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Thermal-Hydraulic Crossflow Applications

by

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ABSTRACT

The thermal-hydraulic analysis can now be performed using the crossflow computational tools of LYNX1, LYNX2, and LYNXT. These thermal-hydraulic crossflow codes have demonstrated improvements in departure from nucleate boiling ratio predictions over previous closed-channel analyses. This report identifies the methods and criteria used in developing the crossflow models and the application of the models in licensing design analyses for 177-fuel assembly plants.

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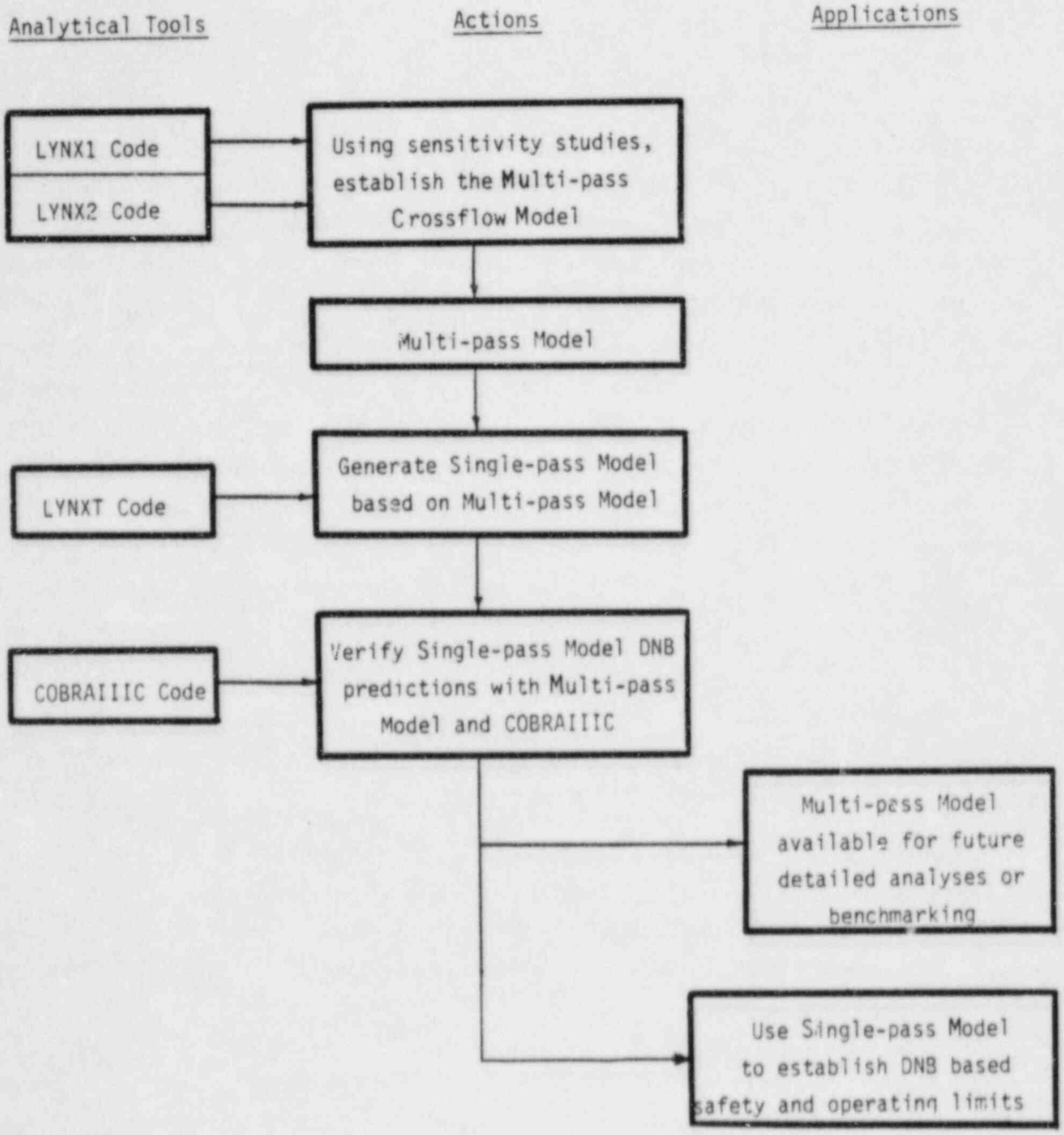
1. INTRODUCTION

Reactor core thermal-hydraulic design and licensing analyses have traditionally used conservative methods which provide significant real but relatively unquantified margins to fuel design limits. The traditional methods use "closed-channel" computer codes in combination with an analysis technique wherein allowances for uncertainties, tolerances, and measurement errors are all considered simultaneously in the most adverse manner. "Crossflow" computer codes, which can predict flow redistribution effects within an open lattice reactor core, provide a significant improvement in modeling accuracy, thereby providing additional departure from nucleate boiling ratio (DNBR) margins relative to the traditional closed-channel modeling. This report describes a revision to the traditional analysis method which, while retaining the inherently conservative treatment of uncertainties, tolerances, and error allowances, incorporates crossflow analysis codes in place of the closed-channel codes.

This report describes the crossflow models developed for LYNX1[1], LYNX2[2], and LYNXT[3] computer codes and demonstrates their accuracy and applicability for DNB calculations. Figure 1-1 shows the approach taken in developing the crossflow code models for licensing application. The LYNXT code, with its single-pass model, will be used for plant licensing analyses. The LYNX1/LYNX2 codes, with their more detailed multi-pass model, will be used for benchmarking.

Figure 1-1. Crossflow Applications Flowchart



2. MODEL DESCRIPTION

The implementation of crossflow modeling in thermal-hydraulic evaluations is performed by using the LYNX1 and LYNX2 computer codes for multi-pass modeling and LYNXT for single-pass modeling. These codes consider the mass and energy exchange between adjacent channels to more accurately predict coolant axial flow behavior. The LYNX1 and LYNX2 codes are run in series (multi-pass modeling) to yield steady-state hot bundle and hot subchannel predictions, respectively. The LYNXT code can simulate the hot channel performance with a single-pass model for steady-state and transient conditions.

The multi-pass model has been established for benchmarking the single-pass model and for providing detailed subchannel hydraulic behavior. The single-pass model has been established for thermal-hydraulic licensing applications.

2.1. LYNX1/LYNX2 Model

The LYNX1 computer program is used to determine the steady-state thermal and hydraulic conditions of the bundle coolant flow in a reactor core by modeling the core on an assembly basis. LYNX1 utilizes one-dimensional conservation equations of mass, momentum, and energy formulated in the axial direction. By using a simplified conservation equation for transverse momentum, LYNX1 can calculate an interbundle crossflow. The forward finite difference numerical solution method is used in converging towards the core exit pressure profile boundary condition. After a converged solution has been obtained, the calculated coolant properties at each axial and transverse location are obtained. The primary output of the LYNX1 code for DNBR analyses is the interbundle diversion crossflow (IBDCF) for the hot bundle (the fuel bundle containing the hot pin). The IBDCF is provided for the four bundle interfaces at each axial location modeled for the hot bundle. The resulting hot bundle IBDCF is then used by the LYNX2 program to establish the governing mass and energy exchange for the hot bundle for each axial location.

The LYNX2 program models the hot bundle and calculates the steady-state subchannel conditions. LYNX2 uses coupled relations for the conservation of mass, energy, and momentum at each axial increment. By incorporating the IBDCF at the periphery of the hot bundle matrix, the edge subchannels transfer mass, energy, and momentum through the periphery of the array. Intersubchannel diversion crossflow is determined from transverse pressure differences. After repeated iterations of the conservation relations at each axial location, the IBDCF propagates throughout the hot bundle. The subchannel critical heat flux (CHF) ratios may then be calculated using local subchannel conditions.

The LYNX1 model used for this study consists of 29 whole or partial fuel bundles representing a symmetric one-eighth portion of a 177-fuel assembly (FA) reactor core. The LYNX1 fuel bundle numbering scheme referred to throughout this report is shown in Figure 2-1. The hot bundle is located in the center fuel location (bundle 1). Interbundle diversion crossflow is provided through the 44 crossflow gaps distributed across the model as shown in Figure 2-2. For design analyses, the axial increment is about 3 inches (as recommended in reference 1).

The control component distribution for the multi-pass model example is shown in Figure 2-3. This distribution is applicable for 177-FA reactor cores containing a combination of burnable poison rod assemblies (BPRAs), control rod assemblies (CRAs), and assemblies with unplugged, or "open," guide tubes. The selection of the control component arrangement is discussed in section 3.1.

The bundle radial peaking distribution, established for design analyses using the multi-pass model, is shown in Figure 2-4. This distribution provides relatively high radial peaks surrounding the centrally located maximum radial peak of the hot bundle. The remaining fuel bundle peaks decrease radially from the hot bundle.

The boundary conditions for the LYNX1 model are a flat exit pressure profile, a flat inlet flow distribution (Figure 2-5), a uniform inlet enthalpy and density profile, and zero crossflow at the inlet. This results in five boundary conditions for the six mass and energy equations. The hot bundle flow factor, for DNBR calculations, is the statistical minimum flow factor (0.95) developed from vessel model flow tests (VMFTs). The selection of these distributions is discussed in section 3.1.

The LYNX2 model comprises 256 linked subchannels and 208 fuel rods. Included in the symmetric model of the hot fuel bundle are 16 control rod guide tubes and 1 instrument guide tube. The subchannel and fuel rod numbering used in the LYNX2 model are provided in Figure 2-6. The power distribution within the hot bundle is referred to as the local peaking distribution where each fuel rod peaking factor is equivalent to the rod absolute power divided by the hot bundle average power. The hot fuel rod exhibits the maximum local peak within the hot bundle with a value of 1.0615. The hot bundle local peaking distribution used in the multi-pass model is shown in Figure 2-7.

The hot channel in LYNX2 is the subchannel in which the minimum DNBR occurs. Since the fuel bundle comprises five types of subchannels (unit, control rod, instrument guide tube, wall and corner), hot channels are designated for each channel type. Hot channel factors and flow area reduction factors are then applied to each hot channel. Figure 2-8 shows the location of the hot channels throughout the hot bundle. The limiting hot channel, possessing the minimum DNBR with the B&W-2 CHF correlation[3], is the hot unit channel for the variety of licensing type operating conditions considered in the development of the multi-pass model.

