HADDAM NECK NUCLEAR POWER PLANT
(Connecticut Yankee Nuclear Power Plant)
362 Injun Hollow Road
Haddam
Middlesex County
Connecticut

HAER CT-185
HAER CT-185

WRITTEN HISTORICAL AND DESCRIPTIVE DATA

REDUCED COPIES OF MEASURED DRAWINGS

HISTORIC AMERICAN ENGINEERING RECORD
National Park Service
U.S. Department of the Interior
1849 C Street NW
Washington, DC 20240-0001

HISTORIC AMERICAN ENGINEERING RECORD

HADDAM NECK NUCLEAR POWER PLANT
(Connecticut Yankee Nuclear Power Plant)
HAER No. CT-185

Location:	362 Injun Hollow Road Haddam Middlesex County Connecticut U.S. Geological Survey Deep River Quadrangle UTM Coordinates 18.708837.80 E -4595337.91 N +41° 28' 56.866" latitude, -72° 29' 54.983" longitude[1]
Date of Construction:	1964-1966
Engineers:	Westinghouse Electric Company Stone & Webster Engineering Corporation
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 Power Plant was one of the earliest commercial-scale nuclear power stations in the United States and the first completed on the East Coast. During its operating history from 1967 to 1996, this plant established several records in electrical production. The plant was eligible for the National Register of Historic Places.
Project Information:	CYAPCO ceased electrical generation at the Haddam Neck plant in 1996. Decommissioning operations started in 1998, subject to authority of the Nuclear Regulatory Commission (NRC). NRC authority brought the protection under the purview of federal acts and regulation protecting significant cultural

[1] The Haddam Neck Nuclear Power Plant was located at latitude +41° 28' 56.866", longitude -72° 29' 54.983". The coordinate represents the center point of the former reactor containment building. This coordinate was obtained on November 4, 2009 using a GPS unit accurate to +/- 5 meters. The coordinates were compared to values obtained on the Google Earth website and USGS Deep River Quadrangle and the accuracy of the coordinates is +/- 15 meters.

resources from adverse project effects.[2] This documentation was requested by the Connecticut State Historic Preservation Office to preclude the possibility of any adverse project effects.

Note: The material in this report is in large part based on Connecticut Yankee Atomic Power Company records that are archived at the University of Connecticut, Dodd Library. The records consist of plant design drawings, plant historical records, employee newsletters, environmental reports, regulatory correspondence, scrapbooks, plaques, historic photographs, and other audiovisual materials. The records are available to the public. For information contact the librarian at:

University of Connecticut
Thomas J. Dodd Research Center
405 Babbidge Road, Unit 1205
Storrs, Connecticut 06269-1205
860.486.4500 / 860.486.4521 (Fax)
http://doddcenter.edu

Project Manager and Historian
Michael S. Raber
Raber Associates
81 Dayton Road, P.O. Box 46
South Glastonbury, CT 06073
860/633-9026

Steam and Electric Power Historian
Gerald Weinstein
Photo Recording Associates
40 West 77th Street, Apt. 17b
New York, NY 10024
212/431-6100

Industrial Archaeologist
Robert C. Stewart
Historical Technologies
1230 Copper Hill Road
West Suffield, CT 06093
860/668-2928

Nuclear Power Consultant
Gerry van Noordennen
Dutchman Consulting
30 Miller Road
Burlington, CT 06013

Graphics Consultant
Gerry Loftus
5 Roberts Road
Marlborough, CT

Connecticut Yankee Liaison
John Arnold
362 Injun Hollow Road
East Hampton, CT 06424

ConnecticutYankee ISFSI Manager
James Lenois
362 Injun Hollow Road
East Hampton, CT 06424

[2] 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).

Historical Information

The Haddam Neck Nuclear Power Plant had an important place in the history of American nuclear power generation as one of the first two large commercial plants in the United States, and the first completed on the East Coast.[1] The plant centered on a pressurized water reactor (PWR). In this design water, kept under pressure to prevent boiling, removed heat from the reactor core and moderated or slowed neutrons from the uranium "fuel" to energies at which the fission process could continue. Additional control was provided by control rods which could be inserted or removed from the core to absorb excess neutrons. Water passing through the core became radioactive and was cycled through a heat exchanger, called a steam generator. Water passing through a secondary system in the steam generator absorbed heat from the pressurized system. It flashed to steam which drove the turbine and generator, was condensed, then cycled back to the steam generator. During its operating history from 1967 to 1996, this plant established several records in electrical production. It typified the steam powered electric plants which combined a primary nuclear steam supply system based on two decades of post-World-War-II development, and secondary systems based on 19th and early 20th century technology for converting steam to electrical energy and recycling condensed steam as feed water for the primary system. The design of the plant, and issues arising from that design, are significant examples of the limitations inherent in the first generation of American nuclear plants.

The Development of American Nuclear Steam Supply Systems, c1945-1960

Nuclear Submarines and the Beginnings of Commercial Nuclear Power

Perhaps the first electricity produced by a nuclear chain reaction occurred when one of the World-War-II-era reactors at Oak Ridge National Laboratory in Tennessee was fitted in the early postwar years with a boiler supplying steam to a small external turbine generator which lit up a light bulb.[2] Most sources credit the Experimental Breeder Reactor (EBR-1), built by the Argonne National Laboratory at the National Reactor Testing Station (NRTS) in Idaho, with being the first nuclear reactor to produce electrical power in December 1951.[3] Less than five years, later one of the NRTS reactors was used to provide all electric power for 1200 people in Arco, Idaho.[4]

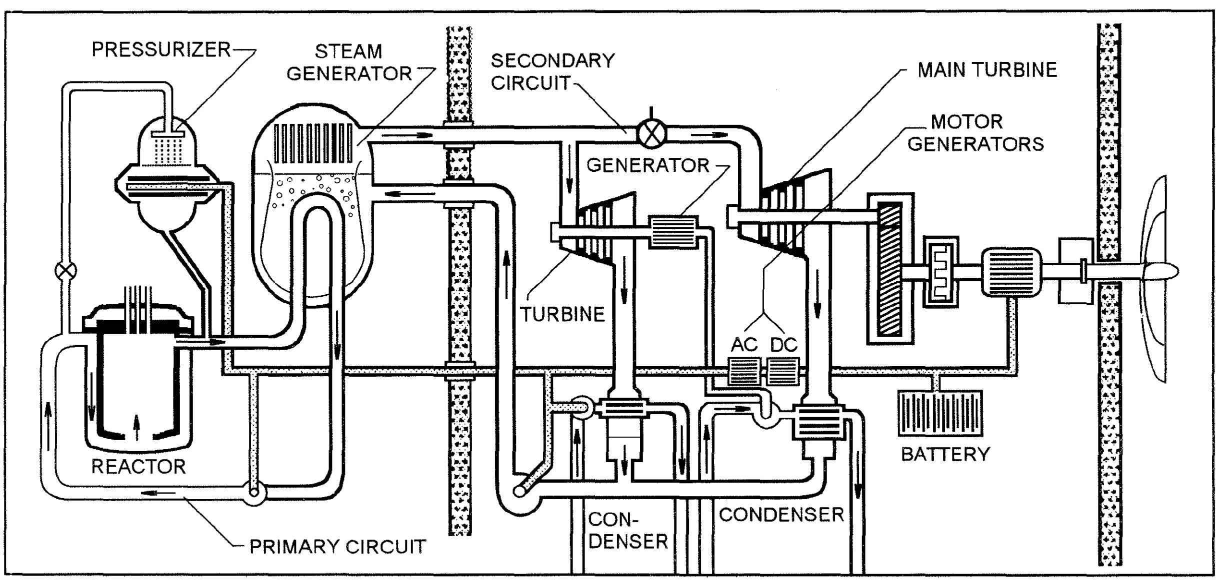
It was, however, Cold War military planning for submarine propulsion which ultimately drove civilian American power reactor development.[5] The idea of a submarine that could travel at high speed underwater was proven by the German Type XXI U-Boat of World War II. The XXI boats had super battery plants and streamlined hulls designed to give greater submerged speed.[6] Taking that a step further, the German navy came up with a design that eliminated the air-breathing diesel-charged battery plants with their limited endurance. The Walter closed-cycle propulsion system in the Type XXVI submarines utilized hydrogen peroxide as an oxidizer to make steam which powered a turbine to drive the propeller, providing extended underwater operation. Both designs were evaluated by the U.S. and British navies after the war, but the peroxide system was unreliable and hazardous.[7] The end of the war and the arrival of nuclear power doomed that technology. Less than six months after the close of the war, nuclear physicists were suggesting using nuclear power to drive ships.[8] In March 1946, the Naval Research Laboratory recommended that nuclear submarines be constructed on high speed hulls based on the German XXVI types.[9] Captain (later Rear Admiral) Hyman Rickover, who had observed operations at Oak Ridge since 1946, took a lead role in the development of the nuclear submarine and American nuclear power plants.[10] As head of the Naval Reactors Branch (Division of Reactor Development) he started working with the General Electric Company (GE) to develop a submarine reactor plant.[11] GE scientists were involved with the isolation of Uranium-235 before the war, and the company took over management of the plutonium production operations at the U.S. Atomic Energy Commission's (AEC) Hanford Laboratories in Richland, Washington in 1946.[12] GE was an advocate of a Submarine Intermediate Reactor (SIR) that utilized relatively fast neutron velocity and was cooled by liquid metal sodium, Concerned about the viability of that technology, Rickover asked the Westinghouse Electric Corporation to develop an alternative design. Westinghouse had also been involved at an early stage of nuclear development as a supplier to the Manhattan Project.[13] They favored a reactor cooled with ordinary (light) water with slowed velocity (thermal type) neutrons. The genesis of both companies' designs was probably the CP-3 Heavy-Water [c] reactor built at Argonne in 1944 which had a core of uranium rods submerged in a water tank cooled by a heat exchanger system. [14] GE's commitments to production of weapons materials at Hanford and difficulties with the sodium system led Rickover to push Westinghouse to provide the first working plant.[15] Their Submarine Thermal Reactor (STR Mark I) propulsion system was designed at the Westinghouse Bettis Laboratory near Pittsburgh, and tested in a mock-up submarine hull at the NRTS.[16] The developed power plant (STR Mark II) utilized a Pressurized Water Reactor (PWR) powered by highly-enriched U-235. The fuel in metallic form was clad with zirconium alloy forming tubes which were arranged in bundles. The nuclear chain reaction occurring in the fuel rods produced heat, and was controlled by hafnium neutron absorber rods inserted into the bundles. The coolant that took the heat out of the reactor core and moderated the reactions was ordinary (light) water, pressurized to prevent boiling as it was pumped back and forth between the reactor and tubing in separate external heat exchanger vessels called steam generators. Feed water pumped around the heated tubes turned to steam which powered turbines geared to the propellers.

The PWR (also known as a closed-cycle water reactor)[17] utilizing highly-enriched uranium was a good choice because its compact size allowed it to fit in the confines of a submarine hull. The light water was easy to handle, and the relatively low fluid and steam pressures outside of the reactor made the plant very reliable, a requirement for a vessel designed for undersea warfare. In comparison to other reactor types under development, the PWR was not very efficient,[18] but it did

---

[c]Heavy-water (D_2O), discovered in the 1930s, has the hydrogen atoms replaced with deuterium. It makes a good moderator and coolant and because it does not capture or waste as many neutrons as light water allowing the use of fuel without enrichment (Oxford English Dictionary 1989: v.4, p. 559 (Deuterium); Nero & Dennis 1984: 391).

not have to be commercially profitable in a naval vessel and was far better than the diesel-electric and peroxide drives that preceded it. Fears of the Soviet Union fielding a large fleet of captured German advanced U-Boats[19] spurred intense, well-organized, and rapid design development among Rickover, Westinghouse and the Electric Boat Division of General Dynamics at Groton, CT. President Truman laid the keel of the first U.S. nuclear submarine, the *Nautilus* in 1952. The shakedown cruises in 1955 set records for underwater distances and the success of its plant was a tremendous impetus for use of that type at sea. Westinghouse subsequently supplied the reactors for the first U.S. Navy surface vessels and many of the second-generation submarines. Despite Westinghouse's success, Rickover still wanted another system as an alternative, and GE continued to work on their design. Its Submarine Intermediate Reactor (SIR)[d] was tested on land at Knolls Atomic Power Laboratory in Milton, New York before installation in *Seawolf*, launched in 1956 as the second nuclear submarine. Surplus power from that plant near Schenectady was distributed locally and may have been the first commercial electricity to be produced by nuclear energy.[20] The *Seawolf* was commissioned in 1957, but the SIR proved unreliable and was replaced by 1960 with a conventional Westinghouse PWR.[21] GE then switched to water-cooled reactors with its twin high-pressure Submarine Advanced Reactors for the *Triton*.[22] Their later-model reactors powered a majority of the navy's aircraft carriers, cruisers and submarines.



Flow diagram for a Pressurized Water Reactor in a Submarine

---

[d]The Nautilus reactor utilized "thermal" neutrons of reduced energy. The SIR used neutrons of intermediate energy. Breeder reactors that create more fuel use high energy "fast" neutrons. (Weinberg and Wigner 1958: 12) Weinberg, Alvin M. And Wigner, Eugene P. The Physical Theory of Neutron Chain Reactors. Chicago: The University of Chicago Press.

Atoms for Peace and Early Commercial Reactor Designs

The 1946 Atomic Energy Act encouraged civilian uses of nuclear power without specifying the means. President Eisenhower's Energy for Peace program and the 1954 Atomic Energy Act opened the way for military research and fissionable materials to reach civilian programs.[23] In August of 1955 the International Conference on the Peaceful Uses of Atomic Energy opened in Geneva. Papers were presented on three broad categories of reactors being considered by seven nations: water-cooled, gas-cooled, and liquid-metal-cooled. Water-cooled reactors were divided into pressurized and boiling types.[24] Types of coolants were light water vs. heavy water in the water reactors, air vs. carbon dioxide in the gas reactors and sodium vs. bismuth in the liquid metal reactors. Moderators which slowed neutrons and aided the reaction included light water and heavy water. Carbon in the form of graphite was also proposed as a moderator in all but the boiling water types. The classes were further divided by types of nuclear fuel (natural vs. enriched)[e] and by fuel configuration (heterogeneous vs. homogeneous).[f]

Heterogeneous loading with uranium dioxide pellets stacked in stainless steel or zirconium alloy (Zircaloy) tubes was the most common arrangement.[25] Most of the reactors were "thermal" types that had neutrons slowed to thermal velocities.[26] An additional type, the Fast Fission Breeder that had no moderator and produced additional fuel, was also described.[27]

---

[e]Civilian reactors using light water require uranium fuel in which the natural percentage of U-235 (about 0.7 percent) has been slightly enriched to typical concentrations of 2-5 percent, less than the concentrations needed for naval submarine reactors. American civilian reactors have relied on government-owned gaseous diffusion enrichment plants; centrifuge-enrichment plants have been built in Europe, Japan, and South Africa. Reactors using heavy water, notably the Canadian Candu models discussed below, can use natural uranium without enrichment (McIntyre 1975; *Power* 1982b).

[f]In the homogeneous fuel arrangement, the uranium or other fuel was suspended in liquid or formed into a slurry which could be pumped in and out of the reactor and replaced at will without shutting down the reactor. The design was adaptable to fuel breeding and recycling was an integral part of the process. The AEC decided not to pursue the concept, or its successor the molten salt breeder and instead advanced the fast breeder reactor to be built at Clinch River. (Weinberg 1994: 117-129)

Considering all the variables of cooling types, moderators, fuels and fuel configurations there were at least 100 feasible arrangements.[28] The United States reported on about ten reactor types completed or in construction under the AEC prototype program at the national laboratories or production plants at Argonne in Illinois, Brookhaven on Long Island, Hanford, Oak Ridge, and the NRTS to assess the various characteristics. A movable "package" reactor for the Army and an aircraft power plant for the Air Force were also being developed. GE and Westinghouse worked closely with some of these facilities and the results of their experimentation would profoundly affect American reactor economics for years.[29] Containment building criteria developed in parallel with reactor types. The dangers from the release of fission materials were well recognized and each reactor design seemed to favor one type of construction over another. While remote siting had originally been the main safety feature, the *Seawolf* test plant near a population center was housed in one of the first vapor-tight steel containment shells.[30]

Clearly, Cold War military competition drove American choices for power station reactors. The AEC wanted nuclear power to advance and the naval program offered the fastest and best chance for that to happen. Private industry may have been enticed by financial assistance from the commission's Power Demonstration Reactor Program (PDRP),[31] and the exponentially greater power of fission: the energy of one pound of fissioned U-235 was equal to the energy in 1000 tons of high quality coal.[32] The government's willingness to provide enriched fuel at nominal prices from its gaseous diffusion plants (built during the war to provide plutonium for bombs) allowed American designers to push the relatively less-efficient light-water reactor that required enriched fuel as the ideal power producer. Based on queries sent by the AEC to private industry in 1951, plans for four private, commercial electric-power generating plants were announced at the 1955 conference.[33] The stations— all with light-water reactors— were at Shippingport, Pennsylvania; Dresden, Illinois; Rowe, Massachusetts (Yankee); and Buchanan, New York (Indian Point).

American and British engineers rightly noted that light-water reactors were going to be limited in their steam pressure and temperature abilities. The parameters of the light-water coolant and moderator were limited by two factors: the difficulty of making large reactor shells pressure proof, and the requirement that the water temperature around the hottest fuel rod areas be kept low enough to prevent film boiling which could cause inadequate cooling.[34] Manufacturing limitations resulted in a maximum working pressure of about 2,000 pounds per square inch (psi) corresponding to a temperature of about 636 degrees Fahrenheit (F). A lower temperature had to be maintained around the fuel rods, resulting in the reactor producing only half as high a heat as that in a contemporary fossil-fueled boiler. It was expected that the poor quality steam would cause moisture problems with steam turbines. In addition, a technology available in fossil-fueled boilers, adding heat to the steam after generation (superheating) was not possible with the water reactors. In some cases plant designers added oil or gas-fired superheaters to raise the steam temperature.[35] The relative inefficiency of the water reactors had another downside that would cause problems later: they produced large amounts of waste heat that had to be dumped into bodies of water unless the utilities opted for large and expensive cooling towers.[36] The engineers were concerned about the safety of PWRs because the large quantities of pressurized water in the reactor coolant loops increased the potential damage from accidents.[37] The gas-cooled system favored by the British had its own share of safety concerns, however. Without plants to enrich uranium cheaply, or the immediate need for naval reactors, Britain chose graphite-moderated carbon-dioxide-cooled reactors fueled with natural uranium, and moved faster than the Americans in opening Calder Hall, the world's first commercial nuclear power station, in late 1956 at 65 megawatts (mw).[38] France also started off with gas-cooled reactors but later went over to PWRs. Canada was an early nuclear power advocate, based on wartime research work at the Chalk River Nuclear Laboratories in Ontario. Assigned to work on natural-fuel, graphite-moderated, heavy-water cooled reactors, the Canadians stayed with that technology in their Nuclear Power Demonstration reactors. This technology led to the successful Candu (Canadian deuterium-uranium) type which uses heavy-water for both the moderator and coolant.[39] The USSR inaugurated their power program with graphite-moderated water cooled PWRs.

Steam generators were a critical component of PWRs. Unlike the steam-generating tubes of fossil fueled boilers which were directly impinged by combustion flames and gases, steam generators were fluid heat exchangers. Heated water pumped out of the reactor was forced through steam generator tubes (primary side) without boiling and then returned to cool the reactor. Feed water was continuously pumped around the tubes and was heated to boiling by contact with their surfaces (secondary side) The generated steam was passed through moisture removal devices and sent to the turbines. Because of the relatively low temperature of the reactor coolant in a PWR, it was necessary to have a large heat-transfer surface to insure reasonable efficiency, calling for almost 4,000 tubes in the Westinghouse units and even more in those of other manufacturers. The cooling water had to flow evenly through all tubes and the feed water around them to insure full heat transfer, requiring complex perforated tube plates and baffles to channel the flow and insure that the primary and secondary water never merged. The tube bundles had to be supported to resist the flows on both sides with a network of braces and connections. The designs proved vulnerable to damage by various foreign substances.

Shippingport

In its prototype reactor program, the AEC supported in whole or in part the construction of small experimental reactor plants that included gas, polyphenol or sodium cooling, fast breeding, homogeneous fuels, etc.[40] It was no coincidence, however, that the first large nuclear electric power utility station in the United States was a PWR built by Westinghouse and supervised by Rickover, a team with a proven track record.[41] That plant, in Shippingport, was essentially a land-based version of a projected naval aircraft carrier reactor and went on line in late 1957. A group of manufacturers got together to build this pioneer plant. Westinghouse was the designer and main supervising contractor of the primary (reactor) systems and fabricated the intricate core assembly consisting of almost 100,000 fuel elements and the critical reactor coolant pumps.[42] Three established fossil-fuel boiler manufacturers supplied other hardware. The reactor vessel was built by Combustion Engineering Inc. (CE). Foster-Wheeler Corp. (F-W) supplied two straight- tube steam generators, and The Babcock and Wilcox Co.(B&W) provided two u-tube generators, each of which was part of a coolant "loop" out of the reactor.[43] Stone and Webster were the architect-engineers with construction shared by Dravo Corporation and Burns and Roe, Inc. Duquesne Light Company supplied the secondary systems (turbine generator, condenser and auxiliaries) and guaranteed to purchase a block of 60 mw power.

The economic efficiency was not expected to compete with conventional plants, even with the AEC supplying most of the development dollars and the enriched fuel.[44] Rickover's standards served as a model for the industry,[45] including 3000-hour core life, redundant fuel rods, backup safety systems, four separate coolant loops to ensure reliability, corrosion resistant materials in the reactor (but not in the secondary systems[46]), and commercially available equipment.[47] Placing safety in the forefront, he insisted that the reactor be partially buried below grade so that a safety injection system could immerse the core with cooling water without needing pumps.[48] If one of the control rods failed, water containing boron (boric acid) to kill the nuclear reactions could be injected into the system.[49] The reactor was contained in its own gas-tight steel chamber, with pairs of steam generators, coolant pumps, and auxiliaries in separate steel chambers, the whole surrounded by more than 5 feet of concrete shielding. The steel shells were designed to resist internal missiles such as valves traveling at high speed.[50] The compact Nautilus reactor was made possible by the use of expensive highly-enriched uranium. The larger reactor for Shippingport had a more economical arrangement with highly-enriched uranium-zircaloy "seed" fuel rods surrounded by natural uranium "blanket" fuel elements of uranium dioxide in Zircaloy tubes.[51] The removable head of the reactor was penetrated for instrumentation and multiple fail safe control rods to start, maintain, and stop the fission reaction. Fuel rods could be individually replaced through fuel ports with the head in place. The lower section of the reactor vessel containing the core was surrounded by three feet of water to reflect neutrons. Coolant water entered the bottom of the vessel, flowed up through the fuel rod bundles taking off their heat and out through nozzles above the core to the steam generators. Each reactor coolant pump was a "canned" leak-proof unit developed for the submarine reactors, without seals between the centrifugal pump and motor and cooled by the primary water. Pressure in the system was closely controlled by electric heaters or water sprays in a separate pressurizer vessel. Following the lead of the Experimental Boiling Water Reactor at Argonne, the spent fuel rod bundles were unloaded underwater (to contain the radiation) by remote handling devices and moved out of containment in a flooded canal to a storage pool in an adjacent fuel handling building which also handled the new fuel.[52] Shippingport was very much a product of national priorities, Rickover's's driving leadership, and perhaps a desire to beat Britain and the USSR.[53] The "forced" nature of the engineering and construction, with speed of design a significant factor, impacted a whole generation of U.S. power reactors.

General Electric's Boiling Water Reactors

While Westinghouse was partnering with the AEC on land based PWR development, GE was working to perfect a boiling water reactor (BWR) to produce electric power from concepts tried out in experimental plants at Argonne and NRTS.[54] In their single cycle BWR, cooling water was allowed to boil in the reactor dome producing steam that was sent directly to the turbine. Starting early with private funding, GE built the Vallecitos boiling water plant in California (AEC license #1) which sent out a small block of power over the Pacific Gas and Electric Co. (PG&E) system in 1958.[55] The benefits of reduced pressure in the core (compared to PWRs) and the elimination of what were to become troublesome steam generators were offset by the fact that the power regulation was poor and that irradiated steam traveled out of containment into the turbine, complicating environmental safeguards and turbine maintenance.[56] Despite these problems, the design had development potential, proved to be equally efficient per kilowatt-hour (kw-h), and became the second most common type in the United States.

The "honeymoon" that occurred during the development of Shippingport between Westinghouse, B&W, and CE did not last as their public relations departments touted the benefits of the reactors and components each was putting on the market. With GE, Alco Products Inc., AMF Atomics, The Martin Company, Allis-Chalmers Mfg. Co., North American Aviation (Atomics International), General Nuclear Engineering Corp., and General Atomic in the mix, there probably were too many suppliers promoting too many design variations.[57] By the early 1960s, power companies were likely attracted to the relatively lower costs of light-water designs, and to the more common models for which supplier profits and experience seemed to assure better potential customer support.

Initial Licensing

The AEC licensing program, established under the Atomic Energy Act of 1954, was critical to the siting, design, construction, start-up and power production of commercial plants. The Act gave the government extensive control over research and nuclear materials, but encouraged private industry to build plants.[58] Fuel and rods were leased from the AEC at rates that were designed to make nuclear power stations competitive with fossil-fueled plants. The AEC exercised its control through extensive licensing procedures. Provisional Operating Licenses, Full Term Licenses, amendments, modifications, safety evaluations, and violations or penalties were the ruling documents. The commission specified everything from the facility location to the limits of worker radiation exposure. Changes to equipment that could in any way lead to radiation releases had to be approved. Resident inspectors were constantly in the field checking start up/shutdown procedures, re-fueling and maintenance. The utilities had to respond to the documents with detailed descriptions of actions taken and also send in annual reports. Commission authority over operational issues was accorded by the Code of Federal Regulations (CFR) and license amendments were printed in the Federal Register.[59]

Early Commercial Plants

All of the plants built into the early 1960s including Shippingport were heavily supported by the government, but some built under the PDRP were mostly financed and built by private utilities with the AEC providing research, development assistance and free fuel for five years.[60] The first of this series, commenced by Commonwealth Edison Company of Chicago at Dresden, Illinois, was GE's first large (180 mw) BWR plant. The dual-cycle design utilizing "secondary" steam generators to supply the low-pressure turbine had better power regulation than the earlier models.[61]

Westinghouse began its first private venture with the Yankee Atomic Power Company plant in Rowe, Massachusetts on the Sherman Pond reservoir of the Deerfield River, the first plant built by a consortium of New England power companies including Connecticut Light & Power Company and other later affiliates of Northeast Utilities. Completed in 1960 as the third American nuclear power plant and the first in New England, the PWR plant with a net output of 167 mw set the trend for subsequent Westinghouse three- and four-loop plants including Connecticut Yankee. Consolidated Edison partnered with B&W to build the 163-mw PWR Indian Point plant on the Hudson River in Buchanan, NY which was designed to use uranium and thorium as fuels.[62] These stations were followed in 1962-3 by the 65- and 67-mw BWR's at Big Rock Point of Consumers Public Power Co. in Michigan and PG&E's Humboldt Bay in California. In 1966 two more stations were completed: Northern States Power's 60-mw Pathfinder BWR plant in Sioux Falls, South Dakota, and Philadelphia Electric Company's 40-mw Peach Bottom High Temperature Gas Cooled Reactor (HTGR) plant. The last plant completed before Connecticut Yankee was the 48-mw BWR La Cross Nuclear Generating Station built in 1967 by the Dairyland Power Company of Wisconsin. The outputs of most of these "demonstration" phase plants were generally less than the fossil-fueled stations already on their respective grids.[63] One other plant usually not included in the history of commercial plants was the N Reactor constructed at Hanford by the government in 1963. While producing weapons grade plutonium it also put a large block of electric power onto the Washington Public Power Supply System grid.[64]

The Yankee Nuclear Plant at Rowe was the direct precursor to Westinghouse's later plants including Connecticut Yankee. While based on the technologies worked out at Shippingport, the design basis was different enough so that project engineers stated in the company magazine *Westinghouse Engineer* that no direct comparisons could be made.[65] Among the changes at Rowe were:

- use of a single, above-ground steel containment sphere;
- modification of reactor coolant flow with entry and exit nozzles above the core to facilitate the admission of emergency core cooling water;

- stainless steel cladding on fuel rods instead of Zircaloy;
- uniform slightly-enriched fuel loading instead of the seed and blanket arrangement;
- silver-indium-cadmium control rods (instead of hafnium) with supports extending below the core;
- no ability to load fuel with the reactor head in place.

