head with a threaded stainless steel adapter on the outside to attach the control rod mechanism assembly. Spare drive locations were sealed with welded pipe caps. Thermal sleeves prevented rod insertion from carrying the cooling water surrounding the mechanism into the reactor. Core deluge admission ports in the head, and instrumentation thermocouples at the top and bottom hemispheres had similar penetrations without thermal sleeves.[54] The reactor could be vented during filling by local control, or remotely from the Control Room by solenoid-operated vent valves after an accident to prevent a gas buildup from blocking coolant from reaching the core.[55] Remote operation required connection to environmental systems and redundant dc power supplies.

The reactor vessel head closure system was fifty seven-inch-diameter studs with nuts and washers. To insure that all the fasteners were adequately and evenly tightened, each one was stretched to enable the nut to be torqued down, and then the stretch of each stud was measured.[56] Lifting eyes could be threaded into the studs for placement by the crane. An O-ring sealing system was installed in two concentric grooves machined into the upper head flange. The O-rings were silver-plated metallic circular tubes lying in the grooves which were "energized" (i.e. enlarged by the pressurized reactor coolant) to form a seal against coolant in the shell forcing its way out. The outer ring was a backup seal. Leakage past the O-rings was collected from drilled holes in the flange and monitored. If leakage increased, sealing pressure on the inner seal could be increased, but leakage past both rings could require reactor shutdown. A backup, weldable, emergency "omega" seal was never used.[57] The massive forged, welded, bolted, and sealed construction of the reactor vessel was required because system heat efficiency was directly related to the 2,500 psia internal

working pressure of the coolant.[i] The ASME specifications required that the vessel be proof tested with pressurized water (hydrostatic test) to over 3,700 psig. to insure integrity.[58] After refueling or shutdowns for servicing, the reactor was pressure tested again in three stages up to 2,000 psig to check that there were no leaks in Reactor Coolant System boundary. Some time after the construction of the Connecticut Yankee reactor, research on the stresses in reactor shells brought to light potential operational problems in reactor shells that had been considered immune to failure.

During cool-down, a stress-producing temperature gradient could be set up on the inside surface by a too rapid cooling when the reactor was pressurized. The resulting combination of thermal and pressure stress known as Pressurized Thermal Shock (PTS) could lead to rupture and loss of coolant.[59] To prevent PTS, emergency operating procedures were instituted to warn operators of impending problems. Overpressure also placed stresses on the vessel and was controlled by the Pressurizer, safety relief valves and the Reactor Protection System (RPS). Overpressure in a cold condition placed even more stress on the reactor since the shell material was less ductile at lower temperature. To mitigate that risk, a Low Temperature Overpressure Protection System with redundant pressure relief valves was retrofitted to the reactor in the late 1980s.[60]

Reactor Internals

The reactor internals directly supported the fuel assemblies in the core, the control rods entering the core from the removable head, and channeled the flow of cooling water into and out of the core. Components were identified by region: the lower internals assembly and the upper internals assembly.[61] An in-core instrumentation support structure was actually split between the top and bottom hemispheres. In the lower internals assembly (also known as the lower core support structure), the core barrel and core support plate served to support the core and formed a "can" containing the nuclear furnace (Figure 10). The barrel was formed from two stacked steel cylinders welded together, and was 20.8 feet high by 12.3 feet wide.

The upper rim of the core barrel had a flange that was suspended from a ledge around the inside of the vessel under the bolting flange. The upper part of the core barrel surrounded the upper internals assembly but was considered part of the lower assembly since it stayed in place during upper assembly removal.[62] Twelve nozzles in the flange area directed

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[i] Psia referred to the pressure over true "0" as opposed to Psig (gage) which was pressure over about 14.7 pounds atmospheric at sea level. Light water reactors had the disadvantage of very low output temperature in relation to fossil-fuel boilers. Temperature could be increased by either raising the pressure (limited by manufacturing constraints to 2,500 psia = 650F) or adding superheat, which was not possible in the Westinghouse system.

cooling water to the upper head. Four nozzles in the upper core barrel mated with the four main outlet nozzles welded to the shell which allowed the coolant to exit the reactor. To facilitate core removal, there was no seal between the inner and outer nozzles allowing some coolant entering the reactor to flow into and cool the reactor head.[63] In other areas in the reactor, small quantities of coolant were directed to cool areas bypassed by the main flow. The lower core barrel around the core region was surrounded by a 4-inch-thick steel thermal shield designed to limit irradiation of the reactor vessel. The spaces between the core barrel and the thermal shield, and between the shield and the reactor shell formed annulae through which cooling water entering from the "cold leg" nozzles was forced down past the core before turning up to enter the core through the lower core support plate. During the 1989-1990 re-fueling outage, the thermal shield was removed due to the failure of some of the support fasteners.[64] This modification required the entire removal of the fuel core to the spent fuel pool (see HAER No. CT-185). The circular lower core support plate, which was welded to the core barrel, was perforated with 100 holes to allow entrance of the coolant. Supported above the plate on columns was the lower core plate which directly supported the fuel assemblies and had alignment pins to aid in loading.

Between the two plates was a perforated diffuser plate which ensured that the coolant entering the core was well distributed and also served to locate the fuel assemblies during loading. Proper distribution of the coolant flow was so critical that multilayer baffle assemblies were used to ensure that the coolant did not bypass the center of the core. The baffles themselves were separately cooled with flow through small holes in the barrel. The main radial support of the core barrel which maintained its position relative to the reactor shell were six keys welded around the bottom of the barrel which engaged in keyways bolted onto blocks welded on the inside of the shell. The main radial support which maintained its position in the core barrel was lowered into the reactor.[65] A secondary support assembly (core support vertical stop) with energy absorbing rods was a backup to the main core barrel support ledge, so that even in the "incredible event"[66] of fracture the core would not drop more than 1-1/4 inch[67], maintaining control rod orientation and insertion ability.[68]

The upper internals assembly (upper core support structure) consisted of the upper support assembly, upper core plate, upper core support columns and the control rod guide tubes (Figure 10). The upper support assembly was a circular plate located near the mating flange between the head and main reactor body. The plate was perforated to admit the control rod assemblies, core deluge pipes, and temperature measuring devices. It served as a barrier to prevent the bulk of the coolant from flowing into the reactor head. The plate had girders welded on its underside for stiffness and attachment points for columns which passed through the nozzle region to attachment points on a circular upper core plate. This plate served the same purpose as the lower core plate: to support and align the fuel assemblies. The upper internals assembly was removed and replaced as an entire unit during refueling.

The assembly was guided during insertion by flat pins and slots. Guide pins on the upper core plate extended down to engage fuel assemblies as the unit was lowered in. Axial (up and down) movement of the upper assembly was resisted by a large circumferential spring resting on top of the unit that was compressed between the reactor head and vessel mating flanges. The upper and lower internals had to resist a number of structural and flow loads. The weight of the core fuel assemblies had to be supported when the reactor was cold. That load and those from the working of control rods and provision for vertical earthquake acceleration were transferred down through the lower internal components to the bottom support forging (lower core support plate) and then up through the core barrel shell and flange to the main vessel bolting flange. Once coolant began to flow down through the annulus and up through the core (over 100 million pounds per hour), forces from water passing through nozzles and baffles which tended to raise the core had to be resisted. Transverse (cross) loads from earthquakes, coolant cross flow, and vibration were shared by the core barrel shell, core barrel flange and lower radial supports to the vessel wall.[69] In the upper assembly region, vertical loads were transmitted through the upper core plate; support columns, top support plate and then to the reactor head via the circumferential spring that was compressed between the flanges.[70]

The upper part of the in-core instrumentation support structure guided and supported the thermocouples[j] entering the head. The thermocouples measured water level in the vessel based on temperature correlations, and coolant temperature in the head and at multiple locations as the coolant exited the core. The neutron flux[k] in the core was measured by detectors entering the vessel bottom hemisphere from a sealing table outside the reactor. The detector probes in pressurized thimbles passed through welded conduit penetrations and instrument guide tubes in the vessel for insertion into the fuel assemblies.[71] The development of in-core instrumentation allowed the up-rating of reactors from their initial licensed outputs (Figure 11).[72] Materials used in the construction of the core components (and to a lesser extent the reactor vessel) had to be chosen for an ability to resist radiation damage which could include increases in hardness and tensile strength, decreases in thermal conductivity and ductility. Metals which tended toward brittleness could become more so and irradiation reduced the amount of energy needed to fracture materials.[73]

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[j] Thermocouples are thermoelectric devices invented in the early 20th century for remote measurement of temperature by utilizing the current generated by 2 different metals at a junction point.

[k] A measurement of the rate of flow of matter or energy crossing a given area in a given time (Oxford English Dictionary 1989, V: 1112.), neutron flux was the most direct measure of the fission rate -- and thus power -- in a reactor (Stephenson 1954: 283.).

Nuclear Fuel

The uranium dioxide (UO2) powder fuel was pressed into can-shaped (right circular cylinder) 3/8-inchdiameter, approximately 19/32-inch-high pellets with concave ends. The formed pellets were placed in a furnace and sintered at high heat forming a hard ceramic material with increased density and ability to contain gases produced by fission.[74] U02 in ceramic form was chosen because it was chemically inert to cladding materials and fission products while being resistant to attack by hot acidic reactor coolant water.[75] If the reactor cladding was damaged the ceramic matrix would reduce the amount of radioactive gas entry into the coolant.[76] The fuel suppliers (Nuclear Materials and Equipment Corporation and others.)[77] then assembled the pellets end to end in 10.5-foot-long, 7/16-inch-diameter wide stainless steel tubes known as fuel rods, or fuel pins, with 1/64-inch-thick walls.[78] The use of continuous rods of fuel equal to the full height of the Connecticut Yankee core gave greater fuel density than the earlier Westinghouse segmented and compartmented fuel assemblies.[79] For experimental purposes, early cores also had small numbers of Zircaloy clad rods. Zircaloy cladding and greater enrichment provided 24 months of core life. Connecticut Yankee had completely switched over from stainless cladding at plant closure.[80]

Clearance was provided between the pellet sides and the inside surface of the rods to allow the fuel to swell and to provide room for gas accumulation. The dished ends of the pellets provided additional gas space and compensated for the increased axial expansion (lengthening) in the center of the pellets where the highest temperatures occurred.[81] The fuel filled about 121 inches in the rods which were sealed by welded plugs to prevent release of fission products into the coolant. A void of about 5 inches at the top end of each rod contained a spring which pre-loaded the pellets to maintain position during assembly into groups, shipping and loading into the core. The rods were pressurized to 40 psig with helium to facilitate initial burn up of the fuel and argon to aid in the end cap welding. The raw materials and completed rods were subjected to inspections and quality control tests.