Transient DNB predictions are obtained with the RADAR[9] code for the multi-pass model. RADAR is a transient closed-channel code widely used in traditional DNB analyses. In the multi-pass modeling scheme, RADAR is initialized to the LYNX2 hot channel minimum DNBR at the beginning of the transient by matching the RADAR hot channel flow rate to that predicted by LYNX2 for the hot subchannel at the axial location of minimum DNBR and by the use of an enthalpy rise factor ($FL\Delta H$) to obtain the desired DNBR value. After the DNBR is initialized, the inherent conservatism of the RADAR code results in conservative transient DNBR predictions.

Figure 2-1. Bundle Location Numbering System (LYNX1)

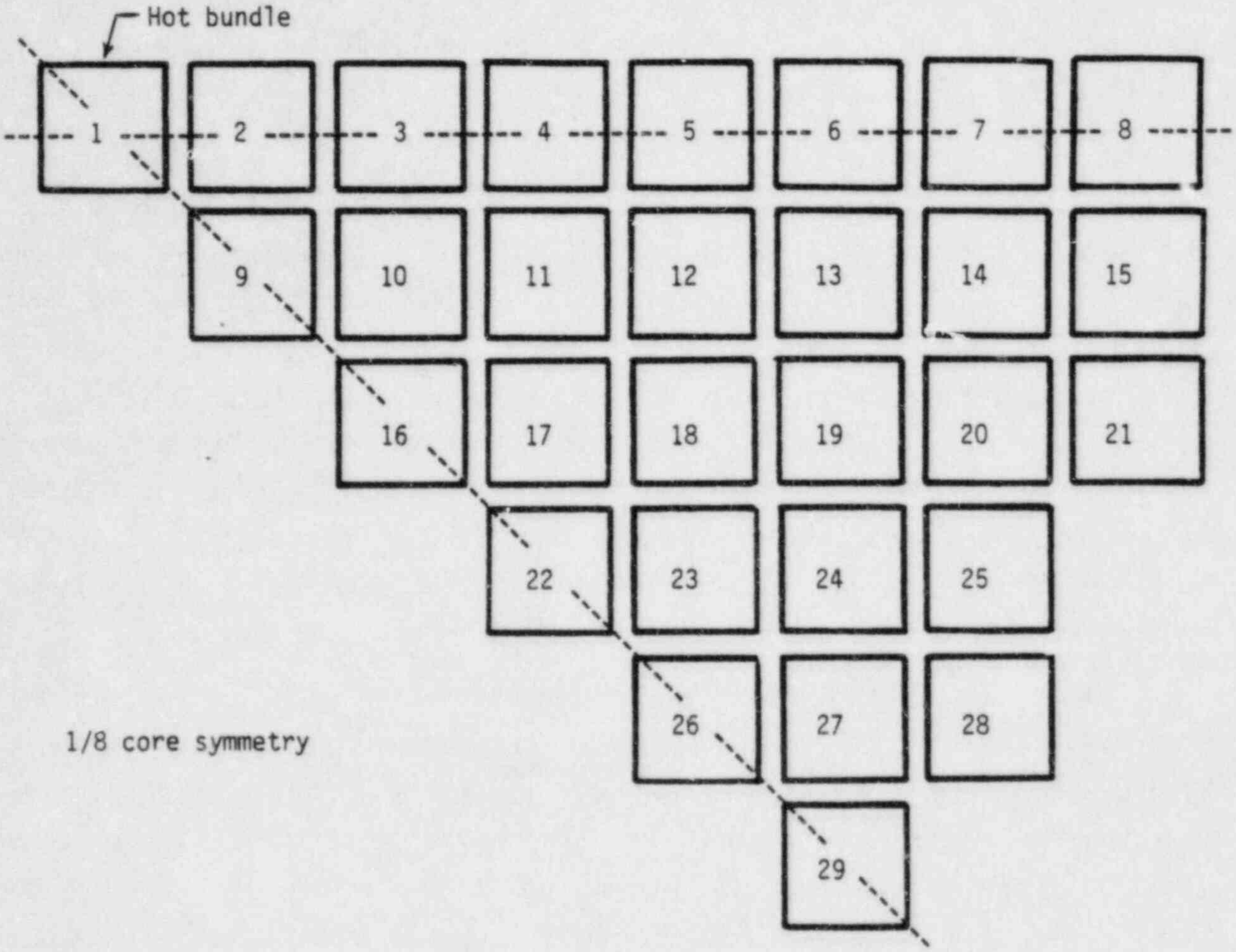


Figure 2-2. Crossflow Gap Numbering System (LYNX1)

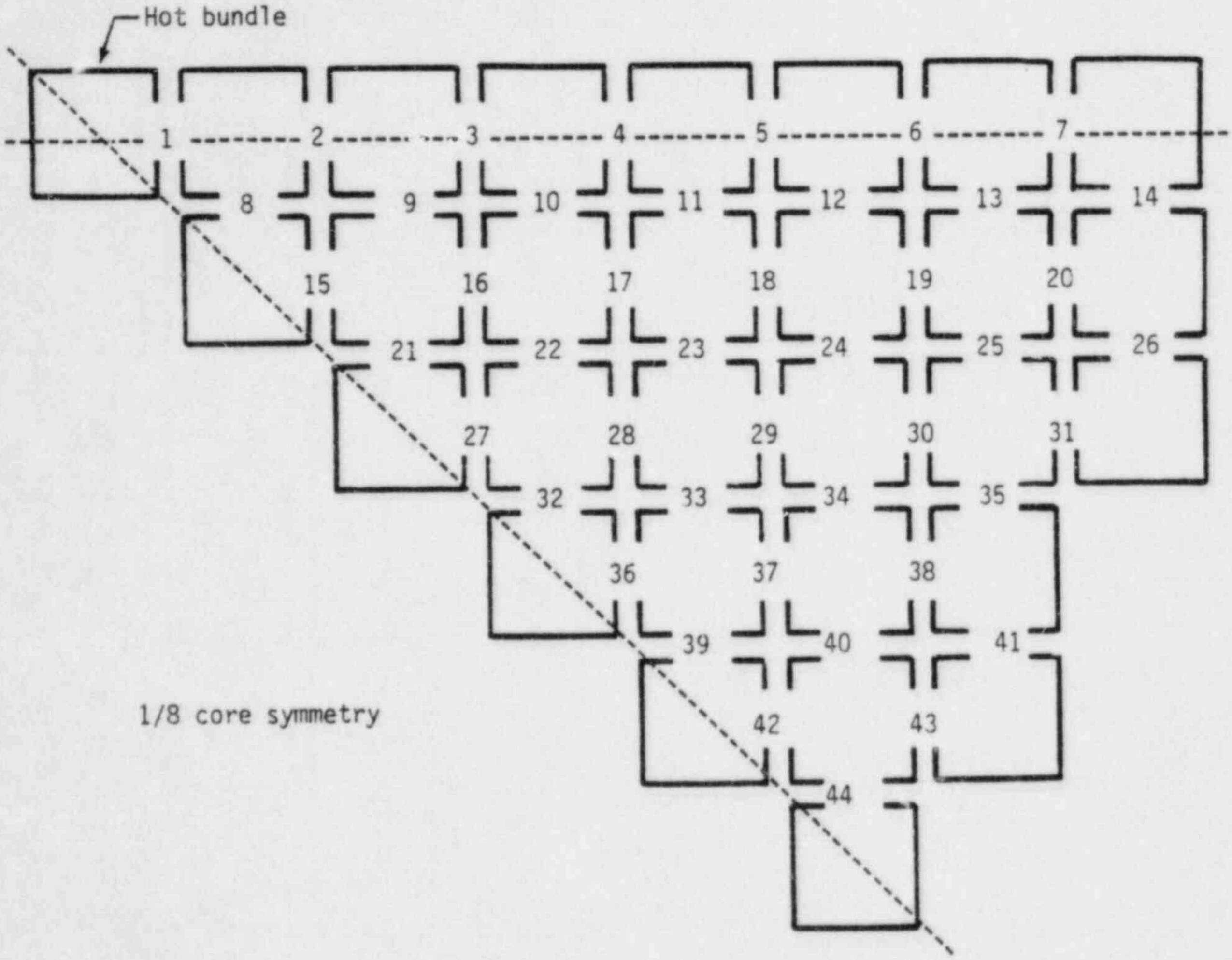
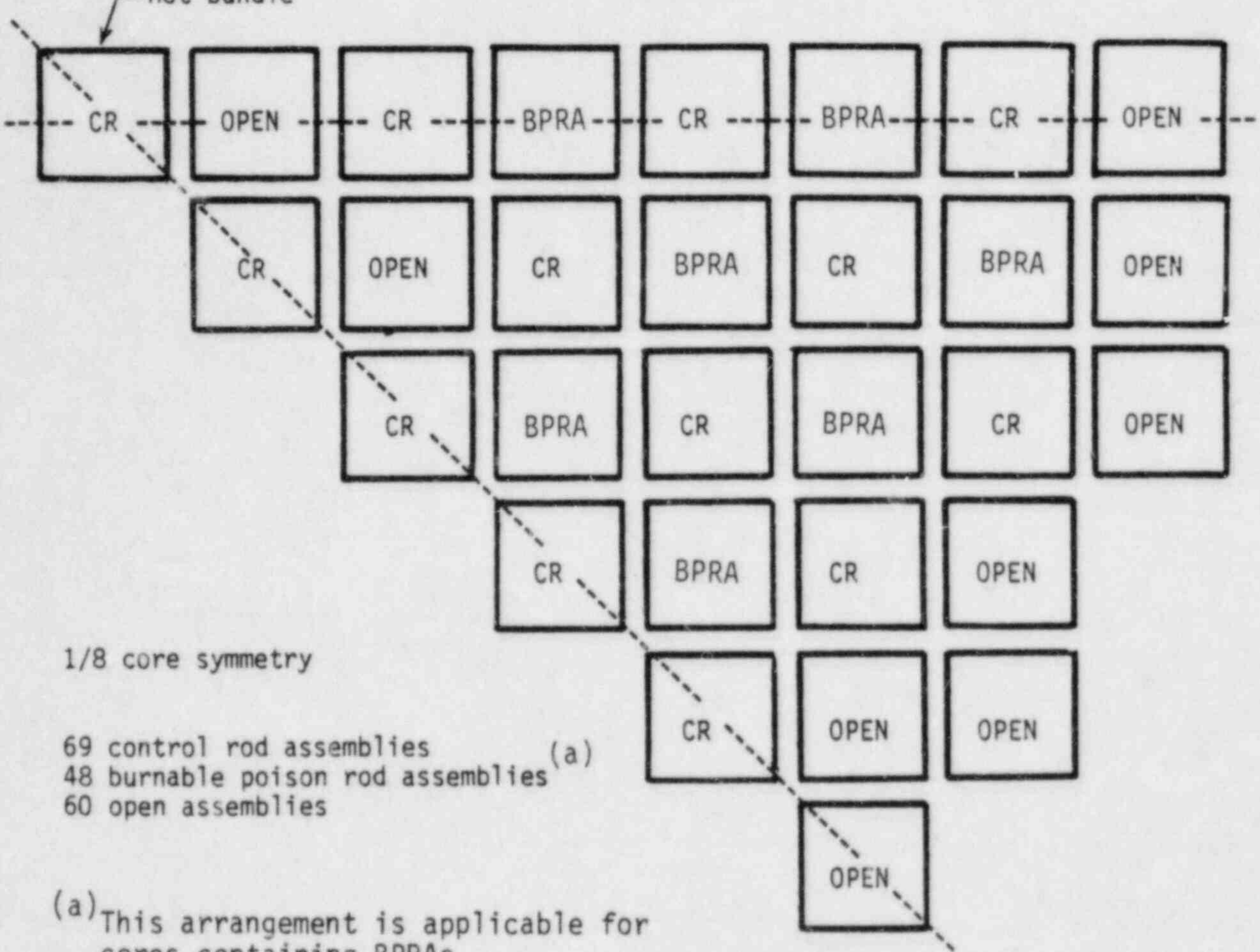