Boron injection into the coolant aided in normal shutdowns and was also used in the safety injection system (later known as the emergency core cooling system.) Westinghouse-designed vertical U-tube steam generators were used in place of the contractor-built horizontal straight- and u-tube types at Shippingport. The elevated position of the reactor required an inclined water filled chute in which a transfer car carried the spent or new fuel rod bundles and control rods to and from the fuel building on ground level.[66] Overall, the goals were safety (the location near the Vermont border was considered remote[67]) and low first cost to make the plant economically viable.[68] Construction was supervised by Stone and Webster.

Containment Structures

The containment structures of these early commercial stations began to assume the features that were standardized in the 1970s.[69] The predominant shapes were either spheres or cylinders with hemispherical tops/bottoms. Dresden had a steel sphere modeled on the earlier Milton shell.[70] Yankee Rowe containment was an above ground spherical steel vapor container surrounding a reinforced-concrete reactor support structure.[71] Indian Point had a domed concrete cylinder with a separate internal steel vapor containment sphere.[72] These early containment structures were built under local building codes and American Society of Mechanical Engineers pressure vessel codes.[73] While containing radiation was relatively easy, they also had to resist a pressure build up from a release of the stored energy from the coolants and moderators, requiring a pressure rating of around 20 to 30 pounds per square inch gage[g] (psig).[74] Release to the atmosphere was to be restricted but not necessarily prohibited.[75]

The AEC's first standardized requirements for siting distance and emissions control were not proposed until 1961.[76] As a result of objections to aspects of that draft, actual criteria of the 1962 document loosened the definition of "population center distance," and led to a trend of reliance on engineered safeguards for protection rather than remote locations.[77] It suggested that meteorological conditions be considered, and that no reactor be located within a quarter mile of a

---

[g]Steam or gas pressure was stated as pounds per square inch gage (psig) which was the pressure over the 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).

on all environmental factors of seismology, meteorology, geology, and hydrology in the CFR was limited to four paragraphs.[78] Planning engineers could use conventional methodology for assessing seismic and wind loading.[79] Assumptions were based on experience with non-mechanical-system-filled structures such as offices.[80] Thus the earliest commercial plants were often sited in areas that later would probably have been prohibited. The 1959 Santa Susana Station was sited by the Southern California Edison Co. sixteen miles from the San Gabriel fault in an area that was described by project engineers to be "...as free from seismic disturbances as any in the vicinity of Los Angeles."[81] The siting of the Indian Point plant near the Ramapo fault was another example. The first detailed criteria from the AEC occurred well after the commercial phase plants were built.[82] It was also some years before effective models were devised to show how critical nuclear components would interact with structures during earthquakes. By 1970 prestressed concrete (previously instituted in French nuclear plants) had taken over from simple reinforced construction.[83]

Early Insurance Issues

While prevention of release via engineered containment buildings and safety systems was generally accepted by the industry, there was developing resistance to insurance and siting criteria in place during the demonstration phase of reactor construction. As early as the 1860s, private insurers in partnership with boiler makers had arrived at specifications and inspections procedures to protect the public from power boiler explosions.[84] The insurance industry was understandably uncertain about extending fossil-fuel-powered boiler insurance programs to nuclear reactors.[85] To spread the risk they set up insurance pools (syndicates) and instituted rating plans to assess various "nuclear perils"[86] As a result of a 1957 AEC report noting the possibility of up to four billion dollars in costs and thousands of fatalities from a major accident, most public liability was transferred from reactor manufacturers and operators to taxpayers though an amendment to the Atomic Energy Act.[87] The substitution of engineered safeguards for remote siting led to challenges to the construction permit for the Fermi Station, located within 35 miles of Detroit and Toledo. A federal court of appeals' decision to rescind the AEC's construction permit ended up in the U.S. Supreme Court which reversed the lower court's decision.[88]

Advanced Reactors

While the licensing trend was leaning towards water-cooled reactors, the AEC was still sponsoring attempts at developing systems that could develop higher pressure and superheated steam in commercial plants. The 1963 prototype Carolinas Virginia Nuclear Power Associates (CVNPA) plant in South Carolina which like reactors in Canada used a heavy water moderated, pressure tube design operated for just 4 years.[89] [h] Experimental superheating reactors were built at Argonne and

[h] In the Canadian reactors, the fuel rods were contained in individual pressure tubes through which the heavy water coolant flowed, eliminating the need for a reactor vessel containing a large volume of pressurized water surrounding the core. (Mcintyre 1975: 18)

Vallecitos but the technology was not adopted by the industry.[90] Sodium cooling with graphite moderation and superheating was tried at the AEC-sponsored, Atomics International (AI) built, experimental Santa Susana plant and their subsequent 77-mw Hallam plant in Nebraska built in late 1963 for the Consumers Public Power District.[91] Hallam was followed by the AI built 61-mw, Detroit Edison Co., Enrico Fermi Liquid Metal Fast Breeder Reactor (LMFBR) plant in Monroe, Michigan which operated for over ten years.[92] There were start-up difficulties in these plants with damage and leaking. While such problems were to be expected with advanced technology, it may have soured the concept with utilities that were experiencing reliability with their light-water thermal plants. Many in the nuclear industry thought that the LMFBR technology was the only economical route in the long term due to limited uranium reserves.[93] In 1972, a major research and development project to create an advanced LMFBR was started by the AEC and two utilities at Clinch River, Tennessee.[94] Three competitors, Westinghouse, GE and AI, combined forces for this project. In 1977 the DOE placed a new breeder core in the Shippingport reactor which operated until plant shutdown in 1982.[95] The Fermi plant closed in 1972, Congress ended funding for the Clinch River project in 1983[96], and with ample enriched uranium supplies there was little impetus for further breeder development in the United States.[97]

The Economics and Efficiency of Early Nuclear Generation

All commercial nuclear plants were designed to meet efficiency objectives which originated over a century earlier The overall efficiency of a steam-powered electric generating station, regardless of fuel source, was determined by a close interaction of all components in the system from the heat source through the prime mover and condenser, with inputs to and from the feed water heating and other auxiliary systems. The goal was to achieve an overall operational efficiency based on an ideal number drawn from the 19th-century theoretical works of Carnot and Rankine.[i] Beginning in 1922, power station operators used the extraction method of feed heating, in which exhaust steam was withdrawn from the turbines to pre-heat the feed water going back to the boiler to boost overall station economy.[98] A few years later, re-heating the steam between separate high- and low-pressure turbine casings also improved efficiency while reducing erosion in turbine blades from moisture.[99] With so many stages of heat utilization, the calculations required to achieve maximum efficiency were complex. A new concept called Heat Balancing, (also known as BTU auditing), treated the heat utilization as a balance sheet in which the usage in the components had to balance for maximum efficiency.[100] Tables to plot the heat flows were augmented in the late teens by heat balance diagrams in which all the important components producing and using heat in a plant were

---

[i]Sadie Carnot (1796-1832), a French natural philosopher, founded the science of thermodynamics in 1824. His work "Reflections on the Motive Power of Heat and on the Machines Adapted to Develop This Power" described an "ideal" cycle of steam through an engine and was later developed by Lord Kelvin (1824-1907) and others as the Carnot Cycle, the basis of a practical measure of the maximum possible output from a given power system (Wilson 1981:137, Engineering 1907: 847). In England, William J. M. Rankine (1820-1872) wrote a paper entitled "On the General Law of the Conservation of Energy" in 1853 followed by other writings expanding on thermodynamics. The cycle he described called the Rankine Cycle is used to measure the comparative efficiency of turbine power systems (Engineering 1873: 14, Babcock & Wilcox: 1960: 10-6).

drawn in a simplified manner to illustrate the relationship of the components and heat flows.[101] As the technique was perfected engineers of some plants showed all the important parameters of temperature, pressure, and quantity of flow on the diagrams to help designers plan the most efficient arrangements.[102] With these factors included, plant designers and operators could clearly project how changes of temperature, pressure and flow quantity in one component would affect other components upstream and downstream, relative to the total plant output as measured in gross heat rate of BTUs per kilowatt hour. Designers of commercial nuclear plants prepared diagrams for various heat rates, from initial startup through licensed maximum output.[103]

Heat balance considerations were only one set of factors in determining the economic viability of the early commercial stations, whose owners used complex formulas based on assumptions regarding costs of construction, operation, and government-supplied fuel, measured against current and expected costs of coal, the main competitor. Nuclear fuel costs were part of a larger cycle including mining, enrichment, loading, burn up, and recycling or storage of spent fuel (Figure 1). In theory, it was expected that the government would take responsibility for all but the loading and burn up components of this cycle, but the unresolved issues of recycling and storage introduced costs to utilities which were not fully factored into early cost calculations.

While there were several methods of measuring the costs of power stations including cents per BTU, or dollars per kw, the standard measure was the mill (one thousandth of a dollar) per kilowatt hour.[104] Plant costs were broken down into fixed (structures and equipment), operation, and fuel. Conventional fossil-fuel plant costs during the critical planning period of the first commercial stations were about seven mills total.[105] This was a number that nuclear plants had to approach to be viable. The fixed costs of nuclear plants were much higher due to their still-experimental nature and the requirements for remote siting or containment structures. Shippingport was projected to come in at 64 mills, but Duquesne Light Company was buying the steam at only eight mills reflecting the extent of the government support.[106]

Nuclear planners expected that perfected (larger) designs, cheap nuclear fuel costs, a credit for burned fuel, increasing demand, and stable or rising coal costs would change that imbalance.[107] For several reasons their assumptions proved inaccurate. In 1964, the AEC amended the 1954 act to require private ownership of enriched fuel which would continue to be enriched at AEC facilities (known as toll enrichment) until at least 1970.[108] Private recycling facilities were to be set up and the fuel price was stabilized.[109] However, nuclear fuel cost projections failed at the tail end of the nuclear cycle because the expected credits from recycled fuel never materialized due to poor planning[j].[110] [Moved note 108 and deleted "The fuel economy..."] Starting in 1958 the AEC sponsored the development of rail, truck and barge shipping casks for the spent fuel rods.[111] The failure early on[112] to set up storage locations, and possible public objection to projected routes,

---

[j]Up to 1971, no commercial reactors had shipped any fuel for recycling (Osbourn and Larson 1971:247)

resulted in the used fuel being stored in plants' increasingly- crowded spent-fuel pools.[k] This came to be known as the "stowaway cycle."[113] In projecting profitability for the nuclear industry, planners took as a given that demand for electricity in the US doubled every ten years and that coal burning plants would not be able to meet that load.[114] A graph provided by the AEC in a 1959 study of operating costs showed nuclear stations of increasing size (450 mw) dropping to meet a flat line for coal station costs.[115] More conservative projections aimed at establishing a nuclear parity with coal plants stated that to be competitive, the plants would have to reach 1,000 mw.[116] Any belief that coal costs which had been dropping from 1948 to 1958 would rise in the 1960s[117] was also in error.

Seeing nuclear power as a second threat after oil to their hegemony, the coal industry came up with new methods and technology that generally kept prices stable and actually lowered costs in some areas.[118] Coal companies encouraged or built power plants close to coal mines that were worked with giant stripping shovels or advanced mining machines. Though these "Mine-Mouth" power plants were sited far from load centers, they were made feasible by extra high voltage (EHV) transmission lines (500 kilovolt amperes) which could send power economically hundreds of miles at five mils.[119] At the same time, railroads concerned about competition from oil-or natural gas-fired stations, coal slurry pipelines, and the Mine-Mouth/EHV technology came up with the unit/integral coal train concept.[120] The unit train of all coal cars had favorable rates (up to 35% cheaper than regular trains[121]) as it shuttled between a mine and power station in the load center. The permanently coupled integral train took the concept even farther. The unloading process was streamlined by providing the coal cars with rotary couplers allowing automatic high speed discharge from car dumpers.[122] One utility, Commonwealth Edison, expected to save over four million dollars per year from these innovations.[123] At the same time improved fuel-burning technologies and super-high-pressure boilers were driving down the number of BTUs required to produce a kw-h of power.[124] Thanks to these advances, and the use of large marine colliers for delivery of coal to plants on navigable waterways, costs of coal-powered generation were not much more than the Jersey Central Power & Light Company Oyster Creek Plant in Toms River, New Jersey, the nuclear industry leader at four mills.[125] As early as 1965 nuclear industry planners were acknowledging that improvements in coal technology had changed the equation but they claimed that it benefitted the nation as a whole.[126] By 1967, the Tennessee Valley Authority (TVA) was projecting its new fossil-fuel and nuclear plants to come in at under three mills.[127] Thus during the critical early phase of commercial nuclear power, there was considerable pressure on profitability.

Another factor that had to be considered in nuclear station economics was the "fit" with the fossil-fueled stations on the grid. The fact that nuclear stations had to be of large size to be economic[128] posed a problem for the utilities since their intermittent fueling meant that during shut-down for refuel, a large block of power had to be replaced. Still committed to water reactors, Westinghouse

---

[k]Between 1974 and 1980, Connecticut Yankee shipped a total of 83 fuel assemblies to General Electric and Battelle, with the remaining 1,019 removed assemblies stored in the spent fuel pool (van Noordennen 2005). Battelle is a Columbus, Ohio based, global, non-profit scientific research and management enterprise founded in 1929 which assists the DOE in operating the national laboratories at Brookhaven, Oak Ridge, and Idaho. (Battelle 2003).

sponsored development of a homogeneous breeder reactor system in which nuclear fuel in a slurry loop could be replaced without shut-down.[129] Their Pennsylvania Advanced Reactor (PAR) concept never reached construction. Continuous on-line fueling was also a goal of Combustion Engineering's proposed heavy-water-moderated, organic-cooled 500-mw plant.[1] CE was aiming for generating costs of three to five mills in full-scale plants but their technology never came to fruition.[130] The only widespread use of continuous fueled reactors in North America are those of the Canadian Candu series.[131] Despite the AEC's encouragement of (and international use of) diverse and more efficient reactor types, the pressure of getting plants on line and making a profit led American utilities to concentrate on the apparently-reliable, somewhat-efficient light-water reactors.

The First Full-Scale Commercial Nuclear Plants and Construction of Connecticut Yankee

By 1963, the stage was set for the first nuclear power stations that could function in multi-station grids on a nearly equal power production and operational cost footing with coal- or oil-powered units. These were important criteria because it was clear that the nuclear stations were not economical unless they were putting large blocks of power into their respective grids.[132] The first "full scale" stations were the three-loop 436-mw San Onofre Nuclear Generating Station of the Southern California Edison Co. and San Diego Gas and Electric Co. in San Clemente, California and the four-loop 616-mw Connecticut Yankee Nuclear Generating Station in Haddam, Connecticut. A 1968 article in *Scientific American* titled "The Arrival of Nuclear Power" noted the importance of these plants in the maturation of commercial atomic energy.[133] Both were Westinghouse-designed plants and both began commercial operation on January 1, 1968.[134] They were closely followed by the 650-mw Oyster Creek station and Niagara Mohawk Power Corp's Nine Mile Point 610 mw-plant in New York State to round out what has been called the "commercial phase."[135]

The long lead times, large size, and problems of engineering containment buildings virtually assured that only a few design and construction concerns in the United States would share the work.

---

[1] Organics are a class of compounds (diphenyls, terphenyls, etc) derived from or containing hydrocarbon radicals. They were first produced by Faraday in 1850 through compression of oil gas (*Oxford English Dictionary* 1989: v.X, p. 675[Phenyl] and v.XI, p.920 [Organic]). Organics have many benefits as a reactor coolant or moderator: providing a compact core, low system pressure, lack of reactions with fuels or water, compatibility with standard metals, and production of higher temperature steam (Balent 1959: 120).

Ebasco Services, Inc.; Sargent & Lundy, Engineers; Burns and Roe, Inc.; Stone & Webster Engineering Corporation, and Bechtel Power Corporation engineered or built many of the early and commercial phase plants and the latter two would figure in the history of Connecticut Yankee.

Planning and Corporate Organization for Connecticut Yankee

The early success of the Yankee Rowe station, which began commercial operations in 1961, and the increased demand for electricity in Connecticut prompted the state's three largest utilities — Connecticut Light & Power Company (CL&P), Hartford Electric Light Company (HELCO), and United Illuminating Company (UI) — to consider another PWR plant in April 1962. Initially organized as the Nutmeg Electric Companies Atomic Project, this consortium soon concluded that a nuclear station could be competitive against fossil-fuel generation over the life of the plant, using the then-common assumption that coal (and to a lesser extent oil) costs would rise. As discussed above, this assumption later proved false, although in regional terms the construction of more nuclear generating capacity contributed to lower costs in conjunction with pressure on coal prices, the introduction of larger generating units and higher-voltage long-distance transmission facilities, and increased coordination among power companies. Nutmeg Electric moved quickly to option the 500-acre site of what became Connecticut Yankee in Haddam Neck, and by the end of 1962 selected Westinghouse to produce the major plant components and Stone and Webster to design, engineer, and build the plant.[m] At the same time, the considerable costs involved, plus the model of the Yankee Rowe consortium and the long history of cooperation among New England power producers,[n] led to the dissolution of Nutmeg Electric and the creation of a new corporation to build Connecticut Yankee in December 1962. The Connecticut Yankee Atomic Power Company (CYAPCO) expanded the consortium beyond Connecticut to include eleven utilities. As plant construction was about to begin, the project became a factor in the negotiations leading to the 1966 creation of Northeast Utilities, an affiliation of CL&P, HELCO, and Western Massachusetts Electric Company (WMECO), the latter also an owner of CYAPCO. The NU system, later expanded by the absorption of Holyoke Water Power Company and Public Service Company of New Hampshire, immediately became the largest utility in New England and one of the twenty largest in the nation.[136]

---

[m] The Boston company was founded by Charles Stone and Edwin Webster in 1889 as an electrical testing lab. The company grew to provide worldwide engineering consulting with a particular emphasis on design and construction of power stations. In 2000, it became a subsidiary of the Shaw Group of Baton Rouge (European Construction Institute 2005: Website, 1; *Hoovers* 2005: Website, 1).

[n] The Connecticut Valley Power Exchange, consisting of CL&P, HELCO, and Western Massachusetts Electric Company (WMECO), was the nation's first electric power pool when created in 1925 (Northeast Utilities 2005).

Connecticut Yankee Initial Licensing, Construction, and Initial Operation

Provisional and final construction permits for Connecticut Yankee were issued in May and June 1964. Plant siting had to conform to the 1962 AEC Reactor Site Criteria which allowed engineered safeguards to replace remote siting. An additional document, Calculation of Distance Factors for Power or Test Reactors (TID-1844) guided the process which included the maximum allowed releases, containment capability, and environmental conditions at the proposed site to arrive at an exclusion zone.[137] The efficacy of the safeguards including structures, safety injection, water sprays, and filters was balanced against the TID's recommended distance factor. In the case of Connecticut Yankee, the plant's engineered safeguards reduced the exclusion radius from about a mile to 1,700 feet.[138] In addition to the exclusion zone, low population zones and population centers were considered. For instance, the low population zone was one where it could be expected that the residents could be protected from a hypothesized major accident while receiving only a specified radiation limit.[139] Although Connecticut was considered to be a seismically stable area, borings were taken and the plant was designed to be able to shut down safely in a "moderately strong earthquake..."[140]

Concrete pouring for the containment and turbine pedestal foundations began in August 1964. The reactor vessel was installed in May 1966 and construction was completed in early 1967. The plant received its provisional license (No. DPR-14) from the AEC in June of that year and initial reactor criticality followed in August[141] AEC licenses governed the power level of the reactor which was measured in megawatts thermal (abbreviated mwt).[o] For startup, the reactor was limited to 1473 mwt out of a possible 1825 mwt.[142] Electricity generation began in August. Under the provisional license the AEC closely monitored start-up activities. During the period before full power operation, adjustments were made to equipment, leaks were sealed and turbine stop valves were modified on two occasions. Most of the work was done on the secondary systems outside of containment with some power production continuing. A repair to a steam generator access door a few months after start-up did require an output drop to less than 50 mw. Commercial operation began in 1968. The amendment to the provisional license for full power operation was not granted until February-March 1969 and 600-mwe generation was not achieved until January 1970.[143]

---

[o]From the beginnings of the electric power industry, the power of boilers and turbines was rated in horsepower and the power of generators in watts, kilowatts (a thousand watts), and megawatts (a thousand kilowatts.) From their inception, the output of power reactors was measured in megawatts of heat (Ford 1955: 492.) and later in thermal megawatts. The electrical output in megawatts was lower than the thermal number due to loses in the nuclear steam supply system and generator. In this historical overview, the capacities of nuclear power stations are given in megawatts of electrical output as described in contemporary and later documents. Published figures for a station can vary because the AEC/NRC often allowed increases in output over the life of the plant. The abbreviation "mwe" came into use in the 1960s and was common in the 1970s. Dates of stations may vary between sources due to the length of time between completion and commercial generation.

Connecticut Yankee Containment and Primary Systems

Connecticut Yankee containment structure and primary systems combined some elements of the Shippingport and Yankee Rowe stations reflecting ten years of development. Shippingport's proximity to a population center and undeveloped standards for mechanical engineered safeguards led the designers to place the steel reactor, steam generator and auxiliary system vessels largely underground. The Yankee Rowe plant had all its reactor systems surrounded by concrete in a completely aboveground steel sphere.[144] Connecticut Yankee reverted to a sub-grade reactor location inside a newer above- and below-ground, industry-standard reinforced-concrete straight-walled cylinder with a hemispherical top known as a "right circular cylinder."[145] The steel vapor shell was attached to the inside surface of the outer concrete wall. The main components within containment were the reactor; four steam generators and coolant pumps; pressurizer, emergency core cooling system (ECCS), ventilation and filter equipment; refueling systems; and overhead crane. The reactor was enclosed in a separate concrete chamber by a primary shield wall which isolated the coolant pumps and steam generators from radiation, allowing access shortly after shutdown.[146] Additional protective walls separated the pumps and generators into pairs. A second concentric circular concrete wall (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 over the reactor and pumps provided a surface for access to the reactor head for refueling. Between the secondary shield and the outer wall of containment were auxiliary systems. The containment building was closely abutted to the spent fuel building and turbine building to ease fuel bundle transfers and keep steam pipe runs short.

The Connecticut Yankee reactor was generally similar to the Rowe reactor but the increased output required a wider and higher vessel, weighing over twice as much and operating at greater pressure. The lower control rod supports used at Rowe were eliminated so the core sat lower in the reactor. Unlike Rowe, the bottom head of the reactor was penetrated for instrumentation devices. The nuclear fuel was clad with stainless steel. While the Zircaloy cladding used at Shippingport had superior nuclear properties, it was considered hard to fabricate and not worth the cost at that time.[147] In later years, Zircaloy rods were tested at Rowe, and two complete assemblies of Zircaloy clad rods were included in early Connecticut Yankee cores for testing.[148] As a result of these tests which showed the potential for longer core life, the fuel rods were being completely converted to Zircaloy cladding in the years before shutdown.[149] Control rod materials were the same as at Rowe. The "canned" main coolant pumps used in the first two Westinghouse stations were succeeded by a new shaft-seal design with almost three times more output. They also incorporated flywheels which insured vital extra seconds of pumping power after a power failure. Since the reactor was nearly at sub-grade a horizontal fuel canal connected the reactor cavity and the spent fuel pool.

Secondary Systems for Electrical Generation and Feed water Control

Steam Turbine Design, Construction, Operations

At Connecticut Yankee, the heat energy to rotative energy conversion device was a Westinghouse/Kraftwerke Union (KWU) three-casing, tandem- compound turbine direct-connected to a Westinghouse generator. The nominal turbine output was 619,328 kw with a maximum output of 648,527 kw or over 800,000 hp.[150] At that load, the turbine was taking 7.463 million pounds of steam per hour. The turbine included one high-pressure and two low-pressure units, which was typical of many large nuclear turbines of the era. The turbines of the Connecticut Yankee unit were all on a single shaft with a high-pressure element exhausting into twin low-pressure units. This design was known as a tandem-compound arrangement to distinguish it from cross-compound types which had two or three separate turbine shafts.[151] Turbines were also typed according to the directional flow of steam through the casing. The Connecticut Yankee units were double-axial flow types, in which the steam entered the center of the casing and flowed outward to each end.[152] One advantage of this design was that the thrust on the blades was well balanced which simplified the design of the support bearings.[153] The Connecticut Yankee turbines were also categorized by their exhaust arrangements. The steam exhausted each casing from two ports at each end, called quadruple exhaust. The splitting of the exhaust path allowed a greater flow without greatly increasing the size of the casing ends. This was an additional benefit of the double-flow design.[154] The steam flow volume dictated the size of the exhaust ports, which in turn dictated the length of the last row of blades in each stage. As discussed below, blade size is a critical factor in turbines because of centrifugal forces acting to pull the blades out by the roots. Casing size and blade length had to be increased as steam pressure dropped and steam volume increased during the flow of steam through the turbine.[155] The constraint of relatively poor steam conditions from the pressurized water reactor generators on exhaust-port design and blade-tip speed required larger-diameter blading in the last stages than were found in fossil-fuel power stations.

Developing steam turbine designs to operate with the first generation of full-scale nuclear reactors of the late 1960's proved be to an engineering challenge for Westinghouse and General Electric.[156] The pressurized water reactors (PWR) favored by Westinghouse, Babcock & Wilcox (B&W) and Combustion Engineering had a design limitation: their use of ordinary water as the reactor coolant severely limited the pressure and temperature of delivered steam.[157] The transfer of heat from the reactor to the steam generators by an indirect heat exchange loop contributed to this problem.[158] Even the boiling water reactors (BWR) of General Electric were limited in their output temperature.[159] The Connecticut Yankee reactor produced steam at 690 psi and 501° F. Coal and oil fired central stations of the early 1960's generally had boilers operating at over 3000 psig and 1000° F.[160] The direct impingement of combustion gases on the water filled generating tubes explained part of their higher operating conditions. In addition, fossil fuel plants utilized superheaters to add extra heat to the steam by running the steam back through the boiler before it went to the turbines. The high temperature steam was very dry which simplified the engineering of the turbines. The pressurized water reactors of the 1960s could not provide any superheat. In an attempt to achieve higher temperatures, some early plants such as Con Edison's B&W-built Indian Point Plant of 1965 had an oil-fired superheater to improve the steam conditions.[161] The nuclear power industry did not pursue that solution. B&W turned to "Once Through" steam generators in the early 1970's which gave a modest degree of superheat.[162] Westinghouse and Combustion Engineering continued on with their proven U-tube generators producing saturated steam, with a temperature the same as that of the water from which it was liberated.[163] Having made that decision, Westinghouse attempted to design turbogenerators that could effectively utilize huge amounts of relatively poor quality steam. These were machines that were as large or larger than existing high-pressure, high-temperature fossil-fuel designs — and had to be because the economics of relatively small nuclear plants were poor.[164]

In designing the Connecticut Yankee turbines for relatively low-pressure, low-temperature steam conditions, Westinghouse had to build units working on steam conditions not common in large power stations since the late 1920s,[165] by which time a number of reliable designs were available. The Connecticut Yankee use of the three-casing, tandem-compound turbine direct-connected to a generator was a direct descendant of the groundbreaking reaction turbine design patented by Sir Charles Parsons in England in 1884, for which the Westinghouse Electric Corporation of East Pittsburgh was the original American licensee.[166] The three-casing arrangement was an efficient, practical way of handling the huge increase in volume that occurs as steam works its way through the turbine.[167] In Parsons' early machines, a number of increasing diameter blade wheels in a single unit utilized the energy of the steam as it flowed through the blades, losing pressure and gaining in volume. As steam pressures got higher in the twentieth century, builders split the turbine blade stages into high pressure (hp) and low pressure (lp) casings (called compounding) with the steam passing out of the hp turbine via exhaust ports and into the lp turbine.[168] Each casing was larger to accommodate the increase in volume. In addition, each casing was enlarged at the exhaust end to provide free flow for the steam.[169] The stationary and moving turbine blades increased in length along the steam flow path to fill the casings.