The individual rods were assembled by the supplier into groups called fuel assemblies or fuel bundles. Each assembly was a complex square fabrication which supported 204 fuel rods in a 15-by-15 array of rods, maintained separation, and channeled cooling water (Figure 10). The main physical requirement for the structure was to form a lattice which located the rods close enough together (1/8 inch) to be in the range of fission fragment travel of less than 3/4 inch.[82] It had to stabilize the long, flexible rods against displacement by the cooling water flow, allow some relative movement from expansion due to heat up during operation, and resist corrosion by the acidic boron. The components had both structural and functional capacities. The main structure consisted of top and bottom nozzle boxes and seven grid straps welded to 20 control rod guide thimbles. Fuel rods were supported by the "egg crate" grid straps and were held with spring clips which allowed axial

expansion.[83] Some of the straps had vanes to mix coolant flow. The grid straps were positioned by being welded to the rod control guide thimble tubes that guided the control rods as they were inserted and removed from the assemblies. The thimble tubing was larger at the top for admission of cooling water and for rapid rod insertion. The tubes narrowed towards the bottom to slow the rods with hydraulic resistance (dash pot action). A bottom nozzle box supported the fuel rods on cross bars, channeled the coolant evenly around the fuel rods and supported the weight of the assembly on four corner feet resting on the lower core plate. Coolant was guided out of the assembly by the top nozzle box which formed a plenum to guide heated coolant toward the flow holes in the upper core plate. The stainless steel and Inconel (high strength, anticorrosive, nickel-chromium-iron alloy) materials used in the construction of the rods and assembly structures were chosen because of their low affinity for neutrons to prevent parasitic absorption, and resistance to loss of strength and axial growth due to neutron bombardment.[84]

Control Rods

To control the excess reactivity[1] control rods that killed the reaction were assembled with the core. Once the reactor was running, some of the rods were used to control short and low-level reactivity changes, and others were reserved to trip (scram) the reactor. As the fuel burned up through the fuel cycle and the core flux changed, reactivity was controlled by precise dilution of the boric acid "nuclear poison" that was injected into the coolant loops. There were forty-five rod cluster control assemblies in the Connecticut Yankee reactor. Each assembly had twenty "rodlets" hanging from a central hub known as a spider assembly.[85] The rodlets were stainless steel tubes containing 80% silver, 15% indium, and 5% cadmium neutron-absorbing material. The cadmium was the main thermal neutron absorber with the silver also adding strength to the rodlets. The spider mounting insured that the rodlets entered the fuel assembly guide thimbles with the proper spacing and at the same rate. The assemblies were divided into two types. Sixteen assemblies in two groups were withdrawn prior to criticality and were used to shutdown the reactor, with the remaining twenty-nine in two groups serving to control the reactor power output.[86]

Control Rod Actuation

The control rods and their actuation mechanisms were perhaps the most critical component that made a controlled nuclear chain reaction possible. The rod cluster control assemblies were raised and lowered by control rod drive mechanisms which consisted of latch

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[1] To compensate for loss of fissionable material during core life, the initial and re-loaded cores had more fuel loaded than was need to maintain a self sustaining reaction. This "excess reactivity" was kept in check by control rods inserted in the core (Nero 1979: 13).

assemblies, operating coil stacks, drive shaft assemblies, and pressure vessels. The Connecticut Yankee control rod system was derived from the crude but effective devices used in the first atomic pile at Chicago. The CP (Chicago Physics)-1 pile was assembled from alternating layers of plain and uranium-containing graphite blocks with additional uranium lumps. As the pile was built up, neutron absorbing cadmium rods were inserted in holes running through the pile to prevent a reaction. When the time came to inaugurate the reaction on December 2, 1942, an automatic rod was slowly withdrawn until detection instruments signaled that a chain reaction was occurring.[87] Hanging over the pile from a rope were emergency control rods. If the reaction exceeded controllable conditions for any reason, the rope could be cut and the rods would drop into the pile via gravity and kill the process.[88] The current operators' jargon for emergency shut down, "Scram" originated there. Some sources state it stood for "safety control rod ax man"[89] others claim it was the instruction to clear out in a hurry after emergency shutdown initiation.[m] At CP-1, a team of engineers dubbed the "suicide squad" were also standing by to dump buckets of cadmium over the pile to kill the reaction.[90]

One of the factors in nuclear fission that made control rods a viable way of sustaining steady power production was the presence of delayed neutrons. Most of the neutrons produced during fission multiplied so rapidly that no control rod system could possibly act fast enough to keep the reaction from running away. Fortunately for nuclear power, nearly 1% of the neutrons were delayed from a fraction of a second to over a minute.[91] The delayed neutrons were controllable with absorber rods. PWR's needed relatively few control rods because they were self stabilizing: as temperature and pressure rose the fuel expanded, the space between the atoms increased, and neutrons leaked out.[92] This negative temperature coefficient reduced the number of fissions and slowed the reactor.

Control Rod Drive Assembly and Penetrations

Since the control rods were in the pressurized reactor but their actuating mechanisms were outside of the vessel, the head had to be penetrated for the control rod drive shafts that connected the drive mechanisms with the mounting spiders. There were 69 control rod penetrations in the reactor head of which forty-five had control rod drive assemblies, each consisting of a stainless steel 22 foot long hollow drive rod, disconnect rod, and coupling. The drive shaft penetrated the head of the reactor vessel through an individual pressure vessel with a pressure housing and rod travel housing. The pressure vessel was in contact with the reactor coolant at operating pressure and was threaded onto an adapter that was seal welded to the penetration to maintain the pressure boundary.[93] The drive shaft passed through guide tubes in the upper assembly and the upper core plate. A tapered coupling at

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[m] The Oxford English Dictionary (1989, 14: 705) states that the origin is unknown.

the lower end of the shaft mated with the socket in the control rod spider hub. When they were in position, a disconnect rod dropped down allowing a spring-activated locking button to cam into position locking the shaft to the control rod assembly. If the spring failed, the locking button was held in place by the disconnect rod. Internal Latch Mechanism and Operating Coil Stack.

The control rods were inserted and removed from the core with magnetic jacks (mag-jacks) pioneered at the Yankee Rowe plant.[94] The Shippingport reactor used canned motors driving split roller-nuts acting on threaded lead screws to pull the rods up and down. In the event of a power failure or scram signal the nuts automatically split, releasing the lead screws so the rods could drop by gravity.[95] While a reliable system, it required shaft seals and packing, and the canned motors were difficult to repair. Westinghouse believed the newer stepped-jack mechanism to be better because it eliminated water-cooled motors, rotational force on the control rod clusters, and rod over-travel from motor over speed. The concept was similar to the toothed rack-and-pawl action of railroad and wagon jacks from the late 19th century. The mag-jacks comprised grooves machined on the drive rods, moveable and stationary grippers and operating coil stacks that provided the motive power. The drive-rod grooves spaced 3/8 inch apart were cut at 45 degree angles into the rod.[96] This represented the smallest increment that the rods would travel when making small reactivity adjustments. The latch mechanism was located inside the individual pressure housing and was lubricated by the coolant bath. The latches had 45 degree angle teeth which entered the grooves to lock the drive rod onto the mechanism. An upper set of latches were known as the moveable grippers and could be activated to engage or disengage the shaft and could also move up and down 3/8 inch for each control cycle (step). The lower stationary gripper role was to hold the rod in position while the upper set was being repositioned. In fact it had 1/32 inch of travel to make a smoother transition of load. Each set of latches had pole pieces which acted as the moving element of a solenoid electro-magnet.[n] The motive power for the latch mechanisms was the operating coil stack which was slid over the outside of the pressure housing adjacent to the latches. Electric current pulses, sent into the coils in an on or off sequence, attracted and moved the pole pieces on the latches via magnetic induction through the wall of the housing. The moveable gripper coil engaged the moveable gripper teeth, the lift coil moved the rod, and the stationary gripper coil engaged and disengaged the stationary gripper from the drive rod. Another benefit of the design was that the coils were not contacted by the corrosive reactor coolant and were accessible for repair without breeching the reactor coolant system boundary.[97]

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[n] Solenoids were electro-dynamic devices from c1830 in which an iron plunger rod surrounded by a coil of copper wire was moved when DC current was applied to the coil (Oxford English Dictionary 1989: v.13, p. 556, v. 15, p. 964).

Control Rod Mechanism Operation

At start-up, operators in the control room (see HAER No. CT-185-F) withdrew the sixteen shutdown/starting rod clusters completely from the reactor. During most operation the rods were held slightly in or just above the core in the upper assembly guide tubes in a static position by the moveable gripper coils. Rods used to control slight changes in reactivity or power output were moved one step out of the core in the following sequence. The lift coil was energized raising the rod one step. The stationary gripper coil then induced its gripper to engage and lifted the rod 1/32 inch to transfer the drive rod weight from the moveable to the stationary gripper. The moveable gripper coil shut off releasing its gripper from the rod. The lift coil then shut off causing the gripper to move down one step. The moveable gripper coil was re-energized causing its gripper to engage the shaft. The final action was the stationary gripper coil de-energizing allowing the it's gripper to move down 1/32 inch transferring the weight of the rod back to the moveable gripper, with the coil stack ready for the next adjustment. The jacking action could be repeated up to forty times per minute giving a maximum rod movement of 15 inches per minute.[98] Parts of the mechanisms working in the coolant were fabricated of stainless steel, Inconel and cobalt alloys for corrosion resistance.[99]

The critical reactor trip function (scram) was caused by de-energizing all three coils, either with a trip signal from the control room, primary, and secondary system components or a power failure (fail to a safe position). Since the grippers were engaged by electro-magnetism, as soon as power was cut off, the weight of the drive shaft and rod cluster assembly forced the latches out of engagement and the shutdown rods dropped into the core from gravity.[100]

Power Supply

Since the rod actuation was by electro-magnetism, a source of DC power specifically for the Control Rod Drive Mechanisms (CDRM) was provided by two redundant motor-generator (MG) sets in the "A" switchgear room (see HAER No. CT-185-F).[101] MG sets with an electric motor on a common shaft with a generator were a common way to raise or lower DC voltage, change from AC to DC or DC to AC) current, dating from the earliest electric power stations. At Connecticut Yankee, multiple buses fed the 480-v AC 150-hp induction motors powering AC generators which provided 125-v synchronous power to diode rectifiers for conversion to DC.[o] Flywheels were attached to the shafts to provide energy to continue rotation for one second after a power interruption. The two units operated in

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[o] Rectifiers utilizing semi-conductors were a preferred source of DC as they were easier to service and repair than DC generators. (.....2006: personal communication.)

parallel (both supplying the load) so supply would continue with the loss of one unit. The DC power was routed through isolation circuit breakers to the scram breakers operated from the control room and then to the CDRMs. On signal from the control room or the Reactor Protection System, two scram breakers cut off power to the three coils of all 45 CDRMs allowing the rods to fall into the reactor. Actuation was via various process signals which energized trip coils and de-energized under voltage cells.