Figure 2-3. Control Component Configuration (LYNX1)

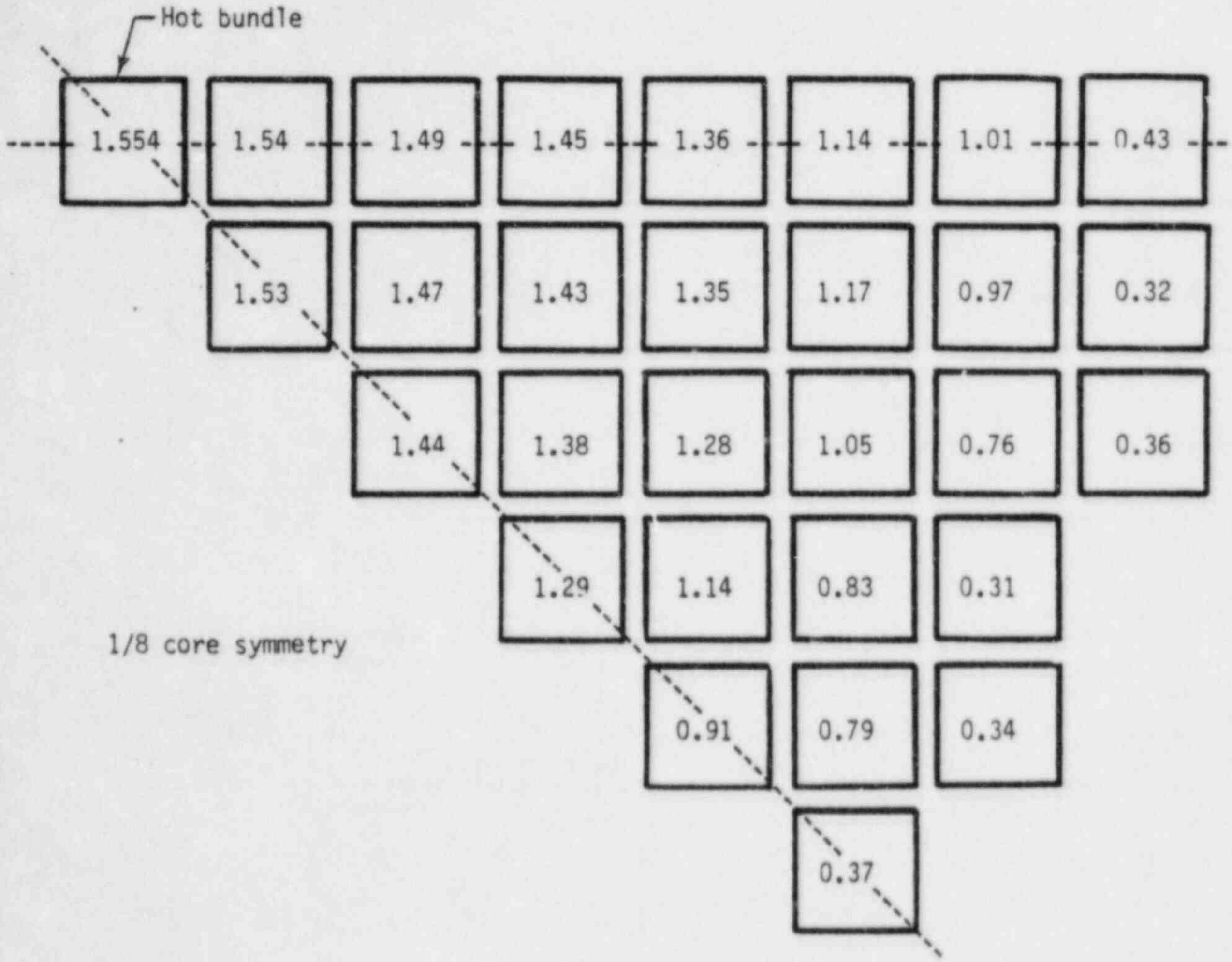


1/8 core symmetry

69 control rod assemblies
48 burnable poison rod assemblies[(a)]
60 open assemblies

(a) This arrangement is applicable for cores containing BPRAs.

Figure 2-4. Bundle Radial Peaking Distribution (LYNX1)



where

$$\text{hot bundle radial peak} = \frac{\text{hot pin radial-local peak}}{\text{hot pin local peak}} = \frac{1.65}{1.0615} = 1.554$$

Figure 2-5. Inlet Flow Factor Distribution (LYNX1)

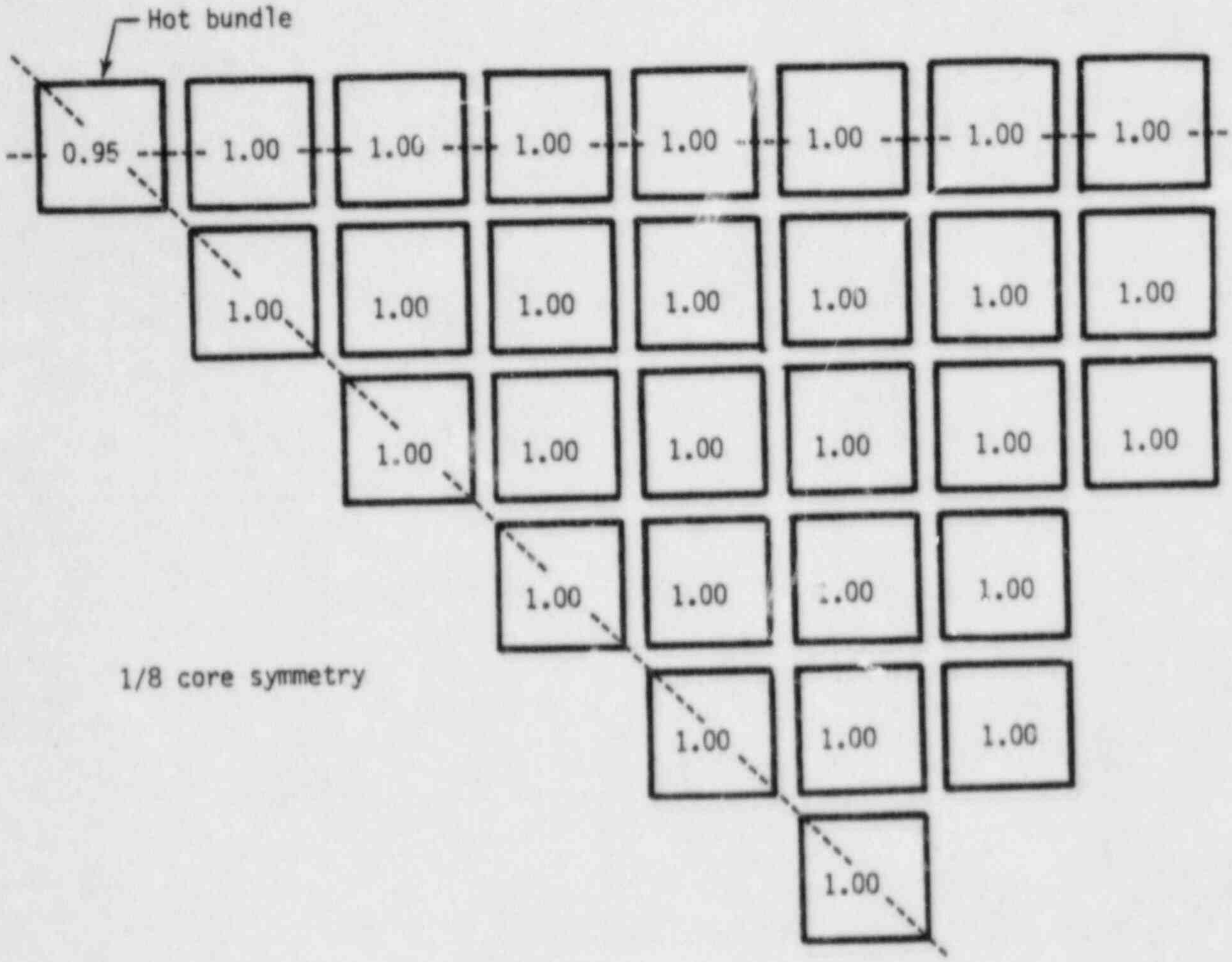


Figure 2-6. Pin and Subchannel Numbering System (LYNX2)