By 1920 the first large central station turbines in the 15,000-60,000 kw range built by General Electric and Westinghouse had several years" operational experience which included a spate of serious accidents. In some cases blades were completely shed from their mounting discs, in others the discs burst at high speed and wrecked the casings which also cracked from temperature stress.[170] It was clear that engineering had not kept up with the size of the machines. The turbine rotors of that period were of built-up design, including forged spindles with cast steel blade attachment discs bolted or pressed on.[171] The bores of the discs were actually machined slightly smaller than their mating spindle diameter. For assembly, the discs were heated to expand the hole and then forced on the spindle with hydraulic pressure. When the components returned to normal temperature they were locked together producing considerable stress at the mating surfaces. A machined steel key was inserted into a slot cut in both the spindle and disc to prevent rotative separation. Investigators used high speed photography on test rotors which showed that the disc wheels were flexing. Metallurgical examinations showed cracks were emanating from keyways, balancing holes and any rough discontinuities in surfaces. Remedies included stiffening the discs, using forgings instead of castings and rounding off corners in the key way areas. It was discovered that cast iron casings "grew" from the higher temperature steam produced by pulverized coal boilers requiring substitution of cast steel. There were also problems with blading clearance, oiling systems and bearings that had to be addressed. At that time the primary problem of turbine builders was blade design. Securing the rotating turbine blades from destruction by vibration required advanced metallurgy and specialized mechanical fastenings. Erosion of the blades from wet steam in the last stages also became a problem.[172]

The solutions to these problems emerged from cooperative engineering between boiler manufactures and the turbine makers. Westinghouse started building one-piece forged rotors in the early 1920s which reduced the risk of assembly flaws.[173] Boiler designers increased the superheat so that the steam stayed dry through the turbine cycle reducing the chance of wet steam damaging the elements. They also increased pressures, which helped turbine designers tackle another high-risk area: the blades in the last rows of the low pressure sections.[174] The longer blades in those areas were particularly susceptible to stress cracking at their attachment roots, wearing along their impingement surfaces (erosion) , and centrifugal force working to tear them out of the discs. At the same time it was recognized that steel under stress was particularly vulnerable to corrosive media, a condition first called "season cracking" in the early twentieth century due to its occurrence during wet weather. It was later known as Stress Corrosion Cracking.[175] This phenomenon was observed by jewelers in the 19th century,[176] and was seen in brass cartridge cases shipped from Britain to arsenals in India in the early 20th century. It resulted from corrosive media attacking metal parts under mechanical stress produced by applied forces, forming operations, or expansion/contraction.[177]

The higher pressures generated by advanced fossil-fuel boilers of the mid-twentieth century mitigated stress corrosion cracking and allowed turbine designers to build smaller machines with high outputs. The smaller machines operating at 3600 rpm had solid forged rotors which were resistant to mechanical failure. It was easier to engineer blading near the smaller exhaust ports and the blades edges were protected from erosion by attached hard metal alloy strips.[178] The high speed engineering also saved money in manufacturing and foundations.[179] By 1950 there were reliable standardized designs putting out 100,000 kw (100mw) at 3600 rpm. The largest turbines of the period still required built-up rotors, and had occasional failures,[180] but the excellent steam conditions produced by fossil fuel boilers of that era ensured reasonable reliability.[181] Output pressure and temperature were slowly increased through the 1950s. By the time the Connecticut Yankee unit was ordered, coal-or oil-fired boiler/turbine generators were producing 600 mw.[182] Thus the trend in the first hundred years of turbine development was to produce the smallest possible machines producing the highest power at high speed with reliability. This achievement was aided by boiler designs that produced huge amounts of steam at very high pressures with high superheat.

A 1966 article on the Connecticut Yankee turbines and their contemporaries in *Westinghouse Engineer*, the company's house magazine, described the extra lengths that their engineers took to make the huge nuclear power station turbines function in the poor steam conditions.[183] Inlet and exhaust ports had to be larger than those at contemporary fossil-fuel plants to do the same work. Rotor speed had to be cut back to 1800 rpm to prevent erosion of the very long last row blades resulting from the large ports.[184] General Electric (Westinghouse's chief competitor) used the same rational for its low-speed nuclear turbines.[185] The lack of superheat required steam drying between the stages to protect the vulnerable low-pressure units from moisture. Live steam from the reactor was then used to bring the temperature back up. This process of reheating between stages was used in fossil fuel stations, but the 90-100 degrees of reheat obtained in the Connecticut Yankee plant and contemporary nuclear plants was very low in comparison to levels obtained in fossil fueled stations. Live steam was even sent direct to the exhaust ends in an attempt to pull out entrained water. Large amounts of steam were used in these areas, but it was not really a problem because the reactor was sized to produce so much more steam than was needed for actually powering the turbines. More water removal occurred at extraction points where steam was bled off to heat the feed water going back into the steam generators.

Turbine Governing

All the calculations for over speed parameters depended on the turbine governing devices doing their job within specified limits. The function of the turbine governing system was to control the speed of the unit to ensure that the generator was producing even, continuous, high quality electric power. In addition the governing devices prevented over speeding which could lead to explosive destruction of critical components. The Connecticut Yankee governing system evolved from the flyball governors used by millers in the 17th century to control speed in their corn mills.[186] This was an early feedback device: a self-regulating mechanism.[187] James Watt later took that design and patented it for steam engine governing. He used it to open and shut a valve on the steam pipe.[188] Later designers adapted the flyball governor to control the admission of steam by varying the settings of the steam admission valves with mechanical linkages. Westinghouse used the same device on its first turbines.[189] By the nineteen teens, the necessity of precise speed control for electricity generation from large turbines led to the oil pressure relay type governor.[190] This utilized the pressurized bearing lubricating oil as a working fluid. The Connecticut Yankee governor dispensed with the mechanical complication of a large rotating flyball governor and instead activated the relays with speed sensitive control oil pressure supplied by a pump impeller on the turbine shaft.[191] A key element in the Connecticut Yankee turbine governor was the servo control system developed in the 1920s in which a powerful control motion was produced from a remote and relatively weak signal via sensors, amplifiers, and servo-motors.[192]

Condensate and Feed water Components

After the steam finished its work in the turbines, it was condensed back to water and recycled. Two surface condensers (nos. 1A and 1B) stood directly below the low-pressure turbines. The condensers were of shell-and-tube construction in which cooling water and exhaust steam were not mixed, a standard design evolved from mid-19th-century steamships which needed fresh water for feeding high pressure boilers.[193] In principal the Connecticut Yankee condensers simply reversed the heat exchange of the steam generators. Water pumped from the Connecticut River flowing through tubes cooled and condensed the surrounding steam. At full load, 93,000 gallons per minute (gpm) of water was required for condensation.[194] The condensed steam (condensate) was the main source of feed water for the steam generators. The collection points were the hot wells which constituted the lower two feet of the condenser shell, and normally held 33,000 gallons -- enough for 2.75 minutes of steaming.[195] Two condensate feed pumps on the Turbine Building ground floor removed water from the hot wells and directed it to Reactor Containment.[196]

The Connecticut Yankee condensate component design was a "once through" system for condensing the used steam and returning it to the boilers. All the cooling water needed was drawn from the Connecticut River and sent back to the river in a heated condition.[197] This was a common choice in the less environmentally-aware early 1960s. The other, more expensive option would have been a closed system in which the condensing water would be cooled in towers and sent back into the condensers.[198] Westinghouse was an early advocate of marine-type shell-and-tube surface condensers for utility steam turbines like those supplied to Connecticut Yankee.[199] Surface condensers were originally necessary for preserving fresh water boiler feed in steam ships operating in salt water. Most early land turbine installations used simpler condenser types operating on barometric or jet mixing principals. The increasing size of turbines in the twentieth century led to widespread reliance on the ability of surface types to condense large amounts of steam and provide high levels of vacuum.[200] Their heavy water flow required plant siting near rivers, lakes, or oceans. Because they were originally designed to operate in corrosive ocean salt environments, they had non-ferrous metal tubes to resist wastage. This technology transferred well to power plants in tidal estuaries where salt or brackish water was the rule. Their complex construction with thousands of closely spaced tubes was still vulnerable to corrosion and fouling by biological organisms.[201] Tube material had to be carefully chosen to suit the particular local water chemistry. Choice of a closed cycle cooling system would have eliminated biofouling and reduced the chance of corrosion.[202] In the Connecticut Yankee units, the bulk of the original tubing was fabricated of Admiralty Brass. The brass tubes deteriorated due to ammonia induced stress cracking, but operation continued by plugging the affected tubes. This could only be considered a stop-gap repair since output would ultimately drop.[203] Failing condenser tubes was a problem in many aging American power plants, and as discussed below eventually led to complete tube replacement at Connecticut Yankee.[204]

Studies done to determine that there would be no impact on fish and bird life in the river adjacent to and downstream from the plant were completed after plant design and construction.[205]

Generator and Transformer Designs

Connecticut Yankee turbines drove a 1,000,000-hp generator which proved far more reliable than the low pressure turbines and condensers. Although the generator also had to be scaled up for such large output at half speed, improvements in mechanical construction, metallurgy, insulation, and cooling during the previous sixty years of development kept pace with the engineering requirements.

The basic form of the Connecticut Yankee generator evolved from designs developed for European systems of alternating-current high-tension transmission. George Westinghouse recognized the superiority of this system over direct current as early as 1885 and he aggressively purchased patents and licenses from several engineers on the continent and from Tesla in this country in an attempt to lock in the technology.[206] The main benefit of AC high-tension distribution was in economy of copper transmission wire, which was a major portion of the capital expense of electrification.[207] A strong influence on emerging technology was the experimental 108-mile 25,000-volt polyphase transmission from Lauffen to Frankfort in Germany in 1891. This installation pioneered high tension, three-phase transmission, with a water powered revolving-field generator and step-up transformers.[208] The use of a three-phase generator gave smoother power, greater capacity and saved money in conductors.[209] In addition the first reliable AC motors worked better on a polyphase system.[210] The main constructional feature was the use of a revolving-field magnet surrounded by stationary armature conductors. This arrangement (which reversed earlier practice in which the armature revolved inside the stationary field magnets), disposed the main elements where they could add to structural simplicity and strength. The copper conductors arrayed in the stationary armature were easier to brace against displacement by electromotive forces. This also eliminated the difficulties of taking high voltages and currents from a moving element.[211] The invention of silicon steel for the conductor-supporting laminations greatly reduced stray currents allowing more output per pound of metal.[212] Placing the relatively simple field magnet wiring on the rotating armature allowed this element to be strongly built to resist centrifugal forces.

Increasingly efficient insulating materials for the copper conductors also played a part in making the early nuclear era generators possible. Varnished paper and cloth used in the first generators gave way to mica in the 1890s.[213] In the 1920s asphalt-bonded mica was the norm. Resin-bonded mica and fiberglass came into use around 1950.[214] The most important factor in making large generators like the Connecticut Yankee unit possible was hydrogen cooling. In the late 1920s, designers realized that improvements in heat removal in natural circulation air-cooled generators would enable them to get higher outputs from smaller machines.[215] Water cooling the air helped, but the Swiss invention of hydrogen cooling in the 1920's paved the way to greater outputs.[216] Hydrogen's lower density and higher heat transfer boosted outputs. At first the pressure was just high enough to keep out air.[217] By increasing the pressure to 60 psi and ducting gas through the conductors, reliable machines of 600 to 1000 mw served the Westinghouse built plants.

Essential to the economics of long-distance transmission was the adoption of alternating currents and step- up transformers. One of the technologies that Westinghouse acquired was the transformer design of Gaulard and Gibbs in Britain.[218] They utilized the principal of electromagnetic induction: current entering the "primary" coil of copper discs at one end of a magnetic circuit produced an electro-magnetic flux which induced a current in an opposite "secondary" coil. Westinghouse engineers rapidly improved this design using copper wire coils and stacked iron plates for the magnetic circuit.[219] By increasing the number of wire turns on the second leg, a relatively low voltage/high current (amperage) incoming current produced an opposite high voltage/low amperage output.[220] This had two features that aided long distance transmission: it allowed for more powerful generators which did not need hard-to-engineer high-voltage connections, and economized the use of copper transmission wire. The high voltage/low amp output of the new transformers allowed much more electricity to be sent though a given wire size that could be economically strung for hundreds of miles.[221] At the receiving end, the same type of transformers reduced the voltage to a safe level for industry or home use. The main areas of development were similar to those for generators: core, insulation, and cooling. The critical core metallurgy was a challenge early on to designers because with ordinary iron, electrical losses increased with time.[222] The same silicon steel used in armature cores solved that problem.[223] Insulation materials evolved along much the same lines as in generators. By the 1930s designers began to design units that could withstand lightning strikes which required a new order of testing, mechanical integrity, and insulation surge resistance.[224] Early transformers tended to be cooled by either forced air or oil, with oil becoming the predominant method for power stations. Natural convection of the heated oil gave way to water cooling of the oil and later to forced oil circulation in external tubed coolers.[225] By mid-century, thermostat-activated fans were added to draw the heat off from the oil in the cooling banks.[226] The Connecticut Yankee output transformer was derived from those "double- and triple-rated" units, as was a step-down transformer which produced a lower voltage to supply the reactor coolant pumps.

Summary of Connecticut Yankee Operations and Repair Issues 1970-1974

The first refueling shutdown began in April 1970 and took about two months, with later refuelings scheduled roughly every year. Each new core required extensive design and engineering of the fuel arrangement, control rods and moderator chemistry to ensure the required power output. Refueling shutdowns also allowed for operational improvements and introduction of other new or re-designed facilities. During the first refueling episode, a new Diesel Generator Building was completed to enhance auxiliary power supply. Enhanced or enlarged facilities to process gaseous and liquid nuclear waste were completed in 1973-74, largely during the fourth refueling shutdown. Until mid-1973, plant designers and operators proclaimed satisfaction with what appeared to be trouble-free operations and the production of some 21 billion kilowatt hours, with particular satisfaction expressed about turbine performance.[227] At about the same time, a GE engineer stated that the erosion rate of their nuclear turbines was no worse than their fossil fuel units,[228] and *Nuclear Safety*, the bimonthly review of the Atomic Energy Commission, indicated no turbine erosion or corrosion problems after reviewing the performance of twenty-eight light water reactors.[229] Soon after these articles appeared, however, significant design problems appeared in the steam generators and low-pressure turbines.

Turbine Repairs 1973-1974 - Emerging Issues of Stress Corrosion Cracking and Erosion

The low-pressure turbines at Connecticut Yankee began to fail in the Spring of 1973.[230] The first repair on the No. 2 unit was in June 1973, and the spindles on both low-pressure units were replaced between July-December of that year, possibly using spares provided by Westinghouse. The rotors must have been severely degraded since the repair was not done during refueling.[231] The station was out of service for over five months. Another month-long repair requiring shut-down began in 1974. The problems included disc cracking, blade root cracking, and erosion of the stationary and rotating blading.[232] Indirect evidence from other plants suggests these repairs reflected stress corrosion problems which began soon after Connecticut Yankee began full-power operations in 1969, if not earlier and a re-emergence of the blade erosion problems that had occurred in the early 20th century.

Beginning in 1965, while Connecticut Yankee was under construction, the steam generators and turbines in one of the pioneer nuclear power stations of the Central Electricity Generating Board (C.E.G.B.) in England began to show damage from feed water impurities.[233] Just before the low-pressure turbine problems at Connecticut Yankee became evident, a groundbreaking report on a turbine failure in another C.E.G.B. station reached the engineering journals. In 1969 the Hinkley Point 'A' Nuclear Station had a catastrophic failure of the low-pressure blade discs on one of its turbines. The unit was very similar in layout and construction to the Connecticut Yankee unit, though smaller and running at 3000 rpm. A 2½-year investigation revealed that the discs failed due to stress corrosion cracking. The report found that the causes of the corrosion were minute impurities in the steam, attacking very tiny defects in the disc attachment points.[234] By this time, stress corrosion cracking was well understood at the molecular level, although it was many years before the engineering caught up with the science.[235] Failure from stress corrosion cracking takes about four years to develop — about the length of time between full-power operation and first turbine repairs at Connecticut Yankee.[236]

The problems afflicting the Connecticut Yankee low-pressure turbines were common in the first generation of Westinghouse units. The near-sister plant to Connecticut Yankee at San Onofre, California had problems with cracking in the keyways that locked in the blade discs to the spindles. The carbon steel of the rotors could not handle the relatively high moisture content.[237] Brookwood #1 (now Ginna) of Rochester Power and Light suffered blade failures with blade ejection requiring operation with the last row blades removed while engineers tried to find solutions.[238] Ten years after delivery of the Connecticut Yankee turbines, some engineers still felt that conditions in low-pressure nuclear units were not very different from fossil-fired units and did not require new engineering.[239] Westinghouse's (and GE's) assumption that lower rotor speeds would reduce the erosion of the last row blades may have been in error as later observations found that high revolutions led to longer blade life.[240] Until c1973, GE turbines operating at PWR and BWR plants had not shown signs of stress corrosion cracking.[241]

While the incidents undoubtedly caused economic harm to the utilities, public notice was probably muted by the lack of safety issues, the subsequent Three Mile Island accident, and later ongoing problems with the more newsworthy steam generator tubing. It is clear that the Connecticut Yankee low-pressure turbine problems were caused by unforeseen engineering decisions and manufacturing methods used by Westinghouse. Excluding the C.E.G.B stations, these types of turbine failures were very much a United States problem that also affected non-nuclear turbines with increasing frequency from 1964 to 1973.[242] For various reasons, similar turbines, even American-made ones, did not fail in Germany or Japan.[243] A critical component in stress corrosion control is the chemical quality of the feed water going into the system. Very slight rises in salinity or minerals could exacerbate the deterioration of components in the steam path.[244] The Hinkley Point investigators even found that stress corrosion cracking could also be induced by certain water purity control chemicals.[245] However, in reviewing all the C.E.G.B. stations that had cracking, they found that though water quality varied it was still within operating specifications. In addition they felt that it would have been impossible to expect the controls to be any better. Their recommendations were that the details of the highly vulnerable disc keyways had to be better engineered.[246] The cooling water intakes for Connecticut Yankee were in theory sited upstream of observed salinity, and the operating engineers kept fairly close controls to prevent its entry into the system.[247] Water quality issues were also central to problems with steam generators at Connecticut Yankee, and at many other nuclear plants.

Connecticut Yankee and Worldwide Steam Generator Problems

Shortly after full power operation began in 1970, leaks in the steam generator tubes were detected.[248] Tube leaks allowed irradiated primary coolant water past the Reactor Coolant System (RCS) boundary, creating a safety hazard. Consequently, federal regulations set maximum gallons per day leakage rate for all steam generators.[249] Initially no repairs were needed. Between June of 1973 and April of 1974, during repairs on the low-pressure turbines, steam generator tube leaks were stopped by explosive plug welding.[250] This technique was developed around 1970 in answer to the difficulties of closing tubes in an irradiated area with manual plugging or welding. On detonation of a small nitroglycerine-based charge, molten metal cleaned the inside of the tube, and weld positioned a wooden plug to block loss of coolant.[251] While plugging was effective in preventing coolant loss, it could only be considered a stop-gap as plant output would drop if too many tubes were plugged.

The history and causes of steam generator problems were related to the issues noted for turbine blades and rotors, and were found in nuclear plants built before Connecticut Yankee. In early 1958, after only a few months of operation, one of the B&W horizontal u-tube steam generators at the

Shippingport station developed a tube leak. Testing with an eddy-current device[P] showed extensive surface stress corrosion cracking. An analysis by Westinghouse and Duquesne concluded that the chemicals then being used in Duquesne's fossil-fueled boilers to control oxygen levels (sulfite) and the pH of the feed water (phosphate) were not as effective in the smaller nuclear (in comparison to fossil-fuel) steam generators.[252] Changes in the type of phosphate, amounts of sulfite and modifications to the tube arrangements controlled the problems.

The British investigations of feed water contaminants in the late 1960s, noted above, found that steam generator tubes were scarred by corrosion.[253] An intensive investigation by the British board indicated several causes including steam generator tube configurations, breakdown of the feed water heater tubing and carryover of the particles, insufficient air removal from condensed steam, and failure of resin filter beds to remove organic compounds from lake-sourced feed water. Improving the filter media, eliminating air from the feed water and adding phosphate water treatment largely resolved the problems.[254] Six years after Yankee Rowe began operation in 1960, leaks were detected in its steam generators with stainless steel tubes, but operators controlled the problems.[255] A few years later, utilities with Westinghouse, B&W, and CE units tubed with Inconel 600 (nickel-iron-chrome alloy[256]) tubes began to have an epidemic of tube and tube support failures that eventually affected forty stations. Some plants had steam generators with 20% of their tubes plugged to prevent leakage.[257] The types of degradation included wastage, pitting, denting, cracking of the tubes, support plate damage, and mechanical damage from vibration and loose parts.

In analyzing steam generator problems, a 1988 NRC report found a complex interaction between the mechanical design, materials, fabrication methods, water treatment chemistry and corrosion products from the plant secondary systems, mainly the condensers.[258] The construction of the steam generators, with thousands of tubes, tube seal plates, and bracing, provided many areas for corrosion materials to accumulate and do damage. Designers of the first PWRs saw the water in the primary reactor/steam generator loop as a potential source of reactor or steam-generator tube damage due to the addition of acidic boron to help control the nuclear reactions. While they specified precise chemical parameters for that system, less attention was given to secondary system feed water control.[259] In both fossil- and nuclear-fueled stations, the secondary system condensers that returned the steam leaving the turbines to water were designed to be the "first line of defense" against the entry of corrosion products, yet they also could be their main source.[260] The thousands of tubes providing the condensing surface area had to be sealed into tube plates at both ends and were acknowledged by American operators to be practically impossible to keep leak-free.[261] The

---

[P]Eddy-current testing of boiler tubes is a remote, non-destructive procedure that utilizes electro-magnetic fields from a probe to find faults by a change in signal intensity caused by variances in wall thickness and cracking. (Singley et al 1959: 753) fact that the condenser shell was operating in a vacuum meant impurities and air in the cooling water drawn from oceans, rivers, or lakes would be sucked out of any leaking tubes to mix with the condensing steam. As a result, air, chlorides, hydroxides and up to ninety other chemical compounds could enter the system and induce stress corrosion cracking in the steam generator tubes and also the turbine blade roots and mounting discs.[262]

"Deaerating" sections of the condensers, air ejectors and chemical water treatment were necessary to mitigate these problems. Recognizing that condensers and mechanical devices alone could not bring oxygen levels low enough to prevent corrosion, engineers of fossil-fueled stations had started using extensive chemical feed water treatment around 1950. Most stations used phosphate to keep water pH below corrosion thresholds.[263] Phosphate treatment did not alleviate the dissolved oxygen content of the water, so some utilities added sulfites or hydrazine as oxygen scavengers in the mid 1950s.[264] Later ammonia or morpholine-q were added to control the pH of the water.[265] Initially positive results were mitigated by increasing evidence that the hydrazine or ammonia attacked the copper tubes in feed heaters and condensers, leading to suggestions that those units be entirely tubed with carbon or stainless steel.[266]

American nuclear plants through the generation including Connecticut Yankee used phosphate treatment until it became evident that the phosphate built up as sludge on tube sheets causing wastage and thinning of the steam generator tubes.[267] For better control of oxygen content and to forestall tube deterioration, both Connecticut Yankee and San Onofre tried hydrazine injection as early as 1970.[268] With the blessing of the manufacturers, American nuclear plants switched to an ammonia- and hydrazine-based "all volatile treatment" (AVT) around 1974.[269] At first AVT reduced tube plugging, but it then began to cause other problems. The ammonia injected into the feed water (or that produced by the breakdown of the hydrazine), in combination with the dissolved oxygen, attacked the copper-based tubing that had been specified for the demonstration- and commercial-phase feed water heaters (feed train) and condensers.[270] The copper shed from the tubes ended up in the steam generators where it acted as both an oxidizer and catalyst for pitting of the tubes.[271] At the same time, the feed train materials and the chlorides from condenser leaks would lodge in the spaces between the steam generator tubes and their drilled, carbon-steel support plates. A resulting buildup of oxide would then squeeze the tubes (denting) leading to cracking and ruptures.[272] Steel and copper corrosion products would also accumulate on top of the lower tube sheet as a sludge that would lead to stress corrosion cracking (also known as inter-granular attack) damage of tubes.[273] The stress set up when the tubes were bent to form an inverted U-shape made that area particularly vulnerable to stress corrosion cracking. Even rigid control of pH did not

---

[q] Morpholine is a common additive, in ppm concentrations, for pH adjustment in fossil fuel and nuclear power plant steam systems systems. Morpholine's volatility is about the same as water, so once it is added to the water, its concentration becomes distributed evenly in both the water and steam. Its pH adjusting qualities then become distributed throughout the steam plant to provide corrosion protection.

guarantee steam generator health because the right level needed to control copper corrosion would allow iron oxidation.[274] The chemistry problems were exacerbated by operations of the plants at low loads and during startup. The mechanical damage was caused either by loose or foreign parts (some left in the units by assembly or repair teams) impacting the tubes or by vibration due to inadequate supports. Clearance between the tubes and their support plates and bars was necessary in some models of generators but it allowed the large volumes of water and gases traveling through the generators at high speeds to cause relative movement (fretting) leading to tube damage.[275]

These problems perplexed manufacturers and plant operators. From 1968 to 1975, San Onofre modified its phosphate chemistry four times, switching between AVT and phosphate before settling on AVT at the request of Westinghouse or because of its own investigations.[276] Connecticut Yankee documents suggest that the utility also tried different chemistries, and that these choices could have been a causative factor in its steam generator tube problems and its heater tube, condenser tube and turbine replacement projects (see HAER No. CT-185-C-Turbine Building).[277] The power companies were evidently getting insufficient help with these problems from their suppliers, and in 1977 operators formed the Steam Generators Owners Group (SGOG) in conjunction with the Electric Power Research Institute (EPRI) to address the problems.[278] SGOG research showed that the chemistry parameters set by the manufacturers were too loose. As an example, the original specification on chlorides allowed 150 parts per billion (ppb) while the EPRI/SGOG guideline limited it to just 20 ppb.[279] While leak problems were usually manageable, more serious issues began in 1975 with a steam generator tube rupture at the Point Beach #1 Plant of the Wisconsin Michigan Power Co., followed by ruptures in other plants. Tube ruptures allowed much more primary coolant to escape, and could lead to other system failures resulting in serious accidents.[280] In 1978, the NRC designated steam generator tube integrity as an unresolved safety issue.[281] The solutions to these problems took years to develop. Tube plugging was augmented with sleeving in which a section of smaller diameter tube was inserted into the leaking one and mechanically sealed allowing coolant flow.[282] Sleeving did not occur at Connecticut Yankee until after 1987.[283] The AEC/NRC instituted inspection programs in which the entire length of every tube had to be examined by eddy-current testing devices. The costs for inspection could reach $500,000 per day (including replacement power and provision for worker rem exposure[r]) in the Westinghouse stress corrosion cracking steam generators — which had fewer tubes than those of the other makers. The problems described above were not limited to American PWR plants. In the 1970s, French stations using Westinghouse-licensed units experienced tube and tube plate corrosion. Their engineers suggested that the Inconel 600 material had to be improved with heat treatment, that the tube support plates be made of stainless steel, or that the tube/plate interfaces had to be upgraded. They felt it imperative that there be no condenser tube leakage.[284] After some problems in German stations, the plants controlled the problems with improved tube metallurgy

[r] Roentgen Equivalent Man=the quantity of radiation having the same effect on human tissue as one roentgen of X-rays. (Oxford English Dictionary 1989: v.4, p. 576.)

(Incoloy 800), largely leak-free and corrosion-resistant condensers, and case-by-case use of either low-phosphate or all-volatile treatment.[285] In addition, European operators were more willing than American utilities to shut down their plants when chemistry upsets or condenser leaks were detected.[286]

For generator problems caused by secondary system impurities, manufacturers recommended installation of "full flow" condensate "polishing" systems which used resin-filled filter beds to purify continuously all the condensate after it left condenser hot wells.[287] This would have provided much greater control than the more common intermittent treatment, but some engineers opposed it because of high capital and operating costs, and because of concerns that the resins used in the system would cause their own problems. Connecticut Yankee did not add a condensate polishing system, probably for these reasons.[288] Another solution — which Connecticut Yankee instituted around 1977 — was to "blow down" the steam generators frequently to clear out deposits.[s] It is undocumented whether this change in operation required additional plant infrastructure or NRC approval. By the 1980s, nuclear engineers realized that slightly brackish cooling water supplies demanded more advanced metallurgy to prevent bio fouling and stress corrosion cracking.[289] To reduce those conditions at Connecticut Yankee, all the tubes were replaced in 1986 with a proprietary stainless alloy — Trent Sea-Cure, which came on the market in 1979 — in an attempt to prevent damage in those sections, contaminant particle carryover, and subsequent denting in the steam generator tubes.[290]

While most of the PWR steam generator problems were caused by the secondary system water, the chemistry of the primary system could also cause damage. Cases of primary-water-induced cracking of steam generator tubes at the stressed U-bends began to appear.[291] Damage to the reactor could also occur and was a factor in the shutdown of Yankee Rowe station.[292] A more serious development was the discovery in France that the boric acid moderator added to the coolant could attack the reactor head. The Toledo Edison-Cleveland Illuminating Co. Davis-Besse plant in Ohio was shut down in 2002 as a result of severe corrosion around the control rod penetrations.[293]

---

[s] Blowing down (also known as blowing off) a boiler was a water purification technique from the earliest days of steam technology (Rankine 1859: 453). Allowing some of the pressurized water inventory to escape from the boiler removed oil, salt and other contaminants which could settle on heating surfaces and restrict heat transfer leading to premature failure. Bottom blow-off and surface blow-off valves cleared out the two regions where substances generally accumulated. While blowing off wasted heat in conventional boilers, in nuclear boilers it also allowed irradiated water outside of the reactor coolant system boundary requiring storage tanks and filters.