Rod Control System

The rod control system was a complex feedback system which controlled the output of the reactor, to make the appropriate quantity of steam to meet the varying needs of the steam turbine governing system as signaled by the electrical load on the generator. Inputs into the system included the amount of heat removal from the reactor, the volume of the reactor coolant which was controlled by the pressurizer, and the amount of steam raised in the generators which was effected by the coolant level and the amount of feed water admitted. Control circuits managed the system by averaging a T (temperature of reactor coolant)-hot and T-cold signal to produce a T-avg (average) signal. Three signals from the coolant loops were averaged to provide an overall plant T-avg which was then compared to a T-reference signal based on the turbine load.[102] The signals were measuring a reactor to turbine power mismatch and then controlled the flow of power to the CDRMs to move the rods in or out a specific distance and speed to maintain appropriate reactor coolant system temperature and steam system pressure as electrical power changed. The system had to allow necessary changes in load without protective tripping. In order to maintain that ability over the life of the core, the poisoning value of the control rods had to be closely monitored. Complex calculations determined the "rod worth" at various locations in the core, and for different boron concentrations to give shutdown margins of 1-3%.[103]

Rod Position

It was critical for safe operation of the reactor that the operators know the positions of the control rods. Two independent systems were used to allow comparison between indications.[104] One system monitored electrical pulses to the lift coil measuring the rod group height and displaying it on the main control board with odometers (step counters) the 3/8 inch movements (steps) made on the rods by the latches. The actual position of each individual rod was detected by the analog position indication system using linear transformers measuring relative movement between the rods and primary and secondary coils on the control rod housings. The electrical output in proportion to rod position was sent to the plant computer and the main control board (see HAER No. CT-185-F).[105]

Reaction Moderation

Even with fuel rods arrayed in the most efficient manner, and enrichment carried to the highest economical level, there was no chance of a chain reaction beginning. To make the chain reaction possible with the given amount of fuel and its level of enrichment, a moderator was used between the fuel elements to slow down (thermalize) the fast neutrons. The slowed neutrons were more likely to be absorbed by nuclei in the fuel which effectively had a higher fission cross section.[106] Moderator efficiency had a direct influence on the design of the reactor, since it affected the distance neutrons could travel before causing fission of the U235 nucleus which was a constraint on the spacing of the fuel rods and the size of the core.[107] Neutrons were slowed by collisions with moderator nuclei in a process called elastic scattering giving them a zigzag path.[108]

The analogy used to describe the process has been termed the billiard ball principal.[109] At Connecticut Yankee, ordinary (light) water was used as the moderator. In addition to being cheap (in fact obtainable from wells on the site), it could serve in two roles since PWR's were the only reactor system in which the same substance could serve as both moderator and coolant.[p]

Soluble Absorber

In the Shippingport reactor, some of the thirty-two control rods corrected gradual changes in coolant temperature and could also adjust fission levels after large load changes. Twelve rods were fully inserted and were only removed to compensate for fuel burn up and xenon buildup which contributed to the poisoning of the reaction by absorbing so many neutrons in non-fission reactions that criticality could not be maintained.[110] Yankee Rowe, initially adjusted reactivity at start-up with "shimming" rods and boric acid added to the coolant. Later, Westinghouse introduced a new control method there, which was perfected at the NRTS known as "chemical shim."[111] Long-term reactivity was instead controlled by

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[p] Absorbing the energy of a particle of a particular mass and size was best done by using one of the same parameters. A cue ball hitting an eight ball could impart all its energy (and be slowed down) by a direct hit and even a fair amount from a glancing blow. However, a ping pong ball hitting a billiard ball even head on would bounce off with nearly the same energy and velocity imparting very little motion to the heavier object. Light elements such as hydrogen were the best moderators from the energy absorption standpoint, but the hydrogen atoms in light water were not as effective as those in heavy water (Weinberg and Wigner 1958: 181, 184; Landis 1955: 3) The light water hydrogen atoms wasted neutrons by capture without producing much energy (Nero 1975: 6). Fuel enrichment was the only solution to the relative inefficiency of light water as a moderator (Stephenson 1954: 60).

maintaining a precise level of boric acid in the coolant on a permanent basis with control rods almost fully withdrawn from the core.

The system reduced by half the number of control rods needed.[112] At start-up, the level of boric acid was high to counteract the excess reactivity and could prevent a run-away reaction even if the control rods were fully withdrawn. During a fuel cycle, the levels were reduced to compensate for the burn up of fuel and buildup of fission product poisons. Due to the large volume of coolant and impossibility of changing the boron level quickly, some control rods were active in the core to control sudden spikes in reactivity. Control of boron admission occurred in the boric acid system where valves were used to vary the rate and control the amount of boron added. Highly concentrated boric acid was stored in a tank in the Primary Auxiliary Building (see HAER No. CT-185-G).

Neutron Source Rods

With the critical mass present in the first fully-loaded core and subsequent cores with all new fuel, neutron activity in the fuel rods was low enough that with the core full of moderator and the starting group withdrawn the reaction would not reach criticality.[113] To initiate startup in those conditions, primary and secondary neutron source rods filled with antimony and berylliums producing fast neutrons were installed in four of the unused control rod thimbles.[114] The rods provided neutron flux to aid in measuring small changes in core reactivity during startup. Their half life of several months[q] provided enough neutrons to monitor reactivity during shutdown.[115]

It was important to concentrate cooling water around the fuel rods so guide thimbles without control or neutron source rods were sealed by spring compressed thimble plugs. The plugs were short rods mounted on a flat plate which was assembled in the top nozzle of the designated bundle. They were securely held by an upper spring pack that was compressed by the upper core plate when the upper internals were lowered into the reactor.

Fuel Loading

The fuel assemblies were shipped to the plant in containers specially designed to maintain the fuel in a subcritical condition.[116] Neutron activity was very low in the assembled bundles and their quantity was limited. However caution was needed to insure that

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[q] Unstable (radioactive) nuclei disintegrate or decay over a specific period of time. A measure of this type of decay is the "half life", the time required for ½ of the active material present to decay. While theoretically infinite, ten half lives rendered the activity negligible. Radon gas with a half life of 3.82 days would be largely decayed in 30 days. (Oxford English Dictionary 1989: 4: 322, Stephenson 1954: 23).

conditions that would foster criticality did not occur by accident. It was essential that the geometry of the bundles not be altered, and that the bundles remain dry, since water could serve as a moderator to slow neutrons to thermal velocities, increasing the chances for "capture" and subsequent fissioning. The containers were unloaded upon delivery to the site and stored in the new fuel racks adjacent to the fuel pool (see HAER No. CT-185-J). There were 157 fuel assemblies in the Connecticut Yankee core design. For the first core, assembly loading was done dry, a common practice during initial construction to facilitate fitting of all the components.[117] The bundles were loaded into the core with the manipulator crane operating on the charging floor. The lack of a moderator and the presence of control rods inserted into the groups insured sub-criticality during core assembly. The total thermal power projected required determination of the number of fuel rods needed to give a total rate of fissioning, and their surface area for the rate of heat transfer.[118] Other factors that had to be considered were absorption, diffusion and leakage of neutrons; burn up of the fuel, and coolant flow. The enrichment levels of the fuel were varied to provide even "burn up" and power density over the year-long fuel cycle. The core had three concentric regions of 3.00, 3.24, 3.67 % enrichment (with higher enrichment on the outside) which facilitated a core cycle in which only one third of the assemblies were replaced in each refueling.[119] To further facilitate even fuel burn-up in early cores, the pellets in outer region were less dense to allow for greater swelling from radiation damage (possibly deleted in later cores).[120] All those factors were taken into account in determining the critical mass of nuclear fuel to provide that at least one neutron would be produced from each fission to maintain a chain reaction during the projected life of the core. To provide for a long core life, more fuel was added (excess reactivity) than was initially needed to maintain a chain reaction. With 157 204-rod assemblies, there were 32,028 rods with over 82 tons of UO2 in the core.[121]

Final Reactor Assembly

With all the fuel, control, and neutron source rods in place, the final assembly of the reactor could take place. The upper internals were lifted by the polar crane and lowered into the reactor vessel with special lifting rigs. The assembly was guided into final position by flat sided pins in the core barrel.[122] Upon completion of upper internals installation, the control rod drive shafts were installed and the reactor head was lifted in position and bolted down. The coils for CDRMs and the position indicators were already permanently installed on the reactor head. A missile shield was installed above the reactor which also supported plugged cable connections between the head and the charging floor for rod control, measurements, and reactor venting valve wiring. Following control rod assembly, moderator water was introduced into the reactor and pre-start-up circulation was begun. The boron concentration was precisely set at 2,470 ppm. which in combination with the fully-inserted control rods gave a shutdown margin of 9%.[123]

The Chain Reaction

The start-up procedure required a calculation to determine the critical condition for the combination of boron concentration and the height of the rods out of the core. Reactor coolant loops had to be operating, water inventory had to be over 230,000 gallons, and all systems readiness had to be verified. Two Reactor Operators and a Senior Reactor Operator had to be present during the process. Control rods were withdrawn by banks until the neutron count doubled. Rod positions were checked for alignment.[124] With the rods almost completely withdrawn, the boron levels were gradually diluted until the reactor reached a "just-critical" state after which low power physics measurements were taken.[125] Once criticality was achieved the self-propagating reaction began. On the nuclear level, neutrons flying from fuel rod to fuel rod were slowed by collisions with protons in the moderator nuclei and some of those thermalized neutrons were absorbed by nuclei in the fuel. The resulting instability caused the nuclei to split producing one to three highly energetic neutrons, one of which had a good chance of causing another fission after being slowed by the moderator.

In a balanced (continuing) reaction the extra neutrons had to disappear without causing more fission.[126] Some were absorbed by fuel nuclei which were converted to new fissile material elements, some of which (plutonium-239 or U233) could be reprocessed as fuel.[127] Other excess neutrons were absorbed by the coolant, reactor materials, or escaped from the core. The fission of U235 also usually produced two fragments of slightly different mass (new elements) which traveled less than 1/10 mm in the fuel. The fragmentation yielded radioactive fission product isotopes such as krypton, strontium, iodine, and cesium. Gamma and beta rays were also produced which contributed to physical damage to the fuel. The main source of the heat produced in the reactor (80%) was the kinetic energy of these recoiling fission fragments as they came to a stop in their surroundings.[128] Additional heat production came from absorbed gamma rays, fast neutrons and fragment radioactive decay.[129] Ultimately, almost all the fission products were deposited in the fuel rods, the moderator or the reactor materials.[130] As the reactor reached full power, fission resonances in the fuel broadened allowing the capture of neutrons of different energy levels, a condition known as the Doppler effect. The Doppler coefficient is an important factor in reactor control and contributes to the resistance of the core to overpower transients or criticality surges.[131] The start-up procedure was controlled by the Technical Specifications[r] to reduce thermal stress on the reactor by limiting heat up to 60° F. per hour.[132]

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[r] Federal requirements for nuclear plant licensing included specifications derived from safety analysis covering safety limits, limiting safety system settings, and limiting control settings (e.g., 10 CFR 50.36).