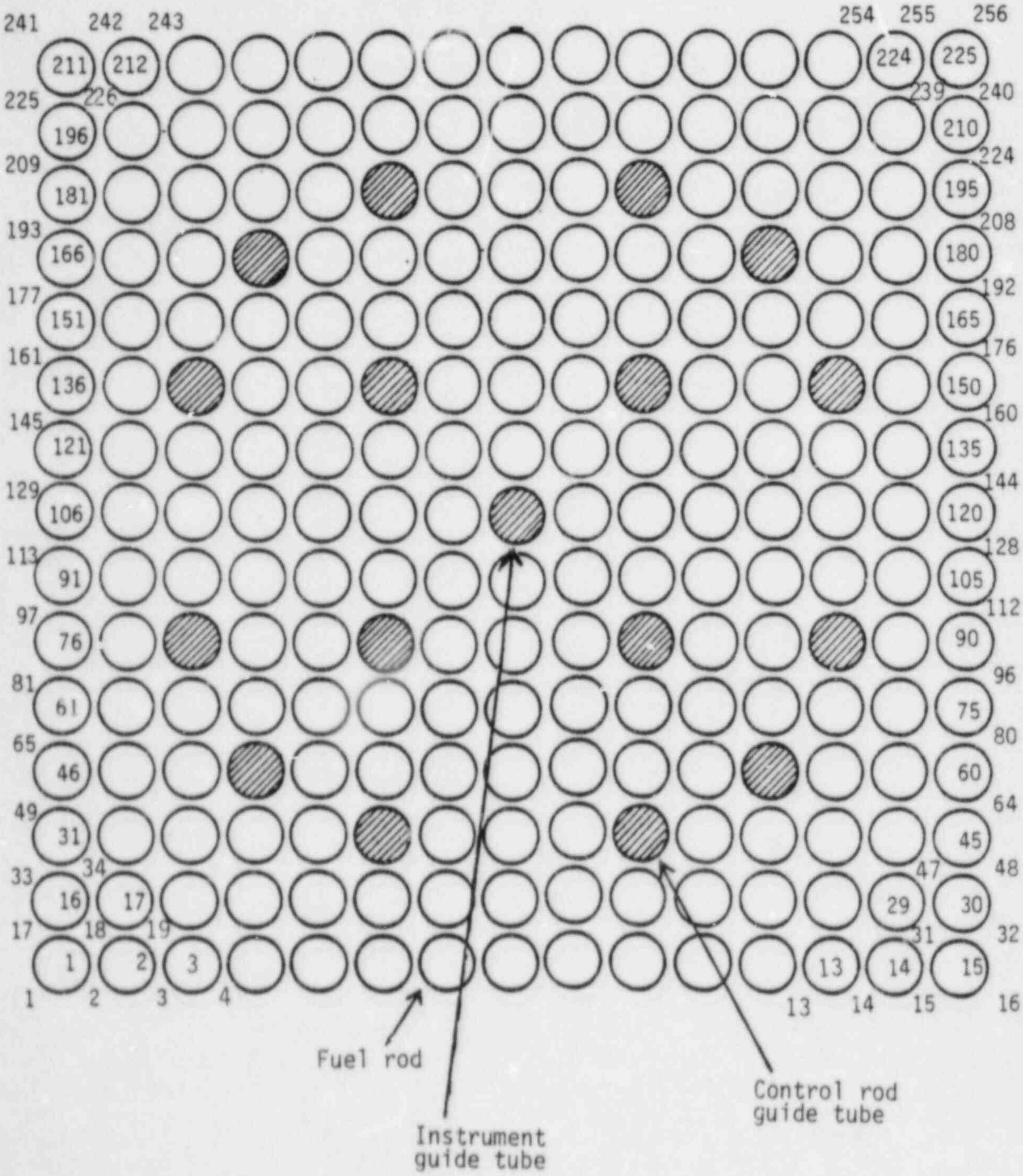
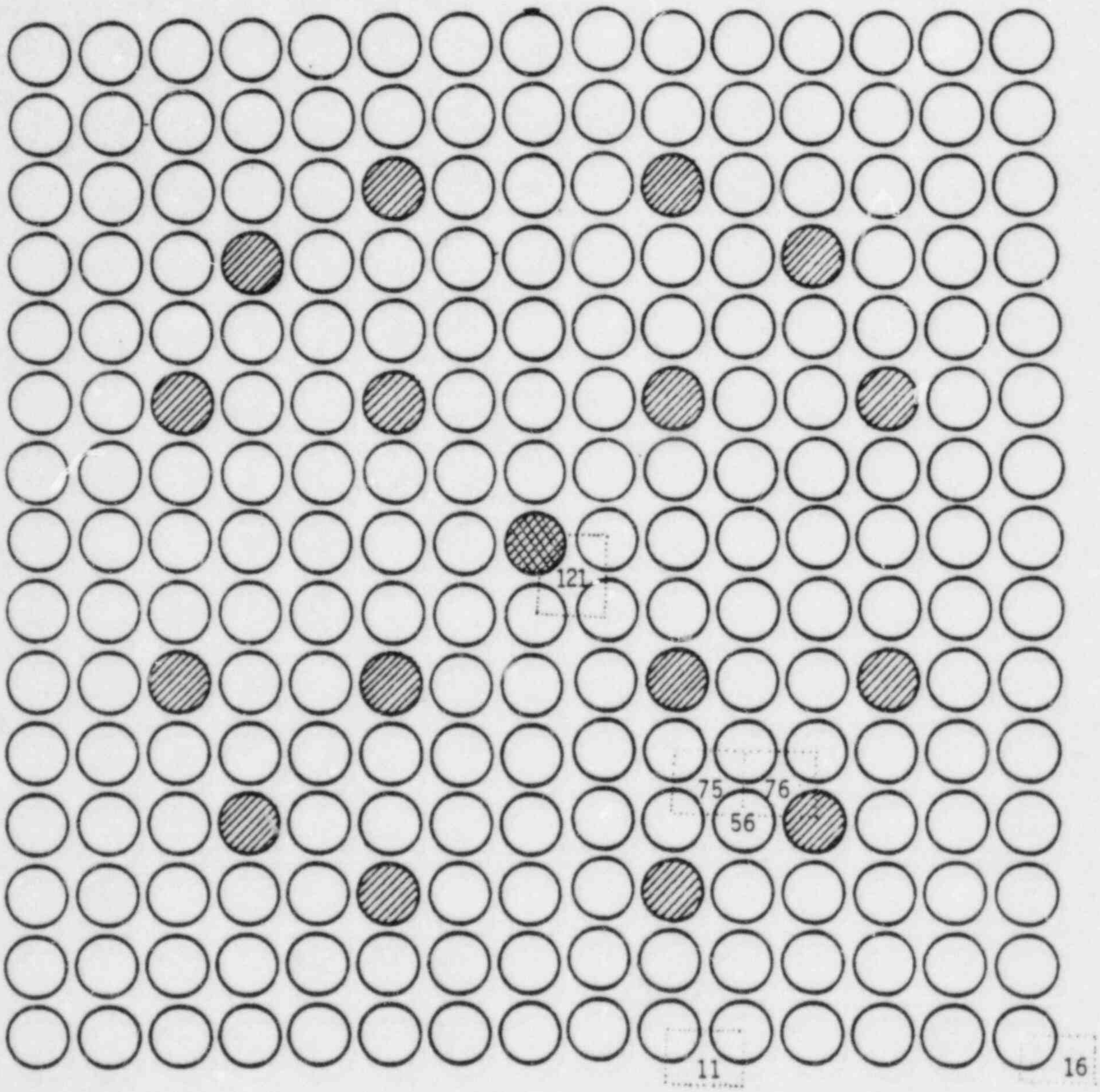


Figure 2-7. Local Peaking Distribution for Hot Bundle

ROW/COLUMN	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15
1	.9618	.9534	.9568	.9676	.9809	.9875	.9767	.9734	.9825	.9825	.9762	.9701	.9626	.9518	.9659
2	.9493	.9593	.9568	.9784	1.0058	1.0200	.9958	.9784	.9992	1.0158	1.0042	.9800	.9618	.9601	.9501
3	.9493	.9551	.9659	1.0067	1.0349	-	1.0125	.9817	1.0158	-	1.0333	1.0075	.9701	.9559	.9510
4	.9584	.9751	1.0049	-	1.0565	1.0366	1.0108	.9859	1.0133	1.0316	1.0540	-	1.0083	.9751	.9593
5	.9701	1.0017	1.0324	1.0557	1.0490	1.0382	1.0166	.9933	1.0191	1.0333	1.0466	1.0557	1.0357	1.0017	.9701
6	.9759	1.0150	-	1.0349	1.0382	-	1.0366	1.0100	1.0391	-	1.0349	1.0349	-	1.0141	.9759
7	.9676	.9933	1.0125	1.0116	1.0183	1.0391	1.0150	1.0141	1.0175	1.0357	1.0183	1.0141	1.0175	.9950	.9709
8	.9643	.9759	.9917	.9867	.9950	1.0125	1.0141	-	1.0166	1.0100	.9950	.9884	.9859	.9776	.9667
9	.9709	.9967	1.0150	1.0150	1.0216	1.0415	1.0175	1.0166	1.0200	1.0382	1.0208	1.0158	1.0191	.9975	.9726
10	.9751	1.0141	-	1.0357	1.0382	-	1.0391	1.0124	1.0407	-	1.0391	1.0391	-	1.0200	.9809
11	.9717	1.0042	1.0357	1.0582	1.0515	1.0399	1.0216	.9975	1.0233	1.0391	1.0524	1.0615	1.0416	1.0083	.9776
12	.9634	.9800	1.0108	-	1.0607	1.0407	1.0175	.9917	1.0191	1.0391	1.0615	-	1.0158	.9842	.9684
13	.9576	.9626	.9734	1.0133	1.0407	-	1.0208	.9892	1.0224	-	1.0415	1.0158	.9784	.9659	.9618
14	.9534	.9643	.9643	.9842	1.0100	1.0233	1.0033	.9850	1.0050	1.0249	1.0133	.9884	.9709	.9717	.9634
15	.9692	.9559	.9643	.9726	.9834	.9892	.9834	.9792	.9850	.9917	.9875	.9784	.9717	.9634	.9762

Hot pin

Figure 2-8. Location of Hot Channels (all types)