During the era in which PWR owners and manufacturers were working on steam generator problems, operators of GE's BWRs were having their own set of troubles. While the successors to the dual-cycle plants were not saddled with PWR steam generators, they began to have a form of stress corrosion cracking in the recirculation and other stainless-steel reactor piping. Following the PWR owners' lead, the utilities and EPRI formed the Boiling Water Reactor Owners Group which successfully addressed the problems.[294]

Summary of Connecticut Yankee History 1974-1984 and Reactor Cavity Seal Failure

The AEC, evidently satisfied with the measures taken by Connecticut Yankee to repair the failed low- pressure turbine rotors and steam generator tubes while operating under the provisional license, authorized full-power operation at 1825 mwt on December 27, 1974 with Facility Operating License #DPR-61.[295] In 1975 the AEC was replaced by the Nuclear Regulatory Commission and two years later the Department of Energy (DOE) was created. Reflecting the ongoing national failure to accommodate spent fuel, the NRC amended Connecticut Yankee's license to allow an increase in the spent fuel pool capacity from 336 to 1172 assemblies in 1976.[296]

Until the mid-1980s, most major changes in plant facilities or operations were driven by national issues in nuclear plant safety. A 1975 fire in the TVA's unfinished Browns Ferry Unit 3 in Alabama led the NRC to require upgrades of fire protection in all American plants. At Connecticut Yankee, resulting improvements included barriers, detection equipment, and fire-fighting capabilities. Work and materials storage methods were changed, with great emphasis on controlling combustibles and ignition sources. On March 28, 1979, the most serious accident in the history of American commercial nuclear power plant operations occurred at the Jersey Central Power & Light Co. Three Mile Island (TMI) unit 2 facility near Middletown, PA. There were no radiation-exposure consequences, but the reactor overheated and fuel melted. Causes included personnel error, design deficiencies and component failures. As a result of this accident, significant changes were made in the industry. The NRC issued amendment No. 42 "TMI Lessons Learned Category "A" Items for Connecticut Yankee.[297] Changes to the plant from this amendment included new accident monitoring systems, new control room instrumentation, seismic improvements to the Service Building housing the control room, and the construction of an Emergency Operations Facility Building in 1980.[298] The largest fire-protection modification at the plant, a new electric switchgear building proposed in 1986 and completed in 1990, was also an outgrowth of increased accident protection measures.[299]

In 1982-83, Connecticut Yankee modified the reactor cavity seal ring, a vital component of the refueling system, prior to the 1983 refueling.[300] During the 1984 refueling that seal ring failed, leading to the most serious accident in the plant's history.

Refueling and the Reactor Cavity Seal Design, Failure, and Reconstruction

The immersed refueling system relied on the principal of water seeking its own level between connected containers, and included the reactor cavity and its adjacent refueling (transfer) canal in containment, the fuel pool in the spent fuel building, and a transfer tube connecting them. Before refueling, water treated with boron to kill any nuclear reactions was pumped from the Refueling Water Storage Tank into the reactor cavity (surrounding the top of the vessel) to an elevation of 46.5 feet above mean sea level (equal to the elevation of the spent fuel pool) to ensure complete coverage of fuel bundles during the refueling process. The head of the reactor was lifted off by the polar crane as the water level rose, and set down in a circular 47-foot-deep concrete pit within containment. With the water levels equalized, the valves and sluice gate that sealed off the transfer tube were opened, providing a continuous water path to convey spent and new fuel rods between the two structures. The refueling water height 24.5 feet above the open top of the reactor vessel provided enough clearance to fully protect the fuel rod bundles as they were pulled out with a manipulator crane on the refueling floor above the cavity. The crane operator then placed the bundles vertically in an upender machine in the transfer canal next to and below the mouth of the reactor. The upender set them in a horizontal position on a wheeled car that carried them through the transfer tube to the fuel pool.[301] Another upender and crane handled the bundles in the pool. When it was time to bring in new bundles from the spent fuel building the process was reversed.

It was necessary to prevent the water in the reactor cavity from pouring down between the shell of the reactor and the surrounding concrete wall, past the neutron shield tank and then into the floor of the containment building. If that occurred when the canal was open during a transfer, in a worst-case-scenario, the water level in both buildings could drop, possibly enough to expose the entire length of bundles being carried by the cranes or a portion of the bundles in the upenders, and the stored fuel bundles in the pool.[302] The subsequent heating of the rods would have produced high doses of radiation to personnel, fuel cladding failure, and possible release of radiation to the atmosphere.[303] The original reactor cavity seal was a circular steel plate bolted in place between a flange around the top of the reactor vessel and the adjacent concrete, covering the annulus (opening) between the two. During refuelings prior to 1983, the seal had small leaks which led to contamination of the lower portion of the vessel.[304] Connecticut Yankee engineers proposed a new seal device which consisted of a plate surrounding the opening with continuous inflatable rubber boots on the inside and outside diameters. On inflation the boots would pull down T-shaped wedges of rubber to plug the openings between the flange of the reactor and the inner edge of the plate, and between the outside edge of the plate and the surrounding structure. The modification was made and it followed the recommended Plant Design Change Record (PDCR) procedures as outlined in the CFR.[305]

On the morning of August 21, 1984, after the cavity had been filled and the head of the reactor removed prior to refueling, the seal failed. In less than half an hour, all 200,000 gallons of water in the cavity drained down through the seal. The water elevation after the accident was at 22 feet, level with the open top of the vessel. Because all the fuel rods were still under water in the reactor, they were not damaged. The transfer tube had not been opened at the time of the accident so there was no loss of pool water and exposure of stored fuel bundles. Operators initiated the correct actions to begin pumping out water from the floor of containment. There was a small filtered release from the ventilation stack. Connecticut Yankee personnel followed procedures to notify the NRC, the state, and declared an "unusual event" in compliance with the Emergency Plan.[306] Refueling was terminated and the event was declared over when the water in the lower portion of containment was pumped out. On dewatering it was found that the corrosive borated water had penetrated insulation on the bottom of the reactor and piping, requiring removal and repairs.

An investigation by Connecticut Yankee engineers found that the seal had been incorrectly designed and tested, allowing a critical part to deform after inflation under the full "head" of water leading to gross failure.[307] The previous design, though not completely watertight, was more failure-proof. Northeast Utilities (NU) notified other licensee's — twenty-seven reactors had a similar seal — through the Institute of Nuclear Power Operations (INPO) network.[308] INPO was created in 1979 to share information between the utilities and the DOE. NU also evaluated the possible impact on upcoming refueling operations at its three Millstone reactors in Connecticut.[309] As a result of the inquiry, hidden flaws in the refuel system design were revealed prompting the NRC to issue a bulletin to almost every operating or planned reactor in the U.S. about the danger of this type of accident.[310] Several corrective measures were taken by Connecticut Yankee to prevent a recurrence. The seal was redesigned with steel rods to prevent the top portion from deforming and a backup seal was added above the main seal. A fixed wall (cofferdam) was added in front of the canal so that even if the seal failed with the transfer tube open there would still be enough water to cover the stored bundles in the spent fuel pool. The additional height of water would also give operators time to activate pool cooling mechanisms. Operators of the manipulator cranes and upenders were trained to quickly place bundles in transit in a safe position during unplanned cavity drainage.[311] The sluice gate in the transfer tube was redesigned to close against a flow of water pouring out of the pool, an event that was not contemplated in the original design.[312] While there would still be water in the reactor after a failure, it was required that the Residual Heat Removal pump would always be activated to provide additional circulation to prevent heating of the rods still in the core.[313] At some point before 1985 the revised temporary cavity seal ring was replaced with a permanent stainless steel ring. It further reduced the chance of failure, allowed for reactor movement, saved refueling time, and eliminated worker radiation exposure.[314]

On December 12, 1984, the NRC issued a "Notice of Violation and Proposed Imposition of Civil Penalty and Order Modifying License" which instituted an $80,000 fine to the Connecticut Yankee Atomic Power Company. Management elected not to contest the fine but also cited a number of compliance actions on their part which they felt should have reduced the fine. These included the prompt notification to NRC and other utilities, in-depth investigation of the event and other potential causes, an extensive redesign process and co-hosting an INPO workshop on seal failure.[315]

Prior to the accident, NRC inspections revealed two other earlier modifications to the plant that the commission felt were not properly instituted. The changes involved radiation monitors and a control valve in the Post Accident Sampling System. The NRC held an enforcement conference in November 1983 to determine if there was a pattern of inadequate design modification processes.[316] The NRC did not find that to be the case, but Connecticut Yankee instituted an improved PDCR process and sent out a letter to all personnel in Nuclear Engineering and Operations asking for in-depth questioning about all possible circumstances (described as "what ifs?") of future design changes.[317]

Summary of Connecticut Yankee Operations 1986-1996

During 1986, the NRC issued an amendment for Connecticut Yankee regarding specifications for three-loop operation.[318] The four-loop design of Westinghouse reactors allowed one loop (coolant pump, steam generator and associated piping) to be shut down for repair while the station operated at reduced output. Connecticut Yankee rarely operated in that fashion. Construction on the new switchgear building, completed in 1990, was begun to meet updated fire protection criteria, and provide enhanced instrumentation and controls for safe plant shutdown.

By the 1980s, nuclear engineers realized that slightly brackish cooling water supplies demanded more advanced metallurgy to prevent bio fouling and stress corrosion cracking.[319] To reduce those conditions at Connecticut Yankee, all the tubes were replaced in 1986 with a proprietary stainless alloy, Trent Sea-Cure, which came on the market in 1979.[320]

During the fourteenth refueling outage in 1987, additional steam generator tubes were plugged and the low pressure turbines were replaced (see HAER No. CT-185-O-Turbine Building). Containment leak integrity was tested by pressurization. Repairs were made to the attachment devices on the thermal shield surrounding the lower core barrel, probably to reduce flow vibration of the shield.[321] Two NRC resident inspectors put in over 4000 hours during the assessment period before and after the shutdown.[322]

Early in 1989, there was a release of radioactive liquid from the Spent Fuel Building into drainage structures at the nearby 115 kilovolt switchyard, which delivered from other power stations in the system almost all the station service power for start-up and shutdown and power production operation. Clean-up after this event required considerable soil removal. During the fifteenth refueling outage in 1989-90, the thermal shield around the lower core barrel was removed, and the entire core was transferred to the spent fuel pool.[323] Generally only one third of the rods were replaced in each refueling operation, so this may have been an unusual incident, the reasons for which are as yet undocumented but may relate to a fuel reconstitution project.[324]

In May 1990 specimens of Asiatic clams were found in the service water system of the plant. The species (*Cobicula fluminea*) spread rapidly in North American fresh waters. Since fouling by these bivalves could comprise important safety systems, Connecticut Yankee was allowed by the Connecticut DEP to continuously chlorinate the system.[325] From 1989 though 1994 there were no amendments on tube plugging so it must be assumed that either plant chemistry was under control or leak rates were under limits specified in the CFR. Tube plugging was resumed during the 18th refueling in 1995 along with roll expansion repairs of tubes. This was an older, more labor intensive process in which plugs were rolled into both ends of the tubes.[326] During 1996, the last year of generation, additional trip mechanisms were added to control containment high pressure and steam generator blowdowns. The Union of Concerned Scientists (a nuclear watchdog group) claimed that the NRC had found that a critical coolant pipe was undersized and had gone undiscovered for 30 years.[327] In October 1996 a gas bubble formed in the Connecticut Yankee reactor, and unnoticed by operators, had flushed out cooling water.[328] It is undocumented whether the 19th refueling cycle was completed in advance of plant shutdown in December 1996 .

Shutdown of Connecticut Yankee

The decision not to seek license renewal and commence decommissioning was based on a study which showed that due to changing market conditions, Connecticut Yankee's customers would save money if the plant was shut down.[329] A review of the physical state of the plant as shown in CY/AEC/NRC documents combined with the economics of light water reactors from the "commercial" generation of plants can give a picture of what might have caused the Connecticut Yankee directors to decline to ask the NRC for an extension past 2007. License extension was an option that the DOE/NRC encouraged since it gave more time to amortize the costs of upgrading the older plants.[330] The ongoing steam generator problems might have been a factor in the decision. In its 1988 report on steam generator failures, the NRC worked out "value-impact" models to show what utilities could expect to spend during the remaining life of their plants under the stepped-up inspection plans being implemented because of the unresolved steam generator safety issue. The modeling estimated inspection times, plugging man-hours, occupational radiological exposure (ORE), and replacement power to give a picture of downstream costs. Included in the NRC report was the possibility of partially or completely replacing a steam generator, some of which had been operating for only 10 years. From 1981 to 1993, nine Westinghouse plants and two CE plants had Steam Generator Replacement Projects (SGRP's) [331]

The first of these replacements, at Virginia Electric Power's Surry #2, a three-loop Westinghouse plant cost over 200 million dollars and involved 2141 man-rems of radiological exposure. NU spent ten years planning the replacement of two CE generators in its Millstone Unit #2 which had over 3,000 plugged or sleeved tubes.[332] In that repair operation, only the bottom portion of the generator containing the tubes was replaced. Improved methods including use of robotics and a full size mock-up building resulted in greatly reduced costs and ORE. To facilitate future replacement projects, the NRC in 1989 allowed utilities to undertake them without prior NRC review or approval.[333]

With the completion of the Millstone SGRP in 1992, NU certainly had the skill sets to assist Connecticut Yankee in a steam generator replacement program. However, it may have been difficult to find lower tube sections that were compatible with the early model Westinghouse generators. If Connecticut Yankee had elected to replace all or part of the generators it would have had to protect them from further corrosive attack. This could have included installing a condensate polishing system, additional air removal devices in the feed water supply and condensate system and further upgrades of the condensers.[334] The shutdown and decision to decommission the NU-affiliated Yankee Rowe station in 1992 and Millstone #1 in 1996 provided the company with experience in an alternative to upgrading.[335] Another factor in Connecticut Yankee's decision-making process may have been the ongoing problem of keeping an older plant up to the contemporary NRC safety codes. Years of "back fitting" before and after TMI had left older plants over-complicated and crowded.[336] In the final analysis though, national economics alone could have influenced the decision. While the nuclear fuel component of Operating and Maintenance (O&M) costs in 1993 was generally lower than fossil fuel costs, overall O&M for nuclear plants had risen higher than their competition by 1987.[337] Only operating efficiencies were going to improve that ratio, and they were going to be hard to come by at a 28-year-old plant that was nearing the end of its operating license.

Connecticut Yankee in Retrospect

The power industry expected Connecticut Yankee and its contemporary full-scale light-water plants to pave the way for nuclear power to be on par with advanced coal-and oil-burning power stations. In the years during Connecticut Yankee's first refueling cycles, over forty power reactors were on order. Even before the Three Mile Island accident, the number of orders was falling sharply, however. While fossil-fueled plant orders also dropped off due to a recession in 1974-75, 2/3 of the cancellations were in nuclear plants due to their much higher construction costs.[338] Cancellations rose after TMI with the numbers of operating reactors peaking in 1990.[339] It is doubtful that a combination of cheap coal, oil and organized protesters could have been the only factors that limited PWR/BWR power production as a percentage of overall U.S. megawatt hours. They were hurt by their own weaknesses and national nuclear policies: poor siting decisions, inefficient heat cycles, technological flaws, their perceived hazard, and a spent fuel liability instead of a credit. There was clearly a gap between the somewhat messianic pronouncements from theoreticians on the necessity of nuclear power no matter what the cost,[340] and the more level-headed analysis of utility executives who were hoping that eventually the technology would be profitable for their stockholders.[341] While the availability of enriched fuel allowed U.S. firms to build plants with lower capital costs,[342] designers had to cut corners in non-safety areas to compete with fossil-fueled plants. Even a critical safety element, the emergency core cooling system pioneered by Rickover may have been shortchanged, as considerable controversy developed as to whether it would function properly in the event of a major loss of coolant.[343] The technology of these early reactors was probably flawed from the outset (for private utilities) because of its reliance on military designs and AEC-sponsored forced development.[344] The result may have been an over-reaction by the AEC/NRC to the operational problems leading to over-regulation (qualifying every weld) and long licensing entanglements.[345]

It is surprising the utilities did not avail themselves of other reactor systems that the AEC/NRC had supported. Fort St. Vrain, the now-decommissioned AEC demonstration gas-cooled plant in Colorado, showed that there were alternatives to the water reactors. The system was 5% more efficient, produced high-pressure and high-temperature steam, had low worker rem exposure, and was resistant to loss-of-coolant accidents.[346] This nuclear technology (and well-designed and -managed water reactors) is part of the reason why Europe and Japan derive such a high percentage of their electricity from nuclear power while in the United States it is now only 20%.[347] A 1999 article in the New York Times compared reactors to Apollo moon rockets— a technology that slipped into history.[348] None of the "demonstration" phase plants and few of the "commercial" phase ones operated for more than 35 years of their 40 year licenses.[349] Many of 800-1000 mw plants that closely followed Connecticut Yankee, and those built under the 1973 AEC standardization program,[350] have had or will need steam generator replacements or other upgrades to reach that point. NRC- mandated inspections of all reactor heads after the 2002 Davis-Besse incident undoubtedly further diminished the profitability of 1970s-era plants.[351] In spite of these costs, these reactors got a second lease on life because of deregulation in the 1990s. New power entities bought the plants from old-line utilities and applied for license extensions and up-ratings of output.[352] Undoubtedly the lessons learned from the problems of the earlier plants will enable their successors to reach and even exceed the 40-year license milestone.

In 2005, after a 30-year hiatus, four power companies have applied for site approvals for new reactors, all of which are PWR or BWR designs.[353] Perhaps reflecting the damage done to the industry by the Three Mile Island accident, the main innovations in these designs appear to be that they will have simplified passive safety systems that do not require backup generators, pumps or operator actions to contain accidents, echoing one of Rickover's goals at Shippingport.[354] The next generation is possibly just over the border in Canada, in Europe or Japan, and perhaps on digital drawing boards in designs combining heavy water or carbon moderators, fuel breeding, gas or liquid metal cooling, continuous fueling, even directly-driven gas turbines[355] and intrinsic safety.

Despite built-in flaws in some of the primary and secondary systems components, Connecticut Yankee engineers and operators achieved some record performances starting with a 1977 "World Light Water Reactor Record Run" of 344 days. In 1984 a record 417 day run was achieved followed by a 461 day run in 1989 becoming the first plant to have twice exceeded 400 days.[356] In addition it was the first internationally to produce 50 billion and later 60 billion kwh of power. In total, Connecticut Yankee generated over 110 billion kwh, saving over 67 million tons of coal[357] or over 260 million barrels of oil[358] during twenty-eight years of operation.

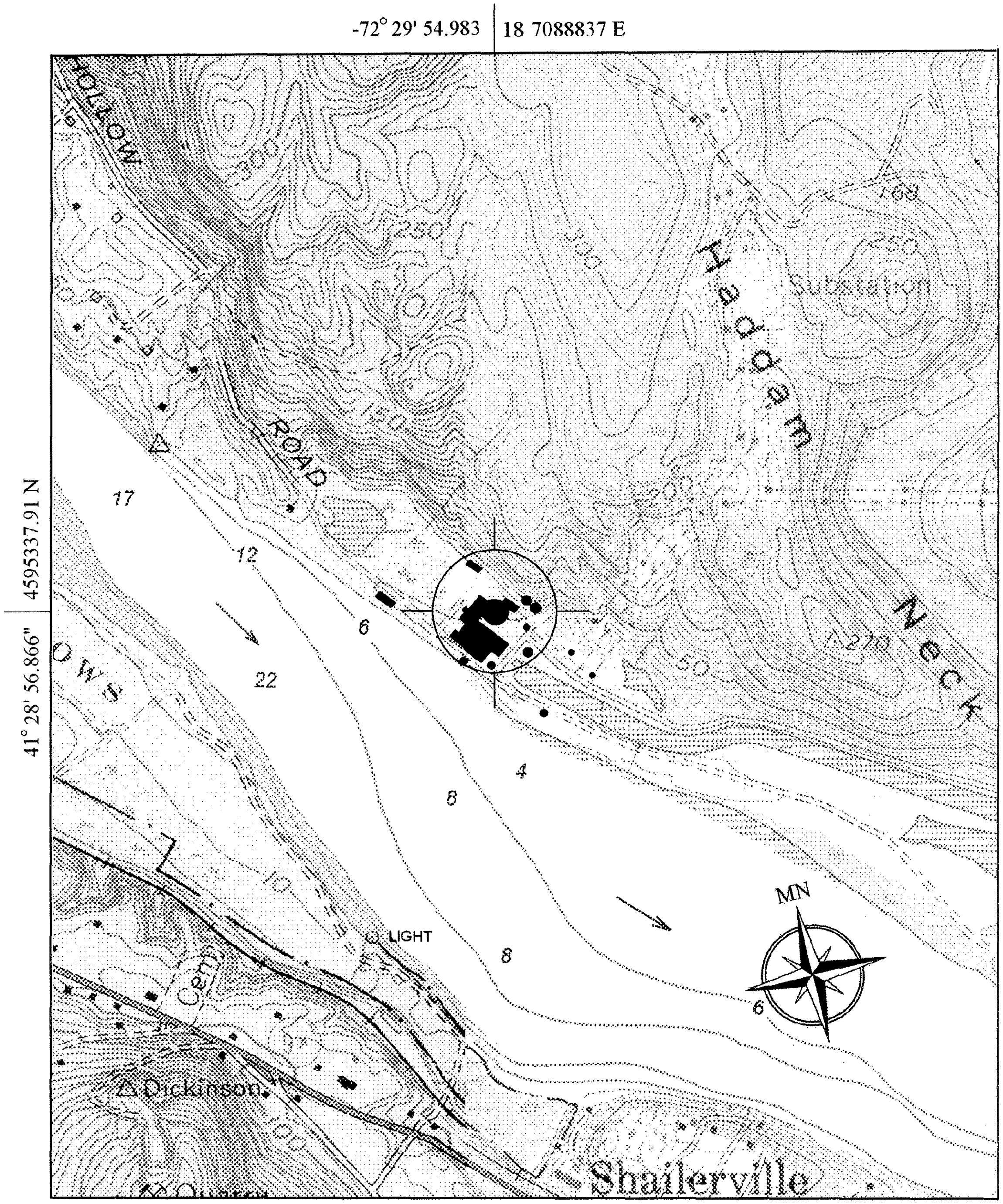


Figure 1 - Location Map - Haddam Neck Nuclear Power Plant
Source: U.S. Geological Survey Deep River, Connecticut Quadrangle - 1961 - rev.1971
(Composite: Map west of site is from Haddam, Connecticut Quadrangle - 1961 - rev. 1971)

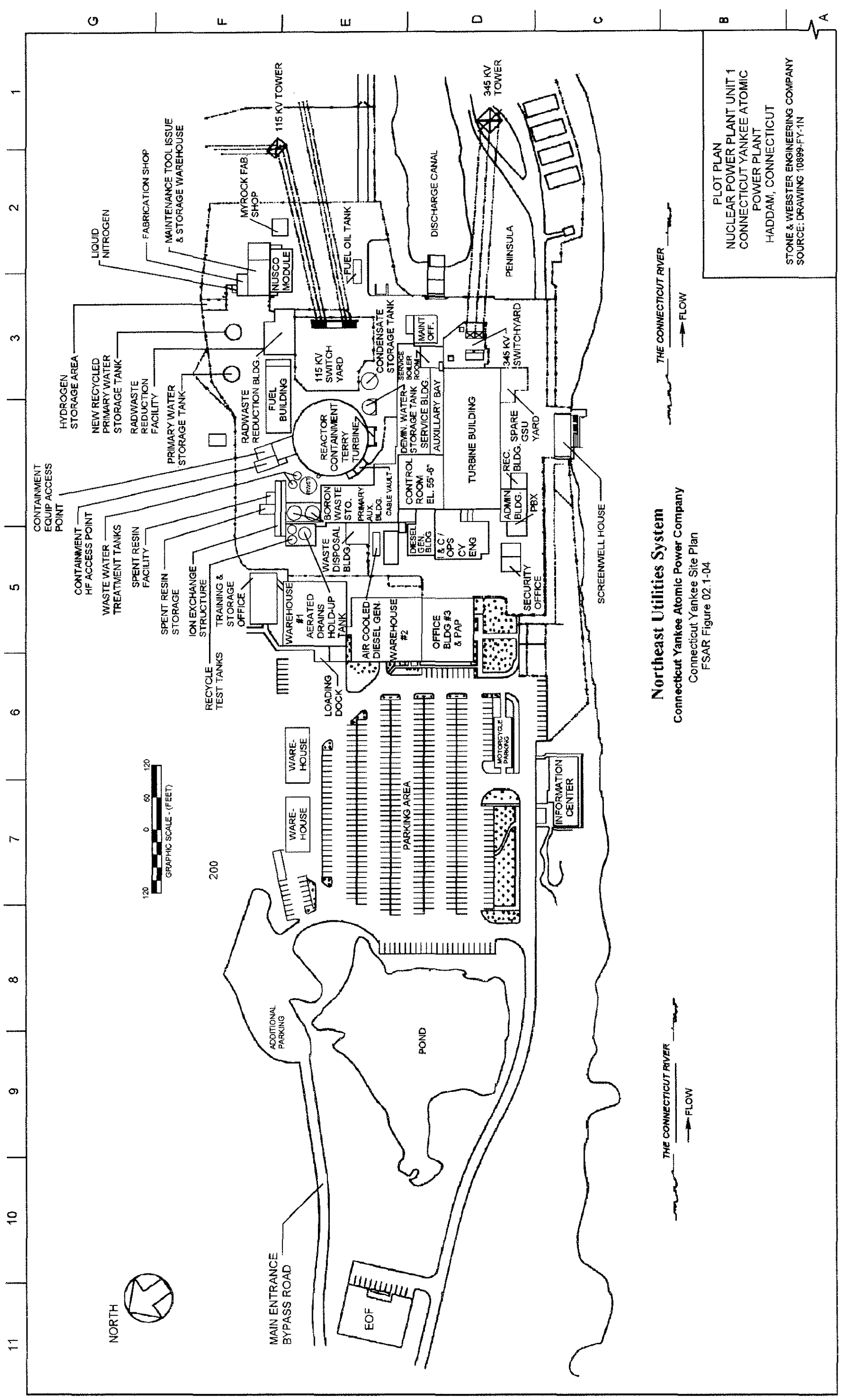


Figure 2 - Site Plan - Haddam Neck Nuclear Power Plant
Source: Stone & Webster Drawing No. 10899-FY-1N

NOTES

---

1. Hogerton 1968: 30, Fortner 2001: 6.

2. Leverett 1955: 23..

3. *Engineering* 1959: 336; Koch 1963: 183; Nuclear Management Company 2002:1.

4. Argonne National Laboratory 1998: 4 of 8. Website.

5. *Nuclear Engineering* 1960: 426.

6. Macintyre and Bathe 1969: 252.

7. *Engineering* 1947: 307; Macintyre and Bathe 1974: 252. Hewlett and Duncan 1974: 27.

8. Hewlett and Duncan 1974: 23.

9. Ibid: 27.

10. Duncan and Moll 1981: 5.

11. Hewlett and Duncan 1974: 30.

12. *General Electric Review* 1955: 15.

13. Hewlett and Duncan 1974: 40.

14. Stephenson 1954: 80.

15. Ibid: 83.

16. Ibid: 170.

17. Simpson 1961: 81.

18. Nuclear Engineering 1958b: 435.

19. Ibid: 72.

20. Ibid: 15.

21. *Janes Fighting Ships* 1967-68: 364.

22. *General Electric Review* 1958: 35.

23. Friend 1964: 613.

24. Ford 1955: 490.

25. Weidenbaum, Sherman and Pashos 1964: 273

26. *Oxford English Dictionary* 1989: v.17, p. 915 (Thermal. 2b. Nucl. Physics.)

27. Ford 1955: 499, Nero and Dennis 1984: 397..

28. Ibid: 500.

29. *Nuclear Engineering* 1960: 425.

30. Haga and Stevenson 1970: 145.

31. Hogerton 1968: 23.

32. Nero and Dennis: 1984: 388.

33. Hewlett and Duncan 1974: 226; Zinn 1957: 20; Ford 1956: 728.

34. Ford 1956: 728, Zinn, Pittman, and Hogerton 1964:32.

35. Babcock and Wilcox 1960: 27-6; Klehm 1963: 47, Hoffman 1958: 78.

36. Agnew 1981: 61; *Power* 1982a: 225.

37. Ford 1956: 730.

38. *Engineering* 1956a: 418.

39. Zinn 1957: 19; Nero 1979: 109.

40. Lewis and Tsui 1960: 24.

41. Duncan and Moll 1981: 6; *Engineering* 1958a:682; Howe 1976: 25.

42. Hewlett and Duncan 1974: 243.

43. *Engineering* 1956b:761, 1958b: 683.

44. *Engineering* 1958a: 682.

45. Hewlett and Duncan 1974: 241, 257.

46. *Engineering* 1958a: 683.

47. Ibid: 682.

48. Duncan and Moll 1981: 11.

49. *Nuclear Engineering* 1958: 239.

50. Niland 1959: 55.

51. *Nuclear Engineering* 1958: 235.