Heat Transfer

The ultimate limit on power production in light water reactors was the maximum temperature obtainable and allowable in the center of the fuel pellets. While in theory the potential heat was practically unlimited, the constraint was the temperature that cost-effective materials and manufacturing processes would allow. The hard ceramic fuel pellets melted at 5000o F. but the requirement that they retain as much of the gaseous fission products as possible limited their operating temperature to about 3600 F. Another constraint was the stainless-steel fuel rod cladding which had to remain intact, limiting the total temperature to around 2,000° F.[133] The amount of heat production was limited by the temperature differential between the central hottest part of the fuel pellets and the coolant surrounding the cladding.[134]

When PWRs were first proposed, there was concern that if boiling occurred around the fuel rods, heat transfer would be lost leading to cladding failure. The cladding contained the gases, fission fragments and new fissile materials in the fuel pellets during the life of the core during storage after refueling. It was determined, however, that a form of boiling called nucleate boiling could occur in which small bubbles forming and leaving the surface of the fuel rods were rapidly replaced with more water, giving a large heat transfer coefficient and safe cladding temperature. [135] That also served as a cladding temperature constraint since a rise could cause a condition called departure from nucleate boiling (DNB)[136] where a complete blanket of vapor (film boiling) formed on the rods creating a 1000% loss of heat transfer ability and cladding damage.[137]

Once DNB began, overheating could propagate in a cascading situation. If the fuel overheated due to DNB or a loss of coolant flow, the pellets would swell and the cladding could perforate allowing the fission products (radionuclides) into the coolant with risk of release into containment or secondary systems and then into the atmosphere. Leakage from the reactor coolant system was minimal while little or none was expected to leak from containment. The most likely path to the environment was though leaking steam generator tubes.[138]

To ensure safe operation of the reactor, designers utilized a concept called "hot channel factors" to set peak conditions that got the most power out of the core without exceeding mechanical and safety limitations. They performed complex calculations to determine gross radial and axial power distribution through the core and between fuel assemblies. The response of the core to load changes and operator adjustments was evaluated through reactivity coefficients coupled to the neutron multiplication factor and variables outside the core.[139]

Fuel Burn up

In a typical converter (as opposed to a breeder) reactor like Connecticut Yankee, the conversion ratio of fissile material produced to that destroyed was about 0.6. At the start of the first fuel cycle, all the nuclei undergoing fission were of U235. For each slow neutron absorbed by U235, two fast neutrons are produced, being then slowed by the moderator to thermal velocity. Some were captured by U235 and the reactor environment (coolant/moderator, structural materials, control rod and chemistry poisons,[140] while a large number were captured by U238 producing plutonium. As the reactor ran, fission product poisons built up and fissile material decreased, but the conversion ratio did not change much. The amount of energy extracted from the fuel was known as "burn up" and the first core was projected to provide 14,100 megawatt days per metric ton of uranium (Mwd per MTU). In subsequent cores, the burn up was raised to over 25,000.[141] The end of the cycle (core life) was defined as the point when the core was just critical producing 1,473 Mwt with no boron in the coolant and the control rods fully withdrawn. At the end of some fuel cycles, plant managers applied to the AEC/NRC for permission to "coast down" to continue power production if there was a peak in electricity demand. In that procedure (also known as stretching), the reactor continued to operate with the power level dropping to 90% before shutdown.[142] When shutdown for refuel (or maintenance) became necessary, it was important to have a slow, controlled cool-down to ease thermal stress on the reactor and the Reactor Coolant System components.

Refueling

As the fuel cycle approached the point where the thermal megawatt output was dropping past full power (unless coast-down was in effect) with the control rods fully withdrawn and boron concentration at zero, preparation began for refueling the reactor. With the completion of cycle I, further re-fuelings were done with the core and the refueling system immersed in borated water with the aid of remote-controlled manipulation devices.[s] Each new core had a complete analysis of core power distribution, control rod programs, hot channel factors and boron concentrations.[143] Refueling was considered a process rather than a system, covering two structures and utilizing many systems and subsystems.[144] At the beginning of the plant life Westinghouse and other fuel vendors performed the refuelings under contract with Connecticut Yankee staff "controlling the process"[145] The main structures used in refueling included the reactor cavity around the head of the reactor, the fuel canal between the wall of the cavity and containment shell, and the transfer tube that

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[s] The complete replacement of the core after the thermal shield removal was done with the cavity immersed to maintain acceptable radiation levels for workers operating remote handled tools above the reactor cavity on the bridge (2006:...Personal communication .

connected the canal and the spent fuel pool in the New and Spent Fuel Building (Drawings HAER Nos. CT-185-B-12 through CT-185-B-13; see HAER No. CT-185-J).

In preparation for refueling the reactor had to be cooled to under 140° F. The head studs in the flange were removed and stored on the charging floor. Since the entire area around the top of the reactor was going to be flooded, the cavity seal around the vessel at the flange was installed and bolted down. As discussed in HAER No. CT-185 there were several generations of cavity seal. The second design failed during the 1984 refueling leading to modifications and then to a permanent seal used until plant closing. The permanent seal required flow-ports (to allow water spray accumulation to pass into the sumps in the event of a loss of coolant accident during operation) to be closed before the cavity was filled with borated water. The water level of 46.5-feet (equal to 230,000 gallons) was at the same elevation maintained in the spent fuel pool, ensuring there would be no flow forces in the system during refueling operations. As water was admitted to the cavity, the head with control rod drives detached was lifted off the reactor vessel by the polar crane which continued to keep the head above the rising water level. The head was then lowered to the floor of containment. Control rod drive shafts projecting out of the top of the reactor were removed and stored in racks along the wall of the cavity. The upper internals package was then removed and stored in the cavity south of the reactor.

Fuel Unloading

With the multi-region core re-fueling concept, one third of the spent fuel rods were removed and replaced at the end of each cycle. In addition, older but still active fuel was moved inward. The process was called a fuel "shuffle", for each spent assembly removed and stored in the fuel pool; a new assembly was removed from the pool and loaded into the reactor.[146] Fuel removal was done by the manned manipulator bridge crane that spanned the cavity and ran on rails on an east-west axis.[147] Travel in the north-south direction was provided by a trolley riding on rails on top of the bridge girder. A two-part vertical tubular rotatable mast on the trolley could be positioned over any assembly location. An auxiliary hoist on the trolley removed the control rod drive shafts from out of the core. An air-operated gripper picked each spent fuel assembly out of the core and raised it up inside the tube. A load cell device was the primary means by which the operator knew that the assembly was suspended in the hoist. With the fuel assembly in the tube the crane traversed over to the fuel canal. The crane lowered the assembly into the canal where operators could either position the assembly for transfer to the spent fuel pool or to have the rod control cluster assembly removed.

These operations occurring in the canal were observed from the charging floor with a swiveling optical periscope.[148] Specialized tools for handling drive shafts, spent or new fuel, and for irradiation sampling were hung from the crane. Assemblies needing control rods

removed were placed in a three compartment cage in the fuel canal. The cage could traverse on floor rails to bring the compartments under a fixed position hoist. Air operated grippers pulled out the spider and rodlets and either stored it in the cage or placed in another assembly.[t]Assemblies going directly to the fuel pool were placed in a fuel transfer basket on the floor of the canal that was brought into a vertical position by a winch powered upender frame. The upender then placed the fuel assembly in a horizontal position over a wheeled fuel transfer car sitting on rails 7 inches apart laid from the fuel canal, through the transfer tube and into the spent fuel pool. The car was moved into the transfer tube by a motor mounted on the charging floor operating a chain drive system and pusher arm fixed to the car (Drawings HAER Nos. CT-185-B-12 through CT-185-B-13).[149]

The transfer tube was actually two concentric tubes over 15 feet long, extending to the west side of the spent fuel pool and breeching the walls of containment and the spent fuel pool by one foot. The outer tube was welded to the steel liners in the canal and spent fuel pool, and supported the 20-inch-diameter inner tube with bellows fasteners. Ten inches of lead shielding surrounded the section of tube passing through the gap between the wall of containment and the fuel building.[150] A manually-operated gate valve closed off the Containment side of the tube and an air operated sluice gate valve sealed the end at the fuel pool. For backup between refuelings a blank flange was bolted on the manual valve preserving containment integrity during operation.[151]

On entering the spent fuel pool with an assembly, the car stopped under an upender that raised the assembly to a vertical position. A bridge crane with two trolleys spanning the entire floor area of the fuel building was used to remove the fuel assembly from the basket and place it in a storage rack on the floor of the pool. As in Containment, specialized tools were hung from the crane and operated by compressed air from the floor of the building. All assemblies and control rods being moved had to remain at least 8 feet under water.[152]

New fuel coming into the plant in shipping containers was stored in racks in a depressed well area adjacent to the fuel pool. When it came time to load in the first new assembly, a hoist on one trolley picked up the fuel and placed the assembly in the new fuel elevator basket which lowered it to the floor of the pool. The other trolley then placed the assembly in the upender, repeating in reverse the spent fuel removal process. The use of a separate hoist for transferring the new fuel to the pool prevented tools used to handle the new fuel from being contaminated by contact with the spent fuel assemblies. Some spent fuel was

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[t] After the cavity seal failure there was an operational change prohibiting more than one fuel assembly in the Rod Control Cluster Assembly (RCCA) change-out fixture at a time. Later all change-outs were done in the spent fuel pool (Connecticut Yankee Atomic Power Company 1987-1995: Chapter 9, page 39.)

shipped to General Electric and Battelle during the plant's service life. For that purpose, the spent fuel pool had a presumably designed-in cask pit in the south east corner. The shipping casks were lowered into the pit and spent assemblies were placed in racks in the casks. Lids were attached and the casks were raised out of the pool.