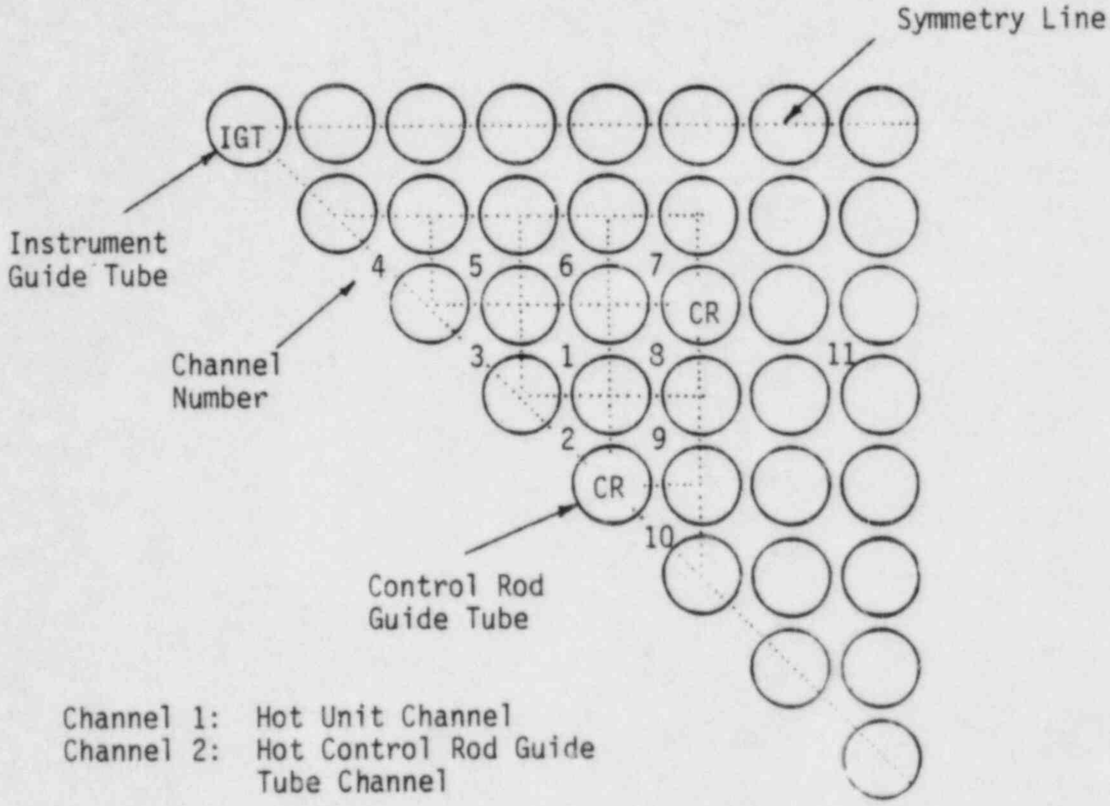


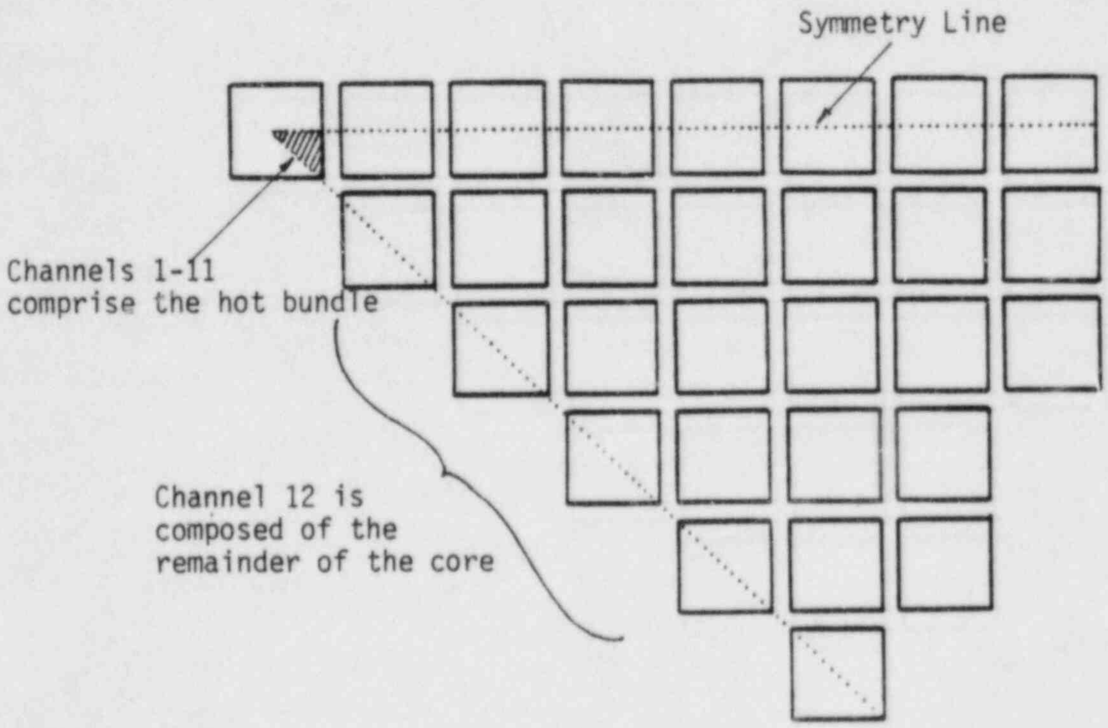
Hot channel locations: Unit cell* 75; Control rod G-T cell 76; Instrument G-T cell 121; Wall cell 11; Corner cell 16

Unit cell*	75	Wall cell	11
Control rod G-T cell	76	Corner cell	16
Instrument G-T cell	121		

*Limiting channel (B&W-2 CHF correlation)

Figure 2-9. Single-Pass Model Channel Numbering System





2.2. Single-Pass Model

The single-pass model utilizes the LYNXT code for determining steady-state and transient flow and temperature distributions within a reactor core. LYNXT is an improved version of the COBRA-IV[5] code developed at Battelle Northwest Laboratories.

Single-pass analyses model subchannels, groups of subchannels, bundles and groups of bundles in one simulation. Historically, core thermal-hydraulic calculations have been performed using multi-pass analysis methods, such as the LYNX1/LYNX2 models discussed above, in which bundles are modeled in an initial "pass" and groups of subchannels in another "pass" which yields the minimum DNBR. LYNX1 and LYNX2 are the B&W multi-pass crossflow calculational tools, respectively. However, LYNXT, with its single-pass modeling capabilities, offers the same accuracy at a lower cost as compared to multipass analyses and therefore will be used for licensing applications.

The LYNXT crossflow model selected for licensing applications is a 12-channel model. The variable-scaling feature of LYNXT permits the simultaneous modeling of the hot subchannel and its surrounding subchannels with the remainder of the core. Figure 2-9 shows the channel modeling scheme. This specific model is applicable to 177-FA core analyses using the B&W-2 CHF correlation for DNB predictions.

3. MODEL DEVELOPMENT AND JUSTIFICATION

The approach followed in developing the single-pass model which is to be used for core thermal-hydraulic analyses was to use a more complex multi-pass model as a benchmark and as a development tool to evaluate DNBR sensitivity to various modeling considerations such as the core power distribution, hot assembly location, inlet flow profiles, etc. Modeling simplifications, desirable for an efficient calculational tool, were made using conservative selections from the options considered. The LYNXT single-pass model was also compared to a similar COBRAIIIC model to provide additional confidence in the final model selected.

3.1. Multi-Pass Model

In developing a multi-pass crossflow model, numerous model sensitivities must be understood and quantified to permit the appropriate selection of model characteristics. The following model characteristics required investigation for the multi-pass model:

- Location of the hot fuel bundle
- Peaking distributions (axial, radial, and local)
- Control component configuration
- Inlet flow profile
- Core exit pressure profile

The sensitivity studies were performed using a basic 112% overpower maximum design case for a 177-FA plant.

3.1.1. Hot Bundle Location

In selecting the location of the hot bundle, various possible locations were considered. The hot bundle was moved from the center location as bundle 1, to numerous locations and ultimately to the peripheral location of bundle 29. A relatively flat hot bundle-to-adjacent bundle peaking gradient was maintained to eliminate any DNBR impact due to bundle power.

The sensitivity study showed that moving the hot bundle throughout the core produces a negligible effect, (0.1% DNB). Since there was no strong DNBR dependency on hot bundle location, additional factors were considered in selecting the appropriate hot bundle. A bundle possessing a higher hydraulic resistance to coolant flow than adjacent bundles would naturally yield more conservative DNB results. Therefore, a bundle location containing a control rod assembly (for this example, bundle 1) was designated the hot bundle since this location could be surrounded by bundles with lower hydraulic resistance.

3.1.2. Power Peaking Distribution

Another model characteristic requiring investigation was the power peaking distribution in the core and the hot bundle. Core thermal-hydraulic analyses use an assumed design peaking distribution which is then imposed on the fuel cycle design as a limiting criterion. The approach followed is to use a "design distribution" for all thermal-hydraulic analyses, both steady-state and transient, then to define combinations of radial and axial peaking factors which provide equivalent DNB protection. These factors are defined as "maximum allowable peaking" (MAP) limits and are used as one of the bases for the power imbalance trip function in the RPS. Similar MAP limits, which correspond to initial condition peaking conditions assumed for accident analyses, are used in the definition of limiting conditions for operation.

The following nomenclature is used throughout the power distribution studies:

The radial peaking factor represents the bundle, or fuel assembly, power relative to the core average power. The local peaking factor represents the power of an individual fuel rod relative to the bundle average. The radial x local peaking factor (commonly denoted as RxL or $F_{\Delta H}^N$) represents fuel rod power relative to the core average. The axial peaking factor (F_z^N) represents the "hot spot," or local, power generation rate relative to the fuel rod average. The total peaking factor (F_Q^N) is the product of radial, local, and axial peaks.

The power distribution study presented herein was oriented toward the development of a model with the following peaking factor limits:

$F_{\Delta H}^N = 1.65$; $F_Z^N = 1.65$; $F_Q^N = 2.72$

This represents a reduction in radial x local peaking and increases in axial and total peaking factors relative to those typically used (1.71, 1.50, and 2.57, respectively), for design and licensing of B&W 177-FA reactor cores. The revised peaking factors were selected to provide a more realistic radial power distribution while, at the same time, accommodating the anticipated need for higher total peaking factors in future very low leakage fuel cycle designs.

Peaking distribution studies were separated into three areas: the bundle radial distribution, the hot bundle local distribution, and the axial distribution. The DNBR sensitivity to the radial peaking distribution was first studied.

Three bundle peaking distributions were considered:

1. A base case (conventional distribution) with the hot bundle in bundle location 10).
2. The same as 1 above, but with a 5% reduction in peaking of the bundles around the hot bundle.
3. The same as 1 above, but with a 15% reduction in peaking of the bundles around the hot bundle.

Figure 3-1 shows the bundle radial distributions for these cases. Note that the peripheral bundle radial peaks were adjusted to maintain peaking normalization. It was observed that the minimum departure from nucleate boiling ratio (MDNBR) was relatively insensitive to the peaking changes of the fuel bundles adjacent to the hot bundle (see Table 4-1). The DNBR differences between the three cases were less than 0.1%, however, the radial peaking distribution with a nearly flat peaking profile around the hot bundle yielded the slightly more conservative DNBR response. This distribution, used in the crossflow model and shown in Figure 2-4, has a radial peaking gradient around the hot bundle similar to the peaking distribution used in traditional closed-channel analyses.

The local peaking distribution was selected after a series of realistic distributions were studied. Results from this study revealed that the minimum DNBR decreased negligibly as long as the hot pin radial-local peak remained constant (see Table 3-2). As the local peaking distribution is flattened, the limiting subchannel, at some point, jumps around the hot bundle, depending on the characteristics of the CHF correlation being used. To avert this behavior in design analyses, a realistic local peaking distribution is selected which yields the lowest DNBR prediction consistently in one channel type.