52. Stolz 1958: 10.

53. Hewlett and Duncan 1974: 251. *Nuclear Engineering* 1960: 426.

54. Howe 1976: 42.

55. *General Electric Review* 1958: 30. *Nuclear Engineering* 1957:121.

56. *General Electric Review* 1955: 21.

57. *Nuclear Engineering* 1960: 425.

58. Ackerman 1961: 46.

59. U.S. Nuclear Regulatory Commission 1986.

60. Hogerton 1968: 23.

61. Love, Darrow and Randolph 1958: 7.

62. Brower 1958: 73; Yankee Atomic Energy Company 2005; Connecticut Light and Power Company 2005.

63. Haueter 1974: 200.

64. U.S. Department of Energy 2005; Ratical.org 2005: 6.

65. Witzke and Voysey 1958: 106.

66. Smith 1960: 476

67. Kilpatrick 1960: 466.

68. Witzke and Voysey 1958: 102.

69. Setlur 1975: 137.

70. Love, Darrow, and Randolph 1958: 5.

71. Seymore 1992: 7.

72. Babcock and Wilcox 1960: 27-8.

73. Bergstrom 1959: 106.

74. Ibid: 102.

75. Stern 1964: 248; Klehm 1963: 49.

76. Office of the Federal Register 1962: 3509.

77. Stern 1964: 247.

78. Office of the Federal Register 1962: 3510.

79. Stolz 1958: 10,

80. Jackson 1971: 724.

81. Downs, Deegan and Boggus 1959: 2.

82. Hadjian 1973: 251. Geologic and seismic siting criteria take up 9 pages in the 2004 CFR (CFR 2004: 552).

83. Haga and Stevenson 1970: 146.

84. Lloyd 1958: 2.

85. Ackerman 1961: 48.

86. Lloyd 1958: 7.

87. Ackerman 1961: 48, Ford 1982: 45.

88. Ibid: 50.

89. Fortner 2001: 6 of 8.

90. Rice and Wallin 1964: 191.

91. *Nuclear Engineering* 1956: 118.

92. Morabito 1964: 258.

93. Seaborg 1965: 25, Golan 1965: 196, Proceedings of the American Power Conference 1971: 48.

94. Van Nort and Copeland 1975: 213.

95. Duncan and Moll 1982: 25.

96. Nuclear Management Company 2002: 4. Website.

97. Nero and Dennis 1984: 397.

98. United Electric Light and Power Company 1926: 40.

99. Engineering 1926: 285; Baumann 1921: 504.

100. Berry and Moreton 1922: 500.

101. Power 1922: 762.

102. Hopping 1922: 483.

103. E.g., Connecticut Yankee Atomic Power Company 1966-1974: 8.2-1--8.2-5.

104. *Engineering* 1955b: 198.

105. Huntley 1955: 767-5.

106. *Engineering* 1958b: 682.

107. *General Electric Review* 1954: 5

108. Graham 1965: 210.

109. Ibid: 217.

110. Roddis 1964: 22., Edlund 1963: 128. Haley 1971:239

111. Ritchey 1964: 298.

112. Ibid: 297.

113. Agnew 1981: 63.

114. Hilberry 1957: 27, Lischer 1965: 2, Vennard 1965: 11.

115. Simpson 1961: 82.

116. Gaines 1965: 292.

117. Reichle 1958: 140, Hogerton 1968:29.

118. Risser 1964: 584, 586.

119. Dillard and Baldwin 1964: 167; Dunn 1964: 17.

120. Snouffer and Hanson 1964: 589.

121. Ibid: 589.

122. Bergstrom and Hamming 1965: 87.

123. Jensen 1964: 380.

124. Reichle 1958: 140.

125. Roddis 1964: 22.

126. Seaborg 1965: 24, Roddis 1965: 284.

127. *Engineering* 1967: 92.

128. Dillard and Baldwin 1964: 169; Hogerton 1968: 27; Spencer & Miller 1973: 24.

129. Johnson, Fax, and Townsend 1957: 642.

130. Rickert 1964: 268.

131. McIntyre 1975: 21.

132. Dillard and Baldwin 1964: 169; Hogerton 1968: 27; Spencer and Miller 1973:24.

133. Hogerton 1968: 30.

134. Connecticut Yankee Atomic Power Company 1988: 1; Southern California Edison 2002: 2.

135. Haueter 1974: 200.

136. Connecticut Yankee Atomic Power Company 1988; Northeast Utilities 2005.

137. Stern 1964: 246.

138. Ibid 252.

139. Ibid: 247

140. Connecticut Yankee Atomic Power Company UFSAR 1998: 1.2-1.

141. U.S. Atomic Energy Commission 1974.

142. U.S. Atomic Energy Commission 1969.

143. Connecticut Yankee Atomic Power Company 1975.

144. Coe 1958: 75.

145. Connecticut Yankee Atomic Power Company 1987-1994: Chapter 7, page 4.

146. Ibid: 18.

147. Brower 1958: 74, Coe 1958:75.

148. FDSA 10/70: 4.1-2, 4.2-13.

149. VanNoordennen 2005: Personal correspondence.

150. Connecticut Yankee Atomic Power Company 1998: 1.2-11; Gray 1917: 14.

151. Morgan 1950: 9.

152. Connecticut Yankee Atomic Power Company 1987-1994: Chapter 22, page 1.

153. Church 1935: 9.

154. Morgan 1950: 8.

155. Church 1953:10.

156. Sinton 1966: 110., Spencer & Miller 1973: 24.

157. Hogerton 1968: 30.

158. Nero 1979: 21; *Power* 1982: 212.

159. Ibid: 210.

160. Sinton 1966: 110.

161. Babcock & Wilcox 1972: 23-1.

162. Johnson 1971: 93

163. Power 1971: 94; MacNaughton 1967: 513.

164. Dillard and Baldwin 1964: 169, Hogerton 1968: 27., Spencer & Miller 1973: 24.

165. MacNaughton 1967: 580.

166. Morgan 1950: 7.

167. Johnson 1919.

168. Johnson 1919: 1100.

169. Sinton 1966:112.

170. Bauman 1921: 630.

171. Richardson 1911: 231.

172. Morgan 1950: 11.

173. Ibid.

174. Hossli 1969: 106.,

175. Swann 1966: 73.

176. Swann 1966: 73.

177. Whitaker 1981: 9.

178. Morgan 1950: 11.

179. Hossli 1969: 110.

180. Morgan 1950: 15.

181. Ibid.

182. Brown and Donahue 1964: 15.

183. Sinton 1966: 113.

184. Ibid: 112.

185. Spencer & Miller 1973: 25.

186. Lardner 1836: 105.

187. Nagel 1952: 46.

188. Lardner 1836: 105.

189. Morgan 1950: 12.

190. Johnson 1919: 1116.

191. Connecticut Yankee Atomic Power Company 1987-1993: Ch. 23, p. 12.

192. Oxford English Dictionary (1989 corrected to 1991): XV p. 44; Brown & Campbell 1952: 46.

193. Main 1893: 224.

194. Connecticut Yankee Atomic Power Company 1987-1994: Chapter 41, page 12.

195. Connecticut Yankee Atomic Power Company 1966-1974: 8.4-1.

196. Ibid.

197. Power 1982: 225.

198. Ibid: 226; Penner, ed., 1976 : 71.

199. Morgan 1950: 14.

200. Koester 1908: 250.

201. Connecticut Yankee Atomic Power Company 1987-1993: Ch. 41, p. 26

202. Power 1982: 228.

203. Kinsman 2001:1.

204. Ibid: 1.

205. Merriman 1970.

206. Passer 1953: 136.

207. Hedges 1892: 165.

208. Ibid 1892: 174.

209. Dawes 1928: 104.

210. Hedges 1892:166.

211. Parsons 1911: 190.

212. Ibid: 22.

213. Lafoon 1950: 21.

214. Ibid: 21.

215. Power 1982: 364.

216. Lafoon 1950: 25.

217. Ibid: 25.

218. Passer 1953: 132.

219. Ibid: 135.

220. Dawes 1902, Vol. 2: 209.

221. Hedges 1892: 174.

222. Snyder 1950: 51.

223. Bowers 1982: 183.

224. Snyder 1950: 53.

225. Ibid: 57.

226. Ibid: 57.

227. Williamson 1973: 15.

228. Spencer & Miller 1973: 25

229. Scott 1973: 508.

230. Connecticut Yankee Atomic Power c1975.

231. Ibid.

232. Connecticut Yankee Atomic Power Company 1987: PDCR 886: page 7.2-1 of 13.

233. Lunn and Harvey 1970: 189.

234. Kalderon 1972: 376., Gray 1972: 389.

235. Swann 1966: 81.

236. Jonas 1985: 9:

237. Chetwin 2002

238. Widay 2002

239. Gruber, H. and Reihard. K., 1979: 11.

240. Ibid: 12

241. Spencer & Miller 1973: 25.

242. Ibid: 10.

243. Ibid: 9.

244. Power 1982: 348.

245. Gray 1972: 379.

246. Ibid: 389

247. Merriman 1970; Clark 2003.

248. U.S. Atomic Energy Commission 1974.

249. U.S. Nuclear Regulatory Commission 1988a: 1-1, 2-37.

250. U.S. Atomic Energy Commission 1974.

251. Coughlin, Stark, Brown, and Johnson 1974: 226.

252. Singley, Welinsky, Whirl, and Klein 1959: 752.

253. Lunn and Harvey 1970: 189.

254. Lunn and Harvey 1970: 191.

255. Randazza 1975: 761; U.S. Nuclear Regulatory Commission 1995: 10.

256. Babcock and Wilcox 1972: 29-16

257. Sopocy 1984: 1013.

258. U.S. Nuclear Regulatory Commission 1988a: 1-1.

259. Hopkinson 1982: 24; Randazza 1975: 760.

260. Randazza 1975: 762; Whitaker 1981: 8.

261. Whitaker 1981: 9; Berge and Vignes 1981: 763; Schuecktanz, Riess, and Stieding 1981:757.

262. Randazza 1975: 762; U.S. Nuclear Regulatory Commission 1988a: 2-31; Whitaker 1981: 9; Hopkinson 1982: 24.

263. Noll 1964: 753; Whitaker 1981: 10.

264. Lorenzi 1952: 21.16; Baker 1957: 707.

265. Riedel 1964: 791.

266. Ibid: 791.

267. Mundis and Noble 1981: 751; van Noordennen 2005.

268. Connecticut Yankee Atomic Power Company 1966-1974: 8.7-2; Britt, Millard and DiFilippo 1975: 753.

269. Sopocy and Kovach 1984: 1014.

270. U.S. Nuclear Regulatory Commission 1988a: 2-31; Randazza 1975: 762.

271. Necci 1993: 590; Sopocy 1984: 1016.

272. Mundis and Noble 1981: 749.

273. Singley et al 1959: 755, Huffman and Malinowki 1981: 768.

274. Sopocy and Kovach 1984: 1016

275. Mundis and Noble 1981: 752.

276. Britt, Millard, and DiFillippo 1975: 753.

277. Connecticut Yankee Atomic Power Company 1966-1974: 8.7-2; 1987-1994: Chapter 18, page 9.

278. Mundis and Noble 1981: 748.

279. Sopocy and Kovach 1984: 1014.

280. U.S. Nuclear Regulatory Commission 1988a: 3-9.

281. Ibid: 1-2

282. Ibid: 4-5.

283. U.S. Nuclear Regulatory Commission 1987.

284. Berge and Vignes 1981: 763.

285. Schuecktanz, Riess, and Stieding 1981:757.

286. Dvorin and Schlesinger 1984: 1005.

287. Huffman and Malinowski 1981: 773.

288. Mundis and Noble 1981: 751; van Noordennen 2005.

289. Power 1982: 228.

290. Trent Tube 2004; Northeast Utilities 1985: 2; Northeast Utilities 1986; Connecticut Yankee Atomic Power Company 1987-1994: Chapter 18, pages 11-12, 19.

291. Mundis and Noble 1981: 752.

292. U.S. Nuclear Regulatory Commission 1995: 1.

293. Wald 2002: A1.

294. Danko and Stahlkopf 1984: 654.

295. U.S. Atomic Energy Commission 1974.

296. Connecticut Yankee 1975-1992: Amendment 7, June 8, 1976.

297. Connecticut Yankee 1975-1992: Amendment #42. October 8, 1981.

298. Van Noordennen 2005: Personal correspondence.

299. van Noordennen 2005; Office of the Federal Register 198?.

300. U.S. Nuclear Regulatory Commission 1984b: Inspection Report page 2 of 8.

301. Connecticut Yankee Atomic Power Company 1987-1994: Chapter 9, page 2.

302. U.S. Nuclear Regulatory Commission 1984a: cover letter signed by Richard C. DeYoung.

303. U.S. Nuclear Regulatory Commission 1984e: 2.

304. U.S. Nuclear Regulatory Commission 1984b: Inspection Report page 2 of 8.

305. U.S. Nuclear Regulatory Commission 1984c: cover letter signed by Walter A. Paulson.

306. U.S. Nuclear Regulatory Commission 1984b: Inspection Report page 4 of 8.

307. U.S. Nuclear Regulatory Commission 1984a: cover letter signed by Richard C. DeYoung.

308. U.S. Nuclear Regulatory Commission 1984d.

309. Northeast Utilities n.d.

310. U.S. Nuclear Regulatory Commission 1984a: cover letter signed by Richard C. DeYoung.

311. Connecticut Yankee Atomic Power Company 1985: Attachment 1, page 3.

312. Connecticut Yankee Atomic Power Company 1984: Attachment 4, p. 1.

313. Ibid: Attachment 2, p. 10.

314. Connecticut Yankee Atomic Power Company 1987-1994: Chapter 9: page 6.

315. Connecticut Yankee Atomic Power Company 1985: cover letter.

316. U.S. Nuclear Regulatory Commission 1984a: letter, Richard W. Starostecki to W.G. Counsil.

317. Connecticut Yankee Atomic Power Company 1985: Program Plan, p. 4.

318. Connecticut Yankee 1975-1992: Amendment No. 91. 12/31/86.

319. Power 1982: 228.

320. Trent Tube 2004; Northeast Utilities 1985: 2; Northeast Utilities 1986; Connecticut Yankee Atomic Power Company 1987-1994: Chapter 18, page 11.

321. Cf. Connecticut Yankee Atomic Power Company 1987-1994, Chapter 3: page 5.

322. U.S. Nuclear Regulatory Commission 1988: Docket No. 50-213. 1-2. Oct 4.

323. Connecticut Yankee Atomic Power Company 1975-1992; 1987-1994, Chapter 3: page 5; 1989.

324. Connecticut Yankee Atomic Power Company 1989 and 1987-1994: Chapter 9, page 1.

325. Van Noordennen 2005: Personal Correspondence.

326. Coughlin, Stark, Brown, and Johnson. 1974: 226; Connecticut Yankee Atomic Power Company 1996.

327. Union of Concerned Scientists 2005: Website 3 of 7.

328. *Suburbanchicagonews* 2005: 3 of 4.

329. Connecticut Yankee Atomic Power Company 2005:1.

330. Clauss, Harrison, and Pickens 1993: 234.

331. Davis, Jacobson and Colburn 1993: 605.

332. Necci 1993: 590.

333. Ibid: 609.

334. Sopocy and Kovach 1984: 1015.

335. U.S. Nuclear Regulatory Commission 1995: 1; Dominion 2005: 1.

336. Martel and Stahlkopf 1981: 741.

337. Slade 1993: 212.

338. *Scientific American* 1976: 60A.

339. Wald 1999: 16.

340. Hilberry 1957: 27-28.

341. Vennard 1957: 100; Simpson 1961: 81.

342. Rhoddis 1964: 26.

343. Nero 1979: 89, Ford 1982: 101.

344. *Nuclear Engineering* 1960: 425, Phung 1984: 889.

345. Ibid: 889; Levenson 1981: 40.

346. Agnew 1981.

347. Nuclear Management Company 2002: 5.

348. Wald 1999: 16.

349. Fortner 2001: 6.

350. Gilbert 1975: 157.

351. Wald 2002: A1.

352. Wald 2000: C1.

353. Wald 2005: A16.

354. Ibid 2005: F3

355. Nero 1979: 11.

356. Connecticut Yankee Atomic Power Company 1988: 2

357. Palo, Emmons, and Gardner 1963: 337. 7,000 tons per day for 600 mw x 300 average power days x 28 years = 67,200,000 tons.

358. Babcock and Wilcox 1960: 2-6. 3.9 bbl equivalent of 1 ton of bituminous coal.

SOURCES OF INFORMATION/BIBLIOGRAPHY

A. Engineering Drawings: Archived at University of Connecticut, Dodd Library-Special Collections.

B. Historic Views: See the archival CD containing CT-185 digital files which is contained in the Special Collections at the Dodd Library, University of Connecticut, Storrs, Connecticut. Over 300 photographs of the plant under construction and on stream taken from 1963 to 2000 were assembled and digitized by Eric Alsing of CYAPCO. The file name is Haddam Neck Construction.pdf

C. Bibliography and Personal Communications

Ackerman, Adolph J.
1961 Atomic Power, A Failure in Engineering Responsibility. *Journal of Professional Practice, Proceedings of the American Society of Civil Engineers*. 187, PP3: 43-51. October.

Agnew, Harold M.
1981 Gas-Cooled Nuclear Power Reactors. *Scientific American*. 244, 6: 63. June.

Argonne National Laboratory
1998 *Nuclear History*. The Beginning. 1-8. Available from World Wide Web: http://www.anlw.anl.gov/anlw_history/gen_hist.html.

Babcock & Wilcox
1960 *Steam*. New York. 37th Edition. New York.

1972 *Steam/Its Generation and Use*. 38th Edition. New York.

Baker, M. D.
1957 Three Years Experience with Hydrazine. *Proceedings of the American Power Conference* 19: 707. Chicago: Illinois Institute of Technology, Technology Center.

Balent, Ralph
1959 Large Organic Cooled Power Reactors. *Proceedings of the American Power Conference* 21: 120-131. Chicago: Illinois Institute of Technology, Technology Center.

Battelle
2003 *About Battelle*. Available from World Wide Web: http://www.battelle.org/more/default.stm. 2.

Bauman
1921 Some Recent Developments in Large Steam Turbine Practice. Part 7. *Engineering*. 111: 630-632. May 20.

Berge, J. P. and A. Vignes
1981 French Nuclear Steam Generator Design Philosophy and Operating Experience. *Proceedings of the American Power Conference* 43: 762-767. Chicago: Illinois Institute of Technology, Technology Center.

Bergstom, Richard N.
1959 Containment of Nuclear Plants. Journal of the Power Division, *Proceedings of the American Society of Civil Engineers.* 85, Paper 2290 PO6, Part 1: 105-119. December.

Bergstrom, Richard N. and Kenneth W. Hamming
1965 Latest Developments in Transporting Coal by Rail. Journal of the Power Division. *Proceedings of the American Society of Civil Engineers.* 91, Paper PO1: 81-91. May.

Berry, C. Harold and Moreton, F. E.
1922 The Heat Balance of the Connors Creek Plant of the Detroit Edison Company. *Transactions of the American Society of Mechanical Engineers.* 43, paper 1803d: 500-515.

Bowers, Brian
1982 *A History of Electric Light and Power.* London: Peter Peregrinus LTD.

Britt, R. D., Millard, R. E. and DiFillippo, M. N.
1975 Operational Experience with PWR Secondary Water Chemistry–A Panel Presentation. San Onofre Unit 1. *Proceedings of the American Power Conference 37*: 753-760. Chicago: Illinois Institute of Technology, Technology Center.

Brower, R. F.
1958 Progress in Nuclear Power Development. Indian Point. *Proceedings of the American Power Conference* 20: 72-74. Chicago: Illinois Institute of Technology, Technology Center.

Church, Edwin, F.
1935 *Steam Turbines.* Second Edition. New York: McGraw-Hill Book Company.

Clauss John M., Dennis L. Harrison, and Terry A. Pickens
1993 The Monticello License Renewal Project. *Proceedings of the American Power Conference* 55-1: 234-238. Chicago: Illinois Institute of Technology, Technology Center.

Cochran, J. Denny and J. E. Owens
1964 Initial Operation of the Sodium Graphite Reactor at the Hallam Nuclear Power Facility. *Proceedings of the American Power Conference* 26: 201-209. Chicago: Illinois Institute of Technology, Technology Center.

Coe, Roger
1958 Progress in Nuclear Power Development. Yankee Plant. *Proceedings of the American Power Conference* 20: 74-77. Chicago: Illinois Institute of Technology, Technology Center.

Connecticut Light and Power Company
2005 The Connecticut Light and Power Company. Available from World Wide Web: http://www.cl-p.com/companyinfo/clphistory.asp

Connecticut Yankee Atomic Power Company (CYAPCO)
1966-1974 Facility Description and Safety Analysis (FDSA) Vol. 1. Topical Report No. NYO-3250-5. Neck Plant, Haddam, Connecticut.

1975 Average Gross Generation-Megawatts Electric (Graph)

1984 Haddam Neck Plant. Reactor Cavity Seal Ring Failure. Docket No. 50-213/B11311. Letter to U.S. Nuclear Regulatory Commission and six attachments. September 12, 1984.

1985 Haddam Neck Plant/Response to Order Modifying License and Notice of Violation and Proposed Imposition of Civil Penalty (Inspection Report No. 50-213/84-23). January 28, 1985. A04547. Letter to U.S. Nuclear Regulatory Commission and three attachments.

1988 Connecticut Yankee History. Collection of typescript press releases, chronologies and draft histories. X-286 10-87 [revised 1989]

1989 Haddam Neck Plant/New Switchgear Building Construction. Letter report to U.S. Nuclear Regulatory Commission, December 15, 1989.

1987-1993 Connecticut Yankee Plant Information Book. 15 vols.

1975-1992 Index. Connecticut Yankee Technical Specification Amendments: Nos. 1-149.

1996 Haddam Neck Plant 10CFR50.59 Annual Report [for 1995 calendar year]. Submitted to U.S. Nuclear Regulatory Commission February 27, 1996.

1998 Decommissioning Updated Final Safety Analysis Report (UFSAR). 3 vols.

2005 CY Decommissioning. Available from World Wide Web: http//www.connyankee.com/html/decommissioning.html: 1-3. March 8.

Coughlin, Steven A., Stark, Louis E., Brown, Elmo L. and Johnson, Wayne R.
1974 Joining Techniques for Remote Maintenance of Nuclear Steam Generators. *Proceedings of the American Power Conference* 36: 226-233. Chicago: Illinois Institute of Technology, Technology Center.

Danko, Joseph C. and Stahlkopf, Karl E.
1984 A Review of the Boiling Water Reactor Owners Group Research Program on Pipe Cracking. *Proceedings of the American Power Conference* 46: 654-658. Chicago: Illinois Institute of Technology, Technology Center.

Davis, A. Bert, Jacobson, J. M. and Colburn, T. G.
1993 Steam Generator Replacement. *Proceedings of the American Power Conference* 55-1: 605-609. Chicago: Illinois Institute of Technology, Technology Center.

Dawes, Chester L. A
1928 *A Course in Electrical Engineering. Alternating Currents.* 3rd edition. 2. Vols. New York: McGraw-Hill Book Company.

Dillard, J.K. and Baldwin, C.J.
1964 Economic Development of Mine-Mouth Power Plants, EHV Transmission, and Nuclear Generation. *Westinghouse Engineer.* 24, 6: 167-173. November.

Dominion
2005 Have You Seen D Today? *Millstone Power Station*. Available from World Wide Web: http://www.dom.com/about/stations/nuclear/millstone/index.jsp.

Downs, Dallas I., Deegan, George E. and Boggus, Robert F.
1959 Engineering Aspects of the Santa Susana Nuclear Power Station. *Journal of the Power Division, Proceeding of the American Society of Civil Engineers.* 85, Paper 2006 PO2: 1-25. April.

Duncan, Francis, and Jack M. Moll
1981 Shippingport: The Nation's First Atomic Power Station. History Division, U.S. Department of Energy. Historic American Engineering Record (HAER) No. PA-81: Shippingport Atomic Power Station. Photographs, Written Historical and Descriptive Data. Mid-Atlantic Region, National Park Service, Department of the Interior.

Dunn, Stephen F.
1964 The New Reliables. *Proceedings of the American Power Conference* 26: 13-20. Chicago: Illinois Institute of Technology, Technology Center.

Dvorin, R. S. And Schlesinger, H. A.
1984 Water Quality Audits Can Improve Availability and Reduce Costs. *Proceedings of the American Power Conference* 46: 1012. Chicago: Illinois Institute of Technology, Technology Center.

Edmund, M. C.
1963 Economics of Recycled Fuel. *Proceedings of the American Power Conference* 25: 128-133. Chicago: Illinois Institute of Technology, Technology Center.

European Construction Institute
2005 *ICI Members*, Company Profiles: Stone & Webster. Available from World Wide Web: http://www.eci-online.org/visitors/memdir/mem-ston.html: 1-3.

Engineering
1926 50,000 KW Parsons Turbo-Alternator For Chicago. March 5.

1947 German Use of Hydrogen Peroxide for Submarine Propulsion. 163, 4239: 307- 308. April 18.

1955a Nuclear Power In Canada. 180, 4672: 198-199. August 12.

1955b Nuclear Power In America. 180, 4673: 232. August 19.

1956a Calder Hall Comes to Life. 182, 4726: 418-421. October 5.

1956b Western Approaches. Atomic Review. 182, 4736: 761-762. December 14

1957 Complementary Angle. Atomic Review. 183, 4753: 474-476. April 12.

1958 Shippingport Reactor. Atomic Review. 185, 4812: 682-684. May 30.

1958 Dresden Boiler. Atomic Review. 185, 4815: 775-776. June 20.

1959 Fast Flux. Atomic Review. 187, 4853: 334-336.

1967 Nuclear Advantage in US Not Decisive. 204, 5283: 92. July 21.

Astride, Herbert
1964 Symposium on the Keystone Station. *Proceedings of the American Power Conference* 26: 381-398. Chicago: Illinois Institute of Technology, Technology Center.

Ford, Daniel
1982 *The Cult of the Atom*. New York: Simon and Schuster.

Ford, G.W.
1955 Power Reactor Projects Throughout The World. *Engineering* 180, 4680: 490-500. October 7.

1956 Two Power Reactors Compared. *Engineering* 181, 4735: 727-730. December 7.

Fortner, Brian
2001 Nuclear Shutdown. *Civil Engineering Magazine*. April. Available from World Wide Web: http://www.pubs.asce.org/ceonline/ceonline01/0104feat.html.

Friend, Walter F.

1957 Inter-American Progress Toward Commercial Nuclear Power. *Proceedings of the American Power Conference* 19: 613-639. Chicago: Illinois Institute of Technology, Technology Center.

Gaines, A. L.

1965 Nuclear Reactor Vessels-1965 Status. *Proceedings of the American Power Conference. Proceedings of the American Power Conference.* 27: 292-301. Chicago: Illinois Institute of Technology, Technology Center.

General Electric Review

1954 General Electric Predicts...Commercial Atomic Power in 5 to 10 Years. 57, 5. July.

1955a Sixteen Years of Nuclear Technology. 58, 15-18. November.

1955b Dual-Cycle Reactor Offers, Good Efficiency, Inherent Safety. 58: 21.

1958 Nucleonics. 61: 30-36. January.

Gilbert, W. D.

1975 Design Capability of Nuclear Power Plants to Withstand Combinations of Natural and Man-Made Hazzards. *Proceedings of the American Power Conference* 37: 157-163. Chicago: Illinois Institute of Technology, Technology Center.

Golan, Simcha

1965 Optimum Nuclear Fuel Resource Utilization. *Proceedings of the American Power Conference.* 27: 196-209. Chicago: Illinois Institute of Technology, Technology Center.

Graham, Richard H.

1965 Nuclear Fuelsl-Cost Trends Under Private Ownership. *Proceedings of the American Power Conference.* 27: 210-220. Chicago: Illinois Institute of Technology, Technology Center.

Gray, Alexander

1917 *Principals and Practice of Electrical Engineering.* New York: McGraw-Hill Book Company.

Hadjian, Asadour H.

1971 Earthquake Forces on Equipment in Nuclear Power Plants. *Journal of the Power Division, Proceedings of the American Society of Civil Engineers.* 97, Paper 8240 PO3: 649-727. July.

Hadjian, Asadour H.
1973 Earthquake Forces on Equipment in Nuclear Power Plants, Discussion. *Journal of the Power Division. Proceedings of the American Society of Civil Engineers* 99, PO1: 251. May.

Haga, P. Blair and John D. Stevenson
1970 PWR Containment Structures Design Experience. *Journal of the Power Division, Proceedings of the American Society of Civil Engineers* 96, Paper 7037 PO1: 145-155. January.

Haueter, Robert L.
1974 Nuclear Power Plant Reliability. *Proceedings of the American Power Conference* 36: 200-209. Chicago: Illinois Institute of Technology, Technology Center.

Hewlett, Richard G. and Francis Duncan
1974 *Nuclear Navy 1946-1962.* Chicago: University of Chicago Press.