As discussed in HAER No. CT-185, a number of physical and operational changes were made in the refueling system as a result of the 1984 cavity seal failure. The NRC required that at least one Residual Heat Removal Pump and Heat Exchanger be operating or operational during the refueling process. The new cofferdam in the transfer canal meant that assemblies in the vertical position in both buildings would have at least 2 feet of water over them in the event of complete reactor cavity drainage. That height would have allowed spent fuel decay heat to boil away enough water to uncover the fuel requiring operators to implement the Seal Failure Emergency Procedure. Actions included adding more water and restoring spent fuel pool cooling and could be implemented from outside the building.[153]

Reactor Coolant System

The Reactor Coolant System (RCS) provided a flow of pressurized water into the reactor that had the triple role of moderating the fission reaction, cooling the reactor core, and removing the heat it produced for the generation of steam to drive the turbine generator.[154] In addition the system removed decay heat from the core following shutdown, served as a barrier against the release of fission products, was the pathway for reactivity control by boron, and provided overpressure protection to the reactor. The main components were the four coolant loops each with associated piping, main coolant pump, and steam generator. A single pressurizer supplied all four loops. By-pass lines, motorized stop valves, check valves, vents, and drains in the system insured reliability and environmental integrity. Chemical addition to the water from the Chemical and Volume Control System (CVCS; see HAER No. CT-185) controlled nuclear reactions, mitigated corrosion, and fouling of heat transfer surfaces. Total water inventory of the RCS was 250,000 gallons minus the water in the reactor vessel.

The system was designed to move 6,228 x 106 Btu of heat per hour from the reactor to the steam generators. Total flow of the four loops was over 100 million pounds per hour of water through the reactor, removing the heat and increasing the temperature 38° F. from inlet to outlet.[155] When the plant was operating at full load, the average water temperature was 564o F. in the vessel and the designed minimum steady state pressure was set at 2,125 psig.[156] Actual operating pressure was regulated by the pressurizer between 1950 and 2050 psig.[u] The high pressure prevented the coolant from boiling. While some localized surface

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[u] Temperature and pressure parameters as stated in FSDA and PIB documents may vary with actual operating numbers as supplied by CY personnel due to changes in procedures over time.

boiling was allowed around the fuel rods, bulk boiling had to be prevented in the PWR design to insure that the rods did not overheat.[157]

Piping and Valves

All the piping connecting the RCS components was constructed of stainless steel, and was insulated to maintain the temperature in the containment building. Most of the piping connections were welded (as opposed to the easier to assemble but less reliable bolted flange method) with the exception of certain valves.[158] Connections in which cooling water could be subject to sudden changes in temperature were insulated with a layer of stagnant water around the weld which reduced thermal shock in the area. The piping was sized (27.5-29.5 inches diameter) to provide adequate flow (40 feet per second) while minimizing erosion.[159] In the four-loop Westinghouse system, the reactor could operate at reduced output (65%) with one of the coolant loops shut down for repair.[160] While the plant operated in this way several times early in its history, the high cost of core analysis for each cycle ended the procedure.[161] Isolation was controlled by two reactor coolant loop stop valves (loop isolation valves[v]) for each loop on the cold and hot legs. These were 23-inch electric-motor- operated gate-type valves in which a plate wedge was forced onto seats to block flow.

The stop valves (and most of the valves in the system) were fabricated of stainless steel and welded on to the pipes. Recirculation flow in an isolated loop was controlled by loop bypass valves. The small flow through the 6-inch valves maintained a stable temperature and boron level in relation to coolant in the other loops.[162] An overpressure spring closing check valve on each loop served as a safety valve when the loop was isolated. Most valves had bonnets (removable heads) that could be welded closed in event of leakage during operation.[163] Since operation of the valves was not vital to plant safety, they were not supplied by uninterruptible power supplies and would fail if the Motor Control Center #5 (MCC-5) 480 ac bus tripped out.[164] The MCC-5 did have two sources of power and was provided with an automatic transfer device.

Reactor Coolant Pumps

One of the most difficult engineering challenges that engineers had on the Nautilus project was the design of the pumps supplying the coolant flow to the reactor. They had to be utterly reliable while handling irradiated water and have essentially zero leakage. The only

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[v] Terminology frequently differs between the main sources, Connecticut Yankee Atomic Power Company 1966-1974: 5.1.4-3, and the Plant Information Book (Connecticut Yankee Atomic Power Company 1987-1995: Chapter 2.1, Page 19).

type of pump capable of handing the high flow requirement was a centrifugal type in which a rotating impeller moved water through a spiral shaped casing.

The point where the motor shaft entered the casing to drive the impeller had to be sealed against leakage with a gland, and the pump motor had to be cooled. In conventional boiler plants minor water loss through seals was not a problem, but such loss could not be ignored with irradiated cooling water. For the Nautilus, Westinghouse in conjunction with Allis-Chalmers and Bryron Jackson[w] developed and supplied vertical "canned" motor pumps in which the electric motor was hermetically sealed in a long, slender, pressure vessel "can."[165] The motor had bearings that were lubricated with the primary water. The windings were water cooled from a separate system with a heat exchanger. Shippingport and Yankee Rowe had the same type of units but the 1825-hp Rowe pumps reaching the limit for that type. Connecticut Yankee was one of first PWRs to use a new type, known as a controlled leakage or shaft seal pump, first used at the Big Rock Point plant in Michigan. They were designed with efficient multiple seal section on the drive shaft between the motor and pump. The new pump design was cheaper, more efficient, and could be repaired in place, unlike the earlier design. Shaft seal pumps could also accommodate an important safety feature; a large, heavy flywheel which provided seconds of additional pumping power from inertia in the event of a power failure.

In addition to forcing flow through the reactor, the pumps supplied the pressurizer spray and ensured that the Boron injection for reactivity control was evenly mixed. One pump was left running during shutdown to ensure those functions continued. The reactor coolant pumps (RCPs) were quarter-sized[x], 61,900[166]-67,000[167] gpm, 2,485-psi units. The pump casing was a stainless steel, cast and welded 8.5-foot-diameter hollow vessel. Water entered from the bottom into the impeller region. The rotating cast stainless steel impeller had vanes which increased the velocity of the water and spun it outwards into a diffuser section which converted the velocity into pressure (known as static head) by increasing the flow through

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[w] The Milwaukee firm of E.P.Allis was founded in 1861. Merging with Fraser and Chalmers in 1901as Allis-Chalmers, the company made steam engines, pumps, generators and both water and steam turbines (Fehring 2006: 31) Byron Jackson was founded in 1872, and seven years later set up shop in San Francisco manufacturing deep well turbine pumps, submersible centrifugal pumps for industry and fire protection. Later they specialized in providing oil field tools and services. The company is now known as BJ Services (BJ Services 2002: 1).

[x] Pumps and motors in the plant were classified as quarter, half, and full sized. There were four RCPs which meant they all had to be operating at full load. The two feed pumps were half size: both were needed at full power. Each condensate feed pump was full size so only one had to run for full output (Houff 2006, van Noordennen 2005-2006).

multiple passages.[168] Hot reactor water in the casing could damage the lower impeller shaft bearing which was protected by a thermal barrier and a coiled tube heat exchanger cooled from the component cooling water system ([CCW] see HAER No. CT-185). The casing was attached to the motor assembly by a main pump flange which was the upper pressure boundary.

The complex seal system[y] that made the newer style of pump possible was in three sections. It was known as a controlled leakage mechanical seal assembly. The main seal (#1) was made up of a runner which rotated with the shaft and a stationary ring attached to the seal housing. A .00045-inch gap (about half the thickness of household aluminum foil) between the runner and the ring allowed a leak rate of approximately 2 gpm at full reactor pressure.[169] Hydrostatic forces (2000 psi) from the coolant forcing its way up the shaft were balanced so the runner/ring gap was constantly maintained with leakage exiting at 50 psi. The #2 seal was a backup with a graphite rubbing seal in a stationary stainless-steel ring and rotating runner which reduced the leakoff pressure. If the #1 seal failed, the #2 seal would control leakage long enough for a controlled shutdown of the RCS. The #2 seal may have been changed to one similar to #1.[170] The #3 seal was also a ring/runner type serving as a vapor shield to prevent leakage from entering the containment building and also to prevent pressurized containment atmosphere from entering the pump.[171] A high-pressure seal injection system forced water from the Chemical and Volume Control System (CVCS) through filters and into the seals to prevent high-temperature reactor coolant from entering the seal chambers.[172]

If seal injection or bearing cooling from the CVCS was lost, the pumps had to be shut down. In normal operation a number of pump support systems had to be functioning before the pumps could be started. The PIB permitted startup procedures to be waived to provide core cooling in an accident however plant operators stated that was not the case.[173] Motive power for the pumps were 3,300 -hp (at plant load), 1,180-rpm AC induction motors of standard construction.[174] Power for the motor came from the 4160-v AC system, with automatic transfer of the loop 2 and 4 motors to offsite power in the event of a reactor/turbine trip.[175] Unlike the earlier canned pump motors, the Connecticut Yankee units powering shaft sealed pumps could be air cooled from a fan on the pump shaft. The upper shaft bearing above each motor was a thrust type with floating babbitted[z] pads and shoes

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[y] The section in the Plant Information Book ([PIB] for the guidance operators) on the shaft seal, its auxiliary systems and controls took up 22 pages, and the importance of the Reactor Coolant Pumps was confirmed by the fact they merited one of the largest chapters in the PIB.

[z] Babbitt antifriction metal was a lead and tin alloy patented in 1839 (Bourne 1846: 228; Knight 1877: 205).

(based on the Westinghouse-marketed Kingsbury bearing of 1911[176]) and carried the combined weight of the pump and motor.[aa] [177] An oil-lift system forced oil between the bearing runner and support shoes, raising the shaft to ease starting effort. While not called a thrust bearing, the lower bearing had the same floating pad construction. Both bearings were operated in an oil reservoir insuring reliable lubrication. Heated oil was cooled in an external heat exchanger. The coast down flywheel was mounted on top of the motor shaft and weighed 7,900 pounds, providing about 45 seconds of pumping power to remove residual decay heat from the reactor after the loss of alternating current power.[178] Pawls on the flywheel rim engaged teeth on a stationary ring when the pump was stopped to prevent reverse rotation and coolant backflow through the pump if other pumps were running. The motor, shaft, and impeller were removable from the unit for repair without disturbing the piping.[179] The mounting supports for the 80-ton pumps with their motors remain undocumented. Each of the pumps supplied cooling water to the reactor through a 27.5-inch diameter "cold leg" welded to a nozzle on the vessel shell.

Coolant Flow in Reactor

Coolant at between 535E F and 554E F (depending on the load) entered the reactor through four "cold leg" nozzles and flowed down the annulus between the core barrel and the vessel wall.[180] The flow was divided by the thermal shield until it was removed in 1989 (see HAER No. CT-185). Coolant then flowed into a plenum in the hemispherical reactor bottom and up into the core. The intermediate diffuser plate, lower core plate, nozzles in the fuel assemblies, and baffles in the core region channeled the flow closely around the rods to insure even heat removal. Ninety-one percent of the coolant flow was used to remove heat from the core.

The remaining bypass flow cooled the outlet nozzles, control rod guide thimbles, the space between the core baffles and the barrel wall, and the upper head flange. The calculated pressure drop between the entrance and exit nozzles was 38 psi.[181] The heated water (583E F designed, 575E F actual operating level[182]) was directed out the four "hot leg" discharge nozzles into 27.5-inch diameter piping leading to the steam generators.