This methodology conservatively considers the most DNB-limiting local peaking distribution expected in-reactor. The local peaking distibution selected for the model, and shown in Figure 2-7, includes a hot pin local peak of 1.0615. Design analyses, using the design local peaking distribution, will result in the MDNBR occuring in the hot unit channel for the B&W-2 CHF correlation.

The axial peak was selected to achieve an increase in total peak relative to those typically used with closed channel methods. The axial peak selected was a symmetric 1.65 P_{max}/P_{avg} cosine shape.

3.1.3. Control Component Distribution

One consideration in the development of the multi-pass model is the presence or absence of a control component in each fuel assembly since this introduces a difference in resistance to flow at the top of the fuel assembly. The 177-FA reactor cores typically include control rod assemblies (CRAs), axial power shaping rod assemblies (APSRAs), burnable poison rod assemblies (BPRAs), neutron source assemblies, and "open guide tube" assemblies. Hydraulic resistance factors of all of the control component types are the same provided it is assumed that CRAs and APSRAs are fully inserted. Open guide tube assemblies have a lower resistance to flow since they do not have control component spiders in the flow stream. For the LYNX1 model, it is assumed that the hot assembly is in a control component location surrounded by open guide tube assemblies. This results in the most conservative model since the increased resistance causes flow to be diverted from the hot assembly into the surrounding assemblies. A representative arrangement is shown in Figure 2-3 and is composed of 69 CRAs, 48 BPRAs and 60 open guide tube assemblies.

The philosophy followed in developing a specific model is to select a configuration which either represents the actual core arrangement or provides a conservative assessment of thermal-hydraulic performance for a given fuel cycle design. This approach has been followed here to develop a model that can be generically applied to cores with various combinations of control components and open guide tube assemblies.

3.1.4. Core Inlet Flow Distribution

Studies were performed to determine the sensitivity of DNBR to the core inlet flow distribution. Cases ranging from flat inlet flow profiles to realistic profiles with higher core interior flow were studied. Results demonstrated that the hot subchannel MDNBR varied by less than 1% in DNBR for the various cases. The situation yielding the lowest MDNBR prediction was the case assuming a flat inlet flow profile with a 5% reduction in flow at the hot assembly location. The 5% reduction is justified in Reference 6. Therefore, the flat inlet flow distribution with the hot bundle modeled at the core center, as shown in Figure 2-5, was selected for use in the DNB analyses.

3.1.5. Core Exit Pressure Profile

Another consideration for DNBR impact with crossflow modeling was the effect of a non-uniform exit pressure profile. An analysis with the multi-pass model demonstrated the DNBR impact for realistic core exit pressure gradients (of 1.5 psi) to be about 0.5% in DNBR. This impact was deemed insignificant in light of the conservatism already incorporated by the selection of the flat inlet flow profile.

3.2. Single-Pass Model

The single-pass model was developed to represent the arrangement modeled by the multi-pass crossflow model. The necessary channel modeling detail was first investigated. Models comprising as few as 12 to as many as 56 channels were studied. DNBR results showed that the differences in the model resulted in a negligible impact. Table 3-3 identifies the relatively uniform DNBR behavior for the models studied. The 12-channel model was selected for application to 177-FA cores due to its accuracy and economical advantages.

The 12-channel single-pass model, using LYNXT, was compared to the multi-pass (LYNX1/LNX2) crossflow model for a broad range of operational conditions. The DNBR predictions for LYNXT were within 1%, on the average, in DNBR of the predictions of LYNX1/LYNX2. The results of this comparison are shown in Figure 3.2. These comparisons covered a range of DNBR predictions from 1.30 to 2.60 (B&W-2 CHF correlation). The DNBR predictions near the CHF design limit of 1.30 were created with single and multiple variations, from the base design overpower case, of the following system conditions:

Condition	Range
Power	90-112%
Pump status	3, 4 pump
Design flow	80%, 106.5% of design flow
Inlet temperature	-15/+35F variation
System pressure	1800-2135 psia

The single-pass model was also found to agree with COBRAIIIC[7] to within 0.25% in DNBR. The COBRAIIIC benchmark model was the same as the single-pass model used in LYNXT.

The active fuel length to be used in single-pass modeling for licensing application is the undensified cold nominal fuel stack height. The selection of this parameter was based on the physical behavior of the fuel pellet stack within the fuel rod. Irradiation effects comprise both densification and swelling phenomena. The densification effects are more predominant at low exposure levels, while the swelling effects are more predominant at higher exposure levels. Fuel stack densification decreases the active fuel length and increases the surface heat flux. Fuel swelling, which occurs once the fuel pores are filled with fission/backfill gases, tends to increase the active fuel length. In addition to the irradiation effects, the active fuel length is affected by thermal expansion. For 95% TD fuel with a typical densification characteristic, the thermal expansion effects are greater than the irradiation effects as shown below:

Cold nominal stack height	141.8 in.
Hot nominal stack height	143.2 in.
Minimum hot densified stack height	142.2 in.

Therefore, the hot rod fuel stack, while being irradiated, will have a length greater than its cold nominal stack height. It is then conservative to consider the cold nominal stack height in DNBR calculations.

The single-pass LYNXT model will be the primary steady-state and transient analysis tool for licensing analyses. The single-pass model DNBR agreement with the multi-pass (LYNX1/LYNX2) model justifies its use for licensing applications. The demonstrated accuracy and economy are two major advantages of the single-pass model.

Table 3-1. Adjacent Bundle Peaking Impact

Case No.	Peaking reduction of bundles around hot bundle, (%)	Subchannel MDNBR (B&W-2)
1	0	2.10
2	5	2.10
3	15	2.10

Table 3-2. Local Peaking Distribution Impact

Case No.	Hot Bundle radial peak	Hot Pin Local peak	Hot Pin RxL	Subchannel MDNBR (B&W-2)
1	1.584	1.042	1.65	2.41
2	1.554	1.062	1.65	2.41
3	1.539	1.072	1.65	2.41

Table 3-3. LYNXT Model DNBR Comparisons[a]

	Minimum DNBR (B&W-2)		
LYNXT model	112% power/ 2568 MWt	112% power/ 2772 MWt	100% power/ 2772 MWt
12-channel	2.19	1.91	2.24
14-channel	2.19	1.91	2.24
17-channel	2.20	1.92	2.25
23-channel	2.19	1.91	2.24
25-channel	2.19	1.91	2.24
28-channel	2.20	1.92	2.24
30-channel	2.20	1.92	2.24
56-channel	2.20	1.92	2.25

[a]All conditions are maximum design.

Figure 3-1. Adjacent Bundle Radial Peaking Impact Distributions

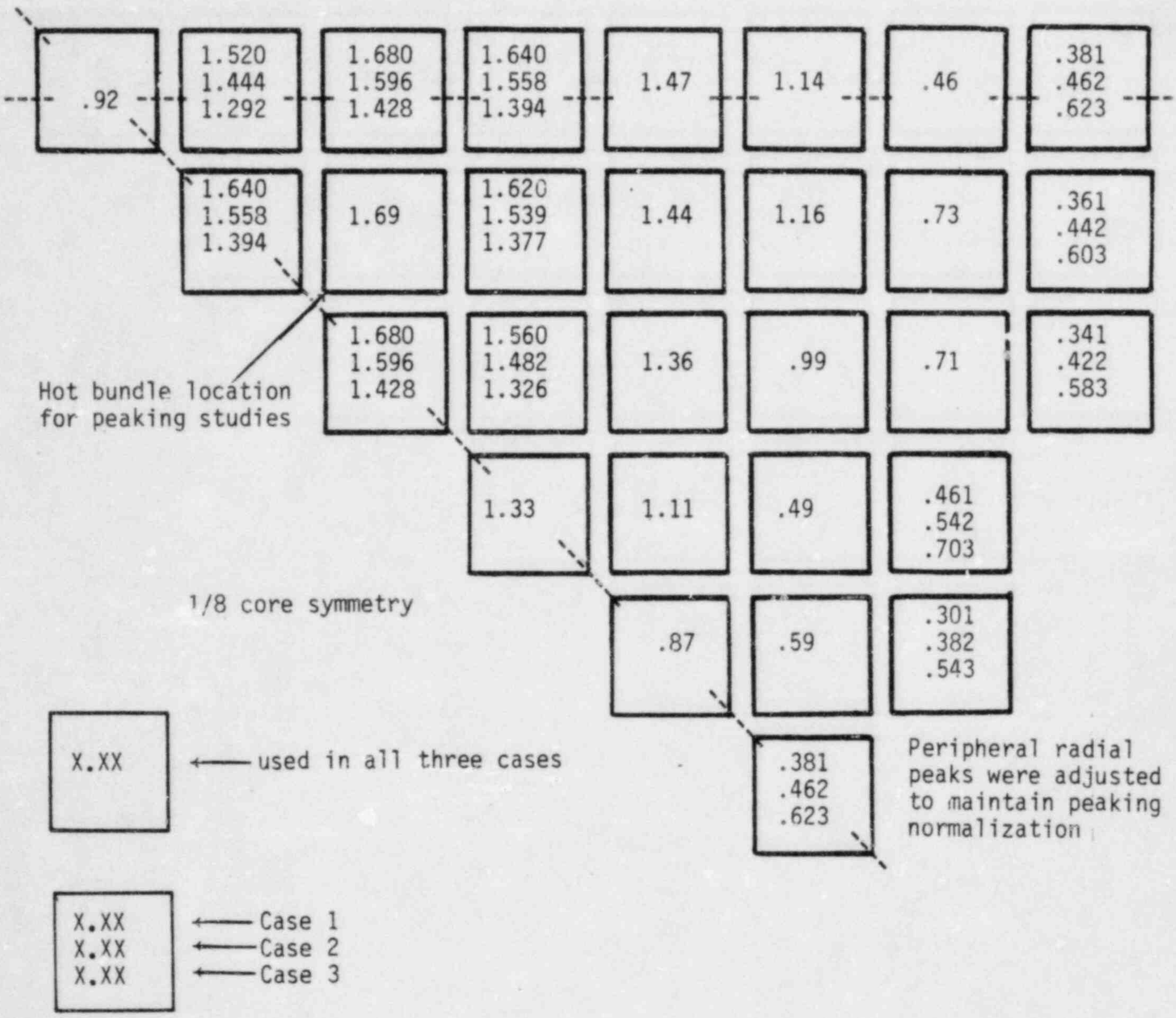
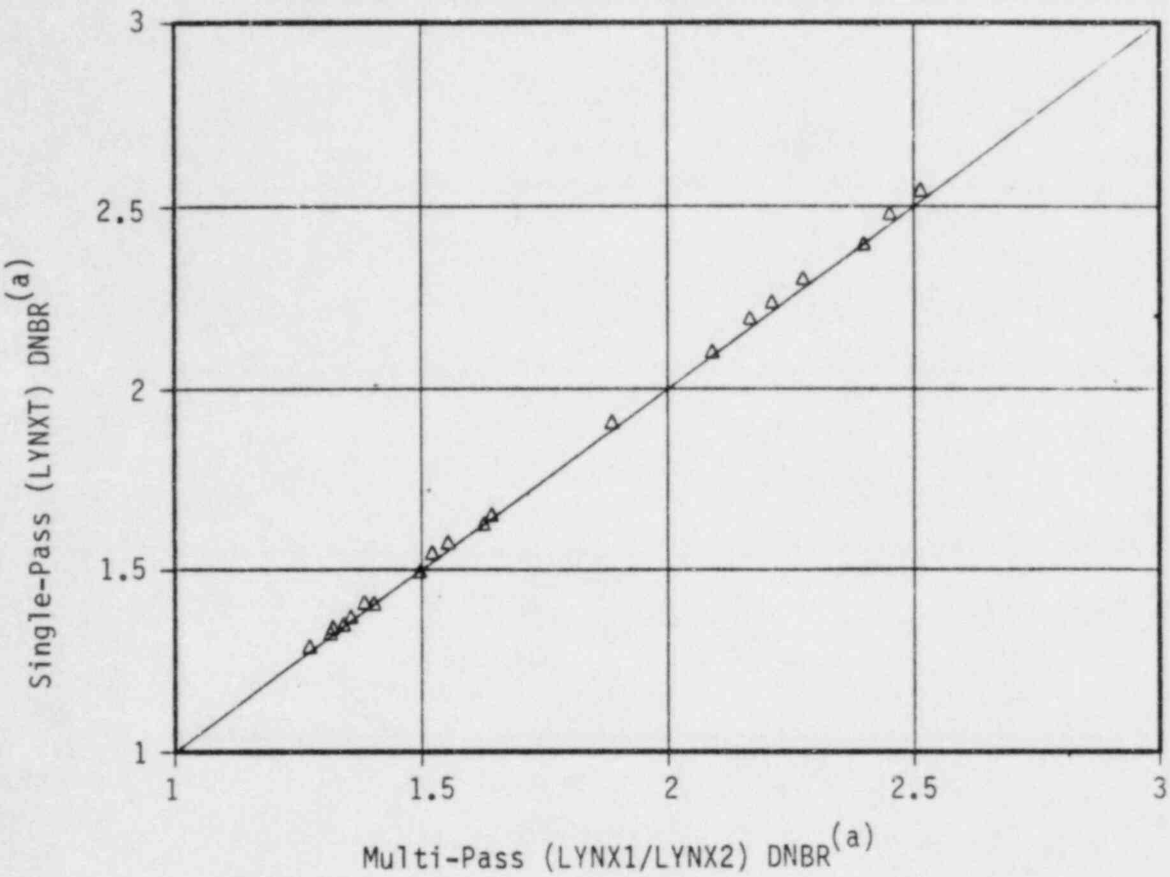