Hillberry, Norman
1957 The Future of Nuclear Power. *Proceedings of the American Power Conference* 19:26-33. Chicago: Illinois Institute of Technology, Technology Center.

Hoffman, J. W.
1958 Progress in Nuclear Power Development. Pathfinder Project. *Proceedings of the American Power Conference* 20:77-80. Chicago: Illinois Institute of Technology, Technology Center.

Hogerton, John F.
1968 The Arrival of Nuclear Power. *Scientific American.* 218, 2: 21-31. February.

Hoovers
2005 *Stone & Webster.* Available from World Wide Web: http://hoovers.com/free/co/factsheet.xhtml?ID=11403: 1-3.

Hopkinson, Jenny
1982 Steam Purity in PWRs. *EPRI Journal.* 7, 6: 22-28. October.

Hopping, E. L.
1922 Auxiliary System and Heat Balance at Delaware Station of the Philadelphia Electric Company. *Transactions of the American Society of Mechanical Engineers.* 43, paper 1803a: 475-486.

Howe, J. P.
1976 Nuclear Fission Energy, in S.S. Penner, ed., *Energy Vol III: Nuclear Energy and Energy Politics*, pp. 1-125. Reading: Addison- Wesley Publishing Company, Inc.

Huffman, K. L. and Malinowski, D. D.
1981 PWR Steam Generator Operating Experience and Its Influence on Steam Generator Design and Manufacture. *Proceedings of the American Power Conference* 43: 768-772. Chicago: Illinois Institute of Technology, Technology Center.

Hughes, Thomas P.
1989 *American Genesis.* New York; Viking Pengquin.

Huntley, H. W.
1955 Prospects for the Use of Nuclear Power. *Proceedings, American Society of Civil Engineers,* City Planning Division. 81, Paper 767. 3-7. August.

Institute of Nuclear Power Operations (INPO).
2005 United States Department of Energy. 1 of 2. Available from World Wide Web: http://www.eh.doe.gov/nsps/inpo/

Janes Fighting Ships
1967-1968 Blackman, Raymond V. B., ed. London: Sampson Low, Marston & Co., Ltd.

Jensen, Harry M.
1964 Unit Train Coal Delivery. *Proceedings of the American Power Conference* 26: 371-380. Chicago: Illinois Institute of Technology, Technology Center.

Jackson, Melvin W.
1971 Discussion in Hadjian, Asadour. Earthquake Research Needs for Nuclear Power Plants. *Journal of the Power Division, Proceedings of the American Society of Civil Engineers* 97, Paper 8240 PO 3: 649-727. July.

Johnson, J. F.
1919 The Large Steam Turbine. *Transactions of the American Society of Mechanical Engineers* 40, paper 1681: 1097-1118.

Johnson, L.E. and McDonald, B. N.
1971 Generating Superheated Steam In Nuclear Boilers. *Power.* 115, 5: 93-94. May.

Johnson, W. E., Fax, D. H. and Townsend, S. C.
1957 The PAR Homogeneous Reactor Project-Plant Design and Operating Problems. *Proceedings of the American Power Conference* 19: 640-650. Chicago: Illinois Institute of Technology, Technology Center.

Kilpatrick, James N.
1960 Yankee Atomic Electric. *Nuclear Engineering.* 5, 53: 466-468. October.

Kinsman, Nicole
2001 Moly Does the Job. IMOA Newsletter. July. Available from World Wide Web: http://www.imoa.info/newsletter July01.pdf: 1-3.

Klehm, W. L.
1963 Civil Engineering for CVNPA Nuclear Power Plant. Journal of Professional Practice. *Proceedings of the American Society of Civil Engineers.* 89, Paper 3645 PO1: 45-65. September.

Koch, L. J.
1963 World Status and Future of Fast Reactors. *Proceedings of the American Power Conference* 25: 178-188. Chicago: Illinois Institute of Technology, Technology Center.

Leverett, M. C.
1955 Highlights of Reactor Technology 1942-1955. *General Electric Review.* 58: 23-26. November.

Levenson, Milton
1981 The State-of-the-Art. *Proceedings of the American Power Conference* 43: 40-41. Chicago: Illinois Institute of Technology, Technology Center.

Lewis, Harold W.
1980 The Safety of Fission Reactors. *Scientific American* 242, 3: 53-65. March

Lewis, L. T. and Tsui, Y. W.
1960 Structural Engineering Aspects of Nuclear Power Plants. *Journal of the Structural Division, Proceedings of the American Society of Civil Engineers* 86, Paper 2059 ST6: 23-46. June

Lischer, L. F.
1965 Opening Remarks. *Proceedings of the American Power Conference. 27: 1-3.* Chicago: Illinois Institute of Technology, Technology Center.

Lloyd, Edward R.
1958 Insurance Aspects of Nuclear Energy. *Journal of the Power Division, Proceedings of the American Society of Civil Engineers* 84, Paper 1599 PO2: 1-9. April.

Lorenzi, Otto de, ed.
1952 *Combustion Engineering.* New York: Combustion Engineering-Superheater, Inc.

Love, Joseph E, Chester S. Darrow, and Burr H. Randolph
1958 Civil Engineering Aspects of the Dresden Nuclear Station. *Journal of the Power Division, Proceedings of the American Society of Civil Engineers* 84, Paper 1600 PO2: 1-21. April.

Lunn, D.C.F. and Harvey, J.
1970 Corrosion at Trawsfynydd. *Engineering* 209, 5416: 189-91. February 20.

Macintyre, Donald and Bathe, Basil
1974 *Man-of-War.* New York: Castle Books.

MacNaughton, Edgar.
1967 *Elementary Steam Power Engineering.* 3rd Edition. New York: John Wiley & Sons, Inc.

Main, T.
1893 *The Progress of Marine Engineering.* New York: The Trade Publishing Co.

Martel, Louis J. and Stahlkopf, Karl E.
1981 Utility Incentives and Plans for Improving Next-Generation Light Water Reactors. *Proceedings of the American Power Conference* 43: 737-742. Chicago: Illinois Institute of Technology, Technology Center.

McIntyre, Hugh C.
1975 Natural-Uranium Heavy-Water Reactors. *Scientific American* 233, 4: 17-27. October.

Morabito, J. J.
1964 Sodium Components in Nuclear Reactor Systems. *Proceedings of the American Power Conference* 26: 257-265. Chicago: Illinois Institute of Technology, Technology Center.

Morgan, D. W. R.
1950 Central-Station Steam-Power Generation. *Westinghouse Engineer* 10, 1: 7-17. January.

Mundis, John A. and Noble, Daniel M.
1981 Steam Generator Technology Update. *Proceedings of the American Power Conference* 43: 748-755. Chicago: Illinois Institute of Technology, Technology Center.

Necci, Raymond P.
1993 Steam Generator Replacement Project. *Proceedings of the American Power Conference* 55-1: 590-593.Chicago: Illinois Institute of Technology, Technology Center.

Nero, Anthony and Jack Dennis
1984 *Nuclear Power.* The Nuclear Almanac. Dennis, Jack, ed. Reading: Addison-Wesley Publishing Company.

Nero, Anthony V
1979 *A Guidebook to Nuclear Reactors.* A General Introduction to Nuclear Reactors. Berkeley: University of California Press.

1959 Design of the Shippingport Reactor Container. *Journal of the Structural Division, Proceedings of the American Society of Civil Engineers* 85, Paper 2258 ST9: 53-64. November.

Noll, Douglas E.
1964 Factors That Determine Treatment for High-Pressure Boilers. *Proceedings of the American Power Conference* 26: 753-761. Chicago: Illinois Institute of Technology, Technology Center.

Northeast Utilities
n.d. Haddam Neck Plant Reactor Cavity Seal Failure: Actions at the Millstone Site. Docket No. 50-213. Report prepared as Section 4 of response to US Nuclear Regulatory Commission 1984a.

2005 Northeast Utilities Formation. Available from World Wide Web: http://www.nu.com/aboutnu/formation.

Nuclear Engineering

1956 American Reactor Program. 1, 3: 116-118. June.

1957 Experimental Boiling Water Reactor. 2, 12:121-122. March.

1958a Shippingport. 3, 27: 233-242. June.

1958b A Survey of Water Reactors. Report from Geneva-5. 3, 31: 436-437. October.

1960 The American Scene. 5, 53: 425-426,

Nuclear Management Company (NMC)
2002 Nuclear Technology Milestones 1942-2000. 1-7. Available from World Wide Web: http://www.nmcco.com/education/facts/history/milestones.htm.

Office of the Federal Register
1962 Code of Federal Regulations. Titles 10-11. Revised as of January 1, 1963. Title 10-Atomic Energy. Chapter 1-Atomic Energy Commission. Part 100-Reactor Site Criteria: 3509-3511. April 12, 1962. National Archives and Records Service.

2005 Code of Federal Regulations. 10 CFR Ch. 1. Part 100.23. Geologic and Seismic Siting

Criteria. 1-1-04 Edition: 552-561. National Archives and Records Administration. Available from World Wide Web: http://www.gpoaccess.gov/cfr/index.html.

Osborn, Kenneth R. and Larson, Howard J.
1971 Spent Fuel Reprocessing. *Proceedings of the American Power Conference 33:247-252.* Chicago: Illinois Institute of Technology, Technology Center.

Oxford English Dictionary
1989 2nd. Edition. Oxford: Clarendon Press.

Palo, George P., Walter F. Emmons, and R.M. Gardner
1963 Symposium on TVA's Bull Run Steam Plant. *Proceedings of the American Power Conference* 25: 331-385. Chicago: Illinois Institute of Technology, Technology Center.

Parsons, R. H.
1939 *The Early Days of the Power Station Industry.* Cambridge: Babcock and Wilcox, Ltd.

Phung, Doan L.
1984 Economics of Nuclear Power, Past Record, Present Trends, and Future Prospects. *Proceedings of the American Power Conference 46: 887-892.* Chicago: Illinois Institute of Technology, Technology Center.

Power
1971 Successful Operation Makes Boilers Generating Saturated Steam the Choice in Many Plants. 115, 5: 94. May.

1982a Fuels and Fuel Handling: Nuclear Fuel Cycle, in *Century of Power Progress. 100th Anniversary Issue. 1882-1982.* Vol. 126, 4: 148-159. April.

1982b Cycle Systems, Circulating Water, in *Century of Power Progress. 100th Anniversary Issue. 1882-1982.* Vol. 126, 4: 224-230. April.

Proceedings of the American Power Conference
1971 Evening Forum. Panel Discussion-the Fast Breeder Program-The Actions Required By The Power Industry. 33: 48-66. Chicago: Illinois Institute of Technology, Technology Center.

Randazza, John
1975 Steam Generator Operating Experience At the Yankee Plant. *Proceedings of the American Power Conference* 37: 760-764. Chicago: Illinois Institute of Technology, Technology Center.

Rankine, William J. M.
1859 *A Manual of the Steam Engine and Other Prime Movers.* London and Glasgow: Richard Griffin and Company.

Ratical.org
2005 Commercial Nuclear Power Reactors in the United States. Appendix VII. 1-6. Available from World Wide Web: http://www.ratical.org/radiation/CNR/PP/app7.html

Reichle, Leonard F. C.
1958 Nuclear Power-Where Are We Headed in Costs? *Proceedings of the American Power Conference. 20: 132-141.* Chicago: Illinois Institute of Technology, Technology Center.

Rice, R. E. and W. R. Wallin
1964 Borax V-A Nuclear Superheat Reactor. *Proceedings of the American Power Conference* 26: 191-200. Chicago: Illinois Institute of Technology, Technology Center.

Richardson, Alexander
1911 *The Evolution of the Parsons Steam Turbine.* London: Offices of Engineering.

Rickert, R. J.
1964 Fuel-Cycle Economics of Large D20-Moderated Organic-Cooled Power Reactors. *Proceedings of the American Power Conference* 26: 67-272. Chicago: Illinois Institute of Technology, Technology Center.

Riedel, Walter
1964 The Control of Metal Pickup in Cycles With Steel Tube Feedwater Heaters. *Proceedings of the American Power* Conference 26:790-797. Chicago: Illinois Institute of Technology, Technology Center.

Risser, Hubert E.
1964 Utility Fuel Trends and Future Supply. *Proceedings of the American Power Conference* 26: 582-588. Chicago: Illinois Institute of Technology, Technology Center.

Ritchey, Neil
1964 Transporting Spent Fuel. *Proceedings of the American Power* Conference 26: 297-302. Chicago: Illinois Institute of Technology, Technology Center.

Roddis, Louis H., Jr.
1964 Private Ownership of Nuclear Fuels. American Power Conference Luncheon. *Proceedings of the American Power Conference* 26: 21-28. Chicago: Illinois Institute of Technology, Technology Center.

1965 The World Nuclear Power Situation in Light of the 1964 Geneva Conference. *Proceedings of the American Power Conference.* 27: 286. Chicago: Illinois Institute of Technology, Technology Center.

Schuecktanz, G., R. Riess, and L. Stieding
1981 German Nuclear Experience. *Proceedings of the American Power Conference* 43: 756-761. Chicago: Illinois Institute of Technology, Technology Center.

Scientific American
1976 Science and the Citizen. Slowdown. Vol. 234. No. 3: 60A-60B. March.

Scott, N. L., M. Mantey, M and R.F.
1960 Additional Aspects of the Enrico Fermi Atomic Power Plant. *Journal of the Power Division, Proceedings of the American Society of Civil Engineers* 86: Paper PO1: 39-55. February.

Seaborg, Glenn T.
1965 Nuclear Power-New Member of the Energy Team. American Power Conference Luncheon. *Proceedings of the American Power Conference. 27: 21-29.* Chicago: Illinois Institute of Technology, Technology Center.

Seymore, Francis W.
1992 Decommissioning Cost Study for the Yankee Nuclear Power Station. TLG Engineering. 10-87.

Setlur, A. V.
1975 Seismic Design Considerations for a Standard Nuclear Plant. *Proceedings of The American Power Conference* 37: 137-145. Chicago: Illinois Institute of Technology, Technology Center.

Simpson, J. W.
1961 The Technical and Economic Status of Central Station Closed-Cycle Water Reactors. *Proceedings of The American Power Conference* 23: 81-86. Chicago: Illinois Institute of Technology, Technology Center.

Singley, W. J., I. H. Welinsky, S. F. Whirl, and H. A. Klein
1959 Stress Corrosion of Stainless Steel and Boiler Water Treatment at Shippingport Atomic Power Station. *Proceedings of The American Power Conference* 21: 748-768. Chicago: Illinois Institute of Technology, Technology Center.

Sinton, Walter
1966 Steam Turbines for Nuclear Power Plants. *Westinghouse Engineer* 26, 4. July.

Slade, Gerald B.
1993 Impact of Competition on Nuclear Production. *Proceedings of the American Power Conference* 55-1: 210-218. Chicago: Illinois Institute of Technology, Technology Center.

Smith, Harley A.
1969 Yankee. Fuel Handling. *Nuclear Engineering*. 5, 53:476. October.

Snouffer, Richard D. and Victor D. Hanson
1964 The Unit Train Concept of Coal Transportation. *Proceedings of the American Power Conference* 26: 589-597. Chicago: Illinois Institute of Technology, Technology Center.

Snyder, F. L.
1950 The Transformer and How it Grew. *Westinghouse Engineer* 10,1: 50-61. January.

Sopocy, Dale M, and John C. Kovach
1984 Chemistry Improvement Programs for PWRS. *Proceedings of the American Power Conference* 46: 1010-1021. Chicago: Illinois Institute of Technology, Technology Center.

Southern California Edison
2002 *San Onofre Nuclear Generating Station.* Unit 1 Decommissioning. Available from World Wide Web: http://www.sce.com/sc3/006_about_sce/006b_generation/006b1_songs/006b1d_songs_unit 1.htm

Spencer, R. C. and E. H. Miller
1973 Performance of Large Nuclear Turbines. *Combustion* 45, 2: 24-30. August.

Stern, Theodore.
1964 Reactor Siting and Engineered Safeguards for Nuclear Plants. *Proceedings of the American Power Conference* 26: 244-256. Chicago: Illinois Institute of Technology, Technology Center.

Stolz, John F.
1958 General Considerations For Reactors and Related Plant Types. *Journal of the Structural Division, Proceedings of The American Society of Civil Engineers.* 84, Paper 1763 ST5, Part 1: 1763-1-15. September.

Suburbanchicagonews.com
2005 Nuclear Timeline: Overview of the Nuclear Power Industry and Zion Station from 1980s through Present. News Sun. Aug. 20, 1997. 1 of 4.

Swann, Peter R.
1966 Stress-Corrosion Failure. *Scientific American* 214, 2: 72-81. February.

Union of Concerned Scientists

2002 Davis-Besse: The Reactor With a Hole in its Head. 1-7. Available from World Wide Web: http://www.ucsusa.org/clean_energy/nuclear_safety/page.cfm?pageID=790.

U.S. Atomic Energy Commission

1969 Docket No. 50-213. Notice of Proposed Issuance of Amendment to Operating License. 1-16. February 3.

1974 Docket No. 50-213. Cover letter to Facility Operating License No. DPR-61: 1-2. December 27.

U.S. Department of Energy

2005 *Hanford Site*. Hanford Site Virtual Tours. 100 N Area: N Reactor: 1 of 2. Available from World Wide Web: http://www.hanford.gov/information/sitetours/?tour=100N

U.S. Nuclear Regulatory Commission

1976 Docket No. 50-213. Cover letter to Amendment No. 7. June 8.

1984a IE Bulletin No. 84-03: Refueling Cavity Water Seal. August 24, 1984

1984b Haddam Neck Special Inspection 50-213/84-23. September 14, 1984.

1984c Summary of September 14, 1984 Meeting [between NRC and CYAPCO representatives]. Docket No. 50-213/LS05-84-09-037.

1984d NRC Region 1 Enforcement Conference October 1, 1984. Special Inspection 50-213/84-23. Industry Use of Pneumatic Cavity Seals. Attachment 2: 3.

1984e Evaluation by the Office of Nuclear Reactor Regulation Relating to Proposed Corrective Actions for the Cavity Seal Failure/Connecticut Yankee Atomic Power Company/Haddam Neck Plant/Docket No. 50-213.

1986 Docket No. 530-213. Cover letter to amendment No. 93. Subject: Steam Generator Plugging. Cover letter. May 14.

1987 Docket No. 50-213. Cover letter to amendment 96. Subject: Technical Specifications for Steam Generator Tube Plugging in The Tube Sheet Region. September 25.

1988a NRC Integrated Program for the Resolution of the Unresolved Safety Issues A-3, A-4, and A-5 Regarding Steam Generator Tube Integrity. Final Report: I-4-28, A-1-A10, B-1-B-3. NUREG-0844. September.

1988b Docket No. 50-213. Cover letter to SALP report: 1-2. October 4.

1995 Docket No. 50-629. Safety Evaluation Report by the U.S. Nuclear Regulatory Commission Related to the Request to Authorize Facility Decommissioning. Yankee Nuclear Power Company. Yankee Atomic Electric Company.

van Noordennen, Gerald (CYAPCO Regulatory Affairs Manager)
2005 Personal electronic communications.

Van Nort, Peter S. and Copeland, Raymond L.
1975 Overview of the Clinch River Breeder Project. *Proceedings of The American Power Conference* 37:213-220. Chicago: Illinois Institute of Technology, Technology Center.

Vennard, Edwin
1957 Opportunities in the Electric Light and Power Business. *Proceedings of The American Power Conference* 19: 89-100. Chicago: Illinois Institute of Technology, Technology Center.

1965 New Horizons in Energy. *Proceedings of The American Power Conference.* 27: 10-13. Chicago: Illinois Institute of Technology, Technology Center.

Wald, Matthew L.
1999 Reactors: Healthy but Dying. *New York Times.* March 7. Section 4. The Nation: 16

2000 Nuclear Power's Second Act. *New York Times.* December 20. C1.

2002 US, Alarmed by Corrosion, Orders Checking of Reactors. *New York Times.* March 25. A1.

2005a Nuclear Power Hopes to Find a Welcome Mat Again. *New York Times:* A16. January 27.

2005b Power Producers Window-Shop for the Latest Models of Reactors. *New York Times.* March 15. F3.

Weidenbaum, Bernhardt, Sherman Naymark, and Thomas J. Pashos
1964 Uranium Dioxide Fuels. *Proceedings of the American Power Conference* 26: 273-282. Chicago: Illinois Institute of Technology, Technology Center.

Weinberg, Alvin M.
1994 *The First Nuclear Era.* New York: American Institute of Physics.

Whitaker, Ralph
1981 Corrosion. *EPRI Journal.* 6, 7: 6-13. September.

Witzke, R. L. and Voysey, Alfred E.
1958 *Westinghouse Engineer.* 18, 4: 102-106. July.

Yankee Atomic Energy Company
2005 Ownership. Available from World Wide Web:
http://www.yankee.com/info_ownership.html

Zinn, Walter H.
1957 Similarities in the Technical Features of Nuclear Power Development in the United States and Other Countries. *Proceedings of the American Power Conference* 19: 19-25. Chicago: Illinois Institute of Technology, Technology Center.

APPENDIX A

Engineered Safety Systems

The substitution of engineered safeguards in place of remote siting was common by the time the first large commercial phase nuclear plants were constructed. While distance was an effective mediator against direct radiation streaming from a plant, it was bound to be less effective against wind driven gaseous clouds of radioactive material. The concrete in the containment shell was able to reduce radiation intensity by over a factor of five.[1] In combination with a gas-tight pressure-resisting steel vapor shell, containment design provided a passive final barrier to release of radioactive materials from the primary system during normal operations or in an accident.[2] That system alone would not have allowed construction at the Haddam site based on the model exclusion distance calculations in the Atomic Energy Commissions (AEC) 1962 report Calculation of Distance Factors for Power and Test Reactor Sites, TID-14844 which followed the main document Reactor Site Criteria (10 CFR Part 100) of April 1962.[3] Reactors designed to operate at over 300 mw could reduce their exclusion distance if any releases were directed up a vent stack, since leakage was considered more dangerous at ground level. The Connecticut Yankee concrete containment building with its attached steel vapor sphere and 175-foot-high vent stack partially fulfilled the "consequence limiting" requirements. Complete compliance with AEC standards required the addition of active engineered safeguards which were specified in further AEC documents of 1963 and 1964.[a]

The systems installed at the plant to insure compliance included an Emergency Core Cooling System (ECCS), a Containment Spray System, In addition, many plant operating systems including the Air Recirculation and Filtration System and the Residual Heat Removal System used in normal operation could be marshaled to aid the primary safety systems in the event of an accident.

Postulated Accidents

To design against accidents, engineers of nuclear plants had to postulate worst-case scenarios and then minimize if not eliminate the hazard to the public from a release of radiation. In a pressurized water reactor, the worst type considered was a loss of coolant accident (LOCA). This could be a double-ended rupture or complete sheering of one of the legs of the primary reactor cooling system, known as a design basis accident or a DBA.[4] While fission would end immediately due to the presumably reliable activation of control rods, heat from decaying fission products could still produce 10% operating power with the rate dropping off to 5% in under a minute. In that short period of time, however, massive damage to the core with subsequent fission release could occur. If the normal flow of cooling water slowed down, the fuel rods would overheat and proceed to boil off the remaining water. The steam produced by the overheating rods, and from cooling water flashing due to the break in the pressure boundary, could raise containment pressure and temperature past design limits.[5] If emergency cooling water did not reach the core soon enough, melting fuel rods could reform into a critical

---

[a]Atomic Energy Commission, "Connecticut Yankee Atomic Power Company Nuclear Plant-Unit Number One Preliminary Hazards Summary Report", Docket No. 50-213. Washington, D.C. The Commission, September 1963, and AEC, "Hazards Analysis by the Research and Power Reactor Safety Branch Division of Licensing and Regulation for Connecticut Yankee Atomic Power Company Nuclear Power Plant -Unit No. 1", Docket No. 50-213, Washington, D. C: The Commission, March 1964. (Stern 1964: 255).

mass triggering a nuclear catastrophe.[6] Tests on rods heated by electricity showed that cooling water had to be reintroduced into the core area within 20 seconds.[7] The ECCS could not mitigate a catastrophic rupture of the reactor vessel, which was considered incredible, as in that case the cooling water would disperse. The ECCS and most other engineered safety systems and summarized below.

Emergency Core Cooling System

The Emergency Core Cooling System (ECCS) was originally (1974) comprised of three separate systems: the High-Pressure Safety Injection System (HPSI), the Low-Pressure Safety Injection (Core Deluge) System (LPSI), and the Charging System.[8] Later, the Residual Heat Removal System described below was classified as part of the ECCS.[9] The systems automatically delivered water to the reactor vessel to cool the core after a loss of coolant accident. In post accident recovery operations the system provided long term core cooling. In the event of a steam line break, the ECCS provided reactivity control. The system was derived from the safety injection systems at Shippingport and Yankee Rowe.[10]

Five minimum requirements for ECCS performance were set by the AEC in 10CFR50.46, although some sections of the system could not meet the passive requirements because the plant was designed and built previous to these rulings:[11]

- Peak Cladding temperature of Zircaloy fuel rods was limited to 2,200° F to limit chemical reactions between the water or steam. The stainless cladding at Connecticut Yankee was allowed a higher limit.

- Maximum Cladding Oxidation was limited to less than 2% of the clad thickness to insure integrity.

- Maximum hydrogen generated from a reaction between the cladding and steam or water was limited to prevent an explosive concentration in containment after an accident.

- Long term cooling had to have sufficient capability to provide decay heat removal from the reactor.

- Extensive redundancy of components, interconnections, leak detection, and power supplies was required, and described as a single failure criterion: every active component (all valves and pumps) had to be duplicated.[12] To achieve that level of protection, the system was configured as two separate trains of equal capacity each consisting of a charging pump, a high pressure injection pump, a low pressure safety injection pump and a residual heat removal pump. The system could be operated locally if the control room became uninhabitable.[13] All the pumps were located in the Primary Auxiliary Building ([PAB], see HAER No. CT-185-G) The system had to be operable whenever the reactor coolant temperature was greater than 350° F.

The automatic feature of the ECCS required the system to be tied in with sensors to components that would be reliably and measurably impacted by the postulated accident conditions. Low pressurizer pressure and high containment pressure were the initiation parameters.[14] Dual safety injection relays with manual or solenoid tripping started the water flow when the pressurizer signaled a drop below 1700 psig resulting from coolant leaving the system. Electrical control functions were powered from 125-volt DC buses with battery backup and provision for powering from AC buses via AC-to-DC-changing motor generators. Immediately after accident initiation, the charging pumps, high- and low-pressure safety injection pumps pulled borated water from the Refueling Water Storage Tank and injected it into the Reactor Coolant System (the injection phase). Water leaving the system was collected in the containment sump and recirculated by the Residual Heat Removal

Pumps and heat exchangers back into the RCS (recirculation phase).[15] Safety injection accumulator tanks which had been used in other PWR plants were not installed at Connecticut Yankee.[16]

Safety Injection Initiation

Redundancy of the critical automatic signal came from two independent circuits with tripping relays on the control board (see HAER No. CT-185-F) Multiple pressurizer channels (sensors) de-energized relays on a pressure drop to 1,700 psig, energizing another relay[b] which in turn energized a solenoid which tripped another relay which then actuated numerous contacts in the system.[17] Automatic actions which followed included: Reactor Trip, Diesel Generator Start, Fans start, trip pressurizer heaters (to insure readiness for cool down), open Safety Injection Loop stop valves and core deluge valves, and signaled "Core Cooling Actuated" alarms and annunciators. Numerous relays, blocking circuits, and switchboard interlocks prevented operators from taking the wrong actions during core cooling and depressurization.[18] The second major initiator of safety injection was an indication of high containment pressure which would directly result from the loss of heated pressurized inventory in the RCS. At 5 psig overpressure (or a safety injection signal) circuits automatically closed many paths through the containment boundary to limit fission product release to the atmosphere.[19] Trains of relays and solenoids initiated automatic events through the plant to control the accident.

The two High-Pressure Safety Injection Pumps provided the entire motive force for the high head safety injection to the four coolant loops.[20] They were horizontal, 6-stage[c] 1,250 hp, 1750 gpm, 1,400 psi discharge centrifugal pumps operating against a head (pressure) equal to a column of water over 2,000 feet high. The stages were arranged with a single entry stage and the 5 following stages separated by the pump motor. Power came from the 4000 vac emergency system with each pump on a different bus. Control circuit breakers were operated by 125 volt dc for redundancy. Water flow was directed to the cold legs of the RCS (between the coolant pumps and the reactor vessel entrance nozzles) to avoid flow into the steam generator u-tubes with subsequent reverse steam generation (from the hot feed water) which could actually lead to higher core temperature.[21] The four safety injection loop motor operated stop valves were normally closed and would open automatically on the safety injection signal. Injection flow was maintained until manually switched to recirculation by operators or until the RWST inventory fell below a set point.