Steam Generators

The primary function of the four steam generators was to transfer heat from the reactor coolant to the secondary water to produce steam to drive the main turbine and auxiliary steam loads. They formed a part of the Reactor Coolant System Boundary - the irradiated coolant was in principal, not allowed to mix with the steam raising feed water. In their shell

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[aa] A similar design was patented by A.G.M.Mitchell in Australia before 1908 and adopted in marine and land power plants (Engineering 1908: 833, Muyderman 1966: 63)

and tube construction they were similar to many power plant heat exchanger auxiliaries (feed heaters or condensers) in which a fluid traveling in tubes heated or cooled a fluid or gas flowing around them (or vice versa) in the shell. They were a necessary auxiliary to the PWR because in that reactor technology, pressure in the reactor prevented boiling (steam raising) which had to occur in separate vessels.[bb] The relatively low temperatures produced in the reactor and the narrow range of coolant temperature increase through the core meant that the heat exchange required an extremely large surface area. Over 3,700 ½ inch tubes with walls under 1/16 inch thick were needed to provide the required steam generating surface (27,000 square feet).

The four steam generators were 45.9-foot-high pressure vessels constructed of carbon steel (Figure 12).[183] Since the inside surfaces of the three to four inch thick shell were not exposed to acidic coolant water they were not clad with stainless steel.[184] The pressure and temperature design parameters were different for the primary side reactor coolant flow and the secondary side feed water flow and steam output. The generators were mounted in pairs on either side of the east/west shield wall close to the coolant pumps, with the upper portions projecting above the charging floor. The 217-ton vessels were provided with side mounting trunnions to allow for movement, and with what appear to be bolted lower support plates.[185] The interaction of these supports is remains undocumented. The generators were divided in two sections: primary and secondary. The primary section in the lower 9.1-foot diameter shell consisted of the coolant channel head, tube plate (sheet), and the tube bundle surrounded by a wrapper. The upper 12-foot-wide secondary section contained the top of the wrapper, the feed water distributing feed ring, and two stages of moisture separators. Bolted and gasketed man ways provided access to the upper section, but access to the wrapper section was only through 6-inch-wide inspection holes above the tube plate. The channel head was the primary flow guide component which formed the bottom dome of the generator. It was a stainless-steel-weld clad-steel casting with an internal divider plate to channel incoming coolant into only the hot leg end of the tube bundle. A man-way on either side of the channel head allowed entry for examination of the tubes and repairs. The ends of each Inconel alloy tube were rolled (expanded) into holes in a circular perforated 21-inch thick tube plate forging that formed the boundary between the primary and secondary water. The bottom of the plate facing the incoming cooling water from the channel head was clad with Inconel and the ends of the tubes were welded to the clad.[186] The tube bundle had to resist flows on the outside and inside of the tubes. Support was by baffles attached to the wrapper and aligned by stay rods attached to the tube plate. The upper section of the bundle at the U-bends was braced with anti-vibration bars which

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[bb] It took many years of development between GE and the National Labs to develop an alternative to the pressurized reactor/steam generator system. Their boiling water reactor eliminated the steam generators by generating steam in the reactor with the reactor coolant/moderator.

had to allow for thermal expansion and contraction. Each steam generator had four safety valves set to open in approximate 15 lbs increments from 1000 and 1030 psia.[187]

There were two flow cycles: coolant and feed water/steam. Heated coolant exited the reactor hot leg nozzles and was piped to the generator channel head at nominally 2000 psig, 575° F.[188] It then flowed up the hot leg end of the tubes, around the U bend, down the bundle, and out the cold leg with a small pressure loss at 535 ° F.[189] Coolant returned to the suction side of the reactor coolant pumps in a 29-inch-diameter cross-over leg.

The steam generators were supplied with 7,600,000 pounds per hour of feed water pre-heated by the feed-heating trains in the Turbine Building and Turbine Building Auxiliary Bay, and pumped by two feed pumps in the Auxiliary Bay (see HAER No. CT-185-C). Feed entered the generators at 720 psig, 425 ° F through a single connection above the tube bundle and was distributed around the shell by a perforated feed ring. The wrapper formed an annulus with the shell through which the feed water flowed down and then entered the tube bundle area via a gap between the wrapper and the top of the tube plate.[190] As the feed water flowed up through the tube bundle it turned to steam while cooling the reactor coolant from 583.5 ° F to 545 ° F as designed [191] and 575 ° F to 535 ° F in actual operation.[192] After transferring a portion of its heat to the feed water the coolant left the tubes, entered the exit side of the channel head and flowed out of the generator through the outlet nozzle.

The 675-psig, 501.5 ° F steam produced in the tube bundle had a high moisture content requiring two stages of moisture removal in the upper section of the generator. Steam was first sent through three swirl vane assemblies which were extensions of the wrapper. The swirl vanes were similar to elongated ship propellers, welded in a fixed position. The steam/water mixture flowing through them was forced into a circular motion, which threw off 75% of the water from centrifugal force. The steam flowed upward into the second stage of moisture removal which occurred in four mist extractors. The extractors were stacked box-shaped assemblies of multiple small vane strips which forced the steam to change direction, shedding water droplets which were collected in the vane support channels. The water from both stages of separation flowed down the generator to a point where it mixed with and added pre-heat to the incoming feed water. The essentially dry steam (0.25% moisture) passed through a distribution plate and then out a nozzle on top of the generator.

Power Fluctuation Effects and Steam Generators

As the power of the turbine changed, the amount of steam drawn from the generators varied. Sudden changes caused the water level to vary due to changes in the density: known as shrink and swell.[193] If the steaming rate increased, the internal pressure decreased along with drop in temperature. The heat transfer rate across the tubes increased as a result producing more and larger steam bubbles which increased resistance to flow in the upper

sections of the generator. If that occurred, the water level (surrounding the tube bundle where the feed water was still in the liquid state) could rise too high. A high level (swell) indication on instrumentation reduced the feed water flow to compensate. The opposite effect occurred during rapid fall off in steaming rate with a reduction (shrink) in the water level, causing an increase in feed flow to restore equilibrium.

Re-circulation of the water removed from the steam in the upper part of the generators was so important to their operation that some sources called them re-circulating boilers.[194] Recirculation flow was dependent on the power level of the turbine. At very low loads (start up or testing) the circulation was low enough to be unstable. At high powers the boiling occurred lower in the bundle, producing drier steam, less moisture removal, and less recirculation. The complex interactions of coolant, heat exchange, feed flow, and water level required a facile control system.

The Steam Generator Water Level Control System (SGWLCS) maintained the water level from low loads up to full power and back by measuring steam flow and feed flow to compensate for shrink and swell.[195] As load increased the system had to vary the density (water to steam) in the generators from 75,000 lbs at no load to 36,700 lbs at full load. Steam flow, feed flow, and water level detectors signaled the SGWLCS to operate the main feed water regulating valves to set the proper water level for steady or transient steaming. Due to the critical nature of steam generator regulation, each generator had a separate vital 120-v AC power supply with automatic backup.[196]

Blowdown

The generators were continuously monitored for radiation; pH, chlorides, conductivity, and sodium on control panels in the Primary Auxiliary Building and the Chemistry Lab (see HAER Nos. CT-185-G and CT-185-F). However, the steam generator designs (of several vendors) proved vulnerable to damage from chemical impurities (see HAER No. CT-185). The only way to remove contaminants during operation was through the Steam Generator Blowdown System. As discussed in HAER No. CT-185, the main components of the system were the blow off tank, tank vent condensers, tank cooler, piping, valves, and instrumentation located in the Primary Auxiliary Building. All four generators were penetrated for upper-surface blow down and lower-bottom blow down above the tube sheet though #1 and #2 were normally isolated and may have been plugged early on.[197] Blowdown water from each unit was routed in a single line through a containment penetration to the Steam Generator Blow off Tank in the Primary Auxiliary Building (see HAER No. CT-185-G).

Pressurizer

The Reactor Coolant system pressure of 2000 psig was maintained by the pressurizer. Plant load changes and temperature fluctuations in the reactor or steam generators could affect the pressure since in a closed constant volume system they were linked.[198] As water is largely incompressible a change in pressure in one component could affect the entire system.[199] If the pressure dropped, cooling water flowing past the fuel rods could boil in quantity (bulk boiling) reducing heat transfer and possibly leading to fuel rod damage.[200] The pressurizer's role was to maintain system pressure to close limits by either producing more steam to increase the pressure or by condensing the steam to reduce it. The unit also served as a pressure relief point for the entire RCS with relief and safety valves.

The pressurizer vessel was a 42-foot-high, 7.5-foot-diameter pressure vessel, constructed of manganese molybdenum steel with a volume of 1,300 cubic feet, weighing 105 tons. Hemispherical heads were welded on each end. Like the reactor, the inside surface of the 5.3-inch-thick shell was clad welded with stainless steel.[201] The unit was vertically mounted, supported from the charging floor structure (to allow for expansion and contraction) with its upper portion projecting above the charging floor. System connection was to the hot leg of loop #four with a 10-inch-diameter surge line and surge nozzle penetration of the shell. The nozzle was protected by a stagnant-water thermal sleeve to minimize temperature shock, and had a perforated, retaining basket-diffuser for flow mixing and to prevent entrance of foreign objects.

The system pressure was maintained by electric heaters and water sprays controlled by the Pressurizer Level Control System (PLCS). In normal operation, the pressurizer maintained an equal volume of steam in the upper section and water in the lower section at 636 F. and 2000 psig. If the turbine load increased, more steam was taken from the generators, which reduced the temperature (and thus the bulk and pressure) of the coolant in the system.[202] Water would as a result exit the pressurizer due to the contraction of the coolant. As discussed above, the loss in pressure resulted in some of the water flashing to steam, but not enough to return pressure to normal. To maintain pressure, ninety emersion heaters in the water section produced steam to increase pressure in the vessel. As system pressure fell, the heaters were gradually energized and as pressure fell, they were gradually de-energized.[203] The Nichrome elements were packed in magnesium oxide powder for electrical insulation. The heaters were divided into groups: 12 heaters were variably modulated by the PLCS; the remaining heaters were full on/full off types controlled manually or by the PLCS. Total capacity was 55E F. per hour increase. Total electrical load with all heaters operating was over 1,300 kW (in an emergency situation) drawn from the three 480-v AC busses.[204] The heaters were mounted through the bottom hemisphere of the vessel and welded into heater wells.

If the turbine load decreased, the temperature and volume of the bulk coolant increased resulting in an inrush of water to the pressurizer and increase in pressure in the unit. Condensation would occur, but not enough to completely normalize the equilibrium. On a signal from the PLCS, water taken from the cold legs of the #3 and #4 loops entered the top of vessel through a nozzle and was sprayed through air operated valves into the steam space to condense more vapor and return parameters to normal. The pressurizer system had to be able to maintain stability of the RCS under ramp-up plant load changes of 5% and momentary changes of 10%.[205] A 16-inch-wide, gasketed, seal-welded diaphragm/man-way in the top dome provided access for inspection and repair.