Figure 3-2. Single-Pass Vs Multi-Pass Steady-State DNBR Comparisons



(a) Steady-state DNBR comparisons (B&W-2 CHF correlation)

4. MODEL APPLICATION

The application of state-of-the-art crossflow models provides improved DNBR margin over closed-channel models which can translate into increased fuel cycle design flexibility and less restrictive operating and safety limits. This benefits both the fuel cycle designer and the plant operator. Although the calculated DNB margin increases with the use of crossflow modeling, the application of the margin is consistent with the licensing analysis methods previously established and utilized with the closed-channel codes. The analysis methods for establishing operating and safety limits will remain the same using crossflow modeling.

As discussed previously, the single-pass crossflow model will be used for DNB analyses with the multi-pass (LYNX1/LYNX2) crossflow model reserved for benchmarking and detailed steady-state analyses.

The single-pass crossflow model will be used for both steady-state and transient calculations including the determination of core protection safety limits. Figure 4-1 shows the comparison of multi-pass and single-pass methods in relation to analysis inputs and results.

4.1. Core Safety Limits

The intent of core safety limits is to establish protection for the fuel and reactor system against various hypothetical accidents and operating transients as well as steady-state operation. These limits are incorporated into the reactor protection system (RPS) in the form of setpoints which cause a reactor trip to occur early enough in plant operation to prevent a condition from exceeding the safety limits. Such safety limits, based on thermal-hydraulic considerations, include the following conditions:

- Maximum permissible core power level
- Permissible combinations of core outlet pressure and reactor outlet temperature
- Flux/flow limit

The maximum permissible core power level for four reactor coolant pumps operating is established by the high flux trip setpoint with appropriate adjustments for measurement allowances. This power level is referred to as the design overpower condition (112% full power). The design overpower condition is used to determine steady-state DNBR-based limits for four reactor coolant pump operation. The corresponding power levels for two- and three-reactor coolant pump operation are based on the flux/flow setpoint with their respective flow measurement error adjustment. Examples of the maximum permissible core power levels for the respective pump operation modes used to define the core protection safety limits are shown in Figure 4-2.

A reactor protection system envelope encompassing permissible combinations of core outlet pressure and reactor outlet temperature, known as a P-T envelope, provides DNBR protection as well as reactor coolant system protection. The DNBR protection is in the form of a limiting safety system setting (LSSS) commonly referred to as the variable low pressure trip function. Figure 4-3 shows a pressure-temperature envelope containing the variable low pressure function. This LSSS bounds a DNBR-based relationship of reactor outlet temperatures and core outlet pressures which yield the DNBR design limit (CHF correlation limit) or exceed the CHF correlation quality limit. These DNB relationships, or P-T limit curves, are typically determined for three and two pump operation as well as for four pump operation. The LSSS is set to bound all the P-T curves for the different pump operation modes. The single-pass crossflow model will be used to define the limiting pressure-temperature curves.

The single-pass model is also used to establish DNB limits for asymmetric axial power distributions. The steady-state power distribution used for determining P-T curves (the "design distribution") utilizes a symmetric axial power distribution. In order to maintain a basis for DNB protection for axially asymmetric power distributions, a series of maximum allowable peaking (MAP) limits are generated in the form of lines of constant minimum

DNBR (1.30 using B&W-2 CHF correlation) for various axial peaks at various axial locations. The MAP limits provide a basis for equivalency between the design symmetric power distributions and asymmetric power distributions. These relationships of allowable peaking are translated into allowable positive and negative core axial offsets. Axial offset is defined as the difference between the power in the top and bottom halves of the core divided by the sum of the power in the top and bottom halves of the core. With allowable axial offset established, allowable axial power imbalance is determined and incorporated into the core protection safety limits of the RPS. Axial power imbalance is defined as the axial offset times the fraction of power. This DNB based axial power imbalance may define a portion of the core power imbalance limits, shown in Figure 4-2, providing there are no other more restrictive safety considerations for the top of the core (Note, in general, only the positive imbalance limit is affected by DNB considerations.)

Another DNBR-based safety limit is the flux/flow function. This protection is necessary to assure fuel integrity during transients involving a partial loss-of-coolant flow. The flux/flow trip is used to provide core protection from a partial loss of coolant flow transient and also provides overpower protection for three and two pump steady-state operation. The flux/flow trip limit is determined by the analysis of either a one or two-pump coastdown, depending on the pump power monitor configuration for the specific plant (pump power monitors, which provide an immediate trip signal on loss of electrical power to the pump motors, are used for protection against the more severe loss of coolant flow transients). The analysis method used to determine the flux/flow limit value is the same as that discussed in reference 8. That is, a flow coastdown transient is analyzed with one or more assumed flux/flow trip limit values to determine the DNBR response to the flow transient. A limit value which insures that the minimum DNBR will be equal to or greater than the design DNBR limit (1.30, B&W-2) is then selected by the use of a cross plot of minimum DNBR versus flux/flow limit. Appropriate error adjustments are then provided to determine the setpoint value. This trip setpoint value may then be adjusted downward to be consistent with partial pump steady state overpower assumptions. In practice with crossflow methodology it is anticipated that the flux/flow setpoint will be determined by the three-pump overpower condition rather than by the transient analysis.

The implementation of single-pass crossflow model into the flux/flow setpoint determination is in the DNBR predictions for the coastdown transient. The LYNXT model can determine the transient behavior of the hot channel as well as the remainder of the core in one model simulation.

4.2. Core Operational Limits

Thermal-hydraulic DNBR analyses are also performed to determine the combinations of power distributions (axial and radial) which yield DNB performance equivalent to that of the design distributions used in the study of thermal-hydraulic transients. A set of peaking criteria, in the form of maximum allowable peaking curves, are generated using the single-pass model. These criteria assure the use of the crossflow model design peaking results in conservative DNBR predictions for transient evaluations. These limiting curves are derived in the same fashion as the MAP limits for the RPS.

4.3. Accident Analysis

The crossflow methodology discussed in sections 2 and 3 of this report will be used for the determination of core DNBR response and fuel temperature response to anticipated operational occurrences and hypothetical design-based transients.