Low-Pressure Safety Injection Pumps were single-stage 1,000 hp, 5550 gpm, 350 psi discharge, centrifugal types operating against a 590 foot head. Power was as for the HPSI pumps. They supplied through 6-inch-

---

b Relays were battery powered electro-magnetic devices used in early telegraph systems to extend transmission length. In 20th century control systems, components were activated by trains of energizing and de-energizing relays (Oxford English Dictionary 1989: 556)

c Multi-stage pumps had two or more impellers with each stage boosting the pressure in succession to achieve higher output pressure.

diameter motorized/manual valves the core deluge piping that entered the reactor head through spare control rod penetrations.[d] The cool borated water entered the core directly, aiding the high pressure injection flow coming

in through the cold legs. The LPSI pumps also supplied the Containment Spray system[22], fire fighting spray for the charcoal filters and #2 loop of the RCS for normal heat removal. Once recirculation was begun, the low-pressure pumps were shut down.

Charging System and Chemical and Volume Control System

Many plant systems had multiple functions and crossed into different buildings from their base location. The Charging System was a portion of the Chemical and Volume Control System (CVCS) which also acted as part of the ECCS by serving as an additional source of high injection.[23] The two charging pumps were horizontal, 13-stage, 2200-gpm, 150-psi discharge centrifugal pumps. Starting was automatic, with power and control from the same group of buses as the HPSI and LPSI pumps. When working in the CVCS system, the charging pumps pumped from and into the Volume Control Tank in the PAB, the reactor coolant loop #2 cold leg, and reactor coolant pump seals. On safety injection initiation they automatically started to inject cool borated water from the Refueling Cavity Water Storage Tank into loop #2. The CVCS then was utilized as part of the ECCS during the recirculation phase.[24]

Residual Heat Removal System (RHRS)

After shutdown the fission products continued to decay producing relatively large amounts of heat. The RHRS provided decay heat removal and circulation through the coolant loops when the reactor coolant pumps were not operating. The recirculation ensured that the boron injected into the RCS for shutdown was evenly concentrated through the system.[25] After being removed from the reactor at 300 F. the coolant was passed through heat exchangers for cold shutdown. The RHRS was not used during normal operations but was lined up for standby ECCS operation. In a Loss of Coolant Accident (LOCA) the system pumped spilled water from the containment sump and recirculated it through the heat exchangers and back into the reactor for long term decay heat removal.

The system had no role in the injection phase, but was manually activated to provide subsequent circulation.[26] The RHRS pumps pulled from the containment sump, collecting the spilled water and sending it through the Residual Heat Removal (RHR) heat exchangers to be cooled by the service water. It was then pumped into two cold- leg motor-operated valves. Other engineered safeguard components which could receive cooled water were the purification system, charging pump suction, Core deluge, containment spray and charcoal filter spray and High Pressure Injection pump suction.[27] In addition, during refueling, the RHRS was used to transfer water between the Refueling Cavity Water Storage Tank and the Refueling Cavity.

---

[d] The core deluge valves would fail to function (open) on loss of power. They were provided with hand wheels but it was accepted that during a serious accident, the radiological conditions would prevent access (Connecticut Yankee Atomic Power Company 1987-1995: Chapter 5, page 41.)

Like many plant systems the RHRS was duplicated with two pumps, and two heat exchangers. The pumps were horizontal single stage centrifugal 2,200-gpm, 150-psi discharge units located in the RHR pit in the Primary Auxiliary Building. Each unit could carry half of the maximum pumping load allowing reasonable flow even if one unit was out of service.[28] Power came from the 480v AC buses with 125v DC control. The heat exchangers were shell and U-tube, 500-psig pressure vessels with cooling water in the shell supplied by the

component cooling water system.[29] During accident conditions, supply was transferred to the Service Water System which was classed as a safety system with higher reliability. As a result of the cavity seal failure in 1984, the RHRS was required to be operable or operating during the entire refueling process to provide circulation.

Water Supply to ECCS

The 250,000-gallon Refueling Cavity Water Storage Tank was the reservoir of borated water for the ECCS and for filling the reactor cavity during refueling.[30] The high-pressure injection, charging, and RHR pumps pulled from a 16-inch-diameter supply line, while the low-pressure injection was fed from a separate 18-inch-diameter line. The tank could supply 100,000 gallons to the RHR for recirculation at which point it was stopped manually at around the 130,000 gallon level.[31]

Containment Sump

The Containment sump and pumps, were designed for use in a serious Loss of Coolant Accident (LOCA), since the 2,000-gallon capacity of the in-core instrumentation ICI sump would quickly be reached leaving the entire ground floor of Containment to become the sump for up to 32,000 to 35,000 gallons of water.[32] 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, the Residual Heat Removal pumps in the recirculation mode pulled water from the sump to the High-Pressure Safety Injection, Low-Pressure Safety Injection, and Charging pumps — components of the Emergency Core Cooling System (ECCS) — for addition to the coolant loops.

Auxiliary Feed water and Other ECCS Systems

As noted above under Injection Initiation, relays automatically closed off penetrations in the containment boundary to prevent releases. Sometime after 1990 the various shut-offs were designated the Containment Isolation System. Control of the heat and pressure produced by a LOCA required numerous active "heat sinks," as well as the passive heat sink effects expected from the concrete mass of the containment building, shield walls, and equipment.[33]

Possibly in response to regulatory changes, the Auxiliary Feed water System (AFW) was also included as an engineered safeguard.[e] It functioned on shutdown of the main feed water system to provided feed water flow to

---

[e] The AFW system was a causative factor in the TMI accident. Operators there had isolated the pumps for testing and failed to restore them, leading to a complete loss of all feed water flow for several critical minutes. (Connecticut Yankee Atomic Power Company 1987-1995: Chapter 21, page 60).

the steam generators so they could continue to absorb heat from the RCS.[34] In addition to their role in steam production for the steam turbine, the steam generators also served as a heat sink for reactor decay heat after shutdown and in accident conditions. To ensure a continuous un-interruptible supply of feed water if the main Feed water system was not functional, Connecticut Yankee had an Auxiliary Feed water System supplied from the Terry Turbine building named after two Terry Turbine driven auxiliary feed pumps.[35] The Terry turbine was developed by the Terry Steam Turbine Company of Hartford, CT early in the 20th century,[36] and became a favored prime mover in the power industry for driving fans and boiler feed pumps due to its ruggedness, relatively high efficiency and high speed.[37] The single forged turbine wheel had multiple semi-circular "buckets" machined directly in the forging and reversing chambers in the surrounding casing. Steam was admitted directly into the buckets moving them from the impact. Steam was then turned 180 degrees and re-admitted to the blades several times until most of the energy was gone. The unit had very large clearances between the turbine wheel buckets and the reversing chambers for reliability and could even continue to operate if the steam supply turned to water.[38]

The pumps were 450-gpm multi-stage centrifugal types which supplied their own lubrication, shaft sealing, and cooling, independent of plant systems. The Terry turbines were the only rotative steam powered auxiliaries in the plant. Even if all plant electric power and backup diesel generators were lost, the Terry turbines could continue to provide pumping power from steam produced by decay heat from the reactor.[39] Steam supply was from the #3 and #4 steam generators through the atmospheric steam dump valve supply header.[40] Special throttling control valves with greater reliability than other types were operated by the plant control air system and would automatically open if the system failed. The system automatically activated if the circuit breakers on the main pumps tripped, or low water level was sensed in two of the steam generators. In addition to the two Terry turbines, a third component of the Auxiliary Feed water System was a manually-operated electric-motor-driven pump located in a separate enclosure south of the Terry Turbine Building. The 725-gpm motor-operated pump could supply feed water when the Condensate and Feed water systems were manually shut down, and provided a back up source if the main pumps and Terry Turbine pumps were involved in a loss of feed incident.[41] On account of its important role in Reactor Coolant System heat removal, the Auxiliary Feed water System was maintained as part of the plant Engineered Safety Systems.[42]

Supply water for the Auxiliary Feed water System came from the Demineralized Storage Water that was located within a protective concrete shield wall south of the Terry Turbine Building. Protection from freezing was insured by a plant heat trace circuit system.

The Reactor Protection System had dual functions. During normal operations, the system maintained the integrity of the fuel and the RCS loops during severe load change and component failure transients resulting from loss of feed water, loss of coolant flow and other conditions which would be expected to occur during the life of the plant.[43] The RPS also worked with the Engineered Safety Systems to limit the amount of fission products released into the atmosphere during an accident.

Operating conditions requiring protection from the RPS were categorized by the estimated frequency or probability of occurrence. Condition 1 Events occurring daily or yearly included normal steady state operation with normal startup, ramped power changes, shutdown and refueling. Condition 2 Events resulted in reactor trip without damage to fuel or RCS overpressure. These events, which included loss of feed water, loss of load and improper dilution of the boric acid, were not expected to prevent the plant from returning to operation and were expected to occur only once a year. During Condition 1 and 2 events, the RPS automatically tripped the reactor to maintain integrity of the fuel and RCS. Safety limits were set by a document called the Technical Specifications which covered all aspects of the plant. Core temperature had to be limited to prevent release of fission products into the RCS loops. Pressure had to limited in the loops or fission products could be released into containment.

Condition 3 Events included small LOCAs, fuel assembly miss loading, and minor steam pipe breaks. Consequences were possible damage to only a few fuel rods, could preclude return to operation and were expected only once in the life of the plant. Condition 4 Events were the worst case scenarios of postulated accidents, including major LOCAs, steam pipe breaks, steam generator tube ruptures, seizure of reactor coolant pumps, ejected fuel rods, and major fuel handling accidents. Though designers never expected them to occur, they were analyzed to determine what if any remedial measures could be taken and what the effects of the postulated accidents would have on the plant, operators, and immediate or surrounding population centers. During Condition 3 and 4 Events, the likely release of fission products into Containment required the RPS to assist the engineered safeguards in restricting release from Containment into the atmosphere around the plant.

The ultimate criteria basis for RPS function was a level of radiation exposure in rems that a person within the exclusion area radius would be subjected to for a period of two hours after the release. That level of 25 rem over their whole body or 300 rem to the thyroid was set by the Code of Federal Regulations in Chapter 10, Part 100. A similar level was set for anyone in the outer boundary of the low population zone.

Since the ECCS, the most important component of the engineered safeguards could not remove all the heat from a reactor at full power, the Reactor Protection System was an essential element of the shutdown system.[44] The critical role was to prevent the failure of the first level of fission product protection, the fuel cladding. It was not possible to actually measure the departure from nucleate boiling (DNB) point during operation, but thermal power and reactor coolant temperatures and pressures could be used to correlate a parameter. A ratio of heat flux to DNB was set up and if it was exceeded the reactor protection system described below would trip the reactor. [45]

The RPS protection system had two separate trains that received inputs from instrumentation circuits at critical components. The inputs could activate protection functions if "coincidences" occurred twice in each train for reliability. On the basis of those inputs the system produced control signals to actuate protective interlocks and reactor trips. The interlocks prevented plant components from operating in a way that could interfere with the accident control process. The various reactor trip signals opened circuit breakers and de-energized the CDRMs allowing all the rods to drop in the core. The entire system was designed with the military dictum "defense in depth" in mind.[46] In addition to redundant and cross checked outputs, the system was designed to be fail-safe: a reactor trip would occur if power was lost to either a protection signal channel or the reactor trip breakers.

Main Steam Line Break (MSLB)

Many types of malfunctions of the reactor and reactor coolant system components were controlled by the Reactor Protection System without activation of the ECCS. A break in any of the main steam lines coming out of the steam generators could be serious enough to require a reactor trip because of resulting cooling of the RCS loops and drop in pressurizer level. A rupture of the 36-inch-diameter main steam header in the Turbine Building Auxiliary Bay could be quickly isolated from the steam generators and reactor by the isolation valves without cool down. A rupture up-stream of the valves in the 24-inch-diameter steam lines exiting the generators was more serious. In that event the Safety Injection would be actuated from the pressurizer signals to insure a flow of borated cooling water to prevent the reactor from returning to power after the trip.[47]

Containment Air Recirculation System (CAR)

The CAR system provided cooling and recirculation of the Containment atmosphere during normal operations with four air recirculation units. Air flow was bypassed only through the cooling coils and fans. During a LOCA or Main Steam Line Break, steam would be released into Containment, 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 re-aligned itself to provide cooling and depressurization of the atmosphere by activating pre-filters and charcoal filters for post-LOCA iodine removal. After an accident the units filtered the air to reduce particulate and iodine concentrations.

The face dampers automatically opened if the containment air pressure rose above a set point. The air-powered dampers used DC power for activation with battery back-up. 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 recirculation. 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 iodines within two hours after an accident via isotropic exchange.[48] Temperature sensors sent alarms to the control room if the filters heated beyond 325 F during post accident filtration. The Residual Heat Removal System provided fire protection water sprays to the filters.[49] 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.

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).[50] 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.[51] 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.[52] 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. [53]

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 RHRS) 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.[54] 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.[55]

Containment Spray System [f]

The Containment Spray System was installed as a backup to the CAR system for depressurization of containment after a Loss of Coolant Accident.[56] 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 a spray ring 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.[57] Initiation in a LOCA was not automatic as the priority was to have the RHR pumps lined up to supply the reactor first.[g] Water could also be taken from the Refueling Cavity Water Storage Tank. 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.[58] 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.[59] The valves would fail on power loss in the closed position[60] 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.

ECCS Controversy

The accuracy of some of the Engineered Safety System design bases were questioned in the 1966-1974 Facility Description and Safety Analysis, and some operational deficiencies were noted in the 1987-1995 Plant

---

[f] 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).

[g] 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.)

Information Book. While no such problems in the ECCS were noted in either document, the efficacy of that system was questioned by some in the industry and by nuclear watchdog groups. There were doubts about a number of assumptions made by the designers of several manufactures' plants regarding the projected way the ECCS would function:[61]

- Flow blockage. Critics doubted that the design process considered the possibility that fuel rods left exposed by coolant draining out at high pressure might heat up rapidly and expand before the core deluge and core cooling water could reach them. The rods were so closely positioned that there was concern that swollen rods might prevent arriving emergency coolant to reach the overheating sections, leaving them to finally rupture and release the encased radioactivity.

- Chemical Reactions. There was concern that the Zircaloy rods used in some stations would undergo physical changes (weakening) at well below the melting point (2,300° F) that the system was designed to control.[62] Whether the stainless steel clad rods in the Connecticut Yankee core were subject to the same problem is un-documented. As noted above, there was concern about the production of hydrogen gas from a stainless steel/coolant reaction.

- ECCS Bypass. There were concerns that the emergency flow could bypass the core and blow out through the leaking section of reactor coolant piping.[63]

- Conflict of Interest. Tests on the ability of core cooling flow to remove heat were done by the manufacturers, not by outside, independent laboratories.[64]

As a result of these issues, the AEC held hearings from January 1972 to December 1973 which determined that the issues had been resolved.[65] The report produced was met with further criticism resulting in a year- long follow-up study that generally supported the earlier methodology.[66] The 1979 Three Mile Island accident was not an effective indicator since operators shut off the ECCS before it could be effective. As of 1980 no full scale tests on the system had been conducted.[67]

NOTES

1. Stern 1964: 244.

2. Connecticut Yankee Atomic Power Company 1987-1995: Chapter 7, page 1.

3. Stern 1964: 253.

4. Howe 1976: 39.

5. Connecticut Yankee Atomic Power Company 1966-1974: 10.3.1-7.

6. Witzke and Voysey 1960: 105.

7. Ibid: 41

8. Connecticut Yankee Atomic Power Company 1966-1974: 5.2.7-1.

9. Connecticut Yankee Atomic Power Company 1987-1995: Chapter 5, page 1.

10. Witzke and Voysey 1960: 105.

11. Connecticut Yankee Atomic Power Company 1987-1995: Chapter 5, page 4.

12. Ibid: 6.

13. Ibid: 115.

14. Ibid: 86

15. Ibid: 3.

16. Connecticut Yankee Atomic Power Company c1974: 2-2.

17. Connecticut Yankee Atomic Power Company 1987-1995: Chapter 5, page 88.

18. Ibid: 90.

19. Ibid: 94.

20. Ibid: 23.

21. Ibid: 8.

22. Ibid: 10.

23. Ibid: 11, 41.

24. Ibid: 12.

25. Connecticut Yankee Atomic Power Company 1987-1995: Chapter 6, page 1.

26. Ibid: 14, Houff 2006: personal correspondence.

27. Connecticut Yankee Atomic Power Company 1987-1995: Chapter 6, page 3.

28. Ibid: page 8.

29. Ibid: Chapter 5, page 66.

30. Ibid: 17.

31. Ibid: 23.

32. Ibid: 21, Chapter 7, page 22..

33. Ibid 1987-1995: Chapter 7, page 37.

34. Connecticut Yankee Atomic Power Company 1993-1998: 1.2.4.4.3-4.

35. Connecticut Yankee Atomic Power Company 1987-1995. Chapter 21, page 1.

36. American Society of Mechanical Engineers 1920: 40.

37. MacNaughton 1950: 493.

38. Connecticut Yankee Atomic Power Company 1987-1995. Chapter 21, page 13.

39. Houff 2006: personal communication.

40. Connecticut Yankee Atomic Power Company 1987-1995: Chapter 21, page 5.

41. Ibid: page 16.

42. Ibid: 31.

43. Ibid: Chapter 75, page 1.

44. Ibid: 5.

45. Connecticut Yankee Atomic Power Company 1966-1974: 4.3-1.

46. Connecticut Yankee Atomic Power Company 1987-1995: Chapter 75, page 11.

47. Ibid 1966-1974: 10.3.3-1

---

48. Ibid: 9.

49. Ibid: 5.

50. Connecticut Yankee Atomic Power Company 1966-1987: 3.6-4.

51. Ibid

52. Ibid: 3.6-5.

53. Ibid: 3.6-6

54. Connecticut Yankee Atomic Power Company 1987-1995: Chapter 56, page: 63.

55. Ibid: 67.

56. Connecticut Yankee Atomic Power Company 1966-1974: 3.6-6.

57. Connecticut Yankee Atomic Power Company 1987-1995: Chapter 5, page 16.

58. Connecticut Yankee Atomic Power Company 1966-1974: 3.6-6A.

59. Ibid: 3.6-6.

60. Connecticut Yankee Atomic Power Company 1987-1995: Chapter 5, page 79.

61. Howe 1976: 41, Nero 1979: 89.

62. Weinberg 1994: 197.

63. Connecticut Yankee Atomic Power Company 1987-1995: 87

64. Ford 1982: 104.

65. Ford 1982: 127, Howe 1976: 41.

66. Lewis 1980: 59.

67. Ibid: 58.

APPENDIX B - SUMMARY OF STRUCTURES, PRIMARY FUNCTIONS, AND MAJOR EQUIPMENT

NAME AND DATES	SUMMARY DESCRIPTION	PLANT SYSTEM	LOCATION	MAJOR EQUIPMENT (usually in sequence of use)
SCREENWELL HOUSE 1964-1966	Screenwell: concrete structure ±65'x19.5' with 4 trash racks on chain assemblies, each ±12' wide x 39' high; fish frightener and de-icing facilities Pumpwell: steel-framed superstructure with insulated Galbestos siding & aluminum louvers, 31.5'x78.8', ±15' high above ground el. 21.5'; concrete foundations to ±el. -18'	SERVICE WATER	Pumpwell	4 Service Water Pumps (P37-1A/1B/1C/1D)
		CIRCULATING WATER	Pumpwell	4 Circulating Water Pumps (P34-1A/1B/1C/1D)
REACTOR CONTAINMENT 1964-1966	circular domed reinforced concrete structure with walls up to 4.5' thick, 144' outside diameter, 170' high aboveground with 70'-high dome section; 218' above bottom of sump at el. -23'; major floors at els. -19.6', 1.5', 16', 22', 48.5'.	REACTOR VESSEL		Reactor Vessel (E-1), Control Rod Drive Mechanisms
		REACTOR COOLANT SYSTEM		Reactor
			el. 16'	4 Steam Generators (E-6-1/2/3/4)
			el. 22'	4 Reactor Coolant Pumps (P-17-1/2/3/4)
			el. 16'	Pressurizer (E-8-1)
			el. 1.5'	Pressurizer Relief Tank (TK-8-1)
				8 motor-operated Loop Isolation Valves, 4 motor-operated Loop Bypass Valves, 4 Loop Over Pressure Check Valves
		MAIN STEAM	el. 16'	4 Steam Generators (E-6-1/2/3/4)
		SERVICE WATER		4 Containment Air Recirculation Fan Motor Coolers (E-77-1/2/3/4)
				4 Containment Air Recirculation cooling coils (E-37-1/2/3/4)
		EMERGENCY CORE COOLING	on top of reactor vessel	2 motor-operated valves supplied borated water from Low-Pressure Safety Injection Pumps (P-92-1A/1B) in Primary Aux.Bldg.
				4 headers to each reactor coolant system loop
				charging header to Loop 2 cold leg, for borated water from

APPENDIX B - SUMMARY OF STRUCTURES, PRIMARY FUNCTIONS, AND MAJOR EQUIPMENT

NAME AND DATES	SUMMARY DESCRIPTION	PLANT SYSTEM	LOCATION	MAJOR EQUIPMENT (usually in sequence of use)
				Centrifugal Charging Pumps (P-18-1A/1B) in Prim. Aux. Bldg.
REACTOR CONTAINMENT (cont.)		CHEMICAL & VOLUME CONTROL		Reactor Coolant System Letdown: motor-operated isolation valve (LD-MOV-200)
		CHEMICAL & VOLUME CONTROL (cont.)	Regenerative Heat Exchanger Cubicle (el. 22')	Reactor Coolant System Letdown: 3 Regenerative Heat Exchangers (E-7-1A/1B/1C)
				Reactor Coolant Pump Seal Water Injection: Reactor Coolant Pump Seal Package
		CONTAINMENT SPRAY	spray header ring inside Reactor Containment liner at el. 110'	nozzles for borated water pumped from 4 sources in Primary Auxiliary Building: 2 Residual Heat Removal Pumps, P-14-1A/1B and Low Pressure Safety Injection Pumps, P-92-1A/1B
		CONTAIN. AIR RE-CIRCULATION	el. 22'	4 fan/filter units (F-17-1/2/3/4)
		CONTAINMENT AIR	el. 22'	2 Containment Air System compressors (C--1A/1B)
			el. 22'	Air Receiver (TK-92-1A)
			el. 22'	Pre-Filter (FL-96-1A)
			el. 22'	Air Dryer (FL-98-1A/1B)
			el. 22'	After-Filter
TURBINE BUILDING 1964-1966	Main Building exterior: gable-end steel-framed structure 268'x110', 122' high with metal roof; enameled fluted aluminum siding except south wall of concrete block & glazed brick; 4.25'-high brick base on north & west sides; on west side, 4 arched louvered aluminum panels above brick	CIRCULATING WATER	Main Ground Floor	2 Main Condensers (E-23-1A, E-23-1B), each with 2 inlet & 2 outlet waterboxes, and 1 priming tank
			Main Mezzanine	Vacuum Priming Tank (TK-27-1A)
			Main Ground Floor	2 Vacuum Priming Pumps (P-36-1A, P-36-1B)

APPENDIX B - SUMMARY OF STRUCTURES, PRIMARY FUNCTIONS, AND MAJOR EQUIPMENT

NAME AND DATES	SUMMARY DESCRIPTION	PLANT SYSTEM	LOCATION	MAJOR EQUIPMENT (usually in sequence of use)
TURBINE BUILDING (cont.)	base (2 panels removable), & paired 2.5'-wide plastic window sections at alternate columns; on east side, 107' of continuous aluminum-sash windows at el. 62.5'; plastic window panels in north and south pediments Auxiliary Bay exterior: flat-roofed steel-framed structure 240-268'x27', 38' high at southeast corner of main building, with enameled fluted aluminum siding Interiors: circulating water intakes and discharges to el. -6, with discharge tunnel including de-icing line intake; in main and auxiliary sections, ground floor el. 21.5', mezzanine level el. 37.5; main section reheater level el. 47.5', operating floor el. 59.5' with crane rail top el. 98.25'. From ground floor, reinforced concrete pedestals supported turbines & generator on operating floor; steel-framed mezzanine & reheater levels independent from turbine-generator foundations. 3 concrete sump areas created in Auxiliary Bay Ground Floor	MAIN STEAM	Aux. Mezzanine	4 24"-dia. Main Steam Lines & 36"-dia. Main Steam Manifold [suspended at el. 47.5']; 2 30"-dia. Main Steam Pipes
			Operating Floor	High-Pressure Turbine (TG-1) with 2 turbine main stop trip valves, & 2 governor control valves on each main stop trip valve
			Main Reheater Fl.	4 Moisture Separator Reheaters (E-28-1A through E-28-1D)
			Operating Floor	2 Low-Pressure Turbines (TG-1A, TG-1B)
			Main Mezzanine	High-Pressure Turbine Steam Dump & Valves
			Main Reheater Fl.	Low-Pressure Turbine Steam Dump & Valves
		MAIN GENERATOR	Operating Floor	Main Generator
		CONDENSATE & FEEDWATER	Main Ground Floor	2 Main Condensers (E-23-1A, E-23-1B)
			Main Ground Floor	2 Condensate Feedwater Pumps (P-35-1A, P-35-1B) [south of condensers]
ca. 1978	Wastewater Treatment Facilities added to south end of Main Ground Floor		Main Mezzanine	2 Priming Air Ejectors (EJ-2-1A/1B) & 2 Main Air Ejectors (EJ-1-A, EJ-1-B)
1979-1982			Main Mezzanine	Gland Steam Condenser (E-64-1A)
			Main Mezzanine	#6A Low-Pressure Feedwater Heater (E-22-1A) [through Main Condenser A]
			Main Mezzanine	#6B Low-Pressure Feedwater Heater (E-22-1B) [through Main Condenser B]
		CONDENSATE & FEEDWATER (cont.)	Main Mezzanine	#5A Low-Pressure Feedwater Heater (E-21-1A) [through Main Condenser A]

APPENDIX B - SUMMARY OF STRUCTURES, PRIMARY FUNCTIONS, AND MAJOR EQUIPMENT

NAME AND DATES	SUMMARY DESCRIPTION	PLANT SYSTEM	LOCATION	MAJOR EQUIPMENT (usually in sequence of use)
			Main Mezzanine	#5B Low-Pressure Feedwater Heater (E-21-1B) [through Main Condenser B]
			Operating Floor	#4A & 4B Low-Pressure Feedwater Heaters (E-20-1A, E-20-1B) [southeast and southwest corners]
TURBINE BUILDING (cont).			Operating Floor	#3a & 3B Low-Pressure Feedwater Heaters (E-19-1A, E-19-1B) [north of LP feedwater heaters 4A & 4B]
			Aux. Mezzanine	#2a & 2B Low-Pressure Feedwater Heaters (E-18-1A, E-18-1B)
			Main Ground Floor	Feedwater Heater Drain Receiver Tank (TK-23-1A) [north of Condenser A]
			Main Ground Floor	2 Feedwater Heater Drain Pumps (P-33-1A/1B) [north of TK-23-1A]
			Aux. Ground Floor	2 Steam Generator Feedwater Pumps (P-31-1A/1B)
			Aux. Mezzanine	#1A & 1B High-Prressure Feedwater Heaters (E-17-1A/1B) [fed common header to 4 Steam Generator Feed Lines]
			Aux. Mezzanine	4 elec. motor-operated valves, feed regulating valves & manual isolation valves per Steam Generator Feed Line
		AUXILIARY FEEDWATER	Aux. Mezzanine	Auxiliary Feed Bypass Valves
		SERVICE WATER	Main Ground Floor	2 Main Turbine Lube Oil Coolers (E-60-1A/1B)
			Main Ground Floor	2 Closed Cooling Water System Heat Exchangers (E-70-1A/1B)
			lower Main Generator casing	4 Main Generator Exciter Hydrogen Coolers (E-62-1A through E-62-1D)

APPENDIX B - SUMMARY OF STRUCTURES, PRIMARY FUNCTIONS, AND MAJOR EQUIPMENT

NAME AND DATES	SUMMARY DESCRIPTION	PLANT SYSTEM	LOCATION	MAJOR EQUIPMENT (usually in sequence of use)
			Main Ground Floor	2 Main Generator Seal Oil Coolers(E-61-1A/1B)
			Main Mezzanine	2 Main Generator Isolated Phase Bus Coolers (E-48-1A/1B)
		SERVICE WATER	Main Mezzanine	Main Exciter Cooler (E-115-1A)
.**TURBINE BUILDING** (cont)		CONTROL AIR	Aux. Ground Floor	2 Control Air Compressors (C-3-1A/1B)
			Aux. Ground Floor	2 Control Air Receivers (TK-29-1A/1B)
			Aux. Ground Floor	4 Control Air Filters (FL-8-1A/1B, FL-9-1A/1B)
			Aux. Ground Floor	2 Control Air Dehydrator (FL-35-1A/1B)
			Main Ground Floor	Control Air Compressor (C-3-1C)
			Main Ground Floor	Control Air Receiver (TK-29-1C)
			Main Ground Floor	2 Control Air Filters (FL-10-1A/1B)
			Main Ground Floor	Control Air Dehydrator (FL-35-1C)
		SERVICE AIR	Aux. Ground Floor	Service Air Intake Filters
			Aux. Ground Floor	2 Service Air Compressors (C-2-1A/1B)
			Aux. Ground Floor	2 Aftercoolers & Moisture Separators (F-49-1A/1B)
			Aux. Ground Floor	Service Air Receiver (TK-28-1A)
		WATER TREATMENT	Aux. Ground Floor	Well Water Filter (FL-44-1A)
			Aux. Ground Floor	Vacuum Deaerator (D-1-1A)
			Aux. Ground Floor	2 Vacuum Deaerator Pumps (P-30-1A/1B)