The pressurizer also contained the over-pressure protection system of the RCS. Three ASME-code steam safety valves opened on increasing pressure points from 2485 to 2585 psig to prevent the coolant from reaching the RCS limit of 2735 psig. Two automatic power-operated relief valves were the preferred method of overpressure control since the spring-closed safety valves could not by code be isolatable, and could potentially remain stuck in the open position after pressure had fallen[206] as happened at Three Mile Island.[cc]

In order to test the valves they had to be removed from the vessel and were the only bolted components in the RCS system to facilitate that requirement.[207] To prevent discharge of irradiated steam into the Containment Building from the safety or relief valves, their outlets were piped directly to the Pressurizer Relief Tank in the lower section of the Containment annulus in proximity to the pressurizer vessel. The tank could accommodate 110% of the steam volume of the pressurizer during full power operation. The tank drained to the Containment sump or the Primary Drains Tank in the Primary Auxiliary Building (see HAER No. CT-185-G). When the RCS was being filled, the pressurizer was vented to the Volume Control Tank in the Primary Auxiliary Building. Solenoid-operated vent valves could be opened to vent non-condensable gases (a hydrogen buildup in the RCS) to the containment atmosphere.[208] When the plant was operating at low levels (RCS at 325E F and 380 psi); over-pressure protection in that range was provided by spring operated relief valves which were isolated until motor-operated isolation valves were activated.

Instrumentation

Operators in the Control Room monitored all aspects of the coolant system. Resistance Temperature Detectors sent data from the hot and cold legs. Flow was measured as a

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[cc] One of the train of malfunctions during the Three Mile Island accident in 1979 was the failure of the B&W system pressurizer, spring and solenoid-operated relief valve to close properly while failing to signal operators of that condition, allowing primary water to escape for over two hours (Lewis 1980: 59, 63).

pressure differential between inlet and outlets on the steam generators. Pressure was measured by transmitters in the hot legs and the pressurizer. Outputs were to indicators, chart recorders and the plant process computer.

Reactor Coolant Supply, Volume Control and Chemistry

Water supply for the RCS came from wells around the plant and was purified (de-mineralized, de-aerated, and de-ionized) in the Water Treatment Plant located in the Turbine Building (see HAER No. CT-185-C).[209] Water was stored in either the Primary Water Storage Tank or Recycle Primary Water Storage Tank before it was sent to the Boric Acid Blender before entering the RCS as "make up." Reactor coolant volume and water quality was maintained by the Chemical and Volume Control System (CVCS; see HAER. No. CT-185).

The CVCS was a stand-alone system but also served as an extension to the RCS.[210] The total average volume in the RCS constantly changed because of fluctuations in power output. The CVCS maintained the proper quantity of water by equating the letdown flow to the sum of the charging flow plus water flowing into the system from the RCP seals.[211] Letdown flow was a continuous bleed of coolant to the chemistry and volume control equipment of the CVCS which cooled, reduced pressure, de-mineralized and filtered the water before it was sprayed into the Volume Control Tank (VCT). The charging flow used in conjunction with the letdown flow for volume measurement was taken from the VCT, pumped by charging pumps through a heat exchanger to the coolant system. Part of the seal water that came out of the controlled leakage seal was cooled in a heat exchanger and returned to the charging pumps where it formed the third reactor coolant volume measuring component. The CVCS system was also used to fill and pressure test the RCS and it controlled the addition of the chemical neutron poison boron into the coolant. The CVCS also provided continuous purification of the RCS to reduce corrosion products and fission products.[dd]

[dd] Demonstration phase reactors were generally cooled with pure water. The early commercial plants began to use water treatment chemicals such as hydrogen and hydroxides to reduce radioactive waste from corrosion, buildup of "crud" (corrosion product oxide), pitting, cracking, and fouling of heat transfer surfaces in the core and generators. Water chemistry became more complex when chemical "shim" plants using boric acid to control reactivity came on line (Connecticut Yankee Atomic Power Company 1966-1974: 5.1..2-1)

Oxygen could enter the system either from radiolysis[ee], from makeup volume, or when the system was opened for refueling. Dissolved oxygen could accelerate corrosion and was controlled with hydrazine or hydrogen. When the plant was under construction (1966), a chemical addition tank was constructed to give operators the option of adding chemicals to control the pH of the water.[212] Probably early on, Lithium Hydroxide and Hydrazine were regularly used for pH control.[213] Other CVCS functions were to provide seal water injection, auxiliary pressurizer spray flow, and emergency core cooling flow.

Containment Engineering Safety Systems

As described in HAER No.CT-185, Connecticut Yankee operated with a large number of engineering safety systems, most of which included components in multiple structures to protect against accidents or system failures in the Reactor Containment Building. Some of these systems involved alternative functions of equipment used in normal operations. The Containment Sump, Containment Air Recirculation and Containment Spray systems had significant components within Containment, and are described below.

Containment Sump

The containment sump was filled by condensation from the CAR fan coolers and leakage from the Service Water System and had to be pumped routinely. Water was led to the sump by trenches. In normal operation the Incore Instrumentation (ICI) sump was never pumped but following a serious Loss of Coolant Accident (LOCA), the 2,000 gallon capacity would quickly be reached, and then the entire ground floor of containment would become the sump for up to 32,000 to 35,000 gallons of water.[214] Overflow water was automatically treated with tri-sodium phosphate from flooded baskets in the area of the sump to control the pH, reducing iodine release into containment atmosphere. In the Primary Auxiliary Building (HAER No. CT-185-xx), the Residual Heat Removal pumps in the recirculation mode pulled water from the sump (but not the overflow) and sent the water to the Low Pressure Safety Injection Pumps, High Pressure Safety Injection and Charging pumps — components of the Emergency Core Cooling System (ECCS) — for addition to the coolant loops.

Containment Air Recirculation System (CAR)

As described above under Containment operating systems, CAR system provided cooling and recirculation of the containment atmosphere during normal operations with four air

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[ee] Radiolysis: the breakdown of compounds from ionizing radiation in the core, producing oxygen and hydrogen (Ibid: 5.2.1-8).

recirculation units.[215] Air flow was bypassed only through the cooling coils and fans. During a LOCA or Main Steam Line Break, steam would be released into the Containment Building, raising the pressure and temperature above the design limits. The CAR system was designed to keep conditions within specification while other emergency systems were in operation. On a safety injection signal or rise in containment pressure, the CAR system would automatically start to cool and depressurize the atmosphere by re-aligning the air flow through mechanical pre-filters and charcoal filters for post LOCA iodine and particulate removal.

The face dampers automatically opened if the containment air pressure rose above a set point. The dampers were air powered with DC power for activation. In addition, they were spring loaded to open if all supplies failed. In conjunction with that, the bypass dampers would fail shut to insure the "safeguard condition" accident flow path. Since moisture in the post accident air flow could reduce the effectiveness of the filters, the air was first passed through two stages of removal: a chevron separator and fiberglass pad mist eliminators. The pleated glass asbestos particulate filter removed solid matter that could foul the critical charcoal filter during post-accident re-circulation. The charcoal filters were arranged in banks of 120 two inch thick cartridges in each unit. They were expected to be 99% efficient at removing radioactive organic iodides within two hours after an accident via isotropic exchange.[216] Temperature sensors sent alarms to the control room if the filters heated beyond 325 F during post accident filtration. The Residual Heat Removal System (RHR) provided fire protection water sprays to the filters.[217] The last stage of air handling was the cooling coil section for both normal and post accident operation. The transverse flow finned-coil banks were supplied by the Service Water System (see HAER. No. CT-185).

The calculations used to determine the design parameters for reduction of iodine (95-99%) in containment post accident were derived from testing done at Oak Ridge National Laboratory (ORNL).[218] While the range of factors considered was wide, an undocumented party questioned whether they actually covered conditions in a severe LOCA with temperatures over 250 F, pressures in the 30-40 psig range, and 100% relative humidity. During 1966 Connecticut Yankee arranged to do full scale tests under incident conditions. It is undocumented how the conditions would be replicated and what the results were.[219] At the same time ORNL noted that if filters were wet their ability to remove methyl iodide was only about 13-54%. Plant management resolved to do full scale testing with accident condition air stream mixtures.[220]

Expecting verification from testing, the Connecticut Yankee FDSA stated "...conclusively that the filtration system is a reliable and efficient means of protection against any incident which might release fission products to the containment atmosphere.[221] During the life of the plant, some operational deficiencies were discovered in the CAR units. In 1974 a number of the charcoal filter modules were found to be fouled with boric acid deposits from

the Residual Heat Removal System due to partially-closed spray valves. It was determined that the filter flow in an accident may have been compromised requiring better attention to valve alignments.[222] In 1984 an Integrated Plant Safety Assessment found that Service Water flow to the coolers may have been over estimated due to changes in valve settings. Also in 1984, it was found that three of the fans could not meet the 50,000 cfm flow required in an accident due to out of adjustment fan controls and leakage. It was discovered that internal specifications for normal operation had not actually required that amount, while specifications for LOCA conditions did.[223]

Containment Spray System[ff]

The Containment Spray System was installed as a backup to the CAR system for depressurization of containment after a Loss of Coolant Accident.[224] If during an accident pressure exceeded the 40 psig design limit by 10 pounds, containment spray valves were opened by the personnel in the Control Room. In the Primary Auxiliary Building, Low-Pressure Safety Injection pumps or the Residual Heat Removal pumps forced water into spray rings (and later possibly a single ring[225]) in the containment dome which spread out and fell down over 60 feet, absorbing heat and iodine from the atmosphere. Spray water would also provide an additional liquid film barrier over the steel liner. Water was collected in the Containment sump for recirculation by the residual heat removal pumps which supplied the system.[226] Initiation in a LOCA was not automatic as the priority was to have the RHR pumps lined up to supply the reactor first.[gg] Water could also be taken from the Refueling Cavity Water Storage Tank (see HAER No. CT-185). If flow from the ECCS was limited, a separate inexhaustible supply was available from the Connecticut River through the Fire Water System powered by diesel driven pump.[227] At the time the system was designed, it was acknowledged that the experimental work done was not sufficient for accurate prediction of the efficacy of the spray system in actual accident conditions.[228] The valves would fail on power loss in the closed position[229] and it is unclear how and if operators would be allowed to access the hand wheels during accident conditions if diesel powered backup power failed.

[ff] In the Plant Information Book, both the Charcoal Filter Spay System and the Containment Spray System were considered part of the Emergency Core Cooling System (Connecticut Yankee Atomic Power Station 1987-1995: Chapter 5, page 15).