Figure 4-1. Multi-Pass and Single-Pass Methods

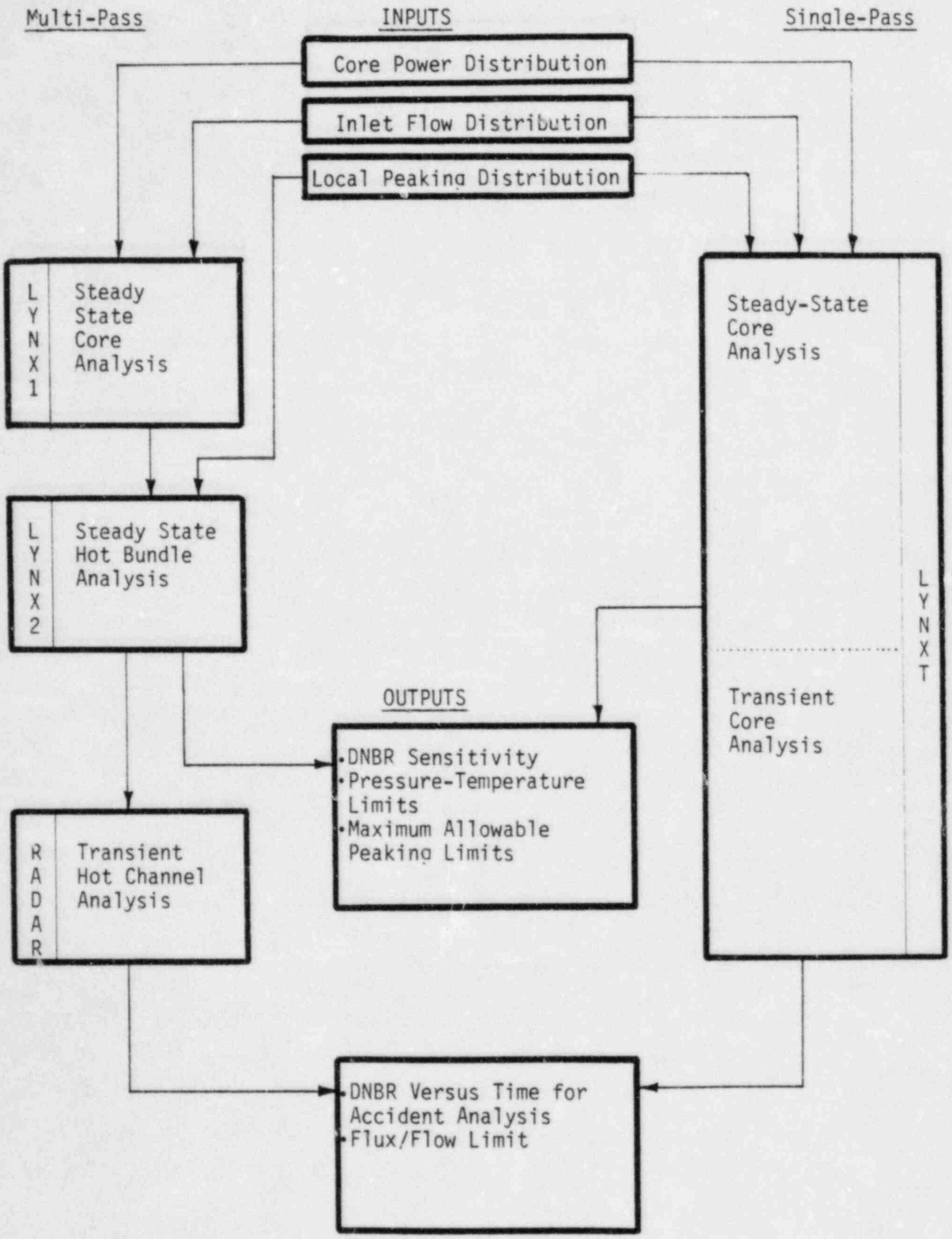


Figure 4-2. Core Protection Safety Limits

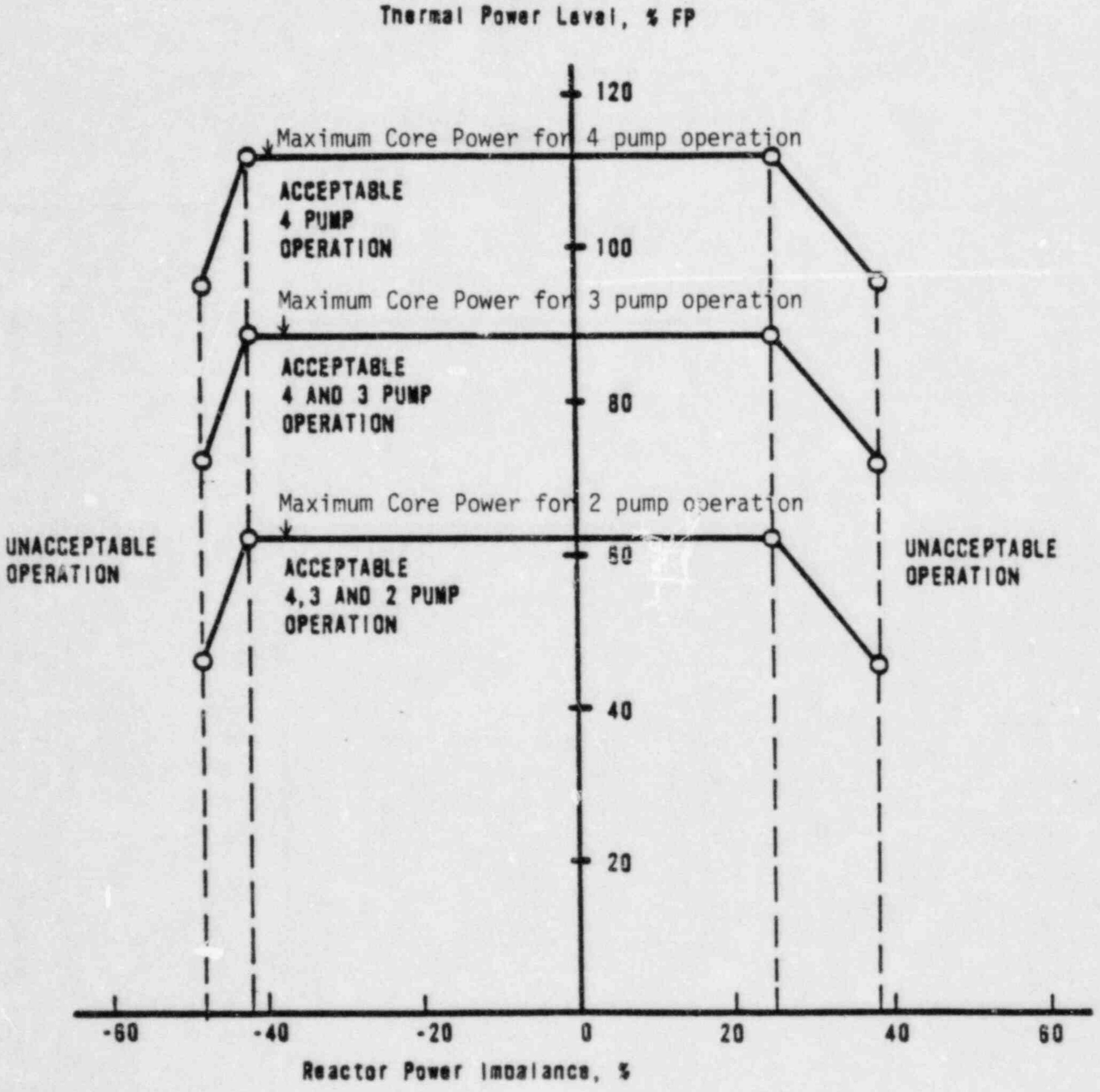
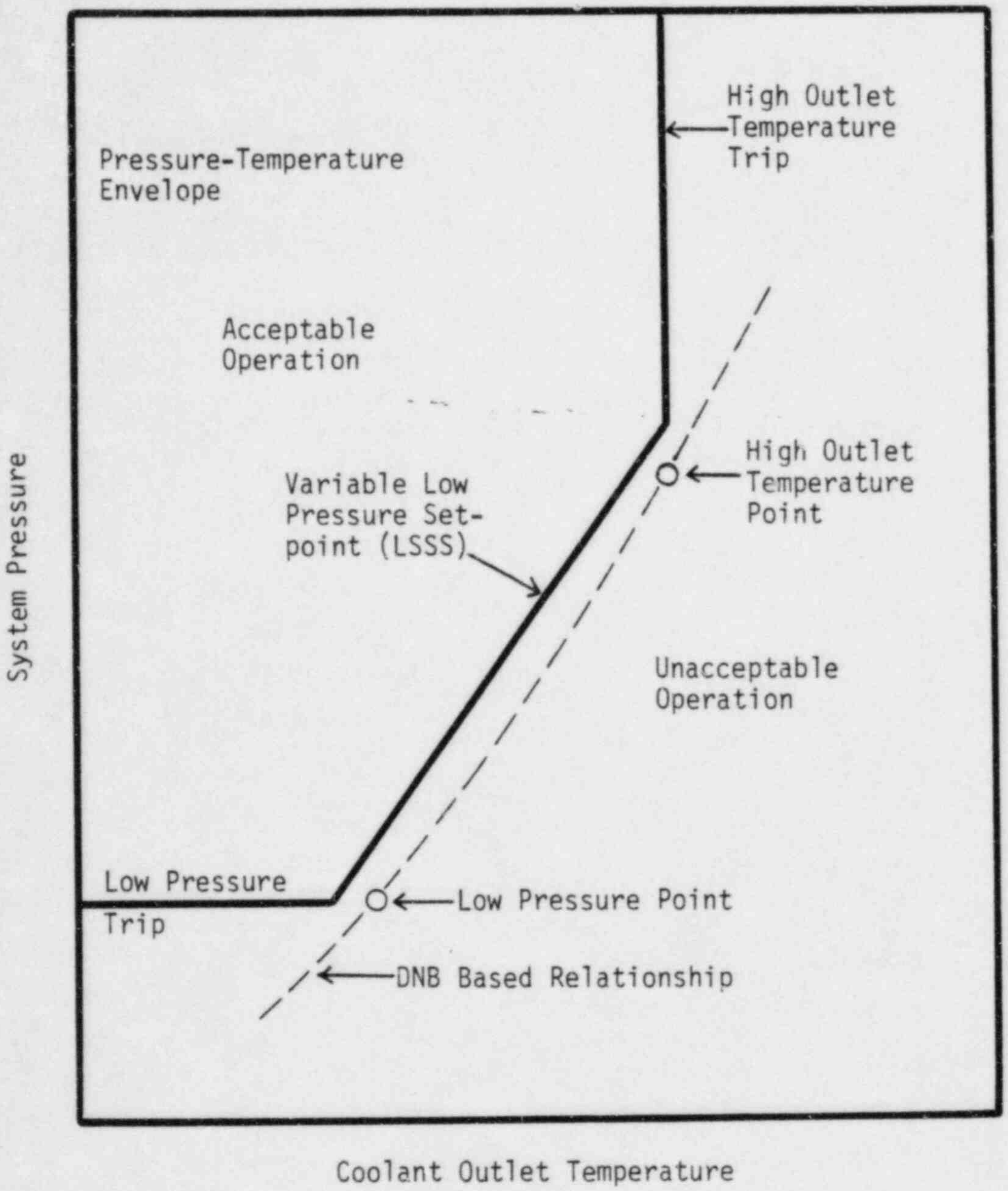


Figure 4-3. Pressure-Temperature Envelope (Protection System Setpoints)



5. SUMMARY

This report describes the models and analysis methods to be used for reactor core thermal-hydraulic analyses. The LYNXT code, with a single-pass core model, will be used for the analysis of core thermal-hydraulic performance for both steady-state and transient conditions. Sensitivity studies have been performed to evaluate various modeling options, such as the number of channels and subchannels to be modeled, the hot assembly location, the core inlet flow distribution, etc., and the results of these studies have been used to select the final model. Benchmarking studies have also been performed with multi-pass LYNX1/LYNX2 analyses and provide justification for use of the single-pass model with LYNXT for core thermal-hydraulic analyses, including the development of core protection safety limits and analyses of response to limiting transients and postulated accidents.

6. REFERENCES

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