APPENDIX B - SUMMARY OF STRUCTURES, PRIMARY FUNCTIONS, AND MAJOR EQUIPMENT

NAME AND DATES	SUMMARY DESCRIPTION	PLANT SYSTEM	LOCATION	MAJOR EQUIPMENT (usually in sequence of use)
			Aux. Ground Floor	2 Forwarding Pumps (P-35-!a/!b)
			Aux. Ground Floor	Demineralized Water Filter (FL-45-1a)
DISCHARGE CANAL 1965-1966	±7000' from Turbine Building discharge tunnel to Connecticut River; upstream ±162' is steel-braced, timber-sheet-pile-sided flume ending at rock weir, drops from el.-5.5' to -10' into earthen canal 65'-80' wide at bottom, 130'-160'-wide at outer ends of berms with gravel or rip-rap inside slopes; 700' wide at river	SERVICE WATER; CIRCULATING WATER		
12R SWITCHYARD		345 KV		319 Main Transformer, 309 Reactor Coolant Bus Transformer, main transformer secondary side disconnects, main transformer output motor-operated disconnects, 320 Line to 14B Switchyard
14B SWITCHYARD		345 KV		ground disconnect, manual disconnects, power circuit breakers, motor-operated disconnects, blockhouse
TERRY TURBINE BUILDING & NON-RETURN VALVE STATION 1964-1966	Steel-framed structure with Galbestos siding, 40' wide, 43.5' high & extending 11' west of the Reactor Containment: Ground Floor el. 21.5' contained 2 auxiliary feed water pumps. Remaining elevations (el. 31', 41', 49', 57') contained structural steel used to support the main steam and feed water system piping and valves; non-return valve station at upper level.	MAIN STEAM	Terry Turbine Bldg	2 Auxiliary Steam powered Steam Generator Pumps (P-32-1A/1B)
			Non-Return Valve Station	Non-Return Valves (2 each on 4 steam lines)
AUXILIARY FEEDWATER PUMP SKID ENCLOSURES A & B		MAIN STEAM		Electrical Auxiliary Steam Generator Feed Pump (P-32-1C)

APPENDIX B - SUMMARY OF STRUCTURES, PRIMARY FUNCTIONS, AND MAJOR EQUIPMENT

NAME AND DATES	SUMMARY DESCRIPTION	PLANT SYSTEM	LOCATION	MAJOR EQUIPMENT (usually in sequence of use)
SERVICE BUILDING 1964-1966	Steel-framed, concrete-walled structure with Galbestos siding, 304'x43'-87.5', 20'-55' high: Ground floor el. 21.5' includes 1-story, 20'-high warehouse & maintenance shop 179.5'x43'&62', & locker rooms/ offices 124.5'x62'&87.5'; Mezzanine el. 41.5' 62'x103' ; Operating Floor at el. 59.5 85'x103', overlying part of Turbine Bldg Aux. Bay.		Ground Floor	Cable Spreading Area
			Mezzanine	Switchgear Room A
			Operating Floor	Control Room
			Operating Floor	Process Computer Room
1981-1984	renovations including new roof and concrete walls for seismic protection, raised floor in Control Room, & Chemistry Lab			
PRIMARY AUXILIARY BUILDING 1964-1966	Reinforced-concrete structure ±70x150', 32' high with ≥1'-thick walls, concrete floors. 2 main levels, each with 4-ton monorail systems: ground floor at el. 21.5', second floor at el. 35.5.' Sections above second floor generally steel framed with insulated Galbestos siding. At east end, Residual Heat Removal Pit, ±70x35.5' extends to el. -19.' Other sections/levels include: 3-level Boron Recovery Room, ±29.5' square at elevations. 35.5-36.5', 25.5' & 15.5'; High-Pressure/Low-Pressure Safety Injection Cubicle at el. 15.5; Blowdown/Sample & Non-Radioactive Valve Room, ±15x25' at el. 22'; Radioactive Valve Room ±10x19' at el. 25'; Charging & Metering Pump Rooms at el. 15.5'; Seal Water Filter Cubicle at el. 13.4.' Pipe galleries below central longitudinal axis at elevations of 13-14'	PRIMARY WATER	Ground Floor (east end)	2 Primary Water Transfer Pumps (P-29-1A/1B)
			Ground Floor (east end)	2 Recycled Primary Water Transfer Pumps (P-118-1A/1B)
		SERVICE WATER	Second Fl. (near W. end) through floor	2 Component Cooling Heat Exchangers (E-4-1A/1B)
			Boron Recovery Room (upper level)	Boron Recovery Overhead Condenser (E-14-1A)
			above Second Floor, el. 40'	Steam Generator Blowdown Tank Vent Condenser (E-78-1A)
			Ground Floor	2 Steam Generator Sample Chiller Condensers (C-16-1A/1B)
			Residual Heat Removal Pit, el. -19'	2 Residual Heat Removal Heat Exchangers (E-5-1A/1B)

APPENDIX B - SUMMARY OF STRUCTURES, PRIMARY FUNCTIONS, AND MAJOR EQUIPMENT

NAME AND DATES	SUMMARY DESCRIPTION	PLANT SYSTEM	LOCATION	MAJOR EQUIPMENT (usually in sequence of use)
			Second Floor	2 Adams Filters (FL-53-1A/1B)
			Boron Recovery Room (lower level)	Boron Recovery Distillate Cooler (E-15-1A)
			Second Fl., near top of Boric Acid Mix Tank	Boric Acid Mix Tank Vent Condenser (E-78-1A)
PRIMARY AUXILIARY BUILDING (cont.)			Primary Drains Tank Cubicle in Residual Heat Removal Pit	Primary Drains Tank Vent Condenser (E-11-1A)
		COMPONENT COOLING WATER	Second Floor (northwest corner)	2000-gal. Component Cooling Surge Tank (TK-5-1A)
			Ground Floor (northwest corner)	3 Component Cooling Water pumps (P-13-1A/1B/1C) [below Component Cooling Surge Tank]
		COMPONENT COOLING WATER	Ground Floor (northwest corner)	2 Component Cooling Water Heat Exchangers (E-4-1A/1B) [below Component Cooling Surge Tank]
		BORON RECOVERY	Boron Recovery Room, 2nd level	2 Boron Recovery Waste Liquid Transfer Pumps (P-22-1AA/1B)
			Boron Recovery Room, 2nd level	Boron Recovery Distillate Feed Heat Exchanger (E-12-1A)
			Boron Recovery Room, 1st level	Boron Recovery First Stage Evaporator Bottoms Pump (P-23-1A)
			Boron Recovery Room, 1st level	Boron Recovery First Stage Evaporator Boiler (E-43-1A)
			Boron Recovery Room, 1st level	Boron Recovery First Stage Evaporator (EV-1-A)
			Boron Recovery Room, 1st level	Boron Recovery Second Stage Evaporator Bottoms Pump (P-25-1A)

APPENDIX B - SUMMARY OF STRUCTURES, PRIMARY FUNCTIONS, AND MAJOR EQUIPMENT

NAME AND DATES	SUMMARY DESCRIPTION	PLANT SYSTEM	LOCATION	MAJOR EQUIPMENT (usually in sequence of use)
			Boron Recovery Room, 1st level	Boron Recovery Second Stage Evaporator Boiler (E-44-1A)
			Boron Recovery Room, 2nd level	Boron Recovery Second Stage Evaporator (EV-2-A)
			Boron Recovery Room, 3rd level	Boron Recovery Evaporator Overhead Condenser (E-14-1A)
			Boron Recovery Room, 3rd level	Boron Recovery Distillate Accumulator (TK-18-1A)
			Boron Recovery Room, 2nd level	2 Boron Recovery Distillate Pumps (P-26-1A/1B)
			Boron Recovery Room, 1st level	Boric Acid Recovery Cooler (E-16-1A)
			Ground Floor	Liquid Waste Control Board
			through Second Fl.	Boric Acid Mix Tank (TK-2-1A)
		STEAM GENERATOR BLOWDOWN	Blowdown Room	Steam Generator Blowoff Tank (TK-22-1A)
		STEAM GENERATOR BLOWDOWN	above Second Fl.	2 Steam Generator Blowoff Tank Condensers (E-90-1A/1B)
			Residual Heat Removal Pit	Blowoff Tank Cooler (E-91) [at el. -8.6']
		RESIDUAL HEAT REMOVAL	Residual Heat Removal Pit	2 Residual Heat Removal Pumps (P-14-1A/1B) [at el .-19']
			Residual Heat Removal Pit	2 Residual Heat Exchangers (E-5-1A/1B) [between el. -5' & -19']

APPENDIX B - SUMMARY OF STRUCTURES, PRIMARY FUNCTIONS, AND MAJOR EQUIPMENT

NAME AND DATES	SUMMARY DESCRIPTION	PLANT SYSTEM	LOCATION	MAJOR EQUIPMENT (usually in sequence of use)
		CHEMICAL & VOLUME CONTROL	Metering Pump Room	Reactor Coolant Letdown: Non-Regenerative Heat Exchanger (E-76-1A)
			Second Floor	Reactor Coolant Letdown: Volume Control Tank (TK-6-1A)
			Ground Floor	Purification Pump (P-12-1A)
			through Second Fl.	Boric Acid Mixing Tank (TK-2-1A)
			Ground Floor	2 Boric Acid Pumps (P-9-1A/1B)
			Ground Floor	Boric Acid Filter (FL-15-1A)
			Ground Floor	Boric Acid Blender (M-9-1A) [atop Boric Acid Filter]
			Ground Floor	Boric Acid Strainer (ST-6-1A)
			Charging Pump Room	2 Centrifugal Charging Pumps (P-18-1A/1B)
			Metering Pump Room	Charging Metering Pump (P-11-1A)
			Seal Water Filter Cubicle, el. 13.4'	Reactor Coolant Pump Seal Water Injection: 2 Seal Water Supply Filters (FL-59-1A/1B)
			Seal Water Filter Cubicle, el. 13.4'	Reactor Coolant Pump Seal Water Injection: 2 Seal Water Return Filters (FL-36-1A/1B)
			Residual Heat Removal Pit	Reactor Coolant Pump Seal Water Injection: Seal Water Heat Exchanger (E-45-1A)
			Second Floor	Chemical Addition Tank (TK-7-1A)
PRIMARY AUXILIARY BUILDING (cont.)		CONTAINMENT PURGE	Second Floor	2 Pure & Dilution Air Fans (F-50A/1B)
		EMERGENCY	High-Pressure/Low-Pressure Safety	2 Low-Pressure Safety Injection Pumps (P-92-1A/1B);

APPENDIX B - SUMMARY OF STRUCTURES, PRIMARY FUNCTIONS, AND MAJOR EQUIPMENT

NAME AND DATES	SUMMARY DESCRIPTION	PLANT SYSTEM	LOCATION	MAJOR EQUIPMENT (usually in sequence of use)
		CORE COOLING	Injection Cubicle	2 High-Pressure Safety Injection Pumps (P-15-1A/1B)
			Charging Pump Room	2 Centrifugal Charging Pumps (P-18-1A/1B) [shifted suction from Vol. Control Tank (TK-6-1A) to Refueling Cavity Water Storage Tank (TK-4-1A) outside Prim. Aux. Bldg.]
		CONTAINMENT SPRAY	Boron Recovery Room, 1st level	2 Low-Pressure Safety Injection Pumps (P-92-1A/1B)
			Residual Heat Removal Pit	2 Residual Heat Removal Pumps (P-14-1A/1B)
		LIQUID WASTE	Residual Heat Removal Pit	2 Aerated Drains Tanks (TK-12-1A/1B)
			Residual Heat Removal Pit	2 Aerated Drain Tank Pumps (P-20-1A/1B)
			Residual Heat Removal Pit	2 Aerated Liquid Waste Strainers (ST-1-1A/1B)
			Ground Floor	2 Waste Test Tank Pumps (P-28-1A/1B)
			Drumming Room, Ground Floor	Chemical Nuclear Processing Skid
			Ground Floor	Liquid Waste Control Board
		WASTE GAS	pipe gallery below Blowdown Room	Valve Stem Leakoff Cooler (E-85)
			Primary Drains Tank Cubicle in Residual Heat Removal Pit	Primary Drains Collecting Tank (TK-11-1A)
			Primary Drains Tank Cubicle in Residual Heat Removal Pit	Primary Drains Tank Vent Condenser (E-11-1A)
			Residual Heat Removal Pit	2 Primary Drains Tank Pumps (P-19-1A/1B)

APPENDIX B - SUMMARY OF STRUCTURES, PRIMARY FUNCTIONS, AND MAJOR EQUIPMENT

NAME AND DATES	SUMMARY DESCRIPTION	PLANT SYSTEM	LOCATION	MAJOR EQUIPMENT (usually in sequence of use)
			Ground Floor	Waste Gas Control Board
DIESEL GENERATOR BUILDING 1964-1966	1-story reinforced-concrete structure, 25'x32', 14' high from el. 21.5', off southwest corner of Primary Auxiliary Building	SERVICE WATER, EMERGENCY GENERATION		3 Diesel Generators
NEW DIESEL GENERATOR BUILDING 1969-1970	1 story reinforced-concrete/concrete block structure, 91'x45'&30'	SERVICE WATER, EMERGENCY GENERATION		2 Diesel Generators (2A/2B), control cabinets, emergency buses
ION EXCHANGE AREA 1964-1967	2 adjacent reinforced- concrete structures: Ion Exchange Structure, 21.8'x73.5', 18' high from el. 14.6', with deck at el. 22.5' in front of ion exchanger & demineralizer facilities which are arrayed within Ion Exchange Structure in 17'-high chambers; Spent Resin Storage Pit, 13.3'x17.3', 17' high from el. 22' with chamber for removable liner, over pit extending to el. 7.5'; within pit, Ion Exchange Sump Pump (P-63-1A) to el. 2.5'.	BORON RECOVERY	Ion Exchange Struct.	Waste Liquid Ion Exchanger (I-6-1A)
			Ion Exchange Struct.	Waste Liquid Transfer Filter (FL-13-1A)
		WASTE LIQUID	Ion Exchange Struct.	Aerated Drains Demineralizer (I-3-1a)
			Ion Exchange Struct.	Aerate Drains/Spent Fuel Pit Filter (FL-3-1A/FL-65)
		CHEMICAL & VOLUME CONTROL	Ion Exchange Struct.	Reactor Coolant Letdown: 2 Purification Demineralizer Ion Exchangers (I-1-1A/1B)
			Ion Exchange Struct.	Reactor Coolant Letdown: Reactor Coolant Pre-filter (FL-5-1A)
			Ion Exchange Struct.	Reactor Coolant Letdown: Reactor Coolant Post-Filter (FL-11-1A)
			Ion Exchange Struct.	Purification: Deboronating Ion Exchanger (I-2-1A)
		SPENT FUEL COOLING	Ion Exchange Struct.	Spent Fuel Pool Ion Exchanger (I-1-1C)
			Ion Exchange Struct.	Aerate Drains/Spent Fuel Pit Filter (FL-3-1A/FL-65)

APPENDIX B - SUMMARY OF STRUCTURES, PRIMARY FUNCTIONS, AND MAJOR EQUIPMENT

NAME AND DATES	SUMMARY DESCRIPTION	PLANT SYSTEM	LOCATION	MAJOR EQUIPMENT (usually in sequence of use)
ION EXCHANGE STRUCTURE ADDITION 1973-1974	Reinforced concrete addition to Ion Exchange Structure, 21'x30', 10'-20' high from el. 12.8', with 4 17'-high chambers	WASTE GAS	Ion Exchange Add.	Degasifier Pre-Filter (FL-67)
		WASTE LIQUID	Ion Exchange Add.	Waste Liquid Polishing Demineralizer (I-9)
			Ion Exchange Add.	Waste Evaporator Distillate Filter (Fl-69)
		WASTE LIQUID SOLID WASTE	Spent Resin Storage Pit	Spent Resin Storage Tank (TK-102-1A) Spent Resin Transfer Pump (P-156-1A)
ION EXCHANGE STRUCTURE ADDITION (cont.)		BORON RECOVERY	Ion Exchange Add.	Boron Evaporator Distillate Filter (FL-68)
			Ion Exchange Add.	Boron Recovery Polishing Demineralizer (I-8-1a)
SPENT RESIN STORAGE FACILITY 1979-1980	Part of original Spent Resin Storage Pit removed, replaced to northeast with 22.2'x29.2' lead-lined reinforced-concrete structure with fl. el. 19.2', 10.8' high with 23.5'-high walls on north & east sides, containing 11 5.2'-dia., 10.8'-high steel cells	SOLID WASTE WASTE LIQUID		Removable storage liners in steel cells; 3" PVC pipe drains from cell bottoms into adjacent Spent Resin Storage Pit
NEW & SPENT FUEL BUILDING 1964-1966	48'x117.3' with ground floor el. 21.5'. Spent Fuel Pit 48'x49', with top of 33.5'-deep, 35'x37' steel-lined reinforced-concrete pit at el. 47' & steel-framed, Galbestos-sided structure above to el. 75.5'. Pit has 1168 5'-dia., 14'-high fuel casks & a skimmer system at top of pool; 6-ton bridge crane above pool at el. 67.3'. New Fuel Building 48'x38.2', 54' high above ground floor with reinforced concrete to floor level at el. 47', steel frame & Galbestos siding above; ground floor serves as Spent Fuel Pit pump cubicle; floor at el. 35' supports PVC racks for 114 1'-dia., 12'-high fuel assemblies; 3-ton bridge crane at el. 67.3'.	SPENT FUEL COOLING	Spent Fuel Pit pump cubicle	2 Spent Fuel Cooling Pumps (P-21-1A/1B)
			Spent Fuel Pit pump cubicle	1 Spent Fuel Pool Tube-Type Heat Exchanger (E-10-1A) 1 Spent Fuel Pool Plate-Type Heat Exchanger (E-10-1B)
			Spent Fuel Pit pump cubicle	2 Spent Fuel Pool Skimmer Pumps (P-90-1A/1B)
			Spent Fuel Pit pump cubicle	2 Spent Fuel Pool Skimmer Filters (FL-65-1A/1B)
		SERVICE WATER	Spent Fuel Pit pump cubicle	2 Spent Fuel Pool Heat Exchangers (E-10-1A/1B)

APPENDIX B - SUMMARY OF STRUCTURES, PRIMARY FUNCTIONS, AND MAJOR EQUIPMENT

NAME AND DATES	SUMMARY DESCRIPTION	PLANT SYSTEM	LOCATION	MAJOR EQUIPMENT (usually in sequence of use)
	Reinforced-concrete Radiation Control Area 48'x30', 25.5' high with 3-ton bridge crane at el.39.5'	REFUELING	Spent Fuel Pit el. 49.5'	Fuel Elevator (FU-5); Sluice Gate (FU-7)
WASTE DISPOSAL BUILDING 1973-1974	41'x42' reinforced concrete structure, 55.5' high with floors at el. 0', 18.5' & 21.5', & 35.5'	WASTE GAS	through middle level	Degasifier Preheater (E-86)
			through upper level	Degasifier (TK-58-1A) with Degasifier Vent Cooler (E-89) & Degasifier Vent Condenser (E-87)
			middle level	2 Degasifier Liquid Transfer Pumps (P-106-1A/1B)
			middle level - el. 30'	Degasifier Effluent Cooler (E-88)
WASTE DISPOSAL BUILDING (cont.)		WASTE GAS (cont.)	upper level - el. 50'	Degasifier Vent Cooler (E-89)
			lower level	Waste Gas Surge Tank (TK-59-1A)
			upper level - el. 38.3'	2 Waste Gas Compressors (C-13-1A/1B)
			lower level	3 Waste Gas Decay Tanks (TK-60-1A/1B/1C)
				Waste Gas Sample & Release Header
		WASTE LIQUID	middle level - el. 30'	Waste Evaporator Feed Distillate Exchanger (E-96)
			lower level	Waste Evaporator Reboiler Pump (P-114-1A)
			lower level	Waste Evaporator Reboiler (E-92)
			middle/upper levels	Waste Liquid Evaporator (EV-4)
			middle level	Waste Evaporator Distillate Tank Pump (P-115-1A)
			upper level - el. 47.3'	Waste Evaporator Overhead Condenser (E-93)
			upper level	Waste Evaporator Distillate Tank (TK-64)

APPENDIX B - SUMMARY OF STRUCTURES, PRIMARY FUNCTIONS, AND MAJOR EQUIPMENT

NAME AND DATES	SUMMARY DESCRIPTION	PLANT SYSTEM	LOCATION	MAJOR EQUIPMENT (usually in sequence of use)
			el. 30.' /upper level	Waste Evaporator Distillate Cooler (E-94)
			lower level	Waste Evaporator Bottoms Pump (P-116-1A)
			lower level	Waste Evaporator Bottoms Cooler Preheater (E-97)
			lower level	Waste Evaporator Bottoms Cooler (E-95)
			lower level	Waste Evaporator Bottoms Cooler Circulating Pump (P-120-1A)
			upper level	Floor & Equipment Drain Tank (TK-65)
			lower level	2 Floor Drain Tank Pumps (P-119-1A/1B)
			lower level	2 Equipment Drain Tank Pumps (P-121-1A/1B)
		COMPONENT COOLING WATER	through upper level	Degasifier (TK-58-1A) with Degasifier Vent Cooler (E-89) & Degasifier Vent Condenser (E-87)
WASTE DISPOSAL BUILDING (cont.)		COMPONENT COOLING WATER	upper level - el. 38.3'	2 Waste Gas Compressors (C-13-1A/1B)
			middle/upper levels	Waste Liquid Evaporator (EV-4)
		SERVICE WATER	middle/upper levels	Waste Liquid Evaporator (EV-4)
			through upper level	Degasifier (TK-58-1A) with Degasifier Vent Cooler (E-89) & Degasifier Vent Condenser (E-87)
PRIMARY WATER STORAGE TANK (TK-20-1A) c1964-1966	30'-OD, 150,000-gal. steel tank	PRIMARY WATER	east of New & Spent Fuel Bldg. & yard crane	
RECYCLE PRIMARY WATER STORAGE TANK	22'-OD 150,000-gal. steel tank	PRIMARY WATER	east of Radwaste Reduction Facility	

APPENDIX B - SUMMARY OF STRUCTURES, PRIMARY FUNCTIONS, AND MAJOR EQUIPMENT

NAME AND DATES	SUMMARY DESCRIPTION	PLANT SYSTEM	LOCATION	MAJOR EQUIPMENT (usually in sequence of use)
(TK-62-1A) 1973				
DEMINERALIZED WATER STORAGE TANK (TK-25-1A) c1964-1966	25'-ID 100,000-gal. steel tank	AUXILIARY FEEDWATER	in diked enclosure SW of Reactor Containment	
CONDENSATE STORAGE TANK (TK-25-1B) c1992	25'-ID 100,000-gal. steel tank	AUXILIARY FEEDWATER	SW of Reactor Containment	
2 RECYCLE TEST TANKS (TK-63-1A/1B) c1973	13.6'-OD, 33'-high, 16,000-gal. steel tanks	BORON RECOVERY	concrete-diked enclosure NE of Primary Aux. Bldg	
2 BORON WASTE STORAGE TANKS (TK-14-1A/1B) c1964-1966	26'-OD 75,000-gal. steel tanks	BORON RECOVERY	concrete-diked enclosure east of Primary Aux. Bldg	
AERATED DRAINS HOLDUP TANK (TK-61-1A) c1973	26.3'-high, 24'-ID, 99,280-gal. steel tank	LIQUID WASTE	concrete-diked enclosure NE of Primary Aux. Bldg	2 Waste Evaporator Feed Pumps (P-113-1A/1B)
2 WASTE WATER TREATMENT TANKS (TK-17-1A/1/B) c1964-1966	14'-OD, 16,000-gal. steel tanks	LIQUID WASTE	northeast of Reactor Containment	
REFUELING CAVITY WATER STORAGE TANK (TK-4-1A) c1964-1966	37'-OD, 50.5'-high, 250,000-gal. steel tank	1. CHEMICAL & VOL. CONTROL: PURIFICATION 2. EMERGENCY CORE COOLING 3. CONTAINMENT SPRAY 4. REFUELING	northeast of Reactor Containment	
RADWASTE REDUCTION FACILITY	1-story, metal-framed, metal-sided structure, ±88'x61'&37'	SOLID WASTE	south of Spent Fuel Building	compactors, cleaning facility, storage area for solid radioactive and mixed waste
SWITCHGEAR BUILDING 1987-1990	pile-supported, reinforced-concrete structure 34'x64', 37.5'high above ground fl. el. 21.5';		north of Service Building	instrumentation and control for safe plant shutdown

APPENDIX B - SUMMARY OF STRUCTURES, PRIMARY FUNCTIONS, AND MAJOR EQUIPMENT

NAME AND DATES	SUMMARY DESCRIPTION	PLANT SYSTEM	LOCATION	MAJOR EQUIPMENT (usually in sequence of use)
	basement laundry area el. 13.5'; air handling equipment at el. 27.5; switchgear equipment at el. 41.5'			
115 KV SWITCHYARD 1964-1966		115 KV	southeast of Reactor Containment	115-kv oil circuit breaker, 2 transformers with lightning arrestors, manually- and DC-motor-operated disconnect switches
SERVICE BOILER ROOM 1964-1966	1-story 24'-high, 45'x36.5' steel-framed, concrete-walled structure with Galbestos siding; 2 service doors	HEATING	southwest corner of Service Building	2 oil-fired boilers
ADMINISTRATION BUILDING 1964-1966	2-story, 30.5'-high brick & concrete-block structure 100'x24'(1st fl.) & 36' (2nd fl), with 45'x15' rear section. N,S,&W facades have tripartite facades of glazed brick, fluted enameled aluminum siding, & enameled aluminum sash; W. side has 2 12'-high, 30'-wide arched panels with glass and glazed brick infill & doors in north arched panel		west of Turbine Building	
c1980-1985	1-story ±30'-square Records Building added in similar style to south end, with west face extended north past south arched panel			
INFORMATION CENTER 1964-1967 1977 renovations	1-story concrete-block structure 120'x38'-50', exterior stone base and metal panels		west of main plant	
HEALTH PHYSICS FACILITY 1964-1966	2-story concrete structure, ±35'x51.5', ±30.5' high			count room, processing, office
HEALTH PHYSICS PROJECT TRAILER	±20'x17'			office
HEALTH PHYSICS COUNT MODULE	2-story ±25'x19' temporary			count room
WAREHOUSE #1 1976	1-story steel-framed, metal-sided, gable-roof structure 120'x122.8', 20' high			storage

APPENDIX B - SUMMARY OF STRUCTURES, PRIMARY FUNCTIONS, AND MAJOR EQUIPMENT

NAME AND DATES	SUMMARY DESCRIPTION	PLANT SYSTEM	LOCATION	MAJOR EQUIPMENT (usually in sequence of use)
WAREHOUSE #2 1994	2-story steel-framed, metal-sided, gable-roof structure 120'x145, 24' high			storage
STEAM GENERATOR MOCK-UP BUILDING 1982	±46' dia. Belowground concrete mock-up with wood-framed geospherical superstructure			steam generator mock-up
ENGINEERING OFFICE/ MODULE c1985	2-story wood- and steel-framed modular structure, comprised of several interconnected modules of office space for engineers; connected to the Instrument and Control Operations Building.			
INSTRUMENTATION & CONTROL OPERATIONS BUILDING c1985	2-story wood- and steel-framed modular structure, comprised of several interconnected modules of office space for the Instrumentation and Controls Group and Operations; 2nd floor included a lunch room/ conference room used for daily plant operations meeting; connected to the Engineering Modular.			
UNCONDITIONAL RELEASE FACILITY	Small 1-story steel-framed building located south of Reactor Containment			processing area to release equipment and tools for radwaste control area
TRAINING & STORES OFFICE c1978-1979	metal-framed, metal-sided, gable-roofed structure 96'x50', ±20' high			
EMERGENCY OPERATIONS FACILITY 1980-1981	1-story reinforced-concrete structure 126'x96', 14'-17' high.			
WAREHOUSES A AND B 1982	2 1-story steel-framed metal-sided structures, each ±100'x46'			
OFFICE BUILDING #3 c1989-1994	2-story, gable-roofed, steel-framed & concrete-block structure 100'x120'			