[gg] Use of the Residual Heat Removal pumps for spray header supply was not a "proceduralized" acceptable source of spray water. (Connecticut Yankee Atomic Power Company 1987-1995: Chapter 5, page 78.)

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B. Historic Views

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Connecticut Yankee Atomic Power Company
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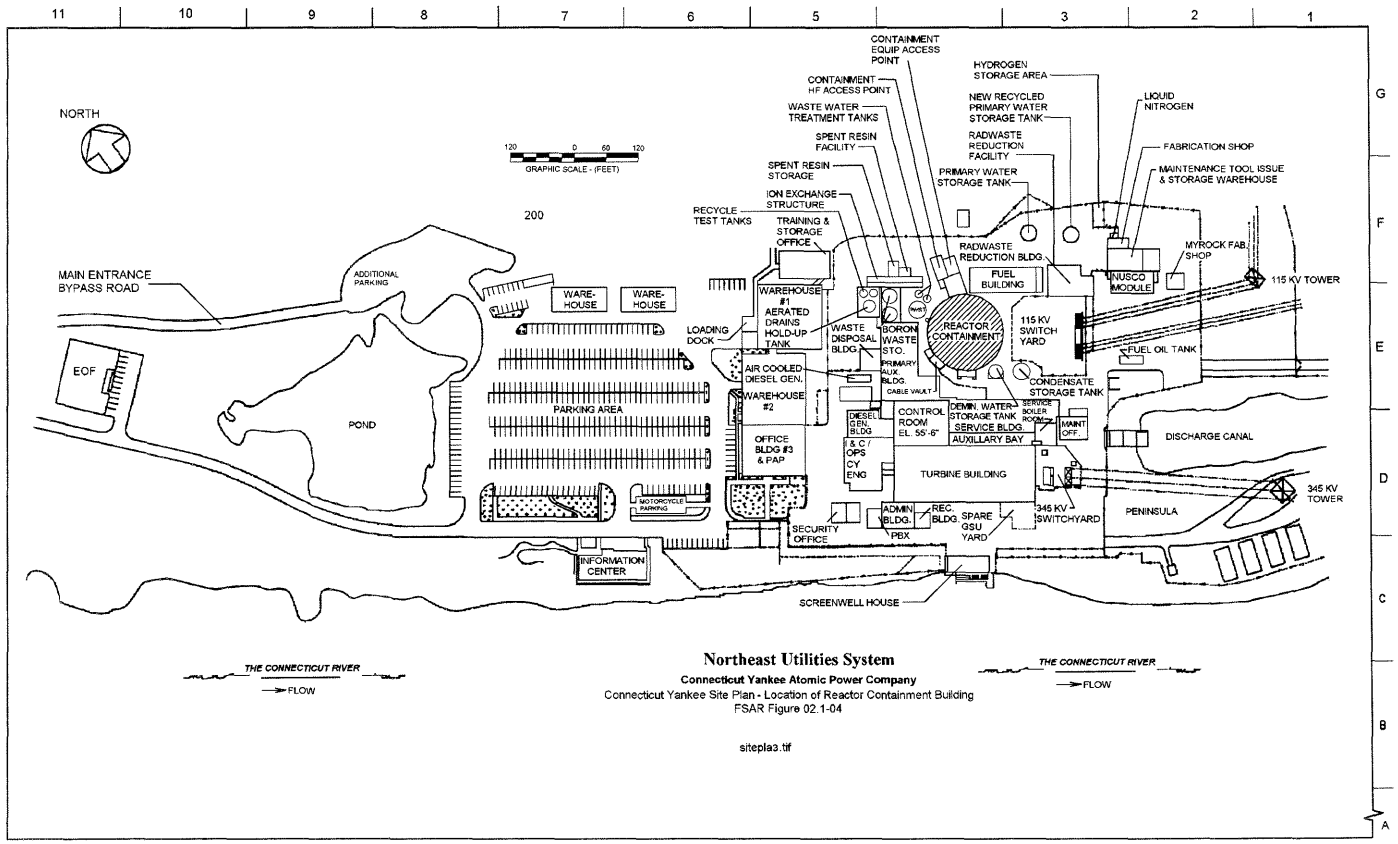

Figure 1. REACTOR CONTAINMENT LOCATION AT MAIN PLANT AREA

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[132] Connecticut Yankee Atomic Power Company 1987-1995. Chapter 3, page 59.

[133] Weinberg 1994: 197-8. At the 1972 Hearings on Acceptance Criteria for Emergency Core-Cooling Systems for Light Water -Cooled Nuclear Power Reactors there was disagreement between manufacturers and nuclear watchdogs about whether the safe temperature was 2,300o F. or much less.

[134] Zinn, Pittman, and Hogerton 1964: 32.

[135] Ibid; Connecticut Yankee Atomic Power Company 1987-1995: Chapter 75, page 6; Houff 2006.

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[136] Davies 1965: 305.

[137] Rice 2007.

[138] Houff 2006.

[139] Connecticut Yankee Atomic Power Company 1966-1974: 4.2-9.

[140] Nero 1979: 91.

[141] Connecticut Yankee Atomic Power Company 1966-1974: 4.2-1.

[142] Van Noordennen 2005-2006.

[143] Connecticut Yankee Atomic Power Company 1966-1974: 4.2-13

[144] Connecticut Yankee Atomic Power Company 1987-1995: Chapter 9, page 1

[145] Ibid: page 3.

[146] Van Noordennen 2005-2006.

[147] Connecticut Yankee Atomic Power Company: Chapter 9, page 16.

[148] Ibid: page 39.

[149] Ibid: page 36.

[150] Connecticut Yankee Atomic Power Company/Stone & Webster Engineering Corp. 1964-1975 [drawings].

[151] Connecticut Yankee Atomic Power Company 1987-1995: Chapter 9, page 14.

[152] Ibid: page 48.

[153] U.S. Nuclear Regulatory Commission 1984: Attachment #3, 11.

[154] Connecticut Yankee Atomic Power Company 1987-1995: Chapter 2.1, page 1.

[155] Connecticut Yankee Atomic Power Company 1966-1974: Table 4.3-1.

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[156] Ibid: 4.3-1, 5.1.3-1; Houff 2006.

[157] Thorp 1960: 472.

[158] Connecticut Yankee Atomic Power Company 1987-1995: Chapter 2.1, page 14.

[159] Connecticut Yankee Atomic Power Company 1966-1974: 5.1.4-4

[160] Connecticut Yankee Atomic Power Company 1987-1995: Chapter 2.1, page 39.

[161] Houff 2006.

[162] Connecticut Yankee Atomic Power Company 1987-1995: Chapter 2.1, page 20.

[163] Connecticut Yankee Atomic Power Company 1966-1974 5.1.4-3

[164] Connecticut Yankee Atomic Power Company 1987-1995: Chapter 2.1, Page 46.

[165] Bunke and Ball 1965: 314.

[166] Connecticut Yankee Atomic Power Company 1987-1995: Chapter 2.1, Page 48.

[167] Ibid: Chapter 2.2, page 43, 49.

[168] Ibid: page 15.

[169] Ibid: page 22.

[170] Houff 2006.

[171] Connecticut Yankee Atomic Power Company 1966-1974: 5.1.4-2.

[172] Ibid: 5.1.4-2.

[173] Connecticut Yankee Atomic Power Company 1987-1995: Chapter 2.2, page 78; Houff 2006.

[174] Connecticut Yankee Atomic Power Company 1987-1995: Chapter 2.2, page 43.

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[175] Ibid: pages 76, 46.

[176] Lafoon 1950: 28, Engineering 1908: 833.

[177] Lafoon 1950: 50.

[178] Connecticut Yankee Atomic Power Company 1987-1995: Chapter 2.2, page 11.

[179] Connecticut Yankee Atomic Power Company 1966-1974: 5.1.4-3.

[180] Ibid: 5.1.1-1

[181] Ibid: 4.3-2.

[182] Ibid: 5.1.3-1; Houff 2006.

[183] Connecticut Yankee Atomic Power Company 1987-1995: Chapter 2.3, page 7.

[184] Ibid: page 7.

[185] Ibid: Figure 7.4.

[186] Ibid: Chapter 2.2, page 6.

[187] Ibid: Chapter 2.3, page 22.

[188] Ibid: page 22.

[189] Ibid: 2.3, page 22.

[190] Ibid: page 3.

[191] Connecticut Yankee Atomic Power Company 1966-1972: 5.1.4-1

[192] Ibid: 5.1.4-1; Connecticut Yankee Atomic Power Company 1987-1995: Chapter 2.3, page 22.

[193] Connecticut Yankee Atomic Power Company 1987-1995: Chapter 2.3, page 40.

[194] Davies 1965: 303.

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[195] Connecticut Yankee Atomic Power Company 1987-1995: Chapter 81, page 1.

[196] Ibid: page 9.

[197] Ibid: Chapter 30, page 3; Connecticut Yankee Atomic Power Company c1972: 15.

[198] Kilpatrick 1960: 467.

[199] Penner 1976:29

[200] Connecticut Yankee Atomic Power Company 1987-1995: Chapter 2.4, page 1.

[201] Ibid: page 9.

[202] Ibid: page 4.

[203] Connecticut Yankee Atomic Power Company 1966-1974: 5.2.2-1

[204] Connecticut Yankee Atomic Power Company 1987-1995: page 16.

[205] Ibid: Page 6.

[206] Ibid: page 7.

[207] Ibid: page 49.

[208] Ibid: page 42.

[209] Ibid: Chapter 53, page 1.

[210] Ibid: Chapter 4, Part 1, page 3.

[211] Connecticut Yankee Atomic Power Company 1966-1974: 5.2.1-1.

[212] Ibid: 5.2.1-8.

[213] Willis 2002: 1 of 11.

[214] Connecticut Yankee Atomic Power Company 1987-1995: Chapter 5, page 21.

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[215] Ibid: Chapter 56, page 1.

[216] Ibid: page 9.

[217] Ibid: page 5.

[218] Connecticut Yankee Atomic Power Company 1966-1987: 3.6-4.

[219] Ibid.

[220] Ibid: 3.6-5.

[221] Ibid: 3.6-6

[222] Connecticut Yankee Atomic Power Company 1987-1995: Chapter 56, page: 63.

[223] Ibid: 67.

[224] Connecticut Yankee Atomic Power Company 1966-1974: 3.6-6.

[225] Connecticut Yankee Atomic Power Company 1987-1995: Chapter 5, page 15; Houff 2006.

[226] Connecticut Yankee Atomic Power Company 1987-1995: Chapter 5, page 16.

[227] Connecticut Yankee Atomic Power Company 1966-1974: 3.6-6A.

[228] Ibid: 3.6-6.

[229] Connecticut Yankee Atomic Power Company 1987-1995: Chapter 5, page 79.